Individual Plant Examination for External Events

Edwin I. Hatch Nuclear Plant

Units 1 and 2

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 EXECUTIVE SUMMARY

1.1 BACKGROUND AND OBJECTIVES

On August 8, 1985, the Nuclear Regulatory Commission (NRC) issued a policy statement in the Federal Register stipulating the need for systematic examinations of all nuclear power plants for plant-specific severe accident vulnerabilities. On May 5, 1988, the NRC issued SECY 88-147 which presented a plan for closure of severe accident issues. The plan discusses six major elements, one of which requires examination of existing plants for severe accident vulnerabilities. As a key part of the policy statement implementation, the NRC issued Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)." Georgia Power Company (GPC) submitted to the NRC on December 11, 1992, an Individual Plant Examination (IPE) for Edwin I. Hatch Nuclear Plant Units 1 and 2 to evaluate internal events including internal flooding.

On June 28, 1991, the NRC issued Supplement 4 to Generic Letter 88-20 requesting that each licensee conduct an Individual Plant Examination for External Events (IPEEE). To establish format and content guidelines for submitting IPEEE results, the NRC issued NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities."

This document contains the results of the IPEEE performed for GPC's Edwin I. Hatch Nuclear Plant Units 1 and 2. The hazards evaluated in this examination are seismic events, internal fires, high winds, external floods, transportation and nearby facility accidents, and other events. The information is presented using the Standard Table of Contents for IPEEE Submittal contained in NUREG-1407, Appendix C, Table C.1.

Performance of the Plant Hatch IPEEE allows GPC to meet the following objectives:

1. To develop an appreciation of severe accident behavior.

2. To understand the most likely severe accident sequences that could occur at the plant under full power operating conditions.

3. To gain a qualitative understanding of the overall likelihood of core damage and fission-product releases.

4. If necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

While a probabilistic risk assessment methodology results in the determination of core damage frequencies, fulfilling objective 3 above, the NRC staff endorsed certain methods for evaluating external hazards which do not produce core damage frequencies. These NRC-recommended methodologies were adopted in the Plant Hatch seismic analysis and in the analyses of high winds, external floods, and transportation and nearby facility accidents.

1.2 PLANT FAMILIARIZATION

Georgia Power Company (GPC) is the licensee for the Edwin I. Hatch Nuclear Plant. GPC is a subsidiary of The Southern Company along with Southern Nuclear Operating Company (SNC) and Southern Company Services, Inc. (SCS). SNC provides technical support, and SCS provides architect-engineering and service support for the plant.

Plant Hatch is located on the south side of the Altamaha River, in the northwestern sector of Appling County, Georgia. See figure 1 for the location of the site. Plant Hatch Units 1 and 2 are General Electric designed, single-cycle boiling water reactors (BWR 4 type) with Mark I containments. The units are almost identical, with a few minor design differences. Significant plant parameters for each unit are as follows:

	Unit 1	Unit 2
On-line (commercial)	12/75	8/79
Thermal rating (MWt)	2436	2558
Fuel assemblies	560	560
Control rod assemblies	137	137

The difference in the thermal rating between the units is a result of a power uprate program implemented for Unit 2 in the fall of 1995. The power uprate program for Unit 1 is scheduled for implementation in 1996.

Feedwater is provided to each reactor by two turbine-driven reactor feed pumps. Two external, motor-driven recirculation pumps inject high velocity water into the internal jet pumps, which entrain additional water for flow through the reactor core. Steam is provided to the main turbine (one high-pressure stage and two low-pressure stages) through four main steam lines. The turbine exhausts to the main condenser. The condenser heat sink consists of three sets of forced-draft cooling towers and one helper cooling tower for each reactor unit.

The ultimate heat sink for decay heat removal is the Altamaha River.

1.3 OVERALL METHODOLOGY

The IPEEE for Plant Hatch was performed using methodologies acceptable to the NRC as described in NUREG-1407. The seismic hazard was addressed by performing a seismic margin assessment in accordance with the guidelines of Electric Power Research Institute NP-6041-SL, Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," and with the enhancements listed in GL 88-20. The methodology for the fire IPEEE (i.e., a fire events probabilistic risk assessment [PRA]) was an extension of the PRA methodology used for the IPE. Both the IPE and the fire events PRA were founded on a scenario-based definition of risk. Other external hazards were analyzed using the progressive screening approach methodology as described in NUREG-1407.

1.4 SUMMARY OF MAJOR FINDINGS

The overall result of the IPEEE is that Plant Hatch has no fundamental weaknesses or vulnerabilities to severe accident risk in regard to the external events related to seismic, fire, high winds, floods, transportation and nearby facility accidents, and other external hazards.

The seismic margin assessment results indicate Plant Hatch Units 1 and 2 have a high-confidence-low-probability-of-failure (HCLPF) capacity of at least 0.3 g peak ground acceleration after certain components were modified to raise their HCLPF capacities as described in section 3.1.4.9.

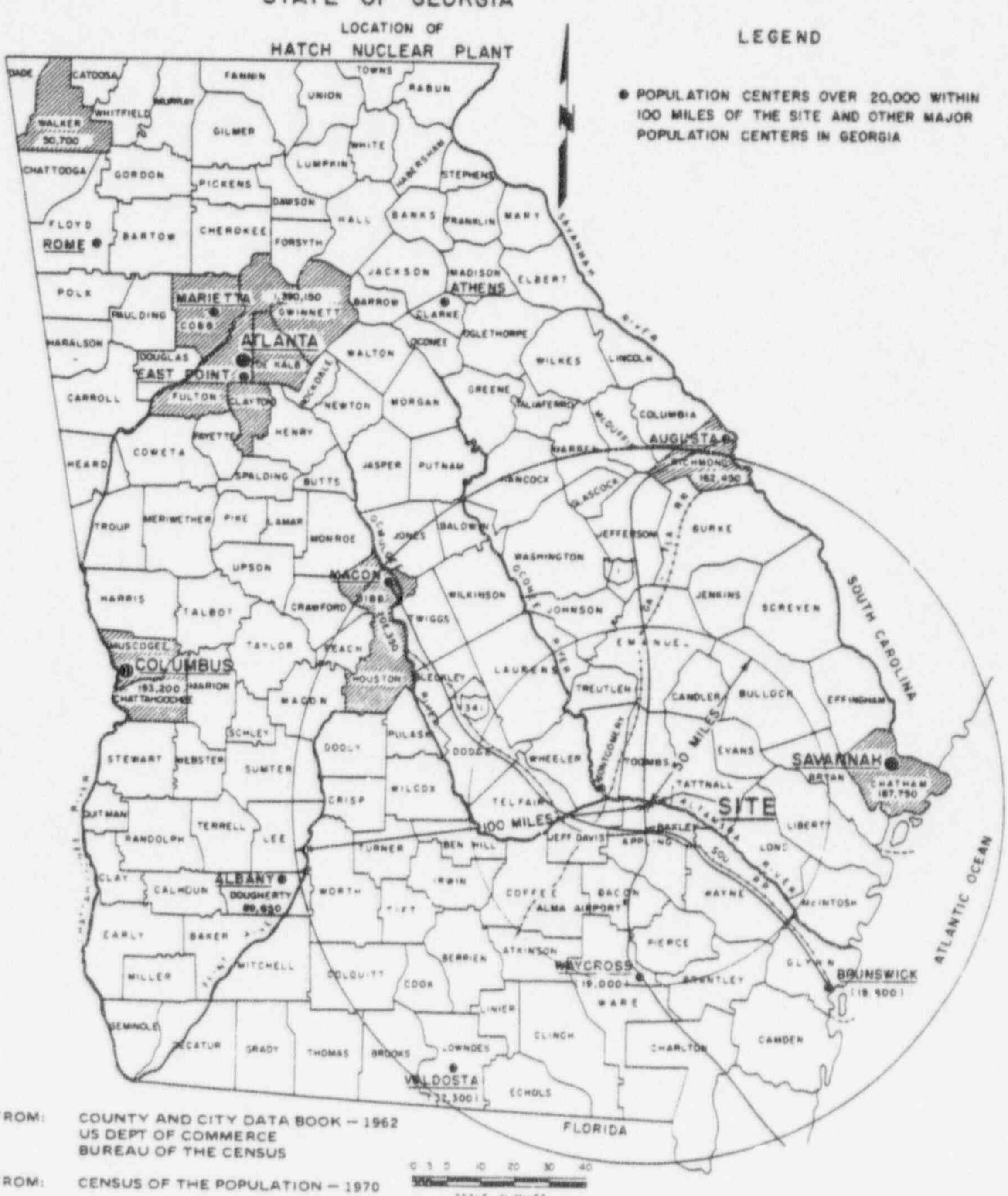
In the fire events analysis, no specific vulnerabilities were identified for Plant Hatch. The core damage frequencies (CDFs) calculated for fire events are consistent with results for other U.S. nuclear power plants. The Level II results for fire events are also consistent with those for internal events at Plant Hatch. Therefore, no containment performance vulnerabilities were identified. The fire-induced CDFs for Unit 1 and Unit 2 were found to be 7.5E-06 per year, and 5.4E-06 per year, respectively. This represents a fire-induced CDF that is slightly less than 36 percent for Unit 1 and 25 percent for Unit 2 of the internal events CDF of 2.1E-05 per year and 2.2E-05 per year, respectively.

In the high winds hazard evaluation, no potential vulnerability was identified. The design of structures at Plant Hatch meets the criteria of the 1975 Standard Review Plan, except that only two of the seven tornado-generated missile types were analyzed. The optional step of applying PRA methods was used to determine the tornado-missile-impact risk. It was concluded that the contribution to CDF from high winds, including tornadoes, is less than 1.0E-06 per year, and the contribution to plant risk is insignificant.

The external flood hazard was evaluated using the more conservative of either the 1975 Standard Review Plan National Weather Service criteria or the design basis criteria. If plant flooding from external causes led directly to core damage, its frequency is estimated to be less than 1.0E-08 per year, which is insignificant.

In the areas of transportation and nearby facility accidents, the existing Plant Hatch design conforms to the 1975 Standard Review Plan criteria. No significant changes were identified which impacted the plant design.

No other plant-unique external events were identified that pose any significant threat of a severe accident at Plant Hatch.



ADAPTED FROM: COUNTY AND CITY DATA BOOK – 1962
US DEPT OF COMMERCE
BUREAU OF THE CENSUS

UPDATED FROM: CENSUS OF THE POPULATION – 1970

Figure 1 Location of Plant Hatch Units 1 and 2

2. EXAMINATION DESCRIPTION

2.1 INTRODUCTION

The Plant Hatch Individual Plant Examination of External Events (IPEEE) was performed to identify and resolve plant-specific severe accident issues arising from external events. Georgia Power Company conducted the IPEEE in compliance with Nuclear Regulatory Commission Generic Letter No. 88-20, "Individual Plant Examination for External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)" Supplement 4, June 28, 1991, and its supporting documentation.

2.2 CONFORMANCE WITH GENERIC LETTER 88-20 AND SUPPORTING MATERIAL

Supplement 4 to Generic Letter (GL) 88-20 (Reference 1) requests that each licensee conduct an Individual Plant Examination for External Events (IPEEE). NUREG-1407 (Reference 2) provides the procedural and submittal guidance for the IPEEE.

Correspondence between Georgia Power Company (GPC) and the NRC (Reference 3) provided GPC's initial response to GL 88-20, Supplement 4, describing the general methodology to be used for the Plant Hatch IPEEE. Subsequently, a request to extend the schedule for final submittal was accepted by the NRC (see References 4 and 5).

In accordance with GL 88-20, Supplement 4, GPC has completed the IPEEE, and the results are documented in this report.

The technical quality of the project was assured by a combination of approaches as follows:

- The assignment of highly competent, experienced personnel to the project team.
- The use of the state-of-the-art methods and software to perform the analyses.
- The involvement of GPC, Southern Company Services, Inc., (SCS), and Southern Nuclear Operating Company (SNC) personnel who are familiar with the design and operation of Plant Hatch to ensure that the models accurately describe the plant, its operating environment, and implementation of plant procedures.
- The performance of independent technical reviews within SNC, GPC, SCS, and PLG, Inc.
- The reviews and comments of the Independent Review Group (IRG).
- The use of appropriate quality assurance procedures.

Georgia Power Company has invested substantial personnel and financial resources in performing the Plant Hatch IPEEE. A core staff of SNC personnel with experience in probabilistic risk assessment, plant design, and specific external hazards were involved in the performance and review of the IPEEE. GPC personnel from the plant were involved in various aspects of the IPEEE, such as the IRG and site reviews. Southern Company Services had the lead responsibility for the performance of the seismic margin assessment (SMA). PLG had the lead responsibility for the internal fire portion of the IPEEE and performed the probabilistic risk analysis associated with the high winds and external flooding analysis.

The NRC was actively involved in the Plant Hatch Unit 1 seismic evaluation performed as part of the Unresolved Safety Issue (USI) A-46 and Electric Power Research Institute SMA pilot project. Reviews were performed by the NRC Seismic Design Margins Working Group, NRC staff, an NRC peer review group composed of industry experts, and an NRC consultant involved in the USI A-46 programs.

REFERENCES

1. U. S. Nuclear Regulatory Commission Generic Letter No. 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f) (Generic Letter No. 88-20, Supplement No. 4)," June 28, 1991.

2. U. S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, June 1991.

3. Letter from J. T. Beckham (GPC), to U. S. Nuclear Regulatory Commission, "Response to Generic Letter 88-20, Supplement 4," HL-1987, December 23, 1991.

4. Letter from J.T. Beckham (GPC), to U. S. Nuclear Regulatory Commission, "Updated Response to Generic Letter 88-20, Supplement 4," HL-2404, September 18, 1992.

5. Letter from K. N. Jabbour (NRC) to J. T. Beckham (GPC), "Review of Response to Generic Letter 88-20, Supplement No. 4 - Individual Plant Examinations for External Events (TAC Nos. M74419 and M74420)," September 7, 1993.

2.3 GENERAL METHODOLOGY

The Individual Plant Examination of External Events (IPEEE) consists of three separate analyses: seismic, fire, and other analyses.

This section provides a brief description of the methodology used in performing these analyses.

2.3.1 SEISMIC

The Plant Hatch seismic events were addressed by performing a seismic margin assessment (SMA) in accordance with the guidelines of Electric Power Research Institute (EPRI) NP-6041 (Reference 1), hereinafter referred to as the EPRI SMA methodology, and as accepted in NUREG-1407 (Reference 2). The EPRI SMA methodology, as applied to Plant Hatch Units 1 and 2, consists of the following tasks:

1. Assessment team selection.

2. Preparatory work prior to the walkdown:

    - Assembly of the Plant Hatch seismic design documents for review by the seismic review team (SRTs).

    - Selection of success paths and identification of associated support systems and their components.

    - Performance of a low-ruggedness relay review.

    - Performance of a soil-structure interaction analysis and development of new in-structure response spectra.

    - Assessment of the seismic margin of soils.

    - Prescreening of structures and equipment (refer to tables 2-3 and 2-4 of the EPRI SMA methodology).

    - Performance of prewalkdown activities.

3. Seismic capability walkdowns.

4. Seismic margin assessment work including:

   - Structural capability evaluations.

   - Equipment and subsystem capacity evaluations.

5. Documentation.

The seismic evaluation of containment performance included an analysis of those functions relating to containment integrity and containment isolation. Early containment failure was addressed by including the containment spray system on the Safe Shutdown Equipment List (SSEL) for both units. For containment isolation, the screening criteria used in the Plant Hatch Individual Plant Examination (IPE) (see References 3 and 4) were also used to construct the list of containment isolation valves to be included on the SSEL.

Plant Hatch is categorized as a focused-scope plant by the Nuclear Regulatory Commission (NRC) in Generic Letter 88-20 (Reference 5). The Unit 1 SMA evaluation was originally performed from 1988 to 1989 as part of a combined Unresolved Safety Issue (USI) A-46 and EPRI SMA project. The Plant Hatch Unit 1 SMA was conducted as a pilot plant study of the EPRI SMA methodology (Reference 6) for EPRI and the NRC, the results of which are documented in EPRI NP-7217-SL (Reference 7). Revision 0 of the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP) (Reference 8) was also used to perform this initial evaluation. This evaluation was subsequently updated to comply with the latest revision of the SQUG GIP (Reference 9) and its supporting documentation, as well as Revision 1 of the SMA methodology.

The Plant Hatch Unit 2 SMA was performed from 1993 to 1994 as part of a combined SMA and USI A-46 Program. The majority of the SMA work was performed by the same personnel who successfully completed the Plant Hatch Unit 1 SMA referenced above. This approach provided consistency between the Units 1 and 2 programs and ensured that lessons learned from the Unit 1 program were incorporated into the Unit 2 effort.

2.3.2 FIRE

For the fire portion of the IPEEE, the objectives described in chapter 1 were accomplished by performing a Level II probabilistic risk assessment (PRA) for Plant Hatch Unit 1 and Unit 2. NUREG/CR-2300 (Reference 10) defines the three levels of PRA work scope. The fire analysis described in this report represents a Level II analysis. The Level II analysis includes an assessment of the frequency of a spectrum of release categories. The scope of accident sequences that are included in this PRA is limited to those initiated by the fire events in conformance with NUREG-1407.

The scenario-based PRA approach systematically and successively evaluates fire hazards and their associated risk impact to Plant Hatch. Figure 4.0-1 in chapter 4 of this report illustrates the major portion of overall approach of the Plant Hatch fire analysis.

The fire IPEEE makes use of the model developed for the IPE as the basis for quantification of external events accident sequences. A major portion of the fire IPEEE effort involved identifying the initiating events and their impact on plant response. The accident sequence model for the Plant Hatch fire IPEEE contains a large number of scenarios that are systematically developed from the point of initiation to termination by utilizing a process of successively screening an initial set of accident scenarios until a credible set is defined and quantified.

Initially a scenario developed for each fire zone was conservatively assumed to damage all plant components and raceways located in the fire zone(s) prescribed by the scenario. This allows the fire zones that have insignificant plant risk associated with fire hazards to be screened from the subsequent analysis at an early stage of the project.

Next, the risk-significant scenarios were quantitatively screened for their risk significance. A scenario was screened from further evaluation if its core damage frequency (CDF) is less than 0.1 percent of the total internal events-induced CDF; i.e., a screening cutoff value of 2.2E-08 per year was used.

The scenarios retained from the quantitative screening were deemed to be relatively risk significant and to require further analysis. In the detailed analysis phase, one or more subscenarios were developed for each scenario that survived the quantitative screening to obtain a more realistic risk impact.

Each subscenario was evaluated iteratively until the subscenario was considered to be relatively risk insignificant or all reduction factors such as geometry factor, severity factors, and fire nonsuppression factor were applied. Upon completion of this process, the CDFs calculated for each scenario/subscenario were added together to determine the total fire-induced CDF for Units 1 and 2. The frequency of fire hazard-initiated core damage sequences was used as a measure of the plant vulnerability to fire.

The containment performance in response to fire-induced initiators was assured by utilizing the IPE analysis to determine the release frequencies and conditional containment failure probability for modes involving containment isolation failures, bypass failures, and overpressure failures.

2.3.3 OTHER

The progressive screening approach described in NUREG-1407 was used to identify potential vulnerabilities at Plant Hatch caused by high winds, floods, and transportation and nearby facility accidents. The progressive screening approach consists of the following steps:

1. Review of the Plant Hatch-specific hazard data and licensing bases, including the resolution of each issue.

2. Identification of significant changes since issuance to the Plant Hatch operating license regarding the following:

- Military and industrial facilities within 5 miles of Plant Hatch.

- Onsite storage or other activities involving hazardous materials.

- Transportation.

- Developments that could affect the original design conditions.

3. Determination of whether the Plant Hatch design meets the 1975 Standard Review Plan criteria (Reference 11).

4. Determination of whether the hazard frequency is acceptably low (optional step).

5. Performance of a bounding analysis (optional step).

6. Performance of a probabilistic risk assessment (optional step).

As specified in NUREG-1407, evaluations for the effects of localized probable maximum precipitation on site flooding and roof ponding consider the revised PMP estimates published by the National Oceanic and Atmospheric Administration in Hydrometeorological Report (HMR) No. 51 (Reference 12).

REFERENCES

1. Electric Power Research Institute, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" (Revision 1), EPRI NP-6041-SL, Palo Alto, CA, August 1991.

2. U. S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, June 1991.

3. Letter from J. T. Beckham (GPC), to U. S. Nuclear Regulatory Commission, "Generic Letter 88-20 Response, Individual Plant Examination Submittal," HL-3039, December 11, 1992.

4. Letter from J. T. Beckham (GPC), to U. S. Nuclear Regulatory Commission, "Revisions to Individual Plant Examination," HL-3491, January 10, 1994.

5. U. S. Nuclear Regulatory Commission Generic Letter No. 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," November 23, 1988.

6. Electric Power Research Institute, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," EPRI NP-6041, Palo Alto, CA, October 1988.

7. Electric Power Research Institute, "Seismic Margin Assessment of Edwin I. Hatch Nuclear Plant, Unit 1," EPRI NP-7217-SL, Palo Alto, CA, June 1991.

8. Seismic Qualification Utility Group, "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 0, June 1988.

9. Seismic Qualification Utility Group, "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 2, February 14, 1992.

10. U. S. Nuclear Regulatory Commission, "PRA Procedures Guide," NUREG/CR-2300, January 1983.

11. U. S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Report for Nuclear Power Plants," NUREG-75/087, LWR Edition, September 1975.

12. National Oceanic and Atmospheric Administration, National Weather Service Hydrometerological Report (HMR) No. 51, Probable Maximum Precipitation Estimates, United States East of the 105th Meridian, June 1978.

2.4 INFORMATION ASSEMBLY

The Plant Hatch Individual Plant Examination of External Events (IPEEE) analysis was conducted by staff personnel with significant probabilistic risk assessment (PRA) experience, Appendix R, and seismic experience; staff personnel with Plant Hatch experience; Plant Hatch staff; outside seismic consultants; and personnel from PLG, Inc. The Peer Review Group, as defined by NUREG-1407 (Reference 1) and referred to as the Independent Review Group (IRG), was formed to review the IPEEE process, as well as specific portions of the analysis.

To ensure the Plant Hatch IPEEE is based on the as-built, as-operated, as-maintained condition of the plant, specific steps were integrated into the Plant Hatch IPEEE process. Calculations generated during the Plant Hatch IPEEE process received an independent technical review prior to approval. IRG members who reviewed the IPEEE report included experienced site and corporate representatives from selected departments who are familiar with the plant's day-to-day operations, analysis, and design. An expert consultant provided the peer review for the seismic margin assessment.

In order provide traceability information was obtained from controlled documents to the extent practical. Information significant to the development of the IPEEE was extracted from proper sources and documented appropriately.

2.4.1 SEISMIC

The Plant Hatch safe shutdown analysis for compliance with 10 CFR 50.48, 10 CFR 50 Appendix R, and the General Electric Company analysis for minimum systems required for shutdown of Plant Hatch following a fire, were heavily relied upon for developing the Safe Shutdown Equipment List (SSEL). Systems engineers also reviewed a variety of plant-specific information including piping and instrumentation drawings, electrical elementary drawings, plant arrangement drawings, plant operating procedures, and selected topical reports.

The Plant Hatch seismic design basis, as described in the Unit 1 and Unit 2 Final Safety Analysis Reports (FSARs) (Reference 2) and other documents, was reviewed by the Seismic Review Teams (SRTs) as needed to determine the conformance of the plant design with the screening guidelines in the seismic margin assessment (SMA) methodology (Reference 3). Original construction drawings were reviewed for the concrete and steel frame structures included in the scope of the Plant Hatch SMA to determine the adequacy of typical construction details used at the plant. The SRTs used equipment location drawings and piping drawings to locate SSEL equipment and valves in the plant. Neat line, reinforcing, and embedded iron detail drawings for concrete equipment pads were used by the SRTs to determine the adequacy of SSEL equipment pads. Vendor drawings, manuals, and qualification reports were reviewed as required by the SRTs.

2.4.2 FIRE

At the beginning of this project, the following information sources were reviewed:

- Site and plant general arrangement drawings.
- Combustible inventories (sources) in each fire zone.
- Edwin I. Hatch Nuclear Plant Individual Plant Examination (IPE) (Reference 4), including internal flood analysis.
- Edwin I. Hatch Nuclear Plant Units 1 and 2 Fire Hazards Analysis and Fire Protection Program (FHA) (Reference 5).
- Relevant sections of Edwin I. Hatch Nuclear Plant Unit 1 and Unit 2 Final Safety Analysis Reports (Reference 2).
- Documentation of the location of equipment and associated cables important to safety.

Information related to the locations of accident mitigation equipment susceptible to fire damage was collected from the internal flooding analysis database (previously developed as part of the IPE) as well as site and plant arrangement drawings. Information regarding cables associated with the accident mitigation equipment was obtained from Appendix R safe shutdown analysis and additional circuit and cable routing analyses. Information related to the impact of fire damage to accident mitigation equipment was taken from the IPE model. The Unit 1 and Unit 2 FSARs and the Edwin I. Hatch Fire Hazards Analysis and Fire Protection Program (FHA) (Reference 5) provide estimates of the in-situ and transient combustible materials and fire loadings within each fire zone. The FSAR and FHA also provide an estimate of the fire severity in terms of fire duration (in hours), a list of fire detection and suppression capabilities, and the fire barrier rating. These information sources were important for the qualitative and quantitative fire analysis. Relevant information was collected and summarized in a set of location characteristics tables for subsequent analyses of the important fire zones. These data sources were also supplemented by information collected during the plant walkdowns as described previously.

In addition to the above information sources, the following data was collected:

- Industry and Plant Hatch fire incident data.
- Fire drill records for Plant Hatch.

Both industry and Plant Hatch fire incident data were organized in accordance with the component faults-induced and human-induced ignition source categories established for this analysis. The fire incident data were used for the calculations of fire ignition frequency. Information contained in the fire drill records was factored into the consideration of fire suppression.

A set of physical parameters (e.g., fire location configurations, dimensions, material rating, etc.) was also collected and used in the fire growth calculations for the fire severity and suppression evaluations. Finally, the IPE model developed for the IPE submittal was used for the calculations of fire scenario core damage and containment event tree end state frequencies.

2.4.3 OTHER

Significant plant documentation reviews were performed and a confirmatory plant walkdown was conducted to evaluate potential vulnerabilities to Plant Hatch from high winds, floods, transportation and nearby facilities accidents. The walkdown was performed by Southern Nuclear Operating Company and from Georgia Power Company personnel familiar with the plant layout. The walkdown concentrated on outdoor facilities that could be affected by high winds, onsite storage of hazardous materials, and offsite developments, as directed by NUREG-1407.

Information assembled for this portion of the submittal consisted of the following:

- Information on plant-specific hazard data and licensing bases.
- Identified significant changes since issuance of the operating license.
- Results of plant design review to the 1975 Standard Review Plan (SRP) (Reference 6).

The following information was also gathered and/or reviewed during performance of this portion of the IPEEE and is stored and/or referenced in supporting work packages as tier 2 documentation:

- Site general arrangement drawings.
- Relevant sections of the FSARs.
- Relevant sections of the 1975 SRP.
- Current information concerning hazardous materials stored onsite or at facilities within 5 miles of Plant Hatch.
- Current information concerning transportation routes within 5 miles of Plant Hatch.
- Current information concerning industrial or military facilities within 5 miles of Plant Hatch.
- Current meteorological reports.
- Plant procedures.
- Electric Power Research Institute reports on tornado missiles.

REFERENCES

1. U. S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, June 1991.

2. Edwin I. Hatch Nuclear Plant Unit 1 and Unit 2 Final Safety Analysis Report, Rev. 13C, April 1995.

3. Electric Power Research Institute, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," (Revision 1) EPRI NP-6041-SL, Palo Alto, CA, August 1991.

4. Edwin I. Hatch Nuclear Nuclear Plant Units 1 and 2 Individual Plant Examination, December 1992.

5. Edwin I. Hatch Nuclear Plant Units 1 and 2 Fire Hazards Analysis and Fire Protection Program, Rev. 10C, April 1995.

6. U. S. Nuclear Regulatory Commission, Standard Review Plan, NUREG-75/087, September 1975.

3. SEISMIC ANALYSIS

3.0 METHODOLOGY SELECTION

The seismic portion of the Individual Plant Examination for External Events (IPEEE) was performed using the methodology described in EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Revision 1 (Reference 1; hereafter referred to as the EPRI seismic margin assessment (SMA) methodology). The seismic IPEEE included the enhancements suggested in Generic Letter (GL) 88-20, Supplement 4, "Individual Plant Examination for External Events for Severe Accident Vulnerabilities - 10 CFR 50.54(f) (Reference 2). The applicable portions of the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP) Revision 2 (Reference 3) used in resolving Unresolved Safety Issue (USI) A-46 were combined with similar portions of the EPRI SMA methodology.

Since combining the EPRI SMA methodology with the SQUG GIP satisfies the objectives of the SMA and the requirements of GL 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," (Reference 4), additional evaluations for many classes of equipment not specified in the EPRI SMA methodology were required. In NUREG-1407 (Reference 5), table 3.1, the NRC staff defines Plant Hatch as a focused scope plant having a review level earthquake (RLE) of 0.3 g peak ground acceleration (pga). The seismic margin earthquake (SME) developed to meet the RLE criteria of 0.3 g pga was used throughout the evaluations as the seismic demand for both SMA and USI A-46. The SME is enveloped by the seismic capacity bounding spectrum shown in figure 4-2 of the SQUG GIP.

The initial Plant Hatch Unit 1 SMA was conducted from 1988 to 1989 as a pilot plant study (for EPRI and the NRC) of the EPRI SMA methodology. The results of the study are documented in EPRI NP-7217-SL (Reference 6). Revision 0 of the SQUG GIP (Reference 7) was also used in performing the initial evaluation, which was subsequently updated to comply with Revision 2 of the SQUG GIP (Reference 3) and Revision 1 of the EPRI SMA methodology presented in EPRI NP-6041-SL. The update did not include a second walkdown of the Unit 1 Safe Shutdown Equipment List (SSEL) components unless additional information was required to address specific changes following the original walkdown.

The NRC was actively involved in the Unit 1 SMA, including reviews by the NRC Seismic Design Margins Working Group, NRC staff, an NRC Peer Review Group composed of industry experts, and an NRC consultant involved in the USI A-46 programs. Reference 8 documents the final report of the Plant Hatch Unit 1 SMA, and References 9, 10, 11, and 12 document the review reports from the NRC Peer Review Group and the NRC Seismic Design Margins Working Group.

The Plant Hatch Unit 2 SMA was performed from 1993 to 1994 as part of a combined SMA and USI A-46 Program. The majority of the SMA work was performed by personnel who successfully completed the Plant Hatch Unit 1 SMA, thus providing consistency between the

Unit 1 and Unit 2 programs. Lessons learned from the Unit 1 program were incorporated into the Unit 2 effort.

Unless noted otherwise, the information presented in this chapter applies to plant Hatch Unit 1 and Unit 2.

The success paths for the SMA were chosen assuming a small-break loss-of-coolant accident (LOCA). The EPRI SMA methodology, as applied to Plant Hatch Units 1 and 2, consists of the following steps:

- Selection of assessment team.

- Preparatory work prior to walkdown:

  - Assembly of Plant Hatch seismic design documents for review by seismic review teams (SRTs).
  - Selection of success paths and identification of associated support systems and their components.
  - Review of low-ruggedness relays.
  - Performance of soil-structure interaction (SSI) analysis and development of new in-structure response spectra (IRS).
  - Assessment of seismic margin of soils.
  - Prescreening of structures and equipment.
  - Prewalkdown.

- Seismic capability walkdowns.

- SMA work:

  - Structural capability evaluations.
  - Equipment and subsystem capacity evaluations.

- Documentation.

The screening criteria summarized in tables 2-3 and 2-4 of the EPRI SMA methodology described in EPRI NP-6041-SL were used as the screening basis for the Plant Hatch SMAs. Appendix A of EPRI NP-6041-SL provides the basis for the screening tables. Since certain portions of the SQUG GIP used to resolve USI A-46 were combined with the IPEEE, the screening criteria found in table 2-4 of the EPRI SMA methodology were expanded, where necessary, to satisfy the SQUG GIP requirements for class-of-20 equipment. Because the screening criteria are intended for components mounted low and in stiff structures, SQUG GIP requirements for comparing seismic capacity to demand for equipment located approximately 40 ft above grade were used.

The anchorage calculations used to screen equipment followed the EPRI SMA methodology. The seismic demand was based on the SME IRS. If the seismic review team was able to judge the minimum lowest natural frequency of the equipment item, it was so noted. The maximum spectral acceleration at that frequency or higher (at the appropriate damping) was used to determine the seismic demand; otherwise, the peak spectral acceleration value was used.

Approximate equipment weights and centers of gravity were based on Appendix C of the SQUG GIP, unless specific equipment data were available. The anchorage capacity was based on Volume 1 of EPRI NP-5228-SL, "Seismic Verification of Nuclear Plant Equipment Anchorage: Development of Anchorage Guidelines," (Reference 13). Volume 3 of Reference 13, "EPRI/Blume Anchorage Computer Program (EBAC)," was used in the majority of anchorage evaluations. Since the median-centered IRS are based on a free-field ground motion that is two or more times higher than the design basis earthquake, the anchorage calculations used in the Plant Hatch SMAs conservatively satisfy the SQUG GIP.

The results of the SMA indicate Plant Hatch Units 1 and 2 have a high-confidence-low-probability-of failure (HCLPF) value of ≥ 0.3 g pga. Outliers identified during the SMA were either shown by analysis to have a HCLPF of 0.3 g pga or modified to achieve a HCLPF of 0.3 g pga. Therefore, the SMA results demonstrate with high confidence the capability of the plant to shut down safely when subjected to an SME.

Existing configuration control programs ensure Plant Hatch is maintained consistent with the design basis. Design of modifications includes consideration of the SME so that the 0.3 g pga HCLPF capacity is preserved.

REFERENCES

1. Electric Power Research Institute, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," EPRI NP-6041-SL, Revision 1, Palo Alto, CA, August 1991.

2. U. S. Nuclear Regulatory Commission, Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," Washington D. C., June 28, 1991.

3. Seismic Qualification Utility Group, "Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment," Revision 2, February 14, 1992.

4. U. S. Nuclear Regulatory Commission, "Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," Washington, D. C., February 19, 1987.

5. U. S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Final Report, NUREG-1407, June 1991.

6. Electric Power Research Institute, "Seismic Margin Assessment of Edwin I. Hatch Nuclear Plant, Unit 1," EPRI NP-7217-SL, Palo Alto, CA, June 1991.

7. Seismic Qualification Utility Group, "Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment," Revision 0, June 1988.

8. Letter from W. G. Hairston, III (GPC) to the U. S. Nuclear Regulatory Commission with enclosed report, "Seismic Margin Assessment of Edwin I. Hatch Nuclear Plant Unit 1," July 20, 1990.

9. Letter from Dan Guzy (RES Co-chairman Seismic Design Margins Working Group) to U. S. Nuclear Regulatory Commission distribution, "Resolution and Closure of All Soils Issues in the Hatch Review," April 29, 1990.

10. Letter from Dr. Michael P. Bohn (Sandia National Laboratories) to Dr. Nilesh Chokshi (U. S. Nuclear Regulatory Commission) with enclosed report, "Independent Evaluation of the Hatch Seismic Margin Assessment - Seismic Building Response and Floor Spectra," July 5, 1991.

11. Memorandum from Dan Guzy (RES Co-chairman Seismic Design Margins Working Group) and Goutam Bagchi (NRR Co-chairman Seismic Design Margins Working Group) to U. S. Nuclear Regulatory Commission distribution, "Final Evaluation of the Hatch Seismic Margins Review," May 2, 1990.

12. Letter from P. R. Davis (PRD Consulting, Chairman, Hatch Seismic Margin Assessment Peer Review Group) to Dan Guzy (RES Co-chairman Seismic Design Margins Working Group) with enclosed report, "Hatch SMA Peer Review Group Final Report: Evaluation of the Application of the NRC and EPRI Seismic Margins Methodologies," May 3, 1990.

13. Electric Power Research Institute, "Seismic Verification of Nuclear Plant Equipment Anchorage," Revision 1, Volume 1: Development of Anchorage Guidelines, and Volume 3: EPRI/Blume Anchorage Computer Program (EBAC), EPRI NP5228-SL, Palo Alto, CA, June 1991.

3.1 SEISMIC MARGINS METHOD

3.1.1 REVIEW OF PLANT INFORMATION, SCREENING, AND WALKDOWN

3.1.1.1 Site Description

Plant Hatch is located in the Coastal Terraces subprovince of the Atlantic Coastal Plain physiographic province. The site is underlain by approximately 4000 ft of relatively unconsolidated Mesozoic and Cenozoic sands, gravels, clays, marls, claystones, sandstones, and limestones. These strata, overlying basaltic basement rock of pre-Cretaceous age, dip and thicken seaward. No structural features affect the material underlying the site. No major or minor fault zones are near the site, nor were any local faults discovered during field mapping, exploratory drilling, and construction.

The Hawthorn Formation of Miocene to Pliocene age, which consists primarily of sand, clay, and cemented sand and clay layers, is the foundation-bearing stratum for the major plant structures. There are no zones of deformation, alternation, or weakness within the Hawthorn Formation.

The site is underlain by both confined and unconfined aquifers. Local and regional groundwater conditions have not been altered by construction and operation of the plant.

The site is located in a region of infrequent seismic activity. No earthquakes occurring within 200 miles of the site produced Modified Mercalli intensities at the site greater than VI. This includes earthquakes in the Charleston, South Carolina, area, approximately 150 miles from the site. Historically, reported earthquakes occurring in other areas have not produced intensities greater than VI at the site. The design basis earthquake (DBE) was conservatively selected as Modified Mercalli intensity VII.

The local topography of the site is relatively flat to gently rolling. The elevation of the plant area is approximately 125 to 145 ft mean sea level (msl). The average river elevation is approximately 71 ft msl. The finished grade elevation around the plant is approximately 129 ft msl. Elevations within an approximate 15-mile radius do not exceed an estimated 310 ft msl. The confluence of the Oconee and Ocmulgee Rivers that forms the Altamaha River is approximately 12 air miles to the west. The two rivers lie in an open, shallow valley that broadens downstream from the site.

3.1.1.2 Seismic Category I Structures

Safety-related structures that are essential for safe shutdown of the reactor and control of radioactive materials are classified as Seismic Category I. Category I structures include the primary containment, the reactor buildings, the diesel generator building, the control building, and the intake structure. These structures are shown in figure 3.1-1.

The following structures were included in the Plant Hatch Unit 1 and Unit 2 seismic margin assessments (SMAs):

A. Intake Structure

The intake structure, shared by Units 1 and 2, is a reinforced concrete structure housing the plant service water (PSW) system and the residual heat removal service water (RHRSW) system pumps.

B. Diesel Generator Building

The diesel generator building is a reinforced concrete structure housing the DGs, local control panels, and emergency switchgear for both Units 1 and 2. Each DG and its control panel is physically separated from the other DGs and associated switchgear.

C. Control Building

The control building is a reinforced concrete structure with steel framing above 164 ft and houses the common control room for Units 1 and 2, station batteries, and associated auxiliaries.

D. Reactor Buildings

The Plant Hatch Unit 1 and Unit 2 nuclear steam supply systems are General Electric single-cycle boiling water reactors (BWRs) 4 with Mark I containments incorporating the drywell/pressure suppression concept.

The reactor buildings are two separate reinforced concrete structures that serve as secondary containments. The area above the refueling floor is common to both units. The exterior walls above the refueling floor consist of structural steel columns and prefabricated concrete panels. A 3-in. seismic gap separates the two reactor buildings.

3.1.1.3 <u>Plant Seismic Design Basis</u>

The following paragraphs summarize the seismic design basis for Plant Hatch structures and equipment.

3.1.1.3.1 Seismic Input

The Unit 1 design spectrum is a Housner-type spectrum with a peak ground acceleration (pga) of 0.08 g for the operating basis earthquake (OBE) and 0.15 g for the DBE. Figures 3.1-2 and 3.1-3 are plots of the Unit 1 OBE and DBE design ground response spectra. Unit 1 Seismic Category I structures that are shared with Unit 2 are also analyzed and designed to meet Unit 2 seismic design bases. The shared structures of interest are the control building, the diesel generator building, and the intake structure.

The Unit 2 design spectrum is a modified Newmark spectrum with the same peak ground acceleration values as the Unit 1 seismic design bases. The Unit 2 OBE and DBE design ground response spectra are shown in figures 3.1-4 and 3.1-5, respectively. The Newmark spectra have higher spectral amplification than the Housner spectra. The design spectra are specified as free-field ground spectra at grade.

3.1.1.3.2 Design and Analysis of Structures

The Seismic Category I structures that are part of the SMA are founded on soil. The input motion of soil-structure interaction (SSI) analyses was conservatively placed at the foundation level of each structure. A refraction survey of the site was performed in 1968 only to obtain a shear wave velocity for use in the SSI analyses. A single (average) composite value of shear wave velocity of 2450 ft/s was obtained for soil characterization. This value was used in the original SSI analyses. The refraction survey results were not correlated to the information from the many soil borings taken in the plant area. The original SSI approach was based on the elastic half-space theory which, at the time of the original analyses, was the standard approach used to address SSI. Later reviews, as discussed in section 3.1.4.7, indicated that a lower composite value of shear wave velocity was appropriate.

For the original seismic design, two-dimensional, horizontal, lumped-mass, stick models with soil springs were developed for each Seismic Category I structure. A vertical dynamic analysis was not performed. Response spectrum analyses were performed to calculate seismic forces on structures. Time history analyses of the building models were performed to obtain horizontal in-structure response spectra (IRS). In 1985, the IRS were reanalyzed using synthetic time histories that more closely enveloped the plant design response spectra obtained using the two-dimensional models. The response spectra were smoothed, and the peaks were broadened by $\pm$ 10 percent in the frequency range.

Forces resulting from one horizontal earthquake direction were combined absolutely with the forces resulting from the vertical earthquake. The maximum damping values used for the OBE were 3.0 percent for structures and 4.5 percent for soils, and for the DBE, 5.0 percent for structures and 5.5 percent for soils. A review of the seismic analysis (Reference 1, Appendix C) showed these calculated responses to be very conservative. This conservatism was principally the result of the treatment of SSI radiation damping effects.

Concrete structures are designed to meet the requirements of American Concrete Institute (ACI) Standard 318-63 (Reference 2), although special reinforcement detailing requirements were followed to produce ductile behavior. Steel structures are designed in accordance with the American Institute of Steel Construction specifications (Reference 3).

The Units 1 and 2 reactor buildings and the control building are reinforced concrete shear wall structures with rigid base mats. These buildings have steel frame superstructures with cross bracing to support bridge cranes and roof loads. The diesel generator building and the intake structure are concrete shear wall structures with rigid base mats. As stated above, all these structures are supported on soil.

3.1.1.3.3 Heating, Ventilation, and Air-Conditioning and Cable Tray Supports

Heating, ventilation, and air-conditioning (HVAC) and cable tray supports are designed by the response spectrum method to withstand the calculated seismic loads using the IRS corresponding to the attachment locations of the supports. The simultaneous application of the horizontal and vertical components which created the highest stresses was used to design the supports. Maximum stresses are limited to 90 percent of minimum yield.

3.1.1.3.4 Mechanical Equipment and Piping

All safety-related mechanical equipment was qualified by analysis for the OBE and the DBE, with a limited number of components qualified by testing. Pressure boundaries of safety-related systems were evaluated to the criteria of the appropriate American Society of Mechanical Engineers (ASME) Code sections. Large-bore piping was dynamically analyzed for seismic response. Seismic Category I piping 2 in. and smaller is supported using standard spacing to produce natural frequencies above the earthquake frequency range of interest. In some cases, a simple conservative static analysis was performed.

3.1.1.3.5 Electrical and Control Equipment

Electrical equipment was primarily qualified by shake table testing of prototype components with input consisting of harmonic sine beat, or similar motions compatible with the appropriate support motion. Unit 2 electrical equipment was qualified per the Institute of Electrical and Electronics Engineers (IEEE) 344-1971 (with amendments) (Reference 4). Unit 1 equipment was not qualified per IEEE 344-1971, since most of the equipment was ordered prior to issuance of IEEE 344-1971. However, Unit 1 equipment is similar to the Unit 2 equipment qualified to IEEE 344-1971. Some passive electrical equipment for both units was qualified by either dynamic analysis or the conservative static coefficient method.

3.1.1.3.6 Seismic Spatial Interaction

Safety systems are separated by structural barriers to provide redundancy in the event of flood or fire. Nonsafety-related equipment whose failure could affect Seismic Category I equipment is designed to meet Seismic Category II/I criteria, ensuring that structural failure will not occur.

3.1.1.4 Qualifications of Seismic Margin Assessment Team

The Seismic Review Team (SRT) members represent the personnel responsible for applying experience and judgment in the implementation of the combined EPRI SMA methodology and Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP). SRT members for Plant Hatch have many years of formal and practical experience in the field of structural and seismic design and analysis. The combined qualifications of the SRT members meet the requirements of both methodologies. All SRT members successfully completed the SQUG and Individual Plant Examination (IPE) Add-On Walkdown Training Course. A brief summary of the qualifications of each seismic capability engineer is given below.

Phillip W. Garrett

Mr. Garrett is a principal engineer in the Consulting and Testing Services (CATS) Department at Southern Company Services, Inc. (SCS). He has a B.C.E. degree from Auburn University and an M.S. degree in civil engineering from the University of Alabama at Birmingham. Mr. Garrett possesses more than 16 years' experience in structural analysis and design of structures for electric utilities. He has developed expertise in the areas of vibration analysis and modal testing, including in-situ vibration and modal testing of fans, motors, pumps, and structures. Mr. Garrett has extensive experience in seismic analysis of civil structures and subsystems and seismic qualification of equipment. He is a registered Professional Engineer in the State of Alabama.

Billy R. Goforth, Jr.

Mr. Goforth is a Professional Engineer in the CATS Department at SCS. He earned B.S. and M.S. degrees in structural engineering at the University of Alabama at Birmingham. Mr. Goforth has more than 22 years' experience in the structural analysis and design of structures for electric utilities. He has performed work for many plants in the Southern electric system, including all three nuclear sites. Mr. Goforth has extensive experience performing seismic analysis, as well as review and approval of vibration and modal testing of equipment, supports, foundations, and structures. He has worked in his current position for several years, providing and approving seismic qualification of Class 1E equipment. Mr. Goforth's experience includes modern plants which meet Standard Review Plan (SRP) requirements, as well as older, SQUG plants. He is a registered Professional Engineer in the States of Alabama and Georgia.

Thomas B. Lantrip

Mr. Lantrip is a senior engineer in the CATS Department at SCS. He earned a B.S. degree in civil engineering from the University of Alabama and completed graduate course work at the University of Alabama at Birmingham and the University of California. Mr. Lantrip has more than 11 years' experience in structural analysis and design of structures for electric utilities, including extensive experience in seismic analysis of nuclear power plant structures and seismic qualification of equipment. He has expertise in the areas of vibration analysis and modal analysis testing of rotating equipment, balancing of high-speed machinery, and in-situ strain testing for nuclear and nonnuclear operating equipment and facilities. Mr. Lantrip is a registered Professional Engineer in the State of Alabama.

Donald P. Moore

Mr. Moore is a consulting engineer in the CATS Department at SCS. He has a B.S. degree in civil engineering from the University of Alabama and an M.S. degree in engineering from the University of Alabama at Birmingham. In addition to being an SRT member, Mr. Moore served as the technical director for the Plant Hatch SMA/Unresolved Safety Issue (USI) A-46 project and was responsible for all technical/seismic aspects of the project. Mr. Moore has more than 23 years' experience in the field of structural engineering with specific emphasis on structural dynamics. Mr. Moore has worked on a broad range of structural engineering activities, including responsibility for the design of structures for nuclear and fossil power plants; seismic analysis and design of structures and supports; analysis and design of foundations for rotating equipment such as coal crushers, fans, and pumps; reviewing and approving seismic qualification reports for Seismic Category I equipment; solving structural vibration problems in operating power plants; and developing and implementing the structural dynamic testing capabilities for SCS. He is also an active member of several American Society of Civil Engineers (ASCE), ACI, and ASME committees that relate to structural dynamics and structural code activities. He was a peer reviewer for the SQUG Walkdown Screening and Seismic Evaluation course, a technical reviewer of EPRI Report TR-102180, "Guidelines for Estimation or Verification of Equipment Natural Frequency," and has been a member of the SQUG Steering Group since 1993. Mr. Moore is a registered Professional Engineer in the State of Alabama.

Keith D. Wooten

Mr. Wooten is the USI A-46/IPEEE-Seismic Program Manager for the Hatch, Farley, and Vogtle nuclear plants. He earned a B.S. degree in civil engineering from Auburn University and an M.B.A. from the University of Alabama at Birmingham. In addition to being the program manager, Mr. Wooten also served as an SRT member for the Plant Hatch SMA. Mr. Wooten has more than 14 years' experience in the analysis, design, and construction of structures and components for large industrial and nuclear facilities, which includes an extensive knowledge of Plant Hatch and its design and analysis basis and requirements. He is a registered Professional Engineer in the State of Alabama.

3.1.1.5 Qualifications of Systems Engineers

Plant Hatch mechanical systems engineers developed the Safe Shutdown Equipment List (SSEL) and assisted the SRT as required. Plant Hatch electrical engineers performed the SQUG relay chatter evaluation, worked with the mechanical systems engineers to develop the SSEL, and assisted the SRT as required. The systems engineers for the Plant Hatch IPEEE project possess considerable experience in the design and analysis of mechanical and electrical systems for nuclear power plants. A brief summary of the qualifications of each system engineer is given below.

Warren T. Barr

Mr. Barr is a senior engineer in the Plant Hatch project group at SCS. He holds a B.S. degree in mechanical engineering from Auburn University. Mr. Barr has more than 20 years' experience in mechanical design and engineering. His experience is primarily in nuclear piping design, nuclear unit startup, system design and operation, and field walkdown and problem resolution. Mr. Barr was involved with the initial startup at Plant Hatch and possesses an in-depth knowledge of the design basis of Plant Hatch mechanical systems. Mr. Barr has attended the SQUG SSEL systems training course.

Peter H. Wells

Mr. Wells is Plant Operations assistant general manager at Plant Hatch. He has a B.S. degree in nuclear engineering from the Georgia Institute of Technology. Mr. Wells served as a United States Naval Officer-PWR Nuclear Engineer qualification for 5 years and was stationed aboard the U.S.S. Dwight D. Eisenhower and the U.S.S. MacDonough. He is a licensed BWR senior reactor operator.

W. Scott Walker

Mr. Walker is a senior engineer in the Plant Hatch project group at SCS. He has a B.S. degree in mechanical engineering from the Georgia Institute of Technology. Mr. Walker has 12 years' experience in design engineering, 10 years of which centered upon the Plant Hatch project. His experience is primarily in piping design, system design and operation, and field walkdown and problem resolution. Through his work in the Plant Hatch project group, he has developed an in-depth knowledge of the design basis of the plant's mechanical systems. Mr. Walker is a registered Professional Engineer in the State of Alabama. He attended the SQUG SSEL systems training course.

Larry D. McWhorter

Mr. McWhorter is a senior engineer in the Plant Hatch project group at SCS. He has a B.E.E. degree from Auburn University, and his work experience totals more than 26 years in utility experience, including work involving power delivery, electrical equipment procurement substation design, and other electrical design activities. He has a total of 5 years' experience on the Plant Hatch support project designing electrical power and control systems. Mr. McWhorter is a member of the IEEE and is a registered Professional Engineer in the States of Alabama, Florida, Georgia, and Mississippi. He attended the SQUG electrical systems training course.

James E. Smith

Mr. Smith is a senior designer in the Plant Hatch project group at SCS. He has more than 19 years' experience in the electrical design of power and control systems for utility projects, with more than 5 years devoted to the Plant Hatch project group. Mr. Smith's responsibilities include the design of control systems and protective relaying for system power plants. He attended the SQUG electrical systems training course.

3.1.1.6 Prescreened Structures and Equipment

Many items were prescreened from a detailed SMA, based on their seismic ruggedness to withstand earthquake forces at the Plant Hatch seismic margin earthquake (SME) level. The prescreening was accomplished using the criteria in tables 2-3 and 2-4 of the EPRI SMA methodology (Reference 5) in conjunction with the seismic capability walkdown. Because the evaluation was combined to meet both IPEEE and USI A-46 requirements, many components required evaluation beyond the requirements of table 2-4. This section contains a discussion of the items that were prescreened per tables 2-3 and 2-4, as well as equipment that required additional evaluation per the SQUG GIP.

The items that were prescreened from the Plant Hatch SMA at a high-confidence-of-low-probability-of-failure (HCLPF) capacity of at least 0.3 g pga are described below.

A. Primary Containment (Drywell)

The drywell is a steel pressure vessel with a spherical lower portion and cylindrical upper portion (figure 3.1-6). Below elevation 111 ft 6 in., the drywell is completely encased in reinforced concrete. Cylindrical shear keys approximately 8 in. high are attached to the drywell to resist seismic and other lateral forces (figure 3.1-7). Reinforcement is welded to the shear keys for structural continuity. Between elevation 111 ft 6 in. and elevation 114 ft 6 in., the drywell is lined with concrete on the inside and a sand pocket on the outside. Between elevation 114 ft 6 in. and elevation 200 ft 1-1/2 in., the drywell is enclosed with concrete, which also performs the function of shielding against radiation. In this area, there is a nominal 2-in. gap between the reactor pressure vessel (RPV) and the concrete enclosure. Above elevation 200 ft 1-1/2 in., the concrete is provided to shield the drywell and does not back the drywell for structural purposes. The drywell is designed for a combined loading of design basis accident pressure, in addition to the DBE. Based on the description above and the requirements of table 2-3 of the EPRI SMA methodology, the drywell is prescreened at a HCLPF capacity ≥ 0.3 g pga. The torus, which is an integral part of the primary containment, is addressed in section 3.1.4.5.

B. Drywell Internal Structures

The SMA screening guidelines state that containment internal structures can be screened out if they are designed to a DBE level of ≥ 0.1 g pga. The Plant Hatch drywell internals are designed to a DBE level of 0.15 g pga, in addition to all forces associated with a postulated loss-of-coolant accident (LOCA). For the SMA, a seismically induced large-break LOCA is not considered a credible event. The exclusion of LOCA loads in the SMA review, including the conservatism inherent in the design, results in a HCLPF capacity > 0.3 g pga.

C. Shear Walls, Footings, and Containment Shield Walls

Shear walls, footings, and containment shield walls are Seismic Category I structures, which are designed for a safe shutdown earthquake (SSE) of 0.15 g pga. This exceeds the value of 0.1 g pga required to prescreen these structures using table 2-3 of the EPRI SMA methodology. Based on these guidelines and the SRT review, these structures are adequate for a HCLPF capacity of at least 0.3 g pga.

D. Seismic Category I Concrete and Steel Frame Structures

The Seismic Category I buildings of interest for the SMA include the reactor buildings, control building, intake structure, and diesel generator building. These structures are designed for a DBE of 0.15 g pga using dynamic analysis techniques. An SMA was required for the reactor building roof structure (see section 3.1.4.4) because of a unique design feature.

The SRT reviewed the original plant design criteria, loading conditions, and typical construction details to ensure that the Seismic Category I structures can be screened out. The soil structure interaction (SSI) evaluation performed for the SMA indicated that the maximum response of the reactor buildings and the control building is in the 1- to 3-Hz range, which was not predicted in the original plant design. To ensure the structures could be prescreened with this difference in response, a sample of the maximum SME responses was compared to the original DBE responses for these buildings. Based on this comparison and the original plant design and construction, the SRT concluded there is no concern with these structures surviving the SME. Therefore, based on the SMA screening guidelines and review by the SRT, Plant Hatch Seismic Category I structures have a HCLPF level of at least 0.3 g pga.

E. Dams, Levees, and Dikes

Any failure of dams, levees, and dike (seismic or otherwise) is discussed in chapter 5 of this report.

F. Nuclear Steam Supply System (NSSS) Primary Coolant Systems and Supports

The NSSS primary coolant systems at Plant Hatch Units 1 and 2 consist of the reactor vessels, recirculation loops with recirculation pumps and valves, main steam piping from the reactor vessel to the first isolation valve, and feedwater piping from the reactor vessel to the first isolation valve. The primary coolant systems are designed to withstand a combined loading condition consisting of SSE and pipe break loads.

Intergranular stress corrosion cracking (IGSCC) has been identified in portions of the primary coolant system piping at Plant Hatch Units 1 and 2. IGSCC is a form of cracking occurring along the grain boundaries of certain stainless steel materials, and is normally found in the heat-affected zone of butt-welded joints along the piping. The three elements that, in combination, cause IGSCC are as follows:

- A susceptible (sensitized) material.

- High tensile stress in the cracking region.

- Water chemistry that promotes initiation and growth of cracks.

The current NRC position regarding IGSCC in BWR austenitic stainless steel is documented in NUREG-0313, Revision 2 (Reference 6).

Georgia Power Company addressed the IGSCC concern for Plant Hatch Unit 1 by using a combination of induction heating stress improvement (IHSI) to limit additional cracked weldments and weld overlays to repair cracked weldments. IHSI consists of heating the outside of the pipe with induction coils to controlled temperatures while cooling water is circulated inside the pipe. The result is to alter the residual stress pattern and put the inner part of the pipe wall in compression, thus inhibiting crack initiation. IHSI is considered a viable mitigation process by the NRC, as stated in NUREG-0313.

Weld overlay reinforcement consists of applying weld metal over the weld and for a specified minimum distance beyond the weld on both sides. All overlays currently installed on Plant Hatch Unit 1 satisfy the requirements of the NRC position stated in NUREG-0313 and are considered to be Category E weldments.

To address IGSCC for Plant Hatch Unit 2, a complete replacement of the recirculation piping and other affected primary coolant system piping, including stainless steel portions of residual heat removal (RHR) pipe and reactor water clean-up pipe, was implemented. Type 316 nuclear grade stainless steel, which has demonstrated a superior resistance to IGSCC in the BWR environment, was used as the replacement pipe material.

To provide additional protection against IGSCC, the hydrogen water chemistry (HWC) system was installed to eliminate, in a timely manner, the chemical conditions in the recirculation water that allow IGSCC. BWRs use high purity water as the primary recirculation coolant in the direct cycle production of steam. Because of radiolytic decomposition of water in the core, the reactor and recirculation water contain a steady-state concentration of 100 to 300 ppb of dissolved oxygen. This amount of oxygen is sufficient to cause IGSCC of highly stressed, sensitized stainless steel.

The HWC system injects hydrogen into the feedwater to mitigate IGSCC in the recirculation piping and regions of the reactor vessel. The injected hydrogen forces a reduction in dissolved oxygen within these areas and lowers the radiolytic production of the hydrogen and oxygen in the vessel core region.

Based on the design loading combination of the SSE and pipe break loads, in addition to the resolution of IGSCC, the Units 1 and 2 NSSS primary coolant systems and piping are prescreened at a HCLPF capacity of at least 0.3 g pga.

An SMA of the reactor vessel and internals was performed and is discussed in section 3.1.4.6.

G. Diaphragms

The floor systems used in Seismic Category I structures at Plant Hatch are designed to support dead loads, equipment loads, laydown loads, piping loads, live loads, and vertical seismic loads. Additionally, floor systems are designed to resist moments caused by tornado loads on adjacent exterior walls and the horizontal shear caused by lateral seismic forces. Reinforcement in the diaphragms is detailed to produce ductile-type behavior. Load paths around cutouts in slabs are considered in the design and detailing of the reinforcement. Based upon these design parameters and the requirements of the EPRI SMA methodology, the diaphragms are adequate for a HCLPF capacity of at least 0.3 g pga.

H. Control Rod Drive (CRD) Housings and Mechanisms

The EPRI SMA methodology requires that the CRD housings have lateral seismic support to obtain a HCLPF level of at least 0.3 g pga. The CRD housings at Plant Hatch have lateral restraints (figures 3.1-8 and 3.1-9) designed to provide lateral support for the vertically cantilevered CRD assemblies. Therefore, for the SMA, the CRD housings and mechanisms are prescreened at a HCLPF capacity of at least 0.3 g pga.

I. Piping

The EPRI SMA methodology states that piping systems in nuclear power plants have HCLPF capacities in excess of 0.5 g pga subject to a walkdown of representative safety-related piping. A walkdown of the high-pressure coolant injection (HCPI) system piping was conducted by one SRT during the seismic capability walkdown. In addition, safety-related piping was observed by the SRTs in areas containing SSEL components. When observing piping, the SRT considered the following items, as outlined in the EPRI SMA methodology:

- Non-ductile joints.

  Plant Hatch Seismic Category I piping is welded and does not contain any threaded, Victaulic, or other mechanical friction-type connections.

- Cast iron pipe.

  Cast iron pipe is not used for Seismic Category I piping at Plant Hatch.

- Branch lines.

  Branch lines from Seismic Category I piping were found to have adequate flexibility.

- Piping connections.

  The connections of pipe into equipment anchor points at Plant Hatch are constructed such that excessive nozzle loads will not occur.

- Valves.

  Valves observed during the piping walkdown have adequate clearance to avoid interaction with structures, components, or other subsystems. Additionally, active valves on the SSEL were inspected during the seismic capability walkdowns; results are summarized in section 3.1.5.2.

- Multiple support failure.

  Seismic Category I piping supports are seismically designed. No potentially weak supports were identified during the walkdown.

  No problems were identified with the flexibility of piping systems attached to equipment with vibration isolation systems.

- Piping across seismic gaps.

  Piping details across seismic gaps observed during the walkdown are designed with adequate flexibility to accommodate building motions. For example, piping from the reactor building torus area to the HPCI room is routed through sleeves with a sufficient gap between the sleeve and the pipe to allow movement.

- Seismic interaction.

  Potentially greater displacement could occur as a result of the increased seismic load of the SME on the piping. During the walkdown, no conditions in which increased displacements would cause an interaction concern were identified.

The EPRI SMA methodology states that, based on numerous probability risk assessments, it is assumed the SME will not cause a large-break LOCA. The results of the SMA piping walkdown serve to reinforce the assumption that a large-break LOCA will not occur at Plant Hatch for the level of the SME under consideration. No case that indicated a potential for a large-break LOCA was found.

Based on the walkdown results described above, Plant Hatch Seismic Category I piping is considered satisfactory for a HCLPF capacity of 0.3 g pga or greater. Buried piping is addressed in section 3.1.4.8.

J. Heating, Ventilation, and Air-Conditioning Ducting and Dampers

HVAC ducts associated with success path components are in the main control room environmental control, the battery room emergency exhaust systems located in the control building, and the HPCI equipment area cooling system located in the reactor building. The control building systems were walked down, and the duct supports were determined to be adequate. The Unit 1 HPCI room cooler duct supports were adequate; however, the Unit 2 HPCI room cooler duct supports did not meet the screening criteria and, therefore, have been modified to obtain a HCLPF level of at least 0.3 g pga.

Cooling units used for the RHR/core spray (CS) pump rooms are located in these rooms and blow directly into them without using any independently supported ducting. The diesel building and intake structure HVAC systems have air intake louvers in the walls and ventilating fans on the roof above the affected rooms and, therefore, do not require ducting.

Dampers in the diesel generator building were evaluated and found to be acceptable for a HCLPF level of at least 0.3 g pga. These dampers are included on the SSEL.

K. Cable and Conduit Raceways

Appendix A of the EPRI SMA methodology states that cable and conduit raceways have HCLPF capacities of at least 0.3 g pga. The EPRI SMA methodology suggests that example cable trays be inspected during the walkdown to verify that they are adequately supported. As part of the requirements of USI A-46, the SQUG GIP requires that an extensive evaluation of cable and conduit raceways be performed. This evaluation, which is discussed below, adequately addresses the requirements of the EPRI SMA methodology.

The Plant Hatch Units 1 and 2 cable tray and conduit raceway review consisted of two parts:

1. A plant walkdown in which the raceways were evaluated against the SQUG GIP inclusion rules and walkdown guidelines.

2. An analytical check of selected worst-case supports using the limited analytical review guidelines found in section 8 of the SQUG GIP.

Portions of raceway systems which did not pass these screening guidelines were classified as outliers and evaluated separately using alternative methods.

In support of previous work at Plant Hatch, extensive walkdowns and analyses were performed on the cable and conduit systems. Data accumulated during these walkdowns were beneficial to the USI A-46 review. Based on USI A-46 walkdowns and previous, non-USI A-46 walkdowns, worst-case analysis samples were identified. After grouping supports of a similar configuration, several cases were chosen for the limited analytical review. All safety-related raceway systems, except for systems located in the drywell, were evaluated during the walkdown. The documentation derived from a previous, non-USI A-46 walkdown was used to satisfy the SQUG GIP cable and conduit raceway walkdown requirements for the drywell.

Because the USI A-46 evaluation was significantly more extensive than that required by the SMA, the raceway evaluation followed the guidance furnished in the SQUG GIP. Detailed results of the walkdown and identification of all outliers associated with the walkdown and the analytical review are included in the Plant Hatch Unit 1 USI A-46 Summary Report (Reference 7). The examination of cable and conduit raceway supports confirmed the HCLPF of at least 0.3 g pga assumed by the EPRI SMA methodology.

The following discussion highlights where additional evaluations were performed beyond that required by table 2-4 of the EPRI SMA methodology. Any outliers found were corrected.

A. Table 2-4 of the EPRI SMA methodology requires no specific evaluation for active valves. During the walkdown, all active valves were screened using the GIP requirement of checking operator height and weight versus pipe diameter to ensure they are within the experience database described in Appendix B of the SQUG GIP.

B. The EPRI SMA methodology does not require a special evaluation of vertical pumps for peak spectral accelerations up to 0.8 g. The plant service water (PSW) pumps required an evaluation addressing the SQUG GIP caveat concerning the limitation of the casing length being less than a 20-ft cantilever.

C. For fans, air handlers, chillers, and air compressors, the EPRI SMA methodology specifies that only units supported on vibration isolators require an evaluation of anchorage. The anchorage of all such equipment included in the Plant Hatch SMA was evaluated against the SQUG GIP criteria, whether or not the equipment was on vibration isolators.

D. For electrical power distribution panels, cabinets, switchgear, motor control centers, and instrumentation and control panels and racks, the EPRI SMA methodology requires a limited walkdown to verify that such equipment is securely anchored to the floor or wall and that the instruments are properly attached to the cabinets. A full walkdown of the anchorage of all electrical equipment on the chosen success paths was performed, including appropriate anchorage calculations, to satisfy the SQUG GIP.

3.1.1.7 Seismic Margin Walkdown

3.1.1.7.1 Equipment Walkdown Procedures

The seismic capability walkdown for SSEL components was performed by enveloping the requirements of the EPRI SMA methodology and the SQUG GIP (Reference 8). Every item on the walkdown list was inspected and evaluated, and a walkdown data sheet was completed for each item. The walkdown concentrated on the following areas: seismic capacity, screening caveats, anchorage, seismic spatial interaction, and internal flooding.

During the seismic capability walkdown, the Screening Evaluation Work Sheets (SEWS) found in the SQUG GIP were used for most components instead of the forms in the EPRI SMA methodology. The GIP forms (SEWS) were chosen because they were more efficient in enveloping both the SMA and USI A-46 programs and were preferred by the SRT. The screening caveats for each equipment class were listed on the SEWS and were addressed as part of the walkdown. A summary of the walkdown results of the screening caveats for each SSEL component is given in the Screening Verification Data Sheets in Appendixes C and D of this report. A sheet was added to the SEWS to address the relay walkdown, which is required for USI A-46, and seismic interaction for flooding. The flooding check is required for SMA, but is not required for USI A-46.

The seismic capacity bounding spectrum, shown in figure 4-2 of the SQUG GIP, was used to represent equipment seismic capacity. The seismic demand was based on the seismic margin earthquake (SME) ground response spectrum (GRS). (See section 3.1.3.1.) Since the SME GRS are enveloped by the bounding spectrum, all Plant Hatch equipment included in the experience data base and located below approximately 40 ft above grade was considered adequate. For equipment located more than 40 ft above grade, the SME IRS were compared to the bounding spectrum multiplied by a factor of 1.5. USI A-46 requires checking seismic capacity versus demand; however, the EPRI SMA methodology does not require this process. The SMA capacity versus demand check is essentially performed by using screening tables 2-3 and 2-4 of the EPRI SMA methodology. If only SMA is being performed, this section of the SEWS is not required.

Equipment anchorage was inspected and evaluated in more detail than the EPRI SMA methodology requires so the more exacting requirements of the SQUG GIP could be met. The anchorage section of the SEWS addresses the requirements of both programs.

In contrast to the SQUG GIP, which requires a thorough inspection of all success path components, the EPRI SMA methodology requires a 100-percent "walk by" for all reasonably accessible success path components. A 100-percent "walk by" is not a complete inspection of each component, and it does not require an electrician or other technician to deenergize and open cabinets or panels for a detailed inspection of all components. Also, if the SRT has a reasonable basis for assuming that a group of components is similar and is similarly anchored, the EPRI SMA methodology requires the inspection of only one component from the group.

Because Plant Hatch combined the USI A-46 and IPEEE resolution efforts, the equipment walkdowns included a thorough inspection of each success path component to comply with the SQUG GIP. An electrician opened electrical components, if possible, to allow inspection of internal elements. Anchor bolts or welds were also inspected, if possible, for each component. This inspection required opening or removing part of most electrical cabinets to access the anchorage. In a few cases, part of the anchorage was inaccessible, but enough could usually be inspected to verify the seismic adequacy of the component anchorage by taking credit for just the accessible anchors.

The inspection of anchor bolts included the following assessments:

- Bolt tightness check, for expansion bolts only, to ensure the bolt was properly installed and would carry the load.

- Washer placement check to ensure washer placement was between the equipment base and the bolt head or nut.

- Bolt spacing check to ensure bolt spacing is at least 10 bolt diameters; i.e., the distance between a bolt and any free concrete edge is at least 10 bolt diameters.

- Soundness of concrete.

- Bolt eccentricity which could cause excessive bolt bending stress.

Embedment length was checked by measuring bolt projection for expansion anchors and reviewing drawings for cast-in-place anchors.

An electrician accompanying the SRTs performed a tightness check of all accessible expansion anchor bolts to ensure the bolts were properly installed, as required by the SQUG GIP. (This check was not intended to test bolt capacity.) The tightness check was performed using an open-end wrench to apply torque by hand until the bolt or nut was "wrench-tight." If a bolt did not tighten and resist the applied torque, it was considered unacceptable and was not used in the analysis.

Welded anchorages were inspected to determine the size and quality of each weld. Deficiencies, such as weld burn-through, were specifically noted and included in any analysis.

While completing the SEWS in the field, the SRTs sketched the pertinent anchorage details to aid in the evaluation. These sketches included the anchorage layout and dimensions, the component dimensions, approximate location of the center of gravity, bolt type, bolt diameter, weld size, and nonstandard details that should be considered in the analysis.

Calculations were performed as required during the walkdown to assess the adequacy of equipment anchorage. These calculations were generally performed in a manner that quickly indicated whether the anchorage capacity was greater than the seismic demand. These calculations were not intended to be rigorous, but were solely made to provide a basis for the SRT to judge whether the anchorage was acceptable. Appendix C of the SQUG GIP was used as the basis for assessing anchorage strength. Anchor loads were determined by hand analysis or, in most cases, by using the EPRI/Blume Anchorage Computer Program (EBAC) (Reference 9).

The component weight used for anchorage analysis was determined in one of three ways. In some instances, the weight was given on the name plate, which was usually the case with transformers. If the weight was not inscribed on the name plate, a conservative estimate of equipment weight was calculated using the guidelines provided in Appendix C, table C-7, of the SQUG GIP. When a more precise weight was required, the vendor drawings for the component were checked.

During the walkdown, the SRT estimated the natural frequency of each component. The component seismic demand was determined from SME IRS at the highest acceleration value corresponding to the component natural frequency or greater. If the natural frequency could not be accurately estimated, or was estimated to be very low, the peak seismic acceleration was taken as the seismic demand.

For components that are welded or attached by other means to embedded plates or channels, the capacity of the embedded item was evaluated by using drawings. The stiffness and strength of the component base were also checked to ensure a sufficient load path existed. Drawings were reviewed to check equipment pads to ensure they were constructed of reinforced concrete and the pads were adequately anchored to the floor slab.

During the walkdown, seismic spatial interaction effects, including proximity, Seismic Category II/I, and internal flooding, were evaluated for all components. Proximity refers to the potential adverse effect from the seismic motion of one component into another component or structure. Seismic Category II/I is the potential impact of a Category II supported component on a Category I component. The internal flooding walkdown focused on internal flooding sources, such as tanks and piping, that could potentially rupture due to seismic interaction or poor anchorage. The results of the internal flooding evaluation are discussed in section 3.1.4.11.

3.1.1.7.2 Subsystem Walkdown Procedures

The EPRI SMA methodology states that piping systems have HCLPF capacities > 0.5 g pga subject to a walkdown of representative safety-related piping. Therefore, a walkdown was performed for a typical safety-related piping system by one SRT during the seismic capability walkdown. The walkdown methodology and results are discussed in section 3.1.1.6.I.

The EPRI SMA methodology also requires a walkdown of representative HVAC ducting systems. Very little ducting is associated with HVAC systems on the SSEL; thus, because of the relatively small scope, all HVAC ducting associated with SSEL components was evaluated. In addition, any HVAC system component, including ducting, in the vicinity of SSEL components was evaluated for potential seismic interaction.

According to the EPRI SMA methodology, all HVAC systems have a HCLPF capacity of at least 0.3 g pga. As suggested by the EPRI SMA methodology, HVAC ducting, within the scope described above, was evaluated to ensure the following:

- Equipment and subsystem supports were adequately anchored.

- Ducting that spans between buildings was adequately designed to accommodate relative displacements.

- No other failure modes existed.

The results of this walkdown and the evaluation are summarized in section 3.1.1.6.J.

Cable and conduit raceways are, in general, judged to have a HCLPF capacity of at least 0.3 g pga by the EPRI SMA methodology. An inspection of sample raceway systems is also suggested by the EPRI SMA methodology. An extensive walkdown inspection and evaluation were performed for all safety-related cable and conduit raceway systems for Plant Hatch Units 1 and 2 as part of the USI A-46 resolution, and the results are summarized in section 3.1.1.6.K.

3.1.1.7.3 Walkdown Results

The Unit 1 Seismic Review SSEL of components requiring a seismic capability walkdown includes 452 components. The Unit 2 Seismic Review SSEL includes 350 components. The Unit 1 SSEL contains several common systems not included on the Unit 2 SSEL. Appendixes A and B of this report contain copies of the seismic review SSEL for Units 1 and 2, respectively. Results of the seismic capability walkdown for each component are shown on the Screening Verification Data Sheet (SVDS), Appendix C of this report for Unit 1 and Appendix D for Unit 2.

3.1.2 SYSTEMS ANALYSIS

3.1.2.1 Identification of Success Paths

In accordance with the EPRI seismic margin assessment (SMA) methodology (Reference 5), success paths are specified which will bring the plant from full-power operations to a safe condition (defined as hot or cold shutdown), and which will maintain that condition during a 72-hour period following the initiating event.

The objective of this assessment is to identify a primary and an alternate success path of safe shutdown components for each unit which could then be demonstrated as operable following the specified seismic margin earthquake (SME). To achieve a safe condition, the safe shutdown components must survive the event and remain operable to perform the safety functions necessary to preclude core damage. The safety functions addressed by the EPRI SMA methodology are:

- Reactivity control.
- Pressure and inventory control.
- Decay heat removal (DHR).

In accordance with the EPRI SMA methodology, a primary and an alternate success path are selected for each unit. Both paths are capable of performing the required safety functions listed above in the event of a small-break LOCA.

The first step in the selection of the shutdown paths is to identify the various alternate paths that can be used to achieve the previously defined safety functions listed above. The paths are defined by the available front-line systems and are presented in the success path logic diagram (figure 3.1-10).

In selecting the primary and alternate shutdown paths, emphasis is placed on compatibility with existing plant procedures, ease of use by operators, and minimizing the number of components required for performance of the plant walkdown and verification. The safe shutdown equipment list (SSEL) comprising the primary and alternate success paths are categorized by system in this report in Appendix E for Unit 1 and Appendix F for Unit 2. A summary of required operator actions is included in Appendix G.

For the Plant Hatch assessment, the primary and alternate paths are independent of each other, with the following exceptions:

- The control rod drives (CRDs) and hydraulic control units (HCUs) are common to both paths. (See section 3.1.2.5.1.B.)
- Unit 1 plant service water (PSW) valve 1P41-F312. (See section 3.1.2.6.E.)

- The main control room environmental control (MCREC) system has shared ductwork between Unit 1 and Unit 2. (See section 3.1.2.6.H.)

- The intake structure HVAC system has a shared fan. (See section 3.1.2.6.I.)

Path independence is accomplished by using separate power supplies, different systems, opposite system trains, and redundant components within systems, to perform the safety functions.

3.1.2.2 Criteria and Assumptions

Each set of equipment comprising the safe shutdown paths must meet the following criteria:

1. The reactivity control function shall be capable of achieving cold shutdown reactivity conditions.

2. The reactor coolant makeup function shall be capable of maintaining the reactor coolant level above the top of the core; however, the core is allowed to be uncovered momentarily if the safe shutdown operation involves depressurization and a low-pressure makeup system.

3. The reactor heat removal function shall be capable of achieving and maintaining DHR.

4. The process monitoring function shall be capable of providing direct readings of the process variables necessary to perform and control the above functions.

5. The supporting functions shall be capable of providing the process cooling, lubrication, etc., necessary to permit the operation of the equipment used for safe shutdown functions.

6. The equipment and systems used to achieve and maintain safe (hot or cold) shutdown conditions shall be capable of being powered by an emergency onsite power system.

7. The equipment and systems used to achieve and maintain cold shutdown conditions shall be capable of performing these functions for 72 hours following the initiating event.

The reactor is assumed to be operating at 100-percent power at the time of the event with normal water level and steady-state conditions. The following conditions and assumptions are considered in selecting the safe shutdown path equipment:

1. Offsite power may not be available for up to 72 hours following the earthquake.

2. All systems and components are functional and in their normal configuration prior to the earthquake.

3. Only seismically induced transient events and small-break LOCAs are addressed in accordance with the EPRI SMA methodology. Based on the representative piping walkdown described in section 3.1.1.6.I of this report, and the inherent ruggedness of piping

as demonstrated in the experience data base plants, the SME should not cause a large-break LOCA.

4. Equipment within the IPEEE scope, but not in the USI A-46 scope, was evaluated for low-ruggedness relays. The potential effects of seismically induced relay and contactor chatter were evaluated, as well as the operator actions required to recover from any such effects, as part of the USI A-46 program. The relay evaluation is discussed in section 3.1.4.10.

   The following types of equipment were identified for use in the USI A-46 relay chatter evaluation:

   - Active, electrically powered or controlled equipment identified as needed for safe shutdown.

   - Electrically powered or controlled equipment considered passive, but whose inadvertent operation as a result of relay chatter could adversely affect the accomplishment of a safe shutdown function.

5. Where operator actions are relied upon to achieve and maintain safe shutdown, assurance is provided that appropriate procedures are available.

6. The equipment identified for seismic evaluation includes:

   - Active mechanical and electrical equipment that operates or changes state to accomplish a safe shutdown function.

   - Instrumentation required to confirm that the three safe shutdown functions have been achieved and are being maintained.

   - Instrumentation required to operate the safe shutdown equipment.

   - Tanks and heat exchangers used by or in the identified safe shutdown path.

   - Passive electrical equipment.

7. Passive mechanical equipment that is not required to change state, but which could be affected by relay chatter, is identified for relay evaluation, but is not seismically verified by walkdown.

   The following equipment types are not identified for evaluation:

   - Passive equipment, such as manual valves, filters, and electrical penetration assemblies that do not operate or change state.

   - Self-actuated check valves.

3.1.2.3 Primary Shutdown Path

The primary shutdown path provides reactivity control, high-pressure makeup, depressurization, low-pressure makeup, and low-pressure DHR. The primary shutdown path consists of the CRD system, the high-pressure coolant injection (HPCI) system, the automatic depressurization system (ADS) safety-relief valves (SRVs), one loop of the residual heat removal (RHR) system for the suppression pool cooling mode, and one loop of the core spray (CS) system.

Reactor protection system (RPS) instrumentation is provided to initiate an automatic scram signal, if required by operating parameters; e.g., loss of offsite power (LOSP), low reactor water level, or a manual scram can be performed if necessary. Upon a scram signal, the control rods are inserted, thus providing the negative reactivity to shut down the reactor. Control room operators verify the reactor is shut down and enter into the shutdown procedure.

An LOSP will most likely occur with the seismic margin earthquake (SME). If an LOSP occurs, the control room operators will immediately verify operation of the diesel generators (DGs). If the DGs do not auto-start as a result of relay chatter, the emergency operating procedures (EOPs) direct an operator to be dispatched to the diesel generator building to assess conditions. Indication on the relays will identify the relays which are tripped. The operator can place the relays in the proper condition and start the DGs within approximately 10 minutes. This is well before ac power is required for low-pressure makeup or heat removal.

According to procedure, control room operators verify the HPCI system is operating and providing makeup. The HPCI system, which is used to maintain inventory until the reactor is depressurized to the operating range of the low-pressure makeup systems, can rapidly restore reactor inventory. If the system trips on high reactor level, it can be restarted as necessary. The reactor can be depressurized by HPCI operation. HPCI suction can be taken from the condensate storage tank (CST) (which is not on the SSEL) or the suppression pool if the CST is not available.

Vessel depressurization is accomplished by manually opening SRVs. The ADS will automatically initiate upon a high drywell pressure or a low reactor water level signal from reactor vessel instrumentation after certain interlock criteria are met. One loop of RHR is available for operation in the suppression pool cooling mode and can be started if HPCI operation increases pool temperature.

The available loop of CS can be manually started to recover any inventory loss following depressurization. The same signals that initiate the ADS automatically start the CS system pumps. Injection does not occur until the pressure is reduced to CS operating levels. For shutdown via the HPCI system, operators are able to control the sequence of events and perform the required actions as time and conditions permit. All primary path manual operator actions, other than restoration of the DG relays, are performed from the main control room.

If CS is not in operation, it is manually started. After makeup flow is established with CS, one of the available SRVs is manually opened to establish an alternate shutdown cooling path. CS is used to slowly raise the reactor water level to establish a flow path through the open SRV back to the suppression pool. The RHR system is started in the suppression pool cooling mode if it is not already running. Cooldown is established by CS taking suction from the suppression pool, injecting into the reactor, and returning to the suppression pool through the open SRV. The suppression pool cooling mode of RHR removes heat from the suppression pool.

3.1.2.4 Alternate Shutdown Path

The alternate shutdown path provides reactivity control, depressurization, and low-pressure makeup. The alternate shutdown path consists of the CRD system, the SRVs, and one RHR loop with low-pressure coolant injection (LPCI) and alternate shutdown cooling modes.

A reactor scram is initiated as described for the primary shutdown path. A high drywell pressure and/or low reactor water level signal automatically initiates the LPCI mode of RHR and the ADS. The SRVs which are dedicated to ADS auto-open upon receipt of these signals and depressurize the reactor if the LPCI pump is running. Actuation of the SRVs can be inhibited from the control room by the plant operator per the EOPs to allow manual control of the depressurization rate. Since no high-pressure makeup systems are assumed available following a scram, reactor inventory is reduced if the SRVs open to provide overpressure protection. According to NEDO-24372, "Minimum Systems Required for Safe Shutdown During a Fire in Edwin I. Hatch Nuclear Power Station Units 1 and 2" (Reference 10), water level drops to the top of the active fuel (TAF) in approximately 1330 seconds (22 minutes). The reactor is then depressurized by manual actuation of three SRVs. When the pressure falls within the operating range of the LPCI subsystem, the LPCI injection valve opens to permit one RHR pump to reflood the core. The core is uncovered for approximately 428 seconds, during which time peak cladding temperature is calculated to be 1208°F. This temperature is well below the temperature that could result in fuel cladding damage.

As in the primary shutdown path, the DGs may not auto-start as a result of relay chatter. The relays can be reset in approximately 10 minutes, as previously described.

Approximately 42 minutes into the event, reactor water level is restored, and pressure is within the shutdown cooling operation limit. The alternate shutdown cooling mode of RHR is used to bring the plant to cold shutdown. Suppression pool water is injected into the reactor to maintain water level using the LPCI flow path with flow going through the RHR heat exchanger. Residual heat removal service water (RHRSW) system operation is required to cool the pool water. Water in the reactor is discharged to the suppression pool through manually operated SRVs.

3.1.2.5 Description of Front-Line Systems

Plant-specific systems available to perform the safety functions identified in section 3.1.2.1 are shown on the Success Path Logic Diagram in figure 3.1-10. These systems are categorized as front-line systems because they perform a direct safety function. The front-line systems are as follows:

3.1.2.5.1 Reactor Reactivity Control

A. Reactor Protection System (multiple system MPL numbers)

The reactor protection system (RPS) monitors plant parameters and initiates a rapid, automatic shutdown (scram) via the CRD system. The scram is initiated in time to prevent fuel cladding damage following abnormal operational transients or accidents. The RPS is designed to be fail safe. The RPS automatically provides a scram signal if an LOSP occurs; however, it is not assured than an LOSP will occur as the result of a seismic event.

Components of the RPS required to provide an automatic scram signal are provided for both shutdown paths. Instrumentation is available to monitor reactor pressure and level so that a scram will occur if the pressure or level reaches the scram trip setpoint. Limit switches on the main steam isolation valves will also initiate a scram signal if the valves are not fully open. Capability for a manual scram is also available. Instrumentation is provided for both paths to monitor critical plant parameters so that if a scram does not occur, the operator can determine whether a manual scram is necessary. Normal operator action following any automatic scram is to perform a manual scram.

B. Control Rod Drive System (C11)

When the RPS initiates a scram, the CRD system inserts the negative reactivity necessary to shut down the reactor. Each control rod is controlled by an HCU. When a scram signal is received, high-pressure water from an accumulator for each rod rapidly forces each control rod into the core.

Reactivity is controlled by insertion of the control rods. A primary path and an alternate path of CRD system components are included on the SSEL to provide insertion of the control rods. The paths are independent of one another except for the HCUs and the CRDs, which are included in both paths. Reliability for the HCUs and CRDs is provided by the fact that the reactor is capable of being shut down with the failure of any one individual HCU or CRD.

C. Standby Liquid Control System (C41)

The standby liquid control system (SLCS) serves as a backup to the control rods for reactivity control. However, the SLCS is not considered a viable alternative for reactivity control for the SMA because of the time and operator actions involved for initiation. This is consistent with the requirements of the EPRI SMA methodology.

3.1.2.5.2 Pressure and Inventory Control

A. Automatic Depressurization System (B21)

A pressure relief system, consisting of 11 SRVs mounted on the main steam lines, is provided to prevent excessive pressure inside the nuclear boiler system following abnormal operational transients. Reactor steam is discharged to the suppression pool.

SRVs in each success path open automatically to provide overpressure protection when system pressure exceeds their setpoints. The SRVs can also be manually actuated to reduce reactor pressure. Equipment for both automatic and manual operation of the SRVs is included in the SSEL.

B. High-Pressure Coolant Injection System (E41)

The HPCI system provides high-pressure inventory makeup for the primary safe shutdown path. The HPCI system includes a turbine-driven pump powered by reactor steam and is the preferred source of high-pressure makeup. It is sized to provide adequate coolant inventory to prevent fuel clad damage in the event of a small-break LOCA.

Industry HPCI system experience has indicated that the system is moderately reliable, with individual system unavailabilities being in the low-percent range. For primary path shutdown, high-pressure makeup is desirable, but not necessary, to achieve a safe condition. In the event of HPCI system unavailability, depressurization and low-pressure makeup capability is available for inventory control.

HPCI system initiation automatically occurs on a low reactor water level or high drywell pressure signal. The system can also be started manually. Normal operator action on a plant scram is to verify HPCI operation at the panel in the main control room. If an automatic initiation has not occurred, the system is started manually from the panel with a minimum of operator action using normal operating procedures. Instrumentation required for system operation and to monitor performance is available. The HPCI system is not assumed to start until 10 minutes following the initiating event (Reference 10).

The normal HPCI suction source is the CST; however, the CST is not included on the SSEL. The suppression pool is used as the suction source. Automatic transfer to the suppression pool normally occurs on a low CST level or high suppression pool level.

HPCI suction from the CST is taken from the bottom of the tank. The postulated failure at the CST is a buckle or crack at the bottom of the tank. If this occurred, the water would be contained within the Seismic Category I shield wall around the tank and the suction source would still be available. If the water source or suction line from the CST is lost and auto-transfer to the suppression pool does not occur, HPCI trips on low suction pressure. If the water source from the CST cannot be confirmed, HPCI is manually aligned using normal operating procedures to take suction from the suppression pool and is restarted. Full HPCI flow is injected into the reactor via the suppression pool suction path. HPCI pump suction pressure indication is available in the main control room.

NEDO-24372 presents an analysis for a shutdown scenario that is essentially the same analysis used for the SMA primary shutdown method. Section 6 of NEDO-24372 states, "The CST is the preferred water source for makeup inventory. Suppression pool as a water source is also acceptable."

The reactor core isolation cooling (RCIC) system is the alternate high-pressure makeup system to HPCI; however, it is sized to maintain hot shutdown conditions and is not designed to provide makeup for a design basis LOCA. For Plant Hatch, ADS is considered the redundant system to HPCI; therefore, the alternate shutdown path provides depressurization and low-pressure makeup. The RCIC system is not included in the alternate safe shutdown path.

C. Core Spray System (E21)

The CS system consists of two independent pump loops that deliver cooling water to spray spargers located over the core. One loop of CS is included in the primary shutdown path. The system is actuated by conditions indicating that a breach in the nuclear system process barrier exists, but water is delivered to the core only after reactor vessel pressure is reduced. The CS system provides the capability to cool the fuel by spraying water onto the core. Either CS loop is capable of preventing fuel clad melting following a LOCA. CS may be used to provide low-pressure makeup, if desired, before DHR systems are placed in operation.

In the event of a small-break LOCA, the reactor vessel will depressurize slowly or not at all. Therefore, the HPCI system can be used to maintain inventory until time and conditions permit use of CS. Automatic depressurization and CS initiation is available but not required. The plant operators, using the EOPs, can manually depressurize by opening SRVs, and open the injection valves after the CS pumps are started. Instrumentation required for CS operation and monitoring system performance is available.

D. Residual Heat Removal System - Low-Pressure Coolant Injection Mode (E11)

The RHR system has several functional modes of operation which include reactor coolant inventory control and reactor and containment DHR. The system is designed to provide these functions during normal operation and during design basis accident conditions. The RHR system consists of four pumps and two heat exchangers divided into two independent loops, each having two pumps and one heat exchanger, plus the associated instrumentation, valves, and piping.

The alternate shutdown path uses the LPCI mode of RHR for reactor coolant inventory control. One RHR pump and the associated loop of the RHR system, including the suppression pool suction, LPCI pump suction, LPCI discharge, and injection valves, are included for use in the LPCI mode as the front-line low-pressure makeup system. The LPCI pump injects cooling water at low pressure into the reactor via a recirculation loop. LPCI is automatically actuated by conditions that indicate a breach in the nuclear system process barrier, but water is delivered to the core only after reactor vessel pressure is reduced. LPCI operation, in combination with the core shroud and jet pump arrangement, provides the capability of core reflooding following a LOCA in time to prevent fuel clad melting.

Instrumentation required for automatic LPCI operation and monitoring system performance is available. LPCI injection can also be performed manually by the operators.

3.1.2.5.3 Decay Heat Removal

A. Residual Heat Removal System - Alternate Shutdown Cooling Mode (E11)

The alternate shutdown cooling mode of RHR provides a method of DHR when normal shutdown cooling systems are not available. It is a necessary front-line system that provides an alternate to the normal shutdown cooling mode. It is used for both primary and alternate shutdown path DHR using opposite loops of RHR. The SSEL includes instrumentation to monitor RHR system operation. Normal shutdown cooling may be used for alternate path shutdown; however, a manual action to open valve E11-F008 is required to provide a suction path from the recirculation loop to the RHR pumps.

For primary path shutdown, the alternate shutdown cooling mode consists of one loop of CS, one loop of RHR with one RHR pump in the suppression pool cooling mode, and SRVs. Alternate shutdown cooling is initiated by placing one loop of RHR (the loop not used for LPCI in the alternate shutdown path) in the suppression pool cooling mode. After suppression pool cooling is established, one of the SRVs, which was verified for pressure control, is manually opened. CS, which may already be providing makeup, is used to raise the level and establish a flow path through the SRV back to the suppression pool. Decay heat is removed by suppression pool cooling. Operation of this alternate shutdown cooling mode is defined in the EOPs.

For alternate path shutdown, the alternate shutdown cooling mode consists of one loop of RHR with the RHR pump in the LPCI mode, and SRVs. The RHR system is already in the LPCI mode which is providing makeup water to the reactor. Alternate shutdown cooling is initiated by starting the residual heat removal service water (RHRSW) pumps and closing the RHR heat exchanger bypass valve to force more of the water flow through the heat exchanger. Decay heat is removed by the RHRSW in the heat exchanger. The cooldown rate is controlled by the position of the heat exchanger bypass valves. The flow path is basically the same as for the LPCI mode. Operation of this alternate shutdown cooling mode is defined in the EOPs.

B. Residual Heat Removal - Suppression Pool Cooling Mode (E11)

The suppression pool cooling mode of RHR limits the water temperature in the suppression pool following a design basis LOCA. In the suppression pool cooling mode of operation, the RHR pumps take suction from the suppression pool and pump the water through the RHR heat exchangers where cooling takes place by transferring heat to the RHRSW system. The fluid is discharged back to the suppression pool.

C. Residual Heat Removal - Shutdown Cooling Mode (E11)

The normal shutdown cooling (SDC) mode of RHR can be used for DHR in the alternate path shutdown; however, a manual action is required. SDC is initiated by placing the RHR loop that was used for LPCI in the SDC mode. Suction for the SDC mode is taken from the recirculation piping. The valves in the suction line, E11-F008 and E11-F009, are containment isolation valves. Power for E11-F008 is derived from a primary shutdown path power supply, while power for E11-F009 is derived from an alternate shutdown path power supply. Opposite path electrical supplies are required for inboard and outboard containment isolation to assure at least one valve can be closed. Therefore, to maintain independent paths, valve E11-F008 must be manually opened using the handwheel. Plant procedures provide for manual opening of this valve if it cannot be operated from the main control room. The EOPs provide for use of alternate SDC, which requires no manual action; therefore, normal SDC is not credited in either path.

3.1.2.5.4 Containment Performance

A. Residual Heat Removal - Containment Spray Mode (E11)

The containment spray system is an operating mode of the RHR system. Spray nozzles located in both the drywell and torus area provide post-accident containment atmosphere temperature and pressure control. Suction for this mode is taken from the suppression pool, routed through the shell side of the RHR heat exchangers, and diverted to the spray headers in the drywell and torus. Water sprayed into the drywell drains back to the suppression pool for reuse.

B. Containment Isolation Valves

Containment isolation valves in systems used for the safe shutdown paths are included by default in the SSEL. Containment isolation valves in systems not used for the safe shutdown paths are also included in the SSEL. The valves are discussed in the Unit 1 FSAR, sections 5.2 and 7.3 (Reference 11), and the Unit 2 FSAR, sections 6.2 and 7.3.2 (Reference 12).

3.1.2.6 Description of Support Systems

Systems categorized as support systems must remain operable during and after the SME so that front-line systems can perform their required safety functions. The following support systems provide a support function to front-line systems and/or to other support systems, and are shown in the support and front-line systems dependency matrixes (see tables 3.1-1 and 3.1-2 of this report).

A. Reactor Protection System (multiple system MPL numbers)

The portion of the RPS providing the reactor scram function is considered a front-line system. The RPS also serves a support system function by providing reactor and reactivity monitoring instrumentation.

Reactor instrumentation providing operators with information on the primary process variables used to ensure the plant is in a safe condition is included in both safe shutdown paths. Examples of primary process variables are reactor coolant system pressure, temperature, and water level.

In accordance with the EPRI SMA methodology, instrumentation is provided for control and actuation of systems needed to perform safety functions and for monitoring critical plant parameters, including monitoring of reactor shutdown status.

The normal method for verifying that the reactor is shut down following a scram is by rod position indication that all rods are full in. If this indication is not available, the EOPs direct the operators to determine from other plant parameters that the reactor is subcritical and to proceed to the shutdown procedures. Although rod position indication is not included in the shutdown paths, sufficient reactivity monitoring instrumentation has been screened to allow the operators to make a determination and proceed with shutdown.

B. Standby ac Power Systems (R20)

Power to the emergency buses is normally supplied from offsite power. Because offsite electrical power is assumed to be unavailable up to 72 hours following an SME, and the switchyard is assumed unavailable, electrical power for the operation of the safe shutdown components must be supplied by the standby electrical power system. Standby power is supplied by two diesel generators (DGs) dedicated to Unit 1 equipment (1A and 1C), two DGs dedicated to Unit 2 equipment (2A and 2C), and one DG that can power either unit (1B). The swing DG has a Unit 1 parts list number and is included in the Unit 1 SSEL. Should an LOSP occur, the DGs automatically start and provide power to the ac power distribution system safety-related 4160-V buses and lower voltage distribution system components.

Diesel generators 1B and 1C are the Unit 1 primary path diesels, while generator 1A is the Unit 1 alternate path diesel. DG 2C is the Unit 2 primary path diesel, while DG 2A is the Unit 2 alternate path diesel. The DGs have auxiliary components that are required to support diesel start and continued operation included in the SSEL. These components include fuel transfer pumps and diesel generator building ventilation equipment.

C. dc Power Supply (R42)

In the event normal plant power sources become unavailable, the plant battery systems provide power for control of plant safety functions required for safe shutdown. The station service dc power distribution system receives power through battery chargers from the ac power system and from the station batteries. The main dc system includes two 250- to 125-V-dc switchgear buses which provide power to various dc motor control centers and 125-V-dc distribution panels in the control and reactor buildings.

Each DG has its own separate battery system which consists of two battery chargers, batteries, and a 125-V-dc distribution panel in the diesel generator building.

Two 24- to 48-V instrument dc power systems powered from the ac power system have separate battery chargers, batteries, and dc panels.

D. Drywell Pneumatic System (P70, T48)

The drywell pneumatic system, used in conjunction with the nitrogen inerting system, provides gas of suitable quality and pressure to supply the air-operated equipment in the drywell. The SRVs are the only air-operated valves included in the SSELs. All other air-operated valves included in the SSELs remain in, or fail to, the proper position on loss of gas supply. The SRVs have pneumatic actuators providing capability for selective operation at system pressures below the set pressure. Since selective SRV operation is relied upon in the shutdown paths, a pneumatic gas supply is a necessary support system.

The normal drywell pneumatic supply used at Plant Hatch is from the drywell inerting system nitrogen storage tank. Valves are included to isolate nonessential portions of the system if failure in these areas causes a loss of gas inventory. Backup nitrogen bottles are available as an alternate gas supply. The bottles have manual valves which can be opened to supply nitrogen, if necessary, using the normal system operating procedure.

Short-term SRV pneumatic supply is provided by individual accumulators for each SRV. The accumulators are included on the SSEL to ensure piping integrity for the pneumatic supply and are sized to ensure two SRV actuations within the first half hour at 70-percent drywell design pressure. For events which result in no breaks within the drywell, the accumulators will provide multiple actuations for greater than 2 hours.

E. Plant Service Water System (P41)

The PSW system consists of four, one-third-capacity wet-pit service water pumps located in the river intake structure, one standby service water pump dedicated to DG 1B, and distribution piping and controls.. Three PSW pumps are required for plant startup and normal operation, although only one pump is required for shutdown and emergency shutdown. If the operating pumps cannot maintain the required system pressure, the PSW pump(s) in standby, start automatically. The PSW system is manually aligned and operated from the main control room using normal operating procedures.

A PSW train containing one pump, associated valves and piping, and instrumentation is available for each shutdown path. The PSW system is included to provide cooling water to the DGs, the RHR pump coolers, and the area coolers in the HPCI and emergency core cooling system pump rooms. The PSW pump dedicated to DG 1B is also included. CS pumps are cooled by water from the CS pump discharge.

Isolation valves (1P41-F310A and B for Unit 1, and 2P41-F316A and B for Unit 2) are provided in each PSW train to isolate the turbine building and other nonessential areas of the plant. These valves are normally open and will isolate on an LOSP. If an LOSP does not occur, the DGs are not required; thus, service water to the DGs is not required.

Unit 1 PSW valve 1P41-F312, which is a normally closed motor-operated valve, is a part of both safe shutdown paths. It remains closed to prevent service water from being diverted back to the river. The failure mode for this valve is for it to open inadvertently; however, there are no automatic signals to open this valve. Any relay chatter associated with valve 1P41-F312 causes it to close or remain closed, which is the desired position. There are no postulated failures which could cause the valve to inadvertently open. A deliberate operator action is required to open the valve. This same situation exists for the Unit 2 primary path with PSW valve 2P41-F310. The Unit 2 alternate shutdown path for PSW does not communicate with the dilution line containing 2P41-F310.

F. Residual Heat Removal Service Water System (E11)

Each division of the RHRSW system supplies river water to the RHR heat exchangers to remove heat during both normal and accident conditions. The RHRSW system is a required support system for the suppression pool cooling and shutdown cooling modes of RHR. The RHRSW system consists of two independent loops, each with two pumps and associated valves and piping. RHRSW system pumps are located near the river in the Seismic Category I intake structure. A steel barrier between opposite loop RHRSW system pumps provides protection from jet impingement to the RHRSW pump motors and associated equipment.

An RHRSW train, which contains one pump, associated valves and piping, and instrumentation, is available for each shutdown path. The RHRSW system is manually placed in service using the normal operating procedure for the RHR system.

G. Reactor Building Heating, Ventilation, and Air-Conditioning Systems (T41)

The reactor building safety-related equipment area cooling is a required support system for HPCI, RHR, and CS to perform their safety functions. Coolers are provided in the equipment areas for each system to maintain operating temperatures within design allowables. The PSW system supplies water to the cooling coils and serves as the heat sink for the equipment area cooling systems.

H. Control Building Heating, Ventilation, and Air-Conditioning System (Z41)

The control building HVAC system is composed of several components and subsystems that maintain the temperature in specific areas of the control building below the operational limits of the equipment contained in that area, and prevent a radioactive environment in the main control room. The control building HVAC equipment required to remain operable for safe shutdown includes the following:

Main Control Room Environmental Control System

The MCREC system is designed to detect and limit the introduction of radioactive material into the main control room, ensuring habitability during accident conditions. The MCR is a shared facility divided into two adjacent open areas, with one area serving Unit 1 and the other serving Unit 2. Each area is served by an independent air-conditioning system consisting of a fan, direct-expansion cooling coil, condensing unit, supply and return ducts, and room- and duct-mounted thermostats. One common air-conditioning system serves as a standby for either area. All three systems are fed by either of two independent filter trains and booster fans for outside makeup air. The PSW system provides cooling for the condensers in the systems. Two of the three systems are included in each shutdown path, and instrumentation controlling system operation is included in the SSEL. The recirculated air taken from the Unit 1 and Unit 2 control room areas share ductwork; therefore, the two shutdown paths are not completely independent. However, there is no common failure that would negatively affect both shutdown paths. Although all three air-conditioning systems have Unit 1 Master Parts List identification numbers, the system serves both Units 1 and 2. The "C" air-conditioning system supplies air to the Unit 1 area. The "A" air-conditioning system supplies air to the Unit 2 area, and the "B" system is capable of serving either unit. All components required for the MCREC system are listed on the Unit 1 SSEL.

Station Service Battery Room Exhaust System

The station service battery room exhaust system provides the necessary ventilation to prevent excessive hydrogen concentrations in the station battery rooms, the RPS battery rooms, and the vital AC room when normal ventilation is not available. During normal operation, these rooms are ventilated by the control building ventilation system. The station service battery room exhaust system consists of two emergency exhaust fans, ductwork, and instrumentation and controls. Air exhausted from the battery rooms is replaced by air drawn from the normal control building ventilation system ductwork. A flow path for both the intake and exhaust air is provided. In the event of an LOSP, one of the two emergency fans provides the necessary ventilation. These fans operate independently of one another, with power provided by the emergency diesels. Emergency power is provided until normal power is restored. The components required for the primary and secondary shutdown paths are based on the batteries being used for that shutdown path.

I. Intake Structure Heating, Ventilation, and Air-Conditioning System (X41)

The intake structure HVAC system consists of three 50-percent capacity roof-mounted exhaust fans and associated thermostats and air intake dampers. The intake structure is a shared building housing Unit 1 and Unit 2 PSW and RHRSW pumps. Only one fan is required per unit, with one fan shared between them as a backup. This system is designed so that operation of two of the three fans limits the average ambient temperature in the Unit 1 and Unit 2 PSW pump bay areas to acceptable levels with three PSW pumps and one

RHRSW pump operating. The dampers are gravity operated (negative pressure) and auto open when the fans come on. Air is drawn into the louvers, circulates through the building, and is expelled through the fans.

The intake structure HVAC system components included in the SSEL are the three fans, a control thermostat for each fan, and four gravity-operated louvers. The fans are powered from three different power supplies:

- 1X41-C009A is powered by division I, Unit 1.
- 1X41-C009B is powered by division II, Unit 1.
- 1X41-C009C is powered by division I, Unit 2.

The controls for these fans allow manual operation from independent control stations or automatic control via the thermostat. For the purpose of path designation, the Unit 2 powered fan is used for both paths. The Unit 1 division II fan is a primary path component, while the Unit 1 division I fan is an alternate path component. All four intake air louvers are designated for both paths. All components required for this system are listed on the Unit 1 SSEL.

J. Diesel Generator Building Heating, Ventilation, and Air-Conditioning System (X41)

The diesel generator building HVAC system consists of two fans, an air intake louver, and associated controls for each diesel generator room, switchgear room, battery room, and oil storage room. Both fans for each area provide 100-percent flow capacity, with one acting as the primary fan and the other as the standby fan in the event the primary fan fails. The primary oil storage and battery room fans run continuously, while operation of the DG and switchgear room fans are thermostatically controlled to maintain the room temperature within equipment design parameters. The Unit 1 fans required to operate are in diesel generator rooms 1A, 1B, and 1C; switchgear rooms 1E, 1F, and 1G; and battery rooms 1A, 1B, and 1C. The Unit 2 fans required to operate are in diesel generator rooms 2A and 2C, switchgear rooms 2E and 2G, and battery rooms 2A and 2C.

Room cooling or ventilation is required to support operation of the DGs. Only one fan for each room is required to maintain the temperature or provide ventilation. The logic for these fans requires that the room louver be open for the fan to start. Thermostats in the diesel generator rooms and switchgear rooms control louver position and, therefore, must be operable. The air flow switches on each diesel generator room fan that control switching from the primary fan to the secondary fan, in the event the primary fan fails, are included in the SSEL to prevent unintended switching to the secondary fan. All louvers and fans may be manually opened/started by their respective control switches, which are located on panels in the diesel generator rooms, and are used to override the control logic.

3.1.3 ANALYSIS OF STRUCTURE RESPONSE

This section provides an overview of the seismic margin earthquake (SME) soil-structure interaction (SSI) analysis and development of the in-structure response spectra (IRS) used for the Plant Hatch Unit 1 and Unit 2 SMAs.

Plant Hatch buildings are founded on soil, with some being significantly embedded in the soil. Both the frequency and the amplitude of response to seismic excitation for these buildings are greatly affected by SSI. The original SSI analyses of the Plant Hatch buildings were performed conservatively. For example, spatial variation of the ground motion with depth was not used, and the limitation on soil damping, including effects of radiation damping, was set not to exceed 5.5 percent of critical damping. The NRC's review of the seismic analysis of Plant Hatch Unit 1 (Reference 13) indicated significant conservatism in the original SSI analysis. Based on the conservatism of the original SSI analysis and other parameters originally used, scaling the original IRS was not appropriate. Therefore, median-centered responses of the major structures were generated for the SME. The new SSI analysis was performed in accordance with the EPRI SMA methodology (Reference 5).

The IRS developed for the SMA were used to assess the high-confidence-low-probability-of-failure (HCLPF) capacity of components and equipment supported in the buildings. To properly assess the HCLPF levels for the plant, the calculated responses should be median centered; therefore, the SSI evaluation, structural models, and parameter values were median centered.

As discussed in section 3.0, the NRC was actively involved in the Plant Hatch Unit 1 SMA. Reference 14 documents the final report of the Plant Hatch Unit 1 SMA, and References 15, 16, 17, and 18 document the review reports from the NRC Peer Review Group and the NRC Seismic Design Margins Working Group. The NRC has reviewed and accepted the use of the SME IRS described in this report for use in the resolution of USI A-46 for both Unit 1 and Unit 2. (The same IRS were scaled down by the ratio of the design basis earthquake (DBE) peak ground acceleration to the SME peak ground acceleration for USI A-46.) This position is documented in Reference 19.

3.1.3.1 <u>Seismic Margin Earthquake</u>

The SME was developed using 0.3 g pga. The spectrum shape was based on NUREG/CR-0098 (Reference 20) median-centered spectra, but was modified for magnitude effects using the procedure proposed by H. B. Seed and I. M. Idriss (Reference 21). Adjustments were made to reflect a magnitude of m = 6.25. The values of v/a = 100 cm/s/g (39.4 in./s/g) and $ad/V^2 = 5$ were selected. The spectra amplification factors were those specified in NUREG/CR-0098 for a median spectra shape.

Figure 3.1-11 is a plot of the 5-percent damped SME ground response spectra (GRS). For comparison, figure 3.1-11 also includes a plot of the Plant Hatch Unit 1 DBE described in section 3.1.1.3.1. A similar plot of the Plant Hatch Unit 2 DBE and the 5-percent-damped SME GRS is shown in figure 3.1-12.

One vertical and two horizontal synthetic time histories were developed to be statistically independent of each other. The absolute value of the correlation coefficient was < 0.16.

Synthetic accelerograms having a zero period acceleration of 0.3 g pga and spectral ordinates that provide a reasonable fit to the smooth response spectrum were developed. Figure 3.1-13 is the accelerogram of one of the two horizontal time histories. Figure 3.1-14 is a comparison of the spectrum of horizontal time history to the SME spectrum. In addition, the power spectral density (PSD) and the cumulative plots were developed to ensure adequate energy over the frequency range of interest for the synthetic time histories. Figure 3.1-15 is the PSD of one of the horizontal time histories scaled to 1.0 g pga. The adequacy of the PSD was demonstrated by comparing it to the average PSD of 14 records of the 1971 San Fernando earthquake. The average PSD of these 14 records scaled to 1.0 g pga is illustrated in figure 3.1-16. Figure 3.1-17 shows the cumulative PSD for the synthetic time history compared to the cumulative PSD for the average PSD of the 14 records from the San Fernando earthquake.

Because the PSDs for the selected synthetic time histories are comparable to or exceed those for the average of the 14 San Fernando records, and a reasonable fit of the spectra of the synthetic time histories to the SME spectrum exists, the selected synthetic time histories were judged adequate.

The SME is specified at the free-field ground surface. The effect of the spatial variation of the ground motion is discussed in section 3.1.3.2.

3.1.3.2 <u>Soil Profiles and Their Variation</u>

This section briefly addresses the development of the lower-, intermediate-, and upper-bound soil profiles for the SSI analysis of each of the major plant structures. Development of the basic plant soil profiles, Profiles I and II, is discussed in section 3.1.4.7.1. For this discussion, the soil profiles from section 3.1.4.7.1 are referred to as "best estimate" or "average." A more detailed discussion and a tabulation of the soil profiles used for the SSI analysis of each building can be found in EPRI NP-7217-SL, Appendix C, Section C3, Enclosure B (Reference 1).

The strain-compatible dynamic soil properties were obtained from the response analyses of Profiles I and II using the computer program SHAKE (Reference 22). The synthetic time history was applied at the ground surface. The strain-compatible shear moduli were obtained by using the average modulus reduction for sands. Strain-compatible values of material soil damping were computed using the average damping curve and the lower range damping curve for sands (Reference 21).

Three additional sets of moduli were also analyzed. One set was considered a reasonable lower range and was set at 0.75 of the best estimate values. Another set, considered a reasonable upper range, was set at 1.8 multiplied by the best estimate values. Finally, an intermediate range was obtained by using 1.2 multiplied by the best estimate values. The results of the SHAKE analyses are included in EPRI NP-7217-SL, Appendix C, Section C4, Enclosure 1.

The results of the strain-compatible dynamic soil properties analyses were evaluated. The strain-compatible moduli were not significantly affected by the damping curve used. Also, for the cases in which the same damping curve was used, the strain-compatible damping ratios for different sets of moduli were almost identical. The results of these analyses were used to develop recommendations for strain-compatible moduli and soil material damping values used in the SSI analyses for the intermediate-, lower-, and upper-range modulus cases.

The recommendations consisted of an intermediate soil profile (shear modulus and soil material damping percentage for soil layers having depths ≥ 10 ft). The lower- and upper-range modulus cases were specified as 0.6 and 1.6 multiplied by the modulus values of the intermediate case. The use of 0.6 and 1.6 multiplied by the strain-compatible intermediate modulus corresponds to the use of 0.75 and 1.8 multiplied by the low-strain best estimate modulus, respectively. The soil material damping percentages listed for the intermediate case are the average of the strain-compatible damping values obtained using the lower-range damping curve and the average damping curve. Intermediate-case soil material damping was also recommended for the lower- and upper-bound cases.

The EPRI SMA methodology recommends the use of median-centered SSI analyses, structural models, and parameter values. Median-centered SSI analyses require that full credit be taken for vertical spatial variation of ground motion, kinematic interaction, and radiation of energy from the structure into the soil. The procedures and parameter values are median centered. However, considerable uncertainty exists in soil-structure system frequency estimates and in items such as vertical spatial variation of free-field ground motion. The SMA accounts for this uncertainty by shifting soil stiffness properties over a range to encompass the effects of approximately ± 1 standard deviation parameter variation. The consideration of many variations in soil dynamic properties as discussed above properly addresses these uncertainties.

3.1.3.3 Structural Models

The original two-dimensional seismic models were reviewed to ensure they were adequate for determining the responses for the SME. Two structures, the control building and the river intake structure, could potentially have significant torsional responses; therefore, these models were converted to three-dimensional models. The other three structures, the two reactor buildings and the diesel generator building, are essentially symmetrical; therefore, the use of two-dimensional models was sufficient. In addition, over the past several years, the mass and stiffness characteristics of the seismic models were reviewed and upgraded to reflect as-built conditions; therefore, no additional reviews were necessary. Also, the stiffness of concrete was judged to be a second-order effect on response compared to the effect of soil stiffness. The large variation of soil stiffness largely masks the effects of reduced concrete stiffness caused by cracking; therefore, consideration for the case with cracked concrete properties was not justified.

Structural damping was also evaluated for use in the structural response analysis to obtain seismic demand estimates that are essentially median centered. A rough estimate of the SME seismic forces and stresses was compared to the original values. Based on this evaluation and the damping values recommended in table 4-1 of the EPRI SMA methodology, a conservative estimate of the median-damping value was selected as 7 percent of critical. This structural damping value was used for all buildings, except the common roof structure portion of the reactor buildings. The roof structure is a bolted steel structure that begins at and extends approximately 50 ft above the refueling floor and supports the reactor building roof, the building crane, and the precast concrete siding. The roof structure will experience inelastic action at the SME level; therefore, 10-percent damping was specified for the reactor building roof structure. The elastic damping value will not be increased to account for the inelastic energy dissipation of the roof structure.

The seismic building models and their associated fixed-base natural frequencies, modal damping values, and percent-effective masses are found in Appendix H of this report.

3.1.3.4 Soil-Structure Interaction Approach

The SSI and structure response of the Plant Hatch buildings were analyzed using the substructure approach as implemented in the CLASSI system of programs (Reference 23). The substructure approach separates the SSI analysis into a series of simpler analyses, independently solves each problem, and superimposes the results. The steps of the substructure approach, as applied to structures with rigid bases subjected to earthquake excitations, are as follows:

1. Specifying the free-field ground motion.

2. Defining the soil profile (see section 3.1.3.2 of this report).

3. Calculating the foundation input motion.

4. Calculating the foundation impedances.

5. Determining the dynamic characteristics of the structure.

6. Performing the SSI analysis; i.e., combining the previous steps to calculate the response of the completed soil-structure system.

The free-field ground motion is specified at the free surface at the top of finished grade. The control motion is specified as three acceleration time histories that are essentially statistically independent. The spatial variation of motion is defined by vertically propagating waves.

Foundation input motion differs from the free-field ground motion, except for surface foundations subject to vertically incident waves. For one, free-field motion varies with soil depth. For another, the soil-foundation interface scatters waves because points on the foundation are constrained to move according to its geometry and stiffness. The foundation input motion is related to the free-field ground motion by means of a transformation defined by a scattering matrix. The reactor buildings and the intake structure were analyzed as embedded foundations which require the scattering matrixes. Because the control building and the diesel generator building are not as deeply embedded as the reactor buildings and the intake structure, their foundation motion is identical to the free-field ground motion. This simplification is somewhat conservative.

Foundation impedances, which describe the force-displacement characteristics of the soil, depend upon soil configuration and material behavior, frequency of excitation, and geometry of the foundation. CLASSI was used to generate the foundation impedances of all structures.

Uncertainties in the SSI analyses are documented in References 5, 24, and 25. For the Plant Hatch SMAs, the effect of uncertainties was accounted for in the SSI analyses by varying the soil-shear modulus. This method has been accepted as specified in Reference 25 and in the EPRI SMA methodology. The adequacy of this process to account for uncertainty associated with deconvolution is demonstrated by the fact that the resulting envelope spectrum at the foundation level of the reactor buildings generally exceeds 60 percent of the surface free-field motion.

In addition to the variation of the soil shear modulus, sensitivity studies were conducted to evaluate the effect of partial embedment of the reactor buildings and determine the proper treatment of the free-field control motion at the intake structure. For the reactor buildings, the SME was specified at the free-field ground surface to properly account for spatial variation of ground motion. No bonding of the side soil to the structure was assumed, since the reactor buildings are in contact with side soil only on one side and partially on another side. The modeling approach was judged to be more realistic than assuming all sidewalls are fully bonded to the soil. To account for the uncertainty of the effective embedment, the spectra were broadened an additional 10 percent in the frequency domain for subsystems having a fundamental natural frequency of approximately 2 to 3 Hz. This accounts for a possible shift of the major spectral peak of the IRS because of a stiffer foundation caused by side soil contact on a portion of the embedded reactor buildings. For the intake structure, the input motion was defined as the free field of the berm, as opposed to the free field of the riverbed, since, in general, the responses of the structure were higher for input motion at the berm.

3.1.3.5 Soil-Structure Interaction Results

As previously discussed, the purpose of a new SSI analysis was to obtain the median response of the buildings for the SME. These median responses were developed as IRS, and were calculated for the three different soil profiles and the three different directions. The format for plotting was to overplot the response spectra for the three soil profiles while holding all other parameters constant.

The spectra were developed for three different sampling values: 3, 5, and 10 percent. This spread of the spectral damping was adequate to perform HCLPF calculations. The IRS used for the SMA were the envelope of the three IRS associated with the lower-, intermediate-, and upper-bound soil modulus profiles. The IRS for the intermediate-bound soil modulus profile were broadened by ± 15 percent in the frequency domain. Thus, the IRS used to evaluate equipment and subsystems for a given damping value, direction, and location are a single envelope of the following three overlaid IRS: the IRS for the lower-bound soil profile, the IRS for the intermediate-bound soil profile broadened by ± 5 percent in the frequency domain for all frequencies, and the IRS for the upper-bound soil profile. This procedure is in agreement with the procedures discussed in Reference 24.

The SME IRS at 5-percent damping for all building elevations are given in Appendix H of this report. Note that the ± 15-percent peak broadening in the frequency domain of the SME IRS for the intermediate-bound soil profile, as discussed in the previous paragraph, is not shown in these spectra.

3.1.4 EVALUATION OF SEISMIC CAPACITIES OF COMPONENTS AND PLANT

3.1.4.1 Masonry Walls

Table 2-3 of the EPRI seismic margin assessment (SMA) methodology (Reference 5) requires an evaluation to screen out masonry walls. This section discusses the Plant Hatch masonry walls; the response to NRC Information and Enforcement Bulletin (IEB) 80-11, "Masonry Wall Design," (Reference 26); selection of the masonry walls for SMA evaluation; and the SMA evaluation approach and results.

3.1.4.1.1 Description of Masonry Walls

The control building is the only building with masonry walls in the vicinity of safe shutdown equipment. A review of masonry wall drawings showed that all masonry walls are constructed of either 8- or 12-in. blocks. All walls have heavy-rod-truss horizontal joint reinforcement for each block course. The walls are doweled to the supporting floor and anchored to side walls, reinforced concrete columns, or pilasters. Therefore, each wall has lateral support at the base and vertical side.

Masonry walls are either partially or completely filled with concrete. In partially filled walls, one cell of each block is filled with concrete and contains a No. 6 vertical reinforcement bar placed so that the rebar spacing is 2 ft-3-in. on center (OC) maximum. In completely filled masonry walls, all cells of the block are filled with concrete and contain a No. 5 vertical reinforcement bar placed so that the rebar spacing is 16-in. OC. All masonry walls are not load bearing and are not included in the overall building shear wall system. Masonry walls are designed to withstand dead loads, live loads, and seismic loads from the operating basis earthquake (OBE) and the design bases earthquake (DBE).

3.1.4.1.2 Response to Nuclear Regulatory Commission Information and Enforcement Bulletin 80-11

Masonry walls were reevaluated in the early 1980s in response to IEB 80-11. The reevaluation provided verification of the masonry wall adequacy, including support condition, global response of the wall, and local transfer of the load. Supplement 3.8C of the Unit 2 FSAR (Reference 12) provides the design criteria for the reevaluation. The walls are located in the control building shared by both units; therefore, these criteria also apply to Unit 1. Concrete masonry walls were analyzed according to working stress principles. The walls are designed, in general, to span horizontally, and this is typically how they were reevaluated. In some cases, two-way action was considered. All attachment loads to walls were determined from walkdown information. These loads were included in the IEB 80-11 evaluation.

The seismic response of the masonry walls was determined using an iterative procedure. A horizontal strip of the wall modeled as a beam strip was analyzed first assuming uncracked properties, and checked for cracking by applying the calculated inertia loads. If cracking occurred, an effective moment of inertia was determined. A second iteration was initiated to recompute the frequencies, mode shapes, and modal participation factors. This procedure was repeated until convergence was achieved. Both OBE and DBE loads were evaluated. American Concrete Institute (ACI) Standard 531-79, "Building Code Requirements for Concrete Masonry Structures" (Reference 27), was used as the basis for the structural reevaluation and OBE allowables. For some complex wall configurations and highly stressed walls, finite-element models were developed to more accurately calculate seismic responses. No masonry walls were reevaluated or qualified using arching or rigid-body analysis methods.

As a result of the IEB 80-11 evaluation, a limited number of masonry walls were modified. The typical modification provided additional pilasters and, in some cases, lateral support along the top edge of the masonry wall. The typical pilaster was a structural steel, wide-flange beam bolted to the floor and ceiling and along its length to bearing plates on the back side of the masonry wall.

3.1.4.1.3 Selection of Masonry Walls

Based on the conservative procedures used in the IEB 80-11 reevaluation and the subsequent modifications, a high-confidence level that the masonry walls would survive the Plant Hatch seismic margin earthquake (SME) existed. However, the EPRI SMA methodology requires an evaluation for an SME ≤ 0.3 g pga. Since section 5.0 of the EPRI SMA methodology states that the level of review may be minimal if the walls are well engineered, high-confidence-of-low-probability-of-failure (HCLPF) capacities were calculated for two masonry walls using the conservative deterministic failure margin (CDFM) approach.

The selection of two masonry walls, C130-48 and C130-14, was based on a review of the IEB 80-11 calculated stress levels of walls whose failure could be detrimental to the safe shutdown paths. These two walls were among the most highly stressed under the IEB 80-11 reevaluation program, and covered the many variables of Plant Hatch masonry walls. Wall C130-48, a 12-in.-wide block wall with partial concrete fill, has a stress level that barely met the IEB 80-11 reevaluation criteria. This wall was not modified, and its horizontal span of 14 ft is one of the longest. Wall C130-14, an 8-in.-wide block wall with both partially filled and fully filled segments, was modified by adding a pilaster and adding angles to the ceiling to take lateral loads according to the IEB 80-11 reevaluation program criteria. However, Wall C130-14 still had stresses close to the allowable stresses used in the IEB 80-11 reevaluation.

3.1.4.1.4 Evaluation of Masonry Walls

The HCLPF capacities for the selected walls were computed following the CDFM methodology outlined in sections 1, 4, and 6 of the EPRI SMA methodology. The guidelines for the CDFM computation, presented on pages A.94 through A.98 of NUREG/CR-5270 (Reference 28), were used to perform the calculation. The first step was to calculate the ultimate out-of-plane uniform load capacity. Using the ultimate load capacity and the actual weight per square foot of the wall surface, the effective spectral acceleration capacity was determined. The second step was to calculate the elastic natural frequency at the ultimate capacity of the wall, including the cracked properties of the wall. The associated spectral acceleration was determined based on the average of the 10-percent damped in-structure response spectra (IRS) of the supporting floor and the floor above. Since the failure mode of the wall is ductile, the elastic computed spectral response was reduced by an inelastic energy absorption reduction factor, K. The final step was to determine the HCLPF capacity by multiplying the ratio of the spectral acceleration capacity to the demand spectral acceleration (reduced for inelastic energy absorption) by 0.3 g.

For wall C130-48, the spectral accelerations increased rapidly for natural frequencies just below the calculated elastic natural frequency of 3.7 Hz. Therefore, a simplified nonlinear analysis was performed. The final effective inelastic natural frequency of out-of-plane flexure, accounting for uncertainty, ranged from 2.9 to 3.9 Hz, producing a slightly higher seismic demand than the elastic-computed spectral response. The resulting HCLPF capacity for wall C130-48 is 0.86 g, almost three times the SME review level of 0.3 g, showing that wall C130-48 has a high seismic capacity.

For wall C130-14, the calculated effective spectral acceleration capacity is 4.19 g. Since this greatly exceeded the 10-percent damped peak spectral acceleration of the average of the IRS above and below the wall, the natural frequency of the wall was not calculated; however, the peak spectral acceleration value, reduced by the inelastic energy absorption factor, was used as the effective spectral acceleration demand. Based on this evaluation, the HCLPF capacity for wall C130-14 is at least 2.10 g, showing that wall C130-14 has an extremely high seismic capacity.

The calculation of HCLPF values significantly higher than 0.3 g for the selected sample provides the basis to screen out all masonry walls at the SME level of at least 0.3 g. Since no other masonry walls were judged to be substantially more vulnerable to earthquake motion than the two investigated walls, the seismic review team (SRT) concluded that Plant Hatch masonry walls have a HCLPF capacity of at least 0.3 g pga.

3.1.4.2 Control Room Ceilings

Earthquake experience shows that suspended ceilings, such as the one in the main control room, are potentially vulnerable. The control room ceiling is composed of light fixtures and ceiling tiles of the same size supported on T-sections, which form a grid system. The grid system is supported vertically by clips and threaded rods attached to a system of parallel globe struts. The ends of the struts are connected to walls and supported vertically along their span with threaded rods connected to the reinforced concrete ceiling. The grid system is also connected to the walls of the control room using standard suspended ceiling hardware. Walkdown results indicated that no lateral ties are used to prevent in-plant distortion of the grid system. Therefore, the SRT concluded that control panels containing soft targets and/or essential relays could be vulnerable to impacts from the ceiling lights during an earthquake and identified the panels as outliers. Should ceiling tiles fall during a seismic event, serious damage is not expected because the tiles are relatively light. Operations personnel are aware that ceiling tiles may fall during an earthquake. Since the light fixtures weigh considerably more than the ceiling tiles, a modification was implemented to wire the lights directly to the reinforced concrete ceiling, thereby resolving the seismic interaction concern with the control room ceiling.

3.1.4.3 Seismic Category II Structures

Structures providing nonessential, commercial functions are classified as Seismic Category II. Screening table 2-3 of the EPRI SMA methodology requires an assessment of Seismic Category II structures that contain safety-related equipment or whose failure could cause the failure of adjacent Seismic Category I structures. The Plant Hatch structures that require evaluation based on these guidelines are the Units 1 and 2 turbine buildings. The failure of the turbine buildings could potentially damage the adjacent control building and the Unit 1 and Unit 2 reactor buildings. No safety-related success path equipment is located in the turbine buildings; however, a cableway passage, which is part of the turbine buildings and contains Seismic Category I cables, is below ground. Therefore, an assessment of the Unit 1 turbine building was performed. This evaluation also applies to the Unit 2 turbine building, since it is structurally identical to the Unit 1 turbine building.

The guidance given in table 2-3 of the EPRI SMA methodology for assessment of Category II structures, for an SME up to a peak spectral acceleration of 0.8 g, states that an evaluation is not required, provided the structure is capable of meeting the 1985 Uniform Building Code (UBC) zone 4 requirements (Reference 29). The original analysis, design criteria, and structural drawings were reviewed. Results of the reviews indicated that the original design was evaluated to demonstrate the turbine building would survive a DBE. Also, the turbine building was designed using UBC 1967 zone 1 requirements for seismic loading; therefore, an additional assessment was performed.

The turbine building is designed to resist tornado winds. Since the lateral forces of the tornado wind loading are quite large, the most efficient method to determine whether the turbine building is capable of meeting the 1985 UBC zone 4 requirements is to calculate the associated lateral seismic forces and compare them to the tornado lateral forces used in the design. The 1985 UBC zone 4 lateral loads were derived using an importance factor, I, of 1.0; a horizontal force factor, K, of 1.0; and C and S factors based on the calculated N-S and E-W first fundamental natural frequencies of the turbine building and soil column resonance obtained from the SMA soils evaluation. The resulting base shears were calculated and the associated lateral loads distributed in accordance with 1985 UBC procedures.

The cumulative seismic shears from the 1985 UBC zone 4 requirements were compared to the tornado wind shears from the turbine room floor down to the top of the base slab. This comparison shows that the E-W tornado wind shears far exceed the UBC seismic shears. Thus, the turbine building is designed for E-W lateral loads larger than the zone 4 seismic load. For the N-S direction, the tornado wind shears exceed the zone 4 seismic shears down to the turbine room floor. Below floor level, the zone 4 seismic loads are slightly larger. These slightly larger shears at the bottom portion of the turbine building are not a concern, since for the worst case, the tornado shear is only 18 percent less than the UBC seismic shear. These shear walls have a very large N-S load capacity, since most of the in-plane stiffness of the reinforced concrete load-bearing walls is N-S; however, as a check, the overall UBC seismic shear stresses were calculated using the effective shear area of the reinforced concrete walls between floors. In all cases, the calculated shear stresses caused by the UBC seismic loads are significantly less than the allowable shear stress of the concrete. The safety margin identified is at least 3.25.

Comparison of the tornado wind lateral loads to the UBC zone 4 seismic loads indicates that the turbine building is designed for larger lateral loads than the UBC seismic load, except for the lower elevations in the N-S direction. At these lower elevations, the capacity to withstand the N-S UBC seismic loads is more than adequate.

Evaluations performed in the early design phase of the plant were reviewed. These evaluations showed the turbine building would survive the DBE. The turbine building was modeled, and the DBE and OBE responses were calculated. The seismic loads on the pilasters were evaluated and shown to be well within acceptable stress limits. The DBE shears and moments were generally larger than the UBC zone 4 lateral seismic forces.

The construction details of the turbine building were reviewed to ensure they meet the intent of the 1985 UBC zone 4 requirements. The roof structure is a steel-frame structure with the same connection details, bracing details, columns, and roof trusses as the control building roof structure, which had previously been screened out. The reinforced concrete columns meet the lateral tie requirements of chapter 26 of the UBC. The reinforcement drawings show proper reinforcement at beam/column connections and cutouts in slabs.

Based on the above assessment, the Units 1 and 2 Seismic Category II turbine buildings have a HCLPF capacity of at least 0.3 g pga.

3.1.4.4 Reactor Building Roof Structure

Using the criteria in table 2-3 of the EPRI SMA methodology, structures such as the reactor buildings would normally be screened out. The reactor buildings were not initially screened out because of a concern involving the connection of the reactor building roof to the reinforced concrete vestibule. Evaluations indicated this connection was susceptible to damage from earthquake motion in the N-S direction; therefore, an SMA evaluation on the portion of the reactor building that could be affected if this connection failed under the SME was performed.

3.1.4.4.1 Unit 1 Evaluation

Roof Structure and Its Connection to the Vestibule

The portion of the reactor building that would be affected by the connection failure is the roof structure, which is a bolted steel structure that begins at the refueling floor and extends approximately 50 ft above the floor to the building roof. The roof structure supports the reactor building roof, the building crane, and the precast concrete siding. Figure 3.1-18 is a plan view of the Unit 1 reactor building roof structure and the reinforced concrete vestibule. Figure 3.1-19 is an isometric drawing illustrating the structural steel framing of the roof structure, and identifies a 14-in.-wide flange anchor beam embedded in the adjacent reinforced concrete vestibule, thereby tying the roof of the reactor building to the vestibule. Figure 3.1-20 is an isometric view of the anchor beam and a portion of the roof structure framing at that connection. The main purpose of the anchor beam is to transfer E-W lateral forces, mainly tornado winds, from the roof structure to the vestibule since the south end of the roof structure has no lateral bracing. The south end of the Unit 1 roof structure is completely open to the adjacent north end of the Unit 2 reactor building roof structure, thus forming a common open area above the Units 1 and 2 refueling floors. This facilitates the use of one common building bridge crane that can travel between Unit 1 and Unit 2.

Roof Connection Vulnerability

Since the roof connection to the vestibule was not originally designed to take N-S forces, it was assumed these forces were fully supported by the cross-bracing in the east and west walls. However, because of the stiffness of the vestibule and the location of the connection, the N-S seismic loads would result in significant forces on the connection.

Previous evaluations indicated that the weakest link is at the connection between the roof and the vestibule, because of the limited transfer of shear and axial forces from the roof structure to the relatively stiff vestibule structure. These previous evaluations also showed that the next weakest link is the roof structure cross-bracing members. Therefore, the response of the roof structure was calculated for two cases:

- Case 1: Roof Structure Connected to Vestibule.

  Conditions under which the anchor beam could transfer shear and axial forces, or an evaluation of the likelihood of the anchor beam connection failing, resulting in disconnection from the vestibule.

If Case 1 proved possible:

- Case 2: Roof Structure Disconnected From Vestibule.

  Conditions under which the anchor beam was assumed to be ineffective in load transfer, or an evaluation to ensure the roof structure could survive the SME without being connected to the vestibule.

Analytical Approach

Three-dimensional space-frame models for each case were used to develop equivalent stick model properties of the roof structure for the reactor building seismic models. Previous work showed that lateral stiffness of the cross-bracing system was less than the stiffness calculated using the full cross-sectional area because of the stiffness limits of the brace member in compression. Therefore, the bracing cross-sectional area was reduced by 10 percent. Unit loads were placed on the space frame, and from these results, 2 by 2 flexibility matrixes were developed for both the N-S and the E-W directions. The flexibility matrixes were inverted to obtain the stiffness matrixes. From the stiffness matrixes, equivalent beam properties were developed for the roof structure for both Case 1 and Case 2 for the N-S and E-W directions. These properties were made part of the reactor building seismic model. Figure 3.1-21 depicts the seismic model of the reactor building for Case 1 (roof structure connected to vestible) and figure 3.1-22 illustrates the seismic model of the reactor building for Case 2 (roof structure disconnected from the vestibule). SME responses were calculated for both cases.

- Case 1: Roof Structure Connected to Structure Vestibule

  Seismic results of Case 1 indicated potential axial and shear force overloads on the embedded anchor beam.

  A further assessment of Case 1 was performed to ensure all other elements and connections of the roof structure have sufficient seismic capacity for both connection to and disconnection from the vestibule.

  The resulting maximum accelerations for mass points at elevations 280 ft and 256 ft-6 in. were applied to a realistic mass distribution on the appropriate roof structure three-dimensional models. The forces and moments on all members and connections were calculated. The collinear forces and moments caused by N-S, E-W, and vertical earthquake components were combined according to the square root of the sum of the squares (SRSS) rule.

  The roof structure is a ductile structure and, therefore, would have ductile failure modes. Considering inelastic energy absorption resulting from the ductility of the roof structure, computed seismic stresses were reduced 20 percent in the capacity evaluation in accordance with the CDFM approach. Dead loads were combined with seismic loads. The largest seismic loads from the analyses were used, based on the three soil profiles. To ensure the maximum loads were found, the lower- and upper-bound soil profile results were used.

  The SME demand for Case 1 (roof structure connected to the vestibule) was evaluated to assure the maximum seismic demand on the critical roof members was identified. The maximum accelerations were calculated for all three soil profiles using the seismic model shown in figure 3.1-21. These maximum accelerations were multiplied by the appropriate measures to obtain the inertia forces, which were then applied to the three-dimensional space-frame model of the roof structure connected to the vestibule. Load combinations were the same as those used in Case 2 (roof structure disconnected from vestibule).

  The seismic demand was compared to that of Case 2 (roof structure disconnected from vestibule). Cross-bracing loads in the east and west walls were reduced. Cross-bracing loads in the north wall were larger. Some loads on structures increased, especially adjacent to the vestibule connections. However, in all cases, the seismic capacities of the members and connections under question were larger than the seismic demand.

- Case 2: Roof Structure Disconnected to Vestibule

  The largest seismic loads for Case 2 were in the east and west wall cross-bracing. The initial approach used to evaluate the shear capacity of a bay of cross-bracing was to calculate the full-tensile capacity of the tension member and the compression capacity of the compression brace using the unbraced length from its connection at the column to the center where the tension member crosses, and is bolted to, the compression member. The moment attributable to the eccentric gusset support was added. The combined horizontal component of these capacities is defined as the shear capacity of that bay of cross-bracing. The capacity of the tension brace and the compression brace is based on 1.6 multiplied by the value in Part I of the American Institute of Steel Construction Code (Reference 3) for the SMA evaluation. Using the factor of 1.6 is conservative because the CDFM approach for SMA allows the use of the factor of 1.7. Based on this approach, some of the upper bays of cross-bracing limited the HCLPF to 0.28 g pga.

A review of these results showed that excessive conservatism could be removed for the SME evaluation:

1. Seismic demand in excess of the seismic capacity of one bay will be redistributed by the roof beams to the other bays. (The four bays in the wall are identical.)

2. The simplified approach to compute the shear capacity of a single bay is also conservative.

Using a rigorous approach to calculate the shear capacity per bay resulted in an additional 5- to 10-percent margin. The upper bays of cross-bracing were reevaluated using the more rigorous techniques.

Using rigorous techniques, the capacity of the upper bay of cross-bracing is 299 kips per bay. Therefore, total capacity of the four upper bays is 1196 kips. The SME HCLPF capacity for a ductile bracing system is:

$$\text{HCLPF} = \frac{\text{Capacity}}{\text{K(demand)}}(\text{SME}) = \frac{1196}{0.8(1337)}(0.3\text{ g}) = 0.34\text{ g}$$

The strut members and their connections were evaluated for loads consistent with the methods used to calculate the seismic capacity of the cross-bracing, considering the redistribution of seismic loads when the seismic demand exceeds the seismic capacity of a cross-braced bay. This evaluation showed that the seismic capacity of the struts and their connections was greater than the seismic demand.

An additional evaluation was considered for Case 2 (roof disconnected from vestibule) to determine the effect of diaphragm stiffness of the roof deck on member loads. In the above analyses, diaphragm stiffness of the roof deck was neglected, because it was believed that cracking of the roof deck would significantly reduce the stiffness of the diaphragm. The effect of this assumption on the load distribution was evaluated. The three-dimensional model of the roof structure disconnected from the vestibule was rerun with the in-plane stiffness of the roof deck assumed to be basically rigid. The results show that the loads on the most highly loaded bays of cross-bracing were slightly lower when roof diaphragm stiffness was considered, because the diaphragm more evenly distributes the roof inertia loads to the cross-bracing. Some struts showed larger axial forces when roof diaphragm in-plane stiffness was included, but the seismic capacity of the struts and their connections was larger than the seismic demand. The SRT concluded that the roof structure HCLPF value is not affected by roof diaphragm in-plane stiffness and inclusion of this in-plane stiffness would more evenly distribute the roof inertia loads to all the braced bays.

Other Considerations

Evaluation of the details of the roof structure connection to the vestibule, discussed above, indicates that any release (e.g., loss of load transfer capability) would be localized and would not prevent the roof joint at that location from functioning. Impact between the roof and the vestibule, if release occurs, is judged not to be significant because of the inelastic energy absorption of the reactor building roof structure. Also, local yielding of the roof structure and spalling of the concrete of the vestibule attributable to impacting would not affect the ability of the plant to survive and safely shut down following an SME.

3.1.4.4.2 Unit 2 Evaluation

Since the Unit 1 and Unit 2 reactor buildings are basically identical up to the refueling floor, the Unit 1 reactor building SME IRS for the refueling floor were used as the seismic input to the Unit 2 reactor building roof structure. The resulting seismic loads were applied to the original three-dimensional space frame model of the Unit 2 reactor building roof structure. The resulting member and connection forces at the previously identified vulnerable locations were evaluated using the CDFM approach specified in the EPRI SMA methodology. In all cases, the seismic capacities of the members and connections in question exceeded the seismic demand.

3.1.4.4.3 Conclusions

In conclusion, the results of the evaluations of the Units 1 and 2 roof structures show a HCLPF capacity of at least 0.3 g pga.

3.1.4.5 Torus

The torus is located below and around the drywell. Figure 3.1-23 provides a cross-sectional view of the torus. The torus is supported by columns with sliding bases. Four shear ties are provided to resist the horizontal forces generated by an earthquake. Vertical seismic forces are resisted at the column bases. Eight vent pipes are equally spaced near the base of the drywell. These pipes penetrate the torus and are connected to one common vent header that has the same shape as the torus and is supported by struts from a ring girder in the torus. These struts are hinged at the base and top to allow for differential horizontal movements between the vent header and the torus.

The original design of the torus includes consideration of postulated accident loads which consist of pressure and temperature loads associated with a LOCA, seismic loads (DBE and OBE), dead loads, jet-impingement loads, hydrostatic loads of the water in the torus, overload pressure test loads, and construction loads. Following the establishment of the original design criteria, additional loadings were identified. These loadings were the result of the dynamic effects of drywell air and steam being rapidly forced into the torus during a postulated LOCA, and the torus response to safety relief valve (SRV) operation generally associated with plant transient operating conditions. The torus was modified extensively as a result of these additional loads. Seismic loads are a minor loading consideration in these load combinations.

To determine adequacy at the SME level, previous calculations were reviewed and scaled up as required.

The critical point for resisting vertical seismic loads is at the base of the support columns where 384 Williams rock bolts are used to attach the column to the base slab. Some rock bolts were found to be bent as a result of movement of the torus toward the reactor pressure vessel. Extensive tests and evaluations determined the capacity of the bolts in their existing condition. Georgia Power Company performed a study which showed the loads on the rock bolts at the DBE level were less than the reduced capacity of the bolts (Reference 30). The equation resulting in the worst-case bolt loads included dead load, SRV loads, LOCA loads, and vertical seismic loads. The vertical seismic loads were conservatively scaled up to the SME level and substituted into the equation. The seismic load was increased by 137 percent; however, the total bolt load was increased by only 3.3 percent. This illustrates the extremely small impact of seismic loads on the torus. The resulting bolt load was still well below the capacity of the bolts. Since large-break LOCA loads are not considered for SMA, this evaluation is considered conservative.

The torus seismic shear ties were designed to resist all horizontal loads. The original torus calculations performed by Chicago Bridge & Iron Company were scaled up to the SME level to demonstrate adequacy. The SME loads were determined from the 5-percent-damped IRS at elevation 87 ft. The natural frequency of the torus is given as 17.14 Hz in the Plant-Unique Analysis Report (PUAR) for the Mark I Containment Long-Term Program (Reference 31). Therefore, the maximum SME spectral acceleration was determined at 17.14 Hz or greater. The reevaluation showed that the seismic ties have adequate capacity for a HCLPF of 0.3 g pga.

The torus shell, supports, and internals were also evaluated. The finite-element computer calculations for the Mark I Containment Long-Term Program used the DBE IRS at 2-percent damping. The maximum spectral acceleration for the DBE 2-percent damped IRS was compared to the maximum spectral acceleration for the SME 5-percent-damped IRS to determine the seismic increase in the horizontal and vertical directions. These values were determined at a frequency of 17.14 Hz and above. The seismic increase attributable to the SME is 49 percent in the horizontal direction and 12 percent in the vertical direction. This increase is insignificant given the conservatisms in the original analysis. A summary of these conservatisms, as identified in section 8 of the PUAR, is given below:

1. Since the original analysis was simplistic, the sloshing effect (i.e., percent effective water mass) was not considered in the analysis assumptions. As demonstrated in section 8.3 of the PUAR, the fluid in the torus is not rigid and is often only 20 percent effective. Considering an effective water mass of 20 percent, the seismic stresses can be reduced by more than 50 percent.

2. The torus analysis included LOCA-related loads in addition to SRV, seismic, and normal operating loads. The EPRI SMA methodology does not consider the occurrence of a large- or intermediate-break LOCA. This obviously introduces considerable conservatism into the analysis.

3. Load combinations were combined using the absolute summation method. SRV loads are the major contributor to structural response in any single load combination, while seismic loads tend to be a minor contributor. A large increase in seismic loads typically results in a relatively small increase in the overall stress.

4. American Society of Mechanical Engineers (ASME) Code (Reference 32) service level D allowables may be used with the EPRI SMA methodology. The torus analysis described in the PUAR was performed using service levels B and C allowables. This introduces another source of conservatism into the analysis.

5. The EPRI SMA methodology also recommends using a ductility factor, K, of 0.80 for seismic loads, except in the case of shell buckling or anchorage pullout. This factor was not used in the analysis.

Based on the evaluation described above, the torus is acceptable for a HCLPF capacity of at least 0.3 g pga.

3.1.4.6 Reactor Pressure Vessels and Internals

This section discusses the SMA of the Plant Hatch Unit 1 and Unit 2 reactor pressure vessels and internals (RPV&I) for the SME. Table 2-4 of the EPRI SMA methodology states that insufficient data are available for recommendations to be made to screen the reactor internals out without an assessment.

3.1.4.6.1 Unit 1 Reactor Pressure Vessel and Internals

An SMA analysis of the Plant Hatch Unit 1 reactor building that included a simplified model of the RPV&I was performed. IRS were developed for 3, 5, and 10 percent of critical damping. The nuclear steam supply system vendor, General Electric (GE), was contracted to perform the SMA. Previous GE evaluations of the Plant Hatch Unit 1 RPV&I used seismic forces and moments for the evaluations, and no IRS were available for comparison. For this reason, the SME IRS could not be compared to the results of previous evaluations; however, compatible data existed for a sister plant. The sister plant RPV&I hardware is the same as the Plant Hatch Unit 1 RPV&I hardware, and the IRS from the adequacy evaluation of the sister plant were available. The sister plant hardware had been qualified, and it was, therefore, sufficient for the seismic margin assessment to show the Plant Hatch Unit 1 SME IRS are bounded by the sister plant IRS in the frequency range of interest.

The sister plant is a recently licensed plant for which a complete design adequacy evaluation for seismic and other loads was performed. The sister plant has a Mark II containment; therefore, the RPV&I experience hydrodynamic and seismic loads. This is not the case for Plant Hatch Unit 1, which has a Mark I containment. This means that the sister plant was designed to resist a given seismic load and higher frequency hydrodynamic loadings. Because the sister plant's IRS envelop Plant Hatch's SME IRS in the frequency range of interest, Plant Hatch RPV&I demonstrated a HCLPF capacity of at least 0.3 g pga.

Figure 3.1-21 shows the Plant Hatch Unit 1 reactor building model used in the SMA evaluation. For the RPV&I assessment, the IRS for RPV nodes 19, 20, 21, and 22 were compared with the IRS for similar locations in the sister plant model which has a detailed representation of the RPV&I. The Plant Hatch Unit 1 IRS at the top of the pedestal (node 16) were also compared with the IRS at the top of the pedestal for the sister plant. The seismic input to the RPV&I is transferred primarily through the pedestal and stabilizer.

It should be noted that the Plant Hatch Unit 1 horizontal RPV&I model has a frequency of 4.57 Hz. The fundamental frequency of the Plant Hatch Unit 1 vertical model was not available; however, similar plants have a fundamental natural frequency of approximately 22 Hz.

The horizontal Plant Hatch Unit 1 SME IRS were properly broadened, and the envelope of the three SME IRS was reduced by 20 percent in accordance with the EPRI SMA methodology to account for the inherent ductility of the RPV&I. The reduced SME IRS were then compared to the sister plant's faulted-load combination IRS of DBE, jet reaction, and feedwater line break. The sister plant's IRS were combined by the square root of the sum of the squares (SSRS) rule.

Figures 3.1-24 through 3.1-28 show the horizontal IRS comparison for node points 16, 19, 20, 21, and 22, respectively. Figures 3.1-24 through 3.1-27 compare the Plant Hatch Unit 1 SME IRS for 3 percent of critical damping to the sister plant's 4-percent IRS (SRSS of DBE and jet reaction). The Plant Hatch Unit 1 SME is enveloped by the sister plant's IRS in the frequency range of 2.5 Hz and higher (figures 3.1-24 to 3.1-26) to 3.6 Hz and higher (figure 3.1-27). Figure 3.1-28 compares the Plant Hatch Unit 1 SME IRS for 4-percent critical damping to the sister plant's 4-percent IRS (SRSS of DBE, jet reaction, and feedwater break). For this case, the IRS are enveloped by the sister plant's IRS above 4.35 Hz. All appropriate Plant Hatch Unit 1 SME IRS are enveloped above the fundamental natural frequency of the RPV&I of 4.57 Hz.

Since the fundamental frequency of the vertical RPV&I model is approximately 22 Hz, only the maximum absolute accelerations were compared with the sister plant's values. The top of the pedestal and two RPV locations were selected for comparison. The Plant Hatch vertical accelerations were reduced by 20 percent because of the ductile capacity of the RPV&I. The vertical accelerations of the sister plant were based on the SRSS of DBE, SRV (ADS), and chugging (faulted load case). A comparison of these vertical accelerations demonstrated that the sister plant accelerations are appreciably larger than the Plant Hatch accelerations, demonstrating that the RPV&I have capacities greater than the Plant Hatch SME demand for the vertical direction.

This comparison of the Plant Hatch Unit 1 RPV&I horizontal IRS and vertical maximum absolute accelerations to those of a sister plant having identical RPV&I hardware verifies that the Plant Hatch Unit 1 RPV&I have a HCLPF capacity of at least 0.3 g pga.

3.1.4.6.2 Unit 2 Reactor Pressure Vessel and Internals

For the Plant Hatch Unit 2 SMA, the RPV&I models supplied by GE for Units 1 and 2 were compared. A listing of the first six natural frequencies is provided in table 3.1-3.

A comparison of the natural frequencies and descriptions in table 3.1-3 indicates close agreement between the frequency values and the mode descriptions for the first four modes. The four modes for Unit 2 exist between the frequency range of 4.46 to 14.60 Hz.

Figure 3.1-29 provides the dynamic model of the Plant Hatch Unit 2 RPV&I which is similar to the Unit 1 model. However, some of the member properties are different, as well as the spring boundary elements, such as the refueling bellows and the stabilizer spring. In general, the refueling bellows springs for Unit 1 are stiffer than those for Unit 2, but the stabilizer spring for Unit 2 is stiffer than the stabilizer spring for Unit 1. However, the natural frequencies for the first four modes are not significantly different.

Figure 3.1-30 provides the Plant Hatch Unit 2 reactor building model, which includes mass points 16, 19, 20, 21 and 22. These mass points represent the boundary connection mass point and the simplified internal model mass points of the RPV&I, respectively. A representative comparison of the DBE IRS to the SME IRS is provided in figure 3.1-31. This overlay of the horizontal spectra indicates that the DBE IRS significantly envelop the SME IRS at the frequency range of 2 Hz and above. For the SMA evaluation, this same comparison was made for mass point 22 at the top of the RPV, which is provided in figures 3.1-32 through 3.1-34 for the E-W, N-S, and vertical directions, respectively.

From these comparisons, the horizontal DBE IRS again significantly envelop the SME IRS for the frequency range greater than 2 Hz. Based on these comparisons for all the horizontal natural frequencies of the RPV&I, the original design basis IRS exceed the SME.

A comparison of the vertical DBE IRS shows that, in general, the vertical SME IRS envelop the original DBE IRS. From the Unit 1 SME evaluation of the RPV&I, the estimated vertical natural frequency is approximately 22 Hz. In general, Plant Hatch Units 1 and 2 RPV&I have enough similarities to conclude that their vertical natural frequencies are not significantly different. A summary of Unit 2 RPV&I components is presented with their corresponding stress states and code allowables in table 3.9-4 of the Unit 2 FSAR (Reference 12). A review of this table indicates that the original design basis condition of the major RPV&I components, in general, has stress levels at or below 50 percent of the allowable stresses. Therefore, a significant margin of safety of 2 or more exists for the original DBE loadings. Since the horizontal SME demand is actually less than the original DBE demand, in the frequency range of interest, the margins of safety for SME are expected to be even larger. For the vertical component, the stress summary provides information that the vertical design basis acceleration was applied statically at a value of

0.40 g. This design basis value envelopes the vertical SME at 4.0 Hz and above, where the frequency range of interest is expected to be approximately 22.0 Hz.

Based on the comparisons of the seismic models for Units 1 and 2, the comparison of the Unit 2 DBE IRS and the SME IRS, the comparative analysis provided by GE for Plant Hatch Unit 1 RPV&I, and the low stress levels summarized in table 3.9-4 of the Unit 2 FSAR, it is concluded that the Plant Hatch Unit 2 RPV&I have a HCLPF capacity of at least 0.3 g pga.

3.1.4.7 Soils Evaluation

The EPRI SMA methodology (Reference 5), when applied to the soils, follows these guidelines:

1. The SME is conservatively specified. (The SME is discussed in section 3.1.3.1.)

2. The capacity (e.g., shear stresses required to cause liquefaction) assessment for a given response is selected conservatively.

3. The response of the earth structures (e.g., soil profile, slopes) to the SME is median centered.

This section contains a description of the development of the soil profile, the evaluation of liquefaction potential, the evaluation of ground settlement, and the evaluation of the stability of the soil slopes in the river intake area.

The assessment of the issues relating to soils was performed by Dr. I. M. Idriss and other personnel of Woodward-Clyde Consultants (WCC). The complete WCC Report concerning issues related to soils for the Plant Hatch SMA is included as Appendix C of EPRI NP-7217-SL (Reference 1).

The application of the soil profile for soil-structure interaction (SSI) analysis is described in section 3.1.3.

Plant Hatch is a soil site. A general description of the site geology is given in section 3.1.1.1. Three soil-related issues can affect the success paths: 1) determination of appropriate strain-compatible soil profiles for the plant area, 2) assessment of soil liquefaction, and 3) assessment of the slopes where failure can affect plant shutdown capabilities.

3.1.4.7.1 Development of Soil Profiles

The subsurface conditions in the plant area were evaluated based on the logs of 54 borings performed prior to plant construction (1967 to 1970). Based on these borings, a generalized soil profile was developed as shown in figure 3.1-35.

The shear wave velocities in the plant area that were assumed and used by EQE, Inc., in a prior SSI analysis for the NRC (Reference 13) were reviewed. The correlation of shear wave velocity with standard penetration test (SPT) resistance, N, proposed by Sykora and Stokoe

(Reference 33) was used to determine the reasonableness of the velocities that were previously assumed. This examination led to the conclusion that the shear wave velocities selected for the upper 130 ft of the soil profile appear reasonable.

Based on this examination and the generalized soil profile, two shear wave velocity profiles were constructed for the plant area as shown in figure 3.1-36. The two profiles are identical below a depth of 55 ft. Velocity Profile I reflects the fact that the upper 55 ft were excavated and replaced by backfill around the reactor building and control building, and to some extent below the diesel generator building. The shear wave velocities in the upper 55 ft of Profile I (consisting of fill) were estimated based on shear wave values for comparable backfill. Velocity Profile II reflects the in-situ soil where the upper 55 ft contains cemented sands. These profiles, corrected for large strains, were used for the evaluation of the liquefaction potential for the plant area. (Refer to section 3.1.4.7.2.)

The water level was conservatively assumed to be at elevation 85 ft, 45 ft below grade. This water level coincides with the 10-year flood water level. The normal water level is at elevation 75 ft. Water level measurements taken throughout the past 18 years register the highest water level at elevation 79 ft.

The subsurface condition in the river intake area was examined based on the logs of 45 borings made between 1967 and 1969. Two cross-sections, shown in figure 3.1-37, were examined. Cross-section B-B' was determined to be the most critical. Its stability evaluation is discussed in section 3.1.4.7.4.

Additional studies were made to evaluate the sensitivity of soil material damping on response and spatial variation of the ground motion for these soil profiles. For the SMA of Plant Hatch, the SME is defined as a free-ground response surface spectrum (i.e., elevation 130 ft). Therefore, the input to the foundation is most accurately determined by deconvolution and accounting for kinematic interaction. Figure 3.1-38 is an example of a response spectrum of the free field at the reactor building foundation level (55 ft below grade) for the average soil material damping and the lower range soil material damping in terms of pseudo-relative velocity versus period. The results, using the average soil damping, were found to be basically identical to those using the lower range soil damping curve.

The damping curves used for sands were as found in Seed and Idriss (Reference 21). The average damping curve for sands was used for the evaluation of the liquefaction potential for the plant area. Figure 3.1-39 is the response spectrum calculated in the free field at the reactor building foundation level (55 ft below grade) in terms of spectral acceleration versus period. The term "target" is the SME free-field ground spectrum at ground surface (elevation 130 ft). The "average modulus" is the strain-compatible Profile II. Different spectra were developed by varying the modulus from 0.75 to 1.8 multiplied by the "average modulus." The term "Unit 1" refers to the Plant Hatch Unit 1 DBE ground response spectra. Figure 3.1-39 shows that the SME will represent an increase in motion at the reactor building foundation level as compared to the DBE used for the original design bases.

3.1.4.7.2 Evaluation of Liquefaction Potential

Liquefaction potential attributable to the SME in the plant area was evaluated using observations of liquefaction in previous earthquakes as described in Appendix C of the EPRI SMA methodology. In addition, settlements in the plant area resulting from dissipation of excess pore water pressure were estimated to evaluate the impact of potential liquefaction at depth on the ability of the plant to safely shut down.

Evaluation of the liquefaction potential attributable to the SME in the plant area consisted of the following steps:

1. The shear stresses induced by the SME were calculated using a ground response analysis procedure. The SHAKE computer program (Reference 22) was used with one of the horizontal synthetic time histories applied at the ground surface. The variations of modulus and damping with shear strain were based on the average values published for sands (Reference 21).

   The maximum shear stresses were calculated for both Profile I and Profile II. The maximum shear stresses induced by the SME were calculated to be the average of those calculated using the shear wave velocities of Profile I and Profile II. These average maximum shear stresses were multiplied by 0.65 to convert them to equivalent uniform shear stresses. This factor is recommended by Seed and Idriss (Reference 34) and is commonly used for this purpose in liquefaction studies.

2. The shear stresses required to cause liquefaction were estimated using graphs that relate these shear stresses from field observations to the standard penetration test (SPT) resistance.

   The SPT values were obtained from borings in the plant area. Adjustments were made to these graphs for earthquake magnitude and vertical effective stresses. The raw blow count, N, for each SPT sample was adjusted to account for the fact that a donut hammer had been used; also, the N values were converted for an initial 1 ton/ft^2 effective overburden pressure on the soil element. In addition, blow count values, N, were increased to obtain an "equivalent clean sand" value for soil samples that were not essentially clean sands.

3. The stresses induced by the SME were then compared to those required to cause liquefaction at various depths in the plant area. This comparison was made in terms of the ratio of shear stress required to cause liquefaction divided by the shear stress induced by the SME at the same depth in the soil profile. This ratio represents the margin available to resist liquefaction at a site caused by SME ground motion. This ratio is referred to as the factor of safety against liquefaction.

As expected, SPT data varied within the plant area; therefore, the data were evaluated statistically to obtain mean and standard deviation values within given strata. Thus, the stresses required to cause liquefaction depend on the selected statistical values. The following minimum values of the ratio of shear stress required to cause liquefaction divided by the shear stress induced by the SME were selected for the SMA.

SPT Values Used	Minimum Required Margin
Mean	1.50
Mean - σ/2	1.30
Mean - σ	1.05

The mean represents the arithmetic average of the data; σ is the standard deviation.

The only portion of soil that is susceptible to liquefaction for the level of earthquake under consideration is the range of 45 to 90 ft below grade. Within this range, the minimum calculated margin is 1.42, 1.20, and 0.98 based on the use of the mean, mean - σ/2, and mean - σ values of N, respectively. Figure 3.1-40 graphically shows the equivalent uniform shear stress induced by the postulated SME (pga equals 0.3 g) with curve 1, the plot of the shear stress required for liquefaction based on mean - σ/2 values of N with curve 2, and mean values of N with curve 3.

To meet the minimum margins of 1.5, 1.3, and 1.05, the peak horizontal ground surface acceleration should be reduced to 0.28 g. This assessment of liquefaction relates to excess pore water pressure in a soil zone 45 to 90 ft below grade. The actual importance of this on the HCLPF of the plant is described below.

3.1.4.7.3 Evaluation of Ground Settlement

The significance of liquefaction at such depth on the ability of the plant to safely shut down is best evaluated by estimating settlements in the plant area resulting from dissipation of excess pore water pressure (PWP) generated by the postulated SME. Estimations were made using procedures that utilized volumetric strains. These strains were integrated over the appropriate depth of the soil profile to obtain the estimated settlements.

Appendix C of the EPRI SMA methodology provides three different procedures to estimate settlement in the plant area resulting from dissipation of excess PWP that may be generated by the postulated SME. It should be noted that procedures 1 and 3 use maximum shear strains for an SME of 0.3 g pga to finally obtain volumetric strains from which settlements are calculated.

The estimated settlements in the plant area attributable to an SME range from less than 1/4 in. to 1-1/2 in. These results suggest that the potential settlements that may occur are in the range of 1 in. to 1-1/2 in., with estimated differential settlements ranging from less than 1/2 in. to potentially 1-1/2 in.

These settlements are incorporated into the SMA of components where HCLPF capacity may be reduced because of relative displacement caused by settlement. A review of the plant layout revealed the only vulnerable areas impacted by settlement and differential settlement are buried structures and pipe penetrations. The buildings are separated by at least a 3-in. gap, so no significant relative interaction is judged possible between the buildings that would affect the safe shutdown of the reactor as a result of differential settlement. Thus, the significance of liquefaction on the plant HCLPF capacity is controlled by the impact of settlement and differential settlement on buried structures and pipe penetrations. Section 3.1.4.8 addresses the effects of settlement and differential settlement on buried structures and piping penetrations. This assessment demonstrates a HCLPF capacity of at least 0.3 g pga.

Therefore, the HCLPF for the plant relating to soil liquefaction is tied to the HCLPF of components affected by the corresponding settlement and differential settlement and not the onset of excessive pore water pressure. Such settlements were shown not to have any detrimental effect on the facilities in the plant area.

3.1.4.7.4 Evaluation of Slope Stability

A topographic map of the site was reviewed to determine the critical slopes requiring evaluation. The conclusion was that the slope adjacent to the river intake structure should be assessed for potential instability. Figure 3.1-37 is a portion of that topographic map; cross-section B-B' was judged to be the critical section for this purpose. This section is adjacent to the river intake structure. Buried service water lines and reinforced concrete electrical conduit banks traverse this area.

The available borings in the river intake structure area were examined to assess the general subsurface conditions. Figure 3.1-41 is the generalized soil layering for cross-section B-B'.

A. Overall Stability

Using the soil properties of layers 1 through 4, the pre-earthquake minimum factors of safety against sliding of the four potential slide surfaces were determined to be significantly above 3. The minimum post-earthquake factors of safety were computed assuming that the SME causes the shear strength in layer 4 to decrease to the residual strength. Layer 4, which extends from approximately elevation 60 ft to 50 ft, as shown in figure 3.1-41, is considered liquefiable. Therefore, its strength was assumed to be reduced to the residual strength.

The original estimate of the residual strength was approximately 300 lb/ft^2. Using this residual value, the minimum post-earthquake factor of safety against sliding was calculated and found to be equal to 1.5 or higher, depending on the potential slip surface assumed.

Thus, the slopes in the river intake structure area are unlikely to experience serious instability as a result of the postulated SME. However, limited amounts of lateral movement were assumed to be likely. The best estimate of the amounts of lateral movement is discussed below.

B. Lateral Movement

Figure 3.1-42 shows the same soil profile as given in figure 3.1-41, except that it is divided into potential slide surfaces A, B, C, and D. Procedures used to estimate the lateral movements of the potential slide surfaces A, B, C, and D are based on Newmark's approach (Reference 35) as augmented by Goodman and Seed (Reference 36), and Makdisi and Seed (Reference 37). The procedure considers that a maximum acceleration k_{max} is applied and that the slope has a yield acceleration k_y. If, during shaking, k_{max} exceeds k_y, an amount of permanent lateral movement is imparted to the slope. As shaking continues, and if k_{max} again exceeds k_y, additional permanent lateral movement takes place. At the end of shaking, the total permanent lateral movement is then the sum of the individual movements caused whenever k_{max} exceeded k_y.

A residual strength was determined for layer 4. It was assumed that this strength would be reached at the end of shaking. The value of k_{max} is assumed equal to 0.3 g for potential slide surface A and decreasing linearly to 0.2 g for surface D. These assumptions produced the curve shown in the upper right-hand corner of figure 3.1-42 labeled "best estimate." As illustrated, the largest best-estimate movement is 2-1/2 in. at slope A. The upper bound represents the case where k_{max} equals 0.3 g for all potential slide surfaces. Additional calculations have been made using a conservatively assigned average shear strength for the full shaking. The results of these additional calculations indicate a lateral movement of 3 in. to 4 in. Based on all calculations made, it is estimated that lateral movement along surface A is approximately 2-1/2 in.

The effect of lateral movement of the slope, including settlement on the buried piping and electrical conduit banks, is described in section 3.1.4.8.

3.1.4.8 Buried Structures

The SMA evaluation of buried structures that are part of the chosen safe shutdown paths included the Unit 1 plant service water (PSW) line, the Unit 1 residual heat removal service water (RHRSW) line, the Unit 1 buried electrical conduit duct runs, and the Unit 1 buried fuel oil tanks. The evaluation also included the typical penetrations where underground piping and electrical conduit enter structures or buildings. The Unit 1 evaluation was used to demonstrate the adequacy of the Unit 2 buried structures. Based on the assessments described in the following sections, the Plant Hatch buried structures have a HCLPF capacity of at least 0.3 g pga.

3.1.4.8.1 Seismic Distortions in Buried Pipelines and Conduit Runs

The buried service water lines and electrical conduit duct runs were evaluated for a combination of SME ground shaking, the best estimate of lateral slope movement, and potential differential settlement.

The evaluation of stresses and strains caused by ground shaking (seismic waves) was performed using the procedures given in Reference 24. The source of the SME is considered to be relatively close (25 km). Shear waves were therefore assumed to be the predominant seismic waves that affect these buried lines. In addition to GL 88-20, Supplement 4 (Reference 38), table 7-9 of the EPRI SMA methodology and References 39 and 40 were consulted and used, as appropriate.

A best-estimate lateral movement of the slope south of the intake structure is 2.5 in. This value was used in the evaluation of the buried pipe lines and buried electrical conduit duct runs. The location of this slope movement is approximately 60 ft south of the intake structure. The lateral movement is discussed in detail in the soil evaluation (section 3.1.4.7.4.)

As discussed in section 3.1.4.7.2, only one soil zone, 45 to 90 ft below grade, has a potential for liquefaction. The significance of liquefaction on the ability of the plant to survive the SME and retain the capability of being safely shut down is related to the potential settlement in the plant area resulting from the dissipation of excess pore water pressure in that soil zone. The maximum differential settlement was estimated to be 1.5 in.

The differential settlement was vectorially combined with the best-estimate lateral slope movement adjacent to the intake structure to obtain a resulting displacement of 3 in. The appropriate forces and moments caused by this resultant displacement were combined absolutely with the forces and moments caused by ground shaking to approximate the maximum seismic demand on the buried pipe lines and buried conduit duct runs.

3.1.4.8.1.1 Buried Pipe Lines. Both the PSW lines and the RHRSW lines were evaluated for ground shaking, combined with the resultant displacement caused by best-estimate lateral slope movement and differential settlement. The pipe lines are ductile steel, butt-welded pipe which has a high inelastic capacity and is seismically rugged. Conservative elastic analyses were performed to demonstrate adequate margin.

Stresses caused by internal pressure, stresses resulting from displacement caused by slope movement and differential settlement, and stresses resulting from moment and axial load caused by earthquake ground shaking in the elastic analysis were combined absolutely. Elastic analysis techniques were used to calculate stresses. To calculate the resulting stresses caused by the displacement of the slope movement with differential settlement, a deformed deflected shape was estimated. Both pipes were assumed to deflect with zero rotation at the face of the intake structure and at the point where these pipe lines traverse the slope at section B-B' of figure 3.1-37, where the best-estimate slope lateral movement of 2.5 in. was estimated. This is equivalent to a fixed-fixed beam 62 ft long, with a support displacement of 3 in. The stresses caused by lateral slope movement plus differential displacement account for, on the average, 56 percent of the total stress.

The calculated stress levels were checked against service level D limits. In both cases, the level D allowable is 3.0 S_h, where S_h is the normal allowable stress. Equation 9 of NC-3653.1 (Reference 41) was used with an additional stress. This is the axial force caused by ground shaking divided by the pipe cross-sectional area.

For the 30-in.-diameter PSW pipe, the safety margin was calculated as 1.56. The safety margin is equal to the service level D allowable divided by the maximum calculated stress. The safety margin was calculated as 2.18 for the 18-in.-diameter RHRSW pipe. These margins definitely show adequate capacity of the buried pipes to withstand the combination of SME ground shaking, lateral slope displacement, and differential ground settlement. The calculations were performed to demonstrate a minimum margin for buried structures in a cost-effective manner; however, there is significantly more margin than identified by these calculations. This added margin could be shown by inelastic analysis. The limits of the inelastic analyses are strain limits as given in Reference 40. Additional margin is available through this more sophisticated analysis, but considered unnecessary.

Prior to entering the intake structure, the piping bends in several locations, especially the 30-in.-diameter PSW line. This provides additional flexibility to handle differential displacement along the buried piping and between the buried piping and the intake structure. Also, it should be noted that the seismic displacements of the structures at grade and below, calculated as part of the SMA SSI analysis of these structures, are insignificant.

In all cases, the calculated seismic response displacements of the structures at grade and below are less than 20 mils. The differential displacement between buildings resulting from building seismic displacement and differential settlement is not important because:

1. The relative displacements are small.

2. The long distance between the intake structure and the other buildings negates the significance of seismic displacements and differential settlement.

3. The number of turns or bends in the buried piping between buildings provides flexibility.

4. The piping itself is ductile. Actually, any axial force caused by differential displacement parallel to the piping axis is borne by the skin friction between the pipe and the soil. The buried steel piping has a protective coating of coal-tar enamel which would produce a relatively high coefficient of friction.

Assessment of the seismic adequacy of the typical pipe penetration for underground piping is discussed later in this section.

3.1.4.8.1.2 Buried Conduit Duct Runs. A Unit 1 buried concrete conduit duct run close enough to the edge of the slope to potentially experience a lateral movement was evaluated in a manner similar to that used for the buried service water piping previously discussed. As with the buried piping, the calculation was performed to provide, in broad terms, documentation that the reinforced concrete duct run would survive the SME. An exact determination of the forces and moments in the duct run was not attempted. The ultimate safety concern is the integrity of the cables themselves. The structural integrity of the buried conduit run is secondary.

The buried duct runs are rows of 4-in.-diameter PVC conduits encased in solid reinforced concrete. The conduits encase the cables, whose integrity is the subject of the assessment. The SRT determined that there is sufficient slack in the cables to survive the best-estimate lateral slope movement combined with differential settlement and ground shaking. However, the buried duct run was assessed to determine its structural integrity.

The same procedures, discussed in section 3.1.4.8.1, were used to calculate forces and moments caused by ground shaking. In addition, the forces and moments caused by a differential displacement of 3 in. were calculated. For this case, the assumed deflected shape was a fixed-fixed beam 56 ft long with a support displacement of 3 in. The distance from the point where the Unit 1 buried duct run traverses the slope at section B-B' of figure 3.1-37 to the face of the intake structure is approximately 56 ft. Actually, the duct run turns and enters the intake structure approximately 29 ft past the face of the intake structure. The cracked section properties were used. The axial forces and moments were evaluated using a conservative straight line interaction equation and the ultimate capacities reduced by the appropriate Φ factors as the allowables. A margin of 1.4 was determined from this evaluation.

Based on the judgment that there is sufficient slack in the cables to survive the displacement anticipated and the fact there is still a margin before the ultimate capacity of the duct run is reached, the buried conduit runs will survive the SME without damage to the cables.

3.1.4.8.2 Penetrations

The adequacy of penetration assemblies to allow the buried piping to flex where it penetrates a building or structure was reviewed as part of the SMA of underground piping. Figure 3.1-43 is a schematic of a typical penetration for buried piping entering a structure. Note that this penetration has rattle space that provides flexibility to accommodate relative motion. Also, the 1/4-in. fillet weld would yield before the 30- or 18-in. pipe, thus preventing failure of the piping as a result of differential displacement of the building and the buried piping.

The reinforced concrete portion of the buried conduit duct run terminates next to the exterior face of the building with a 1/2-in. expansion joint (figure 3.1-44). Steel conduits are used at these interfaces. This type of penetration provides flexibility while protecting the cables.

Penetrations are designed to provide flexibility so that piping or cable is not damaged by displacement during an SME.

3.1.4.8.3 Buried Fuel Oil Tanks

As stated in the EPRI SMA methodology, buried tanks are not particularly vulnerable to seismic damage, especially if the SME peak spectral acceleration is no greater than 0.8 g pga. Therefore, the tanks are screened out. Damage may occur at piping connections if there is large motion of the soil surrounding the buried pipe relative to the motion of the tank.

The SRT walked down the buried fuel oil tanks to the extent possible. The manway covers were removed, and the interior of the manways was inspected. No concerns were identified with the fuel oil pump or piping. The manway covers are hinged and are larger than the opening; therefore, a cover cannot possibly fall into a manway.

The fuel oil tanks are in an area that will not experience lateral slope displacement. The 2-in.-diameter fuel oil lines will experience forces and moments caused by ground shaking, in addition to some possible differential settlement between the fuel oil tanks and the diesel generator building. The piping layout drawing for the fuel oil system shows five 2-in.-diameter lines exiting the manway, three of which make 90-degree bends just outside the manway. The largest potential displacement is differential settlement between the buried fuel oil tanks themselves and between the fuel oil tanks and the diesel generator building.

The largest differential settlement is estimated to be only 1-1/2 in. Review of the piping layout drawing for the fuel oil system showed there is a sufficient length of pipe and enough turns so that damage to these small-diameter fuel oil lines is not likely. The effect of ground shaking is also considered to be insignificant because of the flexibility of the 2-in. lines and the length of the assumed seismic wave of approximately 500 ft. The fuel oil lines are classified as Seismic Category I. These lines are Schedule 80 carbon steel pipe and the fittings are socket welded. These lines enter the diesel generator building through penetrations made of 4-in.-diameter pipe sleeves that provide flexibility at that location.

3.1.4.9 <u>Equipment Capacity Evaluations</u>

All Plant Hatch Unit 1 and Unit 2 equipment included in the scope of IPEEE-Seismic was shown either to possess a HCLPF capacity of at least 0.3 g pga, or was modified to achieve that capacity. A list of items requiring modification to obtain a HCLPF capacity of at least 0.3 g pga is included in Appendix I of this report. The equipment identification number, equipment class, equipment description, the plant area where the component is located, a description of the outlier, a description of the outlier resolution, and the modification method for each item are included.

3.1.4.10 <u>Relay Chatter Evaluation</u>

Per section 3.2.4.2 of NUREG-1407 (Reference 42), focused-scope IPEEE plants are not required to perform a full-scope SMA relay chatter evaluation, but are only required to locate and evaluate low-seismic-ruggedness relays as given in Appendix E of EPRI NP-7148-SL (Reference 43). Because Plant Hatch Units 1 and 2 are both USI A-46 plants, a full-scope relay chatter evaluation was performed for all components within the scope of the USI A-46 program. The results of the USI A-46 relay chatter evaluation are documented in the USI A-46 Summary Reports for each unit (References 7 and 44). Since low-seismic-ruggedness relays are identified as part of the USI A-46 evaluation, components that are within the scope of the Units 1 and 2 IPEEE, but which were not included within the USI A-46 scope, were evaluated to determine whether any low-seismic-ruggedness relays were associated with these components. A list of the low-seismic-ruggedness relays identified for each unit, including the amount of each type found, is given below:

Relay Type	Number of Relays Unit 1	Number of Relays Unit 2
GE CEH	3	2
GE CFD	9	6
GE CFVB	3	2
GE HGA	94	80
Westinghouse HU	6	6
Total	115	96

Relay type GE HGA is listed as a low-seismic-ruggedness relay only in the deenergized, normally closed condition. In most cases, the contact configuration was not determined if the relay is nonessential or an operator action is used as the resolution. Therefore, the total number of GE HGA relays listed above includes several relays that are not low-seismic-ruggedness relays, because they are used with a contact configuration that differs from the configuration described above.

All of the low-seismic-ruggedness relays identified as part of the USI A-46 or IPEEE evaluation for both units were resolved at a HCLPF level of at least 0.3 g pga by determining that either malfunction of the relay is acceptable (i.e., nonessential) or operator actions can be used to reset relays or restore systems to operation.

3.1.4.11 Internal Flooding

As part of the seismic capability walkdown, the SRTs evaluated all potential internal flooding sources in areas containing SSEL equipment. The primary focus of the flooding evaluation was on tanks and piping. No concerns with flooding were identified. All tanks in these areas are well anchored and free from adverse seismic spatial interaction effects.

3.1.4.12 Seismic-Fire Interaction

The seismic-fire interaction evaluation for the Plant Hatch Units 1 and 2 IPEEE addressed the following potential interaction concerns.

3.1.4.12.1 Seismically Induced Fires

All hydrogen or other flammable gas or liquid storage vessels in areas with safe shutdown or safety-related equipment are designed and anchored for Seismic Category II/I considerations. Areas with equipment containing significant amounts of combustible liquids have containment curbing or are provided with acceptable drainage systems. As part of the Fire Hazards Analysis and Fire Protection Program (Reference 45), walkdowns are used to verify that the as-built configuration of the plant agrees with the design basis. During the seismic capability walkdown, the SRTs identified no potential seismically induced fire concerns in areas containing SSEL components. Therefore, seismically induced fires are not a concern with a plant HCLPF capacity of 0.3 g pga.

3.1.4.12.2 Seismic Actuation of Fire Suppression Systems

This section addresses the possibility of seismically induced relay chatter potentially resulting in inadvertent actuation of the fire suppression sprinkler system. The Plant Hatch water suppression system consists of both normally dry and wet pipes. The sealed preaction sprinkler system heads are passive components that only open upon a rise of ambient temperature to the melting point of the fusible links on sealed sprinkler heads. Therefore, even if the sprinkler system is flooded, the sprinkler heads will not open in the absence of heat generated by a fire. Actuation of water deluge systems is similarly controlled by a fusible link.

In the unlikely event that inadvertent actuation of a sprinkler head did occur, the effect would be minimized through the design features described below. Class 1E and safe shutdown-related electrical equipment is required to remain functional in the event of actuation of suppression systems. This is accomplished by protecting equipment in NEMA 4 watertight enclosures, sealing openings into the equipment, sealing open-ended conduit which could transmit water into the equipment, or protecting equipment by external spray shields, unless it can be shown that the equipment will otherwise remain operable. In addition, safe shutdown equipment is mounted on raised pads in areas containing water suppression systems. Concrete floors surrounding the pads are sloped to floor drains at low points.

Therefore, inadvertent actuation of the fire protection systems does not result in any deleterious effects to SSEL components with a plant HCLPF capacity of 0.3 g pga.

3.1.4.12.3 Seismic Degradation of Fire Suppression Systems

Fire suppression systems are installed in accordance with good industrial practice and have been reviewed for seismic considerations to ensure suppression system piping and components will not fall and damage safe shutdown components. It is also unlikely that leaking or cascading of the suppressant will result.

During the seismic capability walkdown, the SRTs evaluated the seismic ruggedness of the fire suppression system piping and components in areas containing SSEL components. No concerns with the seismic capacity of the fire suppression system piping were identified. The steel piping is installed with a mixture of welded and threaded connections. The SRTs determined that the threaded connections are adequate since the piping is well supported. Therefore, based on the judgment of the SRTs, the Plant Hatch fire suppression system is considered adequate with respect to Seismic Category II/I considerations with a plant HCLPF capacity of 0.3 g pga.

3.1.5 ANALYSIS OF CONTAINMENT PERFORMANCE

As stated in NUREG-1407 (Reference 42), "The primary purpose of the evaluation for a seismic event is to identify vulnerabilities that involve early failure of containment functions." Therefore, the primary focus of the Plant Hatch containment performance evaluation was containment failure and containment isolation.

3.1.5.1 Containment Failure

Early containment failure was addressed for Plant Hatch, which has a General Electric Mark I BWR containment, by including the containment spray system on the Safe Shutdown Equipment List (SSEL) for both units. Post-accident containment atmosphere temperature and pressure control is provided by injecting water through spray nozzles located in both the drywell and torus area. Additional information concerning use of the containment spray system for the Plant Hatch IPEEE is included in section 3.1.2.5.3.D.

Components of containment spray system included on the SSEL are demonstrated of possess a high-confidence-of-low-probability of-failure (HCLPF) capacity of at least 0.3 g pga, or were modified to obtain this capacity.

3.1.5.2 Containment Isolation

The basic function of containment isolation valves is to provide necessary isolation of the containment in the event of accidents or similar conditions when the release of containment atmosphere cannot be permitted. A list of all Unit 1 containment isolation valves is provided in table 7.3-1 of the Unit 1 FSAR (Reference 11). A list of Unit 2 containment isolation valves is provided in table 6.2-5 of the Unit 2 FSAR (Reference 12). Both tables were reviewed to determine the penetrations containing valves that are normally open, but are required to close. In accordance with NUREG-1407, the screening criteria used in the Plant Hatch IPE (Reference 46) were used for the IPEEE containment performance evaluation. Based on these criteria, the following were screened from consideration:

- Spare penetrations containing no valves.
- Electrical penetrations.
- Penetrations containing only a thermocouple.
- Closed systems that do not come in direct contact with containment atmosphere or reactor coolant.
- Penetrations with normally closed, fail-closed valves, including check valves.

- Penetrations with normally closed valves that are required for accident mitigation and required to open.

- Penetrations 2 in. and smaller. Per the Plant Hatch IPE, containment penetrations less than 2 in. in diameter are considered too small to release a significant amount of fission products to the environment.

The reactor water cleanup system, clean radwaste system, dirty radwaste system, and the containment purge and inerting system isolation valves remained after the screening process. These valves are not required to safely shut down the plant but are needed for containment isolation and are included in the SSEL.

Containment isolation valves included on the SSEL are either demonstrated to possess or were modified to obtain a HCLPF capacity of at least 0.3 g pga.

REFERENCES

1. Electric Power Research Institute, "Seismic Margin Assessment of Edwin I. Hatch Nuclear Plant, Unit 1," EPRI NP-7217-SL, Palo Alto, California, June 1991.

2. American Concrete Institute (ACI), 318-63, Building Code Requirements for Reinforced Concrete, including 1963 Supplement, Detroit, Michigan.

3. American Institute of Steel Construction (AISC), Manual of Steel Construction, 6th Edition, Chicago, Illinois, 1963.

4. Institute of Electrical and Electronics Engineers, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Standard 344-1971.

5. Electric Power Research Institute, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" (Revision 1), EPRI NP-6041-SL, Palo Alto, California, August 1991.

6. U. S. Nuclear Regulatory Commission, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, Final Report," Revision 2, NUREG-0313, January 1988.

7. Georgia Power Company, Unresolved Safety Issue A-46 Summary Report, Edwin I. Hatch Nuclear Plant Unit 1, May 31, 1995.

8. Seismic Qualification Utility Group, "Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment," Revision 2, February 14, 1992.

9. Electric Power Research Institute, "Seismic Verification of Nuclear Plant Equipment Anchorage," Revision 1, Volume 1: Development of Anchorage Guidelines, and Volume 3: EPRI/Blume Anchorage Computer Program (EBAC), EPRI NP-5228-SL, Palo Alto, California, June 1991.

10. General Electric Company, "Minimum Systems Required for Safe Shutdown During a Fire in Edwin I. Hatch Nuclear Power Station Units 1 and 2," NEDO-24372, October 1981.

11. Edwin I. Hatch Nuclear Plant Unit 1 Final Safety Analysis Report.

12. Edwin I. Hatch Nuclear Plant Unit 2 Final Safety Analysis Report.

13. Johnson, J. J., Maslenikov, O. R., and Doyle, D. J., "Review of Seismic Analysis of Hatch Unit 1 and 2: In-structure Response Spectra," UCRL-21015, prepared by EQE, Inc., for Lawrence Livermore National Laboratory for the U. S, Nuclear Regulatory Commission, November 1987.

14. Letter from W. G. Hairston, III (GPC) to the NRC with enclosed report, "Seismic Margin Assessment of Edwin I. Hatch Nuclear Plant Unit 1," July 20, 1990.

15. Letter from Dan Guzy, (NRC, Seismic Design Margins Working Group), "Resolution and Closure of All Soils Issues in the Hatch Review," April 29, 1990.

16. Letter from Dr. Michael P. Bohn (Sandia National Laboratories) to Dr. Nilesh Chokshi (NRC) with enclosed report, "Independent Evaluation of the Hatch Seismic Margin Assessment Seismic Building Response and Floor Spectra," July 5, 1991.

17. Memorandum from Dan Guzy, (NRC, Seismic Design Margins Working Group), "Final Evaluation of the Hatch Seismic Margins Review," May 2, 1990.

18. Letter from D. R. Davis, (Hatch Seismic Margin Assessment Peer Review Group) to Dan Guzy (NRC, Seismic Design Margins Working Group), with enclosed report "Hatch SMA Peer Review Group Final Report: Evaluation of the Application of the NRC and EPRI Seismic Margins Methodologies," May 3, 1990.

19. Letter from Kahtan N. Jabbour (NRC) to W. G. Hairston, III (GPC), "Evaluation of Plant Hatch Units 1 and 2, 120-Day Response to Supplement No. 1 to Generic Letter 87-02," November 20, 1992.

20. Newmark, N. M. and Hall, W. J., "Development of Criteria for Seismic Review of Selected Nuclear Power Plants, NUREG/CR-0098, prepared for U. S. Nuclear Regulatory Commission, May 1978.

21. Seed, H. B., and Idriss, I. M., "Soil Moduli and Damping Factors for Dynamic Response Analysis," EERC 70-10, Earthquake Engineering Research Center, University of California, Berkeley, 1970.

22. Schrabel, P. B., Lysmer, H. J., and Seed, H. B., SHAKE, EERC 72-12, Earthquake Engineering Research Center, University of California, Berkeley, December 1972.

23. Wong, H. L. and Luco, J. E., "Soil-Structure Interaction: A Linear Continuum Mechanics Approach/(CLASSI)," CE 79-03, University of Southern California, Los Angeles, 1980.

24. American Society of Civil Engineers, "Seismic Analysis of Safety Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety Related Nuclear Structures," ASCE 4-86, New York, approved September 1986.

25. "Uncertainty and Conservatism in the Seismic Analysis and Design of Nuclear Facilities," prepared by the Working Group on Quantification of Uncertainties of the Committee on Dynamic Analysis of the Committee on Nuclear Structures and Materials of the Structural Division of the American Society of Civil Engineers, New York, 1986.

26. U. S. Nuclear Regulatory Commission, "Masonry Wall Design," IEB 80-11, May 8, 1980.

27. American Concrete Institute, "Building Code Requirements for Concrete Masonry Structures," ACI 531-79, 1983.

28. R. P. Kennedy, et al., "Assessment of Seismic Margin Calculation Methods," NUREG/CR-5270, prepared for U. S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory, March 1989.

29. Uniform Building Code, International Conference of Building Officials, 1985.

30. Letter from W. G. Hairston, III (GPC) to the NRC with enclosed report, "E. I. Hatch Nuclear Plant Unit 1, Suppression Chamber Support Anchor Bolt Study," March 23, 1989.

31. Georgia Power Company, "Plant-Unique Analysis Report for Edwin I. Hatch Nuclear Plant Unit 1 Mark I Containment Long-Term Program," Revision 3, December 1989.

32. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, 1977 edition with addenda through Summer 1979.

33. Sykora, D. W., and Stokoe II, K. H., "Correlation of In-Situ Measurement In Sands With Shear Wave Velocity," Geotechnical Engineering Report GR 83-33, University of Texas, Austin, 1983.

34. Seed, H. B. and Idriss, I. M., "Simplified Procedure for Evaluating Soil Liquefaction Potential," Journal of the Soil Mechanics and Foundation Division, ASCE, v. 97, no. SMA, pp. 1249-1273, 1971.

35. Newmark, N. M., "Effects of Earthquake on Dams and Embankments," 5th Rankine Lecture, Geotechnique, v. 15, no. 2, 1965.

36. Goodman, R. E., and Seed, H. B., "Earthquake Induced Displacements in Sand Embankments," Journal of the Soil Mechanics and Foundation Engineering Division, ASCE, v. 92, no. SM2, pp 125-146, 1966.

37. Makdisi, F. I., and Seed, H. B., "Simplified Procedure for Estimating Dam and Embankment Earthquake-Induced Deformations," Journal of the Geotechnical Engineering Division, ASCE, v. 104, No. GTF, pp. 849-867, 1978.

38. U. S. Nuclear Regulatory Commission, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," Generic Letter No. 88-20, Supplement 4, Washington, D.C., June 28, 1991.

39. American Society of Civil Engineers, "Seismic Response of Buried Pipes and Structural Components: A Report by the Seismic Analysis of the ASCE Nuclear Structures and Materials Committee," 1983.

40. American Society of Civil Engineers, "Guidelines for Seismic Design of Oil and Gas Line Systems," The Committee on Gas and Liquid Fuel Lifelines of the Technical Council on Lifeline Earthquake Engineering, 1984.

41. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, NC-3653.1, Section III, Division I-subsection NC, 1989.

42. U. S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events for Severe Accident Vulnerabilities, Final Report," NUREG-1407, Washington, D.C., June 1991.

43. Electric Power Research Institute, "Procedure for Evaluating Nuclear Power Plant Relay Seismic Functionality," EPRI NP-7148-SL, December 1990.

44. Georgia Power Company, "Unresolved Safety Issue A-46, Summary Report, Edwin I. Hatch Nuclear Plant Unit 2," May 31, 1995.

45. Edwin I. Hatch Nuclear Plant Units 1 and 2 Fire Hazards Analysis and Fire Protection Program.

46. Edwin I. Hatch Nuclear Plant Units 1 and 2 Individual Plant Examination, December 1992.

Table 3.1-1 Support and Front-Line System Dependency Matrix
Units 1 and 2

	Front-Line System or Function									
Support System	**RPS**	**CRD**	**HPCI**	**LPCI**	**Low-Pressure CS**	**SDC**	**ASDC**	**Pressure Control**	**Suppression Pool Cooling**	**PCIVs**
RPS	-	X	-	-	-	-	-	-	-	X
ac Power	X	-	O	-	-	-	-	-	X[1]	X
dc Power	X	X	X	X	X	X	X	X	X	X
Drywell Pneumatic	-	-	-	-	-	-	-	X	-	X
PSW		-	O	O	O	O	O	-	O[2]	-
RHRSW	-		-	-		X	X	-	X	-
Equipment Area Cooling	-	-	X	X	X	X	X	-	X	-

Legend:

X Direct dependency.
O Dependency exists, but not direct.
- No dependency.

Notes:

1. The HPCI system operates independently of ac power. The indirect dependency for ac power exists for power to the room coolers, which are not required for operation but which are desirable for extended operation.

2. The indirect dependency for PSW exists for PSW to the room coolers in the ECCS pump rooms. These are not required for system operation, but are required for maintaining temperatures in specific areas within design-allowable operating ranges for components within those areas during extended operation.

Table 3.1-2

Support System to Support System Dependency Matrix
Units 1 and 2

Required Support System	RPS	ac Power	dc Power	Drywell Pneumat.	PSW	RHRSW	Equipment Area Cooling
RPS	-	-	-	-	-	-	-
ac Power	X	-	X	X	X	X	X
dc Power	-	X	-	X	X	X	-
Drywell Pneumat.	-	-	-	-	-	-	-
PSW	-	-	-	-		-	X
RHRSW	-	-	-	-	-	-	-
Equipment Area Cooling	-	-	-	-	-	-	-

Legend:

X Dependency exists.
\- No dependency.

Table 3.1-3

Comparison of Plant Hatch Units 1 and 2 Reactor Pressure Vessel and Internals

Mode	Unit 1 Mode[1] (Hz)	Unit 1 Mode Description	Unit 2 Mode[2] (Hz)	Unit 2 Mode Description
1	4.57	Fuel, Steam Separator, and Standpipes	4.46	Fuel, Steam Separator, and Standpipes
2	5.94	Steam Separator, Fuel, and Standpipes	5.95	Steam Separator, Fuel, and Standpipes
3	10.88	Vessel, Guide Tubes, and Shroud	11.51	Shroud and Vessel
4	14.62	Guide Tubes, Short Housings	14.60	Guide Tubes
5	16.53	Steam Separator, Short Housings, and Shroud	15.51	Guide Tubes and Steam Separator
6	17.92	CRD Housings	20.92	Fuel

Notes:

1. GE Company Drawing No. 761E917, Rev. 2, "Hatch Unit 1 Mathematical Model."

2. GE Company Drawing No. 769E500 / Domestic Drawing No. S-32607, "Hatch Unit 2 Mathematical Model , February 1978."

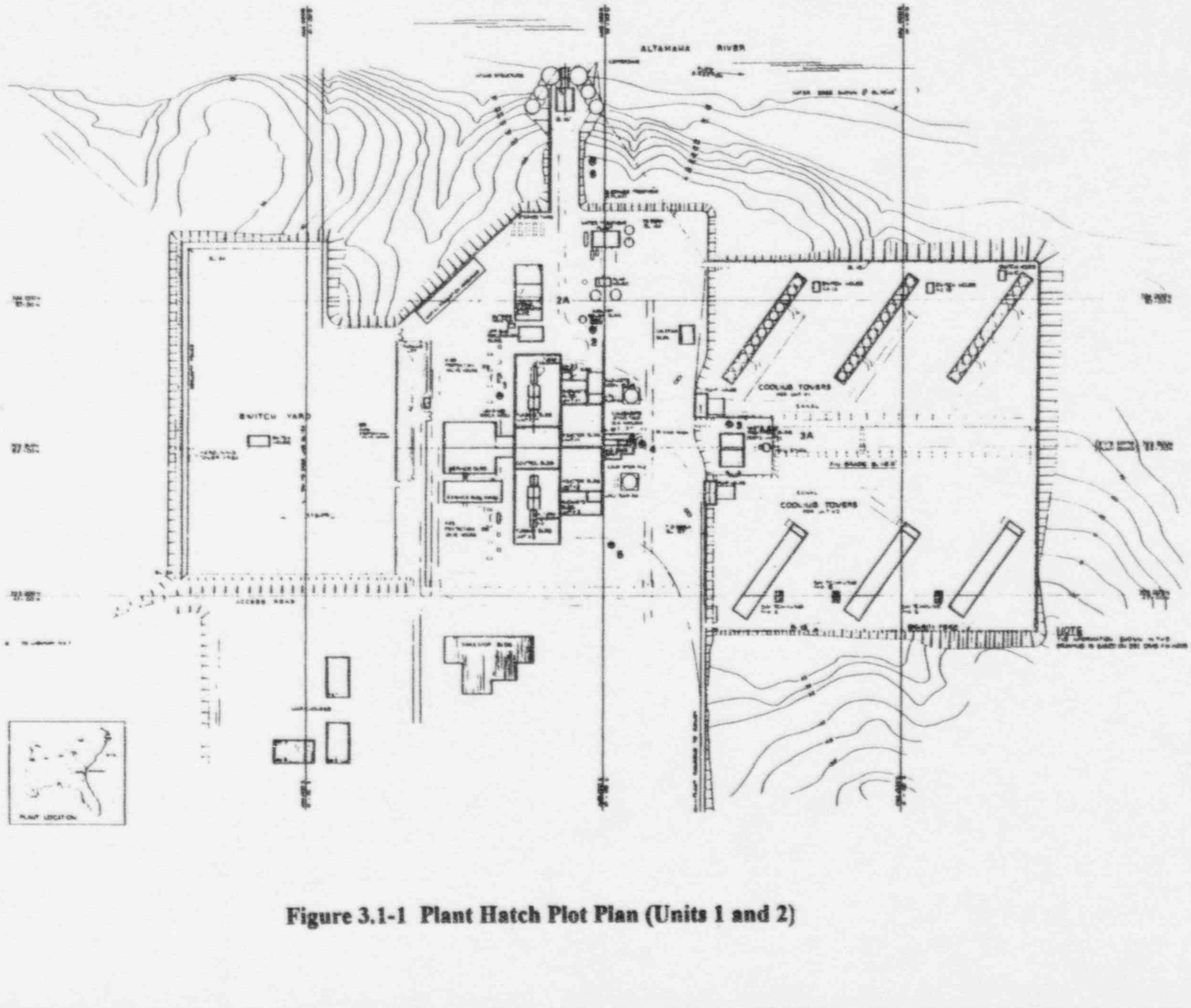


Figure 3.1-1 Plant Hatch Plot Plan (Units 1 and 2)

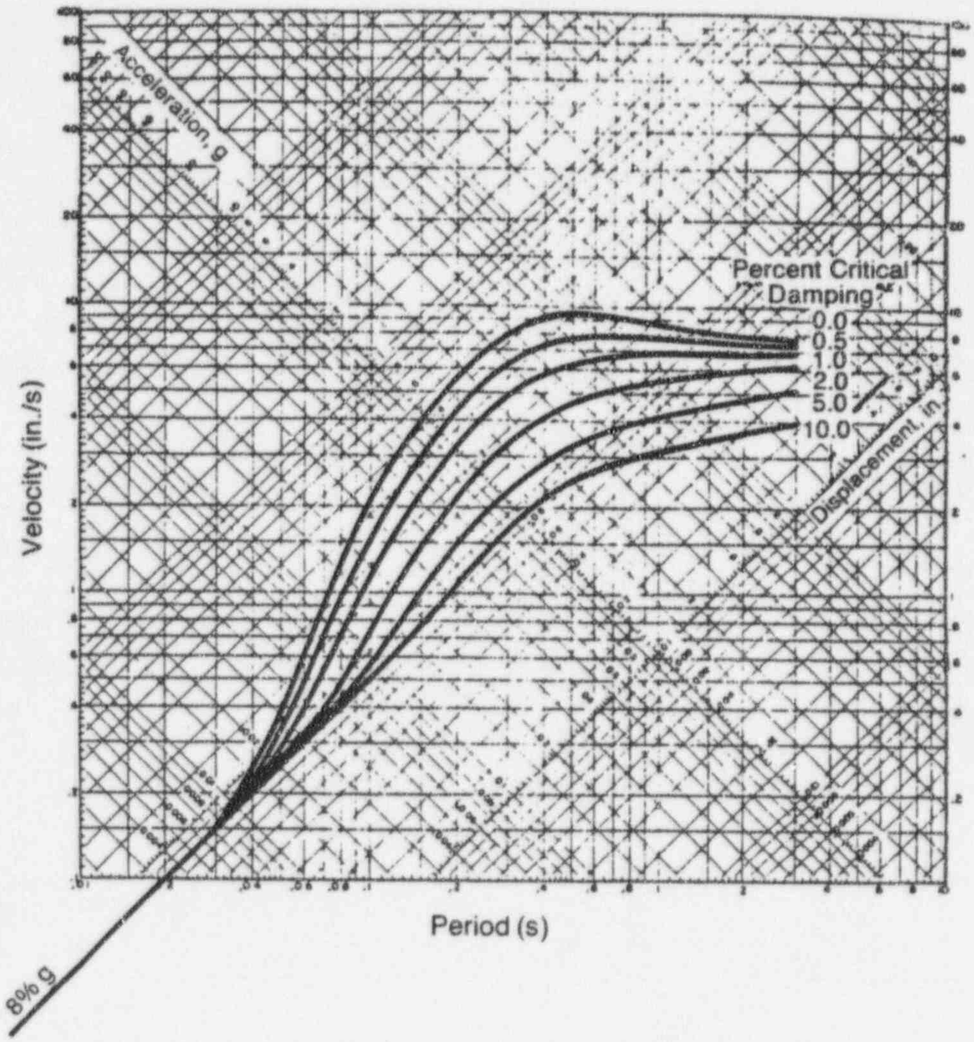


Figure 3.1-2 Plant Hatch Unit 1 Operating Basis Earthquake Design Ground Response Spectra

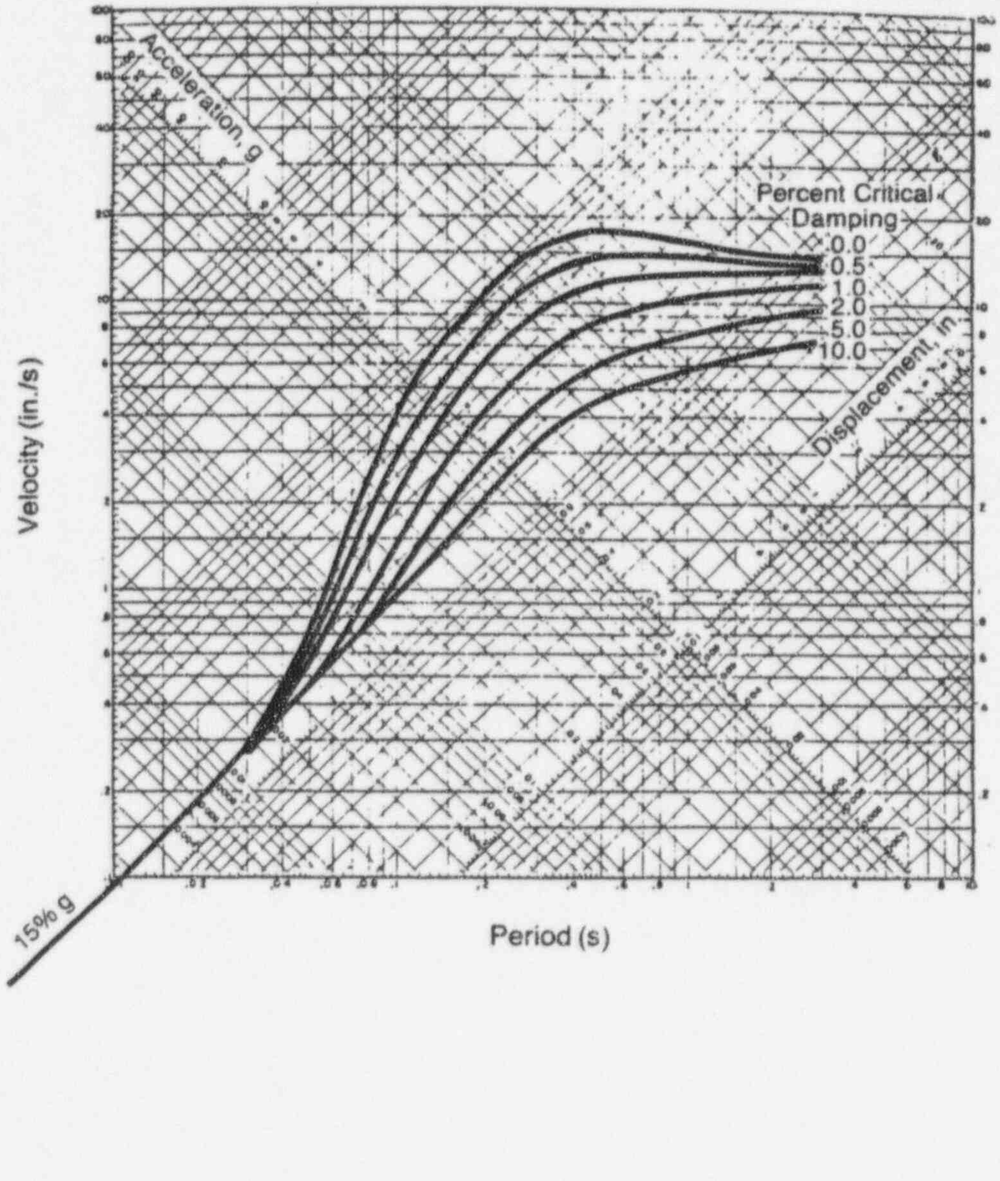


Figure 3.1-3 Plant Hatch Unit 1 Design Basis Earthquake Design Ground Response Spectra

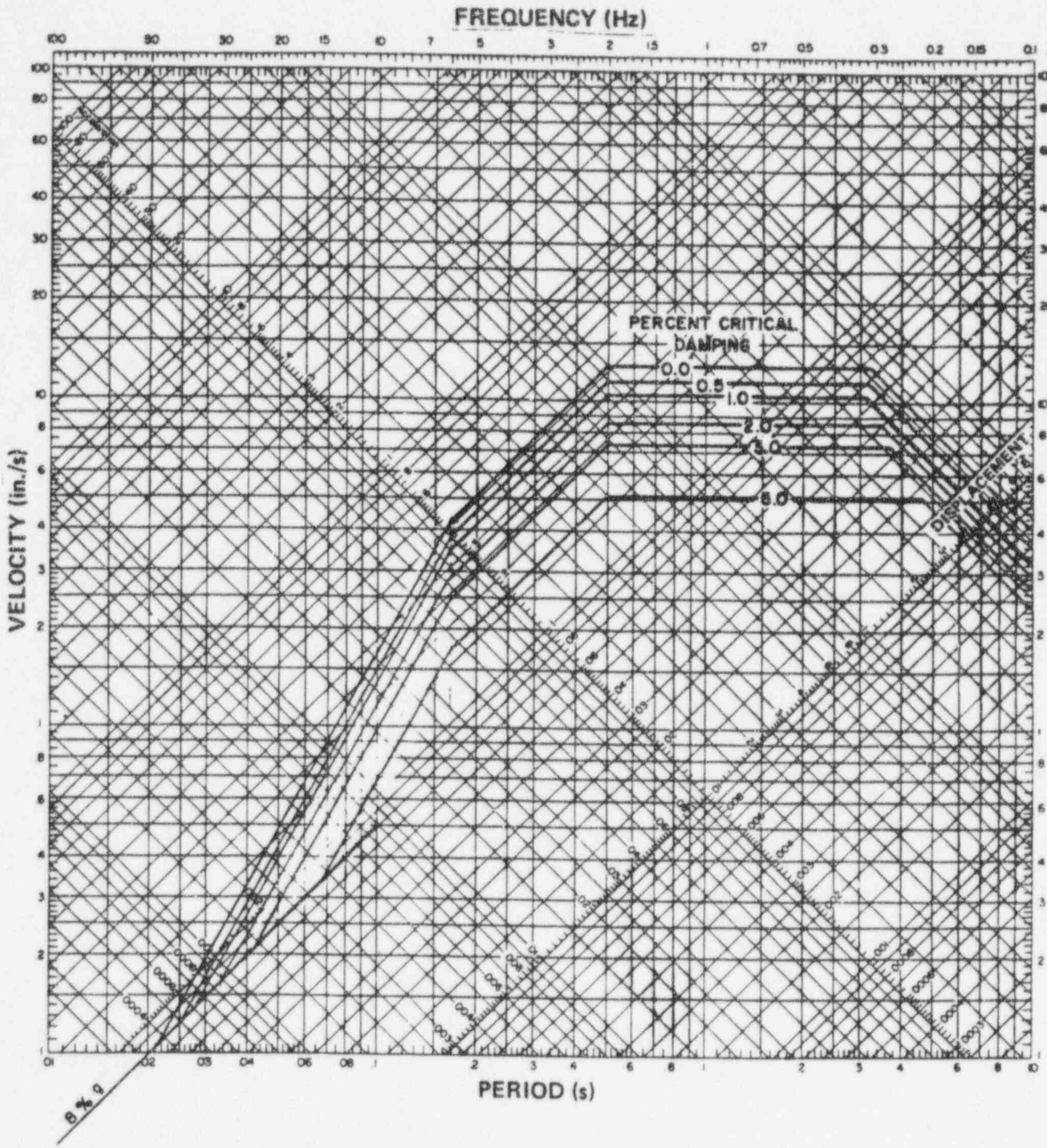


Figure 3.1-4 Plant Hatch Unit 2 Operating Basis Earthquake Design Ground Response Spectra

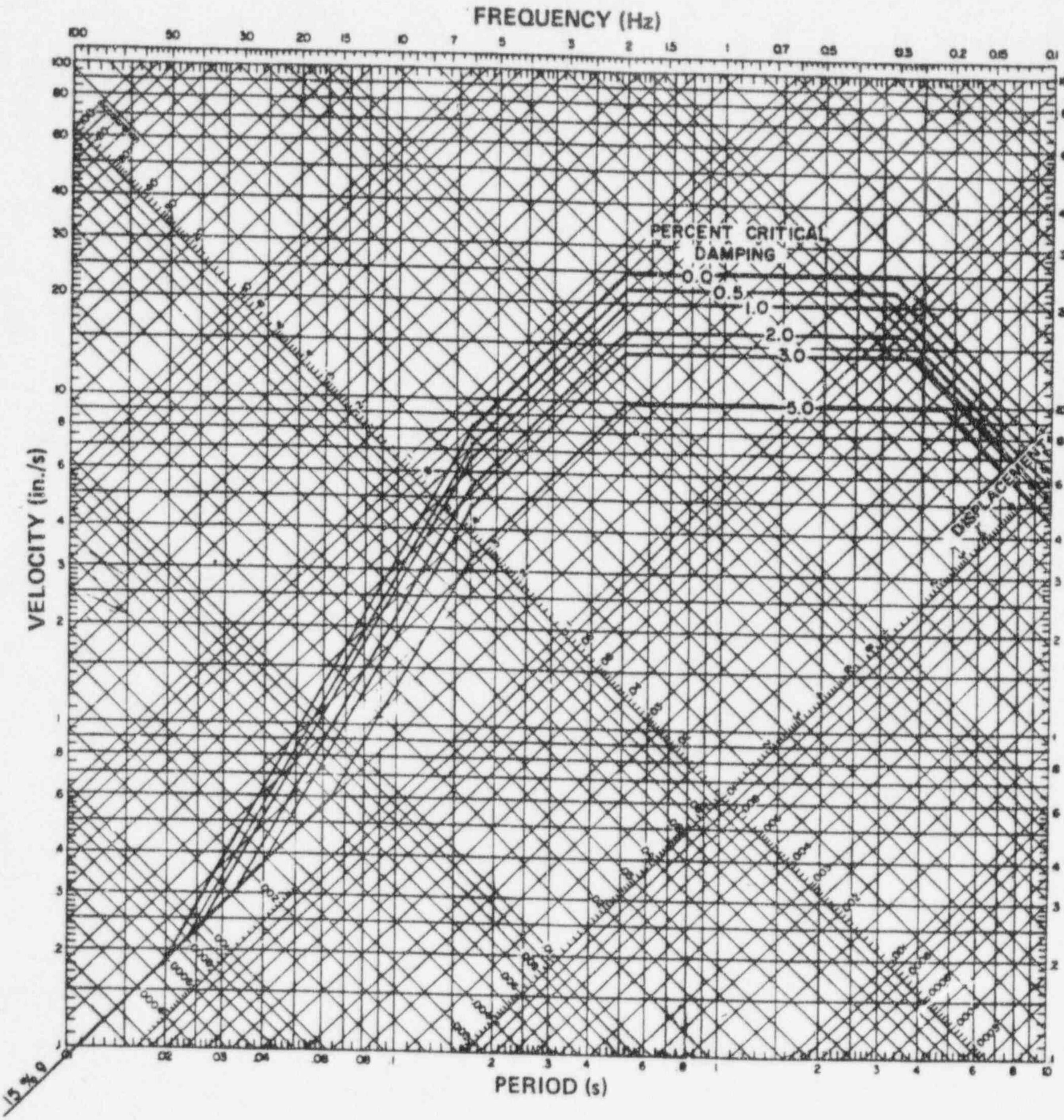


Figure 3.1-5 Plant Hatch Unit 2 Design Basis Earthquake Design Ground Response Spectra

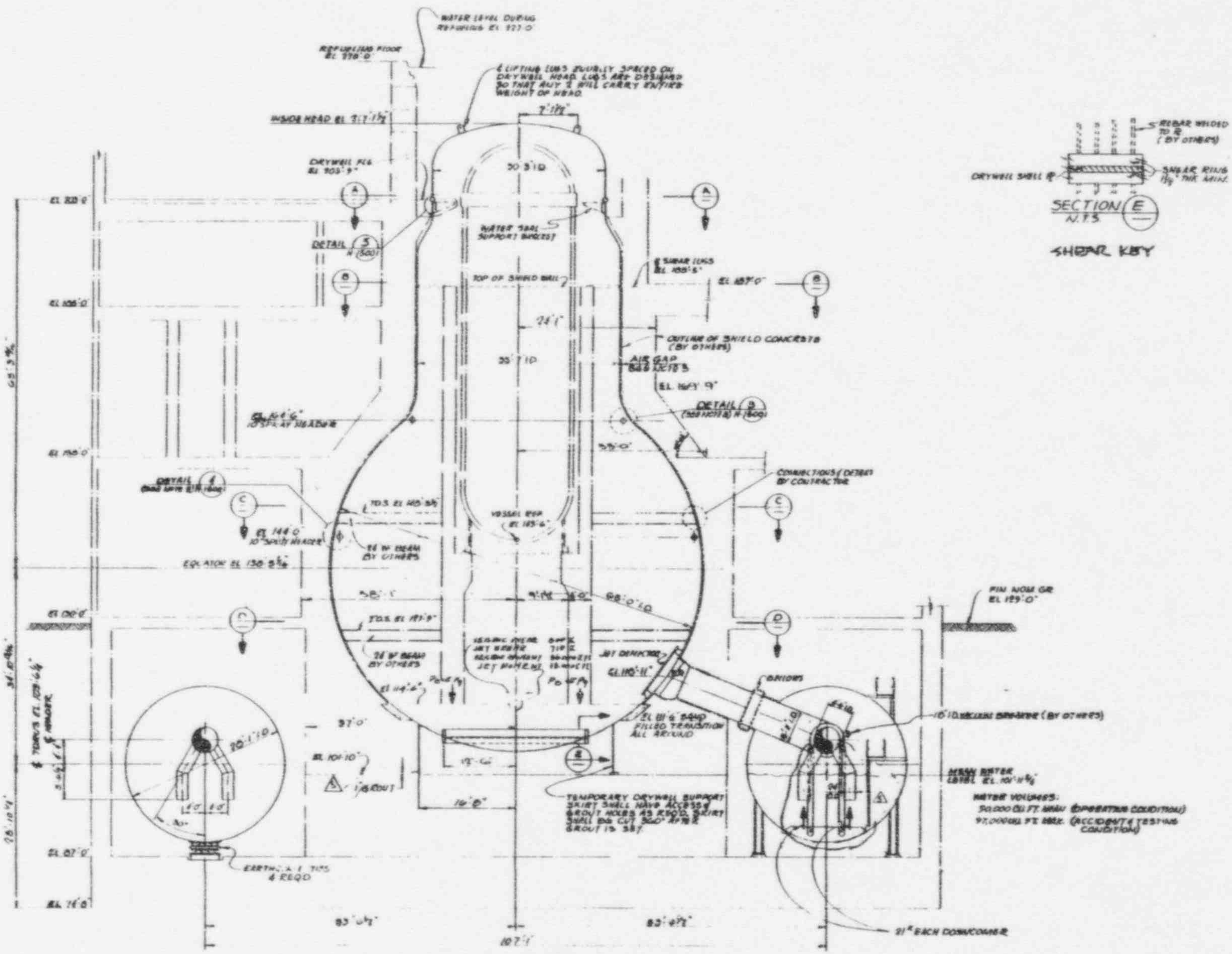


Figure 3.1-6 Plant Hatch Primary Containment

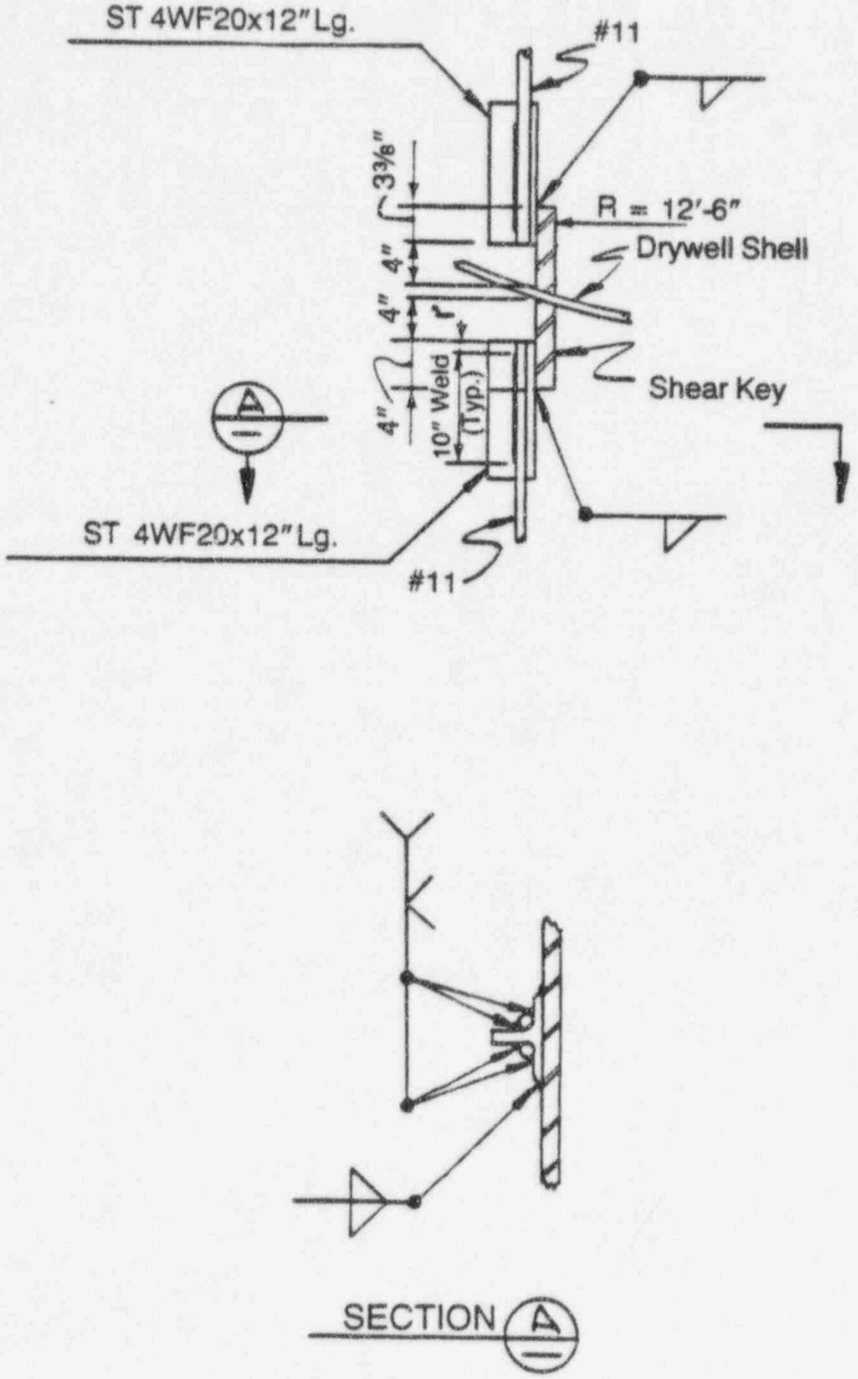


Figure 3.1-7 Shear Key Detail

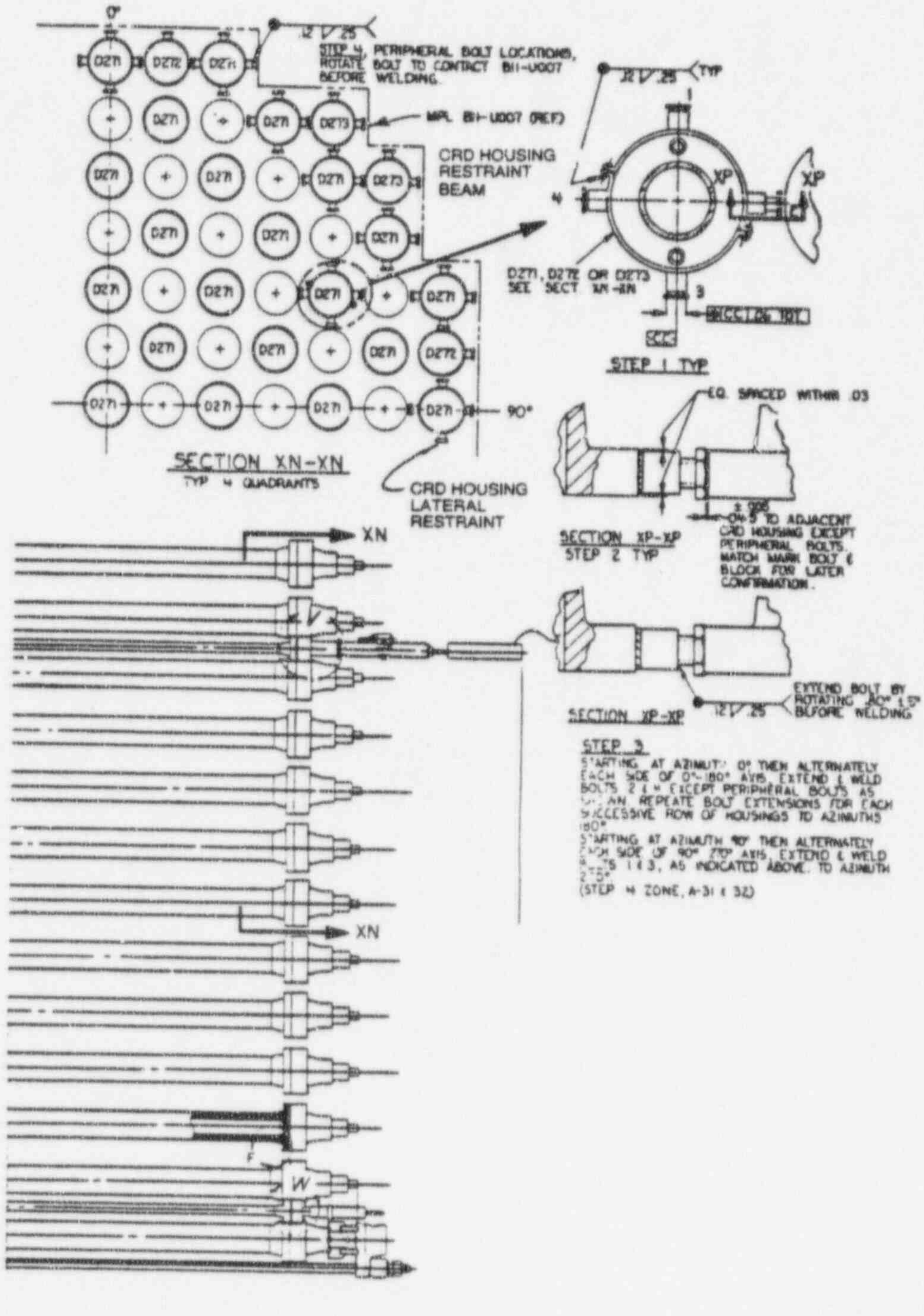


Figure 3.1-8 CRD Housing Lateral Restraints

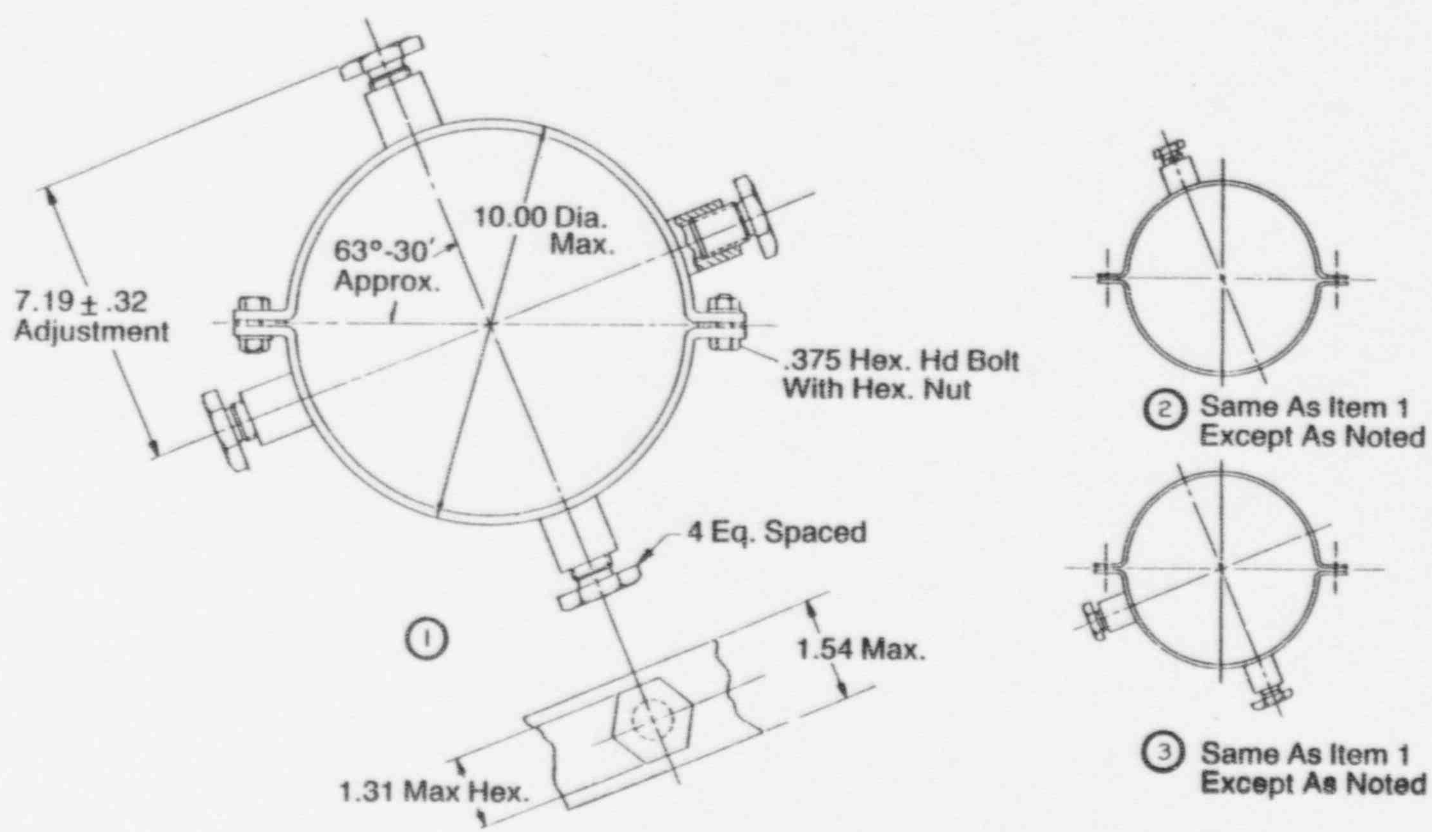
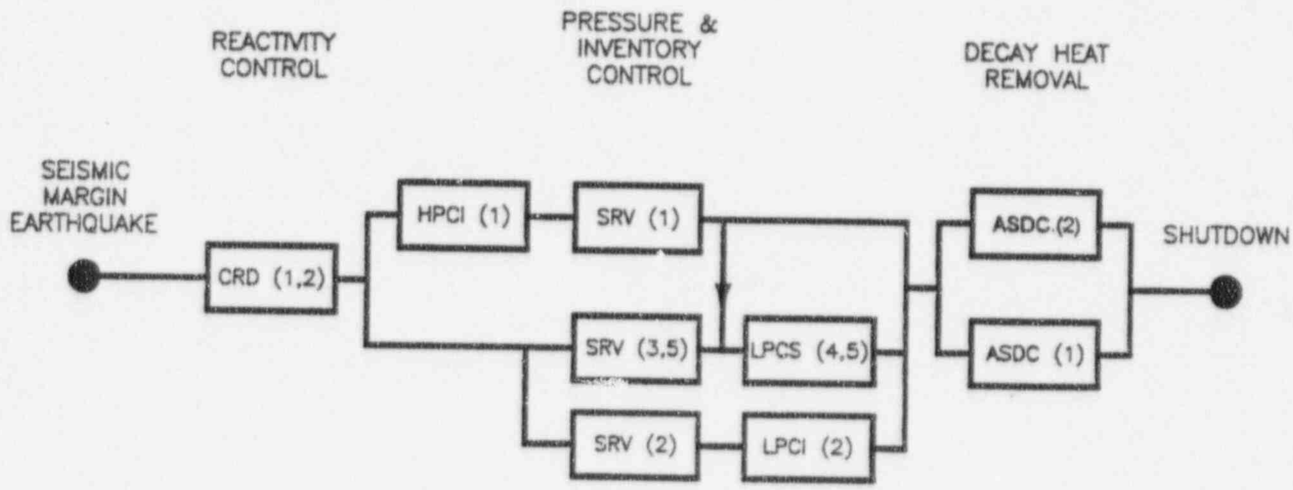


Figure 3.1-9 CRD Housing Lateral Restraints Detail



NOTES:
1. PRIMARY PATH
2. ALTERNATE PATH
3. SAME AS USED IN PRIMARY PATH
4. AVAILABLE FOR PRIMARY PATH BUT NOT REQUIRED.
5. LOW PRESSURE PATH AVAILABLE USING PRIMARY PATH COMPONENTS.

Figure 3.1-10 Success Path Logic Diagram

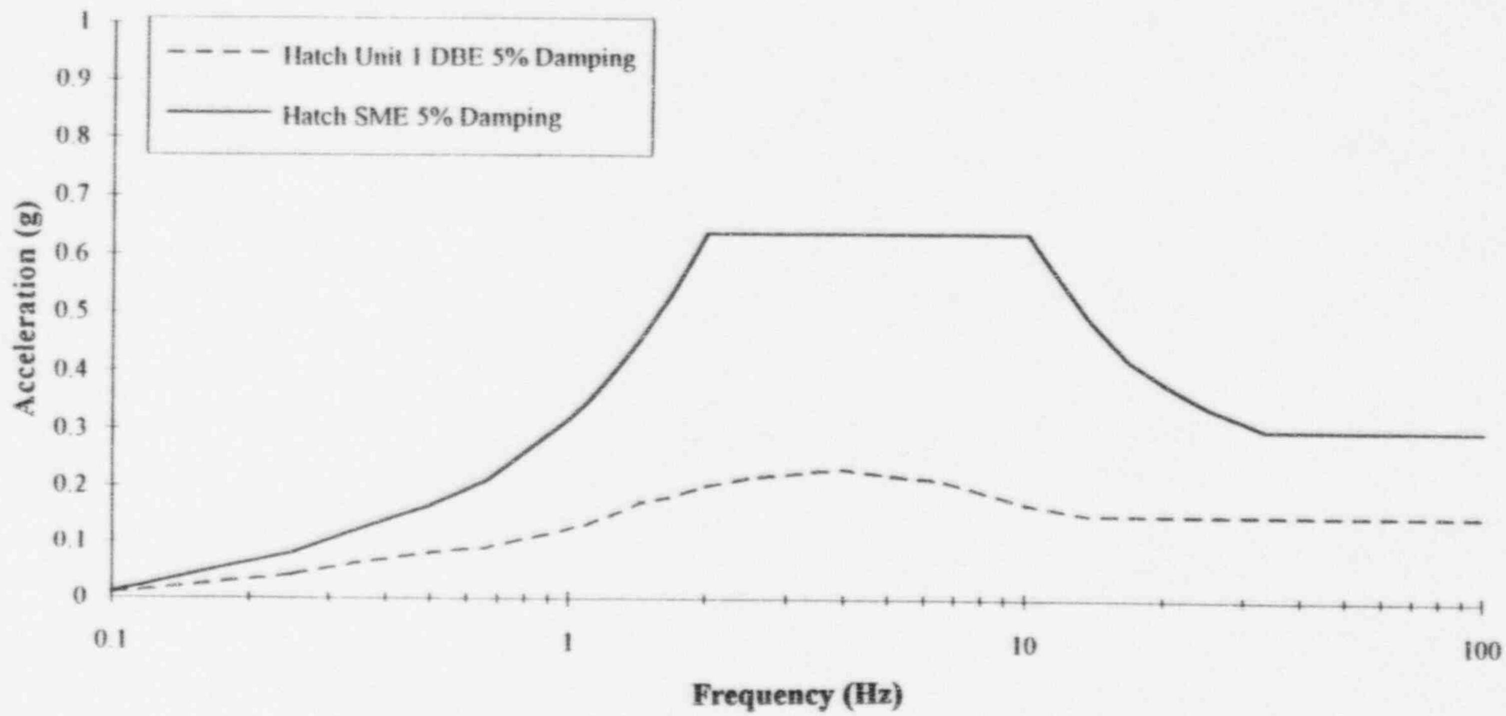


Figure 3.1-11 Plant Hatch Unit 1 Design Basis Earthquake Ground Response Spectrum Versus Plant Hatch Seismic Margin Evaluation Ground Response Spectrum

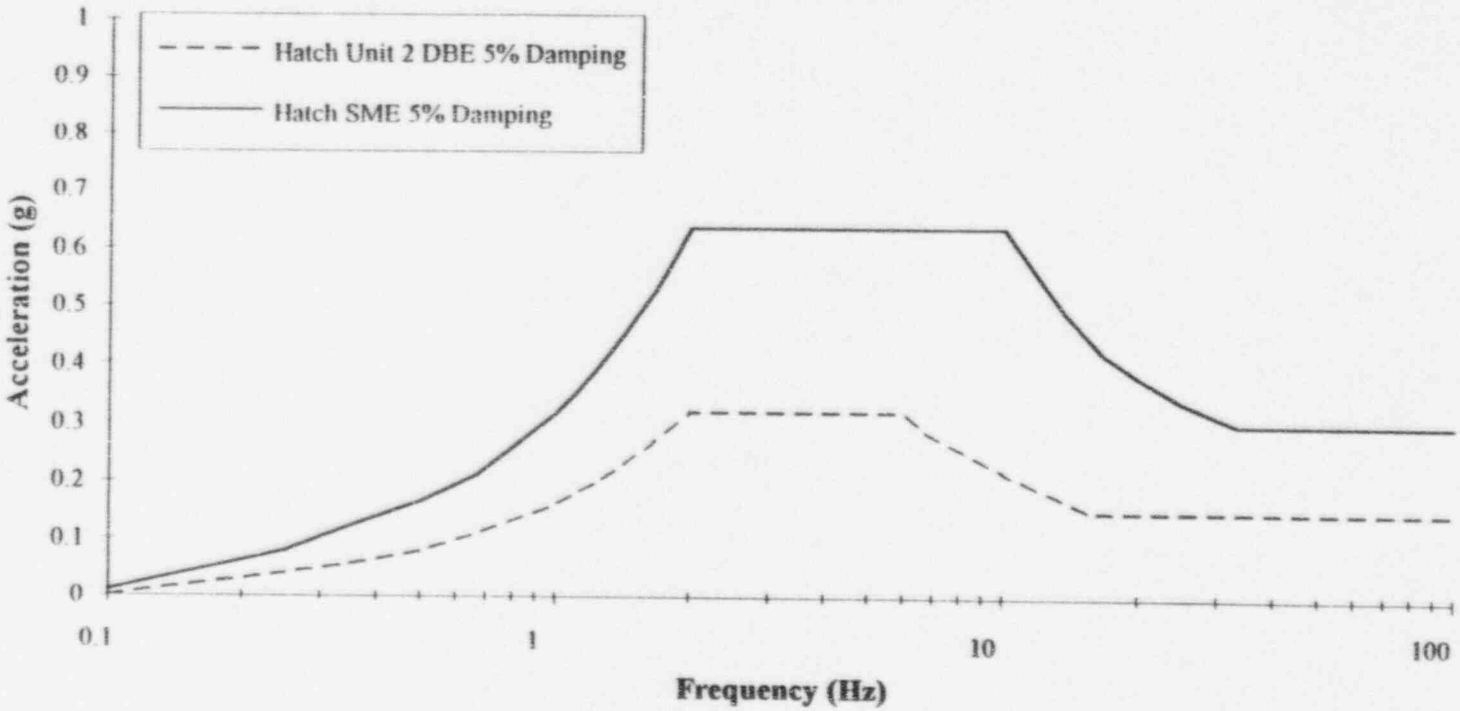


Figure 3.1-12 Plant Hatch Unit 2 Design Basis Earthquake Ground Response Spectrum Versus Plant Hatch Seismic Margin Evaluation Ground Response Spectrum

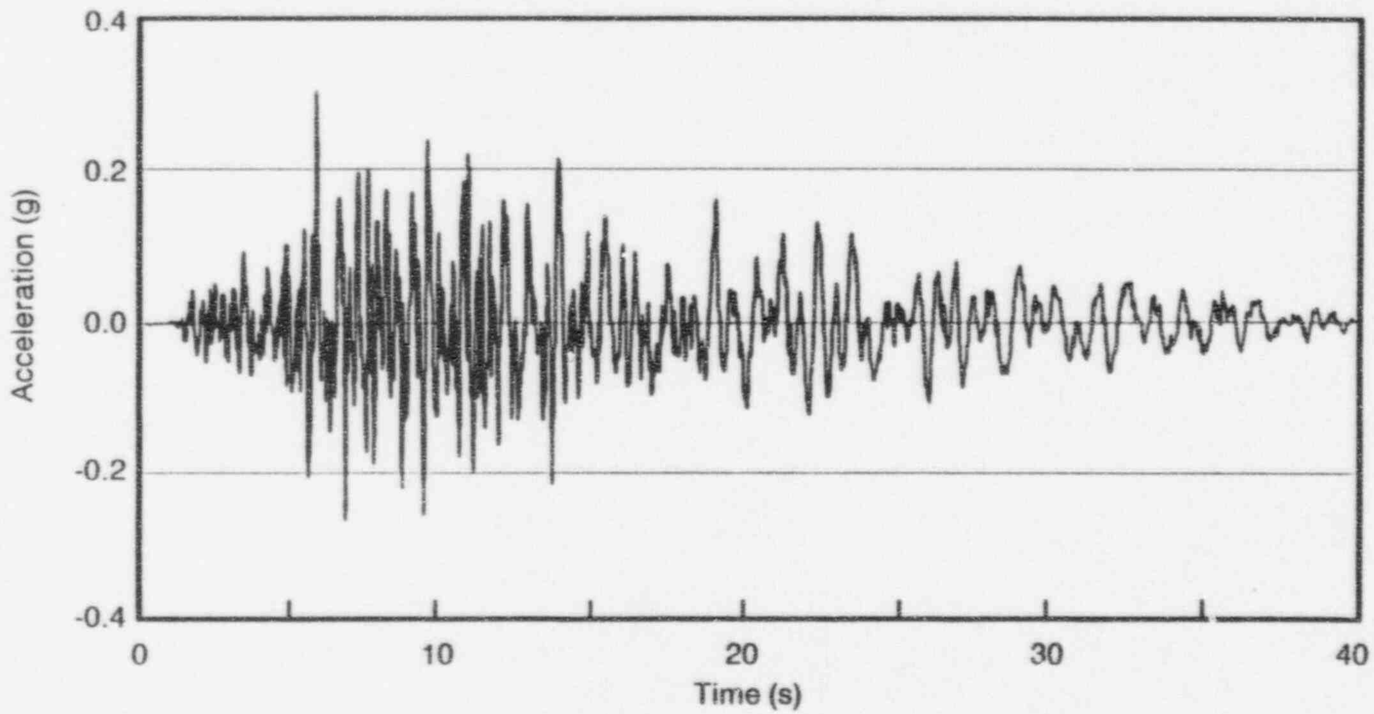


Figure 3.1-13 Accelerogram of Synthetic Time History

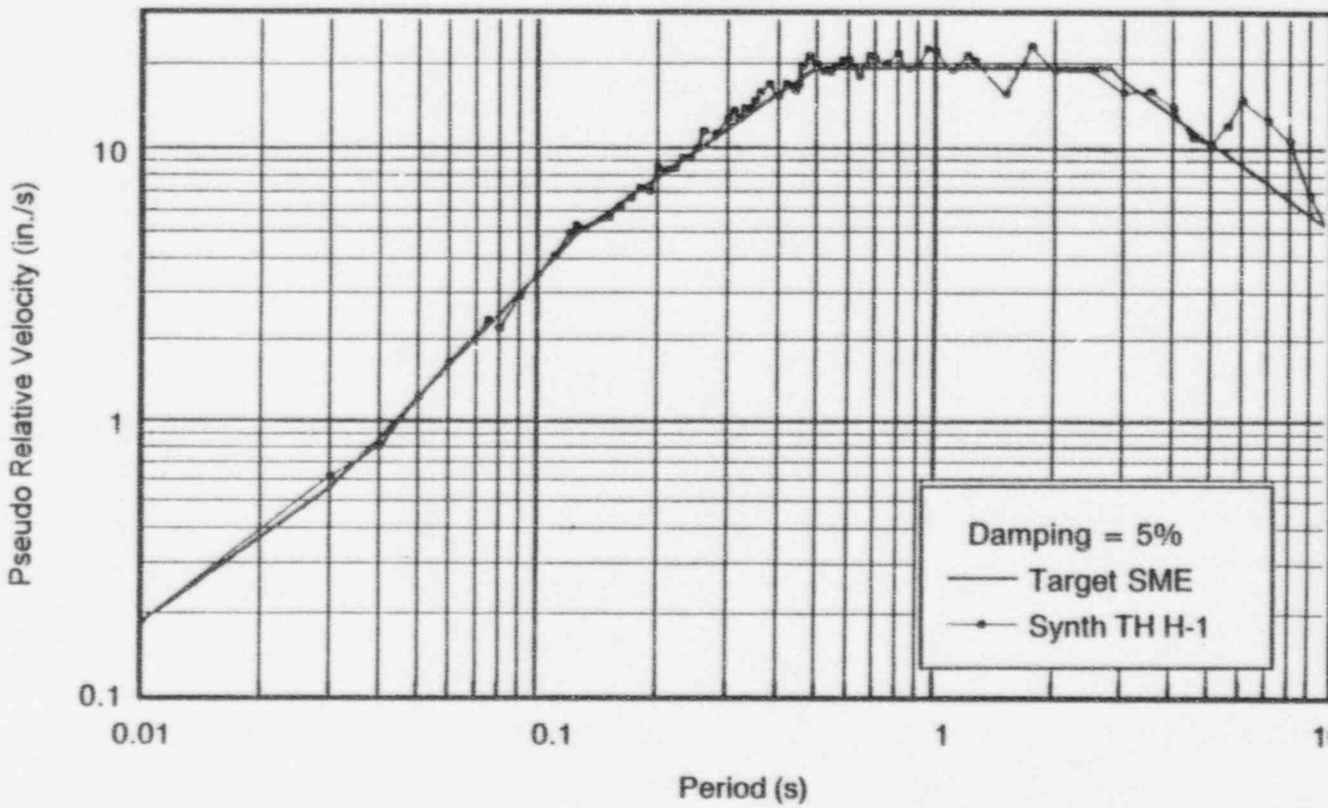


Figure 3.1-14 Plant Hatch Unit 1 SMA Comparison of Target Smooth Spectrum and Spectrum for Synthetic Time History

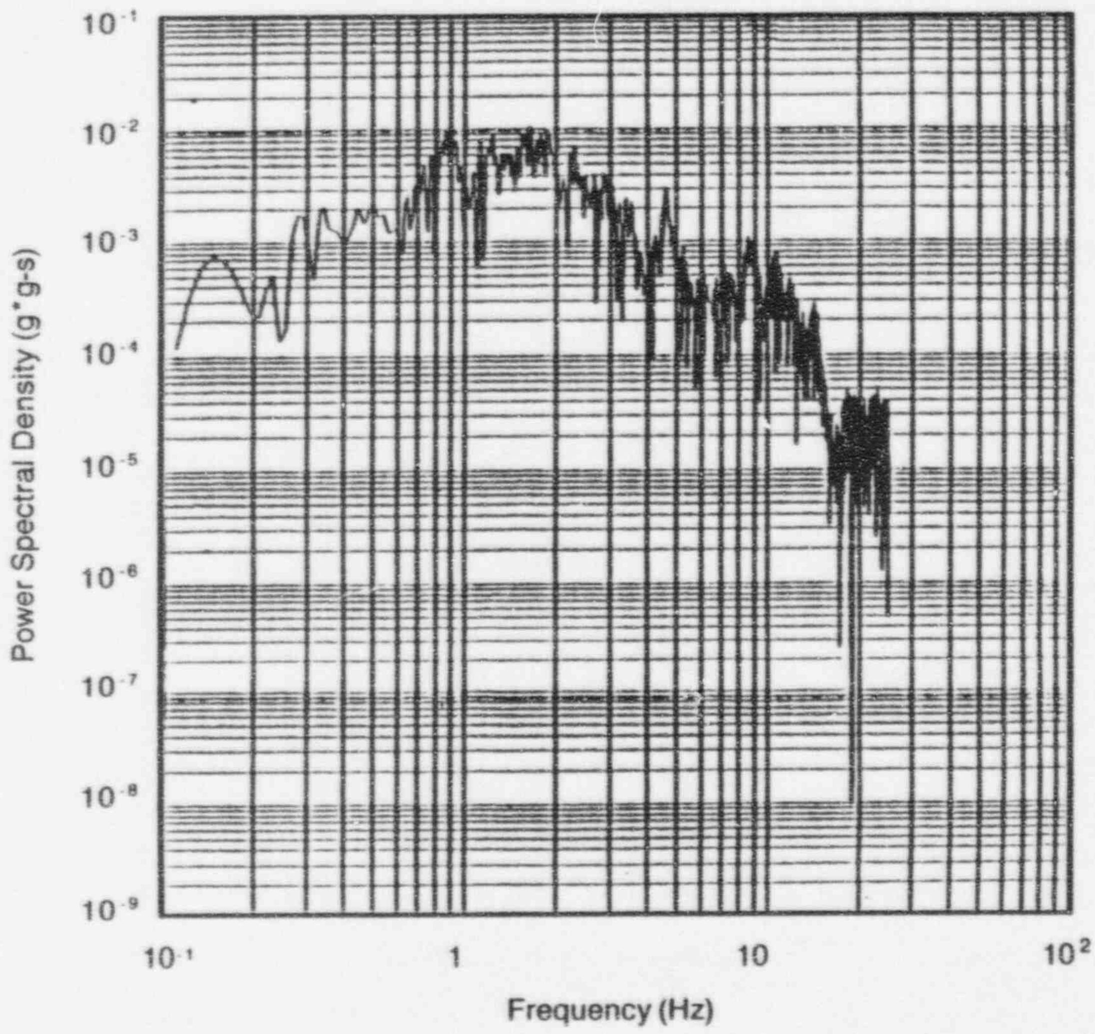


Figure 3.1-15 Power Spectral Density Synthetic Time History

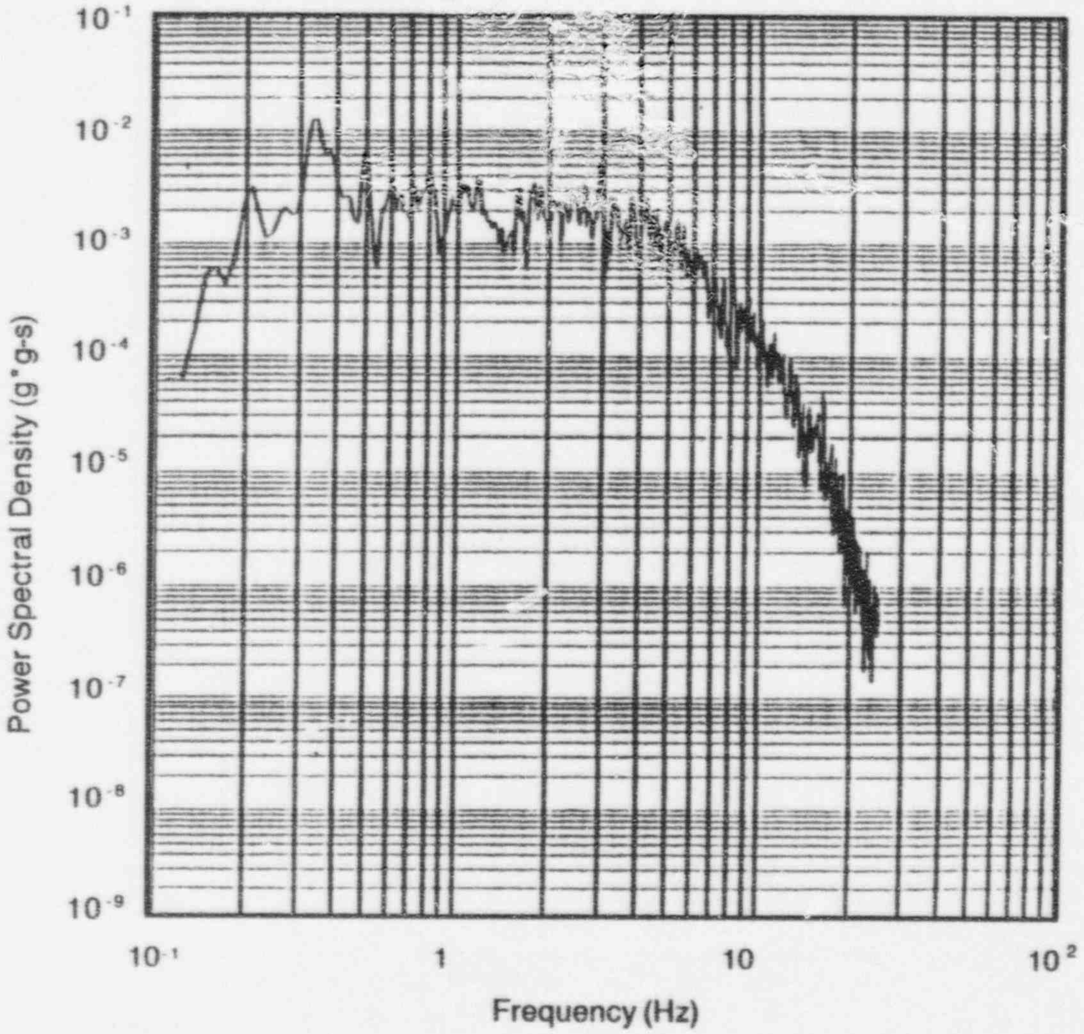


Figure 3.1-16 PSD Average of 14 Records, 1971 San Fernando Earthquake

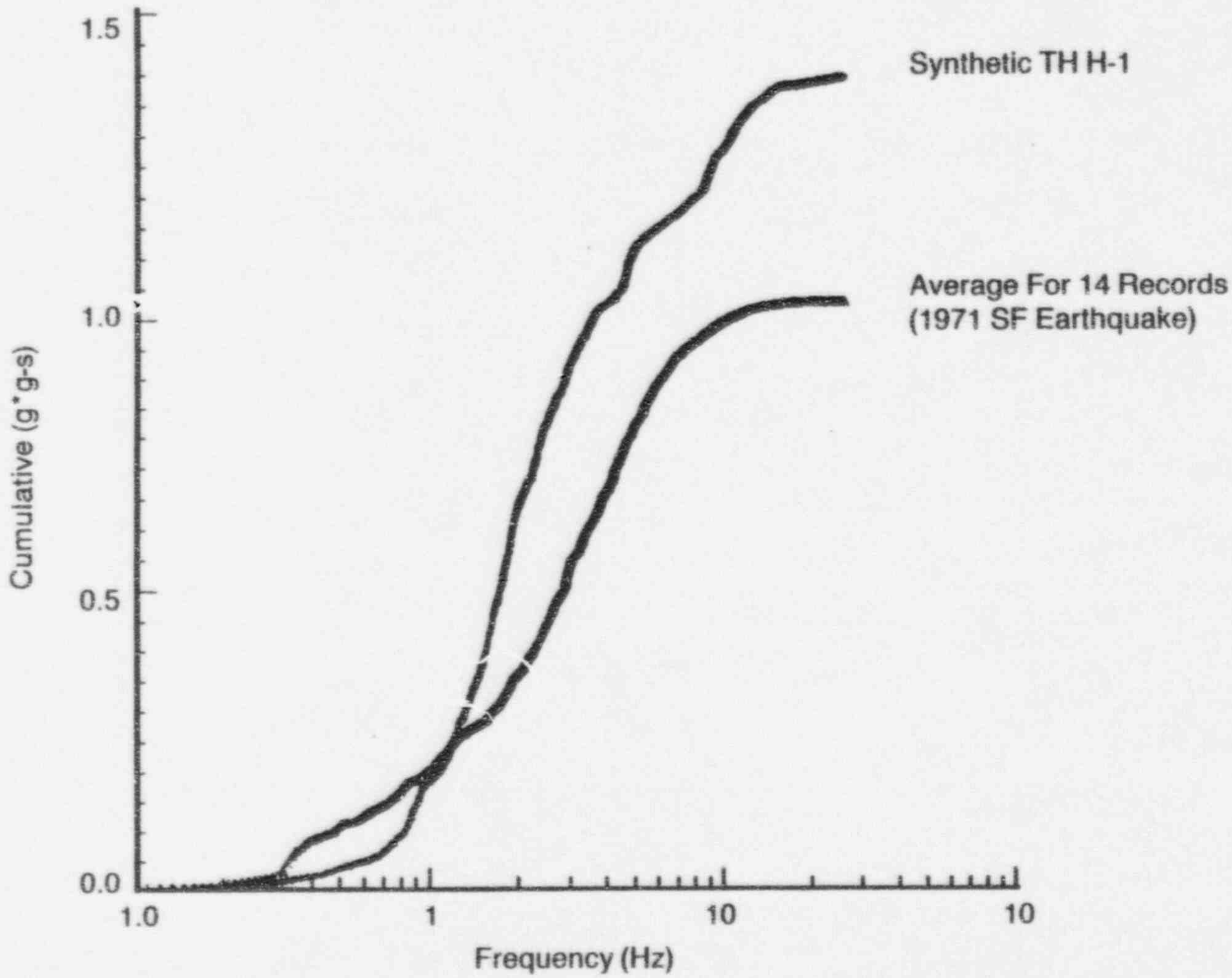


Figure 3.1-17 Cumulative PSD for Synthetic Time History and Average of 14 Records

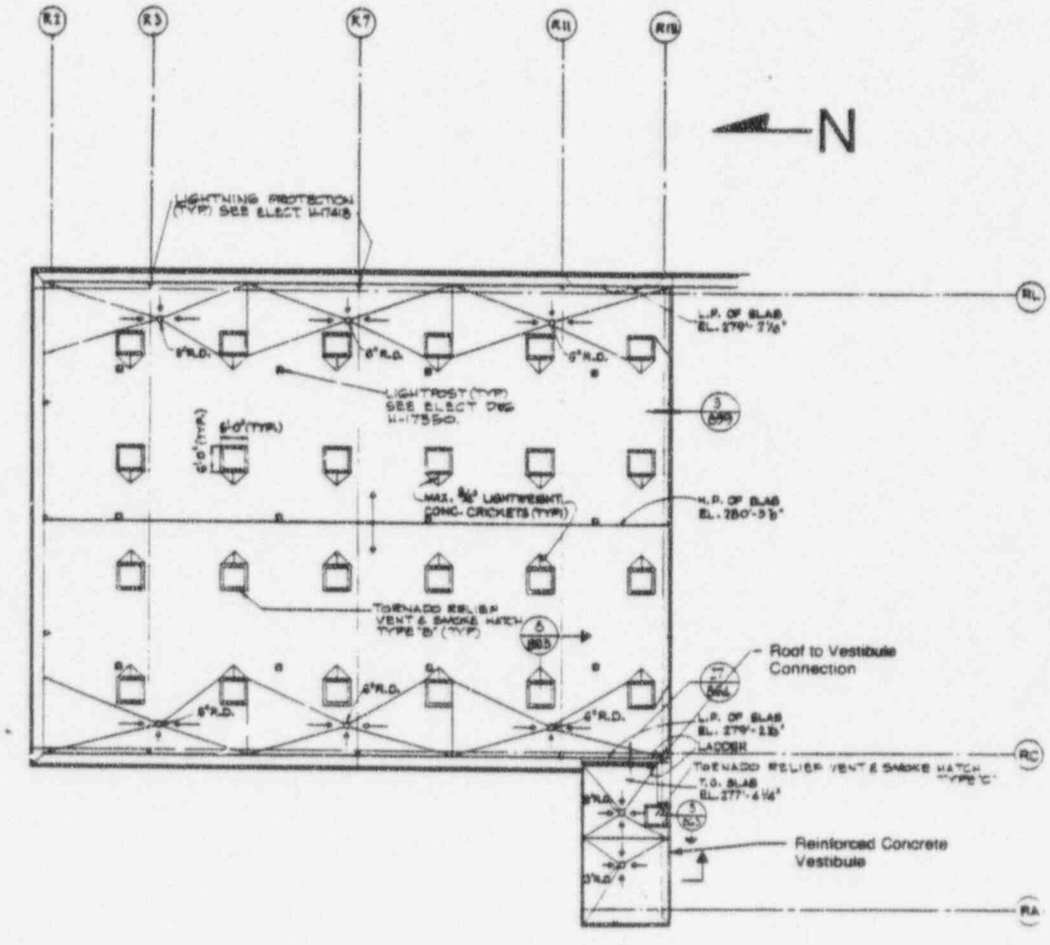


Figure 3.1-18 Plan of Plant Hatch Unit 1 Reactor Building Roof and Vestibule

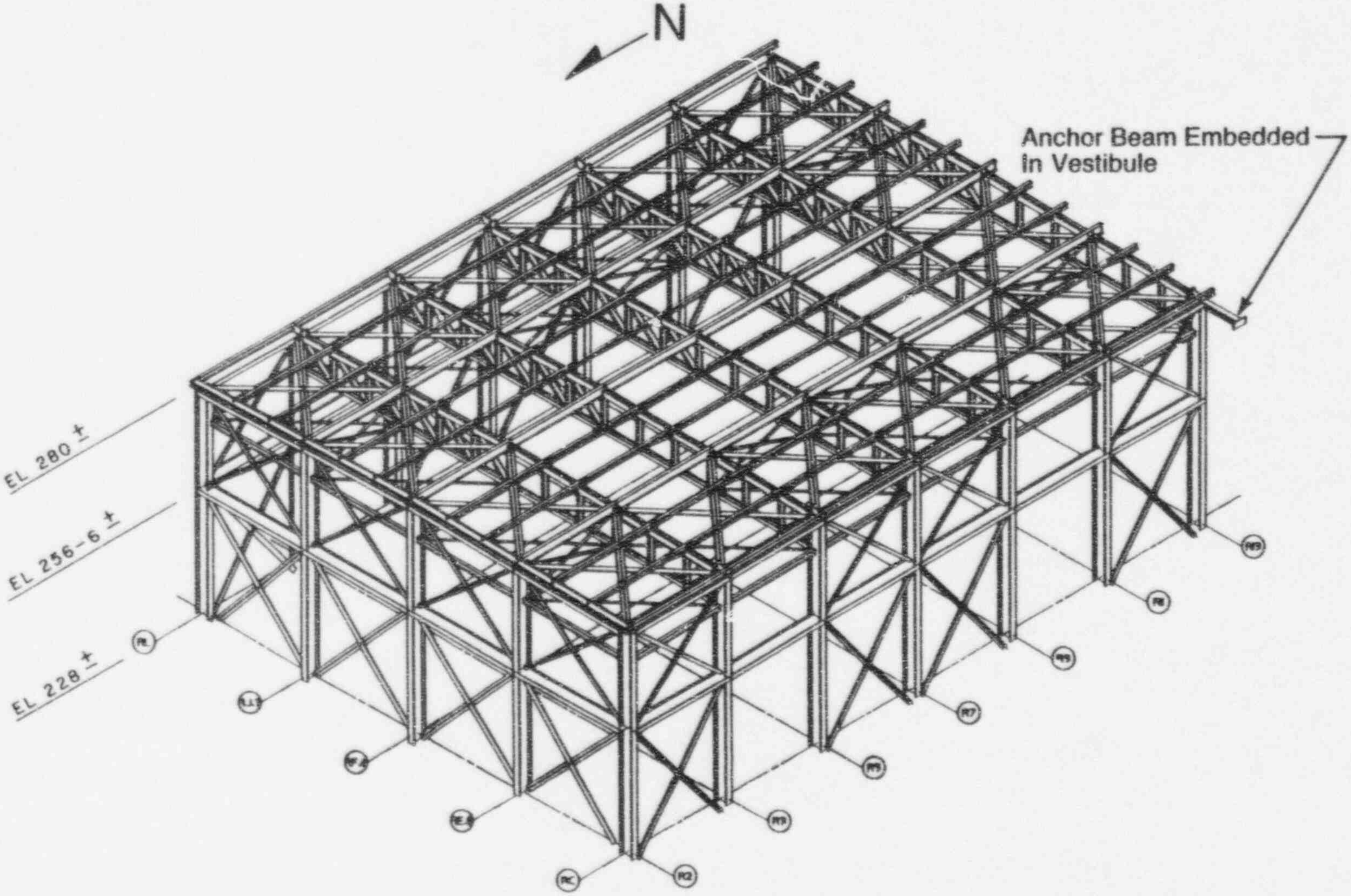


Figure 3.1-19 Plant Hatch Unit 1 View of Structural Steel Framing of Roof Structure

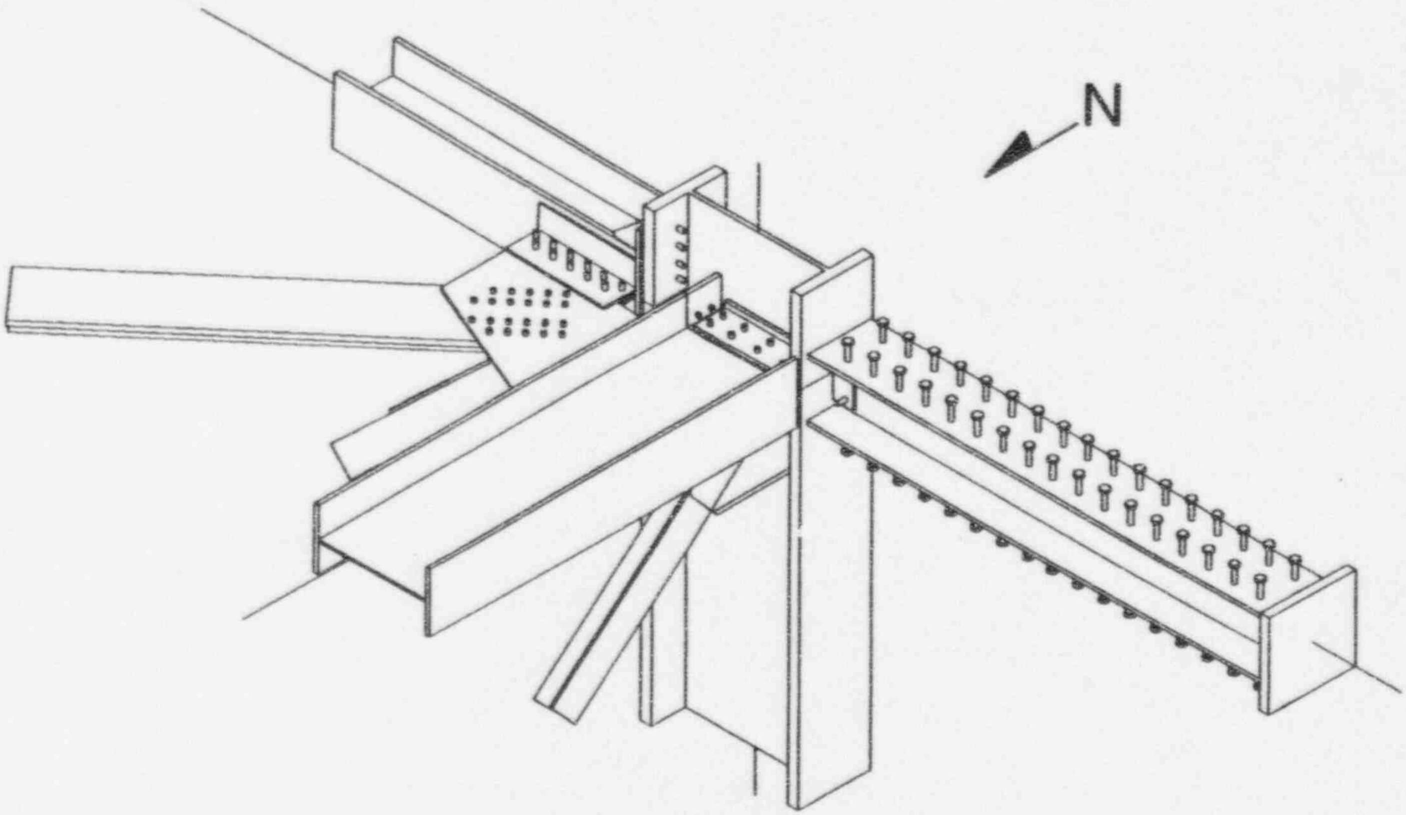


Figure 3.1-20 Isometric View of Detail of Steel Framing at Location of Anchor Beam

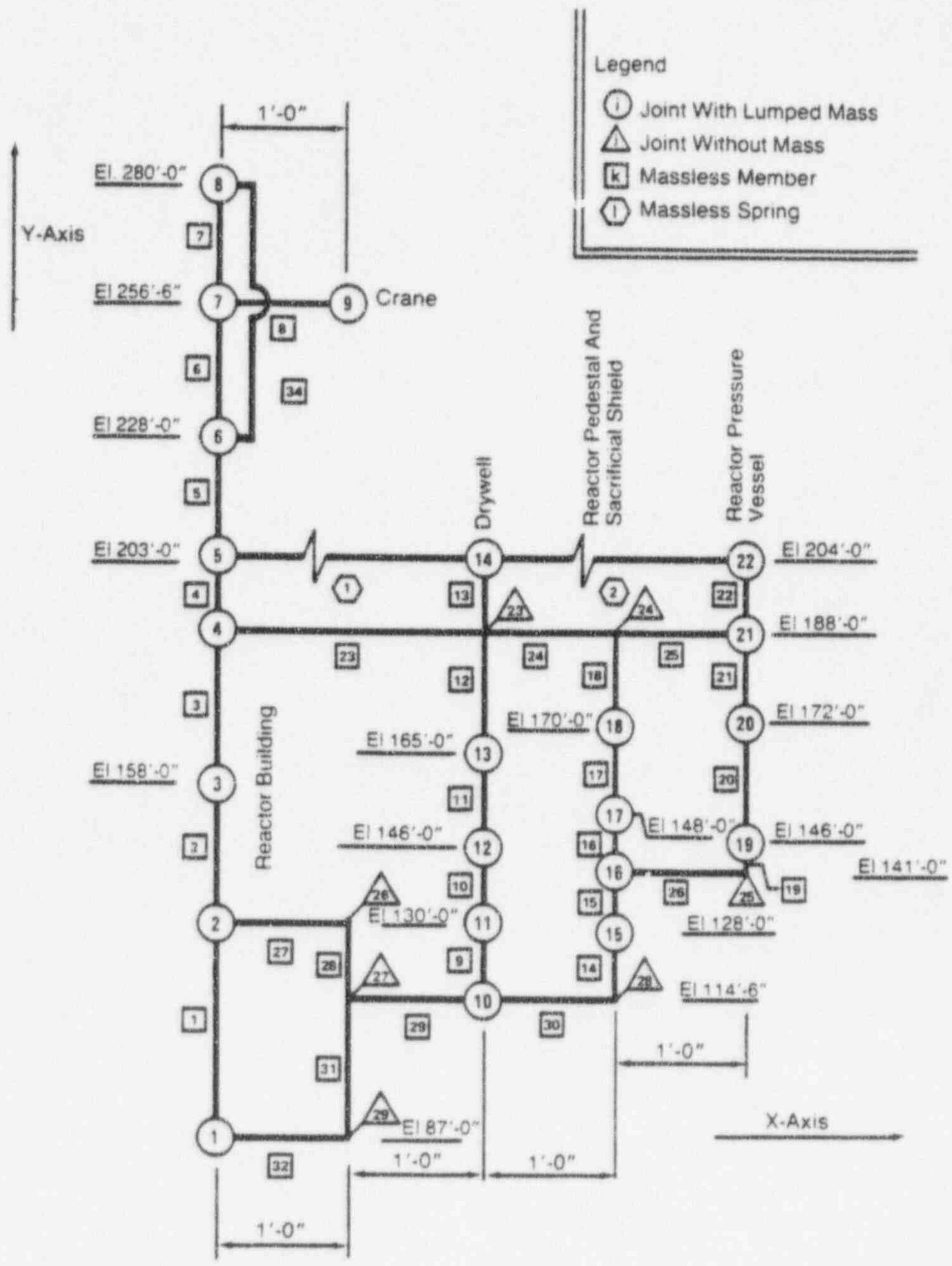


Figure 3.1-21 Unit 1 Reactor Building Seismic Model, N-S and E-W Roof Connected to Vestibule

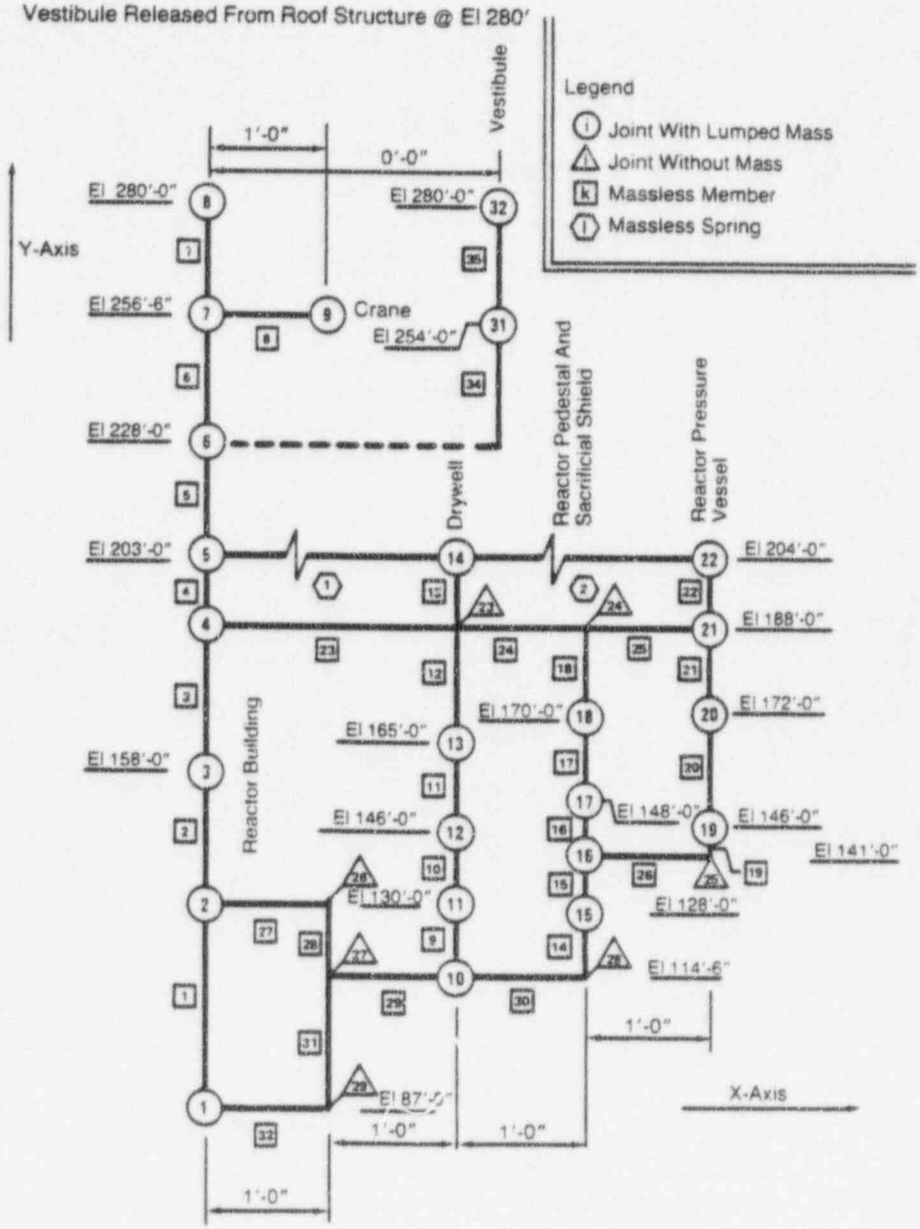


Figure 3.1-22 Unit 1 Reactor Building Seismic Model, N-S and E-W Roof Disconnected from Vestibule

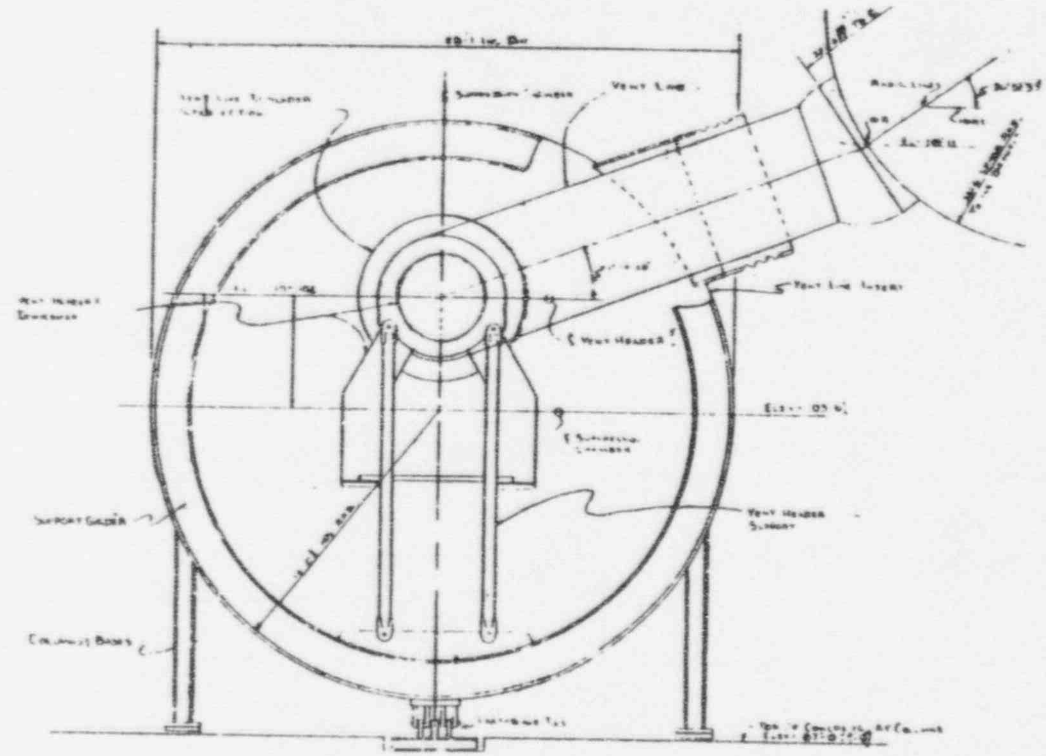


Figure 3.1-23 Torus Cross Section

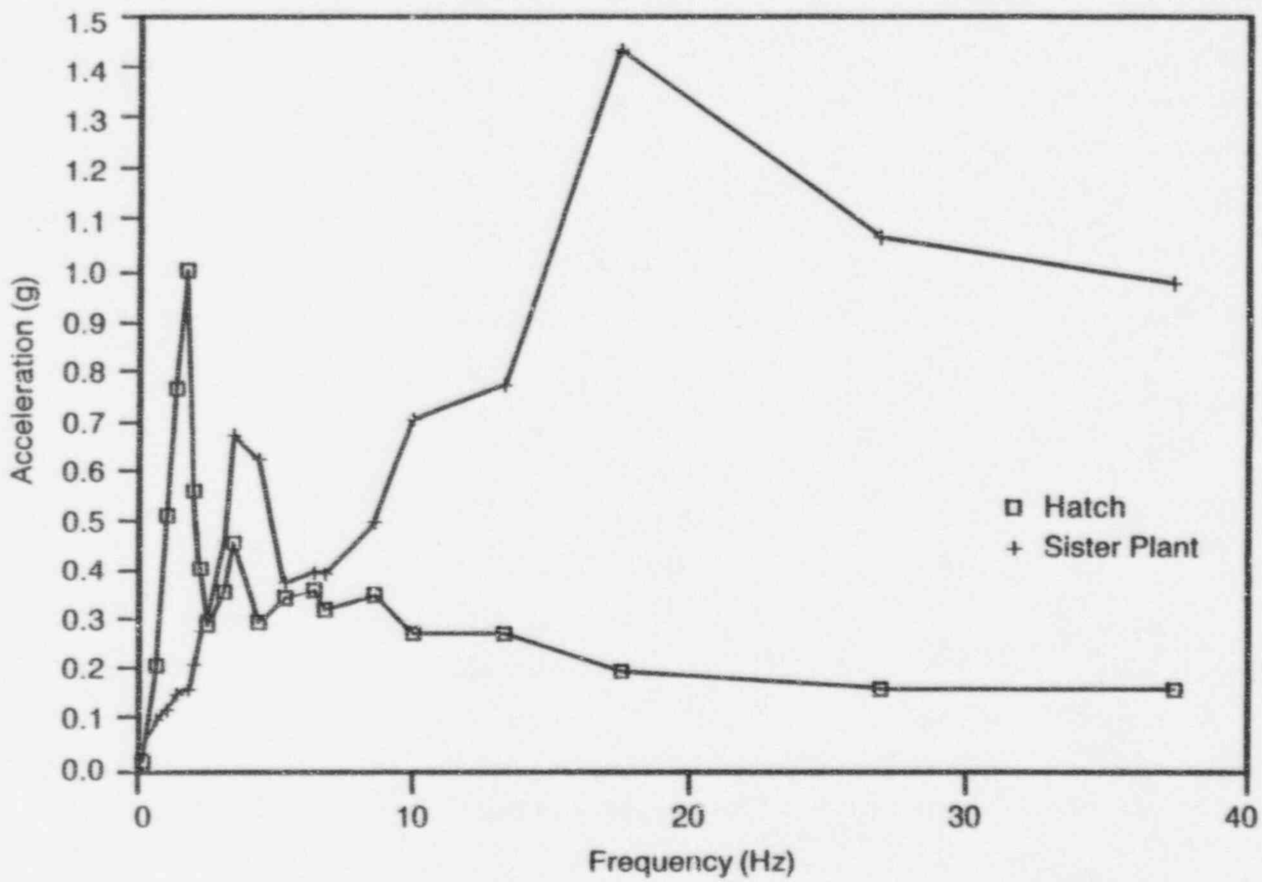


Figure 3.1-24 In-Structure Response Spectra Comparison - Node 16, Top of Pedestal

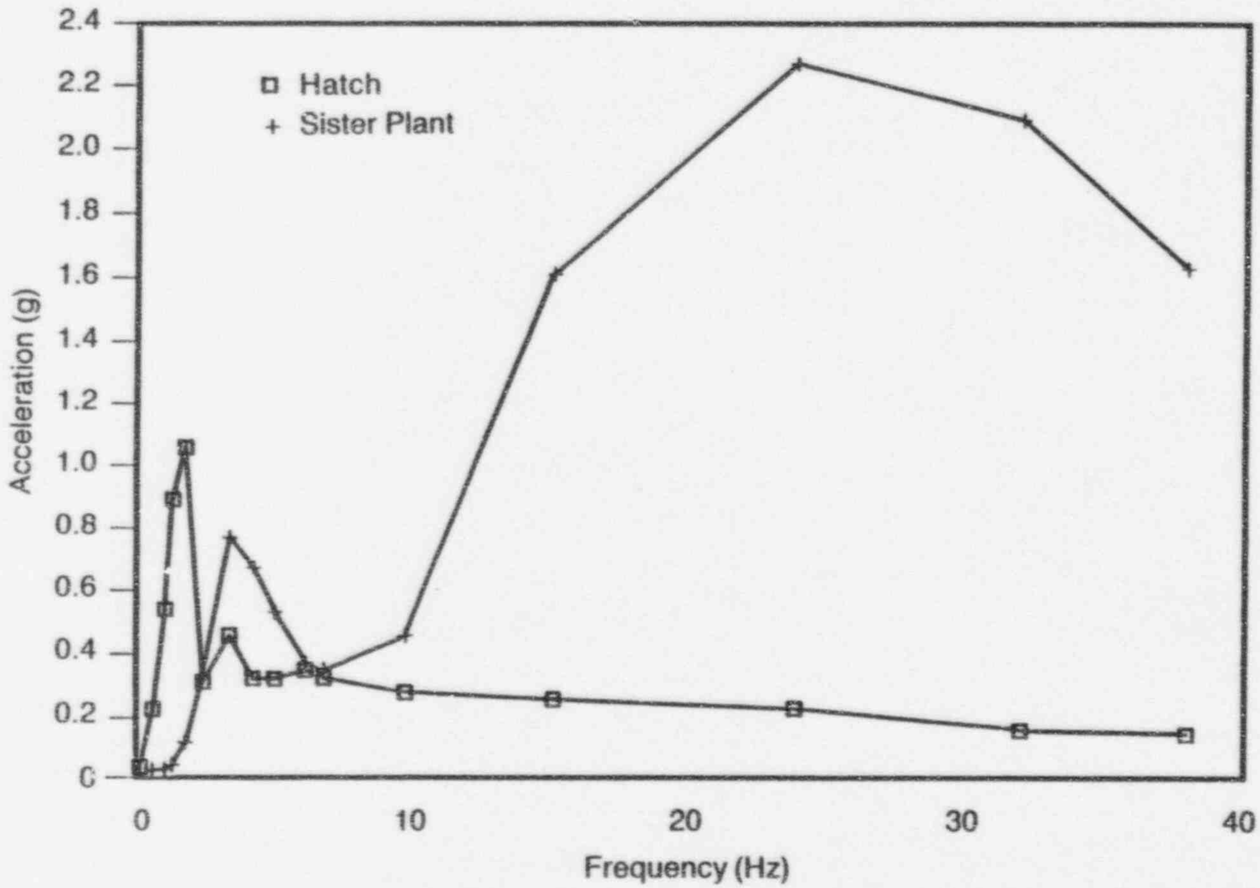


Figure 3.1-25 In-Structure Response Spectra Comparison - Node 19, Reactor Pressure Vessel

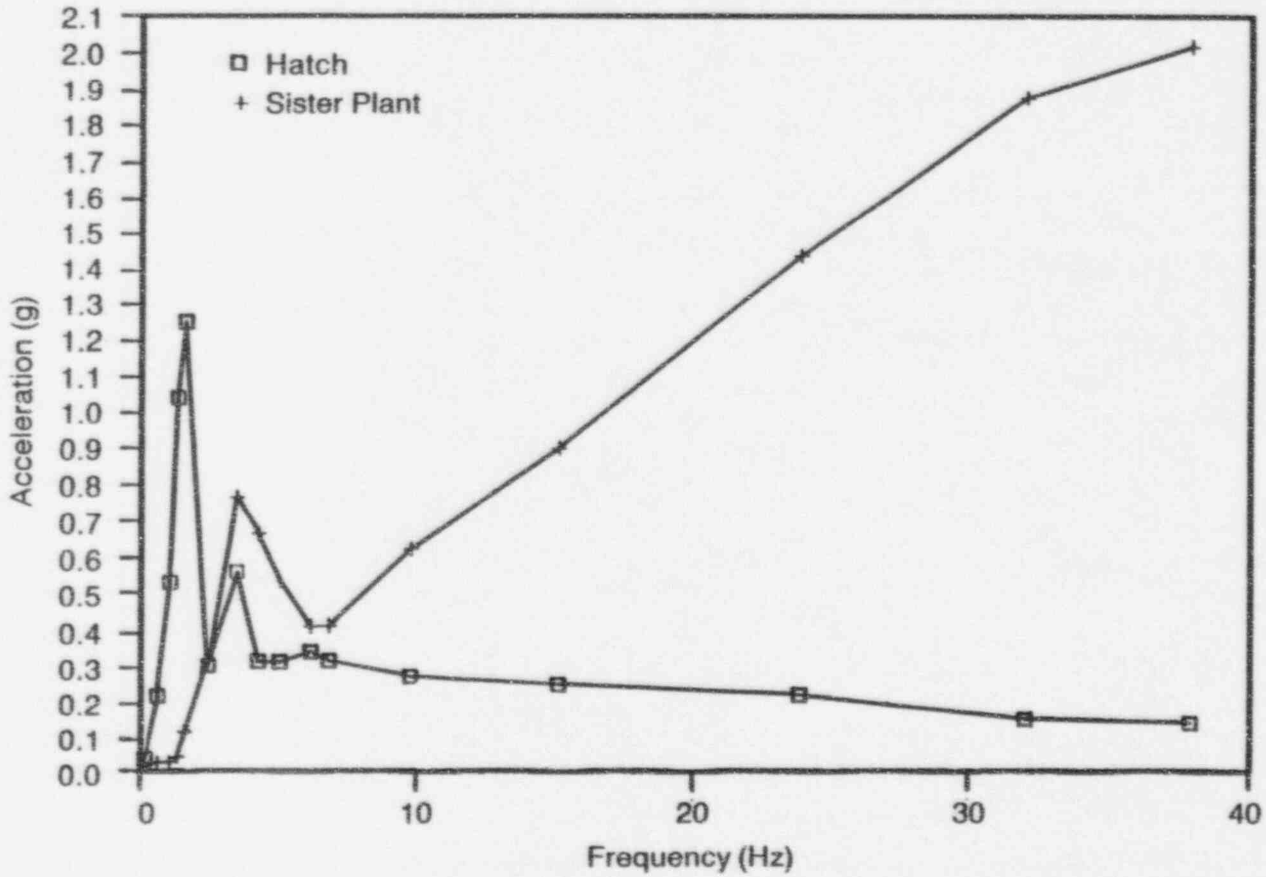


Figure 3.1-26 In-Structure Response Spectra Comparison - Node 20, Reactor Pressure Vessel

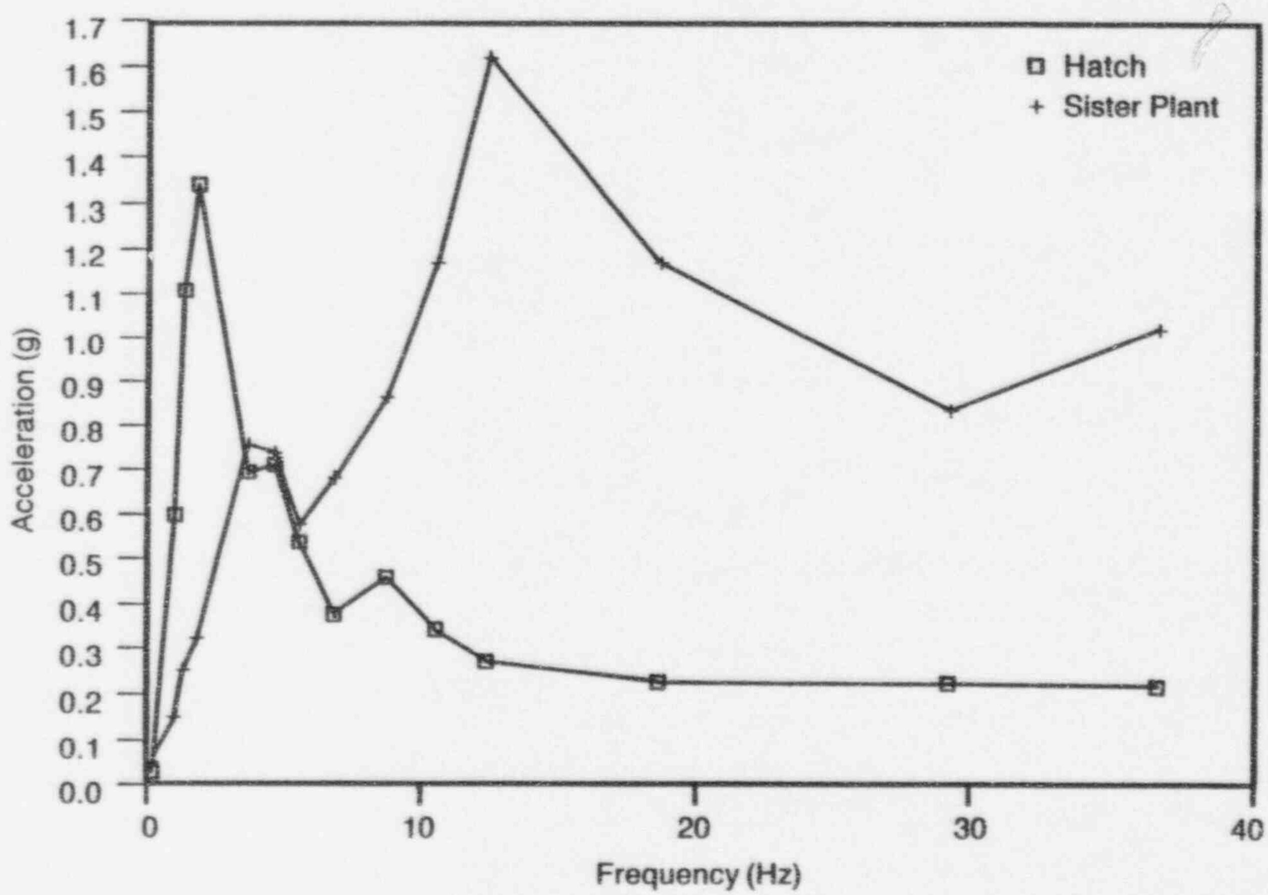


Figure 3.1-27 In-Structure Response Spectra Comparison - Node 21, Reactor Pressure Vessel

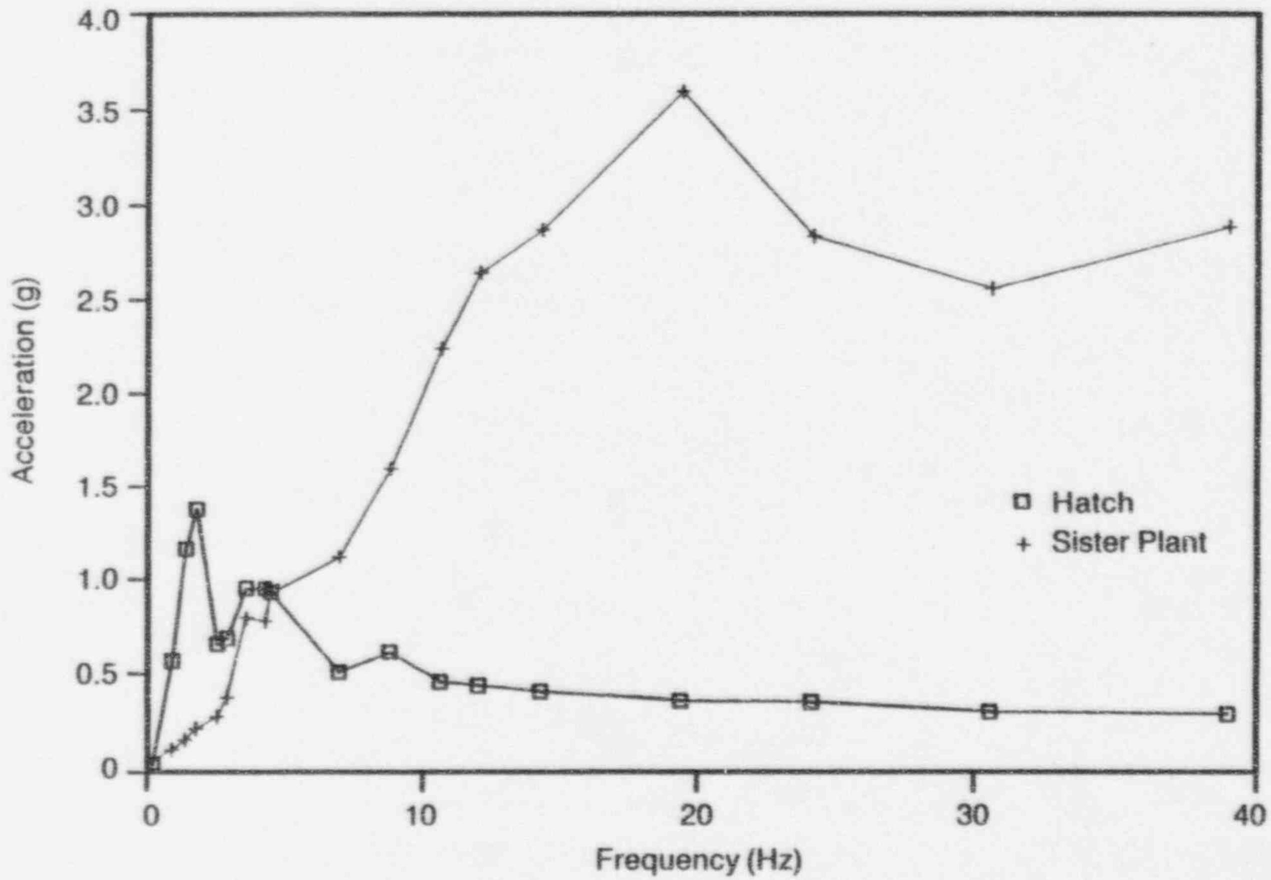


Figure 3.1-28 In-Structure Response Spectra Comparison - Node 22, Reactor Pressure Vessel

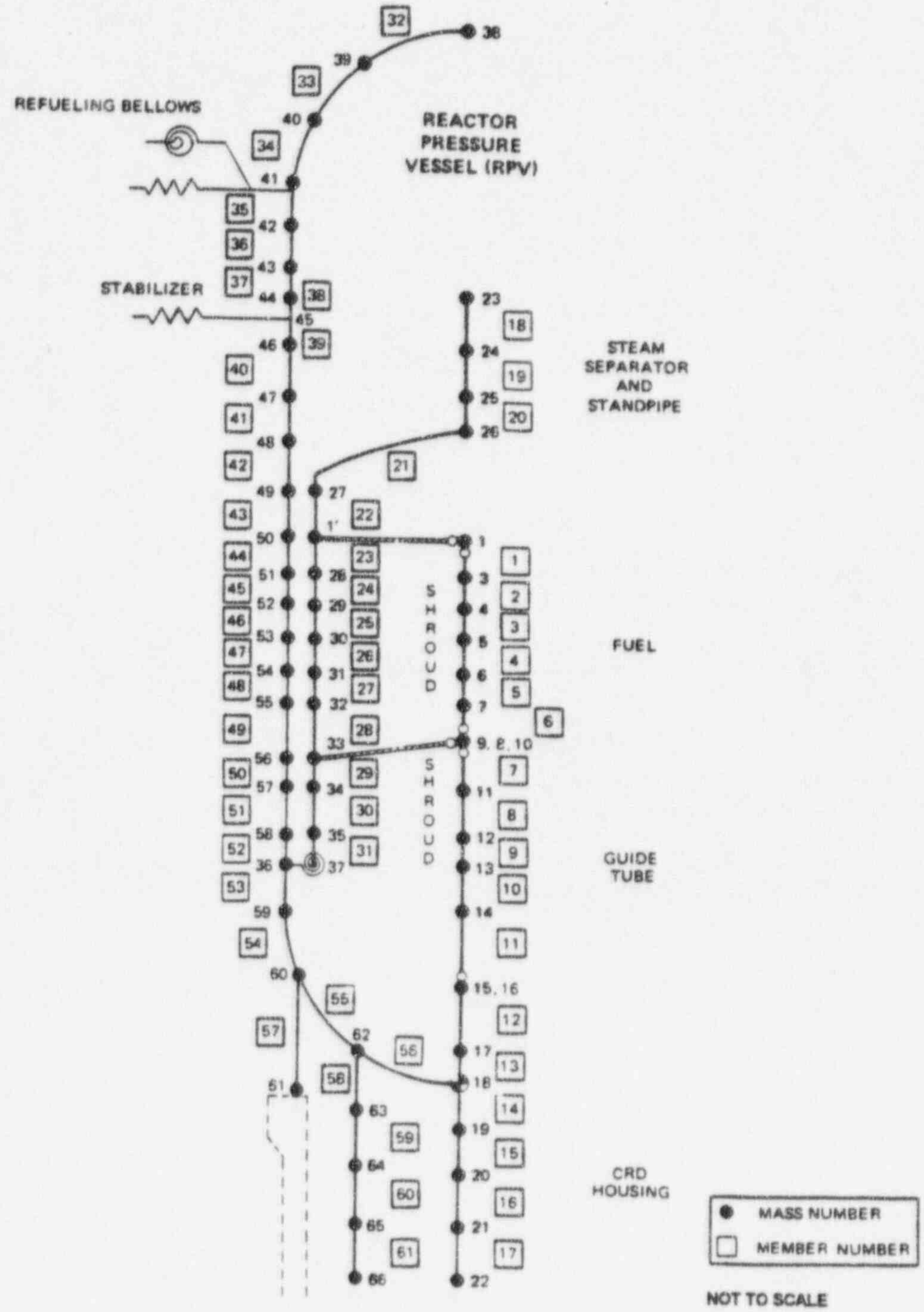


Figure 3.1-29 Dynamic Model of Unit 2 Reactor Pressure Vessel and Internals

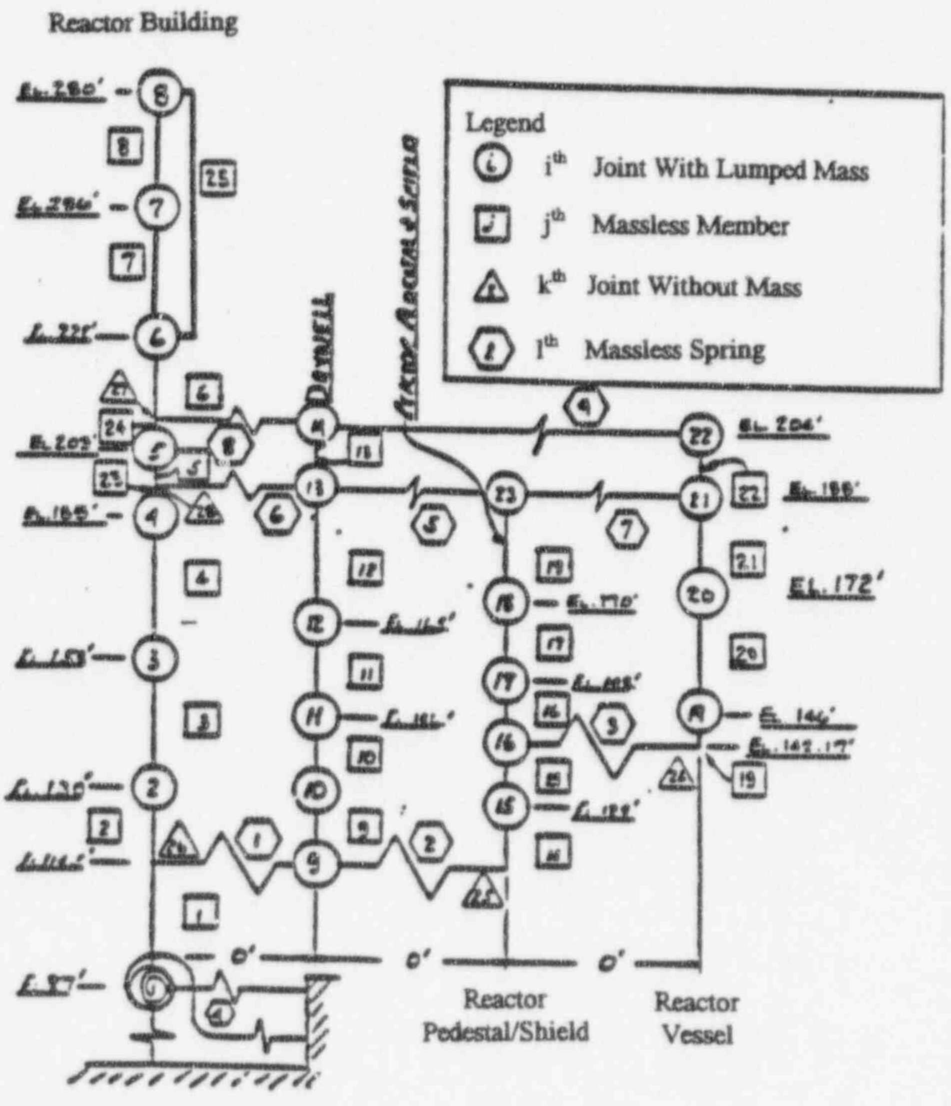
CALCULATION SHEET

JOB NO.	CALC. NO.	REV. NO.	SHEET NO.
6511	C-502.1	0	54

ORIGINATOR	DATE	CHECKED	DATE
[signature]	12/17/85	[signature]	12/17/85

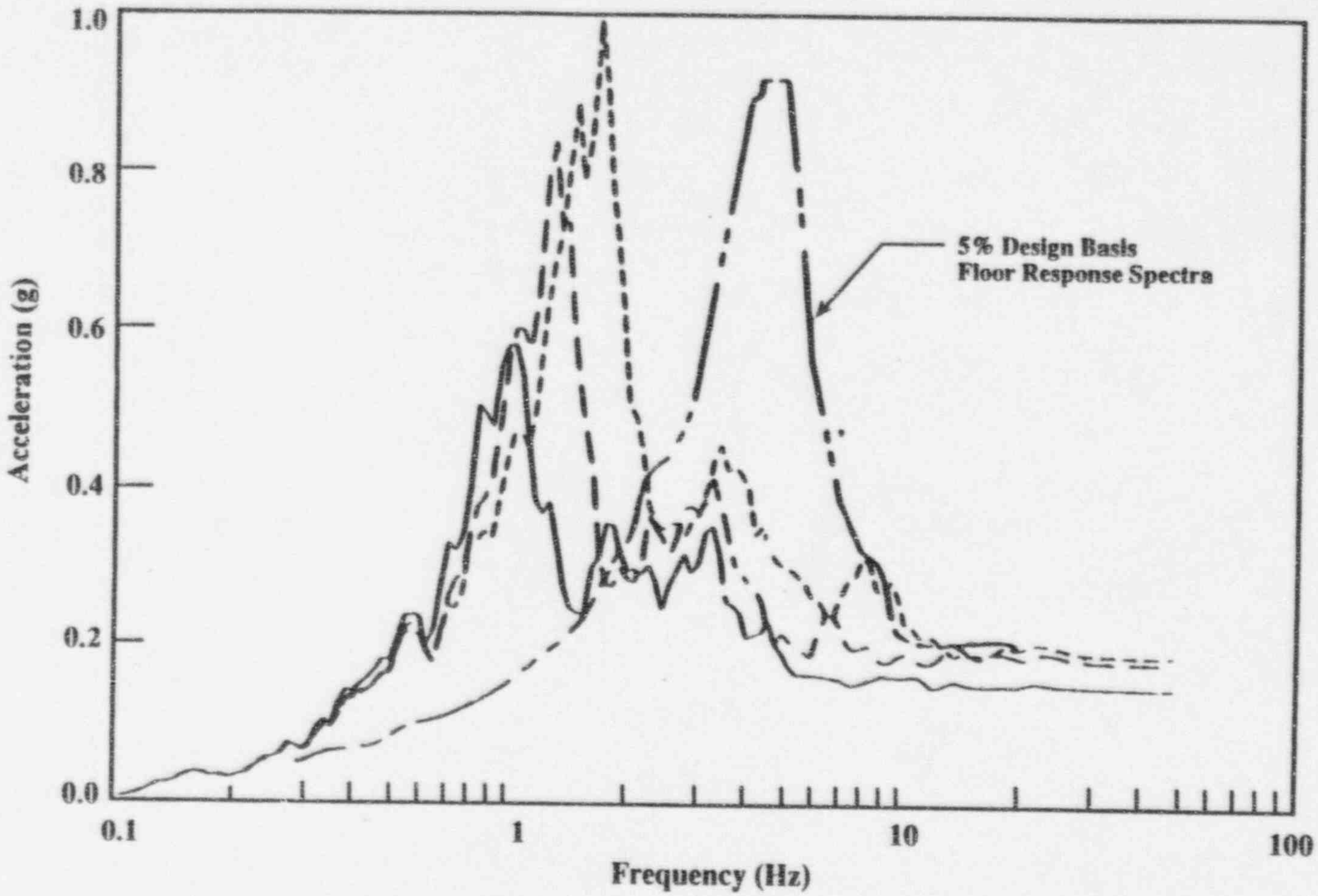
VI.B

1985
Hatch Unit 2 Reactor Building and Internals
Mathematical Model



REV. 1

Figure 3.1-30 Plant Hatch Unit 2 Reactor Building Model



Legend:

Lower Bound
Intermediate
Upper Bound

Note:

5% Spectral Damping

Figure 3.1-31 Reactor Building SSI Analysis SME In-Structure Response Spectra at Mass Point 3, Elevation 158 ft, N-S Direction

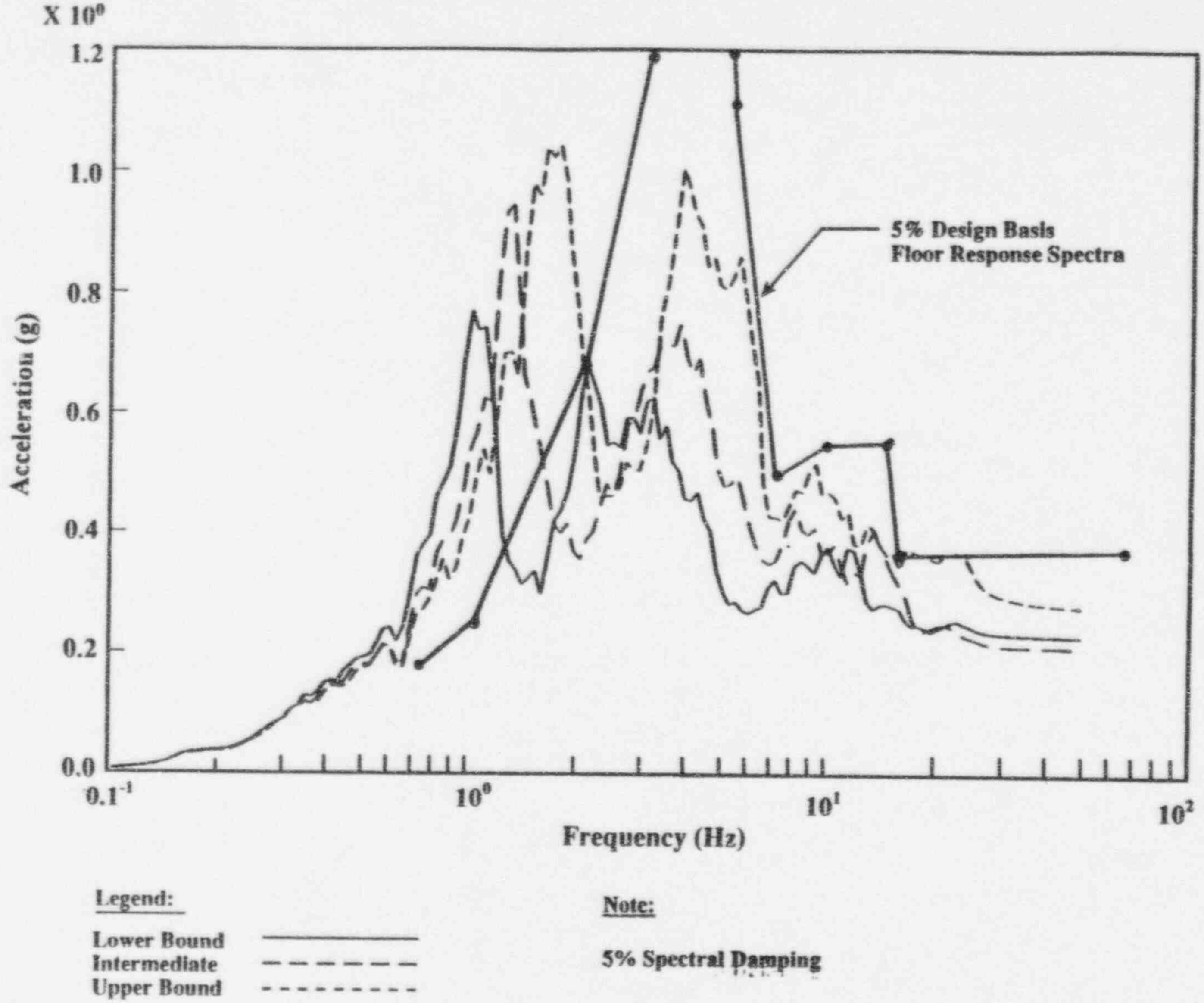


Figure 3.1-32 Reactor Building SSI Analysis SME In-Structure Response Spectra at Mass Point 22, Elevation 204 ft, E-W Direction

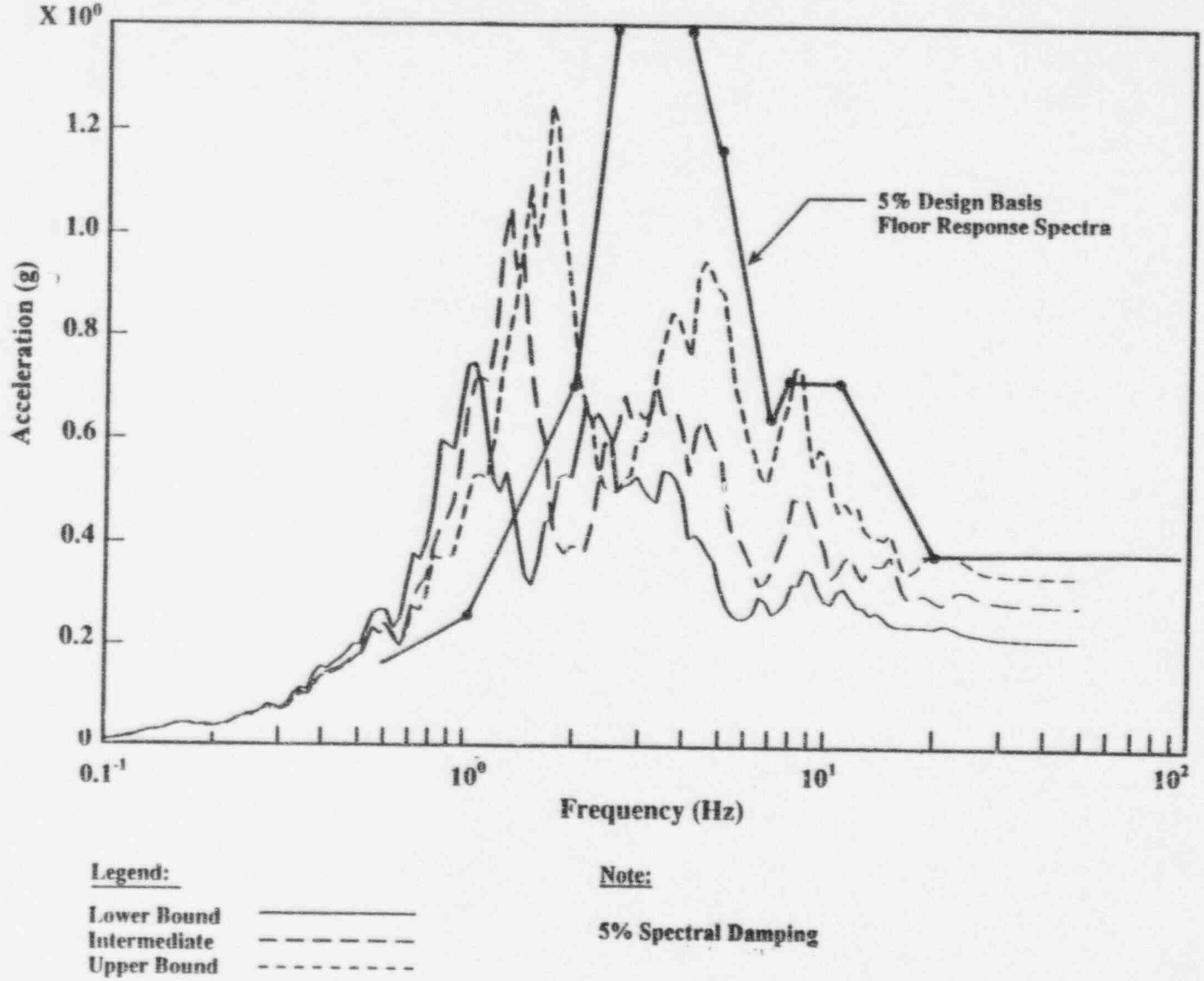


Figure 3.1-33 Reactor Building SSI Analysis SME In-Structure Response Spectra at Mass Point 22, Elevation 204, N-S Direction

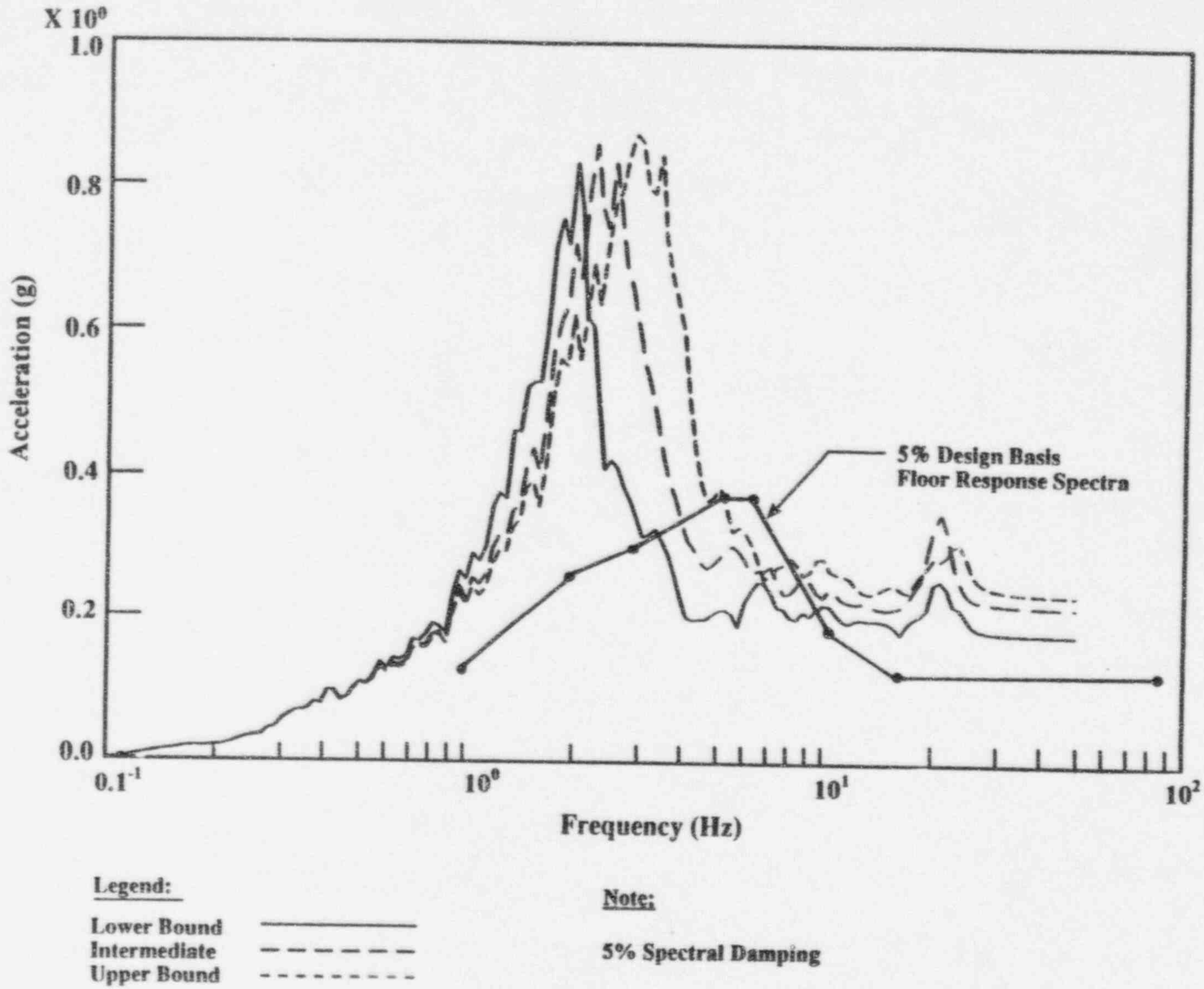


Figure 3.1-34 Reactor Building SSI Analysis SME In-Structure Response Spectra at Mass Point 22, Elevation 204 ft, Vertical Direction

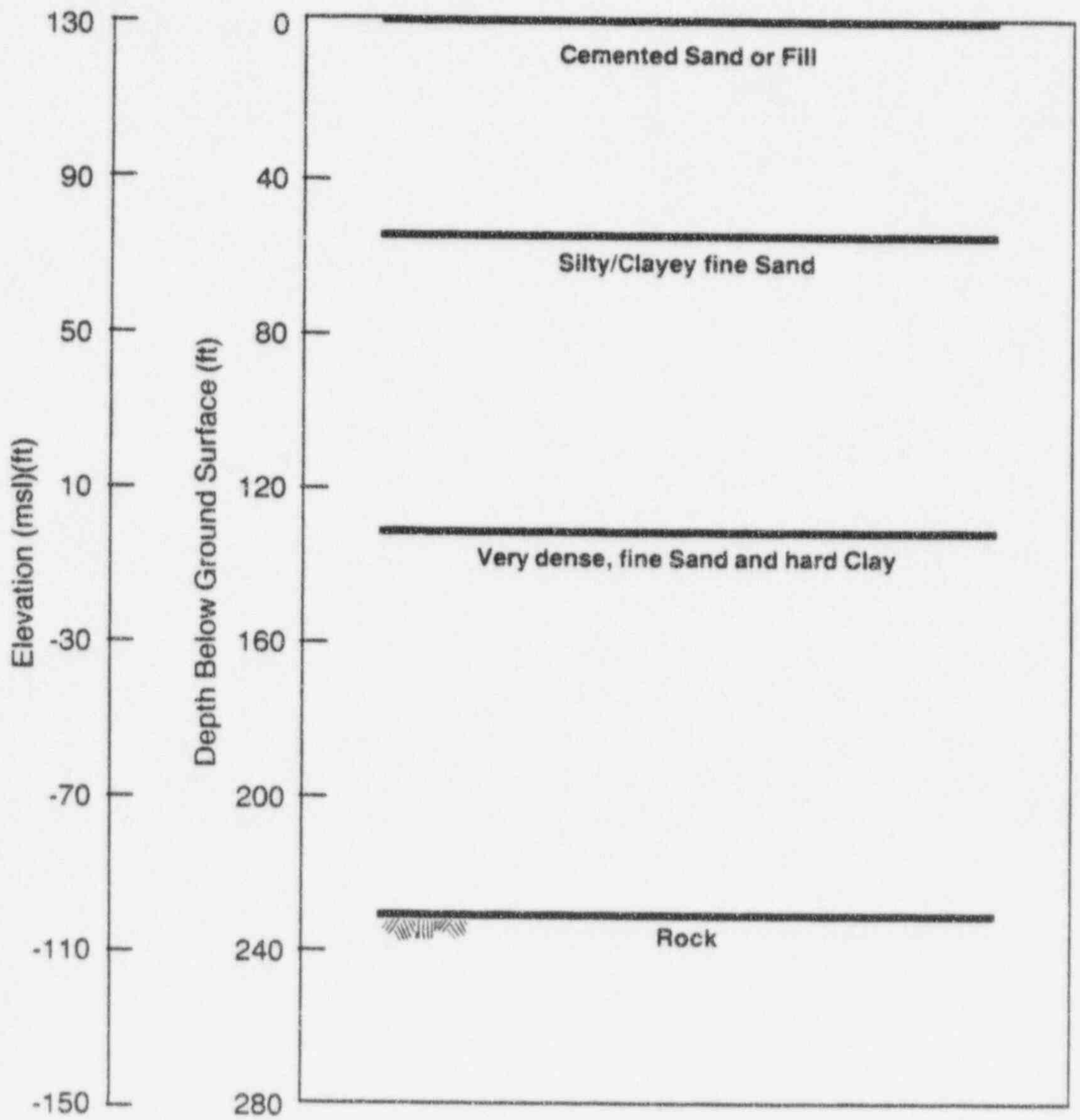


Figure 3.1-35 Plant Area Generalized Soil Profile

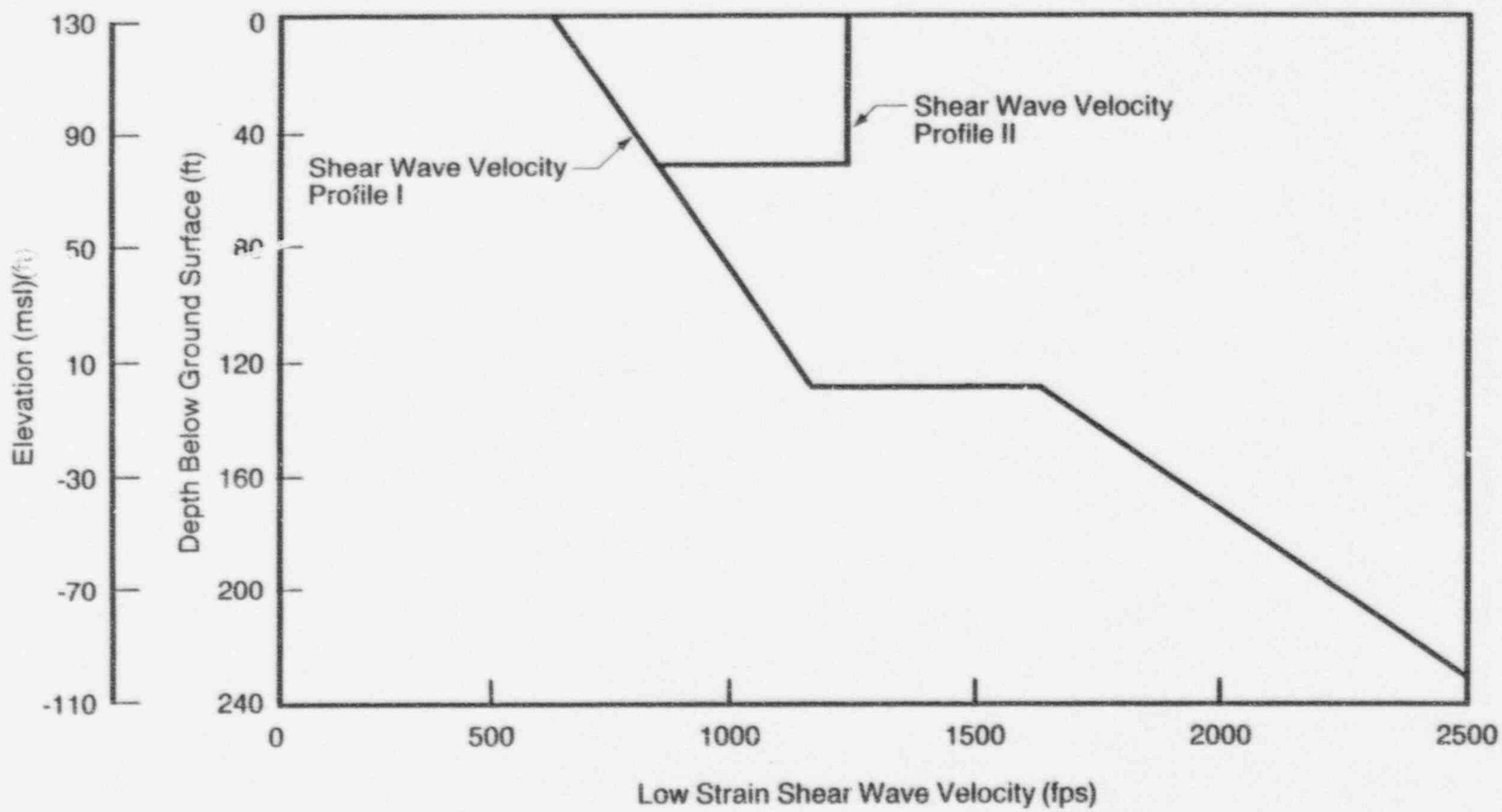


Figure 3.1-36 Best-Estimate Shear Wave Velocity Profiles, Plant Area

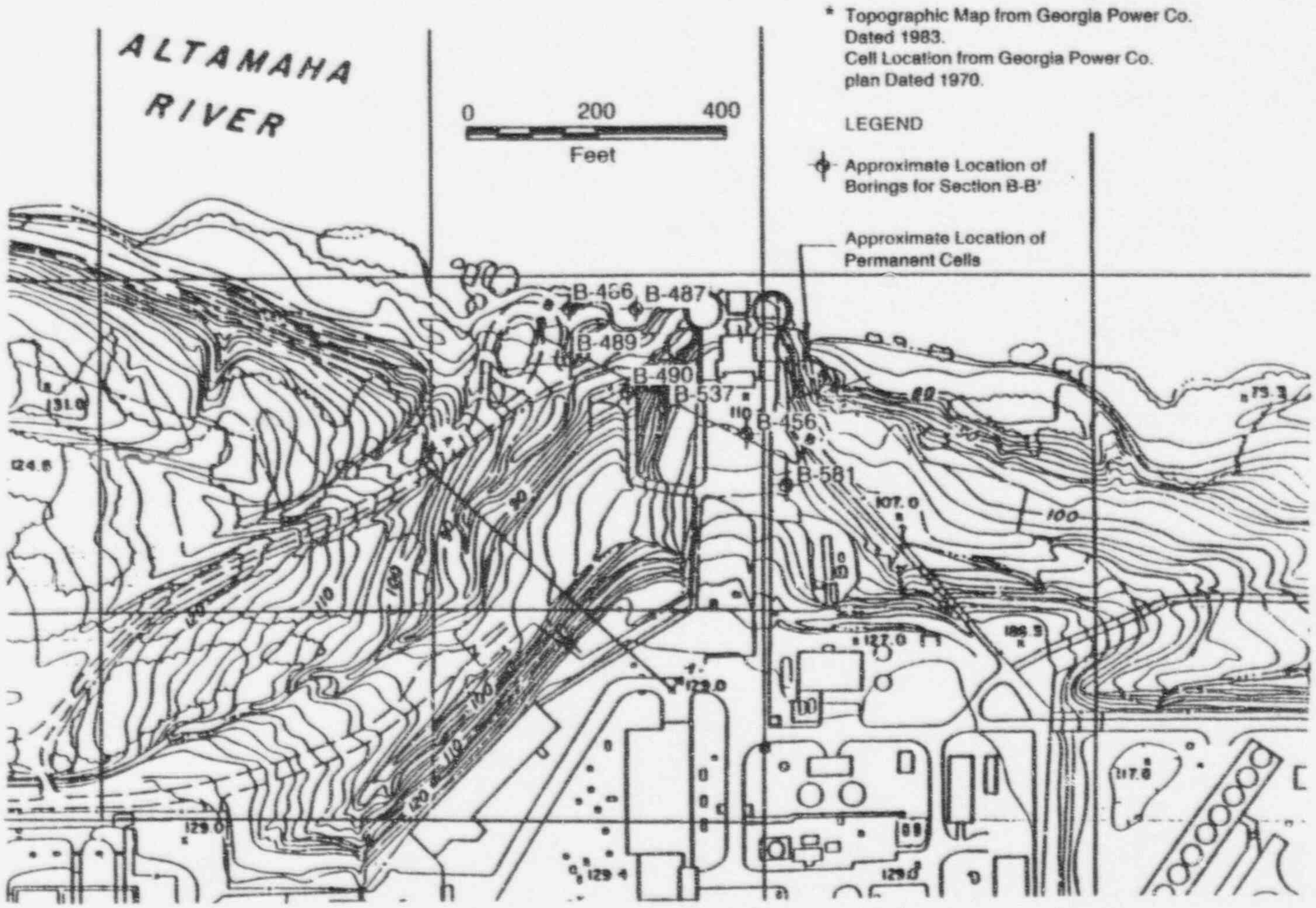


Figure 3.1-37 Boring Locations, Water Intake Area

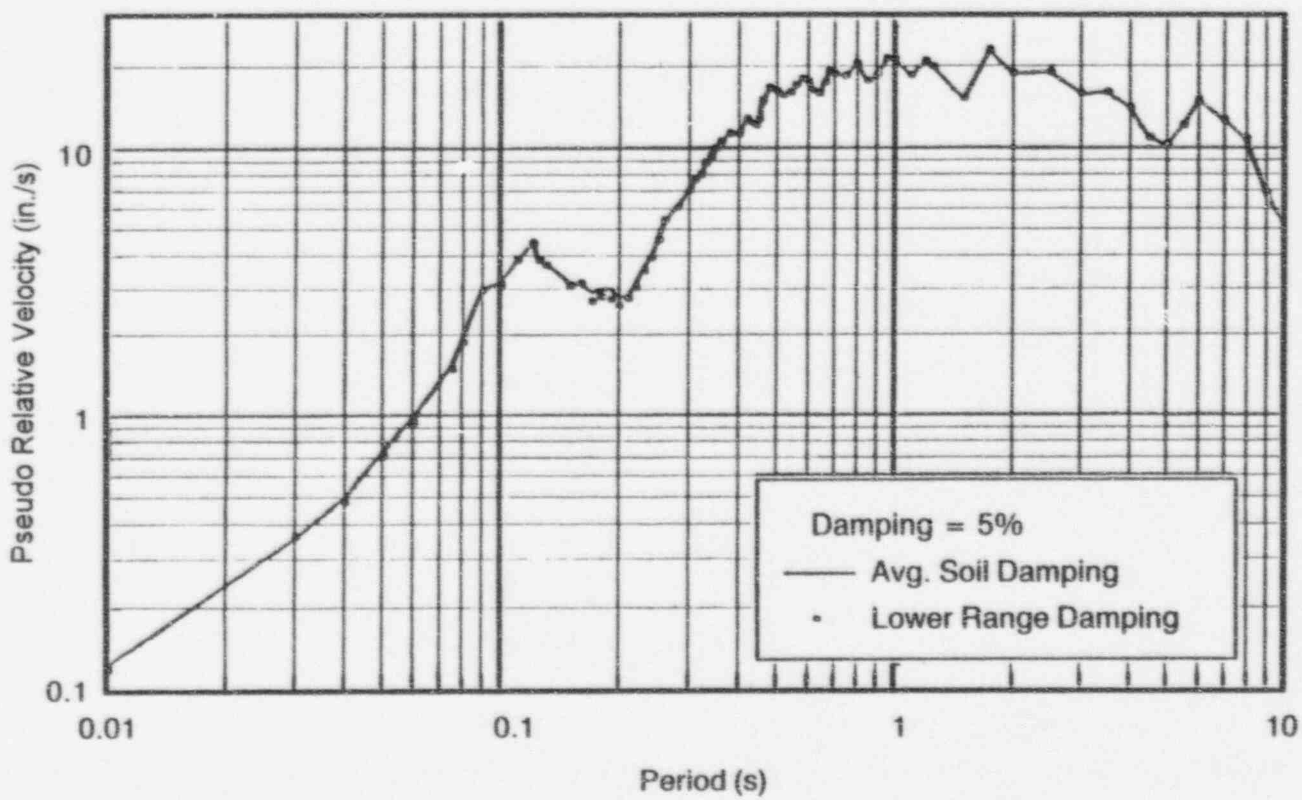


Figure 3.1-38 SMA Spectrum at Foundation Level, Profile II Using 1.2 Multiplied by Average Soil Modulus

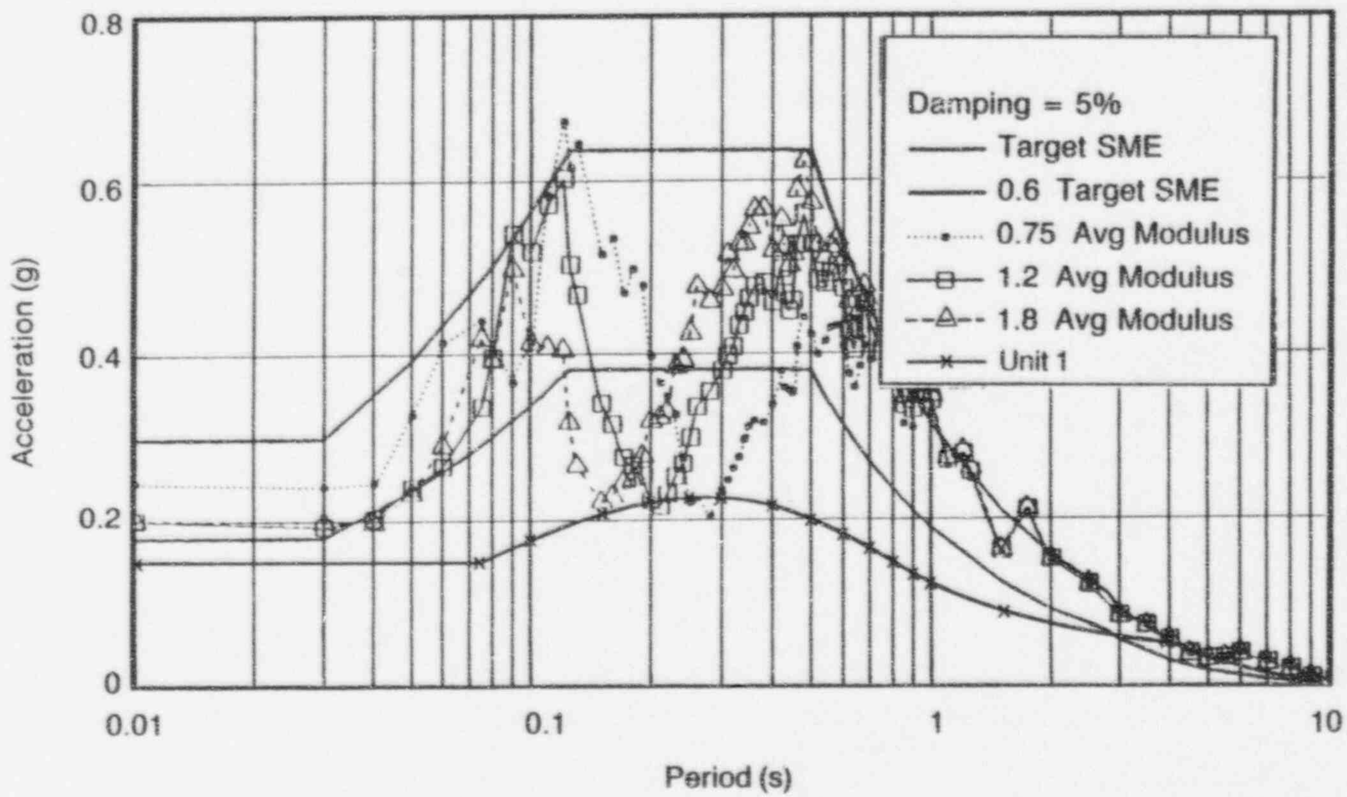


Figure 3.1-39 SMA Target Spectrum and Spectra at Foundation Level Average Soil Damping

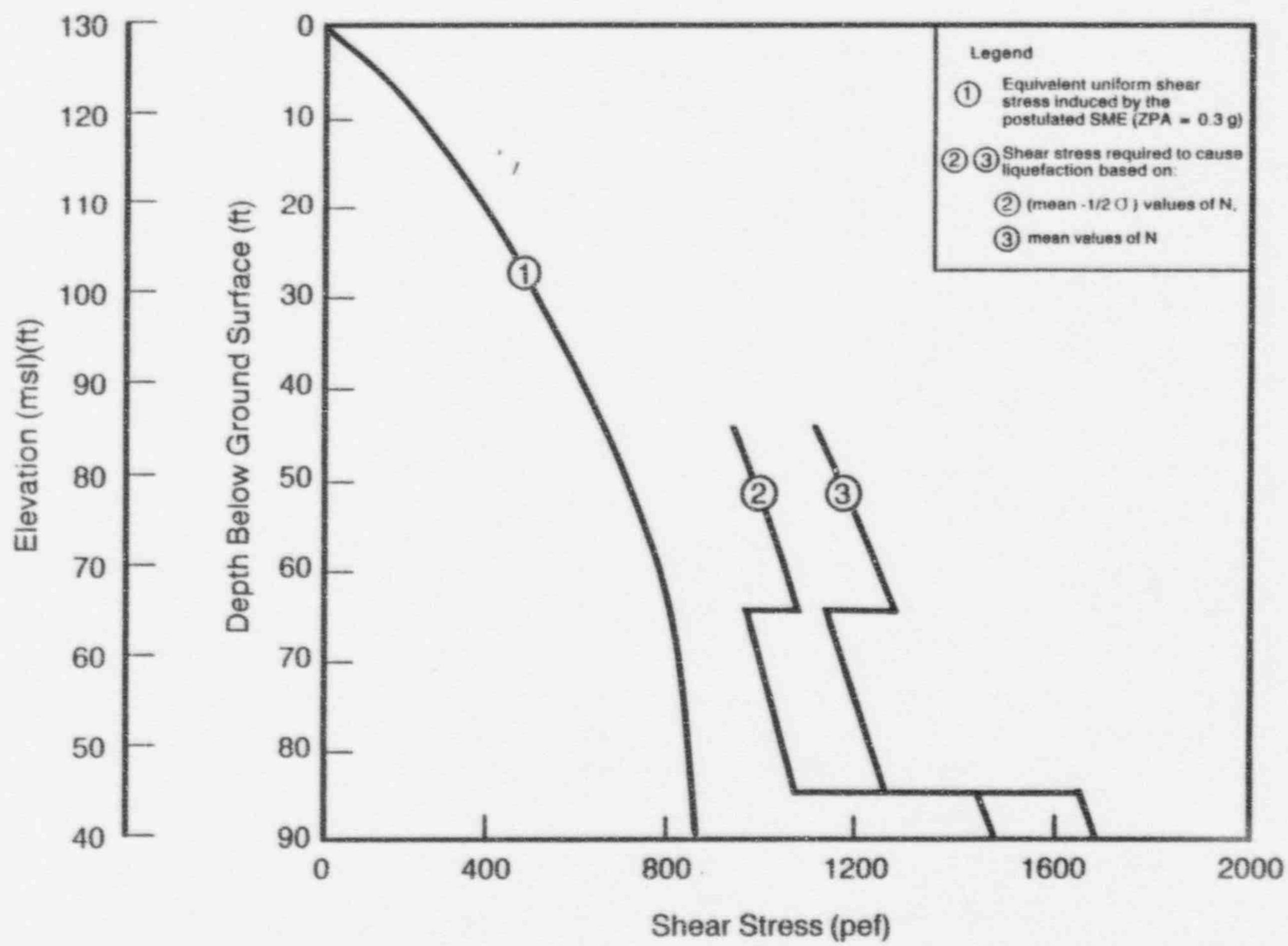
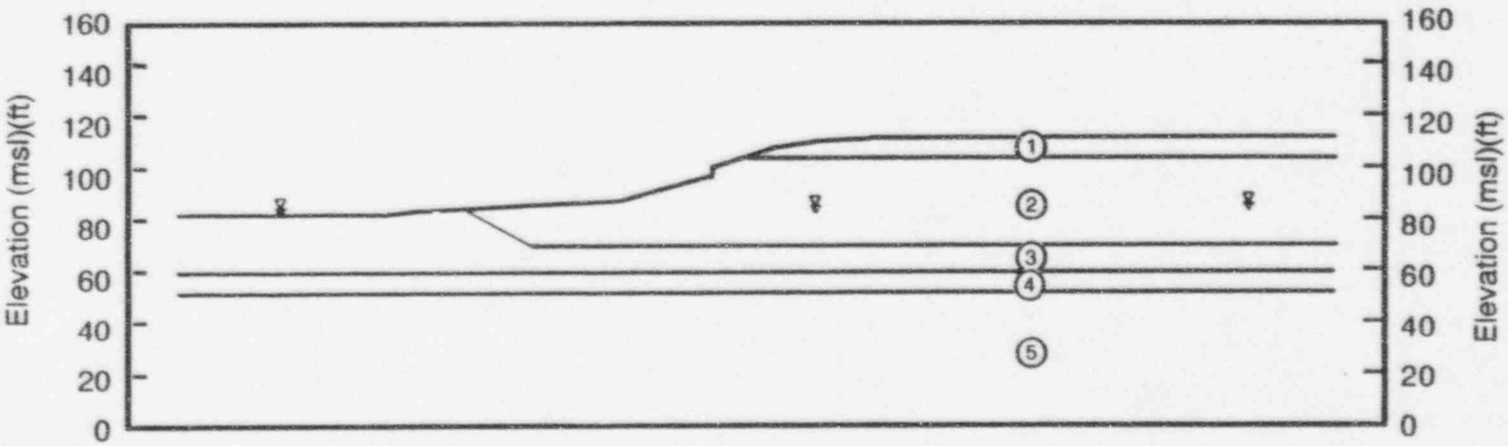


Figure 3.1-40 Comparison of SME-Induced Stresses with Stresses Required to Cause Liquefaction, Plant Area



Legend
1. Silty/Clayey fine Sand
2. Cemented Sand
3. Clay
4. Silty/Clayey fine Sand
5. Very dense Silty/Clayey fine Sand and hard Clay

Figure 3.1-41 Generalized Soil Layering, Section B-B' Water Intake Area

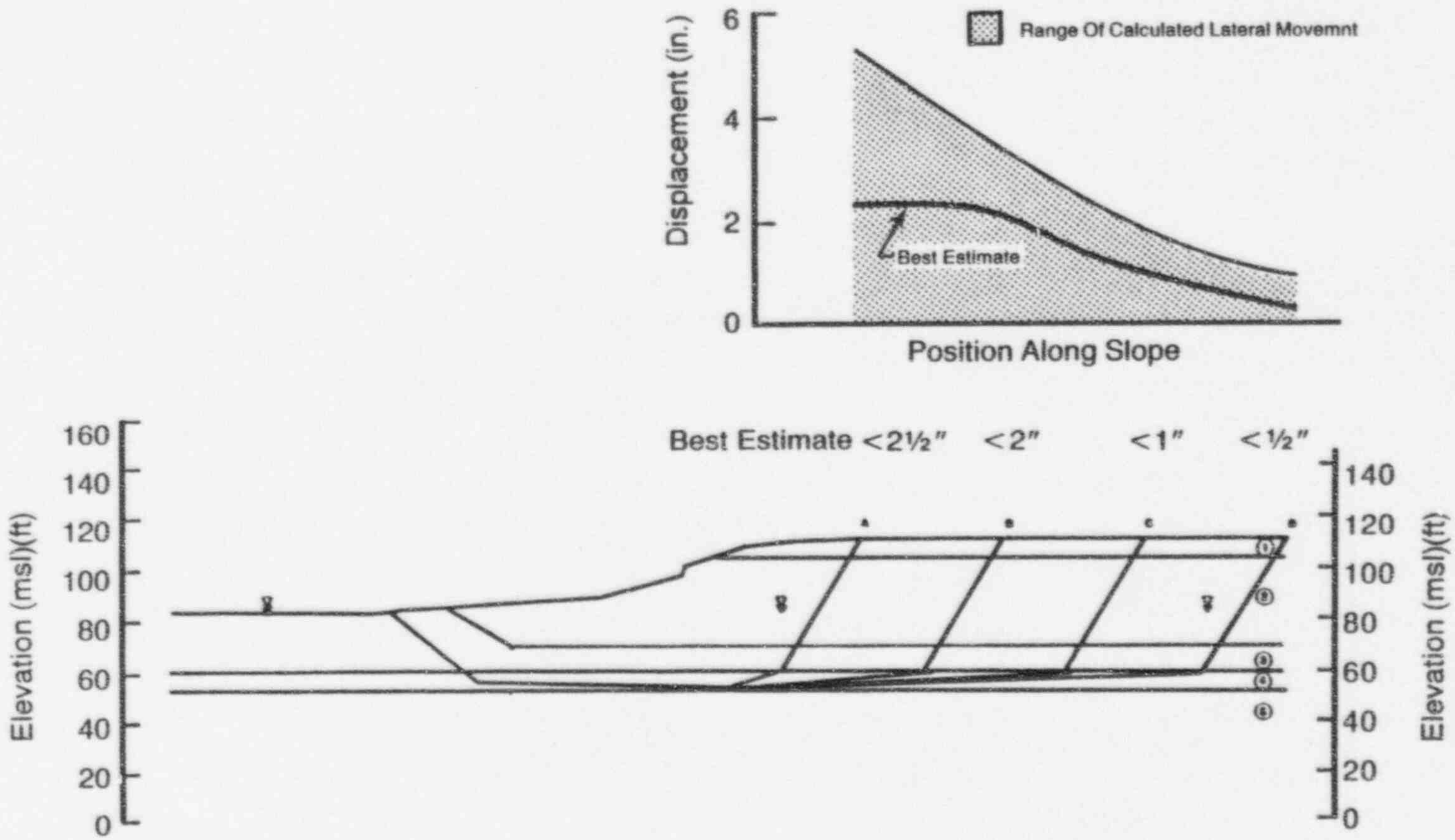


Figure 3.1-42 Estimated Lateral Movement Caused by SME, Section B-B′

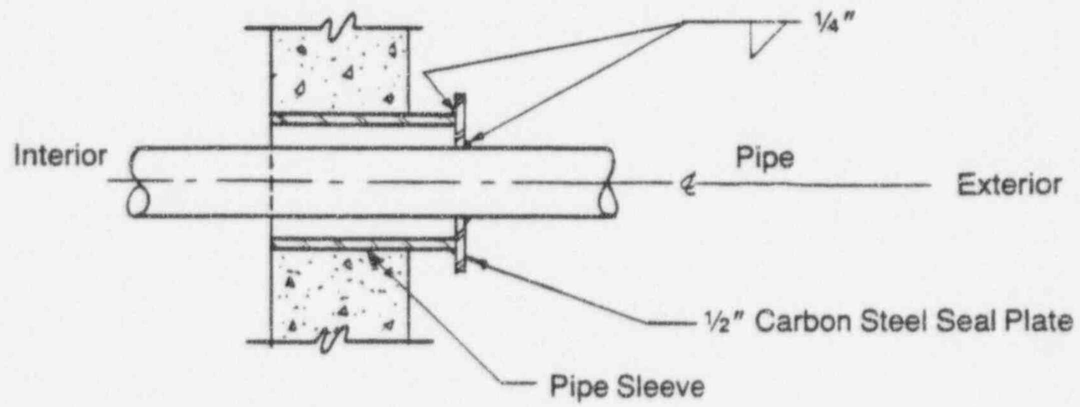


Figure 3.1-43 Typical Penetration Detail for Buried Pipe Entering Building at Exterior Wall

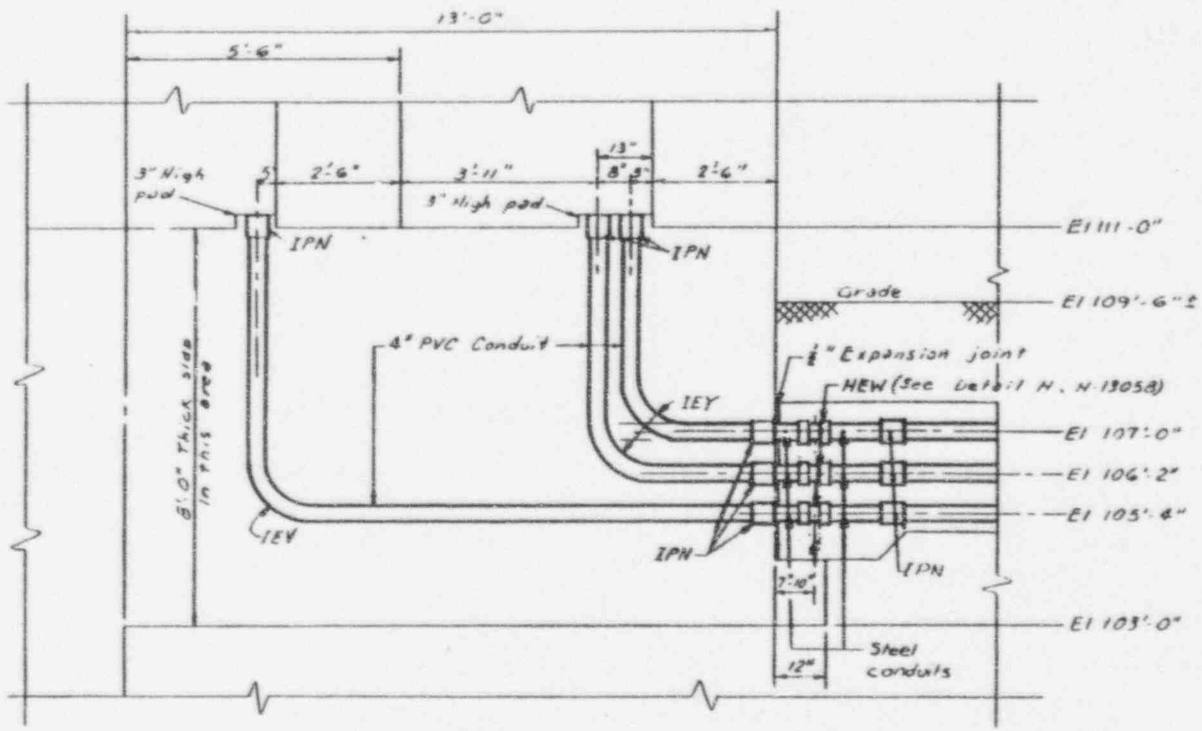


Figure 3.1-44 Interface of Buried Conduit Duct Run and Intake Structure

3.2 USI A-45, GI-131, AND OTHER SEISMIC SAFETY ISSUES

3.2.1 USI A-45, "SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS"

As stated in this report, the high-confidence-low-probability-of-failure capacity for Plant Hatch meets the 0.3 g pga review level earthquake requirement for a focused-scope plant as specified in NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events for Severe Accident Vulnerabilities, Final Report," of June 1991, pending completion of plant modifications as described in section 3.1.4.9. This includes the equipment/components that comprise the decay heat removal system included in the Safe Shutdown Equipment List. Thus, USI A-45, with regard to seismic risk at Plant Hatch, is considered closed.

3.2.2 GI-131, "POTENTIAL SEISMIC INTERACTION INVOLVING THE MOVABLE IN-CORE FLUX MAPPING SYSTEM USED IN WESTINGHOUSE PLANTS"

Because Plant Hatch is a General Electric plant, the issues stated in GI-131 are not applicable.

4. INTERNAL FIRES ANALYSIS

The fire portion of the Individual Plant Examination for External Events (IPEEE) for the Edwin I. Hatch Nuclear Plant Units 1 and 2 was performed in response to Nuclear Regulatory Commission (NRC) Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)" (Reference 1), which requires licensees to identify fire-induced plant-specific vulnerabilities to severe accidents.

4.0 METHODOLOGY SELECTION

4.0.1 METHODOLOGY OVERVIEW

The fire portion of the Plant Hatch IPEEE was performed based on the guidance provided in NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" (Reference 2). The Plant Hatch fire IPEEE used a scenario-based probabilistic risk assessment (PRA) to systematically and successively evaluate fire hazards and their associated risk impact.

The Plant Hatch fire IPEEE was divided into two phases: a spatial interactions analysis phase and a detailed analysis phase. The overall technical approach involving the spatial interactions analysis phase and the detailed analysis phase is illustrated in figure 4.0-1.

In the spatial interactions analysis phase, one or more fire hazard scenarios were developed for each plant location that could potentially initiate a plant transient or affect the ability of the plant to mitigate an accident. Both localized and propagation scenarios were considered. A localized scenario evaluated fires initiated within a fire zone and contained within the same fire zone. A propagation scenario evaluated fires initiated in a fire zone and propagated to one or more adjacent fire zones.

Regardless of fire severity, the amount of combustibles available, or the spatial separation between the fire sources and targets (equipment or cable that may be damaged), a scenario was developed for each fire zone that conservatively assumed damage to all plant components and cables located in the fire zone(s). This assumption allows fire zones that are insignificant fire risk contributors to be screened from the subsequent analysis. This screening allows an effective and efficient use of the resources for the locations that are risk significant.

The occurrence frequency for fires was calculated based on both nuclear power industry fire data and plant-specific data. The plant impact corresponding to each scenario was determined based on the Individual Plant Examination (IPE) (Reference 3) top events in which the affected components (due to fire damage to these components or their associated cables) were modeled. In the spatial interactions analysis, this information was combined so that the analysis scenarios could be quantitatively screened for risk significance. A scenario was screened from further evaluation if its core damage frequency (CDF) is < 0.1 percent of the total internal events-induced

CDF (i.e., a screening cutoff value of 2.1E-08 per year for Unit 1 and 2.2E-08 per year for Unit 2 was used). Instead of discarding the numeric results of the screened scenarios, the CDF of the screened scenarios was aggregated and included in the final CDF calculated in the detailed analysis phase. This approach accounts for all quantified fire-induced risk.

The scenarios retained from the quantitative screening were deemed to be relatively risk significant and required further analysis. In the detailed analysis phase, in order to evaluate the fire risk in more detail, one or more subscenarios were developed. A subscenario was used to describe the progression of a unique fire event initiated by specific ignition source(s) in a particular location. The evaluation of subscenarios also considered in-situ and transient combustible loading, critical components, location-specific features, potential impact from the fire, and possible operator recovery actions. Spatial separation between fire source(s) and target(s) in each subscenario was used to analyze the fire risk in more detail.

Next, a more detailed analysis of the ignition frequency for each fire source was performed. One or several frequency reduction factors (i.e., geometric, severity, and fire nonsuppression) were applied to each detailed analysis subscenario. As each frequency reduction factor was applied (with value ranging from 0.0 to 1.0), conservatism introduced in the earlier phase of the analysis was reduced. Whenever one or more reduction factors led to the conclusion that the risk associated with a detailed analysis subscenario was relatively insignificant, the analysis for that subscenario would be halted and the remaining frequency reduction factor(s) would assume a value of 1.0.

Each subscenario was evaluated iteratively until the subscenario was considered to be relatively nonrisk significant or all frequency reduction factors were applied. The total fire-induced core damage risk was determined by aggregating the CDF associated with each subscenario in the detailed analysis and the scenarios that were quantitatively screened from the spatial interactions analysis. The frequency of fire hazard-initiated core damage sequences was used as a measure of plant vulnerabilities.

The spatial interactions analysis phase and the detailed analysis phase of the fire IPEEE methodology utilized for both Plant Hatch units consists of the following steps:

Spatial Interactions Analysis Phase

- Step 1 Information Gathering and Data Collection
- Step 2 Identification of Important Plant Locations and Qualitative Screening
- Step 3 Development of Location Scenarios
- Step 4 Scenario Occurrence Frequency Assessment
- Step 5 Quantitative Screening of Location Scenarios

Detailed Analysis Phase

- Step 6 Development and Analysis of Subscenarios

The results of the detailed analysis phase were used as inputs for performing an analysis of the decay heat removal system, an uncertainty analysis and an analysis of the containment performance. The containment performance in response to fire-induced initiators was analyzed by utilizing the IPE analysis to determine the release frequencies and conditional containment failure probability for modes involving containment isolation failures, bypass failures and overpressurization failures. The following steps encompass the remaining portion of the Plant Hatch fire IPEEE methodology delineated above.

Step 7 Decay Heat Removal System Analysis
Step 8 Uncertainty Analysis
Step 9 Containment Performance Analysis

The Fire Risk Scoping Study (FRSS) issues and the NRC Safety Issue GI-57 were addressed as part of the Plant Hatch fire IPEEE.

Because of the similarity between Unit 1 and Unit 2, the analysis presented in the following section concentrated on the Unit 1 evaluation. Only the differences in the risk impacts between the two units are presented, even though a full-scope analysis was performed for Unit 2.

4.0.2 BASES AND ASSUMPTIONS

The bases and assumptions of the fire portion of the Plant Hatch IPEEE are as follows:

- The analysis addressed the risk of core damage and impact on containment performance induced by fire hazards in risk significant plant locations. The plant was initially assumed to be at full power.

- The plant design and operation information used in the fire IPEEE analysis reflect the plant configuration as of May 1995. The component failure and maintenance data used were identical to the data in the original IPE submittal. The plant-specific fire incident and fire drill response data were collected as of January 1994.

- The plant-specific data collected and analyzed reflect the plant configuration as of January 1995.

- The cable routing reflects plant design and proposed cable routing for Thermo-Lag® resolution as of May 1995.

- The fire portion of the IPEEE considered fire events initiated by faults from electrical and mechanical plant components, and personnel activities. Fire events initiated by sabotage or arson were not considered.

- The fires in Unit 2 locations were assumed not to interfere with Unit 1 operations except in areas common to both units unless Unit 1 accident mitigation equipment or cables exist in the Unit 2 fire zone.

- A plant location is defined as a designated plant space which can be a compartment, a collection of compartments, or a well-defined open space.

- Consistent with the Edwin I. Hatch Fire Hazards Analysis and Fire Protection Program (FHA) (Reference 4), a fire area is defined as a location surrounded by a 3-hour rated fire barrier (door, wall, and penetration seal).

- Plant locations were evaluated at the fire zone level. A fire zone is a plant location subdivided within a fire area.

- The location and amount of in-situ and transient combustibles affecting each location scenario were identified during the FHA review and plant walkdowns, respectively.

- The plant model used to calculate CDF was based on the full internal events IPE model , using RISKMAN® software (Reference 5). The internal events-induced CDF of 2.1E-08 per year and 2.2E-08 per year were used as the base case values for Unit 1 and Unit 2, respectively.

- The fire frequency associated with each key plant component category as defined in section 4.1.1 was calculated by combining nuclear industry experience (Reference 6) and Plant Hatch-specific data using a two-stage Bayesian data update technique (Reference 7). Separate analyses were performed for Unit 1 and Unit 2.

- The computer program COMPBRN IIIe (Reference 8) was used for calculating the fire-induced damage time of critical components given a particular fire size and duration.

- Automatic fire suppression systems were assumed to be installed per design specifications.

- Fire detectors were assumed to be capable of detecting fire signatures at their location per design specifications. Automatic fire suppression systems were assumed to be sized to effectively mitigate a fire of the maximum postulated size at that location. The fire protection systems were assumed to be maintained regularly in accordance with plant procedures.

- Fire drill records were collected to obtain information relating to fire brigade response time.

4.0.3 LAYOUT OF FIRE IPEEE REPORT

The layout of the fire portion of the Plant Hatch IPEEE report is consistent with table C.1 of NUREG-1407; however, to present the report in the suggested documentation structure, the nine steps stated in section 4.0.2 were tailored to fit the NUREG format. The following table correlates the IPEEE methodology steps to the NUREG format.

NUREG Format	IPEEE Methodology Step
4.1 Fire Hazard Analysis	1 through 5
4.2 Review of Plant Information and Walkdown	1 through 9
4.3 Fire Growth and Propagation	3 through 6
4.4 Evaluation of Component Fragilities and Failure Response	5 and 6
4.5 Fire Detection and Suppression	3 and 6
4.6 Analysis of Plant Systems, Sequences, and Plant Response	6 and 8
4.7 Analysis of Containment Performance	9
4.8 Treatment of Fire Risk Scoping Study Issues	1 through 9
4.9 USI A-45 and Other Safety Issues	7

The results of the spatial interactions analysis are presented in section 4.1, and the results of the detailed analysis including the uncertainty analysis are presented in section 4.6.

The documentation of plant information collected and walkdown observations, the fire event data analysis, the scenario frequency calculation, fire subscenario calculations supporting the detailed analysis, relevant plant records, detailed analysis worksheets, and other supporting analyses are included in the tier 2 documentation.

REFERENCES

1. U. S. Nuclear Regulatory Commission Generic Letter No. 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," June 1991.

2. U. S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, June 1991.

3. Edwin I. Hatch Nuclear Plant Units 1 and 2 Individual Plant Examination, December 1992.

4. Edwin I. Hatch Nuclear Plant, Units 1 and 2 Fire Hazards Analysis and Fire Protection Program, revision 10C, April 1995.

5. PLG, Inc., "RISKMAN® PSA Workstation Software," Version 5, 1994.

6. PLG, Inc., "Database for Probabilistic Risk Assessment of Light Water Reactor Power Plants," PLG-0500, Vol. 8, May 1994.

7. Kaplan, S., "On a 'Two-Stage' Bayesian Procedure for Determining Failure Rates from Experiential Data," Institute of Electrical and Electronics Engineers Transactions on Power Apparatus and Systems, Vol. PAS-102, No. 1, January 1983.

8. Ho, V. S., et al., "COMPBRN IIIe—An Interactive Computer Code for Fire Risk Analysis," UCLA-ENG-9016, EPRI-NP-7282, May 1991.

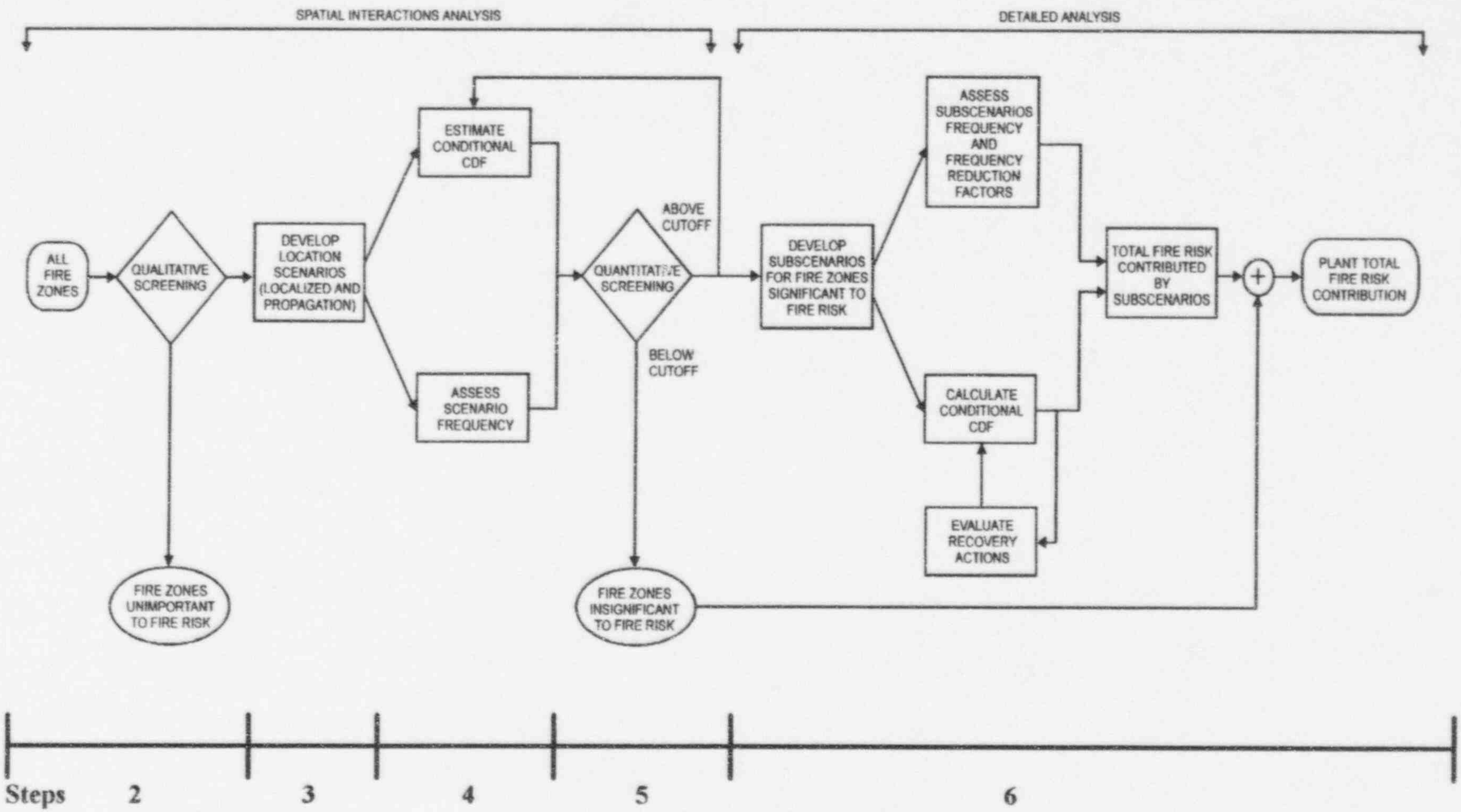


Figure 4.0-1 Overview of Technical Approach

4.1 FIRE HAZARD ANALYSIS

The presence of combustible loadings in a location may not necessarily induce significant risk to plant operation. An ignition source must co-exist with the combustible loadings, and the fire hazard must have the potential to initiate a plant transient or damage plant equipment required to mitigate a core damage accident. Thus, spatial interactions between fire hazards and plant safety components govern fire risk to plant operation. Section 4.1 discusses only the spatial interactions analysis (Steps 1 through 5 of the methodology presented in section 4.0). Section 4.6 presents a discussion of the detailed analysis. The objectives of the spatial interactions analysis performed for Plant Hatch were as follows:

- Collect relevant plant information to support the spatial interactions analysis.
- Perform qualitative screening to identify important locations for subsequent analysis.
- Estimate the fire hazard inventory in those locations.
- Identify fire protection capability in those locations.
- Evaluate the potential of fire propagation between plant locations.
- Develop location scenarios that evaluate the impact of fire hazards to safety-related plant equipment and cables.
- Quantitatively screen out, from detailed analyses, location scenarios that are relatively insignificant to plant risk induced by fire hazards.

The five steps comprising the spatial interactions analysis are discussed in detail below.

Step 1: Information Gathering and Data Collection

The following information sources were reviewed to familiarize analysts with Plant Hatch-specific details:

- Edwin I. Hatch Nuclear Plant Units 1 and 2 Fire Hazards Analysis and Fire Protection Program (FHA) (Reference 1).
- Site and plant general arrangement drawings.
- Combustible inventories (sources) in each fire zone.
- Edwin I. Hatch Nuclear Plant Individual Plant Examination (IPE) (Reference 2), including internal flood analysis.

- Relevant sections of Edwin I. Hatch Nuclear Plant Unit 1 and Unit 2 Final Safety Analysis Reports (Reference 3).

- Location of equipment and associated cables important to safety.

Relevant information was collected and summarized in a set of location characteristics tables (LCTs) for subsequent analyses of the important fire zones. To supplement plant documentation, site-specific walkdowns were completed. (See section 4.2 for further discussion.) Table 4.1-1 lists the fire zones defined and used in the subsequent analyses. The LCTs are retained as part of the tier 2 documentation. Figure 4.1-1 shows a typical LCT for Unit 1.

An essential source of information required for the spatial interactions analysis is the combustible loadings inventory which provides an estimate of the in-situ combustible loadings within each fire zone. The Plant Hatch FHA (Reference 1) provides a list of combustible materials and fire loadings for the plant that includes lube oil, cable insulation, charcoal, paper, and miscellaneous combustible materials within a fire zone. The FHA also provides an estimate of the fire severity in terms of fire duration (in hours), a list of fire detection and suppression capabilities, and the fire barrier rating. Fire duration cited in the FHA was obtained by dividing the fire loadings by the heat rate for a Standard Exposure Fire (80,000 Btu/ft^2-h) established by the National Fire Protection Association.

As previously stated, to initiate a fire risk, an ignition source must co-exist with combustible loadings. The following categories of plant components were considered as possible ignition sources:

- Batteries.

- Battery chargers.

- Electrical cabinets (includes logic cabinets, panels, relays, fuses, and switches).

- Electrical cables (includes power and control cables).

- Control room.

- Diesel generators.

- Generators (includes main generator).

- Heating, ventilation, and air conditioning (includes heaters, fans, chillers, and filters).

- Motors (includes motor-operated valves, motor-generator sets, and starter and strainer motors).

- Motor control centers.

- Pumps and air compressors (includes the motor unit of pumps).
- Switchgear (includes circuit breakers and buses).
- Turbines (includes driver, turbine-driven pump unit, and main turbine unit).
- Large transformers (above 4 kV)
- Transformers (4 kV and below)

In addition to the above categories of component fault-induced ignition sources, certain personnel activities, such as welding or cutting, may create an ignition source. Thus, human activities were also included as an additional ignition source category. Tables 4.1-4 and 4.1-5 provide a complete list of ignition source categories for Unit 1 and Unit 2 respectively.

The above component-induced and human-induced fire hazard ignition source categories were used in the assessment of fire initiation frequency (Step 4 below).

Step 2: Identification of Important Plant Locations and Qualitative Screening

This fire analysis used a top-down approach to evaluate locations within the controlled area of Plant Hatch. The analysis first examined the general layout drawings and general plant description provided in References 1, 2, and 3. The fire areas and fire zones identified in these references are defined as follows:

- A fire area is a plant location that is separated from other locations by a 3-hour-rated fire barrier (e.g., 3-hour-rated fire walls). Each fire area is identified by a unique 4-digit designator.
- A fire zone is a plant location subdivided within a fire area. A fire zone may consist of a compartment or several compartments, or it may be a fire area itself. A fire zone designator consists of two parts. The first part (a numeric prefix) denotes the fire area, and the second part (an alphabetic suffix) is an identifier for the fire zone. For example, fire zone 0007A denotes a location in fire area 0007, while fire zone 0007B denotes a second location in fire area 0007.

Using the fire area and fire zone definitions in the FHA, a qualitative screening was performed to screen out fire zones that do not represent a significant risk to the plant. A fire zone was considered to be potentially important if it satisfied either one of the following criteria:

- The fire zone contained fire-susceptible safety-related plant equipment or its associated power, control, and instrumentation cables, whose damage could cause an initiating event or can interfere with the plant's ability to mitigate accidents. However, if the initiating event frequency due to other causes as calculated in the IPE are much higher (in orders of magnitude) than the fire frequency of the fire zone (estimated at most to be on the order of 10^{-2} per year), the fire zone was screened from further analysis.

- The fire zone contains a sufficient amount of combustible loadings that, if ignited, could result in a fire that could potentially damage safety-related plant equipment or propagate to a fire zone that contains fire-susceptible safety-related plant equipment.

A limited number of fire zones were also screened from further analysis based upon walkdown observations, using the following criteria:

- The absence of a credible fire source or

- The spatial separation between fire sources, combustible loadings, and safety-related components is large and the interactions between them is not credible.

Table 4.1-1 summarizes the results of the qualitative screening analyses for Unit 1 and Unit 2.

Step 3: Development of Location Scenarios

In this step of the spatial interactions analysis, fire zones that were retained from the qualitative screening were evaluated further, prior to quantitative screening. Certain fire zones retained from the qualitative screening were grouped together in the subsequent analyses, based on the following criteria:

- Fires can propagate freely within the grouped fire zones.

- The fire zones contain equipment whose failure causes similar plant impacts.

- The fire zones are protected by similar fire protection capability.

- The fire hazard contents are similar in those fire zones.

- A physical barrier separates the grouped fire zones from the rest of the plant areas, and there is a significant time delay for fires to propagate from the grouped fire zones to other fire zones.

To systematically investigate the risk to plant operations induced by fire growth within a fire zone (grouped fire zones) and fire propagation between fire zones and/or grouped fire zones, one or more scenarios were developed for each fire zone (grouped fire zones) that survived the qualitative screening. Conservative estimates of fire occurrence frequency and impact of plant equipment damage for each scenario were assessed.

Scenarios developed in this step are generally defined at the fire zone level (or for grouped fire zones). Regardless of the initial fire severity, the amount of combustibles available, spatial separation of the fire sources and targets, or the size of the fire zones, the scenarios initially assumed that any fire occurring within the fire zone would damage all plant components and raceways within that zone. Two types of scenarios for each fire zone were developed:

Localized Scenario

A localized scenario describes a fire originated within a fire zone and allowed to spread within the zone. The fire can be initiated from any of the ignition sources within the fire zone, but the fire is assumed to be contained within the fire zone.

Propagation Scenario

A propagation scenario describes a fire originating in one fire zone and subsequently propagating to, and possibly beyond, adjacent fire zone(s). Similar to the localized scenario, all plant components and raceways within the affected fire zones are conservatively assumed to be damaged. A propagation scenario developed for a fire zone usually results in a more severe plant impact than a localized scenario because more plant systems are affected. However, a propagation scenario requires a higher fire severity and a longer fire growth time and, therefore, has a lower occurrence frequency than a localized scenario for the same fire zone.

Fire propagation pathways can be doors, permanent openings, or walls separating the fire zones. In order to identify the propagation pathways between fire zones, a fire zone adjacency matrix was developed. The matrix, based on the layout of the plant, illustrates the connections and potential fire propagation pathways between fire zones at Plant Hatch. The fire zone adjacency matrix was developed by examining layout and arrangement drawings. The fire zone adjacency matrix information for each fire zone is included in the LCT of the corresponding fire area.

A propagation pathway between adjacent fire zones was assumed to be credible only if one of the following criteria is satisfied:

- There is a permanent opening between fire zones or

- The fire severity (duration) of the combustible contents (according to the FHA) in the originating fire zone is > 75 percent of the barrier rating; e.g., door, wall, separating the originating fire zone and its adjacent fire zones, and there is no automatic suppression system in the originating fire zone or in the adjacent fire zone. (This criterion also takes into consideration the failure of fire barriers; e.g., fire door being left open.)

Credible propagation pathways between fire zones were identified by tracing the plant layout drawings and referencing the information contained in the LCTs. It was assumed that fires could propagate in multiple directions. For example, if an originating fire zone has six adjacent fire zones and credible pathways exist between the originating fire zone and four of the adjacent fire zones, the propagation scenario developed for the originating fire zone will affect the originating fire zone and the four adjacent fire zones regardless of the orientation of the fire zones. A fire was allowed to propagate from an originating fire zone to adjacent fire zones and from these zones, continue propagating as long as one of the above criteria was met between adjacent fire zones.

The localized and propagation scenarios developed for the Unit 1 and Unit 2 fire zones that were retained from the qualitative screening are listed in tables 4.1-2 and 4.1-3, respectively. Each scenario is identified by a unique designator. Each designator consists of two parts. The first part denotes the fire zone initiating the fire, and the second part denotes the type of scenario. An "L" stands for a localized scenario, and a "P" stands for a propagation scenario. For example, scenario 0007A-L represents a localized scenario developed for the fire zone 0007A. If more than one propagation scenario was developed from a fire zone, a numeric identifier was added after the letter "P" to distinguish the scenarios.

Step 4: Scenario Occurrence Frequency Assessment

In the spatial interactions phase, location-based scenarios developed for a fire zone describe all possible fire events that can occur in the fire zone and conservatively assume that each fire event can damage all components within the fire zone. Thus, all fire initiators within a fire zone were accounted for. Since more than one component type that can initiate a fire may be found in a fire zone, the fire occurrence frequency for a fire zone accounted for the composite nature of the fire hazard. The primary fire occurrence frequency for each fire zone was assumed to be the sum of the component-based ignition frequency of the components found in the fire zone. (Secondary ignition of the components is accounted for in the detailed analysis phase discussed in section 4.6.)

To accurately reflect the:

- variety of component categories,
- actual inventory of components,
- number of in-situ fuel sources and transients, and
- type of personnel activities at the fire zone,

the component-based fire frequencies were apportioned to different plant locations.

The primary objective of the fire frequency apportionment was to develop a reasonable estimate of the fire ignition frequency, accounting for the actual configuration of equipment in each location. For example, the plant-specific mean fire ignition frequency for the component category "battery-related" was determined to be 2.04E-03 per year. This frequency represents the estimated fire frequency from all battery-contributed fire events throughout the unit and was systematically apportioned to plant areas containing batteries.

Fire frequencies for different ignition sources were assessed on a component-based approach using two-stage Bayesian techniques (Reference 4). Both generic nuclear industry experience and plant-specific experience were used in frequency assessment. Computer software RISKMAN® (Reference 5) was used in the calculation process. Tables 4.1-4 and 4.1-5 summarize the results of the component-based fire ignition frequency assessments for Unit 1 and Unit 2, respectively. The information in these tables is based on the PLG fire database (Reference 6).

After the apportioned component-based fire frequencies were obtained for all fire-related components within a location, the fire occurrence frequency, was obtained by adding together all apportioned component-based fire frequencies identified for that location.

For fire zones for which propagation scenario(s) were developed, the propagation scenario occurrence frequency was assumed to be equal to the localized scenario occurrence frequency. This is conservative, since the resulting sum of the localized scenario occurrence frequency and the propagation scenario(s) occurrence frequency developed for a fire zone is greater than (instead of equal to) the fire frequency apportioned to that fire zone. Tables 4.1-2 and 4.1-3 summarize the occurrence frequency assessed for each Unit 1 and Unit 2 scenario, respectively.

Step 5: Quantitative Screening of Location Scenarios

This section discusses the analysis of failure response in the spatial interactions quantitative screening process. The plant response to component failure (plant impact) was assessed by considering the effects of component and raceway failures on the IPE top events. Table 4.1-6 contains a list of top events. For cables damaged by fires, the associated component was treated as failed. The LCTs list the top events associated with components in that particular zone. The top events associated with each scenario were evaluated to determine whether a component in a particular zone (or group of zones) would completely, or partially fail the top event, or not impact the top event at all.

Based on the top event impact, the conditional CDF was estimated using a simplified core damage sequence model similar to event trees used to model transient initiating events such as reactor scram and turbine trip. Only a few conservative operator recovery actions were included in the simplified core damage sequence model for these conditional CDF calculations. Except for a few recovery actions, IPE split fractions (Reference 7) were used for top events that were not failed by fire damage. The risk impact of each scenario (i.e., the unconditional CDF) was subsequently obtained by multiplying the scenario occurrence frequency by the conditional CDF.

The localized and propagation scenarios were screened based on their unconditional CDF. A conservative screening value of 0.1 percent of the Plant Hatch IPE CDF for internal initiating events was used to ensure that scenarios screened from further analysis were not risk significant. This results in screening values of 2.1E-08 per year and 2.2E-08 per year for Unit 1 and Unit 2, respectively, and provides a reasonable threshold for retaining sequences for detailed analysis that can have a measurable impact on CDF.

Scenarios with an unconditional CDF below the screening value were screened from further analysis. The total unconditional CDF of the screened scenarios was retained, aggregated, and added back to the fire-induced CDF calculated during the detailed analysis. Tables 4.1-2 and 4.1-3 summarize the results of the spatial interactions quantitative screening analyses for Unit 1 and Unit 2, respectively.

The following are examples of quantitative screening:

Localized scenario 0007A-L was developed for Unit 1 fire zone 0007A (table 4.1-2). The scenario occurrence frequency calculated for 0007A-L was 7.92E-04 per year, and the conditional CDF was estimated to be 4.27E-05. The unconditional CDF calculated was 3.38E-08 per year, a value higher than the screening value of 2.10E-08 per year. Therefore, the scenario 0007A-L was retained for detailed analysis.

Localized scenario 0007A-L1 was developed for Unit 2 fire zone 0007A (table 4.1-3). The scenario occurrence frequency calculated for 0007A-L1 was 6.51E-04 per year, and the conditional CDF was estimated to be 1.58E-05. The unconditional CDF calculated was 1.03E-08 per year, a value lower than the screening value of 2.20E-08 per year. Therefore, scenario 0007A-L was not retained for detailed analysis.

The conditional CDFs included in tables 4.1-2 and 4.1-3 assumed all plant equipment and cables within the locations are damaged simultaneously. Thus, the conditional and unconditional CDFs are provided for quantitative screening only.

For the scenarios which survived quantitative screening, subscenarios were developed to account for the realistic risk impact of each ignition source and combustible loadings in each location prescribed in the scenario. The detailed analysis of these subscenarios is presented in section 4.6.

REFERENCES

1. Edwin I. Hatch Nuclear Plant, Units 1 and 2 Fire Hazards Analysis and Fire Protection Program, revision 10C, April 1995.

2. Edwin I. Hatch Nuclear Plant Units 1 and 2 Individual Plant Examination (IPE), December 1992.

3. Edwin I. Hatch Nuclear Plant, Unit 1 and Unit 2 Final Safety Analysis Reports, revision 13C, April 1995.

4. Kaplan, S., "On a 'Two-Stage' Bayesian Procedure for Determining Failure Rates from Experiential Data," Institute of Electrical and Electronics Engineers Transactions on Power Apparatus and Systems, Vol. PAS-102, No. 1, January 1983.

5. PLG, Inc., "RISKMAN® PSA Workstation Software," Version 5, 1994.

6. PLG, Inc., "Database for Probabilistic Risk Assessment of Light Water Reactor Power Plants," PLG-0500, Vol. 8, May 1994.

7. Edwin I. Hatch Nuclear Plant Units 1 and 2 Individual Plant Examination (IPE), revision 1b, June 1995.

Table 4.1-1 Summary of Spatial Interactions Analysis Qualitative Screening Results, Unit 1 and Unit 2 (Sheet 1 of 9)

Zone	Description	Bldg[1]	Elev	Unit 1		Unit 2	
				Qualitative Risk Impact[2]	Qualitative Screening Results	Qualitative Risk Impact[2]	Qualitative Screening Results
0001	Working Floor and Corridor, Annunciator Logic Cabinet	0CB	112	I	Retained	I	Retained
0002A	CB Stairwell	1CB	ALL	N	Screened	N	Screened
0002B	CB Freight Elevator	1CB	ALL	N	Screened	N	Screened
0007A	East Corridor	1CB	112	I	Retained	I	Retained
0007B	U1 Water Analysis Room	1CB	112	I	Retained	N	Screened
0007C	HP Cold Lab Storage Area	1CB	112	N	Screened	N	Screened
0007D	Respirator Room	1CB	112	N	Screened	N	Screened
0007E	HP Cold Lab Test Area	1CB	112	N	Screened	N	Screened
0007F	SCBA Room	1CB	112	N	Screened	N	Screened
0014A	RC Lab	1CB	130	N	Screened	I	Retained
0014B	HP Hallway	1CB	130	N	Screened	N	Screened
0014C	HP Area Storage	1CB	130	N	Screened	N	Screened
0014D	HP Reference Area	1CB	130	N	Screened	N	Screened
0014E	HP Receiving Area	1CB	130	N	Screened	N	Screened
0014F	Decontamination Room and Shower	1CB	130	N	Screened	N	Screened
0014G	HP Counting Room	1CB	130	N	Screened	I	Retained
0014H	Hot Lab	1CB	130	N	Screened	I	Retained
0014I	HP Foreman's Office	1CB	130	N	Screened	N	Screened
0014J	HP Office	1CB	130	N	Screened	N	Screened
0014K	North and South Corridor	1CB	130	I	Retained	I	Retained
0014L	HVAC Room (HP Filter Outside)	1CB	130	N	Screened	N	Screened
0014M	Men's Restroom	1CB	130	N	Screened	N	Screened
0014N	Ladies' Restroom	1CB	130	N	Screened	N	Screened
0024A	Cable Spreading Room Unit 1 and 2, and CO_2 Manual System	1CB	147	I	Retained	I	Retained
0024B	Computer Room	1CB	147	I	Retained	I	Retained
0024C	Control Room	0CB	164	I	Retained	I	Retained
0024D	Control Room Entryway	0CB	164	N	Screened	N	Screened
0025	CO_2 Tank Room	1CB	147	N	Screened	N	Screened
0028	Low Pressure Coolant Injection (LPCI) Inverter Room	1CB	147	I	Retained	I	Retained
0031	Control Room Roof - Filter D004A	0CB	180	I	Retained	I	Retained
0040	Vertical Cable Chase	0CB	130	I	Retained	I	Retained
0101A	Main Turbine Deck - Reactor Feed Pump (RFP) Oil Cond	1TB	164	S, F	Screened	S, F	Screened
0101B	Turbine Bearings Area Bearing #1 U1	1TB	164	I	Retained	N	Screened
0101C	Reactor Feed Pump A-U1	1TB	164	I	Retained	N	Screened

Table 4.1-1 Summary of Spatial Interactions Analysis Qualitative Screening Results, Unit 1 and Unit 2 (Sheet 2 of 9)

Zone	Description	Bldg[1]	Elev	Unit 1		Unit 2	
				Qualitative Risk Impact[2]	Qualitative Screening Results	Qualitative Risk Impact[2]	Qualitative Screening Results
0101D	Reactor Feed Pump B-U1	1TB	164	I	Retained	N	Screened
0101E	East Control Room Entryway	1TB	164	N	Screened	N	Screened
0101F	Control Room Break Area	1TB	164	N	Screened	N	Screened
0101G	Chart Storage Room and Hallway	1TB	164	N	Screened	N	Screened
0101H	Shift Clerk/OPS Supervisor Office	1TB	164	N	Screened	N	Screened
0101I	Turbine Area Bearing #1 - U2	2TB	164	N	Screened	N	Screened
0101J	Main Turbine Deck Area	2TB	164	N	Screened	S, F, N	Screened
0101K	Reactor Feed Pump A-U2	2TB	164	N	Screened	I	Retained
0101L	Reactor Feed Pump B-U2	2TB	164	N	Screened	I	Retained
0201A	Refueling Floor	1RX	228	I	Retained	N	Screened
0201B	Refueling Floor	2RX	228	I	Retained	N	Screened
0401	Diesel Building Hallway	DG	130	N	Screened	N	Screened
0501	Intake Structure	INT	111	I	Retained	I	Retained
0601A	Waste Gas Treatment Working Floors	WG	NA	N	Screened	N	Screened
0601B	U1 Waste Gas Charcoal Absorber	WG	NA	N	Screened	N	Screened
0601C	U2 Waste Gas Charcoal Absorber	WG	NA	N	Screened	N	Screened
0602	Main Stack	YD	NA	N	Screened	N	Screened
0603	Low Level Radwaste Facility	LWRW	NA	N	Screened	N	Screened
0702A	Fire Pump House-Water Pump Room	FP	130	N	Screened	N	Screened
0702B	Fire Pump House-West Fire Pump Room	FP	130	N	Screened	N	Screened
0703	Fire Pump House-Central Fire Pump Room	FP	130	N	Screened	N	Screened
0704	Fire Pump House-East Fire Pump Room	FP	130	N	Screened	N	Screened
0801	U1 and U2 Main 500-kV Auto Transformer	YD	130	N	Screened	LOSP, N	Screened
0802	500-kV Duvall Black - North, South, and Mid	YD	132	N	Screened	N	Screened
0803	Main Meteorological Tower	YD	NA	N	Screened	N	Screened
0804	Backup Meteorological Tower	YD	NA	N	Screened	N	Screened
0805	Chlorine Building	YD	130	N	Screened	N	Screened
0806	Technical Support Center (TSC) - Charcoal Filter	2SB	130	N	Screened	N	Screened
0807	Emergency Offsite Facility	SIM	NA	N	Screened	N	Screened
0808	Central Alarm Station (CAS) Building	CAS	130	N	Screened	N	Screened
1003	Oil Storage Tank Room	1CB	112	N	Screened	N	Screened
1004	Station Battery Room 1A	1CB	112	I	Retained	N	Screened
1005	Station Battery Room 1B	1CB	112	I	Retained	N	Screened
1008	U1 AC Inverter Room	1CB	112	S, I	Retained	N	Screened

Table 4.1-1 Summary of Spatial Interactions Analysis Qualitative Screening Results, Unit 1 and Unit 2 (Sheet 3 of 9)

Zone	Description	Bldg[1]	Elev	Unit 1		Unit 2	
				Qualitative Risk Impact[2]	Qualitative Screening Results	Qualitative Risk Impact[2]	Qualitative Screening Results
1009	Reactor Protection System (RPS) Battery S Room	1CB	112	S, F	Screened	N	Screened
1010	Reactor Protection System (RPS) Battery N Room	1CB	112	S, F	Screened	N	Screened
1013	Reactor Protection System (RPS) Motor-Generator Set Room	1CB	130	I	Retained	N	Screened
1015	Annunciator Room	1CB	130	I	Retained	N	Screened
1016	West 600V Switchgear (SWGR) Room 1C	1CB	130	I	Retained	N	Screened
1017	East 600V Switchgear (SWGR) Room 1D	1CB	130	I	Retained	U	Screened
1018	West DC Switchgear (SWGR) Room 1A	1CB	130	I	Retained	N	Screened
1019	Transformer Room	1CB	130	I	Retained	N	Screened
1020	East DC Switchgear (SWGR) Room 1B	1CB	130	I	Retained	U	Screened
1023	Oil Conditioning Room	1CB	130	I	Retained	N	Screened
1101A	Area Under Main Condenser	1TB	112	I, F	Screened	N	Screened
1101C	Condensate Pump Area	1TB	112	I	Retained	N	Screened
1101D	Steam Jet Air Ejectors (SJAE) Rooms	1TB	112	I	Retained	N	Screened
1101E	Vacuum Pump Room	1TB	112	I	Retained	N	Screened
1101F	Condensate Polishing Room	1TB	112	I	Retained	N	Screened
1101G	Reactor Building Closed Cooling Water (RBCCW) Room	1TB	112	I, F	Screened	N	Screened
1101H	East Corridor	1TB	112	I	Retained	N	Screened
1101I	West Cableway	1TB	112	I	Retained	N	Screened
1101J	Working Floor - Seal Oil Unit, and TB 130 Northwest Switchgear Area	1TB	130	I	Retained	N	Screened
1101K	Main Condenser Area	1TB	130	I	Retained	N	Screened
1101M	NE MCC Area Mezzanine	1TB	147	I	Retained	N	Screened
1101N	NW Switchgear Mezzanine	1TB	147	I	Retained	N	Screened
1102	Northwest Stairwell	1TB	ALL	N	Screened	N	Screened
1103	Northeast Stairwell	1TB	ALL	N	Screened	N	Screened
1104A	East Cableway	1TB	130	I	Retained	I	Retained
1104B	East Cableway Enclosure	1TB	130	I	Retained	I	Retained
1201	U1 Drywell and Torus	1RX	ALL	I, F	Screened	N	Screened
1203A	RB S Torus Chamber - East and West Water Curtain	1RX	87	I	Retained	N	Screened
1203B	RB SE Corner Room - RHR Pump	1RX	87	I	Retained	N	Screened
1203C	RB SW Corner Room - RCIC	1RX	87	I	Retained	N	Screened
1203F	Working Floor - Control Rod Drive (CRD) Area	1RX	130	I	Retained	N	Screened
1203I	RB Stairwell RB-158, 185, 203, and 228 7 Elevation	1RX	ALL	I	Retained	N	Screened
1203K	Working Floor South -Water Curtain	1RX	158	I	Retained	N	Screened
1205A	RB N Torus Chamber E and W Water Curtain	1RX	87	I	Retained	N	Screened

Table 4.1-1 Summary of Spatial Interactions Analysis Qualitative Screening Results, Unit 1 and Unit 2 (Sheet 4 of 9)

Zone	Description	Bldg[1]	Elev	Unit 1		Unit 2	
				Qualitative Risk Impact[2]	Qualitative Screening Results	Qualitative Risk Impact[2]	Qualitative Screening Results
1205B	RB NE Corner Room - RHR Pump	1RX	87	I	Retained	N	Screened
1205C	RB NW Corner Room	1RX	87	I	Retained	N	Screened
1205F	Working Floor - Control Rod Drive (CRD) Area and Water Curtain	1RX	130	I	Retained	N	Screened
1205I	Working Floor North - Water Curtain	1RX	158	I	Retained	N	Screened
1205L	Reactor Water Cleanup (RWCU) Heat Exchanger Room	1RX	158	I	Retained	N	Screened
1205M	Cleanup Phase Separators	1RX	158	N	Screened	N	Screened
1205N	HVAC Room N/S - Supp System No.1 and 2, and Filters	1RX	164	I	Retained	N	Screened
1205Q	Standby Gas Filter Room	1RX	164	I	Retained	N	Screened
1205R	Working Floor North	1RX	185	I, F	Screened	N	Screened
1205S	Working Floor SE Water Curtain	1RX	185	S, I, O	Screened	N	Screened
1205T	Filter Demin Room	1RX	185	N	Screened	N	Screened
1205U	SW Corridor	1RX	185	N	Screened	N	Screened
1205W	Room South of Spent Fuel Pit	1RX	185	N, O	Screened	N	Screened
1205X	Stack Monitoring Room	1RX	203	N, O	Screened	N	Screened
1205Y	Working Floor	1RX	203	I, F, O	Screened	N	Screened
1205Z	HPCI Pump Room - Ceiling	1RX	87	I, F, O	Screened	N	Screened
1210	Recirc Motor-Generator Set Room A	1RX	158	S, F	Screened	N	Screened
1211	East Recirc Motor-Generator Set B	1RX	158	S, F	Screened	N	Screened
1301A	W Condensate Phase Separator Room	1RW	108	N	Screened	N	Screened
1301B	E Condensate Phase Separator Room	1RW	108	N	Screened	N	Screened
1301C	Waste Sludge and Spent Resin Tank and Pump Rooms	1RW	108	N	Screened	N	Screened
1301D	Chemical Waste and Floor Drain Collection Tank Room	1RW	108	N	Screened	N	Screened
1301E	Waste Collector Tank Room	1RW	108	N	Screened	N	Screened
1301F	Waste Sludge and Sample Tank Room	1RW	108	N	Screened	N	Screened
1301G	Working Floor	1RW	108	N	Screened	N	Screened
1301H	Radwaste Control Room	1RW	132	N	Screened	N	Screened
1301I	Chemical Treatment Room	1RW	132	N	Screened	N	Screened
1301J	Working Floor	1RW	132	N	Screened	N	Screened
1301K	RW Exhaust Filter Room - Filter D005/6	1RW	144	N	Screened	N	Screened
1301L	Hopper B Room	1RW	144	N	Screened	N	Screened
1301M	Southeast RW Building	1RW	144	N	Screened	N	Screened
1301N	Hopper A Room	1RW	144	N	Screened	N	Screened
1301P	Working Floor	1RW	156	N	Screened	N	Screened
1301Q	Ventilation Room	1RW	156	N	Screened	N	Screened

Table 4.1-1 Summary of Spatial Interactions Analysis Qualitative Screening Results, Unit 1 and Unit 2 (Sheet 5 of 9)

Zone	Description	Bldg[1]	Elev	Unit 1		Unit 2	
				Qualitative Risk Impact[2]	Qualitative Screening Results	Qualitative Risk Impact[2]	Qualitative Screening Results
1301R	Centrifuge Room A	1RW	156	N	Screened	N	Screened
1301S	Centrifuge Room B	1RW	156	N	Screened	N	Screened
1302A	Concentrated Radwaste Pump Room	1RWA	108	N	Screened	N	Screened
1302B	Chemical Waste Tank Room	1RWA	108	N	Screened	N	Screened
1302C	Chemical Waste Sample Tank Room	1RWA	108	N	Screened	N	Screened
1302D	Concentrator Tank Room	1RWA	108	N	Screened	N	Screened
1302E	Chemical Waste Ntrl Pump Room and Dumbwaiter Hall	1RWA	108	N	Screened	N	Screened
1302F	Working Floor RW ADD-132	1RWA	132	N	Screened	N	Screened
1302G	Radwaste Concentrate Tank Room	1RWA	132	N	Screened	N	Screened
1302H	Floor Drain Sample Tank Room	1RWA	132	N	Screened	N	Screened
1302I	HVAC Room	1RWA	150	N	Screened	N	Screened
1302J	Working Floor RW ADD - Filter D007/8	1RWA	150	N	Screened	N	Screened
1302K	Floor Drain Demin Room	1RWA	150	N	Screened	N	Screened
1302L	Solidification Area	1RWA	132	N	Screened	N	Screened
1401	Day Tank Room 1C	1DG	130	I, U	Screened	I, U	Screened
1402	Battery Room 1C	1DG	130	I, U	Screened	I, U	Screened
1403	Diesel Generator Room 1C	1DG	130	I, U	Screened	N	Screened
1404	Switchgear (SWGR) Room 1G	1DG	130	I	Retained	U	Screened
1405	Day Tank Room 1B	1DG	130	I, U	Screened	U	Screened
1406	Battery Room	1DG	130	I, U	Screened	U	Screened
1407	Diesel Generator Room 1B	1DG	130	I, U	Screened	I, U	Screened
1408	Switchgear (SWGR) Room 1F	1DG	130	I	Retained	U	Screened
1409	Day Tank Room 1A	1DG	130	I, U	Screened	U	Screened
1410	Battery Room 1A	1DG	130	I, U	Screened	U	Screened
1411	Diesel Generator Room 1A	1DG	130	I, U	Screened	N	Screened
1412	Switchgear (SWGR) Room 1E	1DG	130	I	Retained	U	Screened
1601	U1 Service Water Valve Pit 1A	NA	NA	I, U	Screened	N	Screened
1602	U1 Service Water Valve Pit 1B	NA	NA	I, U	Screened	N	Screened
1603	U1 Condensate Storage Tank/Pump	CST	130	I, U	Screened	N	Screened
1604A	Railroad Airlock	1RX	130	N	Screened	N	Screened
1604B	U1 Nitrogen Storage Tank	YD	129	I	Retained	N	Screened
1604C	U1 Dressout Area	YD	NA	N	Screened	N	Screened
1605	U1 Circulating Water Pump Pit	YD	NA	N	Screened	N	Screened
1606	U1 Main Transformer, Service Trfm 1 A/B, Aux. Trfm 1C/D	YD	129	LOSP, L	Screened	N	Screened

Table 4.1-1 Summary of Spatial Interactions Analysis Qualitative Screening Results, Unit 1 and Unit 2 (Sheet 6 of 9)

Zone	Description	Bldg[1]	Elev	Unit 1		Unit 2	
				Qualitative Risk Impact[2]	Qualitative Screening Results	Qualitative Risk Impact[2]	Qualitative Screening Results
1608A	Offgas Recombiner Working Floors	YD	130	N	Screened	N	Screened
1608B	Offgas Preheater #1	YD	130	N	Screened	N	Screened
1608C	Offgas Condenser	YD	130	N	Screened	N	Screened
1608D	Offgas Preheater #2	YD	130	N	Screened	N	Screened
1609	RB-185 Roof Turbine Filter D004/5	1RX	185	N	Screened	N	Screened
1610	Diesel Fuel Oil Storage Tank 1A	YD	130	I, U	Screened	U	Screened
1611	Diesel Fuel Oil Storage Tank 1B	YD	130	I, U	Screened	U	Screened
1612	Diesel Fuel Oil Storage Tank 1C	YD	130	I, U	Screened	U	Screened
1801	U1 "A" Cooling Tower Switchgear (SWGR) Building 1A	YD	118	S, F	Screened	N	Screened
1802	U1 "B" Cooling Tower Switchgear (SWGR) Building 1B/E/F/G	YD	118	S, F	Screened	N	Screened
1803	U1 "C" Cooling Tower Switchgear (SWGR) Building 1C/J/K/L/M	YD	118	S, F	Screened	N	Screened
1804	U1 Frisker Building	YD	130	N	Screened	N	Screened
1805A	Instrument Calibration Room	YD	130	N	Screened	N	Screened
1805B	Respirator Decontamination Room	YD	130	N	Screened	N	Screened
2003	Oil Storage Tank Room	2CB	112	N	Screened	U	Screened
2004	Station Battery Room 2A	2CB	112	N	Screened	I	Retained
2005	Station Battery Room 2B	2CB	112	N	Screened	I	Retained
2006	U2 Water Analysis Room	2CB	112	N	Screened	I	Retained
2008	AC Inverter Unit 2	2CB	112	N	Screened	S, I, F	Screened
2009	Reactor Protection System (RPS) Battery N Room	2CB	112	N	Screened	S, F	Screened
2010	Reactor Protection System (RPS) Battery S Room	2CB	112	N	Screened	S, F	Screened
2013	Reactor Protection System (RPS) Motor-Generator Set Room	2CB	130	N	Screened	I	Retained
2014	U2 Switchgear Access Hallway	2CB	130	N	Screened	I	Retained
2015	Annunciator Room	2CB	130	N	Screened	I	Retained
2016	West 600V Switchgear (SWGR) Room 2C	2CB	130	I, U	Screened	I	Retained
2017	East 600V Switchgear (SWGR) Room 2D	2CB	130	I, U	Screened	I	Retained
2018	West DC Switchgear (SWGR) Room 2A	2CB	130	N	Screened	I	Retained
2019	Transformer Room	2CB	130	I, U	Screened	I	Retained
2020	East DC Switchgear (SWGR) Room 2B	2CB	130	N	Screened	I	Retained
2021	Switchgear Hallway Enclosure	2CB	130	N	Screened	I	Retained
2023	Oil Conditioner Room	2CB	130	I, U	Screened	I, P	Retained
2101A	Area Under Main Condenser	2TB	112	I	Retained	I	Retained
2101C	Condensate Pump Area	2TB	112	I	Retained	I	Retained
2101D	Steam Jet Air Ejectors (SJAE) Rooms	2TB	112	I	Retained	I	Retained

Table 4.1-1 Summary of Spatial Interactions Analysis Qualitative Screening Results, Unit 1 and Unit 2 (Sheet 7 of 9)

Zone	Description	Bldg[1]	Elev	Unit 1		Unit 2	
				Qualitative Risk Impact[2]	Qualitative Screening Results	Qualitative Risk Impact[2]	Qualitative Screening Results
2101E	Vacuum Pump Room	2TB	112	I	Retained	I	Retained
2101F	Condensate Polishing Room	2TB	112	I	Retained	I	Retained
2101G	U2 Offgas Recombiner	2TB	112	I	Retained	I	Retained
2101H	East Corridor	2TB	112	I	Retained	I	Retained
2101I	West Cableway	2TB	112	I	Retained	I	Retained
2101J	Working Floor H_2 Seal Oil Unit and TB 130 SW Switchgear Area	2TB	130	I	Retained	I	Retained
2101K	Main Condenser Area	2TB	130	I	Retained	I	Retained
2101M	SE Switchgear Mezzanine	2TB	147	I	Retained	I	Retained
2101N	SW Switchgear Mezzanine	2TB	147	I	Retained	I	Retained
2102	Southwest Stairwell	2TB	ALL	N	Screened	N	Screened
2103	Southeast Stairwell	2TB	ALL	N	Screened	N	Screened
2104A	East Cableway and Reactor Feed Pump (RFP) Oil Conditioner	2TB	130	I, U	Screened	I	Retained
2104B	East Cableway Enclosure	2TB	130	I, U	Screened	I	Retained
2201	U2 Drywell and Torus	2RX	ALL	N	Screened	I, C, F	Screened
2203A	RB N Torus Chamber E/W Water Curtain	2RX	87	N	Screened	I	Retained
2203B	RB NE Corner Room - RHR Pump	2RX	87	N	Screened	I	Retained
2203C	RB NW Corner Room - RCIC	2RX	87	N	Screened	I	Retained
2203F	Working Floor N Control Rod Drive (CRD) Area and E/W Curtain	2RX	130	N	Screened	I	Retained
2203I	RB Stairwell RB-158, 185, 203, and 228	2RX	ALL	N	Screened	N	Screened
2203K	Working Floor North - Water Curtain	2RX	158	N	Screened	I	Retained
2205A	RB S Torus Chamber E/W Water Curtain	2RX	87	N	Screened	I	Retained
2205B	RB SE Corner Room - RHR Pump Room	2RX	87	N	Screened	I	Retained
2205C	RB SW Corner Room - Control Rod Drive (CRD) Pump Room	2RX	87	N	Screened	I	Retained
2205F	Working Floor South - CRD Area and RB S 130 - E Water Curtain	2RX	130	N	Screened	I	Retained
2205H	Main Steam Pipe Chase - Partial	2RX	130	N	Screened	N	Screened
2205I	Working Floor South - Water Curtain	2RX	158	N	Screened	I	Retained
2205L	RWCU HX Room	2RX	158	N	Screened	I	Retained
2205M	Cleanup Phase Separator Room	2RX	158	N	Screened	N	Screened
2205N	Chiller Room	2RX	164	N	Screened	I	Retained
2205Q	Standby Gas East Filter D001A/B	2RX	185	N	Screened	N	Screened
2205R	Working Floor 185 South	2RX	185	N	Screened	I	Retained
2205S	Working Floor 185 Northeast No. 1/2	2RX	185	N	Screened	I	Retained
2205T	HVAC - RX Building Exhaust Filter D005	2RX	185	N	Screened	N	Screened
2205U	Southwest Corridor	2RX	185	N	Screened	N	Screened

Table 4.1-1 Summary of Spatial Interactions Analysis Qualitative Screening Results, Unit 1 and Unit 2 (Sheet 8 of 9)

Zone	Description	Bldg[1]	Elev	Unit 1		Unit 2	
				Qualitative Risk Impact[2]	Qualitative Screening Results	Qualitative Risk Impact[2]	Qualitative Screening Results
2205V	Exhaust Filter and Demin Room - TB Filter D004/5/7/8	2RX	203	N	Screened	N	Screened
2205W	Area North of Spent Fuel Pit	2RX	185	N	Screened	N	Screened
2205X	Stack Monitoring Room	2RX	203	N	Screened	N	Screened
2205Y	Working Floor	2RX	203	N	Screened	N	Screened
2205Z	HPCI Pump Room	2RX	87	N	Screened	N	Screened
2210	Recirc Motor-Generator Set Room A	2RX	158	N	Screened	N	Screened
2211	Recirc Motor-Generator Set Room B	2RX	158	N	Screened	N	Screened
2301A	Cond Sludge Mix Pump Room	2RW	103	N	Screened	N	Screened
2301B	Condensate Phase Separator	2RW	103	N	Screened	N	Screened
2301C	OHL Skimmer Room	2RW	103	N	Screened	N	Screened
2301D	Floor Drain and Chemical Waste Tank Room	2RW	103	N	Screened	N	Screened
2301E	Decon Solution Concentration Pump Tank Room	2RW	103	N	Screened	N	Screened
2301F	Chemical Waste Neutral Tank and Pump Room	2RW	103	N	Screened	N	Screened
2301G	Spent Resin Pump and Tank Room	2RW	103	N	Screened	N	Screened
2301H	Waste Sludge Phase Separator Room	2RW	103	N	Screened	N	Screened
2301I	Waste Surge Tank Room	2RW	103	N	Screened	N	Screened
2301J	Dry Waste Storage Area	2RW	132	N	Screened	N	Screened
2301K	HVAC Room - Filter D005/6	2RW	132	N	Screened	N	Screened
2301L	Solidification Area	2RW	132	N	Screened	N	Screened
2301M	Conveyor Area	2RW	132	N	Screened	N	Screened
2301N	Floor Drain and Waste Collector Area	2RW	132	N	Screened	N	Screened
2301P	Steam Generator Room	2RW	132	N	Screened	N	Screened
2301Q	Decon Solution Concentrate Room	2RW	132	N	Screened	N	Screened
2301R	CCW Heat Exchanger Room	2RW	148	N	Screened	N	Screened
2301S	Hopper A Room	2RW	148	N	Screened	N	Screened
2301T	Hopper B Room	2RW	148	N	Screened	N	Screened
2301U	Working Floor	2RW	164	N	Screened	N	Screened
2301V	Radwaste Control Room	2RW	164	N	Screened	N	Screened
2301W	Centrifuge Room A	2RW	164	N	Screened	N	Screened
2301X	Centrifuge Room B	2RW	164	N	Screened	N	Screened
2301Y	Chemical Treatment Area	2RW	164	N	Screened	N	Screened
2301Z	Building Supply Ventilation Room	2RW	178	N	Screened	N	Screened
2401	Day Tank Room 2A	2DG	130	N	Screened	U	Screened
2402	Battery Room 2A	2DG	130	N	Screened	I	Retained

Table 4.1-1 Summary of Spatial Interactions Analysis Qualitative Screening Results, Unit 1 and Unit 2 (Sheet 9 of 9)

Zone	Description	Bldg(1)	Elev	Unit 1		Unit 2	
				Qualitative Risk Impact(2)	Qualitative Screening Results	Qualitative Risk Impact(2)	Qualitative Screening Results
2403	Diesel Generator Room 2A	2DG	130	N	Screened	U	Screened
2404	Switchgear (SWGR) Room 2E	2DG	130	I, U	Screened	I, LOSP	Retained
2405	Day Tank Room 2C	2DG	130	N	Screened	U	Screened
2406	Battery Room 2C	2DG	130	N	Screened	I	Retained
2407	Diesel Generator Room 2C	2DG	130	N	Screened	U	Screened
2408	Switchgear (SWGR) Room 2F	2DG	130	I	Retained	I, LOSP	Retained
2409	Switchgear (SWGR) Room 2G	2DG	130	I	Retained	I, LOSP	Retained
2601	U2 Service Water Valve Pit 2A	YD	NA	N	Screened	I, U	Screened
2602	U2 Service Water Valve Pit 2B	YD	NA	N	Screened	I, U	Screened
2603	U2 Condensate Storage Tank/Pump	CST	130	N	Screened	I	Retained
2604	Hot Machine Shop and U2 N_2 Tank	YD	130	N	Screened	N	Screened
2605	U2 Circulating Water Pump Pit, Main Trfm, and Aux Trfm 2A/B	YD	NA	N	Screened	N	Screened
2607	U2 RW Dilution Valve Pit	YD	NA	N	Screened	N	Screened
2608	U2 Startup Aux Transformer 2C/D	YD	129	N	Screened	I, F	Screened
2610	Diesel Fuel Oil Storage Tank 2A	YD	130	N	Screened	U	Screened
2612	Diesel Fuel Oil Storage Tank 2C	YD	130	N	Screened	U	Screened
2801	U2 Cooling Tower Switchgear (SWGR) Building No. 4	YD	118	N	Screened	N	Screened
2802	U2 Cooling Tower Switchgear (SWGR) Building No. 5	YD	118	N	Screened	N	Screened
2803	U2 Cooling Tower Switchgear (SWGR) Building No. 6	YD	118	N	Screened	N	Screened
2804	U2 Turbine Building Back Entrance	NA	NA	N	Screened	N	Screened

Notes:

1. CB Control Building
   CAS Central Alarm Station Building
   CST Condensate Storage Tank
   DG Diesel Generator Building
   FP Fire Pump House
   INT Intake Structure
   LWRW Low-Level Radwaste Building
   NA Not Applicable
   RW Radwaste Building
   RWA Radwaste Building Addition
   RX Reactor Building
   SB Service Building
   SIM Simulator Building
   TB Turbine Building
   WG Waste Gas Building
   YD Yard

2. F Fire area/zone contains equipment whose failure would cause scram but with a relatively lower fire initiating event frequency than other IPE causes. The fire area/zone is screened from further analysis.
   I Fire area/zone contains key equipment (needed for accident mitigation) and/or cables associated with important mitigation equipment and/or safe shutdown equipment.
   L Fire area/zone is screened out based on low impact or fire-induced failure has a relatively lower frequency than other causes.
   LOSP Fire area/zone contains equipment whose failure would cause loss of off-site power.
   N Fire area/zone contains no important mitigation equipment that can cause scram and damage to accident mitigation components. The fire area/zone is screened from further analysis.
   O Fire area/zone is screened out based on walkdown observation.
   P Fire area/zone contains significant amount of combustibles that may propagate to other fire area/zones.
   S Fire area/zone contains equipment whose failure would cause scram.
   U Fire area/zone contains cables associated with Appendix R equipment whose failure would not cause a scram.

Table 4.1-2 Spatial Interactions Analysis Quantitative Screening Scenario Table, Unit 1
(Sheet 1 of 3)

Scenario Designator	Initiating Location[1]	Other Location(s) Affected	Location Scenario Frequency (1/yr)	Top Event(s) Affected	CCDF[2]	UCDF[2] (1/yr)	Retained for Detailed Analysis?
Location Scenarios Survived from Quantitative Screening							
0001-L	0001		1.12E-02	ARI, IA, CW, MSIV Closure, I1(1/2), I2(1/2), RCIC, CS-A, JS-A, HA, HB, QV	1.22E-01	1.36E-03	Yes
0007A-L	0007A		7.92E-04	RCIC, JS-A, S011, CS-A, HA, QT-A, HPCI(1/2 rm clrs), VA(1/2), VB(1/2)	4.27E-05	3.38E-08	Yes
0014K-L	0014K		2.40E-03	NB, BE, BF, BG, BC, S011, BD, S012, DGA, DGB, DGC, D1, D2, D3, I1, I2, SA, SB, ATTS-1, VA, VB, MSIV Closure, RCIC, HPCI(rm clrs), JS, CS, HA, HB, RA(A), RB, QV	2.08E-01	4.99E-04	Yes
0024A-L	0024A		1.00E-02	NB, BE, BF, BG, BC, S011, BD, S012, DGA, DGB, DGC, D1, D2, D3, I1, I2, SA, SB, ATTS-1, ATTS-2, L2, LC, NS, VA, VB, MSIV, SORV/DE, RCIC, HPCI, JS, CS, HA, HB, QT, RA, RB, QV	1.00E+00	1.00E-02	Yes
0024B-L	0024B		5.60E-04	BF, DGA, DGB, DGC, D1(A), D2, I1, I2, ATTS-1, L2, LC, NS, SORV, HPCI, RCIC, CS, JS, RA, RB, HA, HB, QT, QV	1.00E+00	5.60E-04	Yes
0024C-L	0024C		6.37E-02	NB, BE, BF, BG, BC, S011, BD, S012, DGA, DGB, DGC, SA, SB, ATTS, L2, LC, NS, D1, D2, D3, I1, I2, VA, VB, MSIV Closure, SORV, HPCI, RCIC, CS, JS, RA, RB, HA, HB, QT, QV	1.00E+00	6.37E-02	Yes
0031-L	0031		1.01E-03	D3, I1, I2, VM	5.28E-03	5.33E-06	Yes
0040-L	0040		7.47E-04	NB, I1(1/2), I2(1/2), VA(1/2), VB(1/2), MSIV Closure, HPCI(1/2 rm clrs), HA, QV	2.26E-02	1.69E-05	Yes
0501-L	0501		7.27E-03	D1, D2, D3, I1(1/2), I2(1/2), TB, HA, HB	8.68E-03	6.31E-05	Yes
1004-L	1004		8.81E-04	S011, SA, RCIC, JS-A	5.46E-04	4.81E-07	Yes
1013-L	1013		1.61E-03	S065, VA, VB, RPSA, RPSB, MSIV Closure, HPCI(rm clrs), JS-A, HB, QV	2.11E-02	3.39E-05	Yes
1015-L	1015		1.67E-03	S064, VA(1/2), VB(1/2), MSIV Closure, FW, FR, HPCI(1/2 rm clrs), JS-A, HA, QV	5.01E-04	8.37E-07	Yes
1016-L	1016		4.35E-03	BE, DGA, BC, S011, S064, SA, TB, VM(1/3AHUs), RCIC, JS-A	7.63E-04	3.32E-06	Yes
1017-L	1017		4.35E-03	RPSB, DGC, BD, S012, S065, SB, ATTS-2, VM(Partial), HPCI, JS-B	5.10E-05	2.22E-07	Yes
1018-L	1018		2.89E-03	SA, ATTS-1, RCIC, JS-A	1.56E-03	4.51E-06	Yes

Table 4.1-2 Spatial Interactions Analysis Quantitative Screening Scenario Table, Unit 1 (Sheet 2 of 3)

Scenario Designator	Initiating Location[1]	Other Location(s) Affected	Location Scenario Frequency (1/yr)	Top Event(s) Affected	CCDF[2]	UCDF[2] (1/yr)	Retained for Detailed Analysis?
1020-L	1020		1.75E-03	RPT, SB, ATTS-2, I1(1/2), I2(1/2), MSIV Closure, HPCI, JS-B, HB, QV	5.55E-04	9.70E-07	Yes
1101I-L	1101I		6.63E-03	Apply to All Zones: NB, TB, I1, I2, MSIV Closure	1.34E-04	8.85E-07	Yes
1101XX-L	1101XX		7.88E-03	CO, Apply to All Zones: NB, TB, I1, I2, MSIV Closure	1.34E-04	1.05E-06	Yes
1101XX-P	1101XX	1101H	7.88E-03	CO, Apply to All Zones: NB, TB, I1, I2, MSIV Closure	1.34E-04	1.05E-06	Yes
1101H-L	1101H		4.13E-03	Apply to All Zones: NB, TB, I1, I2, MSIV Closure	1.34E-04	5.52E-07	Yes
1101ZZ-L	1101ZZ		2.26E-02	RPST, RPT, FW, BV, Apply to All Zones: NB, TB, I1, I2, MSIV Closure	1.34E-04	3.01E-06	Yes
1104A-L	1104A		3.64E-03	S012, ATTS-2, L2-B, LC-B, NS-B, I1(1/2), I2(1/2), VA, VB, MSIV Closure, HPCI, RCIC, CS-B, JS-B, RB, HB, QT-B, QV	2.23E-02	8.10E-05	Yes
1104B-L	1104B		3.27E-04	S011, VA, VB, HPCI, RCIC, CS-A, JS, RA, HA, QT-A, QV	9.98E-04	3.26E-07	Yes
1203A-L	1203A		1.56E-03	VA, VB, HPCI(rm clrs), RCIC, RA, QT-A, QV, CI(1/2;1T48F310;1T48F311)	5.36E-05	8.37E-08	Yes
1203B-L	1203B		5.01E-03	VA, VB, HPCI, CS-A, JS-A, RA, HA, QT-A, RB(D), QV	7.13E-04	3.57E-06	Yes
1203B-P	1203B	1203A	5.01E-03	VA, VB, HPCI, RCIC, CS-A, JS-A, RA, HA, QT-A, RB(D), QV, CI(1/2;1T48F310;1T48F311)	7.17E-04	3.59E-06	Yes
1203C-L	1203C		2.43E-03	VA, VB, HPCI(rm clrs), RCIC, QV	4.22E-05	1.03E-07	Yes
1203F-L	1203F		1.11E-02	S011, L2-A, LC-A, NS-A, VA, VB, MSIV Closure, SORV, HPCI, RCIC, CS-A, JS, RA, HA, QT-A, RB, QV	1.49E-02	1.66E-04	Yes
1203K-L	1203K		1.95E-03	L2-A, LC-A, NS-A, RPSF(P), RPSI(P), RPSL(P), RPSM(P), ARI-B, SORV, CS-A, RA(A), RB, QV, DWSP-A	2.78E-02	5.42E-05	Yes
1205A-L	1205A		1.56E-03	VA, VB, HPCI, RCIC, JS-B, RB, QT-B, QV	4.28E-05	6.68E-08	Yes
1205B-L	1205B		5.01E-03	VA(1/2), VB, HPCI, CS-B, JS-B, RB, HB	3.36E-05	1.69E-07	Yes
1205B-P	1205B	1205A	5.01E-03	VA, VB, HPCI, RCIC, CS-B, JS-B, RB, HB, QT-B, QV	1.78E-04	8.94E-07	Yes
1205C-L	1205C		2.25E-03	VA, VB, RD, HPCI(rm clrs)	3.38E-05	7.60E-08	Yes
1205F-L	1205F		1.16E-02	S012, S022, L2-B, LC-B, NS-B, NA, VA, VB, ARI, MSIV Closure, RD, HPCI, CS-B, JS-B, RB, HB, QT-B, QV, DWSP-B	5.61E-04	6.53E-06	Yes
1205N-L	1205N		4.83E-03	CS-A, QV, DWSP-A	3.77E-05	1.82E-07	Yes
1404-L	1404		3.53E-03	[LOSP], BG, DGA, DGB, DGC, BD, DC, SB, D2, D3, TB, I1(1/2), I2(1/2), VA, VB, HPCI(rm clrs), CS, RA(A), RB, HB	7.63E-03	2.69E-05	Yes

Table 4.1-2 Spatial Interactions Analysis Quantitative Screening Scenario Table, Unit 1 (Sheet 3 of 3)

Scenario Designator	Initiating Location[1]	Other Location(s) Affected	Location Scenario Frequency (1/yr)	Top Event(s) Affected	CCDF[2]	UCDF[2] (1/yr)	Retained for Detailed Analysis?
1408-L	1408		3.86E-03	BF, DGA, DGB, DGC, BC, OC, OD, DB, SA, D1(C), D2(D), D3, I1, I2, VA, VB, ARI-B, HPCI(rm clrs), HA(C), RA(A), RB	5.93E-03	2.29E-05	Yes
1412-L	1412		3.53E-03	NB, BE, BF, DGA, DGB, DGC, BC, DA, SA, D1, D2, D3, TB, I1, I2, VA, VB, HPCI(rm clrs), CS, RA(A), HA, RB	6.48E-02	2.29E-04	Yes
2101XX-L	2101XX		4.30E-02	Apply to All Fire Zones in this Fire Area: I1, I2, NB	3.02E-05	1.30E-06	Yes
Location Scenarios Screened from Quantitative Screening							
0007B-L	0007B		8.68E-04	SB	2.17E-05	1.89E-08	No
0028-L	0028		2.33E-04	JS-A, JS-B	3.34E-06	7.77E-10	No
0101XX-L	0101XX		1.53E-02	[LOFW], FW	8.93E-07	1.37E-08	No
0201A-L	0201A		1.67E-03	None	3.17E-06	5.30E-09	No
0201B-L	0201B		1.67E-03	None	3.17E-06	5.30E-09	No
1005-L	1005		8.81E-04	SB	2.17E-05	1.91E-08	No
1008-L	1008		4.72E-04	FW, FR, HB, QV	3.95E-05	1.86E-08	No
1019-L	1019		8.50E-04	OC, OD, ATTS-1	5.51E-06	4.68E-09	No
1023-L	1023		4.65E-04	JS-A	3.21E-06	1.50E-09	No
1203I-L	1203I		6.98E-04	None	3.17E-06	2.22E-09	No
1205I-L	1205I		1.95E-03	RPSF(P), RPSI(P), RPSL(P), RPSM(P), L2-B, LC-B, NS-B, CS-B, RB(B), JS-B, QT-B, VA(1/2)	8.07E-06	1.57E-08	No
1205L-L	1205L		1.12E-04	SL(P), CS-B, QV	6.28E-05	7.02E-09	No
1205Q-L	1205Q		1.08E-03	None	3.17E-06	3.44E-09	No
1604B-L	1604B		4.65E-04	N2	3.23E-06	1.50E-09	No
2408-L	2408		2.96E-03	DGB, D3, BF	4.88E-06	1.45E-08	No
2409-L	2409		2.61E-03	DGB, D3, BF	4.88E-06	1.27E-08	No
Location Scenarios Screened Based on Walkdown Observation							
None							

Notes:

1. Room Groupings: 0007XX is 0007A, B, C, D, E, F; 0014XX is 0014A, B, C, D, E, G, H, I, J, N; 0014YY is 0014F, M; 0101XX is 0101A, B, C, D, J; 0101YY is 0101E, F, G, H; 0101ZZ is 0101I, K, L; 1101XX is 1101C, D, E, F; 1101ZZ is 1101J, K, M, N; 2101XX includes all zones in 2101.

2. Conditional core damage frequency (CDF) and unconditional CDF (UCDF) are based on estimates of top event(s) impacted by a fire zone and consideration of appropriate recovery actions and may not reflect realistic conditions and should not be used for any purposes other than the quantitative screening step.

Table 4.1-3 Spatial Interactions Analysis Quantitative Screening Scenario Table, Unit 2
(Sheet 1 of 4)

Scenario Designator	Initiating Location[1]	Other Location(s) Affected	Location Scenario Frequency (1/yr)	Top Events Affected	CCDF[2]	UCDF[2] (1/yr)	Retained for Detailed Analysis?
Location Scenarios Survived from Quantitative Screening							
0001-L	0001		6.97E-03	ATTS-1, BC, S011, I1, I2, IA, JS-A, RCIC	4.47E-05	3.12E-07	Yes
0014H-L	0014H		3.41E-04	DGA, HA, QV	5.00E-04	1.71E-07	Yes
0014K-L	0014K		1.20E-03	BC, BD, BE, BF, BG, DGA, DGC, D1, D2, D3, I1, I2, HPCI, RCIC, CS, JS, RA, RB, HA, HB, QV	1.00E+00	1.20E-03	Yes
0024A-L	0024A		1.18E-02	ATTS, BE, BG, DGA, DGB, DGC, BC, BD, S011, S012, D1, D2, I1, I2, VA(1/2), VB(1/2), SORV, HPCI, RCIC, CS, JS, RA, RB, HA, HB, QT, QV	1.00E+00	1.18E-02	Yes
0024B-L	0024B		5.19E-04	ATTS-1, SORV, DGB, HPCI, RCIC, CS-A, JS-A, HA(F068A)	4.82E-03	2.50E-06	Yes
0024C-L	0024C		2.13E-02	ATTS, BE, BG, DGA, DGB, DGC, BC, BD, S011, S012, D1, D2, I1, I2, VA, VB, SORV, HPCI, RCIC, CS, JS, RA, RB, HA, HB, QT, QV	1.00E+00	2.13E-02	Yes
0031-L	0031		9.02E-04	VM	5.28E-03	4.76E-06	Yes
0040-L	0040		8.28E-04	ATTS-1, DGA, BC, S011, I1, I2, RCIC, HA(F068A), HB(F068B), QV	1.00E+00	8.28E-04	Yes
0501-L	0501		4.21E-03	D1, D2, D3, HA, HB	8.68E-03	3.65E-05	Yes
2004-L	2004		8.30E-04	DGA, BC, S011, SA, ATTS-1, RCIC	5.72E-04	4.75E-07	Yes
2013-L	2013		3.72E-04	S065, HB, HPCI, QV	5.07E-04	1.89E-07	Yes
2014-L	2014		9.40E-04	DGC, BD, S012, SB, ATTS-2, HPCI, JS	4.09E-05	3.84E-08	Yes
2015-L	2015		6.30E-04	DGA, S064, HA, QV	5.00E-04	3.15E-07	Yes
2016-L	2016		2.62E-03	DGA, BC, S011, SA, ATTS-1, RCIC, JS	2.28E-03	5.98E-06	Yes
2017-L	2017		2.62E-03	DGC, BD, S012, SB, JS-B	1.31E-05	3.44E-08	Yes
2018-L	2018		1.94E-03	SA, ATTS-1, RCIC, JS-A	4.09E-04	7.95E-07	Yes
2020-L	2020		1.65E-03	SB, ATTS-2, HPCI, JS	1.45E-05	2.40E-08	Yes
2101-L	2101		4.30E-02	S011, I1, I2, RCIC	1.83E-05	7.86E-07	Yes
2101YY-P	2101YY	0007A, 2101A, 2101XX	1.01E-03	S011, TB, CO, FW, MC, RCIC, JS-A	1.00E+00	1.01E-03	Yes
2104-L	2104		4.27E-03	ATTS-2, S011, S012, I1, I2, VA(1/2), VB(1/2), SORV, HPCI, RCIC, CS, JS, RA, RB, HA(F068A), HB(F068B), QT-B, QV	1.00E+00	4.27E-03	Yes
2104-P	2104	2101ZZ	4.27E-03	ATTS-2, S011, S012, I1, I2, TB, VA(1/2), VB(1/2), SORV, CO, FW, MC, HPCI, RCIC, CS, JS, RA, RB, HA(F068A), HB(F068B), QT-B, QV	1.00E+00	4.27E-03	Yes
2203A-L	2203A		1.14E-03	VB(1/2), HPCI, RCIC, CS-A, RA, HA, QT-A, QV	7.03E-04	7.99E-07	Yes

Table 4.1-3 Spatial Interactions Analysis Quantitative Screening Scenario Table, Unit 2 (Sheet 2 of 4)

Scenario Designator	Initiating Location[1]	Other Location(s) Affected	Location Scenario Frequency (1/yr)	Top Events Affected	CCDF[2]	UCDF[3] (1/yr)	Retained for Detailed Analysis?
2203A-P	2203A	2203B, 2203F, 2205A	1.14E-03	ATTS-1, S011, D2(B), VA(1/2), VB(1/2), DE(9/11SRVs Failed), SORV, HPCI, RCIC, CS, JS, RA, RB, HA, HB, QT, QV	1.00E+00	1.14E-03	Yes
2203B-P	2203B	2203A, F	2.79E-03	ATTS-1, S011, D2(B), VB(1/2), DE(9/11SRVs Failed), SORV, HPCI, RCIC, CS-A, JS, RA, RB(B), HA, HB, QT, QV	1.00E+00	2.79E-03	Yes
2203C-L	2203C		1.79E-03	RCIC	1.21E-05	2.16E-08	No
2203C-P	2203C	2203F	1.79E-03	ATTS-1, S011, D2(B), DE(9/11 SRVs Failed), SORV, RCIC, CS-A, JS, RA, RB(B), HA(F068A), HB, QT-B, QV	1.00E+00	1.79E-03	Yes
2203F-L	2203F		7.43E-03	ATTS-1, S011, D2(B), DE(9/11SRVs Failed), SORV, RCIC, CS-A, JS, RA, RB(B), HA(F068A), HB, QT-B, QV	1.00E+00	7.43E-03	Yes
2203F-P	2203F	2203A, B, I, K, 2205F, H	7.43E-03	ATTS, S011, S012, D2(B), VA(1/2), VB(1/2), DE(9/11SRVs Failed), SORV, HPCI, RCIC, CS, JS, RA, RB, HA, HB, QT, QV	1.00E+00	7.43E-03	Yes
2203K-L	2203K		9.62E-04	ATTS-1, SORV, CS-A, QV	1.54E-03	1.48E-06	Yes
2203K-P	2203K	2203I, 2205I, L, S	9.62E-04	ATTS, SORV, CS, QV	1.00E+00	9.62E-04	Yes
2205A-L	2205A		1.14E-03	D2(B), VA(1/2), CS-B, HPCI, RB, HB, QT-B, QV	5.07E-04	5.77E-07	Yes
2205A-P	2205A	2203A, 2205B, C, H	1.14E-03	BG, D2(B), VA(1/2), VB(1/2), HPCI, RCIC, CS, RA, RB, HA, HB, QT, QV	1.00E+00	1.14E-03	Yes
2205B-L	2205B		2.79E-03	BG, D2(B), VA(1/2), HPCI, CS-B, RB, HB	1.31E-05	3.64E-08	Yes
2205B-P	2205B	2205A, F, Z	2.79E-03	ATTS-2, BG, S012, D2(B), VA(1/2), SORV, HPCI, CS-B, JS-B, RB, HB, QT-B, QV	1.00E+00	2.79E-03	Yes
2205C-P	2205C	2205A, F	1.86E-03	ATTS-2, S012, D2(B), VA(1/2), SORV, HPCI, CS-B, JS-B, RB, HB, QT-B, QV	1.00E+00	1.86E-03	Yes
2205F-L	2205F		7.43E-03	ATTS-2, S012, D2(B), VA(1/2), SORV, HPCI, CS-B, JS-B, RB, HB, QT-B, QV	2.14E-02	1.59E-04	Yes
2205F-P	2205F	2203F, 2205B, C, H, I	7.43E-03	ATTS, BG, S011, S012, D2(B), VA(1/2), DE(9/11 SRVs Failed), SORV, HPCI, RCIC, CS, JS, RA, RB, HA(F068A), HB, QT-B, QV	1.00E+00	7.43E-03	Yes
2205H-P	2205H	2203F, 2205F	3.88E-04	ATSS, SO11, S012, D2(B), VA(1/2), DE(9/11 SRVs Failed), SORV, HPCI, RCIC, CS, JS, RA, RB, HA(F068A), HB, QT-B, QV	1.00E+00	3.88E-04	Yes
2205I-L	2205I		9.62E-04	ATTS-2, SORV, CS-B, QV	1.80E-03	1.73E-06	Yes

Table 4.1-3 Spatial Interactions Analysis Quantitative Screening Scenario Table, Unit 2 (Sheet 3 of 4)

Scenario Designator	Initiating Location[1]	Other Location(s) Affected	Location Scenario Frequency (1/yr)	Top Events Affected	CCDF[2]	UCDF[2] (1/yr)	Retained for Detailed Analysis?
2205I-P	2205I	2203K, 2205F, L, M, R	9.62E-04	ATTS, S012, D2(B), VA(1/2), SORV, HPCI, CS, JS-B, RB, HB, QT-B, QV	1.00E+00	9.62E-04	Yes
2205L-P	2205L	2203K, 2205I, M	2.17E-05	ATTS, SORV, CS, QV	1.00E+00	2.17E-05	Yes
2205N-L	2205N		3.11E-03	S012, D2(B), HPCI, RCIC, JS-B, RB(B), HB, QT-B, QV	5.00E-04	1.56E-06	Yes
2205Q-P	2205Q	2205R	1.24E-03	QV	1.00E+00	1.24E-03	Yes
2205R-L	2205R		1.58E-03	QV	4.40E-05	6.93E-08	Yes
2205R-P	2205R	2205I, Q, S	1.58E-03	ATTS-2, SORV, CS-B, QV	1.00E+00	1.58E-03	Yes
2205S-L	2205S		1.25E-03	QV	4.40E-05	5.48E-08	Yes
2205T-P	2205T	2205R	1.03E-03	QV	1.00E+00	1.03E-03	Yes
2205Y-P	2205Y	2205R, S	1.58E-03	QV	1.00E+00	1.58E-03	Yes
2205Z-L	2205Z		2.80E-03	HPCI	9.56E-06	2.67E-08	Yes
2404-L	2404		2.61E-03	BE, BC, DGA, DA, D1, I1, I2, CS-A, RA(A), RB(D), HA	6.34E-03	1.65E-05	Yes
2408-L	2408		2.96E-03	BF, OC, OD, DGB, D1(C), D2(D), D3, RA(C), RB(D), HA(C)	5.57E-03	1.65E-05	Yes
2409-L	2409		2.61E-03	BG, BD, DGB, DGC, DC, D2, D3, I1, I2, CS-B, RA(C), RB(B), HB	6.46E-03	1.68E-05	Yes
Location Scenarios Screened from Quantitative Screening							
0007A-L	0007A		6.51E-04	S011, RCIC, JS-A	1.58E-05	1.03E-08	No
0014A-L	0014A		5.27E-04	S064	6.21E-06	3.27E-09	No
0014G-L	0014G		6.04E-04	DGC, BD, S012, ATTS-2, HPCI	1.31E-05	7.90E-09	No
0028-L	0028		1.32E-04	JS	3.30E-06	4.35E-10	No
1104-L	1104		1.89E-03	DGB, HPCI, JS-A	3.22E-06	6.07E-09	No
2005-L	2005		8.30E-04	SB	1.28E-05	1.07E-08	No
2006-L	2006		2.09E-04	SB	1.28E-05	2.69E-09	No
2019-L	2019		6.19E-04	JS-B	3.22E-06	1.99E-09	No
2021-L	2021		2.63E-04	ATTS-2, S064	1.31E-05	3.46E-09	No
2023-L	2023		2.63E-04	JS	3.30E-06	8.70E-10	No
2203B-L	2203B		2.79E-03	VB(1/2), CS-A, RA, HA	6.29E-06	1.75E-08	No
2205C-L	2205C		1.86E-03	None	3.22E-06	5.99E-09	No
2205L-L	2205L		2.17E-05	CS-B	3.22E-06	6.99E-11	No
2402-L	2402		2.25E-04	DA	1.72E-05	3.86E-09	No
2406-L	2406		3.02E-04	DGC, DC	1.44E-05	4.34E-09	No
2603-L	2603		5.27E-05	HPCI, RCIC Torus Suction MOVs (probably limit switches)	3.76E-06	1.98E-10	No

Table 4.1-3 Spatial Interactions Analysis Quantitative Screening Scenario Table, Unit 2 (Sheet 4 of 4)

Scenario Designator	Initiating Location(1)	Other Location(s) Affected	Location Scenario Frequency (1/yr)	Top Events Affected	CCDF(2)	UCDF(2) (1/yr)	Retained for Detailed Analysis?
Location Scenarios Screened Based on Walkdown Observation							
0007A-P	0007A	2101H	6.51E-04	No credible fire source to support damage due to fire propagation.	N/A	N/A	No
0014A-P	0014A	0014G, H	5.27E-04	No credible fire source to support damage due to fire propagation.	N/A	N/A	No
0014G-P	0014G	0014A	6.04E-04	No credible fire source to support damage due to fire propagation.	N/A	N/A	No
0014H-P	0014H	0014A	3.41E-04	No credible fire source to support damage due to fire propagation.	N/A	N/A	No
0014K-P	0014K	2014	1.20E-03	No credible fire source to support damage due to fire propagation.	N/A	N/A	No
2014-P	2014	0014K, 2021	9.40E-04	No credible fire source to support damage due to fire propagation.	N/A	N/A	No
2017-P	2017	2014, 2019, 2020	2.62E-03	No credible fire source to support damage due to fire propagation.	N/A	N/A	No
2019-P	2019	2014, 2016, 2017	6.19E-04	No credible fire source to support damage due to fire propagation.	N/A	N/A	No
2021-P	2021	2014	2.63E-04	No credible fire source to support damage due to fire propagation.	N/A	N/A	No

Notes:

1. Room Groupings: 2101YY is 2101F, H; 2101ZZ is 2101J, K, M, N.

2. Conditional core damage frequency (CDF) and unconditional CDF (UCDF) are based on estimates of top event(s) impacted by a fire zone and consideration of appropriate recovery actions may not reflect realistic conditions and should not be used for any purposes other than the quantitative screening step.

Table 4.1-4 Component-Based Fire Ignition Frequency, Unit 1

Item	Fire Category															
	Battery	Battery Charger	Control Room	Diesel Gener-ator	Gener-ator	HVAC	Logic Cabinet	MCC	Motor	Power and Control Cables	Pumps	Switch-gear	Trans-former above 4 kV	Trans-former 4 kV and Below	Turbine	Transient (Human Error)
Total Specialized Generic Events	3	3	3	52	13	5	24	11	4	19	29	23	29	6	14	17
Known Events	3	3	3	48	12	5	23	11	4	15	26	20	25	5	10	14
Unknown Events	0	0	0	4	1	0	1	0	0	4	3	3	4	1	4	3
Prior Mean	2.77E-03	3.41E-03	3.47E-03	5.21E-02	1.56E-02	5.33E-03	2.67E-02	1.17E-02	3.52E-03	2.13E-02	2.55E-02	2.25E-02	2.74E-02	5.74E-03	2.00E-02	1.63E-02
Prior 5th	5.21E-05	1.63E-05	6.21E-05	1.55E-03	1.27E-04	6.29E-05	4.03E-04	2.80E-04	7.51E-05	1.34E-04	2.02E-03	3.85E-04	1.49E-03	1.14E-04	2.30E-04	2.88E-04
Prior 50th	9.36E-04	4.70E-04	1.25E-03	1.82E-02	2.75E-03	1.38E-03	6.86E-03	4.04E-03	1.29E-03	3.68E-03	1.65E-02	6.41E-03	1.78E-02	1.79E-03	6.02E-03	6.45E-03
Prior 95th	1.15E-02	1.46E-02	1.50E-02	2.24E-01	5.14E-02	2.08E-02	9.74E-02	3.83E-02	1.53E-02	7.51E-02	7.18E-02	8.38E-02	7.06E-02	2.21E-02	6.43E-02	4.98E-02
Prior Range Factor	14.9	29.9	15.5	12.0	20.1	18.2	15.5	11.7	14.3	23.7	6.0	14.8	6.9	13.9	16.7	13.1
Unit 1 Event(s)	0	0	0	8	0	0	4	0	0	2	1	0	0	0	0	2
Unit 1 Period (Years) [1]	17.0	17.0	13.3	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0
Posterior Mean	2.04E-03	1.92E-03	2.44E-03	3.89E-01	6.34E-03	3.21E-03	1.69E-01	6.79E-03	2.50E-03	7.87E-02	3.02E-02	9.53E-03	1.63E-02	3.81E-03	8.64E-03	5.80E-02
Posterior 5th	5.61E-05	1.69E-05	6.34E-05	1.58E-01	1.14E-04	6.32E-05	5.61E-02	2.68E-04	7.41E-05	1.28E-02	6.19E-03	3.61E-04	1.10E-03	1.07E-04	2.19E-04	9.88E-03
Posterior 50th	8.91E-04	4.58E-04	1.04E-03	3.87E-01	2.05E-03	1.15E-03	1.64E-01	3.45E-03	1.07E-03	5.82E-02	2.00E-02	4.37E-03	1.44E-02	1.64E-03	4.24E-03	3.83E-02
Posterior 95th	8.22E-03	8.86E-03	9.80E-03	6.70E-01	2.17E-02	1.34E-02	2.91E-01	2.60E-02	9.98E-03	1.83E-01	7.24E-02	2.97E-02	4.40E-02	1.53E-02	2.48E-02	1.84E-01
Posterior Range Factor	12.1	22.9	12.4	2.1	13.8	14.6	2.3	9.8	11.6	3.8	3.4	9.1	6.3	12.0	10.6	4.3

Note:

1. Based on the duration included in the PLG Fire Data Base.

Table 4.1-5 Component-Based Fire Ignition Frequency, Unit 2

Item	Fire Category															
	Battery	Battery Charger	Control Room	Diesel Gener-ator	Gener-ator	HVAC	Logic Cabinet	MCC	Motor	Power and Control Cables	Pumps	Switch-gear	Trans-former Above 4 kV	Trans-former 4kV and Below	Turbine	Transient (Human Error)
Total Specialized Generic Events	3	3	3	53	13	5	27	11	5	19	30	23	29	6	13	18
Known Events	3	3	3	49	12	5	26	11	5	15	27	20	25	5	9	15
Unknown Events	0	0	0	4	1	0	1	0	0	4	3	3	4	1	4	3
Prior Mean	2.76E-03	3.46E-03	3.46E-03	5.13E-02	1.64E-02	5.37E-03	2.92E-02	1.17E-02	4.45E-03	2.18E-02	2.41E-02	2.23E-02	2.68E-02	5.88E-03	1.90E-02	1.86E-02
Prior 5th	5.19E-05	1.70E-05	6.19E-05	1.38E-03	1.19E-04	6.06E-05	3.89E-04	2.78E-04	7.64E-05	1.38E-04	2.16E-03	3.84E-04	1.65E-03	1.36E-04	1.75E-04	2.58E-04
Prior 50th	9.32E-04	4.83E-04	1.25E-03	1.80E-02	2.66E-03	1.34E-03	7.05E-03	4.03E-03	1.38E-03	3.74E-03	1.64E-02	6.29E-03	1.72E-02	1.93E-03	5.13E-03	6.17E-03
Prior 95th	1.14E-02	1.47E-02	1.50E-02	2.20E-01	6.06E-02	2.10E-02	1.40E-01	3.82E-02	1.88E-02	7.30E-02	6.78E-02	8.48E-02	7.14E-02	2.21E-02	6.18E-02	5.95E-02
Prior Range Factor	14.8	29.4	15.6	12.6	22.6	18.6	19.0	11.7	15.7	23.0	5.6	14.9	6.6	12.7	18.8	15.2
Unit 2 Event(s)	0	0	0	7	0	0	1	0	0	2	0	0	0	0	1	1
Unit 2 Period (Years)[1]	13.3	13.3	13.3	13.3	13.3	13.3	13.3	13.3	13.3	13.3	13.3	13.3	13.3	13.3	13.3	13.3
Posterior Mean	2.08E-03	2.02E-03	2.50E-03	4.16E-01	6.88E-03	3.33E-03	4.13E-02	7.10E-03	3.09E-03	9.50E-02	1.64E-02	1.05E-02	1.70E-02	4.24E-03	3.21E-02	3.12E-02
Posterior 5th	5.63E-05	1.87E-05	6.38E-05	1.58E-01	1.08E-03	6.10E-05	3.44E-03	2.73E-04	7.65E-05	1.31E-02	1.97E-03	3.73E-04	1.43E-03	1.48E-04	2.92E-03	3.24E-03
Posterior 50th	8.99E-04	4.79E-04	1.05E-03	4.13E-01	2.03E-03	1.12E-03	2.34E-02	3.56E-03	1.19E-03	6.94E-02	1.34E-02	4.76E-03	1.40E-02	1.84E-03	1.83E-02	1.90E-02
Posterior 95th	8.44E-03	9.28E-03	1.01E-02	7.25E-01	2.46E-02	1.41E-02	1.45E-01	2.77E-02	1.27E-02	2.20E-01	4.42E-02	3.61E-02	4.55E-02	1.63E-02	1.08E-01	1.06E-02
Posterior Range Factor	12.2	22.3	12.6	2.1	4.8	15.2	6.5	10.1	12.0	4.1	4.7	9.8	5.7	10.5	6.1	5.7

Note:

1. Based on the duration included in the PLG Fire Data Base.

Table 4.1-6 Fire Top Events
(Sheet 1 of 4)

THIS REPORT PRESENTS ALL TOP EVENTS SORTED ALPHABETICALLY. THE SYSTEM AND TOP EVENT CODES REFER TO THE CODES PRESENTED IN THE IPEEE FIRE ANALYSIS.

TOP EVENT	SYSTEM	TOP EVENT DESCRIPTION
ARI	ARI	ARI FAILS TO VENT SCRAM AIR HEADER
ARIA	ARI	ARI LOGIC INPUT CHANNEL A FAILS TO PROVIDE SIGNAL
ARIB	ARI	ARI LOGIC INPUT CHANNEL B FAILS TO PROVIDE SIGNAL
BC	BUSC	600-V BUS C AND ASSOCIATED MCCS FAIL TO REMAIN ENERGIZED
BCINIT		TOP EVENT FOR BUSC INITIATOR
BD	BUSD	600-V BUS D AND ASSOCIATED MCCS FAIL TO REMAIN AVAILABLE
BDINIT		TOP EVENT FOR BUSD INITIATOR
BE	BUSE	4KV EMERGENCY BUS E FAILS TO REMAIN ENERGIZED
BEINIT		TOP EVENT FOR LOBUSE INITIATOR
BF	BUSF	4KV EMERGENCY BUS F FAILS TO REMAIN ENERGIZED
BFINIT		TOP EVENT FOR LOBUSF INITIATOR
BG	BUSG	4KV EMERGENCY BUS G FAILS TO REMAIN ENERGIZED
BGINIT		TOP EVENT FOR LOBUSG INITIATOR
BIIT	MACROS	SUPPRESSION POOL TEMP EXCEEDS BORON INJECTION INIT TEMP
BM	BM	DIESEL 1B ALIGNED TO UNIT 1 PSW (S=NO, F=YES)
BV	BV	BYPASS VALVES FAIL TO OPEN FOR PRESSURE CONTROL
CFF	MACROS	ECCS SYSTEMS FAIL BEFORE CONTAINMENT FAILS (SUCCESS = CONTAINMENT FAILS FIRST)
CI	CI	CONTAINMENT NOT ISOLATED
CLASS	MACROS	CET ACCIDENT CLASS (MULTI-STATE TOP EVENT)
CNMT	CET	CONTAINMENT STATUS
CO	FW	CONDENSATE INITIALLY UNAVAILABLE FOR INJECTION
CS	CS	CORE SPRAY SYSTEM FAILS
CW	RBCCW	RBCCW NOT AVAILABLE
D1	PSW	PSW DIV I SUPPLY TO REACTOR BLDG UNAVAILABLE
D2	PSW	PSW DIV II SUPPLY TO REACTOR BLDG UNAVAILABLE
D3	SSWP	DIESEL 1B PSW SUPPLY FAILED OR INADEQUATE
DA	DA	DIESEL BATTERY A NOT AVAILABLE
DB	DB	DIESEL BATTERY B NOT AVAILABLE
DC	DC	DIESEL BATTERY C NOT AVAILABLE
DCREC	DCREC	DC POWER NOT RESTORED TO DEENERGIZED BUS
DE	DE	OPERATORS FAIL TO DEPRESSURIZE VESSEL BEFORE CORE DAMAGE OCCURS
DESC1	MACROS	CNMT PRESS INCREASES, MANUAL OPERATION OF SRVS PRECLUDED
DESC2	MACROS	EMERGENCY DEPRESSURIZATION OCCURS (OR BLOWDOWN VIA BREAK)
DESC3	MACROS	REACTOR STATUS
DESC4	MACROS	CONTAINMENT STATUS/FAILURE MODE
DESC5	MACROS	DEBRIS COOLING STATUS
DESC6	MACROS	TYPE OF RELEASE
DESC7	MACROS	TIMING OF RELEASE
DGA	DGS	DIESEL GENERATOR A FAILS TO SUPPLY BUS E

Table 4.1-6 Fire Top Events
(Sheet 2 of 4)

TOP EVENT	SYSTEM	TOP EVENT DESCRIPTION
DGB	DGS	DIESEL GENERATOR B FAILS TO SUPPLY BUS F
DGC	DGS	DIESEL GENERATOR C FAILS TO SUPPLY BUS G
DGS	DGS	DUMMY TOP EVENT FOR QUANTIFICATION OF DIESELS
DUR	MACROS	LOSP DURATION (MULTI-STATE TOP)
DV	PSW	PSW COMMON DISCHARGE VALVES FAIL TO REMAIN OPEN
DWSP	RHRDWS	DRYWELL SPRAYS UNAVAILABLE FOR FISSION PRODUCT SCRUBBING
DWTC	MACROS	DRYWELL TEMPERATURE CONTROL INADEQUATE - SPRAYS REQUIRED
FC	FC	OPERATOR FAILS TO CONTROL LEVEL DURING ATWS
FR	FW	FEEDWATER/CONDENSATE NOT RECOVERED BEFORE LEVEL 2 REACHED
FW	FW	FEEDWATER PUMPS NOT AVAILABLE OR TRIPPED FOLLOWING TRANSIENT
FWISO	CI	FEEDWATER LINE VALVES FAIL TO ISOLATE FEEDWATER LINE BREAK
GR	GR	OFFSITE GRID NOT RESTORED
HA	RHRSW	RHR SERVICE WATER PUMPS IN DIVISION I FAIL
HB	RHRSW	RHR SERVICE WATER PUMPS IN DIVISION II FAIL
HCU	HCU	CONTROL RODS FAIL TO INSERT GIVEN SCRAM SIGNAL (MECHANICAL PORTION OF RPS)
HI	HI	HIGH PRESSURE CORE COOLING INADEQUATE/NOT RECOVERED
HO	HO	OPERATOR FAILS TO TERMINATE FLOW AND CONTROL LEVEL NEAR TAF
HP	HPCI	DUMMY EVENT TOP FOR HPCI/RCIC
HPCI	HPCI	HPCI FAILS
HPISO	CI	HPCI STEAM LINE ISOLATION VALVES FAIL TO ISOLATE ON STEAM LINE BREAK
HR	HPCI	ATWS: HPCI FAILS TO RESTART FOLLOWING TERMINATION OF INJECTION TO LOWER WATER LE
HS	RHRSW	RHR SERVICE WATER BOTH DIVISIONS - DUMMY TOP EVENT
I1	PSW	1P41-F310A AND C FAIL TO ISOLATE TURBINE BLDG HEADER
I2	PSW	1P41-F310B AND D FAIL TO ISOLATE TURBINE BLDG HEADER
I3	PSW	F310A&D OR B&C FAIL TO ISOLATE TB HEADER AFTER 600-V BUS REALIGNMENT
IA	P52	INSTRUMENT AIR SYSTEM NOT AVAILABLE
IN1	HPCI	INJECTION FAILS TO REMAIN AVAILABLE WHEN REACTOR REPRESSURIZES
IN2	MACROS	INJECTION FAILS TO REMAIN AVAILABLE AFTER CONTAINMENT FAILURE
IS	PSW	TURBINE BLDG ISOLATION - DUMMY TOP EVENT
JS	RHRLPC	LPCI INJECTION PATHS FOR BOTH LOOPS UNAVAILABLE
KMCR	KMCR	MAIN CONTROL ROOM COOLING NOT RECOVERED/PURGE MODE NOT AVAILABLE
KRSDP	KRSDP	TRANSFER OF CONTROL TO REMOTE SHUTDOWN PANEL(S) NOT ACCOMPLISHED
L1OP	L1OP	STATUS OF OPERATOR ACTIONS FOLLOWING LOCA SIGNAL (MULTISTATE TOP)
LBE	DGS	DIESEL A/BUS E LOCA LOAD SHED AND RELOAD FAILS
LBF	DGS	DIESEL B/BUS F LOCA LOAD SHED AND RELOAD FAILS
LBG	DGS	DIESEL C/BUS G LOCA LOAD SHED AND RELOAD FAILS
LBS	DGS	DUMMY TOP EVENT FOR QUANTIFICATION OF DIESEL GENERATOR LOCA LOAD SHEDDING LOGIC
LC	LC	LPCI/CS INITIATION SIGNALS UNAVAILABLE
LLO	LLO	LOW PRESSURE INJECTION POST-VESSEL FAILURE NOT AVAILABLE
LO	LO	LOW PRESSURE INJECTION AND CONTROL INADEQUATE
LOCA	MACROS	LOCA SIGNAL (LEVEL 1 OR HIGH DRYWELL PRESSURE) GENERATED
MC	MC	MSIVS FAIL TO REMAIN OPEN/MAIN CONDENSER NOT AVAILABLE

Table 4.1-6 Fire Top Events
(Sheet 3 of 4)

TOP EVENT	SYSTEM	TOP EVENT DESCRIPTION
MS	MSIVS	MSIVS FAIL TO ISOLATE REACTOR
MSISO	MSIVS	MSIVS FAIL TO ISOLATE REACTOR ON STEAM LINE BREAK OUTSIDE CONTAINMENT
MT	RPS	MANUAL SCRAM NOT SUCCESSFUL
N2	P70	NITROGEN SUPPLY FROM TANK FAILS TO REMAIN AVAILABLE
NA	P51	NONINTERRUPTIBLE INSTRUMENT AIR SUPPLY NOT AVAILABLE
NAU2	No Sys	NONINTERRUPTIBLE INSTRUMENT AIR SUPPLY NOT AVAILABLE - UNIT 2
NB	NB	NORMAL BUSES FAIL TO TRANSFER TO START-UP TRANSFORMERS
NBREC	NBREC	NORMAL BUSES REMAIN DEENERGIZED
NS	NS	LPCI/CS LOW PRESSURE PERMISSIVE SIGNAL FAILS
NSREC	NS	LPCI/CS LOW PRESS PERMISSIVE NOT RECOVERED
02	02	OPERATOR FAILS TO SUPPLY N2 TO DW HEADER FROM EMERGENCY N2 HOOKUP STATION
0B	OPER	OPERATOR FAILS TO OVERRIDE MSIV CLOSURE ON LEVEL 1 GIVEN ATWS
OC	OCD	OPERATORS FAIL TO REALIGN 600-V BUS C TO 4KV BUS F
OD	OCD	OPERATORS FAIL TO REALIGN 600-V BUS D TO 4KV BUS F
OG	GRID	230KV SWITCHYARD FAILS TO REMAIN AVAILABLE FOLLOWING UNIT TRIP
OI	OI	OPERATORS FAIL TO CLOSE TB PSW ISO MOVS, HI DW PRESS ASSUMED
OL	OL	OPERATORS FAIL TO PROVIDE ADEQUATE LONG TERM HEAT REMOVAL
OS	OS	SLCS NOT INITIATED BEFORE BIT REACHED
OW	E11	DRYWELL COOLING NOT RESTORED AND DRYWELL SPRAYS UNAVAILABLE
PN	P70	DRYWELL PNEUMATIC HEADERS FAIL TO REMAIN AVAILABLE
PR	PR	PRESSURE RELIEF INADEQUATE - MEDIUM BREAK LOCA ASSUMED
PS	PSW	PSW PUMP MODELS FOR NON-LOSP CONDITIONS
QC	MC	MAIN CONDENSER UNAVAILABLE
QHR	QHR	CONTAINMENT HEAT REMOVAL NOT AVAILABLE AND NOT RECOVERED
QR	QR	HEAT REMOVAL NOT RECOVERED BEFORE ECCS OR CNMT FAILS
QS	RHRSDC	RHR SHUTDOWN COOLING PATH NOT AVAILABLE
QT	RHRSPC	TORUS COOLING NOT AVAILABLE
QV	CHV	CONTAINMENT VENT NOT AVAILABLE FOR HEAT REMOVAL
QVENT	QVENT	CONTAINMENT VENT STATUS
RA	RHRA	RHR LOOP A PUMP TRAINS FAIL
RB	RHRB	RHR LOOP B PUMP TRAINS FAIL
RCIC	RCIC	RCIC FAILS
RCISO	CI	RCIC STEAM LINE VALVES FAIL TO ISOLATE ON BREAK
RD	CRD	CRD PUMPS ARE UNAVAILABLE FOR INJECTION
RP	RP	RETURN TO POWER OPERATION NOT POSSIBLE OR FAILED
RPOP	MACROS	NORMAL COOLDOWN INITIATED (RETURN TO POWER OP FAILED)
RPS	RPS	AUTOMATIC SCRAM SIGNAL NOT AVAILABLE
RPSA	RPS	RPS BUS A UNAVAILABLE
RPSB	RPS	RPS BUS B UNAVAILABLE
RPSF	RPS	RPS SIGNAL FAILS GIVEN LOSS OF FEEDWATER EVENT
RPSM	RPS	RPS SIGNAL FAILS GIVEN MSIV CLOSURE EVENT
RPST	RPS	RPS SIGNAL FAILS GIVEN TURBINE TRIP EVENT

Table 4.1-6 Fire Top Events
(Sheet 4 of 4)

TOP EVENT	SYSTEM	TOP EVENT DESCRIPTION
RPT	RPS	ONE OR BOTH RECIRC PUMP TRIPS FAIL TO TRIP
RS	RHR	RHR PUMPS IN LOOPS A AND B - DUMMY TOP EVENT
RSREC	MISC	MOTOR COOLING FOR RHRSW PUMPS NOT AVAILABLE/NOT RESTORED
RWISO	CI	RWCU SUCTION ISOLATION VALVES FAIL TO ISOLATE ON BREAK
S001		TOP EVENT MODEL FOR INITIATING EVENT LODC
SA	SA	DC POWER FROM STATION BATTERY A NOT AVAILABLE
SB	SB	DC POWER FROM STATION BATTERY B NOT AVAILABLE
SL	SLCS	STANDBY LIQUID CONTROL SYSTEM FAILS TO INJECT
SORV	SORV	SRV RECLOSURE STATUS (MULTI STATE TOP EVENT)
SW	PSW	PSW UNAVAILABLE GIVEN LOSP (DUMMY TOP EVENT USED TO QUANTIFY CONDITIONAL SPLIT FRACTIONS FOR D1 AND D2)
SWREC	PSWR	PSW NOT RESTORED
TB	PSW	PSW TURBINE BUILDING HEADER FAILS TO REMAIN AVAILABLE
TINJ	MACROS	ATWS - TERMINATION OF HIGH PRESS INJECTION CALLED FOR
TOTAL2	CI	V-SEQUENCE QUANTIFICATION
U1	PSW	DIV I PSW NOT AVAILABLE AFTER AC POWER RESTORATION
U2	PSW	DIV II PSW NOT AVAILABLE AFTER AC POWER RESTORATION
UA	DGREC	DG A NOT RECOVERED BEFORE CORE DAMAGE
UB	DGREC	DG B NOT RECOVERED BEFORE CORE DAMAGE
UC	DGREC	DG C NOT RECOVERED BEFORE CORE DAMAGE
UOLOCA	DGS	HIGH DRYWELL PRESS OR LEVEL 1 RECEIVED ON OPPOSITE UNIT
V18	V18	OPERATORS FAIL TO VENT VIA 18" VENTS TO PREVENT LOCA SIGNAL ON LOSS OF COOLING
VA	VS	RHR/CS LOOP A ROOM COOLING UNAVAILABLE
VB	VS	RHR/CS LOOP B ROOM COOLING UNAVAILABLE
VC	T47	DRYWELL COOLING INADEQUATE TO PREVENT LOCA SIGNAL
VCU2	No Sys	DRYWELL COOLING INADEQUATE TO PREVENT LOCA SIGNAL - UNIT 2
VDPR	MACROS	RX REMAINS AT HIGH PRESSURE UNTIL VESSEL FAILURE
VI	MACROS	INTAKE STRUCTURE HVAC INADEQUATE
VINJ	VINJ	VESSEL INJECTION NOT AVAILABLE AFTER DEPRESSURIZATION
VM	Z41	MAIN CONTROL ROOM COOLING INADEQUATE
VOP	VOP	OPERATORS FAIL TO TRIP UNNEEDED PUMPS ON LOSS OF RHR/CS ROOM COOLING
VS	VS	RHR/CS ROOM COOLING, BOTH DIVISIONS - DUMMY TOP EVENT
XC	SUTC	START-UP TRANSFORMER C UNAVAILABLE OR FAILS TO ACCEPT BUS TRANSFER
XD	SUTD	START-UP TRANSFORMER D UNAVAILABLE OR FAILS TO ACCEPT BUS TRANSFER
Z1	MACROS	DUMMY TOP EVENT
Z5	MACROS	RHR/CS EVENT TREE NOT BYPASSED
Z6	MACROS	DUMMY TOP EVENT

LOCATION CHARACTERISTICS TABLE - UNIT 1

FIRE AREA: 1411

DESCRIPTION: DG ROOM 1A

NO. OF FIRE ZONE CONTAIN IN THIS FIRE AREA: 1

1. FIRE ZONE DESCRIPTION:

FIRE ZONE	FIRE ZONE DESCRIPTION	FLOOR AREA (FT2)	BLDG	ELEV. (FT)
1411	DG ROOM 1A	1598	1DG	130

2. FIRE/SMOKE HAZARDS:

FIRE ZONE	OIL/GREASE (LB)	CABLE (LB)	CLASS A (LB)	CHARCOAL (LB)	PLASTICS (LB)	MISC (LB)	MISC GASES (LB)	FIRE DURATION (HR)
1411	2316	3036	147	0	6	0	0	0.61

3. FIRE DETECTION/SUPPRESSION FEATURES:

FIRE ZONE	SUPPRESSION (TYPE)	HOSE STATIONS	PORTABLE EXTINGUISHERS	DETECTORS (TYPE)
1411	CO2 FLOOD (FULL COVERAGE)	NONE	CO2; DRY CHEMICAL	HEAT DETECTORS (FULL COVERAGE)

4. FIRE BARRIER/ADJACENCY SUMMARY:

FIRE ZONE	ADJACENT FIRE ZONE	BARRIER RATING	FIRE BARRIER DESCRIPTION
1411	0401	3 HR	EAST WALL OF DIESEL GENERATOR ROOM 1A
1411	0401	3 HR	FIRE DOOR ASSEMBLY DIESEL GENERATOR BUILDING EL 130 FT 0 IN.
1411	1410	3 HR	EAST WALL OF THE DIESEL GENERATOR BATTERY ROOM 1A
1411	2409	3 HR	FIRE DOOR ASSEMBLY DIESEL GENERATOR BUILDING EL 130 FT 0 IN.

5. TOP EVENT(S) AFFECTED BY FIRE/SMOKE HAZARDS-SUSCEPTIBLE KEY EQUIPMENT:

FIRE ZONE	TOP EVENT(s) AFFECTED	EQUIPMENT AFFECTED	BASIC EVENT DESCRIPTION
1411	DGA	1R43F015A	AIR START SOLENOID VALVE F015A FAILS TO OPEN ON DEMAND
1411	DGA	1R43F016A	AIR START SOLENOID VALVE F016A FAILS TO OPEN ON DEMAND
1411	DGA	1R43F017A	VENT VALVE F017A FAILS TO CLOSE ON DEMAND
1411	DGA	1R43S001A	DIESEL GENERATOR FAILS
1411	DGA	1X41_GTD_A	STANDBY FAN RELAY GTD OR CONTACTS FAIL
1411	DGA	1X41_TH2X_A	RELAY TH2X OR CONTACTS FAIL
1411	DGA	1X41C002A	DG A EMERG EXHAUST FAN (PRIMARY) FAILS TO START AND RUN
1411	DGA	1X41C002B	DG A EMERG EXHAUST FAN (STANDBY) FAILS TO START AND RUN
1411	DGA	1X41C003A	DG A NORMAL EXHAUST FAN FAILS TO OPERATE
1411	DGA	1X41C005A	DG A VENT SUPPLY LOUVERS FTO OR RELAY GLO FTE
1411	DGA	1X41GLOA	RELAY GLO FAILS TO ENERGIZE OR CONTACTS FAIL TO MAKE
1411	DGA	1X41N004A	DG A THERMOSTAT TH2 FAILS TO PROVIDE START SIGNAL ON RISING
1411	DGA	1X41N005A	DG A THERMOSTAT TH1 FAILS TO PROVIDE VENT START SIGNAL ON RI
1411	DGA	1X41N045A	FAILED FAN LOW FLOW SWITCH N045A FAILS TO OPERATE

6. TOP EVENT(S) AFFECTED BY RACEWAYS ASSOCIATED WITH KEY EQUIPMENT:

FIRE ZONE	TOP EVENT(s) AFFECTED	EQUIPMENT AFFECTED BY LOSS OF CIRCUIT (FIREMPL)
1411	BE	R22-S005
1411	DGA	R43-N013A, R43-N014A, R43-N015A, R43-N016A, R43-S001A, Y52-C001A, Y52-C101A
1411	N/A PWR REMOVED	P41-F403A

7. SUMMARY OF TOP EVENT AFFECTED:

FIRE ZONE	TOP EVENT(s) AFFECTED
1411	BE, DGA

8. WALKDOWN NOTES AND REMARKS (BLANK IF NOTHING NOTED):

Fire Zone 1411 The DG room is equipped with fire doors and fire dampers. There are a limited number cable trays in this room. The area underneath the portable fire extinguisher is labeled as no-combustible area. The DG jacket water surge tank is mounted on the wall in this DG room. There is a protective bracket installed around the tank to prevent the tank from hitting the vital DG equipment. Inside this diesel generator room, CO2 fire suppression is installed. If actuated, the room will be flooded with CO2 within 30 seconds. Equipment in this fire zone and cables going through this area are associated with equipment whose failure would not cause a scram.

Figure 4.1-1 Typical Location Characteristics Table, Unit 1

4.2 REVIEW OF PLANT INFORMATION AND PLANT WALKDOWN (STEP 1)

4.2.1 INFORMATION REVIEW

The information sources listed in section 4.1, Step 1, and the following information sources were utilized throughout the project:

- Cable routing data.
- Plant records of fire incidents.
- Individual Plant Examination plant model.
- Interfacing systems loss-of-coolant accident analysis.
- Fire drill records.
- Fire barrier inspection procedures and problem reports.
- Fire fighting procedures and fire preplans.
- Fire protection and detection maintenance procedures and inspection reports.

Typically, general information was collected first. If required, specific detailed information data was collected to support the analyses. Relevant information was stored in a relational spatial interactions analysis database. A set of location characteristics tables (LCTs) was developed to summarize the stored information in the relational database that is relevant to the spatial interactions analysis. Information contained in the LCTs was verified as appropriate during the walkdowns and continuously updated throughout the project.

4.2.2 PLANT WALKDOWN

Three plant walkdowns were performed in support of the fire IPEEE. The first walkdown was conducted in January 1994 at the beginning of the qualitative screening process. The second walkdown was conducted in April 1995 at the beginning of the detailed analysis phase for Unit 1. The third walkdown was conducted in May 1995 at the beginning of the detailed analysis phase for Unit 2. Each walkdown lasted for 3 to 4 days and was a joint effort performed by a team comprised of PLG and Southern Nuclear Operating Company engineers knowledgeable in the area of probabilistic risk assessment, and a Plant Hatch fire protection engineer.

The objectives of the first plant walkdown were to:

- Gain an appreciation for the spatial interactions of fire hazards and equipment.

- Confirm the information gathered in the LCTs.

- Inspect the amount and location of transient fire hazards.

- Verify potential propagation paths.

The walkdown team physically visited accessible fire zones in Unit 1 and Unit 2. The plant locations that were not visited included those that require special permits (e.g., radiation permit). Photographs, sketches, and notes were made to document complex configurations. In addition to confirming locations of the actual mechanical components (e.g., pumps, valves, electrical switchgear), the team evaluated the personnel activity level, in-situ and transient combustible content, and the applicability of generic fire event data. The walkdown notes are included in the LCTs. Figure 4.1-1 shows an example of an LCT.

The second and the third walkdowns concentrated on the fire zones that were retained after the quantitative screening (Step 4) and were accessible to the walkdown team. The objectives of the second and the third walkdowns were to:

- Confirm results of the quantitative screening process.

- Verify and further screen propagation pathways.

- Identify ignition sources and combustible loadings locations.

- Collect detailed spatial information relative to the safety-related plant equipment and cable raceways (cable trays and conduits).

- Inspect the amount and location of possible transient hazards.

- Develop subscenarios for the detailed analysis.

4.3 FIRE GROWTH AND PROPAGATION

Fire growth and propagation were considered in the spatial interactions analysis and in the detailed analysis phases. In the spatial interactions analysis, fires initiated from any of the fire sources within a fire zone were assumed to grow and propagate uninhibited, damaging all plant equipment and raceways within the fire zone. In the detailed analysis, fire growth and propagation were analyzed by considering the separation distance between the fire sources and the safety-related equipment within a fire zone, the heat release rate of the fire sources, and the duration of fire exposure. These attributes were characterized using a severity factor which is discussed in section 4.4.1 and the COMPBRN IIIe code (Reference 1).

The computer code COMPBRN IIIe was used to predict the heat release rate and the fire exposure duration of a given fire source for calculating the geometric factor (f_G, for transient fires) and the fire nonsuppression factor (f_{NS}). Sections 4.4 and 4.5 describe the quantification of the f_{NS}.

COMPBRN IIIe uses a quasi-static zonal approach to simulate the process of fire growth in an enclosure or an unconfined area. The enclosure is assumed to be divided into three distinct homogeneous, stable zones: the flame region, a hot gas layer, and an ambient air region. The hot gas layer includes the hot gases accumulating under the ceiling due to fire plume entrainment and buoyancy. Objects in this layer are subject to both convective and radiative heat fluxes from the hot gases. A region of ambient air is assumed to form underneath the hot gas layer. This layer is assumed to be thermally inert and contains relatively quiescent cool air, which remains at ambient conditions at all time.

The burning rate of a fire source is used to determine the heat output of the fire source, which can be transported to other objects in the vicinity via radiation. A plant component (e.g., safety-related cable) is considered to be damaged or ignited if the surface temperature exceeds the user-specified damage or ignition temperatures.

The physical models of the COMPBRN IIIe code have been reviewed extensively; e.g., References 3 and 4. In general, the COMPBRN IIIe code is designed with one objective: to predict the time to damage of equipment exposed to a relatively small pool fire in an enclosure for use in a probabilistic fire analysis. Due to the coarse nature of the models used in the COMPBRN IIIe code, the following limitations must be considered in applying the code:

1. The code does not perform well when the fire source is close to the ceiling, within the hot gas layer, or when a target is directly on top of a flame.

2. The code cannot model window openings. Only one doorway can be modeled, but the location of the doorway cannot be specified. The door is not allowed to open or close during the simulation period.

3. The ventilation ports can only be located at the ceiling or on the floor. The fraction of flow entering or leaving the port is specified by the users. Because ventilation to an affected area

will be cut off and the air dampers or door will be closed once a fire is detected, ventilation is normally not modeled.

4. All objects, including the enclosures and cable trays, are assumed to be rectangular in shape and must be oriented parallel to one of the axes.

5. Because the hot gas layer is assumed to be formed within the first time step, the code's predictions are expected to be most reasonable for fire scenarios involving large fuel loads during their pre-flashover burning period. For instance, COMPBRN IIIe should not be used to model fires within cabinets (in such cases, the enclosure would be too small for the fires).

6. Vertical or slanted burning objects cannot be modeled with accuracy.

7. The interference effect of objects in close proximity, such as cable trays piled together with a very small separation distance, cannot be modeled properly.

Some of the fire sources considered in this analysis do not satisfy the requirements of the COMPBRN IIIe models. Surrogate fire sources were then used to simulate the heat release rate and fire duration of the actual fire sources. For example, an oil pool fire of an appropriate pool diameter may be used to simulate a pump-related fire with similar heat release rate.

The uncertainties of COMPBRN IIIe are associated with the physical property data for combustibles and the model parameters for the physical models. The physical property data are used to define the behavior of the objects, and the model parameters are used to represent the parameters of the physical models. Table 4.3-1 lists the point value of the physical parameters using cable trays and oil pools. Table 4.3-2 lists the key physical model parameters. These values are selected from plant data and recommendations from the COMPBRN IIIe code.

REFERENCES

1. Ho, V. S., et al., "COMPBRN IIIe—An Interactive Computer Code for Fire Risk Analysis," UCLA-ENG-9016, EPRI-NP-7282, May 1991.

2. Siu, N. O., and Apostolakis, G., "A Methodology for Analyzing the Detection and Suppression of Fires in Nuclear Power Plants," Nuclear Science and Engineering, 94(1986)213-226.

3. Lambright, J. A., et al., "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," Sandia National Laboratories, prepared for U. S. Nuclear Regulatory Commission, NUREG/CR-5088, January 1989.

4. Ruger, C., Boccio, J. L., and Azarm, M. A., "Evaluation of Current Methodology Employed in Probabilistic Risk Assessment of Fire Events at Nuclear Power Plants," NUREG/CR-4229, May 1985.

Table 4.3-1 Physical Parameter Values Used in the COMPBRN IIIe Simulations

Property Parameter	Cable	Oil
Density (kg/m^3)	1,450	886
Specific Heat (J/kg-K)	2,250	1,884
Thermal Conductivity (W/m-K)	0.1	0.145
Heat Value (MJ/kg)	25.8	48.8
Piloted Ignition Temperature (°K)	761	269
Spontaneous Ignition Temperature (°K)	761	486
Damage Temperature (°K)	450	1.0
Ventilation Controlled Burning Rate Constant	0.11	0.11
Specific Burning Rate Constant (kg/m^2-K)	0.02	0.039
Surface Controlled Burning Rate Constant (kg/J)	2.90E-06	2.00E-06
Combustion Efficiency	0.5	0.9
Fraction of Flame Heat Released as Radiation	0.4	0.5
Absorption Coefficient for Flame Gases (m^{-1})	1.4	1.4
Reflectivity	0.12	0.1

Table 4.3-2 Representative Values for Key Modeling Parameters in the COMPBRN IIIe Simulations

Model Parameter	Representative Values[1]
Flame Heat Transfer Coefficient (W/m^2-K)	22
Hot Gas Layer Convective Heat Transfer Coefficient (W/m^2-K)	10
Coefficient of Inflow Air through Doorway	0.6
Coefficient of Outflow Air through Doorway	0.7
Absorption Coefficient of Hot Gases	1.3
Buoyant Plume Entrainment Coefficient (Unaffected Fires)	2.0
Buoyant Plume Entrainment Coefficient (Fires Next to a Wall)	1.5
Buoyant Plume Entrainment Coefficient (Fires Near a Corner)	1.25

Note:

1. The actual value for these parameters varies with the configuration of fire zone evaluated.

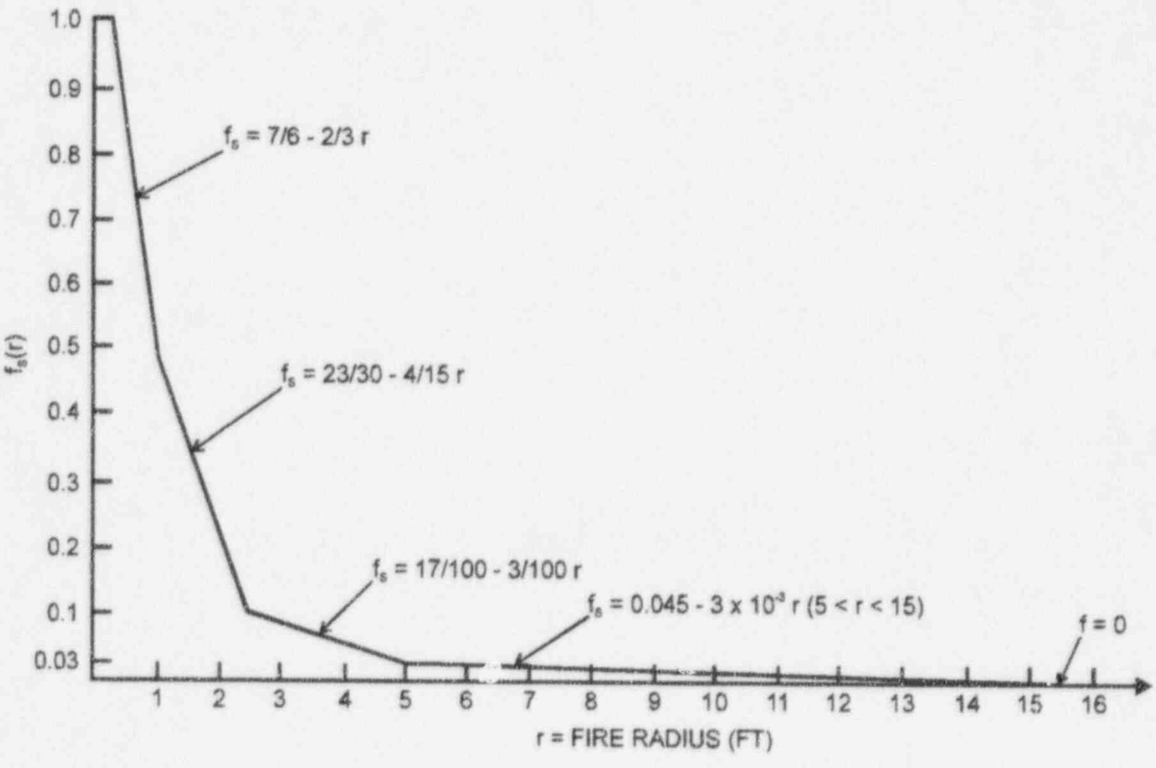


Figure 4.3-1 Fire Severity Curve for Control Room Panel Fires

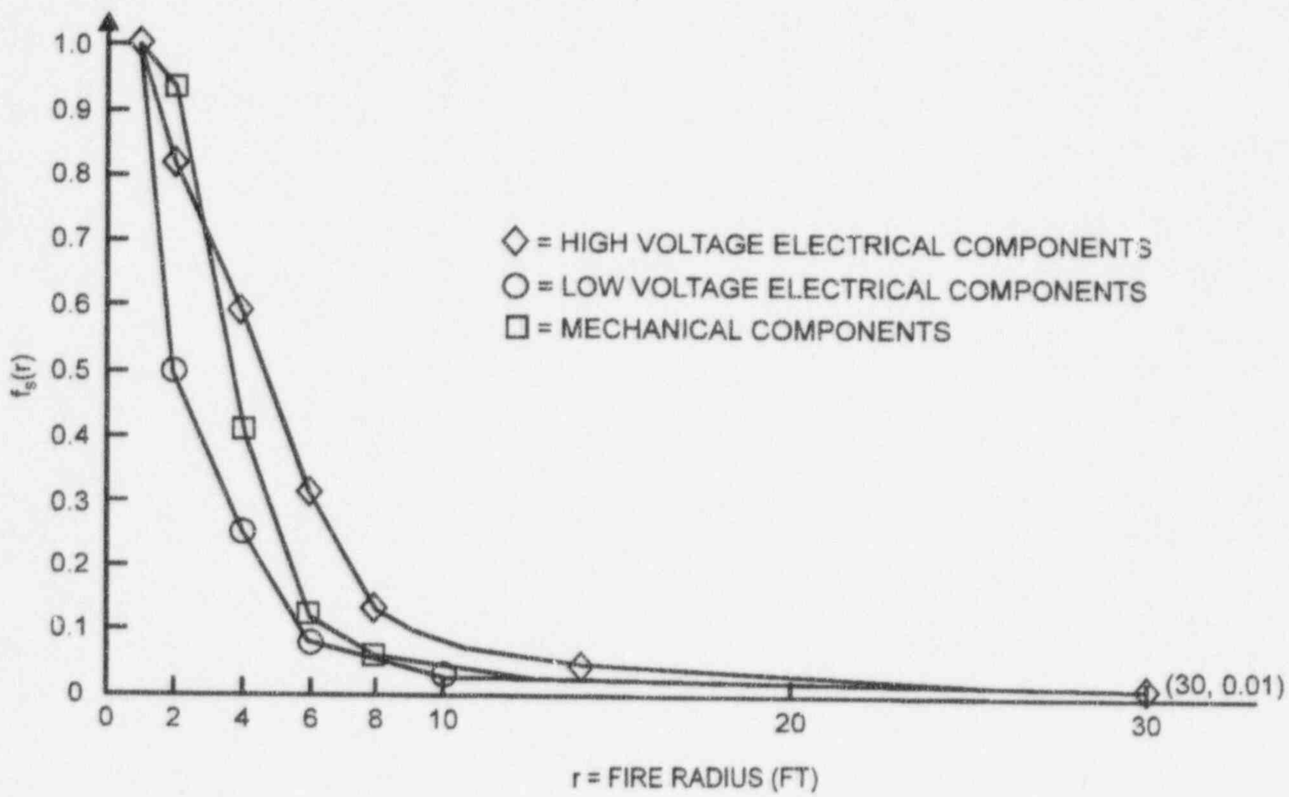


Figure 4.3-2 Fire Severity Curve for Electrical and Mechanical Components

4.4 EVALUATION OF COMPONENT FRAGILITIES AND FAILURE MODES

In the spatial interactions analysis, a component was assumed to be damaged once a fire (of any initial severity from any fire source) occurs within the same fire zone (regardless of the separation distance and suppression efforts). This conservative assumption allows the fire zones with insignificant risk impact to be screened before the detailed analysis performed subsequently.

In the detailed analysis phase where the risk contributions from the important plant locations are required, the component fragility (in terms of fire threats) must be analyzed in more detail. Three factors were used to estimate fire-induced component fragilities: the severity factor ($\mathbf{f_S}$), the geometric factor ($\mathbf{f_G}$), and the fire nonsuppression factor ($\mathbf{f_{NS}}$).

These three factors, $\mathbf{f_S}$, $\mathbf{f_G}$, and $\mathbf{f_{NS}}$, are often referred to as the frequency reduction factors because they were used mainly to reduce the fire frequency assigned to a fire zone for the unique situation modeled by a particular subscenario. For many locations with lower fire risk (i.e., low fire vulnerability), only geometric and severity factors were used.

4.4.1 SEVERITY FACTOR

The severity factor ($\mathbf{f_S}$) was used to assess the damage probability of a component caused by a fire in the vicinity (except transient fires). In calculating the severity factor, it is assumed that if a fire has an initial severity to grow to **r** ft away, then all components within **r** ft from the fire source would be damaged. The severity factor represents a generic probability, and it does not consider the configurations of the specific fire source(s) and targets, and the FPS at a particular location. For example, if a component is **r** ft away from the fire source, $\mathbf{f_S}(\mathbf{r}$ ft) gives the probability that a fire initiated by a fire source would propagate to the component. The component is then assumed to fail, regardless of the duration of fire exposure and available fire suppression efforts.

Figures 4.3-1 and 4.3-2 define the severity factor as a function of radial distance from the fire source for control room panel fires, and electrical and mechanical plant equipment fires. The curves in figures 4.3-1 and 4.3-2 were estimated from nuclear power plant fire incident data and engineering judgment (Reference 2).

4.4.2 GEOMETRIC FACTOR

The geometric factor ($\mathbf{f_G}$) was used to further apportion the component-based fire frequency associated with a location scenario to a particular subscenario developed from that fire zone based on the component population included in the subscenario and the characteristics of the subscenario. For transient fires, multiple runs of COMPBRN IIIe were performed by moving a transient fire source from directly underneath a safety-related cable tray outward to a point where the cable tray would not be affected by the transient fire. The horizontal separation distance between this point and the cable tray is referred to as the critical radius (beyond which the cable tray would not be damaged by the transient fire source). The ratio of the floor area covered by the critical radius to the total floor area provides an estimate of the influence of transient fires to a specific cable tray in a location. The product of the subscenario apportionment factor and the area ratio (for transient fires only) is called the geometric factor in this analysis.

In the detailed analysis, the severity factor was typically used to refine the fire frequency for subscenarios with equipment as the fire source; i.e., pump, motor, transformer, etc. The geometric factor was typically used to refine the fire frequency for subscenarios involving transient fires (human error initiated fires), in addition to its use to apportion the component ignition frequency.

4.4.3 FIRE NONSUPPRESSION FACTOR

In locations where the fire vulnerability is relatively more severe, the fire nonsuppression factor ($\mathbf{f_{NS}}$) was used to account for the fire duration, fire barrier, orientations and configurations of the fire sources and targets, and the interactions of suppression efforts. The fire nonsuppression factor was used to assess the likelihood of a component being damaged by a given fire in the presence of fire suppression (manual or automatic) efforts.

If a fire has the potential to grow and propagate a certain distance, plant components within that distance may still be functionally intact if the fire can be controlled, or is self-extinguished, before the components reach their damage threshold; e.g., damage temperatures. The fire nonsuppression factor is used to account for the time element that is missing in the quantification of the severity factor and geometric factor. If the severity factor and geometric factor are used to answer the question of whether a fire source is strong enough to damage a component at a certain distance, then the fire nonsuppression factor is used to answer the question of whether the fire burns long enough to damage a particular component or set of components.

The quantification of the fire nonsuppression factor requires assessment of the time intervals between fire initiation, time to damage of the critical components, and time required for fire detection and suppression. These different time intervals can be summarized by two characteristic time factors: the fire damage time or fire growth time ($\mathbf{t_G}$), and the fire control time or fire hazard time ($\mathbf{t_H}$).

The fire growth time is defined as the time taken for a given fire to damage the component. The fire hazard time is defined as the time required to control the fire or the fire exposure time during which the component can be damaged by the fire. The fire hazard time factor includes time to detect the fire, time to actuate automatic suppression system (if equipped), time to summon the fire brigade, fire brigade response time, and time to control the fire growth. It is noted that a fire can be controlled long before extinguishment. A fire is said to be controlled if the heat release from the fire source(s) can no longer affect the target(s). Uncertrainity in both the fire damage time and fire hazard time is expressed as a probabilty distribution.

Thus, fire growth time and fire hazard time depend on the characteristics and spatial relationship of the fire and the targets, while the fire hazard time also depends on the characteristics of the FPS at the location. A component is considered to be damaged if its fire growth time is less than the fire hazard time. The fire nonsuppression factor can then be expressed as:

$$\mathbf{f_{NS}} = \text{Frequency}\ (\boldsymbol{t_G} < \boldsymbol{t_H}) \tag{4.1}$$

where t_G and t_H are the random values associated with the probability distributions of $\mathbf{t_G}$ and $\mathbf{t_H}$, respectively.

4.4.3.1 Fire Damage Time

In the detailed analysis, the fire growth time is based on physical models and fire hazard time is determined from generic response times of industry wide fire drills and responses to actual fires. The uncertainities (eg., knowledge model) in the distribution of growth time are believed to be greater than the random variability (eg., physical properties) of fire growth time and the uncertainties in the distribution of fire hazard time overwhelmed uncertainties in the fire growth time. Thus, fire growth time can be represented by its mean value τ_G of the distribution described in Equation (4.1):

$$\mathbf{f_{NS}} = \text{Frequency}\ (\tau_G < \boldsymbol{t_H}) \tag{4.2}$$

which can then be expressed as:

$$\mathbf{f_{NS}} = 1 - F_{t_H}\ (\tau_G) \tag{4.3}$$

where F_{t_H} is the cumulative probability function for $\mathbf{t_H}$.

Given a fire of a certain initial fire size, the COMPBRN IIIe code (section 4.3.2) predicts the surface temperature of the safety-related components (mainly safety-related cables) using a set of user-defined parameters. COMPBRN IIIe can also compare the surface temperatures with a user-specified damage temperature of the components. If the surface temperature of a component is higher than its damage temperature, then the component is considered to be damaged, and τ_G is the time at which the component is declared damaged (with the consideration of the parameter uncertainties). The mean damage temperatures used in the detailed analysis were conservatively assumed to be 450°K (based on the qualification temperature of the cables at Plant Hatch). It is recognized that the qualification temperature is generally well below the typical ignition temperature of the cable jacket material (typically, in the range of 700°K to 850°K) and the damage temperature (623°K) suggested by References 1, 2, and 3.

4.4.3.2 Fire Control Time and Nonsuppression Factor Analysis

In earlier fire analyses (e.g., Reference 4), the fire nonsuppression factor was often estimated by the following simplified model:

$$\mathbf{f}_{NS} = \exp\left(-\frac{\tau_G}{t_S}\right) \tag{4.4}$$

where τ_G is the time to damage critical equipment predicted by COMPBRN IIIe (section 4.4.4), and $\mathbf{t}_s$ is the mean suppression time. This exponential model was also used in References 1 and 2.

However, because the exponential model presumes the suppression time is exponentially distributed and does not consider nuclear power plant fire incident data, the type of FPS in a location, and the initial fire severity of a scenario, it was found not to be scenario-specific (Reference 5) and nonrepresentative.

Reference 5 presents a more sophisticated detection/suppression transition model that distinguishes the type of FPS being analyzed and breaks down the detection/suppression processes into stages (figure 4.4-1). Briefly, the transition model separates the detection and suppression processes and considers the possible combination of manual and automatic response of different types of detection and suppression systems. Conditional probabilities are then assigned to each transition path based on nuclear power plant fire event data; e.g., Reference 6. References 3, 5, and 7 provide a detailed description of the transition model and, therefore, are not repeated here.

Because the transition model requires extensive numerical manipulation, an approximated form (as a Weibull distribution, W) has been commonly used (References 3, 5, and 8):

$$W(\alpha, \beta, t) = \frac{\beta}{\alpha}\left(\frac{t}{\alpha}\right)^{\beta-1} e^{-\left(\frac{t}{\alpha}\right)^{\beta}} \tag{4.5}$$

and

$$F_{t_H}\left(t|\alpha,\beta\right) = 1 - \exp\left[-\left(\frac{t}{\alpha}\right)^{\beta}\right] \tag{4.6}$$

where α and β are the parameters for the Weibull distribution and are FPS class-specific, and $\mathbf{t}$ is fire growth time.

Following Equations (4.3) and (4.6), $\mathbf{f_{NS}}$ becomes:

$$\begin{aligned} f_{NS} &= 1 - F_{t_H}\left(\tau_G \middle| \alpha, \beta\right) \\ &= \exp\left[-\left(\frac{\tau_G}{\alpha}\right)^{\beta}\right] \end{aligned} \tag{4.7}$$

where τ_G is the damage time predicted by COMPBRN IIIe.

Since the Plant Hatch FPS is similar to other previously analyzed nuclear power plant FPS and there is no evidence that the failure rate of the Plant Hatch FPS is significantly different from that of the generic failure rate for FPS, generic failure rates based on nuclear power plant fire records were used in Equation (4.5) to evaluate the parameters for the Weibull distribution for the different FPS classes. Furthermore, the fire brigade response times obtained from actual fire drill records were comparable to the response times provided by Reference 6; therefore, the generic fire response time was used. The mean fire hazard time for the various room FPS class were estimated to be in the range of 39.0 to 69.5 minutes (References 3, 5, and 7), depending on the initial fire severity modeled. This range of mean fire hazard time is considered to be reasonable and conservative because most fires can be detected by automatic fire detectors within seconds of fire initiation.

Two types of fire severity were modeled in Reference 5: high and low initial fire severity. The Plant Hatch analysis uses the parameters developed for high initial severity fires. Table 4.4-1 summarizes the fire nonsuppression factor calculations for several applicable FPS classes (i.e., II, IV, and V) with high initial fire severity as estimated in Reference 3. The fire nonsuppression factor values were used in the detailed subscenario analysis.

4.4.4 FAILURE RESPONSE

The analysis of failure response in the spatial interactions quantitative screening step is discussed in section 4.1.5. The plant response to component failures caused by fire damage for detailed analysis is described in section 4.6.3.

REFERENCES

1. Lambright, J. A., et al., "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," Sandia National Laboratories, prepared for U. S. Nuclear Regulatory Commission, NUREG/CR-5088, January 1989.

2. U. S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," NUREG-1150, June 1989.

3. Brookhaven National Laboratories, "Fire Risk Analysis of POS 6 and POS 10, Surry Low Power and Shutdown Project," prepared for U. S. Nuclear Regulatory Commission, NUREG/CR-6144, 1993.

4. Pickard, Lowe and Garrick, Inc., Westinghouse Electric Corporation, and Fauske & Associates, "Indian Point Probabilistic Safety Study," 1982.

5. Siu, N. O., and Apostolakis, G., "A Methodology for Analyzing the Detection and Suppression of Fires in Nuclear Power Plants," Nuclear Science and Engineering, 94(1986)213-226.

6. PLG, Inc., "Database for Probabilistic Risk Assessment of Light Water Reactor Power Plants," PLG-0500, Vol. 8, May 1994.

7. Ho, V. S., "Improvement of FNS Input Data," PLG, Inc., PLG-0900, March 1994.

8. PLG, Inc., "Beznau Station Risk Assessment, Plant with NANO," prepared for Nordostschweizerische Kraftwerke AG, PLG-0511, December 1989.

Table 4.4-1 Summary of Fire Nonsuppression Factor Calculations

Damage Time (Minutes)	Fire Protection System Class[(1)]		
	II	IV	V
0	1	1	1
1	0.82	0.79	0.84
2	0.75	0.71	0.78
3	0.69	0.66	0.73
4	0.65	0.62	0.69
5	0.62	0.59	0.65
6	0.58	0.56	0.62
7	0.56	0.53	0.59
8	0.53	0.51	0.57
9	0.51	0.49	0.55
10	0.49	0.47	0.53
11	0.47	0.45	0.51
12	0.46	0.44	0.49
13	0.44	0.42	0.47
14	0.43	0.41	0.46
15	0.41	0.40	0.45
16	0.40	0.38	0.43
17	0.39	0.37	0.42
18	0.37	0.36	0.41
19	0.36	0.35	0.40
20	0.35	0.34	0.39
25	0.31	0.30	0.34

Note:

1. Fire Protection System Class — Definition

Fire Protection System Class	Definition
II	Detectors; no suppression systems.
IV	Detectors, manually actuated suppression systems.
V	Detectors; automatic suppression systems

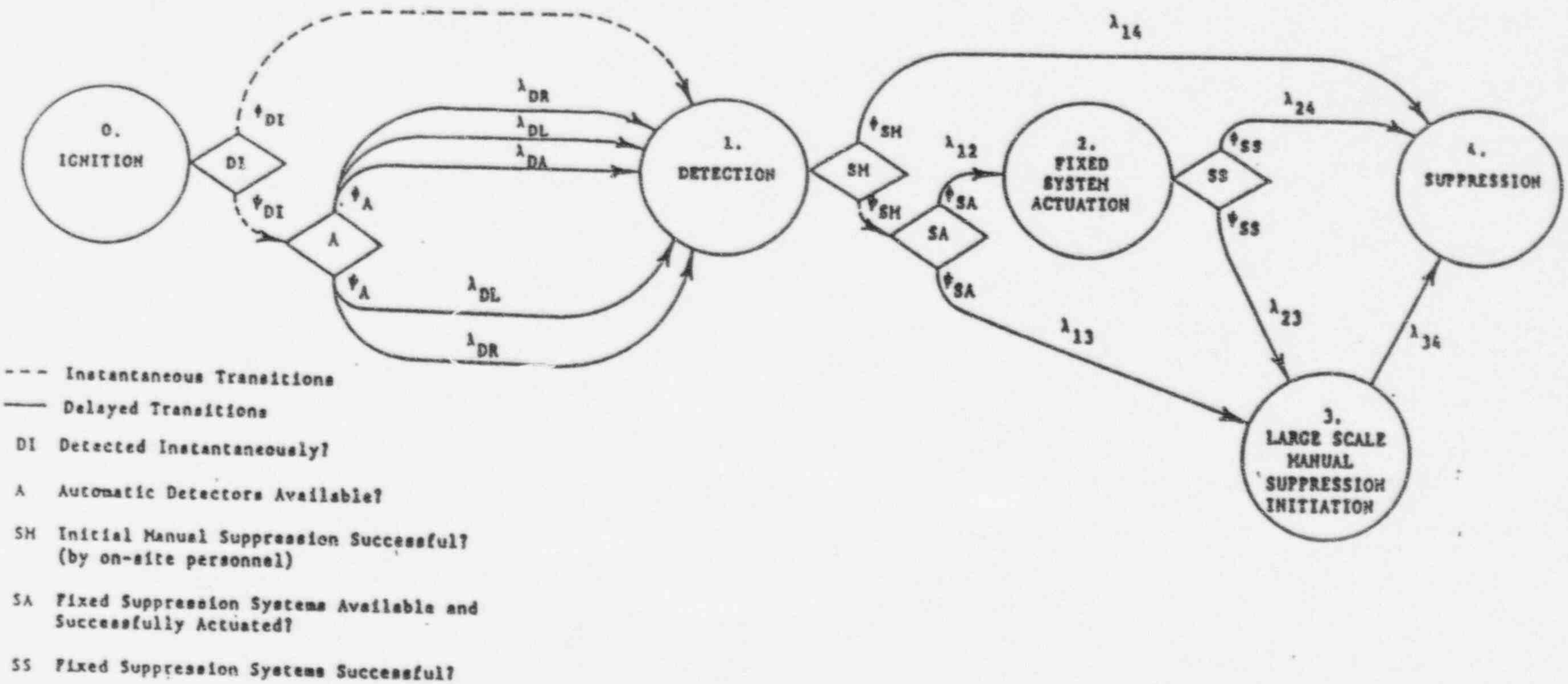


Figure 4.4-1 Transition Model for Detection and Suppression

4.5 FIRE DETECTION AND SUPPRESSION

The type of fire detection and suppression systems within a fire zone determines the fire hazard time, which, in turn, is used to assess the fire nonsuppression factor. The assessment of the fire nonsuppression factor is described in section 4.4.3.

In accordance with the information provided in References 2, 3, and 4, the fire protection system (FPS) in different fire zones at Plant Hatch can be categorized into five FPS classes:

- I No detectors; no suppression systems.
- II Detectors; no suppression systems.
- III No detectors; manually actuated suppression systems.
- IV Detectors; manually actuated suppression systems.
- V Detectors; automatic suppression systems.

The availability and the effectiveness of a FPS often dictates the severity and consequences of a fire. Thus, suppression probability is an important factor in a detailed fire analysis. In the context of a fire risk analysis, it is sometimes more appropriate to consider nonsuppression frequencies rather than suppression probabilities (References 1, 2, and 3).

REFERENCES

1. Lambright, J. A., et al., "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," Sandia National Laboratories, prepared for U. S. Nuclear Regulatory Commission, NUREG/CR-5088, January 1989.

2. Siu, N. O., and Apostolakis, G., "A Methodology for Analyzing the Detection and Suppression of Fires in Nuclear Power Plants," Nuclear Science and Engineering, 94(1986)213-226.

3. Pickard, Lowe and Garrick, Inc., Westinghouse Electric Corporation, and Fauske & Associates, "Indian Point Probabilistic Safety Study," 1982.

4. Edwin I. Hatch Nuclear Plant, Units 1 and 2 Fire Hazards Analysis and Fire Protection Program, revision 10C, April 1995.

4.6 ANALYSIS OF PLANT SYSTEMS, SEQUENCES, AND PLANT RESPONSE

4.6.1 DEVELOPMENT AND ANALYSIS OF SUBSCENARIOS (STEP 6)

Detailed analysis scenarios, called subscenarios, were developed for each scenario that was retained from the spatial interactions analysis quantitative screening. Each subscenario accounts for an individual fire source (or group of fire sources) within a fire zone that can cause a failure to one or more components. Special attention was given to fire growth, fire propagation, fire detection and suppression, timing of failures, previously analyzed plant impacts, and operator recovery actions.

Because the main control room is manned 24 hours a day and the apportioned fire frequency for the cabinet fire source category dominates the fire exposure risk, only the risk-significant panel-initiated fires were considered for the detailed analysis of control room fires.

The core damage frequency (CDF) for the plant , ϕ_i, associated with fire zone **i** can be obtained by summing the plant impact of **j** subscenarios developed for the fire zone:

$$\phi_i = \sum_j \lambda_{i,j}\, f_{R,j}\, Q_{x,j} \tag{4.8}$$

where

$\lambda_{i,j}$ = fire occurrence frequency apportioned to the fire source(s) modeled by subscenario **j** in fire zone **i**.

$f_{R,j}$ = frequency reduction factor, including $f_{S,j}$, $f_{G,j}$, and $f_{NS,j}$, where
- $f_{S,j}$ = severity factor;
- $f_{G,j}$ = geometric factor; or
- $f_{NS,j}$ = fire nonsuppression factor.

$Q_{x,j}$ = conditional CDF, given that the fire described by subscenario **j** has occurred. The frequency of core damage is a combination of random failures and fire-induced failures. The conditional CDF also accounts for the failure probability of operator-initiated plant recovery actions (nonrecovery factors).

As discussed previously, in the spatial interactions analysis phase, the scenario occurrence frequency of a fire zone denotes the total fire occurrence frequency of all fire sources within the fire zone. However, in the detailed analysis phase, individual fire sources are evaluated separately in each subscenario and as a result, the scenario occurrence frequency is further apportioned to the particular fire sources modeled in each subscenario. Thus, the subscenario occurrence frequency, $\lambda_{i,j}$, is the fire occurrence frequency of the particular fire source(s) modeled in subscenario **j** for fire zone **i**.

The frequency reduction factors are described in sections 4.4.1 through 4.4.3 in detail and may have a point estimate value ranging from 0.0 to 1.0. The factor $\mathbf{f_R}$ represents the product of $\mathbf{f_S}$, $\mathbf{f_G}$ and $\mathbf{f_{NS}}$ in the evaluation of $\boldsymbol{\phi}_i$ for fire zone **i**.

The analysis of plant response (development of the conditional CDF) in the detailed analysis is a refinement of the failure response analysis in the spatial interactions analysis (section 4.1). Instead of failing all of the top events associated with a particular fire zone, only impacts on the top events associated with the components failed by the specific subscenario were modeled. Appropriate split fraction values were then used to calculate $\mathbf{Q_x}$, the conditional CDF, of each subscenario. Certain operator recovery actions were also modeled to refine the conditional CDF.

The CDFs for the subscenarios were calculated in two separate iterations:

1. For all subscenarios developed in the detailed analysis, a simplified core damage sequence model (section 4.1, Step 5) was used to calculate the conditional CDFs. The unconditional CDF for each of the subscenarios was calculated by multiplying the occurrence frequency of the fire subscenario by the conditional CDF.

2. For subscenarios with an unconditional CDF thus calculated > 3.0E-08 per year, a second quantification using RISKMAN® software (Reference 1) was performed with the full internal events Individual Plant Examination (IPE) (Reference 2) model by representing the fire-impacted top events with appropriate split fraction values in the event trees. This would accommodate both the fire-induced failures and component/system unavailabilities unrelated to fires in the determination of CDF.

The following event trees were linked and used in the second quantification of fire subscenarios that did not involve a loss of offsite power (LOSP): ELEC1, MECH1, SCRAM, INTER1, REC0, RHRCS, LTC1, CET, and ENDBACK. For fire scenarios involving an LOSP, the following event trees were linked and used in the second quantification: ELEC2, MECH2, LOSP, INTER2, REC3, REC2, RHRCS, LTC1, CET, and ENDBACK. Descriptions for these event trees can be found in the IPE. In the second quantification using the full internal events IPE model, the fire subscenario frequency was used as the initiating event frequency. The conditional CDF was calculated by dividing the CDF obtained from the full model quantification by the fire subscenario frequency; i.e., the initiating event frequency.

Some of the fire subscenario recovery actions were incorporated by splitting the original subscenario into two subscenarios: one with a successfully executed recovery action and one with a failure of the recovery action. The frequency of each of these two subscenarios is the product of the original subscenario frequency and the probability of success or failure of the recovery action. The representation of these two subscenarios was only used in the model quantification. The results were still represented in terms of the original fire subscenarios. The conditional CDF was calculated by dividing the total CDFs associated with these two subscenarios by the original fire subscenario frequency.

Event tree ENDBACK was used to group the core damage sequences into the various containment event tree (CET) endstates needed for the containment performance analysis (section 4.7).

4.6.2 RESULTS OF FIRE RISK ANALYSIS

Table 4.6-1 presents a summary of the total fire-induced CDF for both Unit 1 and Unit 2. As shown in this table, the total CDF is 7.5E-06 per year and 5.4E-06 per year for Unit 1 and Unit 2, respectively. This represents approximately 36 percent and 24 percent of the CDF from the internal events for Unit 1 and Unit 2, respectively. Part of the reason for the smaller CDF for Unit 2, compared to Unit 1, is because a more detailed evaluation of the effects of fire damage to cables was performed for Unit 2. As such, a more conservative representation of the impact of fire damage to cables was modeled for Unit 1 than Unit 2.

4.6.2.1 Risk-Dominant Fire Scenarios

Tables 4.6-2 and 4.6-3 summarize the top fire subscenarios contributing to the fire-induced CDF for Unit 1 and Unit 2, respectively.

4.6.2.1.1 Selected Unit 1 Scenarios

The following descriptions of selected Unit 1 fire subscenarios expand on the information provided in table 4.6-2. These are all of the Unit 1 fire subscenarios with an unconditional CDF > 1.0E-07 events per year. Because of the level of detail included in the models, none of the Unit 1 fire subscenarios presented represent more than 5 percent of the fire-induced CDF. The numbers in parentheses correspond to the rank number in table 4.6-2.

(1) Small Cable Fire in Cable Spreading Room Damaging One Cable Tray (0024A-L-C11)

CDF: 1.4E-06 events per year.

This subscenario models small cable fires damaging one of the following critical cable trays: TFA7-1,2,3,4; TFB7-1,2; TFE7-1; TFU8-3,4,5,6; TFY8-2,3,6,7,8,9,10; TFZ7-2,5; TME7-5,6,7; TMS8-1,5; TMT7-18,19,25; TMU7-1; or TSJ8-1. The cable tray with the most severe impact was used to model the fire damage; i.e., TFZ7-2. The major equipment failed due to fire damage to all cables in cable tray TFZ7-2 include 4-kV emergency bus 1E, 4-kV emergency bus 1G, residual heat removal (RHR) loop A, core spray loop A, containment venting, drywell spray (failed by cable trays other than TFZ7-2), one of two containment isolation valves in a number of penetrations (e.g., X-18, X-19, X-14, X-205), and both containment isolation valves for penetration X-26, and two RHR shutdown cooling suction isolation valves.

Failure of emergency buses 1E and 1G is caused by closure of the alternate supply breakers for these buses due to shorting of two separate circuits. Due to the design of the normal and alternate supply breakers for the 4-kV buses, closure of the alternate supply breakers trips open the respective normal supply breakers resulting in an undervoltage condition on these buses. Loss of buses 1E and 1G de-energizes 600-V ac buses 1C and 1D, respectively. Unavailability of bus 1C causes a loss of feedwater and a subsequent reactor trip on low water level signal.

Since the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems are not damaged by the fire, they can provide vessel water invertory control. If the operators successfully realign the supply of either 600-V ac bus 1C or 1D to transformer 1CD (fed by 4-kV emergency bus 1F), dc power supply to one of the high-pressure injection sources will be available as well as the room coolers for HPCI. However, if ac power is not restored to either buses 1E or 1G, both RCIC and HPCI will become unavailable when the torus water heats up to a temperature at which the RCIC or HPCI turbine can no longer be cooled.

The 4-kV emergency buses 1E and 1G can be restored to service by de-energizing the control power for the alternate supply breakers, manually opening the alternate supply breakers, and reclosing the normal supply breakers. The amount of time available for the operators to perform this recovery action is at least 4 hours; i.e., the duration until RCIC/HPCI fails. If this recovery action is unsuccessful, core damage would occur. This is because the condensate system, feedwater system, and main condenser are unavailable due to the loss of turbine building plant service water (PSW) resulting from a loss of power to two PSW pumps and closure of the turbine building PSW isolation valves. The core spray system is unavailable due to the loss of pump motive power. Decay heat removal (DHR) through RHR loop B is lost due to the loss of power to both residual heat removal service water (RHRSW) loop B pumps. With containment venting unavailable due to fire damage, the mitigation system operable is RHR loop A which is assumed to be able to provide either vessel injection or DHR functions.

(2) Switchgear 1E Fire Causing Loss of Offsite Power (1412-L-G1)

CDF: 7.9E-07 events per year.

This subscenario models fires on the west end of switchgear 1E causing a loss of offsite power supply to all three 4-kV emergency buses, failure of 4-kV emergency bus 1E, and failure of 125-V dc power associated with diesel generator 1A. The 4-kV offsite power supply cables from startup transformers 1D and 1C are located about 4 ft above the west end of switchgear 1E. Since there is no protective device between the startup transformers and the switchgear, grounding of both of these supply cables in fire zone 1412 could cause the disconnect switches between the two 230-kV buses and the two startup transformers (i.e., 440, 450, 470, and 480) to trip open. This would result in the loss of offsite power to all three 4-kV emergency buses.

In addition to the above fire-induced failures, if diesel generators 1B and 1C fail due to causes other than fire, a loss of ac power on all three 4-kV emergency buses would occur. The most likely sequences involving failure of diesel generators 1B and 1C include loss of the two diesel generators due to failure of the diesel equipment itself, its cooling water supply, or diesel control power.

Initially, after the loss of ac power on all three emergency buses, both HPCI and RCIC are available to provide vessel water level control. However, HPCI will soon become unavailable due to the loss of room cooling. RCIC can operate for an extended period of time without the room cooling. As the division I station battery is depleted, RCIC is assumed unavailable. It is also assumed that offsite power has not been restored at the time the station battery is depleted, and attempts to recover a failed diesel generator have been unsuccessful. Core damage will then occur after loss of all water injection sources.

(3) Switchgear 1G Fire Causing Loss of Offsite Power (1404-L-G1)

CDF: 7.7E-07 events per year.

This subscenario is similar to 1412-L-G1(Rank no. 2), except it is associated with switchgear 1G and diesel generator 1C.

(4) Switchgear 1F Fire Causing Loss of Offsite Power (1408-L-G1)

CDF: 4.5E-07 events per year.

This subscenario is similar to 1412-L-G1, except it is associated with switchgear 1F and diesel generator 1B. As explained previously, the total impact of switchgear 1F failure is less severe than that of switchgear 1E or 1G. This results in a lower CDF for this subscenario.

(5) Transient Fire in Switchgear 1E Room Causing Loss of Offsite Power (1412-L-B3)

CDF: 3.6E-07 events per year.

This subscenario is similar to 1412-L-G1, except the fire source is transient combustibles. Offsite power supply cables, 4-kV switchgear 1E, and 125-V dc control power associated with diesel generator 1A are damaged by the fire.

(6) Transient Fire in Switchgear 1G Room Causing Loss of Offsite Power (1404-L-B3)

CDF: 3.6E-07 events per year.

This subscenario is similar to 1404-L-G1, except the fire source is transient combustibles. Offsite power supply cables, 4-kV switchgear 1G, and 125-V dc control power associated with diesel generator 1C are damaged by the fire.

(7) Small Cable Fire in Cable Spreading Room Damaging One Cable Tray (0024A-L-C7)

CDF: 2.2E-07 events per year.

This subscenario models small cable fires damaging one of the following critical cable trays: TFG9-1,2,3,7; TME7-10; TMI9-1,2,4; TMJ7-11,12; or TMT7-35. The cable tray with the most severe impact was used to model the fire damage; i.e., TMT7-35. The major equipment failed due to fire damage to all of the cables in cable tray TMT7-35 include turbine building PSW, HPCI, RCIC, core spray loop A, division I analog transmitter trip system (ATTS), drywell spray, and one of the two containment isolation valves for penetration X-205. Failure of the turbine building PSW would result in loss of the condensate system, feedwater system, and main condenser.

Since the fire has damaged all of the high-pressure injection systems in this subscenario, core damage would occur if vessel depressurization fails.

(8) Transient Fire in Switchgear 1F Room Causing Loss of Offsite Power (1408-L-B3)

CDF: 2.1E-07 events per year.

This subscenario is similar to 1408-L-G1, except the fire source is transient combustibles. Offsite power supply cables, 4-kV switchgear 1F, and 125-V dc control power associated with diesel generator 1C are damaged by the fire.

(9) Control Room Fire Damaging Panel 1H11-P652 (MCR-13)

CDF: 1.5E-07 events per year.

This subscenario models cabinet fires in the main control room damaging all three sections of panel 1H11-P652. The controls provided at this main control room panel include 4-kV emergency buses 1E, 1F, and 1G; diesel generators 1A, 1B, and 1C; 600-V ac bus 1C; station service transformers 1C, 1D, and 1CD; PSW turbine building isolation valves; and a number of motor control centers (MCCs). It is conservatively assumed that all of the preceding equipment will become unavailable when panel 1H11-P652 is damaged by fire. Loss of the turbine building PSW due to failure of turbine building isolation valves was assumed to cause failure of the condensate system, feedwater system, and main condenser.

The 4-kV emergency buses and 600-V buses can be restored to service by manually operating equipment from the 4-kV and 600-V switchgear rooms. If this operator recovery action fails, a loss of all essential ac power and core damage are assumed to occur.

(10) Cabinet Fire in Switchgear 1G Room Causing Cable Failures (1404-L-D1)

CDF: 1.2E-07 events per year.

This subscenario models fires from the dc cabinets in switchgear 1G room causing cable trays located 3 to 4 ft above the cabinets to fail. It is assumed that 4-kV emergency bus 1G and 125-V dc power associated with diesel generator 1C fail as a result of the fire damage. Since bus 1G supplies motive power to PSW pump B, and control power for PSW pumps B and D is provided by the dc power associated with diesel generator 1C, turbine building PSW is assumed to be lost due to insufficient number of PSW pumps being available. As a result, the condensate system, feedwater system, and main condenser are assumed to be unavailable. HPCI is also assumed to be unavailable due to the loss of room cooling. In addition, RHRSW loop B and core spray loop B are lost due to failure of bus 1G. RCIC is the only system available for high-pressure injection. If RCIC fails due to causes other than fire and vessel depressurization fails, core damage would occur. Furthermore, torus cooling via RHR loop A and containment venting are the only means of DHR given the fire damage. If both of these DHR mechanisms fail due to causes other than fires, core damage would also occur.

(11) Large Cable Fire in East Cableway Damaging Multiple Cable Trays (1104A-L-A3)

CDF: 1.0E-07 events per year.

This subscenario models large cable fires in the east cableway damaging the following cable trays: TCG8-04, TCJ8-02, TCK8-02, TCL8-01, TCM8-01, TCV0-02, and TCW0-01. Impact on the plant due to fire damage to these cable trays is failure of HPCI, core spray loop B, RHR loop B, room coolers for reactor building southeast corner room (1203B) and northeast corner room (1205B), division II ATTS, containment venting, drywell spray loop B, and one of two containment isolation valves for penetration X-14.

Since both RHR loop B and containment venting fail due to fire damage, core damage would occur if RHR loop A and the main condenser are also lost. The most likely failures leading to the loss of main condenser are failures of the condensate system, normal ac power, main condenser itself, steam bypass valves, turbine building PSW, etc.

(12) Large Cable Fire in East Cableway Damaging Multiple Cable Trays (1104A-L-A1)

CDF: 1.0E-07 events per year.

This subscenario is similar to 1104A-L-A3. The cable trays damaged by fire are TCC4-01,02; TCJ8-01; TCV0-02; THD4-01; and THJ8-01. Impact on the plant due to fire damage to these cable trays is failure of reactor building MCC 1B (1R24-S012), HPCI, RHR loop B, room coolers for reactor building southeast corner room (1203B) and northeast corner room (1205B), division II ATTS, automatic closure of one of two turbine building isolation valves in both reactor building PSW headers, containment venting, and one of two containment isolation valves for penetration X-14. Since reactor building MCC 1B is lost, core spray loop B becomes unavailable due to loss of motive power to the loop B normally closed isolation valve. As a result, the impact of this subscenario is very similar to subscenario 1104A-L-A3.

The frequencies of all of the remaining subscenarios are < 1.0E-07 events per year.

4.6.2.1.2 Selected Unit 2 Scenarios

The following descriptions of selected Unit 2 fire subscenarios expand on the information provided in table 4.6-3. These are all of the fire subscenarios with an unconditional CDF > 1.0E-07 events per year. Because of the level of detail included in the models, none of the Unit 2 fire subscenarios presented represent more than 4 percent of the total fire-induced CDF. The numbers in parentheses correspond to the rank number in table 4.6-3.

(1) Switchgear 2G Fire Causing Loss of Offsite Power (2409-L-G2)

CDF: 8.4E-07 events per year.

This subscenario is similar to 1404-L-G1 for Unit 1.

(2) Small Cable Fire in Cable Spreading Room Damaging One Cable Tray (0024A-L-C4)

CDF: 7.1E-07 events per year.

This subscenario models small cable fires damaging one of the following critical cable trays: CAA801; DAB702, 703; DBB701, 702, 703; DBD701; DBG701, DCB701, 702, 703; DCG701; DEA801, 802; DEB801; DED801, 802, 807; DGC801; DGD801, 802, 803, 804, 805; DGF801; or DJA701. The cable tray with the most severe impact on the plant was used to model the fire damage; i.e., DGF801. The major equipment failed due to fire damage to all of the cables in cable tray DGF801 include 4-kV emergency bus 2G, 600-V ac bus 2D, division II PSW, division II ATTS, safety relief valve (SRV) , HPCI, RCIC, pump C of RHR loop A, RHR loop B, core spray loop B, containment venting, drywell spray (failed by cable trays other than DGF801), one of two containment isolation valves in a number of penetrations (e.g., X-18, X-19, X-14, X-205, etc.), both containment isolation valves for penetration X-26, and one of two shutdown cooling isolation valves. Loss of either division of PSW (division II in this case) would result in failure of the condensate system, feedwater system, and main condenser due to the loss of turbine building PSW.

Failure of 4-kV bus 2G and 600V bus 2D can be recovered by manually controlling them from the switchgear rooms. Similarly, loss of division II PSW pumps can be restored by controlling the PSW pumps from switchgear rooms in the diesel generator building. RCIC and RHR loop B can be recovered from the remote shutdown panel.

Core damage would occur if the above recovery actions are unsuccessful and if:

- The reactor is not depressurized because water injection would not be available since all high-pressure injection systems (i.e., feedwater, HPCI, and RCIC) are lost.

- The DHR via RHR loop A is unavaialble. This may be caused by nonfire related failures of the RHR loop A, RHRSW loop A, or division I PSW. This implies that all means of DHR is lost.

If the recovery actions are successful, core damage would occur if:

- The RCIC failure was followed by a failure of reactor vessel depressurization.

- The DHR via both RHR loops fails.

(3) Switchgear 2E Fire Causing Loss of Offsite Power (2404-L-G2)

CDF: 6.9E-07 events per year.

This subscenario is similar to 1412-L-G1 for Unit 1.

(4) Switchgear 2F Fire Causing Loss of Offsite Power (2408-L-G2)

CDF: 4.2E-07 events per year.

This subscenario is similar to 1408-L-G1 for Unit 1.

(5) Large Station Service Transformer Fire (2016-L-A1)

CDF: 2.0E-07 events per year.

This subscenario models large fires from the station service transformer 2C causing failure of cable tray 2CSA701. The impact of transformer and cable tray failures is loss of division I 600-V bus 2C, reactor building MCC 2C (2R24-S011), essential cabinet 2C (2R25-S036), 125/250-V dc switchgear 2A, division I ATTS, and RCIC. As a result of failure of bus 2C and dc switchgear, the condensate system, feedwater system, and main condenser are assumed to be lost. The reactor will trip due to low water level. In addition, the RHRSW loop A flow control valve at the RHR heat exchanger outlet (E11-F068A) and RHR torus cooling valves would also fail due to loss of power. However, these valves can be manually opened using a hand wheel.

If the operator successfully recovers the RHRSW flow control valve and the RHR torus cooling valve fails, the following nonfire failures were assumed to likely cause core damage:

- Failure of HPCI followed by vessel depressurization failure.

- Failure of 600-V bus 2D or its power supply.

- Failure of dc power supply from division II dc switchgear.

- Failure of torus cooling in conjunction with failure of containment venting due to instrument air or vent valve failure.

If the operator fails to recover the RHRSW flow control valve and RHR torus cooling valve fails, the following nonfire failures were assumed to likely cause core damage:

- Failure of DHR via RHR loop B in conjunction with failure of containment venting due to instrument air or vent valve failure.

- Failure of HPCI followed by vessel depressurization failure.

(6) Small Cable Fire in East Cableway (2104-L-A3)

CDF: 1.5E-07 events per year.

This subscenario models small cable fires in the east cableway damaging the following cable trays: 2CAA801, 2CAB801, 2CCA001, 2CCD001, and 2CHA001. Impact on the plant due to fire damage to these cable trays is a stuck open SRV, failure of division II ATTS, HPCI, RCIC, RHR loop B, containment venting, and drywell spray.

Because of the stuck open SRV, it was conservatively assumed that the feedwater system cannot be used for the entire mission time and the decay heat must be removed by either torus cooling or containment venting. Since both RHR loop B and containment venting fail due to fire damage, core damage would occur if DHR via RHR loop A is lost due to nonfire causes.

(7) Control Room Fire Damaging Panel 2H11P652 (MCR-13)

CDF: 1.5E-07 events per year.

This subscenario is similar to MCR-13 for Unit 1.

(8) Large Battery Fires in Division I Battery Room (2004-L-A1)

CDF: 1.2E-07 events per year.

This subscenario models large battery fires failing the division I station battery and the following raceways: 2E11639, 2BMA301, and 2BNA501. Impact of the fire damage to this equipment and raceways is failure of dc power supply from division I 125-V dc switchgear 1A, reactor building MCC 2C (2R24-S011), division I ATTS, and RCIC. Failure of the dc power supply from 125-V dc switchgear 1A was assumed to result in a loss of condensate system, feedwater system, and main condenser. The reactor will trip due to low vessel water level. In addition, the RHRSW loop A flow control valve at the RHR heat exchanger outlet and RHR torus cooling valves will also fail due to loss of power. However, these valves can be manually opened using the hand wheel.

If the operator successfully recovers the RHRSW flow control valve and the RHR torus cooling valve failures, the most likely nonfire failures that were assumed to cause core damage include:

- Failure of HPCI followed by vessel depressurization failure.
- Failure of 600-V bus 2D or its power supply.
- Failure of dc power supply from division II dc switchgear.

If the operator fails to recover the RHRSW flow control valve, the following nonfire failures were assumed to likely cause core damage:

- Failure of HPCI followed by vessel depressurization failure.
- Failure of 600-V bus 2D or its power supply.
- Failure of DHR via RHR loop B in conjunction with failure of containment venting due to vent valve or instrument air failure.
- Failure of dc power supply from division II dc switchgear.

(9) Large Switchgear 2G Fires Causing Cable Failures (2409-L-G1)

CDF: 1.1E-07 events per year.

This subscenario models large fires from switchgear 2G causing cable trays located 4 ft above the switchgear to fail. It is assumed that 4-kV emergency bus 2G and 125-V dc power associated with diesel generator 2C fail as a result of the fire damage. Since bus 2G supplies motive power to PSW pump B, and control power for PSW pumps B and D is provided by the dc power associated with diesel generator 2C, turbine building PSW is assumed to be lost due to insufficient number of PSW pumps being available. As a result, the condensate system, feedwater system, and main condenser were assumed to be unavailable. HPCI was also assumed to be unavailable due to the loss of room cooling. In addition, RHRSW loop B and core spray loop B are lost due to failure of bus 2G. RCIC is the only system available for high-pressure injection. If RCIC fails due to causes other than fire and vessel depressurization fails subsequently, core damage would occur. Furthermore, heat removal via RHR loop A and containment venting are the only means of DHR given the fire damage. If both of these DHR mechanisms fail due to causes other than fires, core damage would also occur.

The frequencies of all of the remaining subscenarios are < 1.0E-07 events per year.

4.6.2.2 <u>Risk-Dominant Plant Locations</u>

Tables 4.6-4 and 4.6-5 list the risk-dominant plant locations, based on the detailed analysis, for Unit 1 and Unit 2, respectively.

4.6.2.2.1 Selected Unit 1 Locations

For Unit 1, the top fire locations include the three 4-kV emergency switchgear rooms in the diesel generator building, the cable spreading room, and the main control room. Each of the top eight Unit 1 plant locations for the fire-induced CDF contribution is described below.

1. **4-kV Emergency Switchgear Rooms (1412, 1408, and 1404).**

   CDF: 3.5E-6 per year

   Fire damage to equipment in each of the 4-kV emergency switchgear rooms can result in failure of one 4-kV ac division and one 125-V diesel dc division. In addition, a segment of the power supply cables routed from startup transformers 1D and 1C to 4-kV emergency switchgears 1E, 1F, and 1G are located in each of the switchgear rooms. This represents a potential for a loss of offsite power supply to all three emergency buses if both of these cables are grounded as a result of a fire. Furthermore, equipment damaged from a fire in these switchgear rooms cannot be easily recovered. As a result, the 4-kV emergency switchgear rooms are among the major contributors of fire-induced CDF. The fire-induced CDF associated with bus 1F switchgear room is lower than the other two switchgear rooms because the overall safety significance of the loads served by 4-kV ac bus 1F and the associated 125-V dc control power is less than that for the loads served by the other two divisions (see Reference 2).

2. **Cable Spreading Room (0024A).**

   CDF: 1.9E-6 per year

The cable spreading room is one of the highest contributors to fire-induced CDF. This fire zone contains a large amount of combustible material; i.e., cables. Cables of both divisions and both units are routed through this room. No cables with voltage 600-V ac or 250-V dc or greater are located in this fire zone. The cables in this room include the control and instrumentation cables for most of the accident mitigation equipment. Therefore, fire damage to these cables can compromise many of the accident mitigation functions. Fire protection features include an early warning smoke detection system and an automatic water suppression system. Necessary equipment disabled due to fire damage in this room can be recovered by using controls available at the remote shutdown panel and other locations.

3. **Main Control Room (0024C).**

CDF: 7.1E-7 per year

The main control room contains a large number of control and instrumentation cabinets and panels. The most vulnerable fire hazard in the main control room is the panel or cabinet fires. Cabinets are equipped with a full metal partition between cabinets that reduce the likelihood of fire propagation from one cabinet to the adjacent cabinets. Fire detectors are not provided inside the cabinets. Some cabinets are open in the back, and many panels are not equipped with doors. The main control room is equipped with a ceiling mounted, early warning smoke detection system. Nevertheless, the most important aspect of the main control room is that it is manned 24 hours a day. Manual fire suppression should be extremely effective.

4. **East Cableway (1104A).**

CDF: 2.1E-7 per year

This fire zone consists of significant cable concentration located on each side of the cableway. Each stack consists of multiple levels of cable trays. The bottom cable trays are provided with solid bottoms. The 4-kV cables are routed to the east cableway from the diesel generator building via the turbine building. These cables are then routed to the reactor building and the 600-V switchgear room in the control building. The traffic level in this area is relatively high; there is no maintenance activity in this fire zone. Fire protection features include an early warning smoke detection and an automatic water suppression system. The floor area is labeled to prevent storage of materials. Cable fires are the most likely fire source for this fire zone. Significant impact on plant operations may result from fire damage to cables in this fire zone. The most important fire scenarios in this area involve fire damage to cables leading to failure of RHR loop B, core spray loop B, HPCI, containment venting, and division II ATTS.

5. **North Reactor Building Working Floor at Elevation 130 ft (1205F).**

CDF: 2.0E-7 per year

The north reactor building working floor at elevation 130 ft is another significant contributors to fire-induced CDF. There is a high level of traffic in this area. This fire zone is well maintained and no large quantity of transient combustibles were found. The division II reactor building 600-V ac MCC 1B (1R24-S012), 600-V ac MCC 1D (1R24-S014), and 250-V dc MCC 1B (1R24-S022) are located along the north reactor building wall on this elevation. Also located to the north of the drywell are one-half of the control rod drive (CRD) accumulators and associated scram equipment. There is a substantial amount of cabling and raceways above the MCC and CRD accumulator area. There are some vertical cable trays located behind the MCCs. Both the top and the bottom cable trays in this area are enclosed, and line-type thermal detectors are mounted underneath the bottom trays. Due to the close proximity, MCC fires can potentially damage nearby cables including those located above and behind the MCCs. Loss of the division II MCCs and nearby cables would not only fail the RHR loop B and core spray equipment, but also render containment venting capability unavailable. Compared to this fire zone, the CDF contribution from the south reactor building working floor at elevation 130 ft (1203F) is much lower because fire damage to cables near the division I MCCs in 1203F would not result in failure of the containment venting capability.

6. Division I Station Battery Room (1004).

CDF: 1.5E-7 per year

The major fire sources in this fire zone are battery and transient fires. This fire zone is a significant CDF contributor because the fire damage to cables routed through this area could affect the division I 125-V dc switchgear 1A and the 600-V ac MCC 1A (1R24-S011). Loss of division I dc switchgear could render the condensate system, feedwater system, RCIC, main condenser, and RHR loop A DHR unavailable. As compared to this fire zone, the fire-induced CDF contribution from division II station battery room (1005) is much lower since HPCI and RHR loop B DHR would be the only major mitigating systems affected.

7. Vertical Cable Chase (0040).

CDF: 1.0E-7 per year

This fire zone is a very small area. The major source of fire in this location is cable. This area is fully protected by an automatic water suppression system. As a result, the likelihood of fire propagation and damage to multiple raceways is limited. However, some of the raceways located in this fire zone could cause significant impact on the plant's ability to mitigate the consequence of fire damage. The most significant fire target is cable tray TEN7-01. If this cable tray is damaged, feedwater, the main condenser, containment venting, and DHR via RHR loop A could be lost.

8. Annunciator Room (1015).

CDF: 1.0E-7 per year

This room contains the annunciator logic cabinet, instrument bus 1A (1R25-S064), 120/208-V vital ac 1R25-S063, etc. The major fire source in this fire zone is cabinet fires, cable fires, and transient fires. The most significant fire target is cable tray TEN7-01. If this cable tray is damaged, feedwater, the main condenser, containment venting, and DHR via RHR loop A may be lost.

4.6.2.2.2 Selected Unit 2 Locations

For Unit 2, the top fire locations include the three 4-kV emergency switchgear rooms in the diesel generator building, the cable spreading room, and main control room. The following top eight Plant Hatch Unit 2 plant locations for the fire-induced CDF contribution are described below:

1. **4-kV Emergency Switchgear Rooms (2404, 2408, and 2409).**

   CDF: 2.5E-6 per year

   The description of these fire zones is similar to that for Unit 1.

2. **Cable Spreading Room (0024A).**

   CDF: 8.6E-7 per year

   The description of this fire zone is similar to that for Unit 1.

3. **Main Control Room (0024C).**

   CDF: 7.1E-7 per year

   The description of this fire zone is similar to that for Unit 1.

4. **Division I 600-V ac Switchgear Room (2016).**

   CDF: 2.9E-7 per year

   This fire zone contains division I 600-V ac switchgear, station service transformer 2C, RCIC panel, control building 125-V dc instrument cabinet 1A (2R25-S001), and cable raceways. The major fire sources are the 600-V switchgear, the station service transformer, and cable trays. This fire zone has a significant fire-induced CDF contribution because fire damage to equipment and cables in this room could result in the loss of the condensate system, feedwater system, main condenser, RCIC, core spray loop A, and RHRSW loop A. As compared to this location, the fire-induced CDF contribution from division II 600-V ac switchgear room (2017) is much lower since HPCI, core spray loop B, and RHRSW loop B would be the only major mitigating systems affected.

5. **East Cableway (2104).**

   CDF: 1.9E-7 per year

   The description of this fire zone is similar to that for Unit 1, except that the most important fire scenarios in this area involve fire damage to cables leading to a stuck open SRV and failure of division II ATTS, HPCI, RCIC, RHR loop B, containment venting, and drywell spray.

6. **Division I Station Battery Room (2004).**

CDF: 1.4E-7 per year

The description of this fire zone is similar to that for Unit 1.

7. **South Reactor Building Working Floor at Elevation 130 ft (2205F).**

CDF: 9.5E-8 per year

The south reactor building working floor at elevation 130 ft is another significant contributor to fire-induced CDF. There is a high level of traffic in this area. The fire zone is well maintained, and no large quantity of transient combustibles were found. The division II reactor building 600-V ac MCC 2B (2R24-S012), 600-V ac MCC 2D (2R24-S014), and 250-V dc MCC 2B (2R24-S022) are located along the south reactor building wall on this elevation. Also located to the south of the drywell are one-half of the CRD accumulators and the associated scram equipment. There is a substantial amount of cabling and raceways above the MCC and CRD accumulator area. There are some vertical cable trays located behind the MCCs. Both the top and the bottom cable trays in this area are enclosed, and line-type thermal detectors are mounted underneath the bottom trays. Due to close proximity, MCC fires can potentially damage nearby cables including those located above and behind the MCCs. Loss of the division II MCCs and nearby cables could not only fail HPCI and the division II RHR and core spray equipment, but also cause a stuck open SRV and render the containment venting capability unavailable.

8. **North Reactor Building Working Floor at Elevation 130 ft (2203F).**

CDF: 7.0E-8 per year

This fire zone is similar to 2205F. The division I reactor building 600-V ac MCC 2C (2R24-S011), 2R24-S013, 2R24-S018A, and 2R24-S018B, and 250-V dc MCC 2A (2R24-S021) are located on this elevation. Loss of the division II MCCs and nearby cables could fail RCIC, division I RHR and core spray equipment, and containment venting capability. Compared to 2205F, the fire-induced CDF contribution from this fire zone is lower because fire damage to cables nearby the division I MCCs would not result in a stuck open SRV.

4.6.3 UNCERTAINTY ANALYSIS

Assessing the frequency of rare events, such as the core damage sequences considered in a fire analysis, is subject to significant uncertainties. In view of this, the expression of uncertainty is a fundamental consideration in generating and presenting the risk analysis results. The Plant Hatch fire analysis inherently has many sources of uncertainty. The format for conveying the analysts' confidence in the results and displaying the risk parameter (e.g., fire-induced CDF) is to embed the frequency of this high consequence, low probability event into a probability distribution.

The probability distributions for the frequencies of fire-induced core damage at Unit 1 and Unit 2 are shown in figure 4.6-1. The median (50th percentile) frequency for fire-induced core damage for Unit 1 and Unit 2 are approximately 5.9E-06 and 4.0E-06 events per year, respectively. For Unit 1 and Unit 2, there is a roughly 50 percent chance that the fire-induced CDF is < 5.9E-06 and 4.0E-06 events per year, respectively, and a 50 percent chance that the frequency is greater. The mean frequency is about 7.5E-06 events per year for Unit 1 and 5.4E-06 for Unit 2.

These probability distributions show that there is a 90 percent confidence that the fire-induced CDFs for Unit 1 vary from 1.9E-06 events per year to 1.6E-05 events per year and for Unit 2 from 1.3E-06 events per year to 1.3E-05 events per year. They reflect the available state of knowledge about the current models, analyses, and data used in the analysis. The highest values in the range of the probability distribution correspond to the small chance that the most pessimistic assumptions that are considered in this analysis apply, combined with the possibility that the most pessimistic data also apply. Conversely, the lowest values correspond with the small chance that the most optimistic assumptions and data apply. There is a small chance (5 percent) that the fire-induced CDF is less than the lower bound, and a 5 percent chance that the CDF associated with the fire scenarios analyzed is greater than the upper bound.

REFERENCES

1. PLG, Inc., "RISKMAN® PSA Workstation Software," Version 5, 1994.

2. Edwin I. Hatch Nuclear Plant Individual Plant Examination (IPE), December 1992.

Table 4.6-1 Summary of Plant Hatch Fire Risk Analysis

Description	Unit 1	Unit 2
Total CDF for Internal Events (per Year)	2.1E-05	2.2E-05
Detailed Analysis Subscenario Total CDF (per Year)	7.4E-06	5.3E-06
Percent Internal Events CDF of Detailed Analysis Subscenarios	35%	24%
Total CDF of Scenarios Screened from Spatial Interactions Analysis Quantitative Screening (per Year)	1.5E-07	8.0E-08
Percent Internal Events CDF of Screened Scenarios	0.69%	0.36%
Unit Total Fire-Induced Core Damage Risk (Events per Year)	7.5E-06	5.4E-06
Percent Internal Events CDF of Total Fire Risk	36%	24%

Table 4.6-2 Rank-Ordered List of Leading Fire Risk Scenarios for Unit 1
(Sheet 1 of 6)

Rank No.	Fire Source and Fire Damage	Guaranteed Failures	Major Recovery Actions for Fire Damage	Most Likely Nonfire Failures	CDF (per Year)
1	0024A-L-C11: Small Cable Fire in Cable Spreading Room • One of cable trays (TFA7-1,2,3,4; TFB7-1,2; TFE7-1; TFU8-3,4,5,6; TFY8-2,3,6,7,8,9,10; TFZ7-2,5; TME7-5,6,7; TMS8-1,5; TMT7-18,19,25; TMU7-1; or TSJ8-1) fails. • TFZ7-2: Buses 1E and 1G, loop A RHR and core spray, hardened vent, drywell spray, 1/2 CI valves in several penetrations (X-14, X-18, X-19, X-205, etc), 2/2 CI valves in penetration X-26, and 1/2 RHR S/D cooling suction isolation valves fail.	PSW pumps A and B, condensate, feedwater, main condenser, HPCI (long term), RCIC (long term), loop B CS, RHR pump B, and RHRSW loop B.	Manually control buses 1E and 1G from diesel generator building switchgear rooms.	Failure of recovery actions.	1.4E-06
2	1412-L-G1: Large Switchgear Fire in Switchgear 1E Room • Cable trays and conduits fail. • Bus 1E, diesel battery A 125V DC panel, and offsite power supply cables from startup transformers 1D and 1C fail.	Offsite power supply to buses 1E, 1F, and 1G, PSW pumps A and C, CS pump A, RHR pumps A and D, RHRSW pumps A and C, condensate, feedwater, and main condenser.	–	Failure of diesel generators 1B and 1C (diesel generator equipment, cooling water, control power).	7.9E-07
3	1404-L-G1: Large Switchgear Fire in Switchgear 1G Room • Cable trays and conduits fail. • Bus 1G, diesel battery C 125V DC panel, and offsite power supply cables from startup transformers 1D and 1C fail.	Offsite power supply to buses 1E, 1F, and 1G, PSW pumps B and D, CS pump B, RHR pumps B and C, RHRSW pumps B and D, condensate, feedwater, and main condenser.	–	Failure of diesel generators 1A and 1B (diesel generator equipment, cooling water, control power).	7.7E-07
4	1408-L-G1: Large Switchgear Fire in Switchgear 1F Room • Cable trays and conduits fail. • Bus 1F, diesel battery B 125V DC panel, and offsite power supply cables from startup transformers 1D and 1C fail.	Offsite power supply to buses 1E, 1F, and 1G, PSW pumps C and D, RHR pumps C and D, RHRSW pump C, condensate, feedwater, and main condenser.	–	Failure of diesel generators 1A and 1C (diesel generator equipment, cooling water, control power).	4.5E-07

Table 4.6-2 Rank-Ordered List of Leading Fire Risk Scenarios for Unit 1 (Sheet 2 of 6)

Rank No.	Fire Source and Fire Damage	Guaranteed Failures	Major Recovery Actions for Fire Damage	Most Likely Nonfire Failures	CDF (per Year)
5	1412-L-B3: Large Transient Fire in Switchgear 1E Room • Cable trays and conduits fail. • Bus 1E, diesel battery A 125V DC panel, and offsite power supply cables from startup transformers 1D and 1C fail.	Offsite power supply to buses 1E, 1F, and 1G, PSW pumps A and C, CS pump A, RHR pumps A and D, RHRSW pumps A and C, condensate, feedwater, and main condenser.	--	Failure of diesel generators 1B and 1C (diesel generator equipment, cooling water, control power).	3.6E-07
6	1404-L-B3: Large Transient Fire in Switchgear 1G Room • Cable trays and conduits fail. • Bus 1G, diesel battery C 125V DC panel, and offsite power supply cables from startup transformers 1D and 1C fail.	Offsite power supply to buses 1E, 1F, and 1G, PSW pumps B and D, CS pump B, RHR pumps B and C, RHRSW pumps B and D, condensate, feedwater, and main condenser.	--	Failure of diesel generators 1A and 1B (diesel generator equipment, cooling water, control power).	3.6E-07
7	0024A-L-C7: Small Cable Fire in Cable Spreading Room • One of cable trays (TFG9-1,2,3,7; TME7-10; TMI9-1,2,4; TMJ7-11,12; or TMT7-35) fails. • TMT7-35: TBPSW, division I ATTS, HPCI, RCIC, CS loop A, drywell spray, and 1/2 CI valves for penetration X-205 fail.	Condensate, feedwater, and main condenser.	--	Failure of vessel depressurization.	2.2E-07
8	1408-L-B3: Large Transient Fire in Switchgear 1F Room • Cable trays and conduits fail. • Bus 1F, diesel battery B 125V DC panel, and offsite power supply cables from startup transformers 1D and 1C fail.	Offsite power supply to buses 1E, 1F, and 1G, PSW pumps C and D, RHR pumps C and D, RHRSW pump C, condensate, feedwater, and main condenser.	--	Failure of diesel generators 1A and 1C (diesel generator equipment, cooling water, control power).	2.1E-07
9	MCR-13: Cabinet Fire in Main Control Room (Panel 1H11P652) • MCR Panel 1H11P652 fails. • Buses 1E, 1F, and 1G, diesel generators 1A, 1B, and 1C, 600V bus 1C, station service transformers 1C, 1D, and 1CD, PSW turbine building isolation valves, and MCCs fail.	Condensate, feedwater, main condenser, HPCI, RCIC, RHR, CS, RHRSW, and hardened vent.	Manually operating equipment from 4-kV and 600V proceduralized switchgear rooms.	Failure of recovery actions.	1.5E-07

Table 4.6-2 Rank-Ordered List of Leading Fire Risk Scenarios for Unit 1
(Sheet 3 of 6)

Rank No.	Fire Source and Fire Damage	Guaranteed Failures	Major Recovery Actions for Fire Damage	Most Likely Nonfire Failures	CDF (per Year)
10	1404-L-D1: DC Cabinet Fire in Switchgear 1G Room • Cable trays and conduits fail. • Bus 1G and diesel battery C 125V DC panel fail.	PSW pumps B and D, HPCI, CS pump B, RHR pumps B and C, RHRSW pumps B and D, condensate, feedwater, and main condenser.	–	• Failure of RCIC and vessel depressurization. • Failure of decay heat removal via loop A RHR and hardened vent. • Failure of division I PSW and hardened vent.	1.2E-07
11	1104A-L-A3: Large Cable Fire in East Cableway • Cable trays TCG8-04, TCJ8-02, TCK8-02, TCL8-01, TCM8-01, TCV0-02, and TCW0-01 fail. • Division II ATTS, HPCI, CS loop B, RHR loop B, room coolers for reactor building SE and NE corner rooms (1203B and 1205B), hardened vent, loop B drywell spray, and 1/2 CI valves for penetration X-14 fail.	N/A	Recovery from loss of room cooling; e.g., reduce number of operating pumps.	Failure of decay heat removal via RHR loop A and failure of main condenser (may be caused by failure of condensate system, normal AC power, main condenser itself, bypass valves, TBPSW, etc.).	1.0E-07
12	1104A-L-A1: Large Cable Fire in East Cableway • Cable trays TCC4-01,02; TCJ8-01, TCV0-02, THD4-01, and THJ8-01 fail. • Reactor building MCC 1B (1R24-S012), division II ATTS, auto-closure of 1/2 TB isolation valves in both reactor building PSW headers, HPCI, RHR loop B, room coolers for reactor building SE and NE corner rooms (1203B and 1205B), hardened vent, and 1/2 CI valves for penetration X-14 fail.	CS loop B.	Recovery from loss of room cooling; e.g., reduce number of operating pumps.	Failure of decay heat removal via RHR loop A and failure of main condenser (may be caused by failure of condensate system, normal AC power, main condenser itself, bypass valves, TBPSW, etc.).	1.0E-07
13	1404-L-G2: Large Switchgear Fire in Switchgear 1G Room • Cable trays and conduits fail. • Bus 1G and diesel battery C 125V DC panel fail.	PSW pumps B and D, HPCI, CS pump B, RHR pumps B and C, RHRSW pumps B and D, condensate, feedwater, and main condenser.	–	• Failure of RCIC and vessel depressurization. • Failure of decay heat removal via loop A RHR and hardened vent. • Failure of division I PSW and hardened vent.	9.7E-08

Table 4.6-2 Rank-Ordered List of Leading Fire Risk Scenarios for Unit 1 (Sheet 4 of 6)

Rank No.	Fire Source and Fire Damage	Guaranteed Failures	Major Recovery Actions for Fire Damage	Most Likely Nonfire Failures	CDF (per Year)
14	0040-L-A1: Small Cable Fire in Vertical Cable Chase • Cable tray TEN7 fails. • MSIV closure and loop A RHRSW, hardened vent, 1/2 room coolers for HPCI room, 1/2 room coolers for reactor building SE and NE corner rooms (1203B and 1205B), hardened vent, auto-closure of 1/2 reactor building PSW header isolation valves for both headers, and 1/2 CI valves for penetration X-205 fail.	Feedwater, main condenser, and decay heat removal via loop A RHR.	Manually open loop A RHRSW flow control valve. (Not proceduralized)	Failure of recovery and decay heat removal via loop B RHR.	7.3E-08
15	1205A-L-C1: Small Cable Fire in Reactor Building South Torus Chamber • Cable trays RAA8-1 and RBA4-1 fail. • HPCI, RHR loop B, room coolers for reactor building SE and NE corner rooms (1203B and 1205B), and hardened vent fail.	N/A	Recovery for loss of room cooling; e.g., reduce number of operating pumps.	• Failure of decay heat removal via RHR loop A and failure of main condenser (may be caused by failure of condensate system, normal AC power, main condenser itself, bypass valves, TBPSW, etc.). • Operator failure to align for decay heat removal.	7.3E-08
16	1004-L-A1: Large Battery Fire in Division I Station Battery Room • Division I station battery, cable trays, and conduits fail. • Division I DC supply from 125/250V switchgear 1A, reactor building 600V AC MCC 1C (1R24-S011), RCIC, and loop A LPCI injection valve fail.	Condensate, feedwater, main condenser, CS loop A, loop A RHR torus cooling valves, and loop A RHRSW flow control valve.	Manually open loop A RHRSW flow control valve and RHR torus cooling valves. (Not proceduralized)	Recovery succeeds: • HPCI failure followed by failure of vessel depressurization. • Division II 125/250V station DC supply fails. • HPCI, loop B LPCI, and CS loop B fail. Recovery fails: • HPCI failure followed by failure of vessel depressurization. • Failure of decay heat removal via loop B RHR and hardened vent. • Division II 125/250V station DC supply fails.	7.1E-08

Table 4.6-2 Rank-Ordered List of Leading Fire Risk Scenarios for Unit 1 (Sheet 5 of 6)

Rank No.	Fire Source and Fire Damage	Guaranteed Failures	Major Recovery Actions for Fire Damage	Most Likely Nonfire Failures	CDF (per Year)
17	1412-L-D1: DC Cabinet Fire in Switchgear 1E Room • Cable trays and conduits fail. • Bus 1E and diesel battery A 125V DC panel fail.	PSW pumps A and C, CS pump A, RHR pumps A and D, RHRSW pumps A and C, condensate, feedwater, and main condenser.	–	• Failure of RCIC, HPCI, and vessel depressurization. • Failure of decay heat removal via loop B RHR and hardened vent. • Failure of division II PSW and hardened vent.	7.0E-08
18	1205F-L-E3: Large MCC Fire in North Reactor Building Working Floor at Elevation 130 ft • Reactor building MCC 1R24-S014 and cable trays RAA0-01, RAA2-01, RAA6-01, RCA8-01, REA4-01, REA8-01, and RFA8-01 fail. • HPCI, CS loop B, RHR loop B, room coolers for reactor building SE and NE corner rooms (1203B and 1205B), hardened vent, drywell spray loop B, and 1/2 CI valves in penetration X-14 fail.	N/A	Recovery for loss of room cooling; e.g., reduce number of operating pumps.	• Failure of decay heat removal via RHR loop A and failure of main condenser (may be caused by failure of condensate system, normal AC power, main condenser itself, bypass valves, TBPSW, etc.). • Operator failure to align for decay heat removal.	6.4E-08
19	1205F-L-E2: Large MCC Fire in North Reactor Building Working Floor at Elevation 130 ft • MCC 1R24-S022 and cable trays RAA6-01, RCA8-01, RDA8-03, REA4-01, and RFA4-02 fail. • 600V reactor building MCC 1B (1R24-S012), reactor building DC MCC 1B (1R24-S022), HPCI, CS loop B, RHR loop B, room coolers for reactor building SE and NE corner rooms (1203B and 1205B), hardened vent, drywell spray loop B, 1/2 CI valves in penetration X-14, and 1/2 RHR S/D cooling suction isolation valves fail.	N/A	Recovery for loss of room cooling; e.g., reduce number of operating pumps.	• Failure of decay heat removal via RHR loop A and failure of main condenser (may be caused by failure of condensate system, normal AC power, main condenser itself, bypass valves, TBPSW, etc.). • Operator failure to align for decay heat removal.	6.4E-08

Table 4.6-2 Rank-Ordered List of Leading Fire Risk Scenarios for Unit 1 (Sheet 6 of 6)

Rank No.	Fire Source and Fire Damage	Guaranteed Failures	Major Recovery Actions for Fire Damage	Most Likely Nonfire Failures	CDF (per Year)
20	1205F-L-E1: Large MCC Fire in North Reactor Building Working Floor at Elevation 130 ft • MCC 1R24-S012 and cable trays RDA8-02, RFA4-02, and RGA4-01 fail. • 600V reactor building MCC 1B (1R24-S012), HPCI, CS loop B, RHR loop B, 1/2 room coolers for reactor building SE and NE corner rooms (1203B and 1205B), hardened vent, drywell spray loop B, 1/2 CI valves in penetration X-14, and 1/2 RHR S/D cooling suction isolation valves fail.	N/A	--	• Failure of decay heat removal via RHR loop A and failure of main condenser (may be caused by failure of condensate system, normal AC power, main condenser itself, bypass valves, TBPSW, etc.). • Operator failure to align for decay heat removal.	6.3E-08

Table 4.6-3 Rank-Ordered List of Leading Fire Risk Scenarios for Unit 2 (Sheet 1 of 5)

Rank No.	Fire Source and Fire Damage	Guaranteed Failures	Major Recovery Actions for Fire Damage	Most Likely Nonfire Failures	CDF (per Year)
1	2409-L-G2: Large Switchgear Fire in Switchgear 2G Room • Cable trays and conduits fail. • Bus 2G, diesel battery C 125V DC panel, and offsite power supply cables from startup transformers 2D and 2C fail.	Offsite power supply to buses 2E, 2F, and 2G, PSW pumps B and D, CS pump B, RHR pumps B and C, RHRSW pumps B and D, condensate, feedwater, and main condenser.	--	Failure of diesel generators 2A and 1B (diesel generator equipment, cooling water, control power).	8.4E-07
2	0024A-L-C4: Small Cable Fire in Cable Spreading Room • One of cable trays (CAA801; DAB702,703; DBB701,702,703; DBD701; DBG701; DCB701,702,703; DCG701; DEA801,802; DEB801; DED801,802,807; DGC801; DGD801,802,803, 804,805; DGF801; or DJA701) fails. • DGF801: Buses 2G and 2D (600V), division II PSW, division II ATTS, SORV, HPCI, RCIC, RHR pump C, RHR loop B, CS loop B, 1/2 room coolers for reactor building SE (2205B) and NE (2203B) corner rooms, hardened vent, drywell spray, 1/2 CI valves in several penetrations (X-14, X-18, X-19, X-205, etc), 2/2 CI valves in penetration X-26, and 1/2 RHR S/D cooling suction isolation valves fail.	Condensate, feedwater, and main condenser.	Manually open loop B RHRSW flow control and RHR torus cooling valves.	• Failure of decay heat removal via loop A RHR. • Failure of vessel depressurization.	7.1E-07
3	2404-L-G2: Large Switchgear Fire in Switchgear 2E Room • Cable trays and conduits fail. • Bus 2E, diesel battery A 125V DC panel, and offsite power supply cables from startup transformers 2D and 2C fail.	Offsite power supply to buses 2E, 2F, and 2G, PSW pumps A and C, CS pump A, RHR pumps A and D, RHRSW pumps A and C, condensate, feedwater, and main condenser.	--	Failure of diesel generators 1B and 2C (diesel generator equipment, cooling water, control power).	6.9E-07
4	2408-L-G2: Large Switchgear Fire in Switchgear 2F Room • Cable trays and conduits fail. • Bus 2F, diesel battery B 125V DC panel, and offsite power supply cables from startup transformers 2D and 2C fail.	Offsite power supply to buses 2E, 2F, and 2G, PSW pumps C and D, RHR pumps C and D, RHRSW pump C, condensate, feedwater, and main condenser.	--	Failure of diesel generators 2A and 2C (diesel generator equipment, cooling water, control power).	4.2E-07

Table 4.6-3 Rank-Ordered List of Leading Fire Risk Scenarios for Unit 2
(Sheet 2 of 5)

Rank No.	Fire Source and Fire Damage	Guaranteed Failures	Major Recovery Actions for Fire Damage	Most Likely Nonfire Failures	CDF (per Year)
5	2016-L-A1: Large Station Service Transformer Fire in Division I 600V AC Switchgear Room • 600V AC station service transformer 2C and cable tray 2CSA701. • 600V bus 2C, division I reactor building MCC 2C (2R24-S011), division I essential cabinet 2C (2R25-S036), 125/250V division I station DC power, division I ATTS, and RCIC fail.	Condensate, feedwater, main condenser, CS loop A, loop A RHRSW flow control valve, and loop A RHR torus cooling valves.	Manually open loop A RHRSW flow control valve and RHR torus cooling valves.	Recovery succeeds: • Failure of HPCI followed by vessel depressurization failure. • Failure of 600V bus 2D or its power supply. • 125/250V division II station DC power. • Failure of decay heat removal via RHR and hardened vent (instrument air or vent valve failure). Recovery fails: • Failure of decay heat removal via loop B RHR and hardened vent (instrument air or vent valve failure). • Failure of HPCI followed by vessel depressurization failure.	2.0E-07
6	2104A-L-A3: Cable Fire in East Cableway • Cable trays 2CAA801, 2CAB801, 2CCA001, 2CCD001, and 2CHA001 fail. • Division II ATTS, SORV, HPCI, RCIC, RHR loop B, hardened vent, and drywell spray fail.	Feedwater and main condenser.	–	Failure of decay heat removal via RHR loop A.	1.5E-07
7	MCR-13: Cabinet Fire in Main Control Room (Panel 2H11P652) • MCR Panel 2H11P652 fails. • Buses 2E, 2F, and 2G, diesel generators 2A, 1B, and 2C, 600V bus 2C, station service transformers 2C, 2D, and 2CD, PSW turbine building isolation valves, and MCCs fail.	Condensate, feedwater, main condenser, HPCI, RCIC, RHR, CS, RHRSW, and hardened vent.	Manually operating equipment from 4-kV and 600V switchgear rooms.	Failure of recovery actions.	1.5E-07

Table 4.6-3 Rank-Ordered List of Leading Fire Risk Scenarios for Unit 2 (Sheet 3 of 5)

Rank No.	Fire Source and Fire Damage	Guaranteed Failures	Major Recovery Actions for Fire Damage	Most Likely Nonfire Failures	CDF (per Year)
8	2004-L-A1: Large Battery Fire in Division I Station Battery Room • Battery S001A, cable trays 2BMA301 and 2BNA501, and conduit 2E11639 fail. • Division I DC supply from 125/250V switchgear 2A, reactor building 600V AC MCC 2C (2R24-S011), division I ATTS, and RCIC fail.	Condensate, feedwater, main condenser, CS loop A, loop A RHR torus cooling valves, and loop A RHRSW flow control valve.	Manually open loop A RHRSW flow control valve and RHR torus cooling valves.	Recovery succeeds: • HPCI failure followed by failure of vessel depressurization. • 600V bus 2D or its power supply. • Division II 125/250 station DC supply fails. Recovery fails: • HPCI failure followed by failure of vessel depressurization. • 600V bus 2D or its power supply. • Failure of decay heat removal via loop B RHR and hardened vent. • Division II 125/250 station DC supply.	1.2E-07
9	2409-L-G1: Large Switchgear Fire in Switchgear 2G Room • Cable trays and conduits fail. • Bus 2G and diesel battery C 125V DC panel fail.	PSW pumps B and D, HPCI, CS pump B, RHR pumps B and C, RHRSW pumps B and D, condensate, feedwater, and main condenser.	--	• Failure of RCIC and vessel depressurization. • Failure of decay heat removal via loop A RHR and hardened vent. • Failure of division I PSW and hardened vent.	1.1E-07
10	2205F-L-F1: Large HVAC Fire in South Reactor Building Working Floor at Elevation 130 ft • Cable trays RAB001, RBC804, and REA801,802 fail. • Division II ATTS, SORV, HPCI, CS loop B, RHR loop B, 1/2 room coolers for reactor building NE corner room (2203B), hardened vent, drywell spray, and 1/2 CI valves in penetrations X-14 and X-18 fail.	Feedwater and main condenser.	--	Failure of decay heat removal via RHR loop A.	6.6E-08

Table 4.6-3 Rank-Ordered List of Leading Fire Risk Scenarios for Unit 2 (Sheet 4 of 5)

Rank No.	Fire Source and Fire Damage	Guaranteed Failures	Major Recovery Actions for Fire Damage	Most Likely Nonfire Failures	CDF (per Year)
11	0024A-L-C5: Small Cable Fire in Cable Spreading Room • One of cable trays (CCD001; DBA701; DBF701; DDG901,902,903,904,905; DED805,806; DEE801; DGB801; DGF802; DHD004,005,006; DHF001; DNA701,702; or DPA901) fails. • DNA701: Division I ATTS, SORV, HPCI, RCIC, CS loop A, LPCI loop A, RHRSW loop A, drywell spray, 1/2 CI valves for several penetrations (X-14, X-18, X-19, X-205, etc.), 2/2 CI valves for penetration X-26, and 1/2 RHR S/D cooling suction isolation valves fail.	Feedwater, main condenser, and RHR loop A.	--	• Failure of decay heat removal via RHR loop B and hardened vent. • Failure of vessel depressurization. • Failure of all low pressure injection systems. • Failure of condensate and low pressure injection permissive signal.	5.9E-08
12	2404-L-G1: Large Switchgear Fire in Switchgear 2E Room • Cable trays and conduits fail. • Bus 2E and diesel battery A 125V DC panel fail.	PSW pumps A and C, CS pump A, RHR pumps A and D, RHRSW pumps A and C, condensate, feedwater, and main condenser.	--	• Failure of RCIC, HPCI, and vessel depressurization. • Failure of decay heat removal via loop B RHR and hardened vent. • Failure of division II PSW and hardened vent.	5.4E-08
13	2016-L-B1: Large Switchgear Fire in Division I 600V AC Switchgear Room • 600V AC bus 2C, RCIC panel 2H21P051, 125V DC cabinet 2R25-S001, cable tray 2CSA701, and conduits 2E12018, 2E12019, 2E12020, 2E12021, 2E12024, 2E12025, 2E12079, 2E12133, 2E12134, and 2E12135. • Division I reactor building MCC 2C (2R24-S011), division I essential cabinet 2C (2R25-S036), 125/250V division I station DC power, division I ATTS, and RCIC fail.	Condensate, feedwater, main condenser, CS loop A, loop A RHRSW flow control valve, and loop A RHR torus cooling valves.	Manually open loop A RHRSW flow control valve and RHR torus cooling valves.	Recovery succeeds: • Failure of HPCI followed by vessel depressurization failure. • Failure of 600V bus 2D or its power supply. • 125/250V division II station DC power. • Failure of decay heat removal via RHR and hardened vent (instrument air or vent valve failure). Recovery fails: • Failure of decay heat removal via loop B RHR and hardened vent (instrument air or vent valve failure). • Failure of HPCI followed by vessel depressurization failure.	5.0E-08

Table 4.6-3 Rank-Ordered List of Leading Fire Risk Scenarios for Unit 2 (Sheet 5 of 5)

Rank No.	Fire Source and Fire Damage	Guaranteed Failures	Major Recovery Actions for Fire Damage	Most Likely Nonfire Failures	CDF (per Year)
14	2203A-L-C1: Small Cable Fire in Reactor Building North Torus Chamber • Cable trays 2RAA701,702,703; 2RAB701; and 2RAC701 fail. • RCIC, RHR loop A, 1/2 room coolers for reactor building SE corner room (2205B), hardened vent, and 1/2 CI valves in penetrations X-18, X-19, and X-205 fail.	N/A	–	• Failure of decay heat removal via RHR loop B and failure of main condenser (may be caused by failure of condensate system, normal AC power, main condenser itself, bypass valves, TBPSW, etc.). • Operator failure to align for decay heat removal.	4.6E-08
15	2203F-L-E2: Large MCC Fire in North Reactor Building Working Floor at Elevation 130 ft • MCCs 2R24-S011, 2R24-S013, and 2R24-S021 and cable trays 2RAB701,702,703; RAC701,702; RCA702,703,704; and RDA702,703,704 fail. • 600V reactor building MCC 2C (2R24-S011), reactor building DC MCC 2A (1R24-S021), RCIC, CS loop A, RHR loop A, hardened vent, 1/2 drywell spray loops, 1/2 CI valves in penetrations X-14, X-18, X-19, and X-205, and 1/2 RHR S/D cooling suction isolation valves fail.	N/A	–	• Failure of decay heat removal via RHR loop B and failure of main condenser (may be caused by failure of condensate system, normal AC power, main condenser itself, bypass valves, TBPSW, etc.). • Operator failure to align for decay heat removal.	4.1E-08
16	2205A-L-C1: Small Cable Fire in Reactor Building South Torus Chamber • Cable trays 2RBA803,804; 2RBC801; and 2RBD801 fail. • PSW pump B, HPCI, CS loop B, RHR loop B, 1/2 room coolers for reactor building NE corner rooms (1203B), hardened vent, 1/2 drywell spray loops, and 1/2 CI valves in penetrations X-18 and X-205 fail.	N/A	–	• Failure of decay heat removal via RHR loop A and failure of main condenser (may be caused by failure of condensate system, normal AC power, main condenser itself, bypass valves, TBPSW, etc.). • Failure of division I 600V AC switchgear or 125/250V station DC power and failure to recover main condenser. • Operator failure to align for decay heat removal.	4.0E-08

Table 4.6-4 Dominant Fire Risk Locations for Unit 1

Rank	Fire Zone	Fire Zone Description	Fire Zone Total UCDF (1/yr)	Percentage Internal CDF
1	0024A	Cable Spreading Room	1.93E-06	9.17
2	1404	4-kV Switchgear Room 1G	1.45E-06	6.90
3	1412	4-kV Switchgear Room 1E	1.38E-06	6.58
4	0024C	Main Control Room	7.10E-07	3.38
5	1408	4-kV Switchgear Room 1F	7.04E-07	3.35
6	1104A	East Cableway	2.05E-07	0.98
7	1205F	Reactor Building North Working Floor on Elevation 130 ft (Control Rod Drive Area)	2.03E-07	0.97
8	1004	Station Battery Room 1A - Division I	1.49E-07	0.71
9	0040	Vertical Cable Chase	1.04E-07	0.50
10	1015	Annunciator Room	9.94E-08	0.47
11	1016	West 600V Switchgear Room 1C - Division I	5.76E-08	0.27
12	0014K	Control Building North and South Corridor on Elevation 130 ft	5.73E-08	0.27
13	1018	West DC Switchgear Room 1A - Division I	4.26E-08	0.20
14	0024B	Computer Room	4.23E-08	0.20

Table 4.6-5 Dominant Fire Risk Locations for Unit 2

Rank	Fire Zone	Fire Zone Description	Fire Zone Total UCDF (1/yr)	Percentage Internal CDF
1	2409	4-kV Switchgear Room 2G	1.10E-06	4.98
2	2404	4-kV Switchgear Room 2E	8.88E-07	4.04
3	0024A	Cable Spreading Room	8.55E-07	3.89
4	0024C	Main Control Room	7.10E-07	3.23
5	2408	4-kV Switchgear Room 2F	4.66E-07	2.12
6	2016	West 600V Switchgear Room 2C - Division I	2.89E-07	1.31
7	2104A	East Cableway	1.89E-07	0.86
8	2004	Station Battery Room 2A - Division I	1.40E-07	0.64
9	2205F	Reactor Building South Working Floor on Elevation 130 ft (Control Rod Drive Area)	9.51E-08	0.43
10	2203F	Reactor Building North Working Floor on Elevation 130 ft (Control Rod Drive Area)	7.03E-08	0.32
11	0024B	Computer Room	5.70E-08	0.26
12	2101ZZ	Turbine Building Areas on Elevations 130 ft and 147 ft	4.80E-08	0.22
13	2203A	Reactor Building North Torus Chamber	4.62E-08	0.21
14	0001	Control Building Working Floor and Corridor on Elevation 112 ft	4.49E-08	0.20
15	0040	Vertical Cable Chase	4.39E-08	0.20
16	2018	West DC Switchgear Room 2A - Division I	4.16E-08	0.19
17	2205A	Reactor Building South Torus Chamber	3.98E-08	0.18

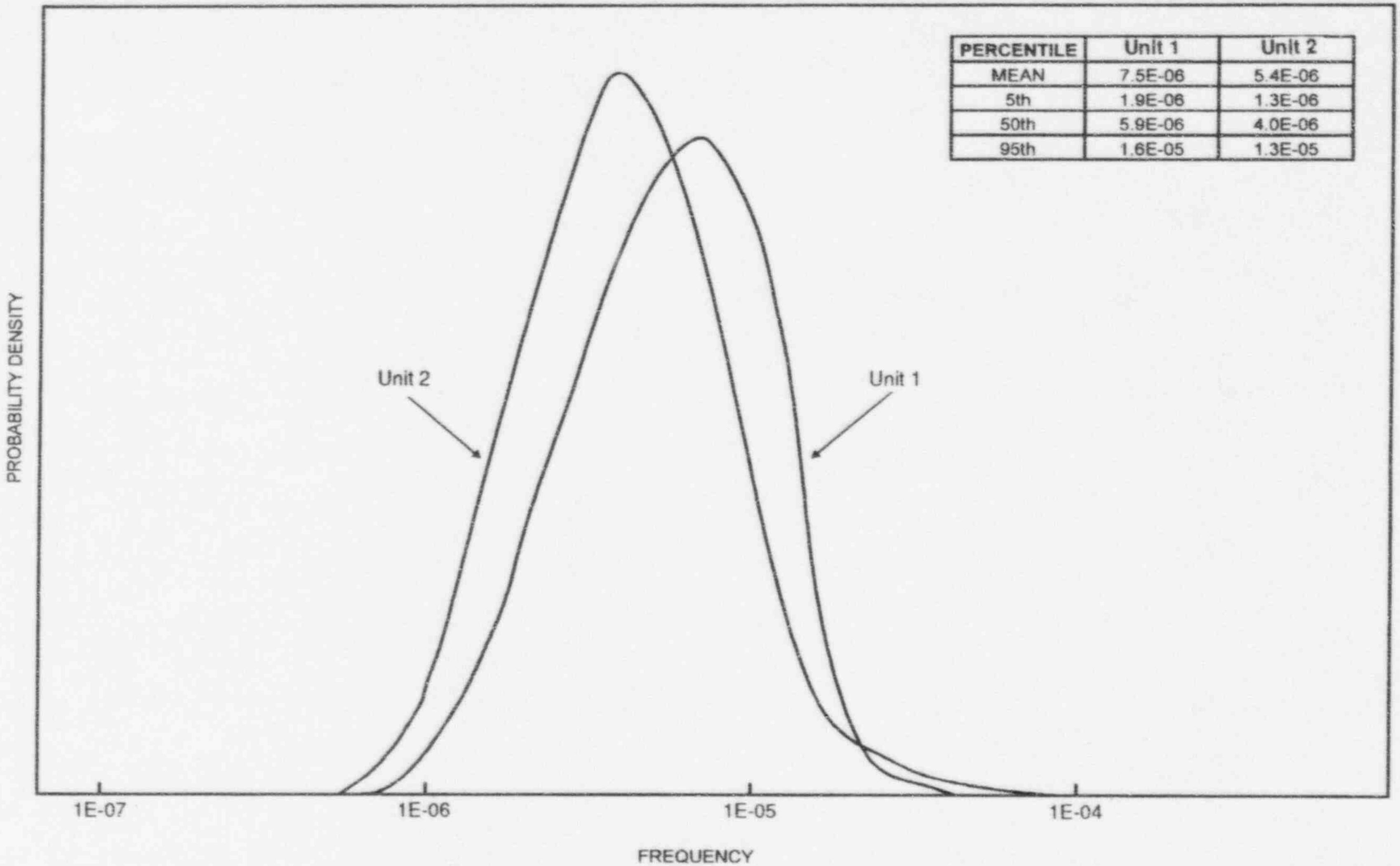


Figure 4.6-1 Uncertainty Distributions for Total Fire-Induced CDF

4.7 ANALYSIS OF CONTAINMENT PERFORMANCE

The Level II quantification for fire-induced core damage sequences was accomplished by linking the containment event tree (CET) developed for the internal events Individual Plant Examination (IPE) (Reference 1) with the Level I event tree models for fires, thereby eliminating the need to define Level I end states for linking with the CET. The event tree ENDBACK was used to group the core damage sequences into the various CET end states needed for the containment performance analysis. Sequences are grouped into accident classes at the beginning of the CET based on functional similarities. The CET end states are binned into plant damage states and release modes. Plant damage states define the state of systems and equipment important to the source-term analysis, and release modes characterize the sequence based on magnitude and timing of the release. The dependence of split fractions for many of the CET top events on Level I failures or success is facilitated via the use of this set of accident classes. Accordingly, the analysis of containment performance includes a review of the following:

1. Applicability of the internal events CET for use in fire-induced sequences.

2. Applicability of conditional CET top event split fractions developed for internal events to fire-induced sequences.

3. Applicability of CET end states as well as CET end state mapping to release modes and source term calculation bins previously developed for internal events for fire-induced sequences.

4. Applicability of the source term calculations and the post-severe-accident containment state.

The results of the Level II analysis associated with fires were compared to the results obtained for the IPE to determine whether the containment function is impaired by fires.

4.7.1 SYSTEMS AND STRUCTURES

The analysis of fires disclosed no unique containment structural failure modes. The failure of any containment system leading to complete or partial containment isolation failure, containment bypass, or loss of drywell spray or heat removal due to fires was evaluated in the same manner as the Level I portion of the analysis even though the containment system functions were modeled in the Level II CET. Aside from the potential impact on system or train recovery times due to the nature of the initiating event, once a Level I sequence is characterized in terms of the accident classes, the containment response is essentially independent of the initiating event cause; i.e., external or internal. Therefore, no special consideration was given to containment system or structural failures in the Level II model other than the fire impact on the containment systems.

In the detailed fire scenario analysis, the impact of fire damage on containment systems for each of the subscenarios developed was evaluated. This was performed by first identifying the control and power circuits whose failure is caused by fire damage and may completely or partially fail the containment systems (i.e., containment isolation valves, containment bypass isolation valves, drywell spray valves). The routing of these circuits through the various plant locations was then determined. For each detailed fire subscenario, the fire-damaged equipment and cable trays were analyzed to determine whether any of the containment systems were affected. This information was incorporated into the model quantification (including both Level I and Level II event trees) by using the appropriate containment top event split fraction values to reflect the fire impact on the containment systems. The assignment of fire-induced accident sequences to CET end state was accomplished using the same logic rules developed in the IPE for the CET.

4.7.2 LEVEL I/II INTERFACE CONSIDERATIONS

A review of the dominant core damage sequences resulting from fires did not identify any unique core damage state considerations. Therefore, the accident classes and Level I/II interface defined for the IPE are directly applicable to the fire IPEEE.

4.7.3 APPLICABILITY OF CONTAINMENT EVENT TREE TOP EVENTS

As indicated above, the analysis of fires did not identify any new containment failure modes. Therefore, no additional failure mode top events addressing fires are required in the CET.

In general, the recovery of systems is more difficult if failures of these systems are induced by fire; i.e., the time required for recovery is expected to be appreciably longer. However, since no credit was taken for recovery of systems after core damage occurred in the internal events IPE, no changes to the existing internal events CET were necessary to address the potentially longer recovery times associated with fire-induced events.

4.7.4 APPLICABILITY OF SOURCE TERM CALCULATIONS AND ASSOCIATED BINNING

The categories used to bin the source terms are provided in table 4.7-1 of this report. For each source term calculation bin, the frequency of representative functional sequences in the IPE was compared with the frequencies of the functional sequences in the fire IPEEE. If the IPE representative functional sequence frequency was higher, no additional source term was calculated for the IPEEE. The IPE source term was considered to envelop the source terms of the IPEEE functional sequences of the particular source term calculation bin. If the IPE representative functional frequency was lower, the two functional sequences were evaluated to determine which resulted in the higher source term. This was accomplished by either engineering judgment or new Modular Accident Analysis Program (MAAP) (Reference 2) runs.

As an additional verification of the applicability of each source term calculation bin, the frequency of the representative individual accident sequence of the IPE was compared with the frequencies of the individual accident sequences of the IPEEE. If the IPE representative accident sequence frequency was higher, no additional source term was calculated for the IPEEE because it was considered to envelop the source terms of the individual IPEEE accident sequences of the particular source term calculation bin. If the IPE representative accident sequence frequency was lower, the accident sequences were evaluated to determine whether the IPE or IPEEE sequence resulted in the higher source term. As indicated earlier, this was accomplished by either engineering judgment or new MAAP runs.

Next, the source term calculation bins were grouped by the assigned release categories. For each release category, the total radionuclide release frequency and the conditional probability of release, given core damage for the fire IPEEE, were compared with those calculated in the IPE. The unassigned portion of the fire IPEEE core damage frequency (CDF) was proportionately divided among the four release categories. The definitions for these release categories are found in table 4.7-1. The same plant damage states used in the IPE were used in the fire IPEEE. Finally, the CDF and the percent of fire IPEEE and IPE CDF attributed to each containment state were reviewed for insights.

As discussed previously, the fire analysis did not identify any containment failure modes other than those identified for the internal events IPE. All fire-induced containment isolation failures and bypass events are adequately enveloped by release categories and source terms previously developed for internal events IPE. Releases of material to the environment are not expected to be impacted by the thermal effects of the fires. Thus, no additional release categories needed to be identified or source terms calculated for fires considerations.

4.7.5 LEVEL II RESULTS FOR FIRES

Tables 4.7-2 and 4.7-3 provide a comparison of the source term binning for release frequencies from the IPE and IPEEE fire analyses. These include the frequencies of functional sequence for both analyses and the frequencies of the particular accident sequences used to calculate the representative source term for the source term calculation bin in the IPE. The final column of tables 4.7-2 and 4.7-3 indicates the highest IPEEE fire sequence frequency of each accident class/subclass and CET end state calculated in the IPEEE that is > 1.0E-10 per year.

The most important benchmark for the risk of early offsite health effects is the frequency of Release Category D. This category contains sequences involving the release of > 10 percent of the volatiles from the containment. As shown in table 4.7-4, Release Category D accounts for approximately 10 percent of the CDF associated with fires for Unit 1 and 8 percent for Unit 2. Release Category D accounted for 23 percent of the CDF associated with the IPE for Unit 1 and 22 percent for Unit 2. Thus, the conditional frequency of large releases; i.e., "poor containment performance," associated with fires is determined to be lower than the IPE.

Table 4.7-5 provides the CDF and percent of CDF attributed to each containment state for both the fire IPEEE and the IPE. The unassigned portion of the IPE and fire IPEEE CDF was proportionately divided among the seven containment states. The CDF for each IPE containment state is larger than the corresponding CDF for the fire IPEEE. However, the percent of the CDF is higher for the fire IPEEE containment isolation (CI) failure, overpressure (OP) failure, and overtemperature (OT) failure. For the IPE, accident initiators, except for perhaps station blackout events, do not disable containment systems such as suppression pool cooling and drywell spray. However, for fire IPEEE, the accident initiator is a fire which could disable systems used for containment isolation or heat removal. A review of the top fire scenarios revealed cases in which these containment systems were disabled. Therefore, it is not unreasonable that CI, OP, and OT failures would have a higher percentage of the CDF for the fire IPEEE than for the IPE. From these results, it was concluded that the fire IPEEE analysis does not result in any containment failure modes distinctly different from those found in the IPE.

REFERENCES

1. Edwin I. Hatch Nuclear Plant Units 1 and 2 Individual Plant Examination (IPE), December 1992.

2. EPRI Modular Accident Analysis Program-BWR, Version 3B, revision 10.

Table 4.7-1 Definitions for Level II Analysis
(Sheet 1 of 5)

Release Categories

Release Category	Percentage of Radionuclides Released			
	Nobles	Volatiles		Non-Volatiles
A	< 100 %	< 0.1 %	and/or	< 0.1 %
B	< 100 %	0.1 - 1 %	and/or	< 0.1 %
C	< 100 %	1 - 10 %	and/or	< 2 %
D	< 100 %	> 10 %	and/or	> 2 %

ACCIDENT SEQUENCE SUBCLASSES

Accident Sequence Class	Accident Sequence Subclass	Definition
Class I	A	Transient With Loss of High Pressure Injection and Failure to Depressurize Accident sequences involving loss of high-pressure coolant inventory makeup in which reactor pressure remains high at core damage .
	B	Station Blackout Accident sequences involving a loss of AC power (station blackout) and loss of coolant inventory makeup.
	C	ATWS With Loss of Injection Accident sequences involving a failure to scram (ATWS) which induces a coincident loss of all coolant inventory makeup.
	D	Transient With Loss of Low Pressure Injection Accident sequences involving a loss of all reactor coolant inventory makeup in which reactor pressure has been successfully reduced to low pressure (< 200 psi) at core damage.
Class II	--	Loss of Decay Heat Removal Accident sequences involving loss of containment heat removal leading to containment failure and subsequent loss of coolant inventory makeup.

Table 4.7-1 Definitions for Level II Analysis
(Sheet 2 of 5)

Release Categories

Class III	A	RPV Rupture Accident sequence involving inadequate coolant makeup, leading to core vulnerable conditions, initiated by reactor vessel rupture due to failure of SRVs to open where containment integrity is not breached in the initial time phase of the accident.
	B	LOCA With Loss of High Pressure Injection and Failure to Depressurize Accident sequences initiated or resulting in small or medium LOCAs with failure of or inadequate high pressure coolant inventory makeup for which the reactor cannot be depressurized.
	C	LOCA With Loss of Low Pressure Injection Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is depressurized and low pressure coolant inventory makeup fails or is inadequate.
	D	LOCA With Failure of Vapor Suppression Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging containment integrity.
CLASS IV	--	ATWS With Inadequate Heat Removal Accident sequences involving failure to scram and failure to inject boron leading to a high pressure challenge to the containment, and subsequent loss of inventory makeup.
CLASS V	--	Unisolated LOCAs Outside Containment Unisolated LOCA outside containment leading to loss of effective coolant inventory makeup

Table 4.7-1 Definitions for Level II Analysis
(Sheet 3 of 5)

Release Categories

PLANT DAMAGE STATES

CODE = X YY Z

X: REACTOR STATUS (three states)

- R (arrest/recovery in-vessel) - recovery in-vessel assumed for any sequence in which in vessel injection occurs between time of potential core damage and vessel failure. (The Plant Hatch IPE does not take credit for in-vessel recovery.)
- L (vessel failure at low pressure) - characterized by depressurization of reactor vessel without operation of in-vessel injection system
- H (vessel failure at high pressure) - high pressure melt ejection

YY: CONTAINMENT STATUS (seven states) - physically describes status of containment as it relates to potential for releases

- CN (containment intact)
- VW (torus venting)
- VD (drywell venting)
- OP (overpressure failure)
- OT (overtemperature failure)
- CI (containment isolation failure)
- CB (containment bypassed)

Z: DEBRIS COOLING (three states)

- N (debris cooling not needed)
- S (debris cooling successful - either containment spray or vessel injection)
- F (debris cooling failure - both containment spray and vessel injection)

Table 4.7-1 Definitions for Level II Analysis
(Sheet 4 of 5)

Release Categories

RELEASE MODES

CODE = VV W

VV: TYPE OF RELEASE

N (intact containment/reactor vessel at accident termination)

N1 (no containment/vessel failure with successful RHR DHR) - no release (beyond normal leakage)

N2 (no containment/vessel failure with successful torus venting)

N3 (no containment/vessel failure with successful drywell venting)

N4 (vessel failure with intact containment - successful RHR DHR) - no release (beyond normal leakage)

N5 (vessel failure with intact containment - debris cooling; successful torus venting)

N6 (vessel failure with intact containment - debris cooling; successful drywell venting)

S (containment failure occurs; release with torus scrubbing through large containment penetration like failed torus gas space)

S1 (no vessel failure)

S2 (vessel failure/debris cooling)

E (early failure - preceding or coincident with reactor vessel failure; torus bypassed)

E1 (small containment failure with vessel injection successful)

E2 (large containment failure with no vessel failure)

E3 (large containment failure with vessel failure/containment sprays successful)

E4 (large containment failure with vessel failure/vessel injection successful)

E5 (large containment failure with vessel failure/no vessel injection or containment spray)

Table 4.7-1 Definitions for Level II Analysis
(Sheet 5 of 5)

Release Categories

L (late containment failure - delayed significantly following reactor vessel failure permitting settling of aerosols; torus bypassed)

L1 (containment sprays successful)

L2 (vessel injection successful)

L3 (no vessel injection or containment spray)

I (failure to isolate containment prior to the initiation of core damage; small containment failure)

I1 (no vessel failure)

I2 (vessel failure/containment sprays successful)

I3 (vessel failure/vessel injection successful)

I4 (vessel failure/no vessel injection or containment spray)

B (radionuclides bypass containment directly through unisolated LOCA outside containment/interfacing system LOCA)

W: TIMING OF RELEASE - approximate timing of release respective to time at which general emergency is declared

N (no release) - releases limited to normal leakage

E (early) - releases expected during first 6 hours of event

I (intermediate) - releases expected during first 24 hours of event (overlaps release mode E)

L (late) - releases expected to occur well into the event, on the order of a day or more

Table 4.7-2 Plant Hatch Unit 1 IPE/IPEEE (Fire) Source Term Binning Comparison
(Sheet 1 of 5)

Source Term Calculation Bin Number	Accident Sequence Class/Subclass	CET End State	Plant Damage State	Release Mode	Release Category	Unit 1 IPE Funct Freq	Unit 1 IPEEE Fire Funct Freq	Unit 1 IPE Analyzed Sequence Freq	Unit 1 Highest IPEEE Fire Sequence Freq
1	IA	15	H CN S	N4 N	A	4.43E-06	3.75E-07		4.59E-08
	IIIB	15	H CN S	N4 N		2.82E-06	9.86E-09	4.56E-07	6.53E-10
	IB	15	H CN S	N4 N		5.44E-07			
	IA	19	H CN S	N4 N		8.17E-08	2.27E-07		1.05E-07
	IIIB	19	H CN S	N4 N		5.24E-08	1.39E-10		
2	IB	23	H OT F	L3 I	D	2.31E-06	1.29E-08	6.59E-07	
	IA	23	H OT F	L3 I		1.07E-06	4.08E-09		
	IIIB	23	H OT F	L3 I		1.75E-09	1.39E-11		
3	II	125	H OP F	E5 L	B	2.43E-06	1.99E-06	4.97E-07	7.11E-07
	II	121	L OP F	E5 L		1.33E-06	1.61E-07		
4	II	133	H OP F	E5 L	D	6.41E-07	5.27-07	1.32E-07	1.89E-07
	II	129	L OP F	E5 L		3.50E-07	4.22E-08		
	II	141	H OP F	E5 L		2.54E-09	2.24E-09		
	II	137	L OP F	E5 L		1.36E-09	1.33E -10		

Table 4.7-2 Plant Hatch Unit 1 IPE/IPEEE (Fire) Source Term Binning Comparison
(Sheet 2 of 5)

Source Term Calculation Bin Number	Accident Sequence Class/Subclass	CET End State	Plant Damage State	Release Mode	Release Category	Unit 1 IPE Funct Freq	Unit 1 IPEEE Fire Funct Freq	Unit 1 IPE Analyzed Sequence Freq	Unit 1 Highest IPEEE Fire Sequence Freq
5	V	224	L CB F	B E	D	1.71E-07		8.71E-08	
6	IA	16	H VW S	N5 L	A	9.38E-07	1.09E-08	2.30E-07	
	ID	7	L VW S	N5 L		1.15E-07	5.16E-08		9.01E-09
	IA	20	H VW S	N5 L		3.73E-09	1.10E-09		
	IB	16	H VW S	N5 L		1.57E-10			
	IC	16	H VW S	N5 L		4.26E-11			
	IB	7	L VW S	N5 L		2.94E-11			
	ID	11	L VW S	N5 L		1.70E-12			
	IC	20	H VW S	N5 L		1.52E-12			
7	IC	15	H CN S	N4 N	A	1.02E-07		2.88E-08	
	IC	19	H CN S	N4 N		1.81E-09			
	IC	6	L CN S	N4 N		7.82E-12			
8	IA	6	L CN S	N4 N	A	1.83E-06	2.79E-08	8.16E-09	
	IB	6	L CN S	N4 N		2.48E-07			
	IIIC	6	L CN S	N4 N		4.51E-08	2.15E-08		3.54E-09
	ID	10	L CN S	N4 N		2.14E-09			

Table 4.7-2 Plant Hatch Unit 1 IPE/IPEEE (Fire) Source Term Binning Comparison (Sheet 3 of 5)

Source Term Calculation Bin Number	Accident Sequence Class/Subclass	CET End State	Plant Damage State	Release Mode	Release Category	Unit 1 IPE Funct Freq	Unit 1 IPEEE Fire Funct Freq	Unit 1 IPE Analyzed Sequence Freq	Unit 1 Highest IPEEE Fire Sequence Freq
	IIIC	10	L CN S	N4 N		1.28E-11			
9	ID	14	L OT F	L3 I	C	4.37E-07	2.05E -07	1.05E-07	2.24E-08
	IB	14	L OT F	L3 I		1.80E-07	2.55E-06		2.23E-07
	IIIC	14	L OT F	L3 I		2.66E-09	2.46E-09		
10	IV	121	L OP F	E5 E	C	2.12E-07		4.58E-08	
	IV	125	H OP F	E5 E		9.67E-09			
11	IV	129	L OP F	E5 E	D	5.59E-08		1.29E-08	
	IV	133	H OP F	E5 E		2.54E-09			
	IV	137	L OP F	E5 E		2.13E-10			
	IV	141	H OP F	E5 E		9.39E-12			
12	IA	18	H OP S	L1 L	C	7.64E-08	6.70E-10	1.47E-08	
	ID	9	L OP S	L1 L		1.30E-08	4.42E-09		
	IA	17	H VD S	N6 L		1.95E-09	2.50E-11		
	IIIB	18	H OP S	L1 L		1.90E-09	1.37E-11		
	IA	22	H OP S	L2 L		8.01E-10	6.89E-10		

Table 4.7-2 Plant Hatch Unit 1 IPE/IPEEE (Fire) Source Term Binning Comparison (Sheet 4 of 5)

Source Term Calculation Bin Number	Accident Sequence Class/Subclass	CET End State	Plant Damage State	Release Mode	Release Category	Unit 1 IPE Funct Freq	Unit 1 IPEEE Fire Funct Freq	Unit 1 IPE Analyzed Sequence Freq	Unit 1 Highest IPEEE Fire Sequence Freq
	IIIB	22	H OP S	L2 L		3.73E-11			
	ID	8	L VD S	N6 L		2.16E-10	7.59E-11		
	IIIB	17	H VD S	N6 L		1.03E-10			
	IB	18	H OP S	L1 L		9.76E-11			
	IB	17	H VD S	N6 L		9.62E-11			
	IB	9	L OP S	L1 L		2.94E-11			
	IB	8	L VD S	N6 L		2.94E-11			
	IA	21	H VD S	N6 L		5.94E-12			
	IIIC	9	L OP S	L1 L		9.90E-12	1.87E-11		
	IB	13	L OP S	L2 L		2.02E-12			
	IIIB	21	H VD S	N6 L		1.62E-12			
13	IIIB	16	H VW S	N5 L	A	3.36E-08	3.20E-10	2.06E-08	
	IIIB	20	L VW S	N5 L		8.77E-10	8.10E-12		
	IIIC	7	L VW S	N5 L		4.16E-10	4.21E-10		

Table 4.7-2 Plant Hatch Unit 1 IPE/IPEEE (Fire) Source Term Binning Comparison (Sheet 5 of 5)

Source Term Calculation Bin Number	Accident Sequence Class/Subclass	CET End State	Plant Damage State	Release Mode	Release Category	Unit 1 IPE Funct Freq	Unit 1 IPEEE Fire Funct Freq	Unit 1 IPE Analyzed Sequence Freq	Unit 1 Highest IPEEE Fire Sequence Freq
14	IB	30	H CI F	I4 E	B	1.33E-08	4.67E-11	3.94E-09	
	IA	30	H CI F	I4 E		7.92E-09	8.11E-11		
	IA	28	H CI S	I2 E		6.13E-09	2.57E-10		
	IB	28	H CI S	I2 E		3.12E-09			
	ID	25	L CI S	I2 E		2.09E-09	2.29E-10		
	IB	25	L CI S	I2 E		1.34E-09			
	IIIB	28	H CI S	I2 E		8.99E-10			
	ID	27	L CI F	I4 E		6.86E-10	2.34E-09		
	IB	27	L CI F	I4 E		8.66E-10	1.49E-08		
	IIIC	25	L CI S	I2 E		1.04E-11	8.64E-12		
	IA	29	H CI S	I3 E		2.81E-11	6.56E-10		
	IC	28	H CI S	I2 E		2.08E-11			
	IIIB	29	H CI S	I3 E		4.55E-12			
	IIIC	27	L CI F	I4 E		4.52E-12			
	IIIB	30	H CI F	I4 E		4.11E-12			

Table 4.7-3 Plant Hatch Unit 2 IPE/IPEEE (Fire) Source Term Binning Comparison (Sheet 1 of 5)

Source Term Calculation Bin Number	Accident Sequence Class/Subclass	CET End State	Plant Damage State	Release Mode	Release Category	Unit 2 IPE Funct Freq	Unit 2 IPEEE Fire Funct Freq	Unit 2 IPE Analyzed Sequence Freq	Unit 2 Highest IPEEE Fire Sequence Freq
1	IA	15	H CN S	N4 N	A	5.46E-06	2.82E-07		4.14E-08
	IIIB	15	H CN S	N4 N		3.40E-06	7.67E-09	4.54E-07	
	IB	15	H CN S	N4 N		5.27E-07			
	IA	19	H CN S	N4 N		1.01E-07	5.65E-09		
	IIIB	19	H CN S	N4 N		6.34E-08	1.85E-07		5.56E-08
2	IB	23	H OT F	L3 I	D	2.29E-06	5.05E-09	6.58E-07	
	IA	23	H OT F	L3 I		1.36E-06	9.46E-09		
	IIIB	23	H OT F	L3 I		6.09E-09	1.46E-09		
3	II	125	H OP F	E5 L	C	2.19E-06	7.50E-07	5.03E-07	2.83E-08
	II	121	L OP F	E5 L		1.15E-06	1.09E-07		
4	II	133	H OP F	E5 L	D	5.77E-07	1.99E-07	1.34E-07	3.05E-08
	II	129	L OP F	E5 L		3.04E-07	2.86E-08		
	II	141	H OP F	E5 L		2.25E-09	8.09E-10		
	II	137	L OP F	E5 L		1.18E-09	9.66E-11		
	II	119	L OP S	S2 L			1.65E-08		
	II	123	H OP S	S2 L			4.57E-08		

Table 4.7-3 Plant Hatch Unit 2 IPE/IPEEE (Fire) Source Term Binning Comparison (Sheet 2 of 5)

Source Term Calculation Bin Number	Accident Sequence Class/Subclass	CET End State	Plant Damage State	Release Mode	Release Category	Unit 2 IPE Funct Freq	Unit 2 IPEEE Fire Funct Freq	Unit 2 IPE Analyzed Sequence Freq	Unit 2 Highest IPEEE Fire Sequence Freq
	II	127	L OP S	E3 L			4.34E-09		
	II	131	H OP S	E3 L			1.20E-08		
	II	135	L OP S	E3 L			1.12E-11		
	II	139	H OP S	E3 L			2.90E-11		
5	V	224	L CB F	B E	D	1.77E-07		8.71E-08	
6	IA	16	H VW S	N5 L	A	9.67E-07	1.87E-08	2.15E-07	7.55E-09
	ID	7	L VW S	N5 L		4.91E-08	1.97E-08		
	IA	20	H VW S	N5 L		3.77E-09	5.36E-10		
	IB	16	H VW S	N5 L		1.74E-10			
	IC	16	H VW S	N5 L		5.56E-11			
	IB	7	L VW S	N5 L		2.80E-11			
	ID	11	L VW S	N5 L		3.17E-12			
	IC	20	H VW S	N5 L		1.89E-12			
7	IC	15	H CN S	N4 N	A	1.30E-07		3.58E-08	
	IC	19	H CN S	N4 N		2.32E-09			
	IC	6	L CN S	N4 N		1.29E-11			

Table 4.7-3 Plant Hatch Unit 2 IPE/IPEEE (Fire) Source Term Binning Comparison (Sheet 3 of 5)

Source Term Calculation Bin Number	Accident Sequence Class/Subclass	CET End State	Plant Damage State	Release Mode	Release Category	Unit 2 IPE Funct Freq	Unit 2 IPEEE Fire Funct Freq	Unit 2 IPE Analyzed Sequence Freq	Unit 2 Highest IPEEE Fire Sequence Freq
8	ID	6	L CN S	N4 N	A	1.55E-06	4.67E-08	8.12E-09	2.25E-08
	IB	6	L CN S	N4 N		2.20E-07			
	IIIC	6	L CN S	N4 N		3.95E-08	3.33E-09		
	ID	10	L CN S	N4 N		2.74E-09			
	IIIC	10	L CN S	N4 N		1.56E-11			
9	ID	14	L OT F	L3 I	C	4.51E-07	3.34E-07	1.06E-07	1.02E-07
	IB	14	L OT F	L3 I		1.72E-07	1.68E-06		2.53E-07
	IIIC	14	L OT F	L3 I		2.95E-09	3.32E-08		8.32E-09
10	IV	121	L OP F	E5 E	C	2.60E-07		6.15E-08	
	IV	125	H OP F	E5 E		1.26E-08			
11	IV	129	L OP F	E5 E	D	6.86E-08		1.64E-08	
	IV	133	H OP F	E5 E		3.32E-09			
	IV	137	L OP F	E5 E		2.62E-10			
	IV	141	H OP F	E5 E		1.25E-11			

Table 4.7-3 Plant Hatch Unit 2 IPE/IPEEE (Fire) Source Term Binaing Comparison (Sheet 4 of 5)

Source Term Calculation Bin Number	Accident Sequence Class/Subclass	CET End State	Plant Damage State	Release Mode	Release Category	Unit 2 IPE Funct Freq	Unit 2 IPEEE Fire Funct Freq	Unit 2 IPE Analyzed Sequence Freq	Unit 2 Highest IPEEE Fire Sequence Freq
12	IA	18	H OP S	L1 L	C	7.52E-08	2.06E-09	1.28E-08	
	ID	9	L OP S	L1 L		6.63E-09	1.80E-09		
	IA	17	H VD S	N6 L		3.52E-09	5.64E-11		
	IIIB	18	H OP S	L1 L		2.55E-09	3.48E-12		
	IA	22	H OP S	L2 L		9.19E-10	6.56E-11		
	IIIB	22	H OP S	L2 L		2.08E-10	5.59E-10		
	ID	8	L VD S	N6 L		1.31E-10	5.93E-11		
	IIIB	17	H VD S	N6 L		1.12E-10			
	IB	18	H OP S	L1 L		9.25E-11			
	IB	17	H VD S	N6 L		9.13E-11			
	IB	9	L OP S	L1 L		2.65E-11			
	IB	8	L VD S	N6 L		2.65E-11			
	IA	21	H VD S	N6 L		6.01E-12			
	IIIC	9	L OP S	L1 L		3.70E-12	3.86E-12		
	IB	13	L OP S	L2 L		2.04E-12			
	IIIB	21	H VD S	N6 L		1.64E-12	2.84E-11		

Table 4.7-3 Plant Hatch Unit 2 IPE/IPEEE (Fire) Source Term Binning Comparison (Sheet 5 of 5)

Source Term Calculation Bin Number	Accident Sequence Class/Subclass	CET End State	Plant Damage State	Release Mode	Release Category	Unit 2 IPE Funct Freq	Unit 2 IPEEE Fire Funct Freq	Unit 2 IPE Analyzed Sequence Freq	Unit 2 Highest IPEEE Fire Sequence Freq
13	IIIB	16	H VW S	N5 L	A	4.52E-08	1.87E-10	2.08E-08	
	IIIB	20	L VW S	N5 L		1.28E-09	1.01E-08		2.92E-09
	IIIC	7	L VW S	N5 L		3.27E-10	1.81E-10		
14	IB	30	H CI F	I4 E	B	1.33E-08	1.76E-11	3.93E-09	
	IA	30	H CI F	I4 E		7.92E-09	2.72E-11		
	IA	28	H CI S	I2 E		6.[illegible]E-09	4.04E-10		
	IB	28	H CI S	I2 E		3.[illegible]E-09			
	ID	25	L CI S	I2 E		2.09E-09	1.77E-10		
	IB	25	L CI S	I2 E		1.22E-09			
	IIIB	28	H CI S	I2 E		1.14E-09			
	ID	27	L CI F	I4 E		6.86E-10	1.46E-09		
	IB	27	L CI F	I4 E		8.66E-10	9.84E-09		
	IIIC	25	L CI S	I2 E		1.04E-11	2.03E-12		
	IA	29	H CI S	I3 E		2.81E-11	3.51E-12		
	IC	28	H CI S	I2 E		2.68E-11			
	IIIB	29	H CI S	I3 E		5.48E-12	9.25E-09		
	IIIC	27	L CI F	I4 E		4.82E-12	9.04E-10		
	IIIB	30	H CI F	I4 E		2.38E-11	7.60E-11		

Table 4.7-4 Source Term Results Summary

Release Category	IPE Radionuclide Release Frequency (per year)	IPE Conditional Probability of Release Category Given Core Damage	IPEEE (Fire) Radionuclide Release Frequency (per year)	IPEEE Conditional Probability of Release Category Given Core Damage
Unit 1				
A	1.120E-05	0.54	9.009E-07	0.12
B	3.800E-06	0.18	2.693E-06	0.36
C	9.340E-07	0.05	3.176E-06	0.42
D	4.710E-06	0.23	7.304E-07	0.10
Unit 2				
A	1.256E-05	0.58	8.162E-07	0.15
B	3.648E-08	0.00	3.120E-08	0.01
C	4.328E-06	0.20	4.098-06	0.76
D	4.790E-06	0.22	4.547E-07	0.08

Table 4.7-5 Containment States

Designator	Definition	Unit 1 IPE CDF % Total	Unit 1 IPEEE Fire CDF	Unit 2 IPE CDF % Total	Unit 2 IPEEE Fire CDF
CN	Containment Intact	1.0E-05 49%	7.9E-07 11%	1.2E-05 53%	7.5E-07 14%
OT	Containment Overtemperature Failure	4.3E-06 19%	3.3E-06 44%	4.6E-06 20%	2.9E-06 54%
OP	Containment Overpressure Failure	5.6E-06 25%	3.3E-06 44%	5.1E-06 21%	1.6E-06 31%
VW	Wetwell Venting	1.2E-06 5%	7.7E-08 1%	1.2E-06 5%	7.0E-08 1%
VD	Drywell Venting	2.6E-09 0%	1.2E-10 0%	4.2E-09 0%	2.0E-10 0%
CB	Containment Bypass	1.9E-07 1%	0 0%	1.9E-07 1%	0 0%
CI	Containment Isolation Failure	3.9E-08 0%	2.2E-08 0%	4.0E-08 0%	3.1E-08 1%
TOTAL CDF		2.2E-05	7.5E-06	2.4E-05	5.4E-06

4.8 TREATMENT OF FIRE RISK SCOPING STUDY ISSUES

NRC Generic Letter No. 88-20, Supplement 4 (Reference 1), lists the following Fire Risk Scoping Study (FRSS) issues to be addressed in the fire portion of the IPEEE:

- Effectiveness of manual fire fighting.
- Fire barrier assessment.
- Seismic/fire interactions.
- Total environment equipment survival.
- Control systems interactions.

The approach taken in addressing the specific concerns regarding each of the above issues and the results of those investigations are described in the following sections.

4.8.1 EFFECTIVENESS OF MANUAL FIRE FIGHTING

A review was conducted of Plant Hatch's fire fighting programs, procedures, and training records to confirm and document that the plant has an effective fire fighting team. Procedures establish the personnel requirements for and responsibilities of the Site Fire Brigade, as well as training and drilling requirements. Procedures provide instructions for both onsite and offsite fire response by the Fire Brigade, and evacuation plans and instructions for verification or recovery of operation of certain safety-related equipment. Annual inspections and audits of the Fire Protection Program are performed and documented.

The average response time for the fire areas evaluated in the detailed analysis is adequate and is far less than the fire control time used in the fire nonsuppression factor calculations (section 4.5). Thus, based on the review of Plant Hatch's fire fighting programs, procedures, and training records, the plant's manual fire fighting capability is considered adequate.

4.8.2 FIRE BARRIER ASSESSMENT

Plant Hatch has an established program for fire barriers in fire-rated assemblies. Barriers include fire doors, penetration seals, fire dampers, and wrapped raceways. The fire barriers are maintained as effective measures that will mitigate the effects from the spread of fire, hot gases, and smoke from one safety-related fire area/zone to another.

Surveillance procedures are performed every 18 months, as required by the Edwin I. Hatch Fire Hazards Analysis and Fire Protection Program (FHA) (Reference 2). The surveillances include a visual inspection of all fire-rated assemblies. Ten percent of each type of penetration seal material is also inspected in the 18-month interval. The inspection results have not identified any major problems with the installation and integrity of the fire-rated assemblies. Plant Hatch also has an administrative program to control combustible materials that are taken into fire areas/zones. Both of these programs are in place as part of the overall fire protection program for Plant Hatch.

The results from surveillance and inspection activities required by the FHA indicate the program is adequate.

4.8.3 SEISMIC/FIRE INTERACTIONS

The seismic plant walkdown included consideration of the following:

1. Potential fire initiators, such as nonseismic hydrogen piping, component lube oil systems, and tanks.

2. Fire protection piping, which could be rendered inoperable due to seismic interaction with its environment (i.e., adjacent piping and block walls).

3. Fire protection systems, which, if actuated (or ruptured) due to a seismic event, could affect the required fire mitigation equipment.

No significant seismic-induced fire scenarios were identified in the seismic analysis. (For a detailed discussion, see chapter 3, section 3.1.4.12 of this report.)

4.8.4 TOTAL ENVIRONMENT EQUIPMENT SURVIVAL

The FRSS expressed concern regarding the potential for adverse effects on plant equipment caused by combustion products released from the fire. Current industry testing/analysis relative to nonthermal fire effects on plant equipment has not quantified the potential problems. The absence of industry conclusions concerning the detrimental short-term effects of smoke on equipment are not believed to be significant. In the nearly 700 events of the generic fire database, there is no indication of any equipment damage due solely to smoke effects. Therefore, the effect of smoke on equipment survivability was considered only with fire effects.

To assess operator effectiveness in performing manual safe shutdown actions, an evaluation of the normally operating plant ventilation systems and their effect on the ability of the operators to access and operate equipment needed to shut down the plant was conducted.

The effect of inadvertent actuation of the fire suppression system on safety-related equipment is addressed in section 4.9.2, Generic Issue-57.

4.8.3 CONTROL SYSTEMS INTERACTION

The control and monitoring circuits of the plant were reviewed to verify that safe shutdown circuits are located physically independent of, or can be isolated from, the control room for an exposure fire that causes a loss of control from the control room. This analysis found no significant contributor to risk due to FRSS issues.

REFERENCES

1. U. S. Nuclear Regulatory Commission Generic Letter No. 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," June 1991.

2. Edwin I. Hatch Nuclear Plant Units 1 and 2 Fire Hazards Analsis and Fire Protection Program, revision 10C, April 1995.

4.9 UNRESOLVED SAFETY ISSUE A-45 AND OTHER SAFETY ISSUES

In addition to the Fire Risk Scoping Study issues, the following safety issues are addressed in this report:

- Unresolved Safety Issue (USI)-A45, "Shutdown Decay-Heat Removal Requirements."
- Generic Issue (GI)-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment."

4.9.1 DECAY HEAT REMOVAL EVALUATION (USI A-45)

Resolution of the external events portion of USI A-45 has been subsumed into the IPEEE requirements that allow plant-specific evaluation of the safety adequacy of decay heat removal (DHR) systems. The Plant Hatch IPEEE fire probabilistic risk assessment (PRA) evaluates the systems and operator actions required to remove decay heat during the 24-hour mission time following a fire-induced plant trip from full-power operation. The results of the fire PRA provide insights concerning the importance of systems that directly perform the DHR function.

Decay heat can be removed by different systems and methods depending on the type of accident sequence. Three means of DHR were considered in the Plant Hatch fire PRA:

- Power conversion system.
- Residual heat removal (RHR), in the long-term cooling modes.
- Containment venting.

Sequences with inadequate DHR are assigned to Accident Classes II and IV. Accident Class IV represents anticipated transients without scram (ATWS) sequences in which the containment failed because reactor power could not be controlled within the capacity of the available heat removal systems. In this accident class, RHR suppression pool cooling and other DHR systems may be available, but reactor power exceeds the capacity of these systems. The results of the Plant Hatch fire PRA indicate the core damage frequency (CDF) associated with Accident Class IV is negligible, because Plant Hatch is equipped with the alternate rod insertion system in addition to the reactor trip function provided by the reactor protection system (RPS). Furthermore, the RPS is a fail-safe system designed to trip on de-energization. Fire damage to the RPS equipment or its associated cables is more likely to result in a de-energization condition due to open circuit failures than a sustained hot short condition. Therefore, this evaluation of the DHR capability focuses on core damage contributors to Accident Class II; i.e., non-ATWS loss of DHR.

The importance of the DHR function to core damage and the causes for loss of DHR can be determined from an analysis of the importance of Accident Class II to the total CDF, leading fire subscenarios contributing to Class II accidents, and the important hardware and human failures contributing to the Class II accidents. Class II accidents contribute approximately 36 percent and 22 percent of the total fire-induced CDF for Unit 1 and Unit 2, respectively.

Only two Unit 1 fire subscenarios contribute more than 5 percent to the fire-induced Class II accident frequency. The subscenario with the highest frequency is small cable fires in the cable spreading room. The conservative plant impact modeled for this subscenario includes failure of 4-kV emergency buses 1E and 1G, containment venting, RHR loop A, and core spray loop A. Only 1 of the 30 raceway segments included in this subscenario involves loss of both buses 1E and 1G. Decay heat removal via the main condenser is unavailable due to the loss of turbine building cooling water as a result of failure of buses 1E and 1G. Decay heat removal through RHR loop B is lost because motive power supply to both RHRSW loop B pumps (from bus 1G) is unavailable. As such, the combination of direct fire damage (RHR loop A and containment venting) and cascaded failures due to loss of support equipment (power conversion system) will lead to the loss of all three DHR methods. This subscenario represents approximately 50 percent of the total Class II frequency.

The subscenario with the second highest frequency (6 percent) is a large switchgear fire in switchgear 1G room resulting in a loss of offsite power to all three emergency buses. This is caused by fire damage to both offsite power supply cables from startup transformers 1D and 1C. Switchgear 1G and the diesel generator 1C dc battery panel were conservatively assumed to be failed by the fire. Diesel generator 1A and containment venting fail due to nonfire causes. Decay heat removal via RHR and power conversion system is unavailable due to loss of buses 1E and 1G and failure of the diesel generator 1C dc battery panel.

Three Unit 2 fire subscenarios contribute more than 5 percent to the fire-induced Accident Class II frequency. They include small cable fires in the cable spreading room (approximately 43 percent), small cable fires in the east cable way (approximately 13 percent), and a large heating, ventilation, and air-conditioning fire on the reactor building south working floor on elevation 130 ft (approximately 6 percent). The plant impact modeled for all three subscenarios includes failure of DHR via hardened vent and RHR loop B. In addition, the power conversion system was also assumed to be ineffective due to the fire-induced stuck-open safety relief valve. If DHR via RHR loop A fails due to nonfire causes, a Class II accident would occur.

Since the frequency of each of the above Unit 1 and Unit 2 fire subscenarios is relatively low (i.e., < 2E-06 events per year), no fire subscenarios in plant locations reflect a plant vulnerability with respect to fire-induced CDF due to loss of DHR.

The issue of DHR vulnerabilities for BWR Mark I containment was investigated in NUREG/CR-4767 (Reference 1). This study proposed modifications to include additional fire barriers to address DHR vulnerabilities from fires. Because of the low Class II accident frequency and based on an evaluation of fire propagation and local scenarios analyzed in the Plant Hatch fire risk analysis, additional fire barriers are not considered necessary. An additional modification considered in NUREG/CR-4767 was an alternate RHR. Because of the high cost of this proposed modification, NUREG/CR-4767 concluded that the alternate RHR system did not have a positive net benefit. Because of the low Class II accident frequency from the fire events, it is reasonable to conclude that an alternate RHR system for Plant Hatch is not justifiable.

In summary, the importance of loss of DHR to the fire-induced CDF is 36 percent for Unit 1 and 22 percent for Unit 2. The total frequency of all Class II sequences is below 3.0E-06 events per year and 1.5E-06 events per year for Unit 1 and Unit 2, respectively. The Class II accident sequence models contain conservatism which, if modeled in more detail, would significantly reduce the importance of loss of DHR to the CDF. Thus, no plant-specific fire-induced vulnerabilities of the Plant Hatch DHR systems were identified.

4.9.2 GENERIC ISSUE-57

No electrical relays containing mercury are used in any plant equipment at Plant Hatch. GI-57, "Effects of Fire Protection System Actuation on Safety Related Equipment," was addressed during the implementation of 10 CFR 50.48 Appendix R modifications. The fire protection suppression systems utilized at Plant Hatch are activated by thermal devices to minimize the incident of inadvertent discharge. Nevertheless, all safe shutdown electrical cabinets and panels susceptible to water damage from the fire suppression systems were equipped with seals to preclude water entrainment as well as spray shields. Equipment pads are also provided to protect the safe shutdown cabinets and panels from minor flooding should a suppression system discharge. Fire suppression was considered as a flooding source in the internal flooding analysis of the Individual Plant Examination (Reference 3). No vulnerabilities associated with internal flooding from any source were identified. The total contribution from internal flooding calculated in the IPE was < 1.0 percent of the total CDF from internal events. No significant risk was identified in response to GI-57.

REFERENCES

1. "Shutdown Decay Heat Removal Analysis of a General Electric BWR4/Mark I," NUREG/CR-4767, July 1987.

2. Meeting Minutes, Advisory Committee on Reactor Safeguards Auxiliary and Secondary Systems Subcommittee Meeting on Review of LaSalle Fire PRA, the Proposed Resolution of Generic Issue-57, and ABWR Fire Risk Assessment, Bethesda, Maryland, July 27-28, 1993.

3. Edwin I. Hatch Nuclear Plant Units 1 and 2 Individual Plant Examination, December 1992.

5. HIGH WINDS, FLOODS, AND OTHERS

The overall methodology recommended by NUREG-1407 (Reference 1) for analyzing plant risk due to high winds, floods, and transportation and other events is a progressive screening approach. The steps for this approach, which are described in section 5 of NUREG-1407, are summarized as follows:

1. Review plant-specific hazard data and licensing bases.

2. Identify significant plant changes that could affect the design conditions since the operating license issuance.

3. Determine whether the plant design meets current criteria as stated in the 1975 Standard Review Plan (SRP), NUREG-75/087 (Reference 2).

4. Determine whether the hazard frequency is acceptably low (if plant design does not conform to the 1975 SRP) (optional step).

5. Perform a bounding analysis if the hazard frequency is unacceptable (optional step).

6. Perform a probabilistic risk assessment if the bounding analysis is unacceptable (optional step).

The first three steps are the starting point in the progressive screening process. If these steps conclude that the 1975 SRP criteria have been met, the optional steps are bypassed. Otherwise, one or more of the optional steps, which are a series of analyses that increase sequentially in level of detail, may need to be performed. If the plant conforms to the 1975 SRP for each external event, the contribution to core damage from that external event is judged to be less than 1E-6 per year and the screening criteria is met. Likewise, the screening criterion for the optional steps is a core-damage contribution of 1E-6 per year (assuming the hazard frequency is less than 1E-5 per year and the conditional probability of core damage is 0.1, given the occurrence of the external event).

REFERENCES

1. U. S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, Draft Report, July 1990.

2. U. S. Nuclear Regulatory Commission, Standard Review Plan, NUREG-75/087, September 1975.

5.1 HIGH WINDS

5.1.1 PLANT HAZARD AND LICENSING BASIS

High winds (including tornadoes) can affect plant structures in at least two ways. If wind forces exceed the load capacity of a building or other external facility, the walls or framing might collapse or the structure might overturn because of the excessive loading. If the wind is strong enough, as is experienced in tornado or hurricane conditions, it may be capable of lifting materials and thrusting them as missiles against some of these critical facilities. Critical components or other contents of facilities not designed to resist missile penetration might be damaged and lose their function.

Wind load capacities for Plant Hatch Seismic Category I structures were selected on the basis of the American Society of Civil Engineers (ASCE) Paper No. 3269, "Wind Forces on Structures" (Reference 1). In accordance with the ASCE requirements, the structures are designed to withstand a basic wind velocity of 105 mph, which is the fastest wind experienced at the Plant Hatch site. The recurrence interval of this wind velocity is estimated by the ASCE paper to occur at least every 100 years.

In addition to wind loading, the Seismic Category I structures are also designed to withstand tornado loading and tornado-generated missiles. The following parameters are applied in combinations producing the most critical condition for structures:

- Dynamic wind pressure.

  The dynamic wind pressure considered is caused by a design basis tornado that has a peripheral tangential velocity of 300 mph and forward progression of 60 mph.

- Pressure differential.

  The structure interior bursting pressure is taken as rising 1 psi/sec for 3 sec, followed by a 3-sec calm, then decreasing at 1 psi/sec for 3 sec. All fully enclosed Seismic Category I structures are designed to withstand the full 3-psi pressure differential.

- Missile impingement.

  Two types of tornado missiles are considered in the plant's structural design:

  - A 12-ft-long piece of wood, 4-in. by 4-in. in size, traveling end-on at 300 mph and striking the structure at any elevation.

  - A 4000-lb automobile, traveling end-on at a speed of 50 mph and striking the structure on an impact area of 20 ft^2, with any portion of the impact area being no more than 25 ft above grade.

Failure of the Seismic Category II structures that are not designed for tornado loads does not affect the ability of Seismic Category I structures to perform their functions for the following reasons:

1. Tornado missiles that may be formed as a result of the failure of Seismic Category II structures do not exceed the force of those missiles postulated by the design requirements from which Seismic Category I structures are designed.

2. The structural frame of the Seismic Category II turbine building has been designed against collapse when subjected to tornado loadings.

A confirmatory plant walkdown was also performed and concluded that no significant unreviewed plant changes exist that impact the Plant Hatch hazard data or licensing bases in regard to winds and tornadoes.

5.1.2 REVIEW OF THE 1975 STANDARD REVIEW PLAN

A review of the 1975 SRP (References 2 through 6) indicates that the Plant Hatch design conforms to the 1975 SRP, except that a more detailed analysis involving a full tornado missile spectrum is required.

Section 3.5.1.4 of the SRP, Missiles Generated by Natural Phenomenon, requires that seven tornado-generated missile types be considered as a design basis for plant structures. The seven missile types are as follows:

1. Wood plank.

2. 3-in. pipe.

3. 1-in. steel rod.

4. 6-in. pipe.

5. 12-in. pipe.

6. Utility pole.

7. Automobile.

The Plant Hatch design considers only two missile types: wood plank and automobile. Incorporating the absent missile types in the calculation for the total tornado load, to verify conformance with the SRP, would require extensive analysis. Consequently, a probabilistic approach (as recommended in the optional steps) to estimate the tornado-generated missile frequency has been utilized.

In the Plant Hatch Unit 2 Final Safety Analysis Report (FSAR) (Reference 7), the tornado occurrence frequency was based on a 13-year study conducted from 1955 to 1967. As part of the IPEEE effort, more recent data from the National Servere Storm Forecast Center (Reference 8) was used in estimating the tornado frequency. Since the wind loading exceedance and tornado missile hazard frequencies are derived from the tornado frequency, reevaluation of these frequencies is required when using the new tornado frequency data.

5.1.3 TORNADO FREQUENCY

The first step in estimating the tornado-generated missile frequency is to determine the tornado striking frequency (or occurrence frequency), ϕ_t, at the Plant Hatch site. In general, the frequency can be estimated by the following expression (see Reference 9):

$$\phi_t = n \frac{A_o}{A} \tag{5.1}$$

where

ϕ_t = the annual frequency of a tornado striking the plant

n = the mean number of tornado occurrences per year in the reference area

A_o = the tornado origin area

A = the reference area wherein the plant is located and throughout which tornado data is accumulated

The tornado origin area is the area within which the tornado center point must be located to result in a strike at the plant. The tornado origin area is given in Reference 8:

$$A_o = W_t L_t + W_t Z_1 + L_t Z_2 + A_p \tag{5.1a}$$

where

W_t = tornado path width

L_t = tornado path length

Z_1 = projection of plant site area in tornado length direction

Z_2 = projection of plant site area in tornado width direction

A_p = Area of the plant site

Figure 5.1-1 illustrates the tornado origin area for the Plant Hatch site. The site is represented by a rectangular area of length, 2000 ft in the north-south direction, and width, 1100 ft. For this analysis, it is conservatively assumed that the projections on the tornado length (Z_1) and width (Z_2) has the maximum possible value of 2280 ft.

Substituting A_o from Equation 5.1a into Equation 5.1, the result is

$$\phi_t = n(W_t L_t + W_t Z_1 + L_t Z_2 + A_p)/A \qquad (5.1b)$$

Based on tornado data from the National Severe Storm Forecast Center (Reference 8), 55 tornadoes occurred in 38 years (from August 1954 to January 1993) in a 36-mi radius of Plant Hatch. The average tornado path length of the 55 events is 2.17 miles, the average tornado path width is 0.051 mile and the reference area (within the 36-mile radius) is approximately 4070 square miles. Applying Equation (5.1b), the tornado striking frequency at the plant site is obtained as 4.09E-4 per year.

The National Severe Storms Forecast Center for the National Weather Service noted that the upper bound tornado frequency is estimated as 2.54E-3 per year, based upon the number of events in the most tornado-frequent year (i.e., 9 events in 1971).

5.1.4 WIND LOADING EXCEEDANCE FREQUENCY

The pressure, or wind load, exerted on a structure is a function of the windspeed. The following equation, as provided in section 3.3 of the Plant Hatch Unit 2 FSAR, relates wind load to windspeed.

$$Q = 0.002558(V^2) \qquad (5.2)$$

where

Q = the windspeed-induced pressure or wind loading (psf)

V = the windspeed (mph)

Figure 5.1-2 illustrates this relationship. The plant's structures are designed to withstand a design basis tornado, which produces winds at a much higher velocity than can be produced by wind sources such as hurricanes or wind gusts.

Because tornadoes produce the highest velocity winds, windspeed exceedance frequency is derived from data of tornado intensity probability. Windspeed exceedance frequency is obtained from two factors:

1. The occurrence frequency of tornadoes.

2. The cumulative probability of tornado intensity.

The annual frequency, ϕ_v, of winds greater than V mph can be expressed by the following relationship:

$$\phi_v = \phi_t\,(\phi_{v|t}) \qquad (5.3)$$

where

ϕ_v = the annual frequency of a tornado striking the plant with a windspeed greater than V mph

ϕ_t = the annual frequency of a tornado striking the plant, given by Equation (5.1b)

$\phi_{v|t}$ = the cumulative probability of tornado intensity or the fraction of tornadoes with a peak windspeed greater than V mph

Knowing the windspeed exceedance frequency from Equation (5.3) and the Plant Hatch-specific relationship of wind load expressed as a function of windspeed, Equation (5.2), a wind load exceedance relationship can be derived.

As stated in section 5.1.3, the annual tornado frequency, ϕ_t, is 4.09E-4 per year. The cumulative probability of tornado intensity provided in section 2.3.1.3.G of the Plant Hatch Unit 2 FSAR is judged to be applicable because it is based on a large population (i.e., 1612 tornadoes). Therefore, the corresponding probability, $\phi_{v|t}$, for the design-basis tornado with a tangential velocity of 300 mph and a forward progression of 60 mph, is approximately 1.44E-4. From Equation (5.3), the resultant frequency (or wind loading exceedance frequency) is calculated as 5.89E-8 per year and is, therefore, several orders of magnitude lower than the screening criterion of 1E-6 per year.

5.1.5 TORNADO MISSILE FREQUENCY AND IMPACT AND DAMAGE FREQUENCY

Nearly every tornado generates missiles as it whirls along its path. The analysis of tornado-generated missiles is a highly complex problem because several factors must be considered: tornado frequency and intensity, plant site characteristics, and potential sources of missiles, etc. Electric Power Research Institute (EPRI) studies (References 9 and 10) applied a probabilistic Monte Carlo simulation to predict the risk to a hypothetical nuclear plant by tornado-generated missiles. The study postulated an initial spectrum of available missiles and evaluated the following:

1. The wind field in the tornado.

2. Missile injection and transportation.

3. Missile-impact velocities.

4. Potential damage to plant structures.

The study also included a tornado data analysis which evaluated the occurrence frequency of tornadoes in different tornado intensity regions. Using a given tornado-occurrence frequency by region, a spectrum of missile types, and a representative number of potential missiles in a plant as input data, the study estimated the annual impact and damage frequency to structures at a hypothetical plant. For example, a typical result of the study for an operating plant in Region I (see Reference 9, Table 3-26), with a tornado occurrence frequency of 2.29E-3 per year, is as follows:

P^N: 7.09E-5 per year

P_L: 3.45E-5 per year

P_U: 3.33E-6 per year

where

P^N is the probability that any tornado-generated missile will impact the plant structures

P_L is the probability that a missile will impact with sufficient force to cause backscabbing (loss of wall material on the building interior) if all plant structures have 6-in. concrete walls

P_U is the probability that a missile will impact with sufficient force to cause backscabbing if all plant structures have 18-in. concrete walls

The results are also based on a typical two-unit plant layout to establish the target envelope and a 26-missile spectrum, which includes the seven missiles types addressed in the 1975 SRP (the total number of missiles evaluated is 1000). The 26-missile spectrum used in the study is considered to be more conservative than the SRP spectrum regarding damage potential; thus, the resultant impact and damage frequency is judged to be applicable to the current analysis.

To apply the results of the EPRI study (Reference 9) to a specific plant structure of interest, the missile impact and damage frequency is combined with a missile strike density. The missile strike density is defined as the ratio of the annual frequency of impact with any plant safety-related structure divided by the total exposed surface area of all safety-related structures. This provides the frequency per year of hitting a unit area of a safety-related building with tornado missiles. By multiplying this strike density by the surface area of any target, the annual strike frequency for that specific target can be calculated as follows:

$$P_i = P_t \frac{A_i}{A_t} \tag{5.4}$$

where

P_i = annual frequency of a tornado missile hitting the i^{th} target

P_t = annual frequency of a tornado missile hitting any plant structure

A_i = exposed area of the i^{th} target

A_t = total exposed surface area of plant structures

The missile impact and damage frequencies shown above (Reference 9) were calculated based on a tornado-strike frequency of 2.29E-3 per year. Scaling these impact and damage frequencies with a Plant Hatch site-specific tornado frequency of 4.09E-4 events per year yields the following results:

P^N: 1.27E-5 per year

P_L: 6.16E-6 per year

P_U: 5.95E-7 per year

The missile impact and damage frequencies given above are the likelihood of missiles hitting and damaging any of the plant structures, including the reactor building, control building, turbine building, radwaste building, service building, intake structure, diesel generator building, etc. Most safety-related systems or equipment are inside plant buildings such as the reactor building, control building, intake structure, and diesel generator building, all of which have 24-in.-thick concrete walls.

Three factors must be considered in the assessment of the tornado-missile-impact risk for the Plant Hatch site:

1. The missile impact and damage frequency that causes backscabbing of the plant's 24-in. walls should be smaller than that for the 18-in. wall case (i.e., 5.95E-7 per year).

2. The area ratio A_i/A_t for target structures given in Equation (5.4) is likely to reduce the frequency by at least a factor of 2 or 3.

3. Missiles that cause backscabbing of concrete walls do not necessarily cause structural failure, nor do they necessarily cause sufficient damage to components near the wall such that they fail.

Therefore, it is concluded that the tornado-generated missiles' contribution to core damage frequency is substantially less than 1E-6 per year and is an insignificant contributor to plant risk.

5.1.6 CONCLUSIONS

The results provided in sections 5.1.4 and 5.1.5 conclude that the Plant Hatch structures are well designed to withstand the hazards associated with high winds and no potential vulnerability is identified. Therefore, it is concluded that the contribution to core damage frequency from high winds, including tornadoes, is less than 1E-6 per year and, therefore, the contribution to plant risk is insignificant.

REFERENCES

1. "Wind Forces on Structures," American Society of Civil Engineers, Paper No. 3269.

2. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 3.3.1, Wind Loadings, NUREG-75/087, September 1975.

3. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 3.3.2, Tornado Loadings, NUREG-75/087, September 1975.

4. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 3.5.1.4, Missiles Generated By Natural Phenomenon, NUREG-75/087, September 1975.

5. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 3.5.2, Structures, Systems, and Components to be Protected from Externally Generated Missiles, NUREG-75/087, September 1975.

6. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 3.5.3, Barrier Design Procedure, NUREG-75/087, September 1975.

7. Edwin I. Hatch Nuclear Plant Unit 2 Final Safety Analysis Report.

8. TORPLOT Results for Hatch Site from National Severe Storms Forecast Center, National Weather Service, July 13, 1994.

9. Twisdale, L.A., Dunn, W.L., and Chu, J. "Tornado Missile Risk Analysis," Electric Power Research Institute, EPRI NP-768, May 1978.

10. Twisdale, L. A., et al., "Tornado Missile Risk Analysis, Appendixes - Analytical Models and Data Bases," Electric Power Research Institute, EPRI NP-769, May 1978.

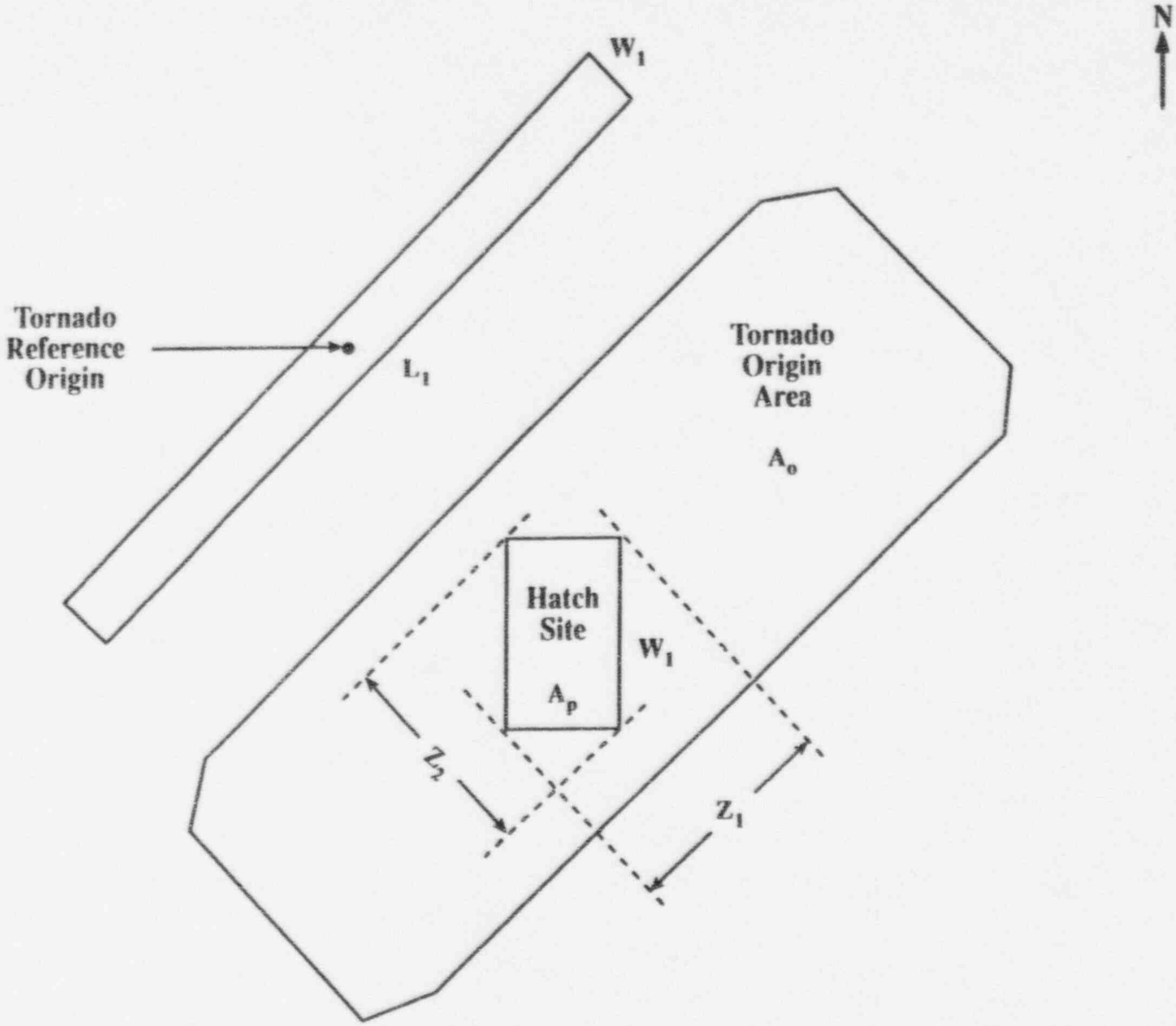


Figure 5.1-1 Tornado Origin Area for the Plant Hatch Site

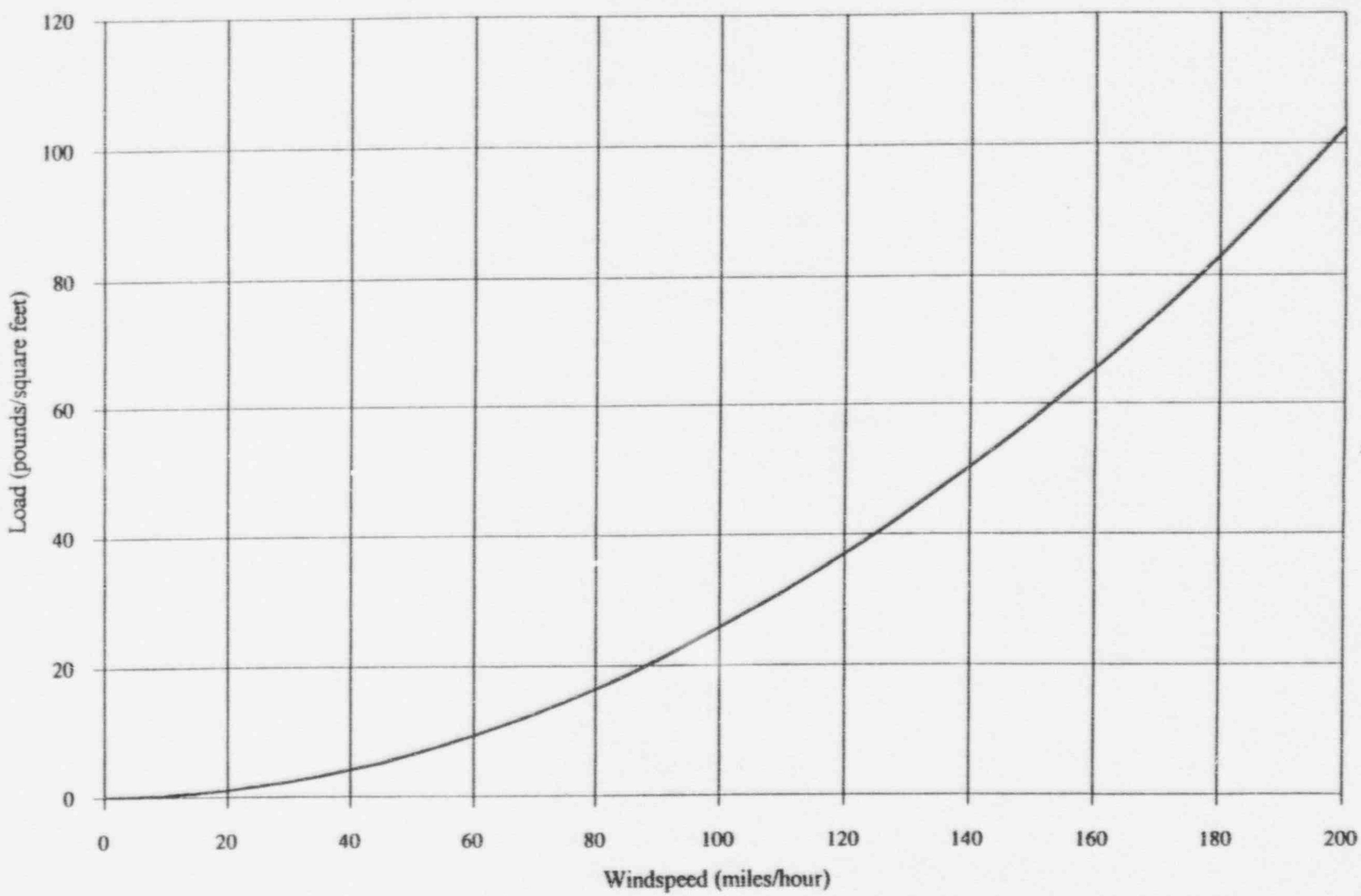


Figure 5.1-2 Wind Load as a Function of Windspeed

5.2 FLOODS

The Edwin I. Hatch Nuclear Plant Unit 2 Final Safety Analysis Report (FSAR) (Reference 1) estimates the drainage area for the portion of the Altamaha River that serves the site at 11,700 mi^2. According to data collected by the National Weather Service (also known as Hydrometeorological Report No. 51 or HMR-51; see Reference 2), for a duration of 72 hours, the theoretically greatest depth of precipitation for a drainage area of 11,700 mi^2, based on the probable maximum precipitation (PMP), is approximately 24.8 in. The value derived from HMR-51 is nearly 5 in. greater than the FSAR value of 19.0 in., which is based on the Elba, Alabama, storm of March 11-16, 1929, per data derived from the FSAR. Figure 5.2-1 of this report is taken from HMR-51 and illustrates the depths expected for durations ranging from 6 to 72 hours for drainage areas ranging from 200 to 20,000 mi^2.

The resultant river discharge rate is directly proportional to the 72-hour rainfall total. The corresponding level of the Altamaha River must be known to predict the peak level of the probable maximum flood (PMF). The Plant Hatch Unit 2 FSAR table 2.4-5 summarizes the Altamaha River's performance (level) for a broad range of discharge values. Figure 5.2-2 of this report is a graphical representation of this table, which presents flood levels for corresponding peak river discharge rates.

After gaining an understanding of the magnitude of the PMF for the site, the frequency of such an event must be estimated. The FSAR assumes that there is a logarithmic relationship between the river discharge rate, or flood, and its frequency, as depicted in figure 5.2-1. Therefore, because the probable maximum precipitation (PMP) is known, the associated river level (PMF) can be determined and the frequency of this event can be estimated.

5.2.1 PLANT HAZARD AND LICENSING BASIS

The external flood hazard licensing bases can be found in the FSAR, and are summarized in this section. The flood design considerations applicable to Plant Hatch are the PMF, local intense precipitation, coincident wind effects, dam failures, river obstruction, and roof loads.

To estimate the PMF conditions for which the plant was designed, the FSAR assumed a base river flow of 75,000 ft^3/sec, corresponding to a river level of approximately 84 ft mean sea level (msl). The Elba, Alabama, storm was transposed and maximized over the Altamaha River drainage area. The peak level that such a storm would produce is estimated at 105 ft msl. If high winds should occur simultaneously, wave runup would contribute up to 3.25 ft to the peak flood level. Flood levels could then potentially reach 108.3 ft. The floor of the intake structure is constructed at 110 ft msl, which exceeds the hypothetical storm. More important, the plant grade where the emergency diesel generators, reactor building, and control building are located is at 129 ft msl.

Flooding can also be caused by local intense rainfall. The topography of the plant site is such that the runoff of rainfall is directed away from the power block area. The design of site structures is based upon the world record point precipitation curve, for which the first hour of rainfall postulates 15.3 in. of rain. The site drainage system is designed for a continuous rainfall rate of

6 in./hour; however, an evaluation was done of the plant site to ensure flooding of safety-related equipment would not occur as a result of the world record point precipitation curve rainfall rate (15.3 in. in 1 hour).

Dam failures were considered in the design of the site. It is assumed the dam failure was coincident with the river crest of the Elba, Alabama, storm. The analysis results placed the flood level below that of the most severe precipitation event. Flood caused by ice blockage of the Altamaha River was considered, but the historically lowest recorded river temperature would not support formation of ice.

5.2.2 CHANGES IN PLANT DESIGN

A plant walkdown was performed and documented to identify any plant changes that could affect the plant's hazard data or the licensing bases regarding flooding. The results of this task indicate that there are no significant plant changes.

5.2.3 REVIEW OF THE 1975 STANDARD REVIEW PLAN

The plant licensing bases contained in the FSAR were compared with the 1975 SRP regarding external flooding events (see References 3 through 9). The results of that evaluation indicate that the PMP used in the FSAR flooding analysis was not derived from the latest National Weather Service maximum precipitation criteria (SRP criteria). Moreover, the storm that was used actually produced less rain over the 72 hours of interest than the National Weather Service source.

5.2.4 ANALYSIS

The external flood hazard was evaluated using the more conservative of either the 1975 SRP National Weather Service criteria or the design basis criteria. For the PMF the HMR-51 rainfall was assumed at the frequency associated with the Elba, Alabama, storm. For the local intense rainfall analysis, the design basis rainfall values were used. For the dam failure evaluation, the design basis criteria was used because the frequency of the dam failure coincident with the HMR-51 PMF level is much lower than the recurrence frequency of the PMF itself.

5.2.4.1 Probable Maximum Flood

According to the American Meteorological Society, the PMP is, "the theoretically greatest depth of precipitation for a given duration that is physically possible over a particular drainage area at a certain time of the year." (See Reference 10.) The resulting peak river discharge and associated stage or level created by the PMP is the PMF. The FSAR assessed the PMP and the PMF based upon the storm centered at Elba, Alabama. Because the FSAR analysis does not meet the NUREG-1407 requirements, this analysis uses the curves found in HMR-51, as required, to estimate the PMP and the resulting PMF. HMR-51 is the most current National Weather Service source for estimating potential maximum rainfall for the durations and drainage areas of interest to Plant Hatch.

The FSAR estimates that the Elba, Alabama, storm (19 in. in 72 hours) would generate a peak flowrate of 612,000 ft^3/sec at the site. Figure 5.2-1, taken from HMR-51, translates the 72-hour rainfall into 24.8 in. of rain over the 11,700 mi^2 drainage area. For this analysis, it is assumed that the peak flowrate is directly proportional to the 72-hour rainfall (24.8 in.) over the 11,700 mi^2 drainage area. The 24.8 in. of rain then corresponds to a discharge rate of 799,000 ft^3/sec generated at the site. Referring to figure 5.2-2, this corresponds to a peak level of approximately 110 ft, approximately 5 ft higher than reported in the FSAR.

Because the PMP produces a greater depth of precipitation than the Elba, Alabama, storm, it is logical that the PMP frequency is smaller than that of the Elba, Alabama, storm. Therefore it is conservative to assume that the PMF occurs at a frequency of the Elba, Alabama, storm. Then, the relationship between river discharge rate and frequency is illustrated by figure 5.2-3. This figure is derived by graphing the logarithmic relationship of discharge exceedance (ft^3/sec) versus recurrence interval (per year) as presented in the FSAR. As a result, the recurrence interval associated with a discharge rate of 799,000 ft^3/sec is approximately 1E+8 years. This new data point allows information to be extrapolated from the known flood data to obtain a new curve as shown in figure 5.2-4. Figure 5.2-5 shows the corresponding revised exceedance frequency for any given flood level of the Altamaha River at the Plant Hatch site.

5.2.4.2 Local Intense Precipitation

The PMP at Plant Hatch generated by a local intense storm was estimated in the plant design basis by using the world record point precipitation curve (see figure 5.2-6), created from the following equation (see Reference 1):

$$R = 15.3(D^{0.486}) \tag{5.5}$$

where R is rainfall in inches and D is rainfall duration hours. Note that the world record point precipitation curve provides a greater precipitation depth than HMR-51 and therefore is more conservative.

This equation is used to estimate the peak rainfall rate that the site might experience for varying durations. The topography of the site is such that rainfall runoff is systematically directed away by culverts, open ditches, and natural drainage channels. The plant drainage system, however, is designed for a sustained 6 in./hour rainfall rate. A storm with a peak rainfall rate of 6 in./hour is estimated to occur with a frequency of 6.35E-5 per year. This estimate is based upon a curve fit of rainfall data presented in the FSAR (see figure 5.2-7 of this report). Although the plant drainage system is designed for this rainfall rate, it is assumed for this analysis that a loss of offsite power (LOSP) is caused by this intense rainfall. An LOSP could occur if the switchyard relay house is flooded, thereby failing the batteries or causing a spurious relay actuation. Relay house flooding could occur because of excessive roof loading caused by rainwater buildup, or by accumulation of water on the ground adjacent to the relay house.

The Individual Plant Examination (IPE) Report for Plant Hatch (Reference 11) states that the core damage contribution caused by an LOSP is 5.51E-6 for Unit 1 and 5.10E-6 for Unit 2. The site LOSP frequency is 2.20E-2 events per reactor year; consequently, the conditional probability of core damage given an LOSP event is 2.50E-4 and 2.32E-4 for Units 1 and 2, respectively (the core damage frequency caused by an LOSP divided by the initiating event frequency). Therefore, the core damage frequency caused by flooding, as a result of local intense precipitation, is the product of its occurrence frequency (6.35E-5 per year) and the conditional core damage frequency caused by an LOSP as a result of flooding (the value for the conditional core damage frequency caused by an LOSP, 2.50E-4 and 2.32E-4). For Unit 1, the estimated core damage frequency caused by local intense flooding is 1.6E-8 events per year, and for Unit 2, the value is 1.5E-8 events per year. Local intense precipitation is, therefore, not an important event for the Plant Hatch site. Thus, Generic Issue 103 (Generic Letter 89-22) has been addressed and is considered closed.

5.2.4.3 Coincident Wind Effects

The FSAR discusses the effect of a coincidental high wind on the peak flood, assuming that a 45-mph wind, sustained for more than 1 hour, would conservatively coincide with the peak flood. If this wind were to affect an 18-mi stretch of the river, the wave-height study indicates that 6.5-ft waves (crest to trough) would develop, adding approximately 3.25 ft to the peak flood level.

5.2.4.4 Dam Failures

There are three major dams located in the Altamaha River basin upstream of the plant. The largest and closest facility Sinclair Dam, which is 169 river miles upstream of the plant. The next nearest dam is Wallace Dam, located on the Oconee River 172.7 river miles upstream of the site. Located furthest away is Lloyd Shoals Dam, situated on the Ocmulgee River 268 river miles upstream of the site.

The Elba, Alabama, storm, transposed, rotated, and optimized for the drainage area feeding Sinclair Dam, would cause the dam to discharge at a peak rate of 389,000 ft^3/sec. If the dam were instantaneously removed, a 27-ft high wave would be created, discharging at approximately 3,000,000 ft^3/sec. A wave decay curve analysis performed for the FSAR indicated that the wave would be reduced to 15 percent of its original size by the time the wave reached the site. This is equivalent to an increase in river discharge flow of 100,000 ft^3/sec, for a total flow of approximately 500,000 ft^3/sec. The resultant flood level would then be approximately 100 ft msl.

A similar analysis performed for the FSAR indicates that if Lloyd Shoals Dam should fail, a 1-ft wave would develop at the site after traveling 268 miles downstream.

If Wallace Dam should be instantaneously removed, a 29-ft wave would be generated. By the time the wave reached the Sinclair Dam, it would have attenuated to 23 ft. If Sinclair Dam should instantaneously fail because of the wave, a combined wave of approximately 33 ft would be generated. By the time the wave reaches the site, the wave size would be reduced to only 5 ft.

5.2.4.5 <u>River Obstructions</u>

According to the FSAR, no record exists in modern times of the Altamaha River freezing over. Moreover, the lowest recorded temperature measured was 37.4°F, which is safely above freezing.

5.2.4.6 <u>Roof Loads</u>

The design of site structures is based on the world record point precipitation curve rainfall values. These values are more conservative than those in HMR-51; therefore, no additional analysis for roof loading is required.

5.2.5 CONCLUSIONS

The PMF for the Plant Hatch site is estimated by HMR-51 as 24.8 in. of rainfall over a drainage area of 11,700 mi^2. The expected Altamaha River flood level is 110 ft, which is the floor level of the intake structure (the diesel generator building and all other safety-related buildings are located well above this level at 129 ft). Should there also be coincidental high-wind-generated waves, as much as 3.25 ft would be added to the 110 ft of flood level. However, only two doors of the intake structure are located below this level, and they are placed at labyrinth offsets and are weather-stripped. Wave run-up created by the coincidental high winds would cause leakage into the building, but the leakage would be mitigated by floor drains and valve pit submersible pumps. The recurrence frequency of such a flood is estimated to be once every 1E+08 years, which makes the scenario unlikely. Moreover, this frequency is conservatively based upon the assumption that the PMP occurs at the same frequency as the Elba, Alabama, storm.

Dam failures contribute only nominally to the flood level. Because of the long distance between the site and the nearest dams, any surges created by an instantaneous failure would be attenuated to a fraction of the original size. For the nearest, largest dam, the Sinclair Dam, the resultant wave from the dam's instantaneous failure would only add 100,000 ft^3/sec of flow. (The failure of the Sinclair Dam is not likely under the conditions of the PMF because of the standards with which it was designed; thus, the scenario in which the dam fails coincident with the PMF would occur at a frequency that is much lower than the recurrence frequency of the PMF, making this scenario unlikely.)

Plant procedures provide measures for protecting all entrances to the intake structure, diesel building, reactor building, and turbine building if the Altamaha River is expected to crest at greater than or equal to 105 ft msl.

Other sources of flooding such as river obstruction and local intense rainfall do not contribute to any significant flooding events. River flooding caused by ice flows is very unlikely because river temperature historically has never dropped low enough. Local intense rainfall even at world-record level, would be sufficiently mitigated by the site's drainage system. Therefore, even if plant flooding from external causes were to lead directly to core damage, its frequency is estimated to be less than the screening criteria, 1E-08 per year, which is insignificant.

REFERENCES

1. Edwin I. Hatch Nuclear Plant Unit 2 Final Safety Analysis Report.

2. National Oceanic and Atmospheric Administration, National Weather Service Hydrometeorological Report (HMR) No. 51, Probable Maximum Precipitation Estimates, United States East of the 105th Meridian, June 1978.

3. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 2.4.2, Floods, NUREG-75/087, September 1975.

4. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 2.4.3, Probable Maximum Flood, NUREG-75/087, September 1975.

5. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 2.4.4, Potential Dam Failures, NUREG-75/087, September 1975.

6. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 2.4.5, Probable Maximum Surge and Seich Flooding, NUREG-75/087, September 1975.

7. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 2.4.10, Flooding Protection Requirements, NUREG-75/087, September 1975.

8. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 3.4.1, Flood Protection, NUREG-75/087, September 1975.

9. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 3.4.2, Analysis Procedure, NUREG-75/087, September 1975.

10. American Meteorological Society, Glossary of Meteorology, Boston, Massachusetts, 1959.

11. Edwin I. Hatch Nuclear Plant Units 1 and 2 Individual Plant Examination, December 1992.

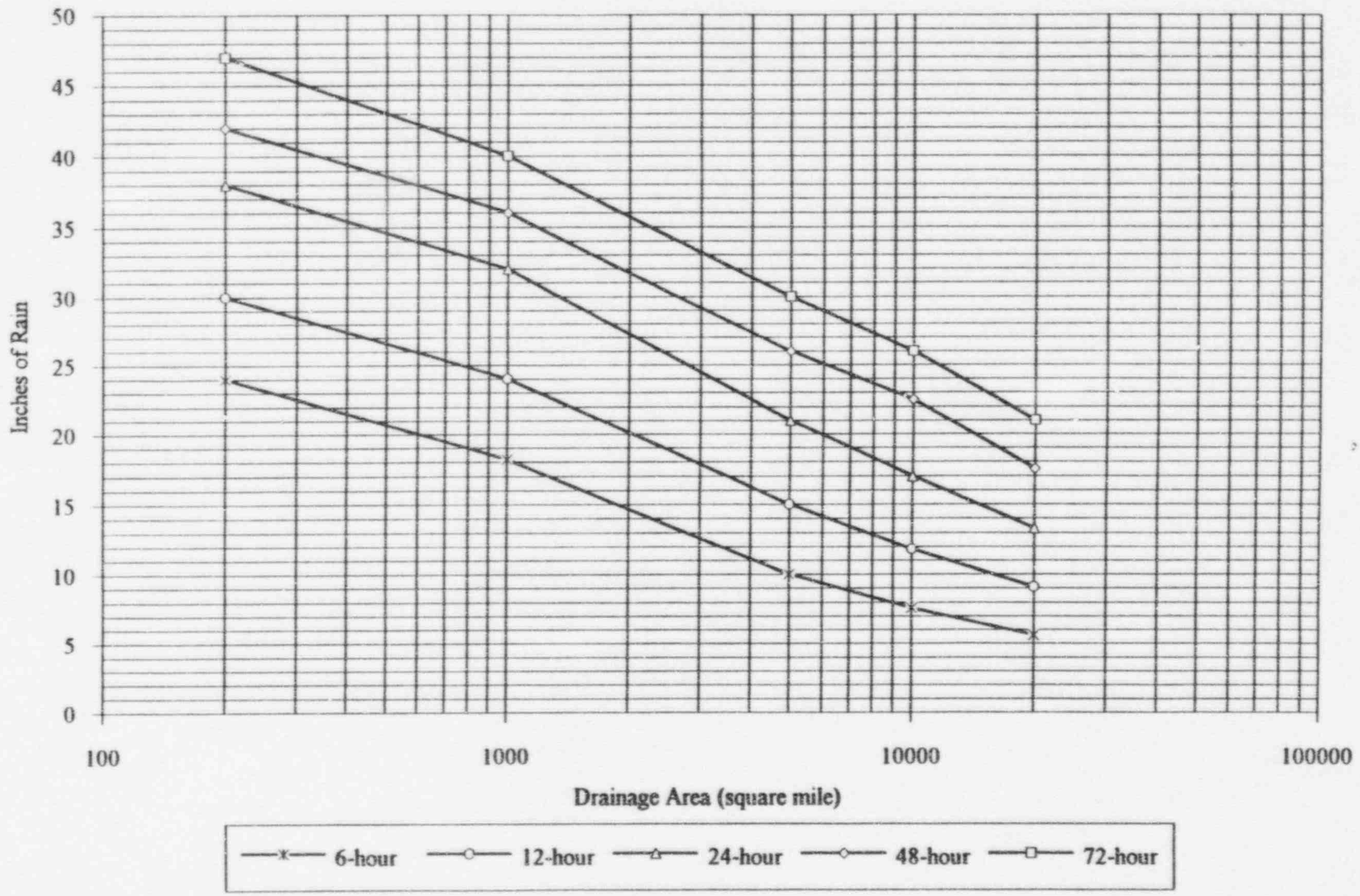


Figure 5.2-1 Probable Maximum Precipitation for Plant Hatch

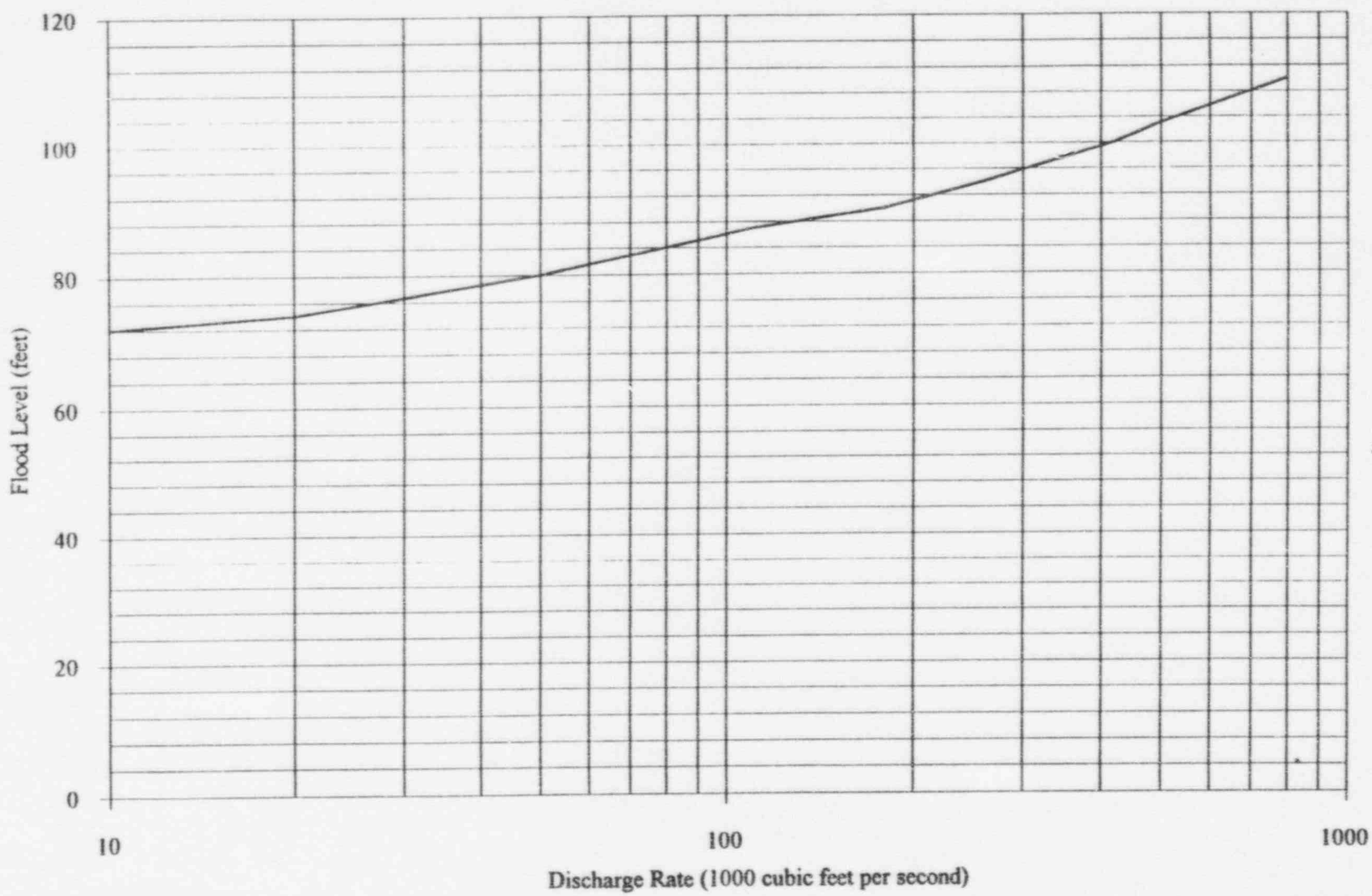


Figure 5.2-2 Flood Level as a Function of Altamaha River Flow Rate

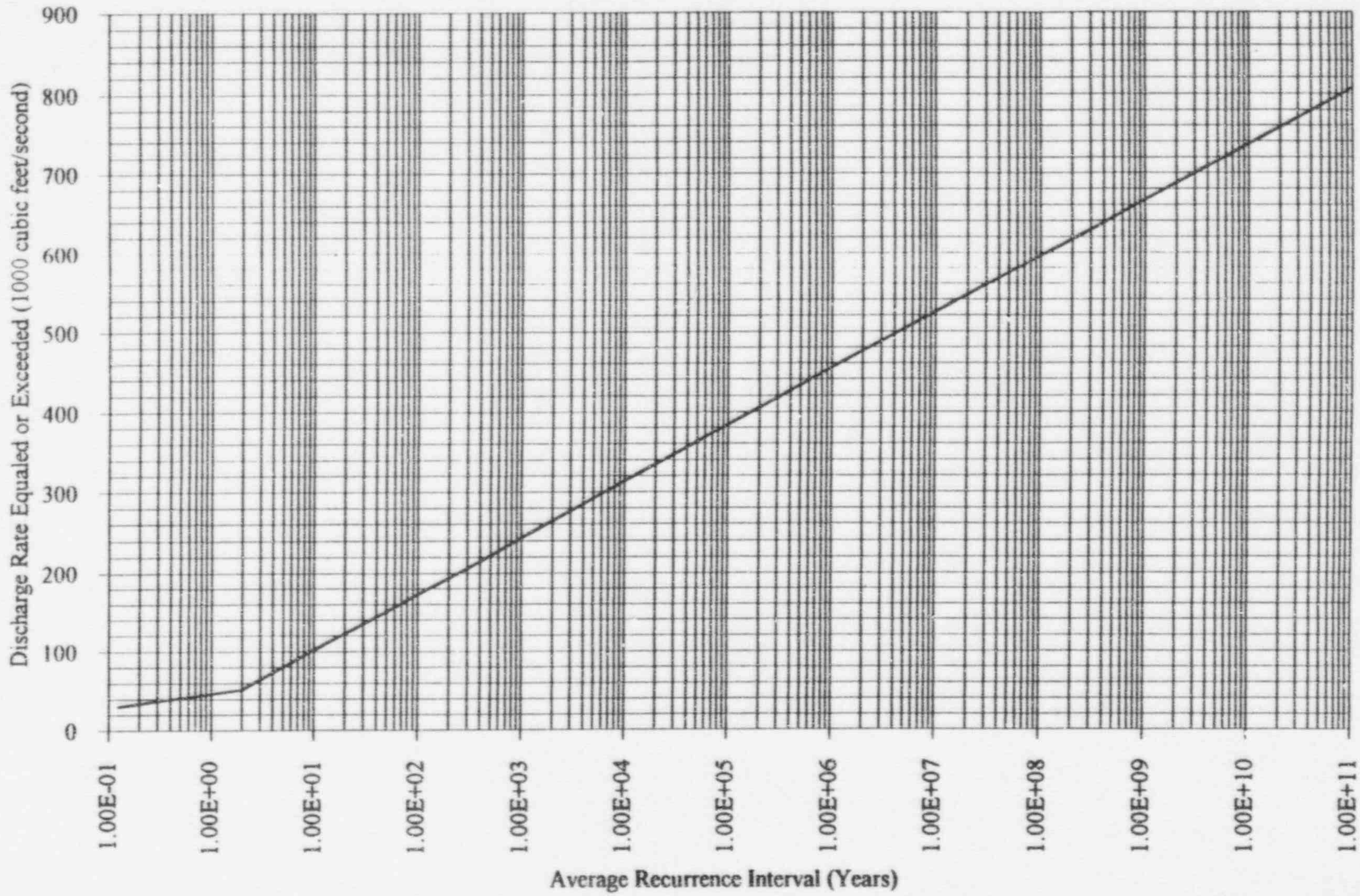


Figure 5.2-3 Altamaha River Discharge Frequency

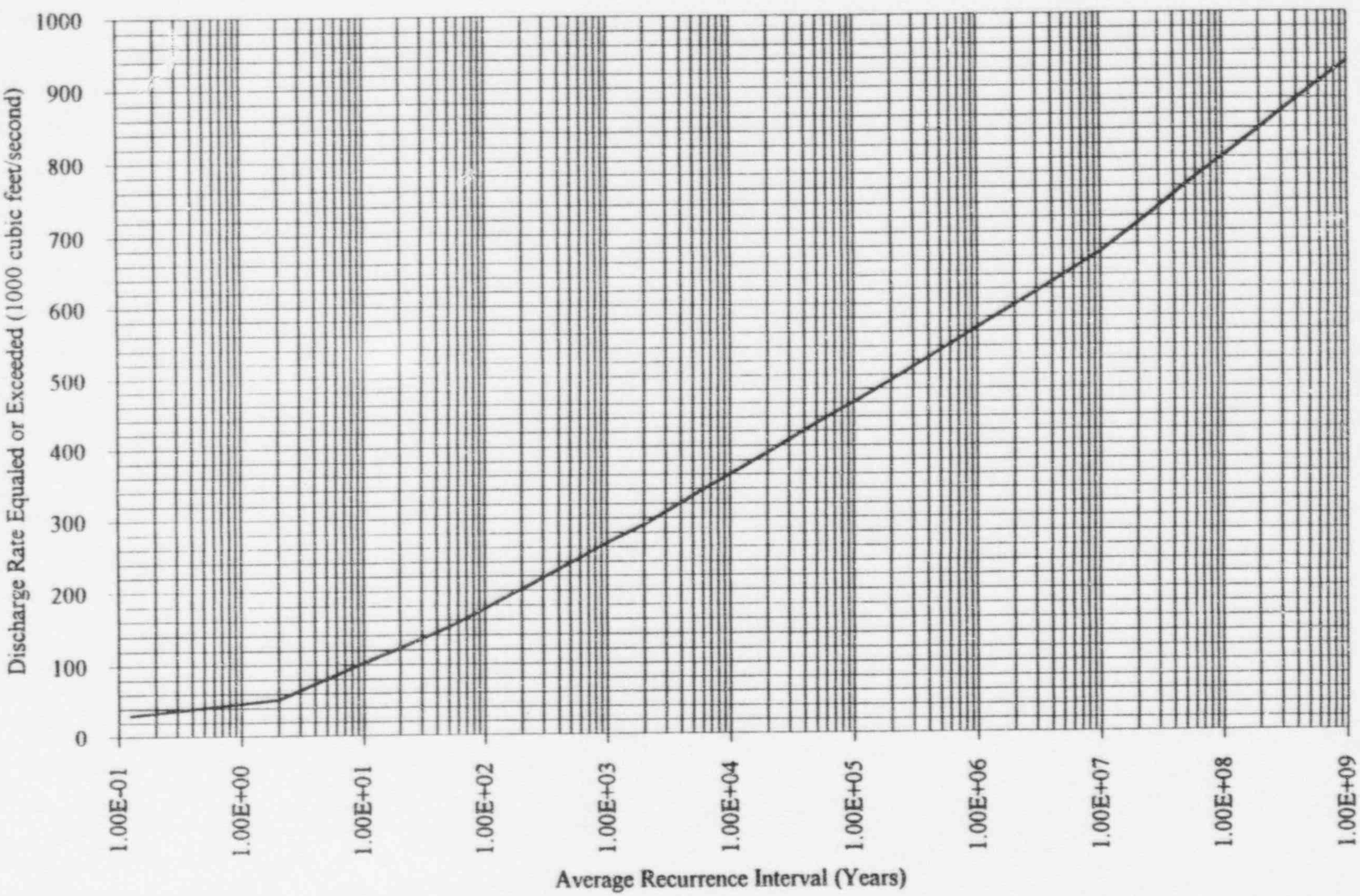


Figure 5.2-4 Revised River Discharge Rate and Recurrence Interval

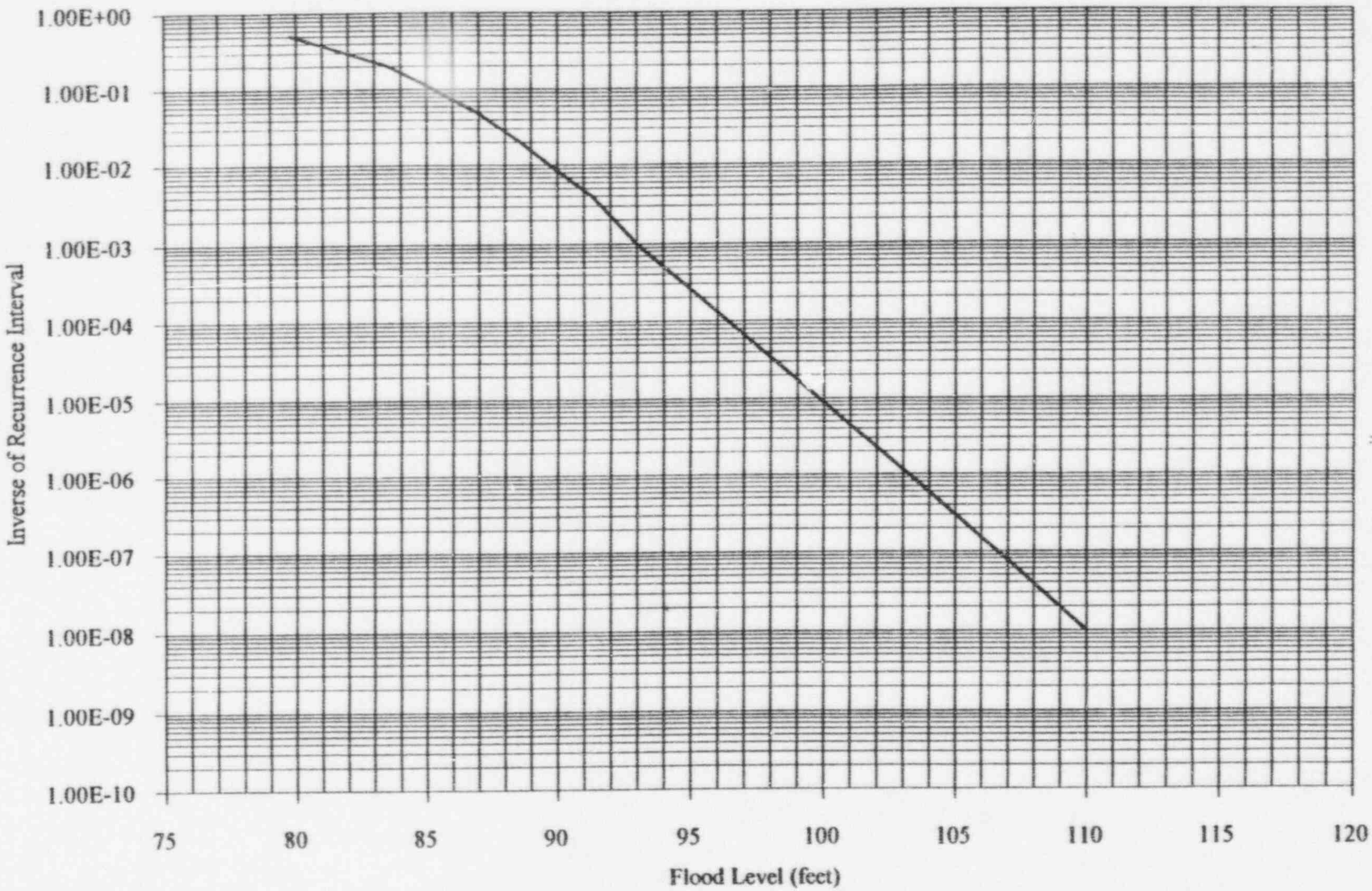


Figure 5.2-5 Revised Flood Level Versus Frequency

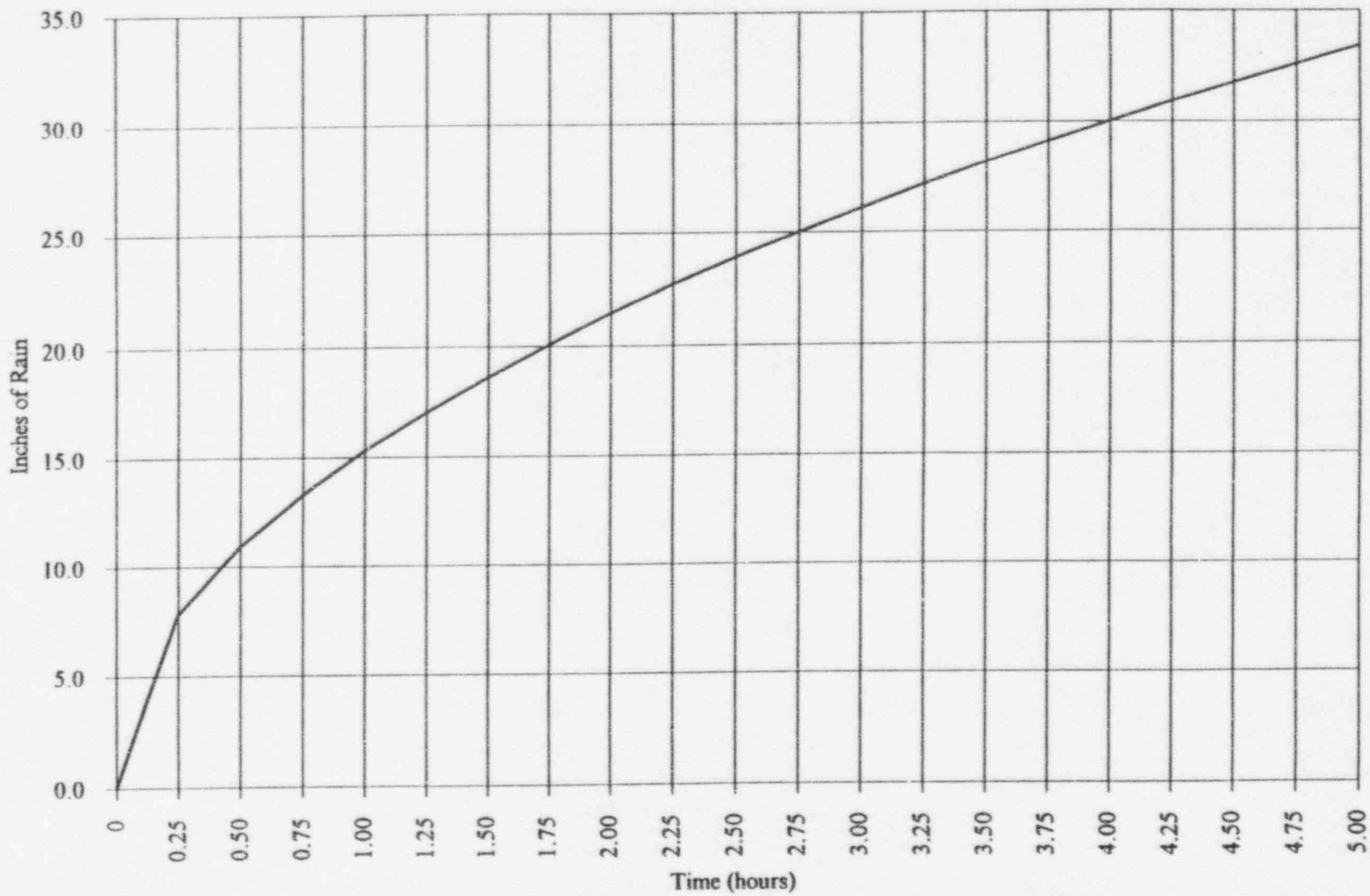


Figure 5.2-6 World-Record Precipitation Curve

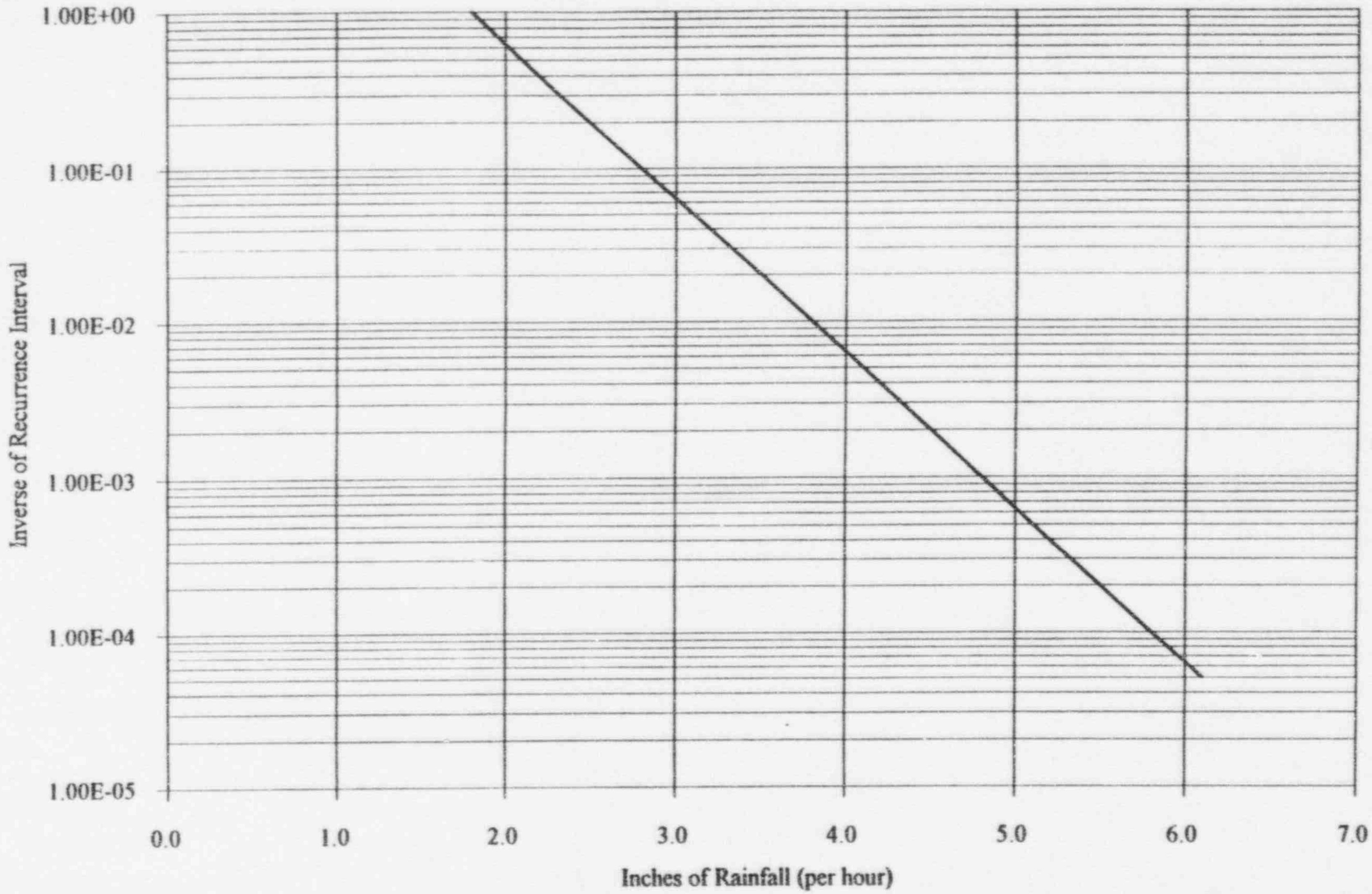


Figure 5.2-7 Peak 1-Hour Rainfall Total Versus Inverse of Recurrence Interval

5.3 TRANSPORTATION AND NEARBY FACILITY ACCIDENTS

5.3.1 TRANSPORTATION

5.3.1.1 Plant-Specific Hazard Data and Licensing Bases Review

Transportation hazards in the Plant Hatch vicinity at the time of operating license (OL) issuance which had the potential to affect Units 1 and 2 were identified and discussed in the Plant Edwin I. Hatch Unit 1 Final Safety Analysis Report (FSAR) (Reference 1), section 2.2, and the Unit 2 FSAR (Reference 2), section 2.2. Per the Unit 2 FSAR, at the time of OL issuance, the only significant transportation routes within a 5-mi radius of Plant Hatch were U.S. Highway No. 1, a rail-spur line leading to Plant Hatch, and the Altamaha River. The rail-spur line leading to Plant Hatch did not carry commercial traffic. There were no known military firing or bombing ranges or aircraft low-level flight holding or landing patterns in the plant's vicinity, nor were there any airports or landing strips within 10 mi of the site. The closest airport with commercial service was located in Waycross, Georgia, approximately 48 mi south of Plant Hatch.

At the time of OL issuance, potential accidents involving transportation hazards were evaluated as follows:

5.3.1.1.1 Explosions

The possibility of an explosion resulting from a transportation accident was evaluated as a potential hazard. Because there was no commercial traffic on the Altamaha River at the time of OL issuance, it was concluded that no potential existed for accidental detonations which could threaten the plant. U.S. Highway No. 1 passes approximately 3400 ft west of Plant Hatch. Transportation accidents involving the detonation of materials or cargo transported on U.S. Highway No. 1 were judged sufficiently distant so as not to pose a threat to the plant.

5.3.1.1.2 Flammable Vapor Clouds (Delayed Ignition)

The possibility of a flammable vapor cloud released during a transportation accident was evaluated as a potential hazard. Because there was no commercial traffic on the Altamaha River at the time of OL issuance, it was concluded that no potential existed for a transportation accident involving a flammable vapor cloud release which could threaten the plant. Due to the distance between U.S. Highway No. 1 and Plant Hatch (approximately 3400 ft) and the relatively small quantities of materials transported by highway, the probability of a flammable vapor cloud reaching the site was judged to be exceedingly low.

5.3.1.1.3 Toxic Chemicals

The possibility of a toxic chemical release as a result of a transportation accident was evaluated as a potential hazard. Because there was no commercial traffic on the Altamaha River at the time of OL issuance, it was concluded that no potential existed for a transportation accident involving a toxic chemical release which could threaten the plant. Due to the distance between U.S. Highway No. 1 and Plant Hatch (approximately 3400 ft), the probability of a potential toxic release reaching Plant Hatch was judged to be exceedingly low.

5.3.1.1.4 Fires

The possibility of a fire resulting from a transportation accident was evaluated as a potential hazard. Fires from possible sources were judged to be too far away from the plant to pose a threat.

5.3.1.1.5 Collisions With Intake Structure

The possibility of barge traffic on the Altamaha River impacting the intake structure was evaluated as a potential hazard. There was no commercial barge traffic on the Altamaha River at the time of OL issuance. The U.S. Army Corps of Engineers operated a snagging barge on the Altamaha River approximately twice per year–one trip upriver and one trip downriver. The probability of a barge passing the intake structure, veering off course, and colliding with the intake structure was evaluated to be 6.3E-08. Because of the infrequent barge traffic that passed the intake structure, the cumulative probability of a collision occurring was judged to be extremely small (substantially less than 1E-06).

5.3.1.1.6 Liquid Spills

The possibility of a liquid spill contaminating the Altamaha River, occurring as a result of river traffic, was not evaluated as a potential hazard because there was no commercial traffic on the Altamaha River at the time of OL issuance.

5.3.1.2 <u>Identification of Significant Changes Since Operating License Issuance</u>

Numerous information sources were reviewed and a confirmatory plant walkdown was conducted to determine whether any changes have occurred since the OL was issued, which were not reported per 10 CFR 50.71(e). As a result of this review and walkdown, no significant changes regarding transportation or other developments were found to impact the plant's original design conditions.

5.3.1.3 Conformance to 1975 Standard Review Plan Criteria

The following Standard Review Plan (SRP) NUREG-75/087 (Reference 3) sections dealing with transportation hazards were reviewed and conformance was evaluated as follows.

5.3.1.3.1 Standard Review Plan Sections 2.2.1 and 2.2.2, Locations and Routes, Descriptions

The criteria of SRP sections 2.2.1 and 2.2.2 were met at the time of OL issuance because all significant transportation routes within 5 mi of the site were identified and descriptive information was provided to allow evaluation of possible hazards. Because there are no new transportation routes or significant developments which impact the original design conditions, the plant conforms to SRP sections 2.2.1 and 2.2.2.

5.3.1.3.2 Standard Review Plan Section 2.2.3, Evaluation of Potential Accidents

Explosions, flammable vapor clouds (delayed ignition), toxic chemicals, fires, collisions with the intake structure, and liquid spills were evaluated as potential accidents resulting from transportation routes in the plant vicinity. The SRP section 2.2.3 criteria were met at the time of OL issuance because no potential accident was evaluated as posing a threat to the plant, due to the distances between the transportation routes and the plant site and the size of the expected hazardous shipments along the transportation routes. Because there are no new transportation routes in the plant vicinity or significant developments which impact the original design conditions, Plant Hatch conforms to SRP section 2.2.3.

5.3.1.3.3 Standard Review Plan Section 3.5.1.5, Site Proximity Missiles (Except Aircraft)

The SRP section 3.5.1.5 criteria were met at the time of OL issuance because potential accidents which could produce missiles (explosions and flammable vapor clouds) were evaluated as not posing a threat to the plant because of the distance between the transportation route and the site, and the size of the expected hazardous shipment along the transportation route. Because there are no new transportation routes in the plant vicinity or significant developments which impact the original design conditions, Plant Hatch conforms to SRP section 3.5.1.5.

5.3.1.3.4 Standard Review Plan Section 3.5.1.6, Aircraft Hazards

The SRP section 3.5.1.6 criteria were met at the time of OL issuance because there were no identified aircraft hazards in the plant vicinity. There were no identified airports or landing strips within 10 mi of the site, nor were there any low-level flight holding or landing patterns in the plant vicinity. The closest airport with commercial service was located in Waycross, Georgia, approximately 48 mi south of Plant Hatch.

No airports or landing strips exist within 10miles of the plant, and there are no airways within 2 statutemiles of the plant. The closest airports with commercial service are located in Brunswick, Georgia, approximately 70miles southeast of the site and Savannah, Georgia, approximately 67miles northeast of the site. Because there are no sources of aircraft hazards in the plant vicinity, Plant Hatch conforms to SRP section 3.5.1.6.

5.3.2 NEARBY FACILITY ACCIDENTS

5.3.2.1 Plant-Specific Hazard Data and Licensing Bases Review

At the time of OL issuance, nearby facilities having the potential to affect Units 1 and 2 were identified and discussed in the Unit 1 FSAR, section 2.2, and the Unit 2 FSAR, section 2.2. Per the Unit 2 FSAR, at the time of OL issuance the only significant nearby facility was a natural gas line located approximately 4.5miles south of the plant. No manufacturing plants, chemical plants, refineries, storage facilities, mining or quarrying operations, military bases, missile sites, or oil and gas wells existed within 5miles of the site.

At the time of OL issuance, potential nearby facility accidents and onsite storage of hazardous materials were evaluated as follows:

5.3.2.1.1 Explosions

Explosions resulting from the process, storage, or use of high explosives, munitions, chemicals, or liquid and gaseous fuels were evaluated as a potential accident. The only facility within 5miles of the plant which stored/transported an explosive material was the natural gas pipeline located 4.5miles south of the site, and which was judged sufficiently distant such that potential detonations posed no threat to the site. No onsite sources of explosive material were identified.

5.3.2.1.2 Flammable Vapor Clouds (Delayed Ignition)

Flammable vapor clouds were evaluated as a potential hazard. At the time of OL issuance, the only facility within 5miles of the plant which was capable of producing a flammable vapor cloud was the natural gas pipeline located 4.5miles south of the site. The natural gas pipeline was judged sufficiently distant such that the resulting cloud would be dispersed below the flammable concentration by the time it reached the site under the most adverse meteorological conditions. No onsite sources capable of producing a flammable vapor cloud were identified.

5.3.2.1.3 Toxic Chemicals

A toxic chemical release was evaluated as a potential hazard. At the time of OL issuance, no facilities within 5miles of Plant Hatch were identified as using or storing toxic chemicals. Sixteen

1-ton cylinders of gaseous chlorine were stored onsite for water treatment purposes. Additional toxic chemicals stored onsite included acids and caustics. Because of its extreme toxicity and the large quantity stored onsite, gaseous chlorine was considered as being the design basis toxic chemical release for the main control room habitability system. As noted in section 5.3.2.3.2, the use of gaseous chlorine onsite has been discontinued.

5.3.2.1.4 Fires

Fires from nearby facilities were evaluated as a potential hazard. At the time of OL issuance, no nearby industrial, chemical, or storage facilities that could pose a threat due to a fire existed within 5miles of the plant. The only facility within 5miles of the site which was capable of producing a large fire was the natural gas pipeline located 4.5miles south of the plant. A fire associated with the pipeline was judged to be sufficiently distant to not affect the plant.

The ground cover surrounding Plant Hatch was judged to be nonconducive to brush or forest fires which could affect the site.

5.3.2.1.5 Liquid Spills

The possibility of liquid spills contaminating the Altamaha River, resulting from an upstream release, was evaluated as a potential hazard. The nearest industrial facility upstream of Plant Hatch was located near Macon, Georgia. It was judged that any spill from this facility would be diluted substantially before it reached the intake structure and, therefore, would not affect the site.

5.3.2.2 Identification of Significant Changes Since Operating License Issuance

Numerous information sources were reviewed and a confirmatory plant walkdown was conducted to determine whether any changes have occurred since the operating license was issued which were not reported per 10 CFR 50.71(e). As a result of this review and walkdown, no significant changes regarding onsite storage of hazardous materials or nearby facility accidents were found to impact the plant's original design conditions.

5.3.2.3 Conformance to 1975 Standard Review Plan Criteria

The following SRP sections deal with hazards from nearby facility accidents and onsite storage of hazardous materials.

5.3.2.3.1 Standard Review Plan Sections 2.2.1 and 2.2.2, Location and Routes, Descriptions

The criteria given in SRP sections 2.2.1 and 2.2.2 were met at the time of OL issuance because all nearby facilities and hazardous materials stored onsite were identified and descriptive information was provided to allow evaluation of possible hazards. Because there are no new manufacturing plants, chemical plants, refineries, storage facilities, mining and quarrying operations, military bases, missile sites, or oil and gas wells within 5 mi of the site which would impact the original design conditions, the plant conforms to SRP sections 2.2.1 and 2.2.2.

5.3.2.3.2 Standard Review Plan Section 2.2.3, Evaluation of Potential Accidents

Explosions, flammable vapor clouds (delayed ignition), toxic chemicals, and fires were evaluated as potential accidents resulting from nearby facilities and onsite storage of hazardous materials. Occurrence of an accidental onsite chlorine release was used as the design basis toxic chemical release for the control room habitability accident analysis. The chlorine detectors in the control room habitability system were designed to detect gaseous chlorine and automatically isolate the main control room air intake. The criteria of SRP section 2.2.3 were met at the time of OL issuance because the effects of a toxic-gas-release design basis event had been considered and analyses indicated the control room habitability system was able to mitigate the consequences of such a release.

There are no new manufacturing plants, chemical plants, refineries, storage facilities, mining or quarrying operations, military bases, missile sites, or oil and gas wells within 5 mi of Plant Hatch which impact the original design conditions. The use of gaseous chlorine has been discontinued at Plant Hatch, and the chlorine detectors for the control room habitability system have been removed. Because of the location, quantities, and administrative and physical controls on hazardous materials currently stored at the site, the original design conditions have not been impacted. Plant Hatch conforms to SRP section 2.2.3.

5.3.2.3.3 Standard Review Plan Section 3.5.1.5, Site Proximity Missiles (Except Aircraft)

The criteria of SRP section 3.5.1.5 were met at the time of OL issuance because potential accidents which could produce missiles (explosions and flammable vapor clouds) were evaluated as not posing a threat to the plant, due to the great distance between the natural gas pipeline and the plant. There are no new nearby facilities which could produce an explosion or flammable vapor cloud; hence, there is no credible potential for missiles from nearby facility accidents. Hazardous material stored at Plant Hatch which has the potential to create explosions or flammable vapor clouds is located at remote locations to preclude the possibility of affecting safety-related equipment. The plant conforms to SRP section 3.5.1.5.

5.3.2.3.4 Standard Review Plan Section 6.4, Habitability Systems

The criteria of SRP section 6.4 were met at the time of OL issuance as documented in NUREG-0411. The main control room habitability system was judged to be adequate to protect main control room operating personnel from the effects of an onsite chlorine release and a fire outside the main control room. Gaseous chlorine is no longer used at Plant Hatch. Physical and administrative controls exist to preclude the possibility of a sodium hypochlorite/acid reaction which could produce chlorine gas. Administrative controls exist to ensure the main control room is notified in the event of a fire. Toxic chemicals stored at the site and smoke resulting from a fire outside the main control room are not a threat to main control room habitability because, in the event of a release or a fire, the main control room air intake can be isolated and self-contained breathing apparatus are available if toxic fumes or smoke enters the main control room. Plant Hatch conforms to SRP section 6.4.

5.3.3 CONCLUSION

The existing Plant Hatch design conforms to SRP criteria for transportation and nearby facility accidents. No significant changes were identified which impact the plant design. There are no potential vulnerabilities attributed to transportation or nearby facility accidents.

REFERENCES

1. Edwin I. Hatch Nuclear Plant Unit 1 Final Safety Analysis Report, July 1993.

2. Edwin I. Hatch Nuclear Plant Unit 2 Final Safety Analysis Report, July 1993.

3. Nuclear Regulatory Commission, Standard Review Plan, NUREG-75/087, September 1975.

5.4 OTHERS

Pursuant to the guidelines of NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991, Plant Hatch is not required to address the following hazards:

- Lightning.
- Severe temperature transients.
- Severe weather storms.
- External fires (forest fires, grass fires.)
- Extraterrestrial activities.
- Volcanic activity.

No other plant-unique external events were identified that pose any significant threat of a severe accident at Plant Hatch.

6. LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

6.1 INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS PROGRAM ORGANIZATION

This section describes the organization of the Plant Hatch Individual Plant Examination of External Events (IPEEE) program and delineates the areas of responsibility presented in figure 6.1-1.

An engineer from the Nuclear Engineering and Licensing Department, Hatch Project, was the IPEEE project engineer and coordinated the IPEEE team activities with the plant, corporate office, architect engineer, Independent Review Group (IRG), and Southern Nuclear Operating Company (SNC) Technical Services (TS) Department. The team selection for the Hatch IPEEE was based upon expertise in one or more areas: probabilistic risk assessment (PRA) methodology including the Edwin I. Hatch Units 1 and 2 Individual Plant Examination (December 1992); seismic capability evaluations; fire protection; and Plant Hatch operations, training, and maintenance.

The seismic margins assessment was performed by Southern Company Services, Inc. (SCS). The SCS Hatch Project Support Group prepared the Safe Shutdown Equipment List (SSEL). SCS Nuclear Services performed the seismic margins analysis.

The Plant Hatch site personnel provided assistance in the IPEEE preparation. The shutdown paths were reviewed by operations personnel to verify that the SSEL success paths are compatible with Hatch procedures and training. The site staff also provided assistance with both seismic and fire walkdowns. Hatch fire protection engineers provided information for the fire risk scoping study issues, as well as details concerning fire training and plant protection systems.

SNC TS coordinated the IPEEE fire activities with PLG, Inc. SNC TS performed the quantifications and assigned the top event impacts utilizing the IPE PRA model. PLG performed the fire assessment and containment analysis. For events other than fire and seismic, SNC TS performed the reviews. SNC Environmental Services performed the hazards analysis for unanalyzed chemicals identified in the IPEEE process. The transportation and nearby facility accident evaluation was performed by SNC TS. Active interface between SNC and the contractors ensured the effective transfer of knowledge and insights gained from the review.

6.2 COMPOSITION OF REVIEW TEAM

Seismic

The seismic margins methodology developed by the Electric Power Research Institute (EPRI) is a well-defined process which requires the utilization of specially trained personnel. The application of an independent expert peer review process was therefore deemed more appropriate than a review by an independent review group. The selection of the shutdown paths was made by system engineers in conjunction with operations staff input. Operations department personnel reviewed the final safe shutdown equipment list (SSEL) to verify that the paths were consistent with plant procedures.

An independent review was performed of the Unit 1 seismic margins assessment (SMA) by Dr. Robert P. Kennedy of Structural Mechanics Consulting. Dr. Kennedy is a noted expert in the field of seismic design and analysis and served as chairman of the Senior Seismic Review and Advisory Panel for USI A-46. Dr. Kennedy's review of the Hatch IPEEE-Seismic evaluation consisted of a plant walkdown of a representative sampling of SSEL components and electrical raceways. In addition, Dr. Kennedy reviewed the Unit 1 screening evaluation work sheets, sample calculations, outlier resolutions, and proposed modification packages. The results of Dr. Kennedy's third-party audit for Unit 1 are documented in a letter from Dr. Kennedy to Keith D. Wooten of Southern Company Services (SCS), "Hatch Unit 1 A46/IPEEE Peer Review," November 26, 1993.

The only open item from Dr. Kennedy's peer review involved a concern with the method used to determine the capacity of the cable spreading room cable tray support anchor bolts. The calculation for the cable spreading room cable tray support anchorage was subsequently transmitted to Dr. Kennedy for his review. As documented in a letter from Dr. Kennedy to Keith D. Wooten of SCS, "Hatch Unit 1 A46/IPEEE Peer Review," October 5, 1994, this issue was satisfactorily resolved.

A peer review was not performed for the Unit 2 SMA due to the following: 1) The seismic review team engineers that performed the Unit 1 SMA also performed the Unit 2 evaluation, 2) The Unit 1 SMA was a pilot program performed for EPRI and the Nuclear Regulatory Commission, and there was extensive review of its implementation, and 3) The Unit 1 and Unit 2 structures and safe shutdown equipment are similar.

The mechanical and electrical systems work consisted of developing the safe shutdown paths and the SSEL by SCS systems engineers. The independent review of these items was performed by other qualified SCS systems engineers.

Fire

The fire IPEEE performance and conclusions were reviewed by personnel familiar with the Plant Hatch fire protection systems and program. These personnel served as a fire-specific Independent Review Group (IRG). It was decided that this type of review was most appropriate since the results provide an insight into relative plant vulnerabilities as a result of a fire. The IRG members were selected based upon their expertise in fire equipment, training, and protection activities at Plant Hatch. Since the PRA aspects of the analysis relied on the PRA developed and reviewed as part of the IPE process, no additional PRA developmental review was deemed necessary.

The IRG was composed of the following personnel:

- Manager, Hatch Engineering and Licensing, Southern Nuclear Operation company (SNC).
- Nuclear Specialist, Hatch Engineering and Licensing, SNC.
- Senior Engineer, Fire Protection, Hatch Project Support, SCS.
- Fire Protection Engineer, Engineering Department, Georgia Power Company (GPC.)
- Engineer I, Appendix R, SCS.

Topics which were presented to and reviewed by the IRG included the following:

- Data Collection.
  - Fire initiation frequency.
  - Equipment and cable location.
  - PRA interface.
  - Site visits.
- Screening Process.
- Review of Results.

IRG reviews were held in January 1994, December 1995, and January 1996. In addition to the IRG reviews, the SNC Hatch Project staff conducted interim reviews of the evaluations as they were in progress.

Other

Then analysis of high winds, floods, transportation and nearby facility accidents and other events was independently reviewed by a Senior Engineer with experience in the overall IPEEE process from the SNC Hatch Engineering and Licensing Department.

Additional Review

The overall IPEEE examination report was also reviewed by GPC site management. This review served the dual function of providing input to the analyses and providing high visibility of the fire IPEEE and its results to plant personnel.

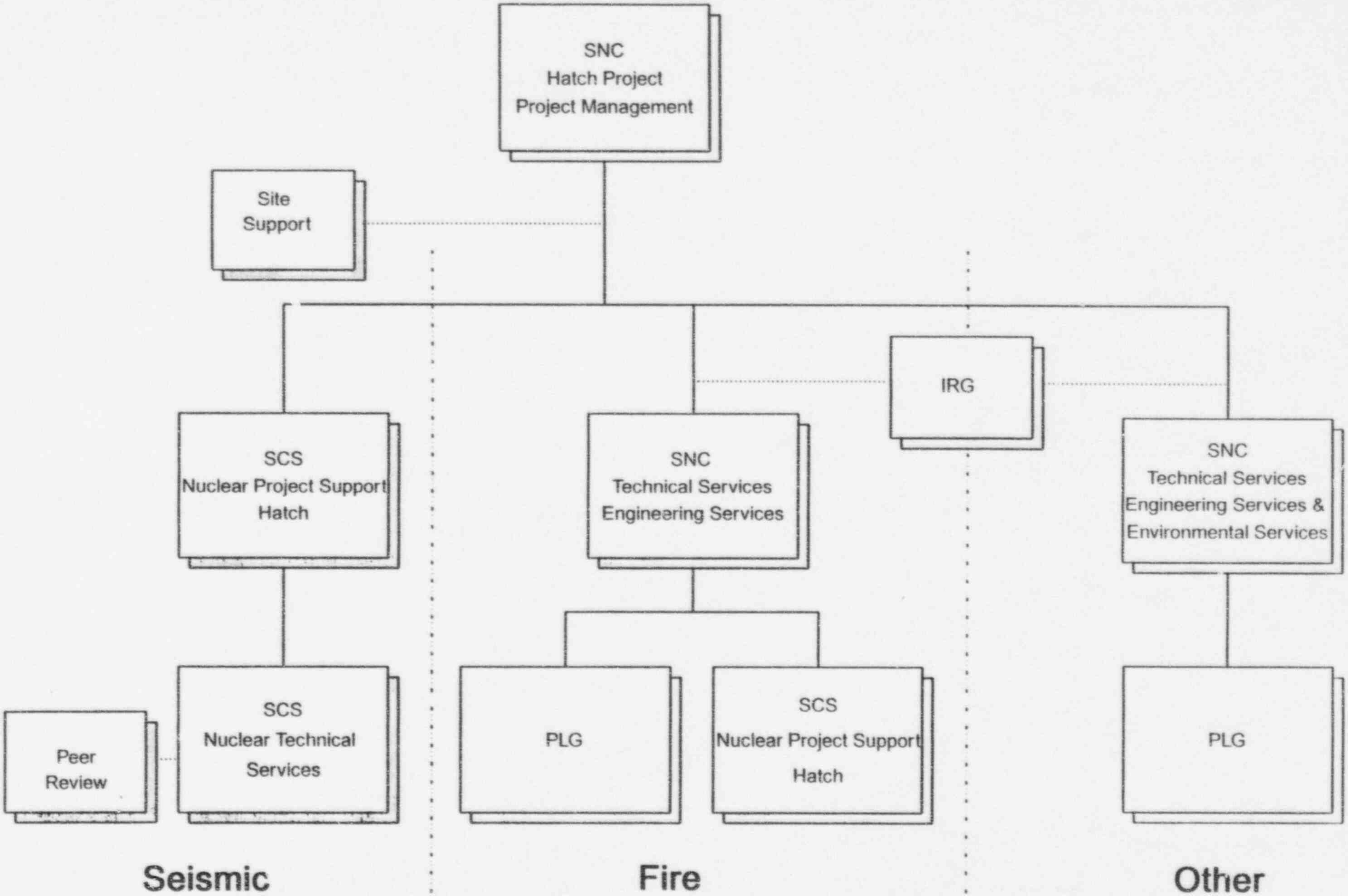


Figure 6.2-1 Hatch IPEEE Organization

6.3 AREAS OF REVIEW AND MAJOR COMMENTS

The technical review of the Plant Hatch Individual Plant Examination for External Events (IPEEE) included all aspects of the analysis. The IPEEE review philosophy was to incorporate reviews by site and corporate personnel at intermediate stages of the analyses to facilitate the addition of comments and provide more exposure to the IPEEE process, as opposed to reviewing only the completed products. Many Independent Review Group comments and questions were editorial or were related to understanding the methodology and assumptions; however, comments were received on all IPEEE topics.

6.4 RESOLUTION OF COMMENTS

During the Independent Review Group (IRG) meetings and during the staff reviews, a technical expert responsible for the preparation of the evaluation was available to address comments. In some cases, the comments resulted in editorial changes; in others, revisions were made to calculations, or additional documentation was gathered to support report conclusions. The final Plant Hatch Individual Plant Examination for External Events report reflects the resolution of comments.

The IRG concurred with the conclusions reached in the IPEEE evaluation.

7. PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

7.1 SEISMIC

7.1.1 SEISMIC PLANT IMPROVEMENTS

The extensive evaluation of the design and location of Plant Hatch summarized in this report resulted in no fundamental weakness or vulnerability to seismic hazards. While no major plant changes were determined to be necessary, the seismic analysis identified modifications of certain Unit 1 and Unit 2 components that were necessary to obtain a high-confidence-low-probability-of-failure capacity of at least 0.3 g peak ground acceleration. Appendix I contains the equipment identification number, equipment class, equipment description, the plant area where the component is located, a description of the outlier, a description of the outlier resolution, and the modification method (i.e., design change or maintenance work order) for each item. Modifications for items listed in Appendix I were completed in 1995.

7.1.2 SEISMIC UNIQUE SAFETY FEATURES

Plant Hatch structures, equipment, piping, electrical raceways, and HVAC distribution systems are all designed and constructed to resist the seismic forces associated with the Plant Hatch design basis earthquake (DBE). The results of the Seismic Margin Assessment demonstrate that, with completion of the modifications described in chapter 3 of this report, the design features described below provide sufficient capacity for Plant Hatch structures and systems to withstand a seismic event at least twice the level of the DBE.

Concrete structures are designed to meet the requirements of American Concrete Institute (ACI) Standard 318-63 (Reference 1), although special reinforcement detailing requirements were followed to produce ductile-type behavior. Steel structures are designed using the requirements of the American Institute of Steel Construction (AISC) specifications from the Manual of Steel Construction, Sixth Edition, 1963 (Reference 2).

The Unit 1 and 2 reactor buildings and the control building are reinforced concrete shear wall-type structures with rigid base mats. These structures have steel frame superstructures with cross bracing to support bridge cranes and the roof loads. The diesel generator building and the intake structure are concrete shear wall structures with rigid base mats.

Cable tray and HVAC supports are designed by the response spectrum method to withstand the calculated seismic loads using the floor response spectra corresponding to the locations where the supports are attached. The simultaneous application of the horizontal and vertical components creating the highest stresses are used to design the supports.

All safety-related mechanical equipment was qualified by analysis for both the operating basis earthquake and the DBE, with a limited number qualified by testing. Pressure boundaries of safety-related systems were evaluated to the criteria of the appropriate American Society of Mechanical Engineers (ASME) code sections. Large-bore piping was dynamically analyzed for seismic response. Two-inch and smaller seismic Class 1 piping is supported using standard spacing to produce natural frequencies above the earthquake frequency range of interest.

Electrical equipment was primarily qualified by shake table testing of prototype components with input consisting of harmonic sine beat, or similar, motions compatible with the appropriate support motion. Unit 2 electrical equipment is qualified per Institute of Electrical and Electronics Engineers (IEEE) 344-1971 (with amendments) (Reference 3). Unit 1 equipment was not qualified per IEEE 344-1971, since most of the equipment was ordered prior to issuance of IEEE 344-1971. However, Unit 1 equipment is similar to the Unit 2 equipment qualified to this standard. Some passive electrical equipment was qualified by either dynamic analysis or the conservative static coefficient method.

Non-seismic Category II equipment whose failure could affect Seismic Category I equipment is designed to meet seismic Category II/I criteria, ensuring that structural failure will not occur.

7.2 FIRE

7.2.1 FIRE PLANT IMPROVEMENTS

No plant-specific vulnerabilities to fire events were identified as a result of the fire IPEEE performed for Plant Hatch. The term vulnerabilities, as used in this report, refers to "those components, systems, operator actions, and/or plant design configuration that contribute significantly to an unacceptably high severe accident risk," and is defined in section 3.4.2 of the Plant Hatch IPE report. As such, no modifications to reduce the likelihood of core damage or radioactive material release were necessary. However, the Plant Hatch fire IPEEE was affected by a cable rerouting modification driven by regulatory issues associated with the use of Thermo-Lag fire retardant. This modification will be completed in the near future, and was incorporated in the fire PRA model.

7.2.2 FIRE UNIQUE SAFETY FEATURES

The following unique features of Plant Hatch provided added protection against severe accidents associated with fire events:

- The likelihood of a complete loss of offsite power due to fires is greatly reduced by the design of separate lines supplying offsite power to the startup transformers and unit auxiliary transformers. The 4-kV emergency buses are powered by the startup transformers, and the balance-of-plant (BOP) power is provided from the unit auxiliary transformers. Tripping open of the 230-kV switchyard breakers supplying the startup transformers would not result in a loss of power to BOP buses. Similarly, tripping open of the 230-kV switchyard breakers supplying the unit auxiliary transformers would not cause a loss of offsite power to emergency buses. Each of the 4-kV emergency buses can be supplied by either one of the two startup transformers. This further reduces the likelihood of a loss of offsite power to the 4-kV emergency buses. Finally, the location of the 4-kV emergency switchgears (i.e., diesel building switchgear rooms) also reduces the likelihood of a fire-induced loss of offsite power to the 4-kV emergency buses because only a very small segment of the offsite power supply lines is susceptible to damage by fires inside plant buildings.

- The existence of three 4-kV emergency buses per unit reduces the significance of the loss of any single division of ac power due to fire damage. Loss of a single emergency bus affects less than half of the emergency loads. The likelihood of losing more than one emergency bus due to fires was substantially decreased by the design of a separate switchgear room for each 4-kV emergency bus. Each of these switchgear rooms is equipped with full-coverage smoke detectors, CO_2, hose station, and CO_2 portable fire extinguisher.

- Each of the four 600-V emergency buses (two per unit) has both a normal and an alternate supply. Diesel generator 1B can be aligned to supply any one of the four 600-V emergency buses; thus, loss of any 4-kV emergency bus does not result in a long-term loss of any 600-V division of power. Each of these four 600-V emergency buses is located in a separate room in the control building. These 600-V switchgear rooms are equipped with full-coverage smoke detectors and fusible-link fire doors. This significantly reduces the likelihood of fire propagation from one switchgear room to the others.

- The dc loads are diversified between diesel batteries and station batteries. The diesel batteries provide control power to the large motors on the 4-kV emergency buses, while the station batteries provide instrumentation and actuation power. Therefore, the loss of a single station battery does not affect the ability to close breakers on emergency buses, and loss of a diesel battery does not affect instrumentation (such as manual and automatic opening of SRVs, actuation of safeguards on the redundant train, etc.). The diesel battery for each 4-kV emergency bus is located in a separate, enclosed room. This minimizes the potential for fire damaging the diesel batteries of more than one emergency bus. Similarly, the station battery for each division is located in a separate room with a normally closed fire door. Each room is equipped with full-coverage smoke detectors. The dc switchgear associated with each station battery is located in a separate room equipped with full-coverage smoke detectors and a fire door with fusible link. These fire protection features reduce the chance of fire damage to the dc power supply of more than one division.

- The station dc system at Plant Hatch is designed and operated in a manner that reduces the potential for loss of a dc bus and for common-cause failure of multiple buses. No bus interconnections are employed, and the batteries are not disconnected from the buses for any routine activities during power operation. Each station battery has three chargers, with only two chargers in service at any time.

- Plant service water (PSW), which provides cooling water for essential loads, is a normally operating system. Standby failure modes are not applicable to this system for fire events. Furthermore, the partitioning design in the intake structure reduces the likelihood of fire propagation causing failure of all four PSW pumps. Thermal detectors and the fire suppression equipment provided in this area also help mitigate the consequence of fires.

- Instrument air does not depend on PSW. The air compressors are supported by a dedicated closed-loop cooling water system and equipped with state-of-the-art, solid state controllers. This design represents a higher reliability and a better plant capability to cope with PSW failures. In addition, the instrument air systems for both units can be manually interconnected within a very short period of time.

- The pneumatic supply for the safety relief valves (SRVs) does not depend on ac or dc power. The nitrogen supply system is essentially a passive system which maintains nitrogen to the equipment within the drywell. Thus, the SRV accumulators are needed only if the nitrogen system isolates spuriously. If the pneumatic supply is unavailable, an emergency nitrogen hook-up station allows operators to use bottled nitrogen to maintain the SRVs. No credit was taken for this operator action in the fire IPEEE.

- The physical layout of the plant reduces dependencies on heating, ventilation , and air-conditioning (HVAC) systems. The reactor building is large and relatively open, with few enclosed areas. Reactor core isolation cooling (RCIC) and one residual heat removal (RHR) or core spray (CS) pump in each corner room could operate for a period in excess of 24 hours with no room cooling. The control building is also relatively open, with electrical switchgear rooms connected to large hallways by normally open fire doors. The 4-kV emergency buses are located in separate rooms in the diesel building, and loss of HVAC is not expected to result in electrical equipment failures. In addition, redundant room coolers are provided in each of the safeguard rooms. Fire damage to cables associated with one room cooler for each corner room would not affect the operation of the corresponding safety equipment in response to fire events.

- The main steam isolation valves for Plant Hatch do not close on high drywell pressure; thus, events that may result in a spurious high drywell pressure signal do not cause loss of the main condenser.

- Eleven SRVs are equipped for each unit. In addition, the seven automatic depressurization system (ADS) valves can be powered from either station battery. Due to the large number of SRVs available and the redundant power supply for ADS valves, hardware failure or fire damage contribution to depressurization failure is insignificant.

- Plant Hatch is equipped with four RHR pumps for low-pressure coolant injection (LPCI) and two core spray pumps for low pressure injection. Nonsafety-related equipment available for low pressure injection include three condensate pumps and three condensate booster pumps. Other low capacity systems are also available for injection. The abundance of low-pressure injection pumps makes loss of low-pressure injection an insignificant contributor to core damage.

- The reliability of RHR in the LPCI mode is improved by the use of the LPCI inverter for the LPCI injection valve and the elimination of LPCI loop-selection logic. The LPCI injection valve can be supplied by both dc and ac power sources.

- The Plant Hatch design allows the injection of fire protection system water into the vessel or drywell sprays. The emergency operating procedures provide direction to perform this action when required. No credit for this action was taken in the fire PRA. The residual heat removal service water (RHRSW) system can also provide an unlimited injection source for low-pressure injection or containment spray. While no credit was taken in the Plant Hatch fire IPEEE, control rod drive pumps can serve as an additional source of high-pressure injection.

- The arrangement of high-pressure and low-pressure injection pumps in the reactor building is such that fire propagation from one corner room to the other corner rooms is not feasible. Therefore, fire damage to multiple safeguards of different division is unlikely.

- The hardened containment vent provides an additional means of decay heat removal independent of the RHRSW and PSW pumps located in the intake structure.

- The drywell for each unit is inerted by nitrogen during normal power operation. This prevents fire occurrences inside the containment.

- The use of metal enclosures and Kaowool fire retardant material for many cable trays helps prolong the time required for fire propagation between cable trays. The provision of bottom metal enclosure for the lowest cable tray in a stack of trays and the design of line-type thermal detectors (e.g., control building hallways) under the bottom trays help prevent transient fires from propagating to the cable trays above.

- The cable spreading room is equipped with dual water and CO_2 suppression system. Many of the cable trays in this fire zone are enclosed with Kaowool fire retardant material or metal tray cover. In addition, line-type thermal detectors are provided under the cable trays. These fire protection measures significantly reduce the fire risk.

- In the main control room, some control cabinets are equipped with a full metal partition between adjacent cabinets. Fire blankets are also installed in some cabinets. These design features reduce the likelihood for fire propagation between cabinets.

- Administrative procedures such as "no combustible" zones are implemented in some plant areas (e.g., east cableway) to minimize the potential for fire initiation. The practice for spark containment during the performance of hot work also reduces the likelihood of human-induced fires.

7.3 <u>OTHER</u>

7.3.1 OTHER PLANT IMPROVEMENTS

The extensive evaluation of the design and location of Plant Hatch summarized in this report found no fundamental weakness or vulnerability to external hazards of high winds, floods, transportation, and nearby facility accidents.

A plant procedure (Reference 4) addressing naturally occurring phenomena provides instructions for plant personnel to take specific measures to prepare the plant for the effects of tornadoes, high winds, and floods. One of these measures for actual or forecasted high winds is to secure equipment inside and outside the protected area which may be damaged by high winds or may damage other equipment if blown. When the Altamaha River level is expected to crest at equal to or greater than 105 ft msl, the building and grounds personnel are notified to prepare enough sandbags to protect all entrances to the intake structure, the diesel building, the reactor building, and the turbine building.

As evaluated in section 5.3 of this report, all aircraft hazard screening criteria were met, and it was concluded that no significant aircraft hazards currently exist at Plant Hatch. However, to support continued safe operation of the plant, a general guideline was issued concerning the conduct of company aircraft operations in the vicinity of the plant. The following guideline has been incorporated into the System Aircraft Department's Operations Manual, section 4 (Flight Operations):

> During take-off and approach to nuclear facility helipads, the pilot will avoid taking the aircraft directly over structures. In general, care shall be taken to avoid flying any aircraft directly over nuclear facility structures.

7.3.2 OTHER UNIQUE SAFETY FEATURES

The following unique features of Plant Hatch provided added protection against severe accidents associated with high winds, floods and transportation and nearby facility events:

- Plant Hatch is located in a remote, sparsely populated, rural area of southern Georgia. No large industrial facilities, manufacturing plants, or military facilities are located within 5 miles of the plant.

- There is no commercial barge traffic on the Altamaha River.

- There are no commercial railroads within 5 miles of Plant Hatch. The rail spur leading to the plant does not carry commercial cargo.

- There are no airfields within 10 miles of Plant Hatch.

- The probable maximum flood for the Plant Hatch site is 110 ft which is below the 129.5 ft grade of the diesel generator building, the reactor building, and the control building.

- Records do not indicate any incidence of the Altamaha River freezing over. Therefore, ice flooding or blockage is not a credible hazard.

- Because of the great distance between the Plant Hatch site and the nearest dams, any surges created by an instantaneous dam failure would be attenuated to a fraction of the original size.

REFERENCES

1. American Concrete Institute (ACI), 318-63, Building Code Requirements for Reinforced Concrete, including 1963 Supplement, Detroit, Michigan.

2. American Institute of Steel Construction (AISC), Manual of Steel Construction, 6th Edition, Chicago, Illinois, 1963.

3. Institute of Electrical and Electronics Engineers, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Standard 344-1971.

4. Georgia Power Company, Edwin I. Hatch Nuclear Plant Abnormal Operating Procedure, 34AB-Y22-002-0S, Revision 0, Naturally Occurring Phenomena.

8. SUMMARY AND CONCLUSIONS

The major finding from this examination is that Plant Hatch has no fundamental weaknesses or vulnerabilities to severe accident risk in regard to the external events related to seismic, fire, high winds, floods, transportation and nearby facility accidents, and other external hazards.

8.1 SEISMIC

The overall result of the Plant Hatch Units 1 and 2 seismic margin assessment (SMA), as documented in this report, is that both units have a high-confidence-low-probability-of-failure (HCLPF) capacity of at least 0.3 g peak ground acceleration (pga) with the modification of certain components to raise their HCLPF capacities, described in section 3.1.4.9.

A total of 802 components were evaluated during the Unit 1 and Unit 2 seismic capability walkdowns. The walkdowns included a check for seismic capacity, anchorage (if applicable), and seismic spatial interaction for each component on the Safe Shutdown Equipment List (SSEL). The potential effects of spray, flooding, and cascade onto equipment caused by possible pipe or vessel rupture were also considered. Appendix I of this report includes a list of equipment that was classified as outliers during the seismic capability walkdowns. A brief description of the equipment, the equipment location, a description of the outlier, and a description the outlier resolution are also included in Appendix I. Equipment listed in Appendix I that could not be shown to posses a HCLPF capacity of at least 0.3 g pga in its as-built condition was modified. Equipment modifications are designed to increase the HCLPF capacity of the equipment to at least 0.3 g pga. The last column of Appendix I indicates the method of the equipment modification where applicable. The SSEL equipment not listed in Appendix I was shown to posses a HCLPF capacity of at least 0.3 g pga during the walkdown.

The following items were prescreened from further review, or a detailed SMA, based on their seismic ruggedness to withstand earthquake forces at the Plant Hatch seismic margin earthquake level of 0.3 g pga: primary containment (drywell); drywell internal structures; shear walls; footings; containment shield walls; Seismic Category I concrete and steel frame structures; dams, levees, and dikes; nuclear steam supply system primary coolant system and supports; diaphragms; control rod drive housings and mechanisms; piping; and heating, ventilation, and air-conditioning ducting and dampers.

An extensive evaluation of Plant Hatch cable and conduit raceways was performed using the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP) methodology. This evaluation, which far exceeds the SMA review recommendations, consisted of two parts: 1) a plant walkdown in which the raceways were evaluated against the SQUG GIP inclusion rules and walkdown guidelines, and 2) an analytical check of selected worst-case supports using the limited analytical review guidelines found in section 8 of the SQUG GIP. This evaluation adequately demonstrates that the Plant Hatch cable and conduit raceway systems posses a HCLPF capacity of at least 0.3 g pga.

The containment spray system is included on the SSEL to prevent early containment failure. Containment isolation valves in systems used for the safe shutdown paths are included by default in the SSEL. Containment isolations in systems not used for the safe shutdown paths are also included in the SSEL. With the resolution of open items described in Appendix I complete, all the containment isolation valves and containment spray equipment included on the SSEL were demonstrated to posses a HCLPF capacity of at least 0.3 g pga.

A full-scope relay chatter review was performed for all Unresolved Safety Issue (USI) A-46 SSEL equipment. A review of all equipment included on the SSEL only for IPEEE, but not in the A-46 scope, was performed to identify any additional low-seismic-ruggedness relays. All of the low-seismic-ruggedness relays identified as part of the IPEEE evaluation for both units were resolved at a HCLPF level of at least 0.3 g pga.

Additional evaluations were performed for masonry walls, main control room ceilings, Seismic Category II structures that could affect Category I structures containing SSEL equipment, the reactor building roof structure, torus, reactor internals, buried structures, and soils issues. The evaluation results confirmed that each of these items have a HCLPF capacity of at least 0.3 g pga.

The evaluations of USI A-45 and Generic Issue (GI)-131 revealed that there are no plant-specific vulnerabilities at Plant Hatch related to these issues.

8.2 FIRE

The fire-induced core damage frequencies (CDFs) for Unit 1 and Unit 2 were found to be 7.5E-06 per year, and 5.4E-06 per year, respectively. This represents a fire -induced CDF contribution that is slightly less than 36 percent for Unit 1 and 25 percent for Unit 2 of the internal events CDF of 2.1E-05 per year and 2.2E-05 per year, respectively. The uncertainty in these fire-induced frequencies is expressed in terms of probability distributions which show that there is 90 percent confidence that the fire-induced CDFs for Unit 1 vary from 1.9E-6 events per year to 1.6E-5 events per year, and for Unit 2 from 1.3E-6 events per year to 1.3E-5 events per year. These distributions were obtained by incorporating many sources of uncertainty and reflect the available state of knowledge about the current models, analyses, and data used in the study.

The fire IPEEE did not uncover any unique containment structural failure modes. In addition, the IPE source term was determined to envelop the source term of the IPEEE. Since the fire-induced CDF is relatively low and the conditional frequency of large releases due to fires is lower than that for the IPE, it can be concluded that the containment function is not significantly affected by fires. The evaluations of fire risk scoping study issues, decay heat removal requirements, and GI-57 revealed that there are no plant-specific vulnerabilities at Plant Hatch related to these issues.

Based on the results of the detailed analysis of fire scenarios, a number of observations relating to plant design and operation were made. These observations pertain to the major contributors to the fire-induced CDF, the relative importance of various design features in buildings that have accident mitigation equipment, significance of fire protection features, and other aspects of fire initiation, propagation, and suppression. The key observations include:

- 4-kV emergency switchgear rooms, cable spreading room, and main control room are the most important plant locations from the fire-induced CDF standpoint.

- Significant contribution to fire-induced CDF results from cable fires in all of the locations with a substantial amount of cable within a small area. This includes the cable spreading room, main control room, east cableway, vertical cable chase, computer room, and control building north and south corridor on elevation 130 ft.

- Each of the 4-kV emergency switchgears is located in a separate room of the diesel generator building. Since the 4-kV emergency switchgear rooms are the only locations inside any plant buildings that contain the offsite power supply cables from both of the startup transformers to the 4-kV emergency switchgears, these are the only areas where a loss of offsite power supply to the emergency buses can be caused by fires.

- The location contributing significantly to the fire-induced CDF in the reactor building is the reactor building working floor at elevation 130 ft. A large number of cables are located within close proximity of the motor control centers in this area. Fire damage to these cables could cause failure of containment venting in addition to one loop of residual heat removal (RHR) and core spray. The reactor building safeguards areas (i.e., high-pressure coolant injection and corner rooms) are not high contributors to fire-induced CDF since fire damage to equipment and cables in these areas can only affect one high-pressure injection system or one loop of core spray and RHR systems. Fire propagation to impact more than one of these areas is not feasible. The upper floors of the reactor building are not high contributors to fire-induced CDF because the impact of fire damage to equipment and cables in the upper floor areas is much less severe.

- The locations contributing significantly to the fire-induced CDF in the control building include the cable spreading room, main control room, vertical cable chase, division I station battery room, division I 600-V ac switchgear room, and division I dc switchgear room. The fire-induced CDF contribution of division I fire zones for 600-V ac and 125-V dc sources is much higher the equivalent locations of division II in that loss of division I 600-V ac or 125-V dc power would result in the loss of condensate system, feedwater system, and main condenser.

- The only location contributing significantly to fire-induced CDF in the turbine building is the east cableway. The other areas in turbine building are not important because the impact of fire damage to equipment and cables is less severe.

- The fire-induced CDF contribution of the intake structure is significant for three reasons. First, due to the partitioning design, fire propagation resulting in failure of all four plant service water pumps is relatively unlikely. Second, the impact of loss of one PSW division is not relatively severe and loss of cooling to most of the room coolers is recoverable. Third, fire damage to the residual heat removal service water pumps would not cause a reactor trip.

- The only location contributing significantly to fire-induced CDF in the diesel generator building is the 4-kV emergency switchgear rooms. Fire damage to equipment and cables in most of the other locations in this building would not cause a reactor trip.

- The other buildings in the plant are not significant contributors to the fire-induced CDF because they do not contain core damage mitigation equipment and its associated cables.

- Due to limited combustible loading along the border between fire zones with credible fire propagation pathways, propagation between fire zones is not likely

- The use of metal enclosures and Kaowool fire retardant material for cable trays helps prolong the time required for fire propagation between cable trays. The provision of bottom metal enclosure for the lowest cable tray in a stack of trays and the design of line-type thermal detectors under the bottom trays help prevent transient fires from propagating to the cable trays above.

8.3 OTHER

In the high winds hazard evaluation, no potential vulnerability was identified. The design of structures at Plant Hatch meets the criteria of the 1975 Standard Review Plan (SRP), except that only two of the seven tornado-generated missile types were analyzed. The optional step of applying PRA methods was used to determine the tornado-missile-impact risk. It was concluded that the contribution to core-damage frequency from high winds, including tornadoes, is less than 1E-6 per year, and the contribution to plant risk is insignificant.

The external flood hazard was evaluated using the more conservative of either the 1975 SRP National Weather Service criteria or the design basis criteria. The probable maximum flood (PMF) for the plant site is estimated by National Weather Service Hydrometeorological Report No. 51 to result in an Altamaha River flood level of 110 ft which is the floor level of the intake structure (the diesel generator building and all other safety-related buildings are located well above this level at 129 ft). The river intake structure will protect the service water pumps from coincidental high-wind-generated waves. The recurrence frequency of such a flood is estimated conservatively at once every 1E+8 years, which makes the scenario unlikely. The dam failure scenario coincident with the PMF would occur at a frequency that is much lower than the recurrence frequency of the PMF; therefore, this scenario is very unlikely. Local intense rainfall, even at world-record level, would be sufficiently mitigated by the site's drainage system and topography. Therefore, if it is conservatively assumed that plant flooding from external causes leads directly to core damage, its frequency is estimated to be less than 1E-8 per year, which is insignificant.

In the areas of transportation and nearby facility accidents, the existing Plant Hatch design conforms to the 1975 SRP criteria. No significant changes were identified which impacted the plant design.

8.4 COORDINATION WITH OTHER EXTERNAL EVENT PROGRAMS

This report addresses Fire Risk Scoping Study issues (NUREG/CR-5088), Shutdown Decay Heat Removal Requirements (USI A-45), Effects of Fire Protection System Actuation of Safety-Related Equipment (GI-57), and Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants (GI-131).

Verification of Seismic Adequacy of Equipment in Operating Plants (USI A-46) is addressed in a separate report.

During the evaluation of seismic, fire and other external events, no plant vulnerabilities were identified associated with other USIs or GIs.

APPENDIX A

SEISMIC REVIEW SAFE SHUTDOWN EQUIPMENT LIST
PLANT HATCH UNIT 1

The Seismic Review Safe Shutdown Equipment List (SSEL) is a subset of the composite SSEL included in Appendix E of this report. The Seismic Review SSEL contains all mechanical and electrical equipment, tanks, and heat exchangers for which a seismic evaluation was peformed.

The format of the Seismic Review SSEL is as given by the Safe Shutdown Equipment Manager computer program supplied by the Seismic Qualification Utility Group (SQUG). The Seismic Review SSEL includes the following information, where applicable and available, for each component. Components are sorted by mark number.

Column Number	Description
1	Unique line number
2	Equipment class (see table 1 of this appendix)
3	Equipment identification number
4	Equipment description
5	Building or area in which the equipment is located
6	Floor elevation from which the equipment can be seen
7	Building grid row/column (or panel number) indicating equipment location

APPENDIX A

TABLE 1
EQUIPMENT CLASS DESIGNATIONS

Equipment Class (Column 2)	Description
01	Motor control centers
02	Low-voltage switchgear
03	Medium-voltage switchgear
04	Transformers
05	Horizontal pumps
06	Vertical pumps
07	Fluid-operated valves
08A	Motor-operated valves
08B	Solenoid-operated valves
09	Fans
10	Air handlers
11	Chillers
12	Air compressors
13	Motor generators
14	Distribution panels
15	Batteries on racks
16	Battery chargers and inverters
17	Engine generators
18	Instruments on racks
19	Temperature sensors
20	Instrumentation and control panels and cabinets
21	Tanks and heat exchangers

APPENDIX A
SEISMIC REVIEW SSEL
E. I. HATCH NUCLEAR PLANT - UNIT 1

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
0810	20	2H11-P652	ELECT AUX PWR CONTROL CONSOLE	CONTROL	164	N/A
0770	20	2H21-P231	DIESEL GEN 2B - RELAY PANEL 2B	DIESEL	130	N/A
0439	06	2P41-C002	PSW STANDBY PUMP 1B DIESEL	INTAKE	111	
0440	07	2P41-F340	PSW ISOL VALVE TO 1B DIESEL	DIESEL	138	D/01
0790	03	2R22-S006	4160V SWGR EMERGENCY BUS 2F	DIESEL	130	C/03
0789	01	2R27-S037	LOCAL STARTER FOR 2P41-C002	DIESEL	130	C/02
0823	18	2R43B-M01	DIESEL B UNIT 1/2 MODE SWITCH	DIESEL	130	
0445	0	B21-A003A	AIR ACCUM FOR RELIEF VALVE A	DRYWELL	148	AZ043
0446	0	B21-A003B	AIR ACCUM FOR RELIEF VALVE B	DRYWELL	148	AZ077
0447	0	B21-A003C	AIR ACCUM FOR RELIEF VALVE C	DRYWELL	148	AZ073
0449	0	B21-A003E	AIR ACCUM FOR RELIEF VALVE E	DRYWELL	148	AZ088
0530	0	B21-A003F	AIR ACCUM FOR RELIEF VALVE F	DRYWELL	148	AZ295
0531	0	B21-A003G	AIR ACCUM FOR RELIEF VALVE G	DRYWELL	148	AZ266
0532	0	B21-A003H	AIR ACCUM FOR RELIEF VALVE H	DRYWELL	148	AZ331
0533	0	B21-A003J	AIR ACCUM FOR RELIEF VALVE J	DRYWELL	148	AZ320
0701	07	B21-F013A	RPV PRESSURE RELIEF VALVE	DRYWELL	148	AZ036
0702	07	B21-F013B	RPV SAFETY RELIEF VALVE	DRYWELL	148	AZ054
0703	07	B21-F013C	RPV PRESSURE RELIEF VALVE	DRYWELL	148	AZ064
0705	07	B21-F013E	RPV SAFETY RELIEF VALVE	DRYWELL	148	AZ056
0706	07	B21-F013F	RPV SAFETY RELIEF VALVE	DRYWELL	148	AZ304
0707	07	B21-F013G	RPV PRESSURE RELIEF VALVE	DRYWELL	148	AZ293
0708	07	B21-F013H	RPV PRESSURE RELIEF VALVE	DRYWELL	148	AZ324
0709	07	B21-F013J	RPV SAFETY RELIEF VALVE	DRYWELL	148	AZ306
0693	07	B21-F022A	INBOARD MSIV	DRYWELL	130	RB/R07
0694	07	B21-F022B	INBOARD MSIV	DRYWELL	130	RB/R07
0695	07	B21-F022C	INBOARD MSIV	DRYWELL	130	RB/R07
0696	07	B21-F022D	INBOARD MSIV	DRYWELL	130	RB/R07
0697	07	B21-F028A	OUTBOARD MSIV	REACTOR	130	RB/R07
0698	07	B21-F028B	OUTBOARD MSIV	REACTOR	130	RB/R07
0699	07	B21-F028C	OUTBOARD MSIV	REACTOR	130	RB/R07

Report Date/Time: 12-20-95 / 15:24:50
Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
0700	07	B21-F028D	OUTBOARD MSIV	REACTOR	130	RB/R07
0726	18	B21-N093A	RPV LEVEL 8 LT	REACTOR	158	RH/R08
0730	18	B21-N093B	RPV LEVEL 8 LT	REACTOR	158	RH/R04
0460	08A	B31-F023A	RECIRC PUMP C001A SUCTION VALVE	DRYWELL	117	RF/R07
0466	08B	C11-D001-117	PILOT SCRAM SOLENOID	REACTOR	130	HCU
0467	08B	C11-D001-118	PILOT SCRAM SOLENOID	REACTOR	130	HCU
0468	0	C11-D001-125	SCRAM ACCUMULATOR	REACTOR	130	HCU
0469	07	C11-D001-126	SCRAM INLET VALVE	REACTOR	130	HCU
0470	07	C11-D001-127	SCRAM OUTLET VALVE	REACTOR	130	HCU
0462	08B	C11-F009A	SDV VENT & DRAIN PILOT VALVE	REACTOR	130	RA/R05
0523	08B	C11-F009B	SDV VENT & DRAIN PILOT VALVE	REACTOR	130	RA/R05
0463	07	C11-F010A	SDV VENT VALVE	REACTOR	130	RH/R12
0464	07	C11-F010B	SDV VENT VALVE	REACTOR	130	RH/R03
0465	07	C11-F011	SDV DRAIN VALVE	REACTOR	130	RB/R03
0480	07	C11-F035A	SDV VENT VALVE	REACTOR	130	RH/R12
0481	07	C11-F035B	SDV VENT VALVE	REACTOR	130	RH/R12
0482	07	C11-F037	SDV DRAIN VALVE	REACTOR	130	RB/R04
0479	08B	C11-F040A	SDV VENT & DRAIN PILOT VALVE	REACTOR	130	RA/R05
0524	08B	C11-F040B	SDV VENT & DRAIN PILOT VALVE	REACTOR	130	RA/R05
0491	08B	C11-F110A	BACKUP SCRAM VALVE	REACTOR	130	RA/R05
0492	08B	C11-F110B	BACKUP SCRAM VALVE	REACTOR	130	RA/R05
0001	14	C71-P001	RPS POWER DISTRIBUTION PANEL BUS A	CONTROL	130	TG/T12
0840	18	C71-P003E	PROTECTION PANEL	CONTROL	130	TF/T12
0841	18	C71-P003F	PROTECTION PANEL	CONTROL	130	TF/T12
0839	18	C71-S002	LINE VOLTAGE REGULATOR	CONTROL	130	
0775	20	C82-P002	REMOTE SHUTDOWN PANEL	REACTOR	158	RH/R05
0354	21	E11-B001A	RHR HEAT EXCHANGER A	REACTOR	087	RL/R13
0356	21	E11-B001B	RHR HEAT EXCHANGER B	REACTOR	087	RL/R03
0290	06	E11-C001A	RHR SERVICE WATER PUMP 1A	INTAKE	111	
0357	06	E11-C001D	RHR SERVICE WATER PUMP 1D	INTAKE	111	

Report Date/Time: 12-20-95 / 15:24:50
Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
0292	06	E11-C002A	RHR PUMP 2A	REACTOR	087	RL/R11
0358	06	E11-C002D	RHR PUMP 2D	REACTOR	087	RL/R02
0335	08A	E11-F007A	RHR PUMP 2A & 2C MIN FLOW BYPASS VLV	REACTOR	087	RL/R11
0362	08A	E11-F007B	RHR PUMP 2B & 2D MIN FLOW BYPASS VLV	REACTOR	087	RL/R03
0304	08A	E11-F015A	RHR LPCI DISCHARGE VALVE	REACTOR	130	RJ/R08
1066	08A	E11-F016A	CONTAINMENT SPRAY DISCH VALVE	REACTOR	158	RH/R08
1029	08A	E11-F016B	CONTAINMENT SPRAY DISCH VALVE	REACTOR	158	RH/R08
0340	08A	E11-F017A	RHR LPCI DISCHARGE VALVE	REACTOR	130	RJ/R08
1067	08A	E11-F021A	CONT SPRAY INBRD GATE MOV	REACTOR	158	RH/R08
1030	08A	E11-F021B	CONT SPRAY INBRD GATE MOV	REACTOR	158	RH/R08
0366	08A	E11-F024B	RHR TEST LINE TORUS ISO	REACTOR	087	RL/R05
1064	08A	E11-F027A	SUPP POOL SPRAY VALVE	REACTOR	087	RJ/R10
1027	08A	E11-F027B	SUPP POOL SPRAY VALVE	REACTOR	087	RL/R05
0369	08A	E11-F028B	RHR INLET TO SUPP POOL VALVE	REACTOR	087	RL/R05
0344	08A	E11-F048A	RHR HX A BYPASS VALVE	REACTOR	087	RL/R13
0371	08A	E11-F048B	RHR HX B BYPASS VALVE	REACTOR	087	RJ/R02
0311	08A	E11-F068A	RHR HX A TUBE TO SHELL OUTLET	REACTOR	087	RH/R13
0374	08A	E11-F068B	RHR HX B TUBE TO SHELL OUTLET	REACTOR	087	RH/R02
0317	18	E11-N002A	RHR HX A TUBE TO SHELL DP TRANS	REACTOR	087	H21-P018
0380	18	E11-N002B	RHR HX B TUBE TO SHELL DP TRANS	REACTOR	087	H21-P021
0319	18	E11-N007A	RHRSW HX A FLOW TRANSMITTER	REACTOR	087	H21-P018
0382	18	E11-N007B	RHRSW HX B FLOW TRANSMITTER	REACTOR	087	RL/R03
0322	18	E11-N017A	RHR HX A INLET PRESSURE SWITCH	REACTOR	087	RH/R13
0385	18	E11-N017B	RHR HX B INLET PRESSURE SWITCH	REACTOR	087	RL/R02
0323	18	E11-N017C	RHR HX A INLET PRESSURE SWITCH	REACTOR	087	RH/R13
0386	18	E11-N017D	RHR HX B INLET PRESSURE SWITCH	REACTOR	087	RL/R02
1116	18	E11-N055A	RHR PUMP A DISCHARGE PT	REACTOR	087	RJ/R12
1117	18	E11-N055B	RHR PUMP B DISCHARGE PT	REACTOR	087	RL/R03
1118	18	E11-N055C	RHR PUMP C DISCHARGE PT	REACTOR	087	RJ/R12
1119	18	E11-N055D	RHR PUMP D DISCHARGE PT	REACTOR	087	RJ/R03

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
1120	18	E11-N056A	RHR PUMP A DISCHARGE PT	REACTOR	087	RJ/R12
1121	18	E11-N056B	RHR PUMP B DISCHARGE PT	REACTOR	087	RL/R03
1122	18	E11-N056C	RHR PUMP C DISCHARGE PT	REACTOR	087	RJ/R12
1123	18	E11-N056D	RHR PUMP D DISCHARGE PT	REACTOR	087	RJ/R03
0324	18	E11-N082A	RHR PUMP A & C FLOW DP TRANS	REACTOR	087	RL/R10
0387	18	E11-N082B	RHR PUMP 2B & 2D FLOW DP TRAN	REACTOR	087	RL/R05
1124	18	E11-N094A	DRYWELL PT	REACTOR	158	RC/R08
1125	18	E11-N094B	DRYWELL PT	REACTOR	158	RJ/R04
1126	18	E11-N094C	DRYWELL PT	REACTOR	158	RC/R08
1127	18	E11-N094D	DRYWELL PT	REACTOR	158	RJ/R04
0394	06	E21-C001B	CORE SPRAY PUMP B	REACTOR	087	RL/R02
0397	08A	E21-F005B	CORE SPRAY INLET VALVE	REACTOR	158	RF/R04
0400	08A	E21-F031B	MINIMUM FLOW BYPASS VALVE	REACTOR	087	RL/R02
0403	18	E21-N051B	CORE SPRAY FLOW TRANSMITTER	REACTOR	087	RL/R02
1130	18	E21-N052A	CS PUMP C001A HIGH PRESS	REACTOR	087	RJ/R12
1131	18	E21-N052B	CS PUMP C001B HIGH PRESS	REACTOR	087	RL/R03
1128	18	E21-N055A	CS PUMP C001A HIGH PRESS	REACTOR	087	RJ/R12
1129	18	E21-N055B	CS PUMP C001B HIGH PRESS	REACTOR	087	RL/R03
0109	05	E41-C001	HPCI PUMP	REACTOR	087	RG/R01
0110	05	E41-C002	HPCI TURBINE	REACTOR	087	RL/R02
0111	05	E41-C002-3	HPCI LUBE OIL PUMP	REACTOR	087	RL/R01
0112	08A	E41-F001	HPCI TURBINE STEAM SUPPLY VLV	REACTOR	087	RL/R01
0115	08A	E41-F004	HPCI PUMP SUCT FROM CST	REACTOR	087	RH/R01
0116	08A	E41-F006	HPCI PUMP INBO DISCH VLV	REACTOR	087	RA/R07
0119	08A	E41-F012	HPCI MIN FLOW BYPASS VLV	REACTOR	087	RH/R02
0122	08	E41-F041	HPCI PUMP SUCT FROM SUPP POOL	REACTOR	087	RH/R01
0123	08A	E41-F042	HPCI PUMP SUCT FROM SUPP POOL	REACTOR	087	RH/R02
0125	08A	E41-F059	BAR COND COOLING WATER VLV	REACTOR	087	RL/R02
1081	07	G11-F003	DRYWELL FL DR PMP ISO VLV	REACTOR	087	RE/R03
1082	07	G11-F004	DRYWELL FL DR PMP ISO VLV	REACTOR	087	RE/R03

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
1079	07	G11-F019	DRYWELL EQ DR PMP ISO VLV	REACTOR	087	RB/R05
1080	07	G11-F020	DRYWELL EQ DR PMP ISO VLV	REACTOR	087	RB/R05
1077	08A	G31-F001	RWCU INBOARD ISO GATE VLV	DRYWELL	158	AZ 170
1078	08A	G31-F004	RWCU OUTBOARD ISO VLV	REACTOR	158	RH/R07
0734	20	H11-P601	REACTOR & CONT COOLING & ISOLA CONTROL PANEL	CONTROL	164	N/A
0735	20	H11-P602	REACTOR WTR CLEAN UP & RECIR CONTROL PANEL	CONTROL	164	N/A
0736	20	H11-P603	REACTOR CONTROL PANEL	CONTROL	164	N/A
0737	20	H11-P605A	DIV 1 ANALOG SIG CONV/ISOLATION	CONTROL	164	N/A
0738	20	H11-P605B	DIV 2 ANALOG SIG CONV/ISOLATION	CONTROL	164	N/A
0739	20	H11-P606	START U RANGE NEUTRON MONTORING	CONTROL	164	N/A
0807	20	H11-P608	POWER RANGE NEUTRON MONITORING PANEL	CONTROL	164	N/A
0740	20	H11-P609	CHAN A PRI ISO SYS REAC PROT SYS	CONTROL	164	N/A
0842	20	H11-P610	RPS TEST MON VB	CONTROL	164	TE/T11
0741	20	H11-P611	CHAN B PRI ISO SYS REAC PROT SYS	CONTROL	164	N/A
0742	20	H11-P612	FW & RECIR INSTRUMENTATION PANEL	CONTROL	164	N/A
0808	20	H11-P613	PROCESS INSTRUMENTATION PANEL	CONTROL	164	N/A
1132	20	H11-P614	NUCLEAR STEAM RECORDER VB	CONTROL	164	TE/T11
0743	20	H11-P617	CHAN A RHR CORE SPRAY RELAY PANEL	CONTROL	164	N/A
0744	20	H11-P618	CHAN B RHR CORE SPRAY RELAY PANEL	CONTROL	164	N/A
0745	20	H11-P620	HPCI RELAY COMPUTER EQUIPMENT PANEL	CONTROL	164	N/A
0746	20	H11-P622	INBOARD-PRI CONTROL ISOLATION RELAYS PANEL	CONTROL	164	N/A
0747	20	H11-P623	OUTBOARD-PRI CONTAIN ISOLATION RELAYS PANEL	CONTROL	164	N/A
0748	20	H11-P626	CORE SPRAY CONTR PNL DIV I	CONTROL	164	N/A
0749	20	H11-P627	CORE SPRAY CONTR PNL DIV II	CONTROL	164	N/A
0809	20	H11-P628	AUTOMATIC BLOWDOWN RELAY PANEL	CONTROL	164	N/A
0750	20	H11-P650	TURB, FW & COND CONTROL CONSOLE	CONTROL	164	N/A
0751	20	H11-P651	GEN & STA SER CONTROL CONSOLE	CONTROL	164	N/A
0752	20	H11-P652	ELEC AUX PWR CONTROL CONSOLE	CONTROL	164	N/A
0795	20	H11-P654	GAS TREAT & VENT VERTICAL PANEL	CONTROL	164	N/A

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
0796	20	H11-P657	VENT & DRYWELL INERT VERTICAL PANEL	CONTROL	164	N/A
0797	20	H11-P691	ANALOG SIGNAL CONVERSION/ISOLATION PANEL	CONTROL	164	N/A
0798	20	H11-P700	ANALYZER/VENT & LEAK DETECTION PANEL	CONTROL	164	N/A
0799	20	H11-P921	ATTS RPS PANEL	CONTROL	164	N/A
0800	20	H11-P922	ATTS RPS PANEL	CONTROL	164	N/A
0801	20	H11-P923	ATTS RPS PANEL	CONTROL	164	N/A
0802	20	H11-P924	ATTS RPS PANEL	CONTROL	164	N/A
0802	20	H11-P925	ATTS ECCS PANEL	CONTROL	164	N/A
0804	20	H11-P926	ATTS ECCS PANEL	CONTROL	164	N/A
0805	20	H11-P927	ATTS ECCS PANEL	CONTROL	164	N/A
0806	20	H11-P928	ATTS ECCS PANEL	CONTROL	164	N/A
0786	18	H21-P016	CS/HPCI LEAK DET RACK	REACTOR	130	RF/R04
0772	18	H21-P018	RHR INSTRUMENT RACK-CHANNEL A	REACTOR	087	RL/R13
0777	18	H21-P019	CORE SPRAY SYSTEM B RACK	REACTOR	087	RL/R03
0776	18	H21-P021	RHR CHANNEL B RACK	REACTOR	087	RL/R03
0787	18	H21-P036	HPCI LEAK DET RACK	REACTOR	130	RF/R04
0773	20	H21-P173	SHUTDOWN INSTRUMENT PANEL	REACTOR	130	RL/R03
0753	20	H21-P175	HOT SHUTDOWN PUMP CONTROL PANEL	DIESEL	130	N/A
0754	20	H21-P200	DIESEL GEN 1A - RELAY PANEL 1A	DIESEL	130	N/A
0755	20	H21-P201	DIESEL GEN 1B - RELAY PANEL 1A	DIESEL	130	N/A
0756	20	H21-P202	DIESEL GEN 1C - RELAY PANEL 1A	DIESEL	130	N/A
0757	20	H21-P230	DEISEL GEN 1A - RELAY PANEL 1B	DIESEL	130	N/A
0758	20	H21-P231	DIESEL GEN 1B - RELAY PANEL 1B	DIESEL	130	N/A
0759	20	H21-P232	DIESEL GEN 1C - RELAY PANEL 1B	DIESEL	130	N/A
0760	20	H21-P245	600 VOLT SWGR 1C CONTROL PANEL	CONTROL	130	N/A
0761	20	H21-P246	600 VOLT SWGR 1D CONTROL PANEL	CONTROL	130	N/A
0817	18	H21-P248	250V DC SWITCHGEAR CONTROL PANEL	DIESEL	130	TF/T11
0818	18	H21-P249	250V DC SWITCHGEAR CONTROL PANEL	DIESEL	130	TB/T11
0762	20	H21-P255	MOV & FUEL PUMP CTRL PANEL 1A - DIV I	DIESELDB	130	N/A
0763	20	H21-P256	MOV & FUEL PUMP CTRL PANEL 1B - DIV II	DIESEL	130	N/A

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
0764	20	H21-P257	GEN 1A HEAT & VENT CONTROL PANEL	DIESEL	130	N/A
1101	20	H21-P258	GEN 1B H&V CONT PNL	DIESEL	130	D01
1102	20	H21-P259	GEN 1C H&V CONT PNL	DIESEL	130	E01
0765	20	H21-P266	MOV CONTROL PANEL 1A-DIV I	INTAKE	111	N/A
0774	20	H21-P267	MOV CONTROL PANEL 1B-DIV II	INTAKE	111	N/A
0092	18	H21-P285	125/250V STATION BATT 1A FUSE BOX	CONTROL	112	TF/T11
0093	18	H21-P286	125/250V STATION BATT 1A FUSE BOX	CONTROL	112	TE/T11
0094	18	H21-P287	125/250V STATION BATT 1A FUSE BOX	CONTROL	112	TF/T11
0095	18	H21-P288	125/250V STATION BATT 1B FUSE BOX	CONTROL	112	TC/T11
0096	18	H21-P289	125/250V STATION BATT 1B FUSE BOX	CONTROL	112	TE/T11
0097	18	H21-P290	125/250V STATION BATT 1B FUSE BOX	CONTROL	112	TD/T10
0714	18	H21-P294	TERMINAL BOX	CONTROL	112	TE/T12
0715	18	H21-P295	TERMINAL BOX	CONTROL	112	TE/T12
0767	20	H21-P303	DIESEL 1A LEAD TIMER PANEL	DIESEL	130	N/A
0768	20	H21-P304	DIESEL 1B LEAD TIMER PANEL	DIESEL	130	N/A
0769	20	H21-P305	DIESEL 1C LEAD TIMER PANEL	DIESEL	130	N/A
0781	18	H21-P404A	INSTRUMENT RACK	REACTOR	158	RH/R08
0783	18	H21-P404B	INSTRUMENT RACK	REACTOR	158	RH/R08
0782	18	H21-P405A	INSTRUMENT RACK	REACTOR	158	RH/R03
0778	18	H21-P405B	INSTRUMENT RACK	REACTOR	158	RH/R03
0779	18	H21-P409	INSTRUMENT RACK	REACTOR	130	RH/R10
0780	18	H21-P410	INSTRUMENT RACK	REACTOR	130	RF/R03
0784	18	H21-P414A	INSTRUMENT RACK	REACTOR	087	RJ/R02
0785	18	H21-P414B	INSTRUMENT RACK	REACTOR	087	RJ/R02
0788	18	H21-P434	INSTRUMENT RACK	REACTOR	087	RJ/R02
1113	20	H21-P530A	FAN 1A PANEL	INTAKE	111	N/A
1114	20	H21-P530B	FAN 1B PANEL	INTAKE	111	N/A
1115	20	H21-P530C	FAN 1C PANEL	INTAKE	111	N/A
0406	06	P41-C001A	PLANT SERVICE WATER PUMP 1A	INTAKE	111	
0426	06	P41-C001B	PLANT SERVICE WATER PUMP 1B	INTAKE	111	

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
0837	07	P41-F035B	HPCI ROOM COOLER INLET VALVE	REACTOR	087	RL/R01
0428	07	P41-F036B	RHR/CS ROOM COOLER INLET VALVE	REACTOR	087	RL/R03
0407	07	P41-F037A	RHR PUMP COOLER 2A INLET VALVE	REACTOR	087	RL/R13
0429	07	P41-F037D	RHR PUMP COOLER 2D INLET VALVE	REACTOR	087	RL/R03
0408	07	P41-F039A	RHR/CS ROOM COOLER 2A VALVE	REACTOR	087	RL/R11
0897	08B	P41-F123A	Z41-B025 PSW INLET ISOL SV	CONTROL	180	TB/T11
0972	08B	P41-F123B	Z41-B025 PSW INLET ISOL SV	CONTROL	180	TB/T11
0412	08	P41-F310A	PSW TURBINE BLDG ISOL VALVE	YARD		
0431	08A	P41-F310B	PSW TURBINE BLDG ISOL VALVE	YARD		
0442	08A	P41-F312	PSW RETURN LINE ISOL VALVE	INTAKE	098	
0975	08A	P41-F422A	Z41-B008B ISOL GLOBE MOV	CONTROL	180	TB/T11
0420	18	P41-N200A	PSW STRAINER DP SWITCH	INTAKE	087	
0436	18	P41-N200B	PSW STRAINER DP SWITCH	INTAKE	087	
0976	18	P41-N520	PSW CB A/C-1B DI DPS	CONTROL	180	TB/T11
0935	18	P41-N521	PSW CD A/C-1B DII DPS	CONTROL	180	TB/T11
0525	0	P70-A002A	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RL/R09
0526	0	P70-A002B	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RL/R09
0527	0	P70-A002C	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RL/R09
0450	07	P70-F001A	DW PNEUMATIC NITROGEN SUPPLY VALVE	REACTOR	087	RA/R02
0528	0	P70-F084	EMERGENCY NITROGEN ISOL VALVE	REACTOR	130	RL/R09
0529	0	P70-F141	EMERGENCY NITROGEN ISOL VALVE	REACTOR	130	RL/R09
0454	18	P70-N022A	DRYWELL PNEUMATIC FLOW TRANS	REACTOR	130	RA/R06
0455	18	P70-N022B	DRYWELL PNEUMATIC FLOW TRANS	REACTOR	130	RA/R06
0010	04	R11-S004	45KVA 600-120/208V PWR XFMR 1D	DIESEL	130	D/02
0011	04	R11-S005	45KVA 600-120/208V PWR XFMR 1E	DIESEL	130	E/02
0012	04	R11-S006	45KVA 600-120/208V PWR XFMR 1F	DIESEL	130	F/02
0013	04	R11-S039	45KVA 600-120/208V TRANSFORMER	REACTOR	130	RF/R13
0014	04	R11-S040	45KVA 600-120/208V TRANSFORMER	REACTOR	130	RF/R02
0015	04	R11-S041	112.5 KVA 600-120/208V ESSENTIAL XFMR 1B	CONTROL	130	TF/T11
0016	04	R11-S042	112.5 KVA 600-120/208V ESSENTIAL XFMR 1C	CONTROL	130	TD/T11

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
0815	04	R11-S071	MISC POWER TRANSFORMER	DIESEL	130	
0017	03	R22-S005	4160V SWGR EMERGENCY BUS 1E	DIESEL	130	D/03
0018	03	R22-S006	4160V SWGR EMERGENCY BUS 1F	DIESEL	130	E/03
0019	03	R22-S007	4160V SWGR EMERGENCY BUS 1G	DIESEL	130	F/03
0020	02	R22-S016	250V DC BATTERY DIV 1 SWGR 1A	CONTROL	130	TF/T11
0021	02	R22-S017	250V DC BATTERY DIV 2 SWGR 1B	CONTROL	130	TB/T11
0022	02	R23-S003	600V SWGR EMERGENCY BUS 1C & 4160-600V XFMR	CONTROL	130	TE/T10
0023	02	R23-S004	600V SWGR EMERGENCY BUS 1D & 4160-600V XFMR	CONTROL	130	TC/T10
1103	01	R24-S002	600V/208V MCC 1B	CONTROL	180	TE/T11
1104	01	R24-S003	600V/208V MCC 1C	CONTROL	180	TE/T11
0024	01	R24-S009	600/208V MCC 1A	INTAKE	111	
0025	01	R24-S010	600/208V MCC 1B	INTAKE	111	
0026	01	R24-S011	600V ESS DIV 1 MCC 1C	REACTOR	130	RH/R13
0027	01	R24-S012	600V ESS DIV 2 MCC 1B	REACTOR	130	RF/R02
0028	01	R24-S018A	600V ESS DIV 1 MCC 1E-A	REACTOR	130	RL/R08
0029	01	R24-S018B	600V ESS DIV 2 MCC 1E-B	REACTOR	130	RL/R08
0032	01	R24-S022	125/250V DC ESS DIV 2 MCC 1B	REACTOR	130	RF/R02
0033	01	R24-S025	600/208V ESS DIV 1 MCC 1A	DIESEL	130	D/03
0034	01	R24-S026	600/208V ESS DIV B MCC 1B	DIESEL	130	E/03
0035	01	R24-S027	600/208V ESS DIV 2 MCC 1C	DIESEL	130	F/03
1105	01	R24-S029	600V MCC 1E	CONTROL	180	TD/T11
1112	01	R24-S031	600V MCC 1G	CONTROL	180	TB/T11
0816	01	R24-S048	600/208V MCC 1D	DIESEL	130	
0036	14	R25-S001	125V DC DIV 1 CAB 1A	CONTROL	130	TE/T11
0037	14	R25-S002	125V DC DIV 2 CAB 1B	CONTROL	130	TE/T11
0038	14	R25-S004	125V DC CAB 1D	DIESEL	130	D/02
0039	14	R25-S005	125V DC CAB 1E	DIESEL	130	E/02
0040	14	R25-S006	125V DC CAB 1F	DIESEL	130	F/02
0519	14	R25-S015	24/48 VDC CABINET 1A	CONTROL	130	TG/T12

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
0520	14	R25-S016	24/48 VDC CABINET 1B	CONTROL	130	TG/T12
0041	14	R25-S029	120/208V AC CAB 1J	DIESEL	130	D/02
0042	14	R25-S030	120/208V AC CAB 1K	DIESEL	130	E/02
0043	14	R25-S031	120/208V AC CAB 1L	DIESEL	130	F/02
0862	14	R25-S035	120/208V AC CABINET	DIESEL	130	
0044	14	R25-S036	120/208V AC ESS CAB 1A	CONTROL	130	TF/T10
0045	14	R25-S037	120/208V AC ESS CAB 1B	CONTROL	130	TD/T10
0047	14	R25-S064	120/208V AC VITAL CAB 1A INSTR BUS 1A	CONTROL	130	TG/T11
0048	14	R25-S065	120/208V AC CAB 1B INSTR BUS 1B	CONTROL	130	TG/T12
0049	14	R25-S105	125V DC CAB 1D ESS DIV I	CONTROL	130	TE/T11
0050	14	R25-S106	125V DC CAB 1E ESS DIV II	CONTROL	130	TB/T10
0051	18	R25-S110	120/208V CAB 1A (R25-S064) FUSE BOX	CONTROL	130	
0052	18	R25-S111	120/208V CAB 1C (R25-S065) FUSE BOX	CONTROL	130	
0053	18	R25-S112	120/208V MCC-1A (R24-S025) FUSE BOX	DIESEL	130	D/02
0054	18	R25-S113	120/208V MCC-1B (R24-S026) FUSE BOX	DIESEL	130	E/02
0055	18	R25-S114	120/208V MCC-1C (R24-S027) FUSE BOX	DIESEL	130	F/02
0838	18	R26-M021	RPS DIST CAB THROWOVER SW	CONTROL	130	TG/T12
0059	18	R26-M031A	125V DC 600A THROWOVER SWITCH 1A	CONTROL	130	TE/T11
0060	18	R26-M031B	125V DC 600A THROWOVER SWITCH 1B	CONTROL	130	TE/T11
0061	18	R26-M031C	125V DC 600A THROWOVER SWITCH 1C	CONTROL	130	TB/T11
0062	18	R26-M031D	125V DC 600A THROWOVER SWITCH 1D	CONTROL	130	TB/T11
0063	18	R26-M032A	125V DC THROWOVER SWITCH 1E	DIESEL	130	D/02
0064	18	R26-M032B	125V DC THROWOVER SWITCH 1F	DIESEL	130	E/02
0065	18	R26-M032C	125V DC THROWOVER SWITCH 1G	DIESEL	130	F/02
0513	18	R26-M041A	24 VDC THROWOVER SW 1A	CONTROL	130	TG/T12
0514	18	R26-M041B	24 VDC THROWOVER SW 1B	CONTROL	130	TG/T12
0515	18	R26-M041C	24 VDC THROWOVER SW 1C	CONTROL	130	TG/T12
0516	18	R26-M041D	24 VDC THROWOVER SW 1D	CONTROL	130	TG/T12
0521	18	R26-M073	DISCONNECT SW FOR C11-F040A	CONTROL	130	TG/T12
0522	18	R26-M074	DISCONNECT SW FOR C11-F040B	CONTROL	138	TG/T12

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
0066	18	R26-M077	600V BREAKER	CONTROL	147	TH/T14
0067	18	R26-M078	600V BREAKER	CONTROL	147	TH/T14
0107	01	R27-S005	LOCAL STARTER FOR E11-F017A	REACTOR	130	RL/R09
0103	01	R27-S035	LOCAL STARTER FOR E41-F006	REACTOR	130	RH/R02
0104	01	R27-S036	LOCAL STARTER FOR E41-F007	REACTOR	112	RH/R02
0105	01	R27-S037	LOCAL STARTER FOR E41-F008	REACTOR	112	RH/R02
0106	01	R27-S066	LOCAL STARTER FOR E41-F002	REACTOR	130	RL/R04
0820	0	R34-S004A	NEUTRAL RESISTOR 1A	DIESEL	130	
0821	0	R34-S004B	NEUTRAL RESISTOR 1B	DIESEL	130	
0822	0	R34-S004C	NEUTRAL RESISTOR 1C	DIESEL	130	
0791	0	R34-S005A	SURGE PROT PANEL FOR P41-C001A	INTAKE	111	
0792	0	R34-S005B	SURGE PROT PANEL FOR P41-C001B	INTAKE	111	
0793	0	R34-S006A	SURGE PROT PANEL FOR E11-C001A	INTAKE	111	
0794	0	R34-S006D	SURGE PROT PANEL FOR E11-C002D	INTAKE	111	
0068	15	R42-S001A	125/250V STATION BATTERY 1A	CONTROL	112	TE/T11
0069	15	R42-S001B	125/250V STATION BATTERY 1B	CONTROL	112	TD/T11
0070	15	R42-S002A	125V DIESEL SYSTEM BATTERY 1A	DIESEL	130	D/02
0071	15	R42-S002B	125V DIESEL SYSTEM BATTERY 1B	DIESEL	130	E/02
0072	15	R42-S002C	125V DIESEL SYSTEM BATTERY 1C	DIESEL	130	F/02
0712	15	R42-S017A	BATTERY 1A 24/48 V	CONTROL	112	TC/T11
0713	15	R42-S017B	BATTERY 1B 24/48 V	CONTROL	112	TC/T12
0073	15	R42-S026	125V BATTERY CHARGER 1A	CONTROL	130	TE/T11
0074	16	R42-S027	125V BATTERY CHARGER 1B	CONTROL	130	TE/T11
0076	16	R42-S029	125V BATTERY CHARGER 1D	CONTROL	130	TB/T11
0077	16	R42-S030	125V BATTERY CHARGER 1E	CONTROL	130	TB/T11
0079	16	R42-S032A	125V BATTERY CHARGER 1G	DIESEL	130	D/02
0080	16	R42-S032B	125V BATTERY CHARGER 1H	DIESEL	130	E/02
0081	16	R42-S032C	125V BATTERY CHARGER 1J	DIESEL	130	F/02
0507	16	R42-S051	BATTERY CHARGER 1A	CONTROL	130	TG/T12
0509	16	R42-S052	BATTERY CHARGER 1B	CONTROL	130	TG/T12

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
0510	16	R42-S053	BATTERY CHARGER 1C	CONTROL	130	TG/T12
0512	16	R42-S054	BATTERY CHARGER 1D	CONTROL	130	TG/T12
0629	21	R43-A001A	FUEL DAY TANK 1A	DIESEL	130	D/01
0660	21	R43-A001B	FUEL DAY TANK 1B	DIESEL	130	E/01
0691	21	R43-A001C	FUEL DAY TANK 1C	DIESEL	130	F/01
0630	21	R43-A002A	FUEL STORAGE TANK 1A	YARD		
0661	21	R43-A002B	FUEL STORAGE TANK 1B	YARD		
0692	21	R43-A002C	FUEL STORAGE TANK 1C	YARD		
0627	21	R43-A003A	AIR RECEIVER	DIESEL	130	D/01
0658	21	R43-A003B	AIR RECEIVER	DIESEL	130	E/01
0689	21	R43-A003C	AIR RECEIVER	DIESEL	130	F/01
0628	21	R43-A007A	AIR RECEIVER	DIESEL	130	D/01
0659	21	R43-A007B	AIR RECEIVER	DIESEL	130	E/01
0690	21	R43-A007C	AIR RECEIVER	DIESEL	130	F/01
0622	20	R43-P001A	DSL GEN 1A CONT PNL	DIESEL	130	D/02
0653	20	R43-P001B	DSL GEN 1B CONT PNL	DIESEL	130	E/02
0684	20	R43-P001C	DSL GEN 1C CONT PNL	DIESEL	130	F/02
0099	17	R43-S001A	DIESEL GENERATOR 1A	DIESEL	130	C/02
0100	17	R43-S001B	DIESEL GENERATOR 1B	DIESEL	130	D/02
0101	17	R43-S001C	DIESEL GENERATOR 1C	DIESEL	130	E/02
0085	16	R44-S002	DC/AC INVERTER FOR MCC 1E-A	CONTROL	147	TI/T14
0086	16	R44-S003	DC/AC INVERTER FOR MCC 1E-B	CONTROL	147	TH/T14
0088	04	S11-S009	4160/600V STA SERV XFMR 1F1	DIESEL	130	E/03
0819	04	S11-S012	STA SERV XFMR 1F2 4160/600V	DIESEL	130	
0424	10	T41-B002A	RHR/CS PUMP ROOM COOLER	REACTOR	087	RL/R11
0437	10	T41-B003B	RHR/CS PUMP ROOM COOLER	REACTOR	087	RL/R03
0836	10	T41-B005B	HPCI PUMP ROOM COOLER	REACTOR	087	RL/R03
0456	21	T48-A001	UNIT 1 NITROGEN STORAGE TANK	YARD		
0539	18	T48-N010A	TORUS WATER LEVEL TRANS	REACTOR	087	RF/R13
0545	18	T48-N010B	TORUS WATER LEVEL TRANS	REACTOR	087	RF/R02

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
0517	18	TB1-139	TERMINAL BOX	CONTROL	130	TG/T12
0518	18	TB1-140	TERMINAL BOX	CONTROL	130	TG/T12
0089	18	TB1-211	125V DC BATTERY 1A FUSE BOX	DIESEL	130	D/02
0090	18	TB1-212	125V DC BATTERY 1B FUSE BOX	DIESEL	130	E/02
0091	18	TB1-213	125V DC BATTERY 1C FUSE BOX	DIESEL	130	F/02
0716	18	TB1-229-7	TERMINAL BOX	DIESEL	130	D/02
0717	18	TB1-230-8	TERMINAL BOX	DIESEL	130	E/02
0718	18	TB1-231-8	TERMINAL BOX	DIESEL	130	F/02
0827	09	X41-C002A	DG ROOM 1A FAN	DIESEL	150	
0828	09	X41-C002C	DG ROOM 1B FAN	DIESEL	150	
0829	09	X41-C002E	DG ROOM 1C FAN	DIESEL	150	
0860	0	X41-C005A	DG ROOM 1A LOUVER	DIESEL	130	
0831	0	X41-C005B	DG ROOM 1B LOUVER	DIESEL	130	
0832	0	X41-C005C	DG ROOM 1C LOUVER	DIESEL	130	
0824	09	X41-C006A	DG SWITCHGEAR ROOM 1E FAN	DIESEL	150	
0825	09	X41-C006C	DG SWITCHGEAR ROOM 1F FAN	DIESEL	150	
0826	09	X41-C006E	DG SWITCHGEAR ROOM 1G FAN	DIESEL	150	
0833	0	X41-C007A	DG SWITCHGEAR ROOM 1E LOUVER	DIESEL	130	
0834	0	X41-C007B	DG SWITCHGEAR ROOM 1F LOUVER	DIESEL	130	
0835	0	X41-C007C	DG SWITCHGEAR ROOM 1G LOUVER	DIESEL	130	
1089	09	X41-C008A	BATTERY ROOM 1A FAN	DIESEL	150	
1085	09	X41-C008C	BATTERY ROOM 1B FAN	DIESEL	150	
1086	09	X41-C008E	BATTERY ROOM 1C FAN	DIESEL	150	
0985	09	X41-C009A	INTAKE STRUCTURE VENT FAN 1A	INTAKE	150	
0977	09	X41-C009B	INTAKE STRUCTURE VENT FAN 1B	INTAKE	150	
0978	09	X41-C009C	INTAKE STRUCTURE VENT FAN 1C	INTAKE	150	
0882	0	X41-C017A	DG ROOM 1A ROLL-UP FIRE DOOR	DIESEL	130	
0883	0	X41-C017B	DG ROOM 1B ROLL-UP FIRE DOOR	DIESEL	130	
0884	0	X41-C017C	DG ROOM 1C ROLL-UP FIRE DOOR	DIESEL	130	
1090	0	X41-C027A	BATTERY ROOM 1A LOUVER	DIESEL	130	C/01

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
1087	0	X41-C027B	BATTERY ROOM 1B LOUVER	DIESEL	130	D/01
1088	0	X41-C027C	BATTERY ROOM 1C LOUVER	DIESEL	130	E/01
0987	18	X41-N002A	INTK STRUC VENT FAN 1A THERMOSTAT	INTAKE	111	PUMP RM
0979	18	X41-N002B	INTK STRUC VENT FAN 1B THERMOSTAT	INTAKE	111	PUMP RM
0980	18	X41-N002C	INTK STRUC VENT FAN 1C THERMOSTAT	INTAKE	111	PUMP RM
0885	18	X41-N004A	D/G RM 1A FAN THERMOSTAT	DIESEL	130	C/01
0886	18	X41-N004B	D/G RM 1B FAN THERMOSTAT	DIESEL	130	D/01
0887	18	X41-N004C	D/G RM 1C FAN THERMOSTAT	DIESEL	130	E/01
0888	18	X41-N005A	D/G RM 1A FAN THERMOSTAT	DIESEL	130	D/02
0889	18	X41-N005B	D/G RM 1B FAN THERMOSTAT	DIESEL	130	E/01
0890	18	X41-N005C	D/G RM 1C FAN THERMOSTAT	DIESEL	130	F/02
1109	18	X41-N006A	SWGR RM 1E FAN THERMOSTAT	DIESEL	130	C/03
1110	18	X41-N006B	SWGR RM 1F FAN THERMOSTAT	DIESEL	130	D/03
1111	18	X41-N006C	SWGR RM 1G FAN THERMOSTAT	DIESEL	130	E/03
1106	20	X43-P006A	PNEU-ELECTRO RELAY CAB	DIESEL	130	C/01
1107	20	X43-P006B	PNEU-ELECTRO RELAY CAB	DIESEL	130	D/01
1108	20	X43-P006C	PNEU-ELECTRO RELAY CAB	DIESEL	130	E/01
0625	06	Y52-C001A	DSL 1A FUEL OIL PUMP 1A1	YARD		
0656	06	Y52-C101B	DSL 1B FUEL OIL PUMP 1B2	YARD		
0688	06	Y52-C101C	DSL 1C FUEL OIL PUMP 1C2	YARD		
0898	10	Z41-B003A	CONTROL ROOM AIR HANDLING UNIT	CONTROL	180	TD/T12
0939	10	Z41-B003B	CONTROL ROOM AIR HANDLING UNIT	CONTROL	180	TE/T12
0901	10	Z41-B003C	CONTROL ROOM AIR HANDLING UNIT	CONTROL	180	TF/T12
0899	11	Z41-B008A	B003A CONDENSING UNIT	CONTROL	180	TB/T11
0940	11	Z41-B008B	B003B CONDENSING UNIT	CONTROL	180	TB/T11
0937	11	Z41-B008C	B003C CONDENSING UNIT	CONTROL	180	TB/T12
0903	09	Z41-C012A	D004A BOOSTER FAN	CONTROL	180	TF/T13
0941	09	Z41-C012B	D004B BOOSTER FAN	CONTROL	180	TG/T12
1097	09	Z41-C014	BATT ROOM EMERGENCY EXH	CONTROL	112	TD/T12
1091	09	Z41-C015	BATT ROOM EMERGENCY EXH	CONTROL	112	TD/T11

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
0900	0	Z41-D004A	CONTROL ROOM FILTER TRAIN	CONTROL	180	TG/T13
0938	0	Z41-D004B	CONTROL ROOM FILTER TRAIN	CONTROL	180	TG/T13
0904	07	Z41-F007A	AIR OPERATED DAMPER B003A IN	CONTROL	180	TD/T12
0905	07	Z41-F007C	AIR OPERATED DAMPER B003C IN	CONTROL	180	TF/T12
0943	07	Z41-F008B	AIR OPERATED DAMPER B003B IN	CONTROL	180	TE/T12
0944	07	Z41-F008C	AIR OPERATED DAMPER B003C IN	CONTROL	180	TF/T12
0945	07	Z41-F009B	AIR OPERATED DAMPER B003B IN	CONTROL	180	TE/T12
0947	07	Z41-F010A	AIR OPERATED DAMPER B003B IN	CONTROL	180	TE/T04
0909	07	Z41-F011	AIR OPER DMPR D004A/B BYPASS	CONTROL	180	TG/T12
0949	07	Z41-F012	AIR OPER DMPR D004A/B BYPASS	CONTROL	180	TF/T12
0910	07	Z41-F013A	AIR OPERATED DAMPER D004A IN	CONTROL	180	TH/T13
0957	07	Z41-F013B	AIR OPERATED DAMPER D004B IN	CONTROL	180	TH/T12
0912	07	Z41-F014A	AIR OPERATED DAMPER D004A IN	CONTROL	180	TH/T12
0953	07	Z41-F014B	AIR OPERATED DAMPER D004B IN	CONTROL	180	TH/T12
0914	07	Z41-F015	AIR OPERATED DAMPER D003 BYPASS	CONTROL	180	TH/T12
0956	07	Z41-F018A	AIR OPERATED DAMPER C011A IN	CONTROL	180	TE/T12
0951	07	Z41-F018B	AIR OPERATED DAMPER C011B IN	CONTROL	180	TE/T12
0918	07	Z41-F019	AIR OPERATED DAMPER RESTROOM	CONTROL	180	TH/T13
0919	07	Z41-F020	AIR OPERATED DAMPER RESTROOM	CONTROL	180	TH/T13
0960	07	Z41-F028A	AIR OPERATED DAMPER B010B OUT	CONTROL	180	TE/T13
1099	0	Z41-FD-F004	FIRE DMPR STN BATTERY 1A	CONTROL	112	TE/T11
1093	0	Z41-FD-F005	FIRE DMPR STN BATTERY 1B	CONTROL	112	TD/T11
1092	0	Z41-FD-F006	FIRE DMPR STN BATTERY 1B	CONTROL	112	TC/T11
1098	0	Z41-FD-F020	FIRE DMPR STN BATTERY 1A	CONTROL	112	TF/T11
0925	18	Z41-N003A	B003A DISCHARGE FS	CONTROL	180	TE/T13
0963	18	Z41-N003B	B003B DISCHARGE FS	CONTROL	180	TE/T13
0966	18	Z41-N003C	B003C DISCHARGE FS	CONTROL	180	TF/T13
0923	18	Z41-N005A	C012A DISCHARGE FS	CONTROL	180	TF/T13
0964	18	Z41-N005B	C012B DISCHARGE FS	CONTROL	180	TF/T12
0924	18	Z41-N015A	CONTROL RM OUTSIDE AIR INLET RE	CONTROL	180	TH/T13

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
0965	18	Z41-N015B	CONTROL RM OUTSIDE AIR INLET RE	CONTROL	180	TH/T13
0926	18	Z41-N600A	B003A COMPRESSOR TIS	CONTROL	180	TD/T12
0968	18	Z41-N600B	B003B COMPRESSOR TIS	CONTROL	180	TE/T12
0967	18	Z41-N600C	B003C COMPRESSOR TIS	CONTROL	180	TF/T12

APPENDIX B

SEISMIC REVIEW SAFE SHUTDOWN EQUIPMENT LIST
PLANT HATCH UNIT 2

The Seismic Review Safe Shutdown Equipment List (SSEL) is a subset of the composite SSEL included in Appendix E of this report. The Seismic Review SSEL contains all mechanical and electrical equipment, tanks, and heat exchangers for which a seismic evaluation was performed.

The format of the Seismic Review SSEL is as given by the Safe Shutdown Equipment Manager computer program supplied by the Seismic Qualification Utility Group (SQUG). The Seismic Review SSEL includes the following information, where applicable and available, for each component. Components are sorted by mark number.

Column Number	Description
1	Unique line number
2	Equipment class (see table 1 of this appendix)
3	Equipment identification number
4	Equipment description
5	Building or area in which the equipment is located
6	Floor elevation from which the equipment can be seen
7	Building grid row/column (or panel number) indicating equipment location

APPENDIX B

TABLE 1
EQUIPMENT CLASS DESIGNATIONS

Equipment Class (Column 2)	Description
01	Motor control centers
02	Low-voltage switchgear
03	Medium-voltage switchgear
04	Transformers
05	Horizontal pumps
06	Vertical pumps
07	Fluid-operated valves
08A	Motor-operated valves
08B	Solenoid-operated valves
09	Fans
10	Air handlers
11	Chillers
12	Air compressors
13	Motor generators
14	Distribution panels
15	Batteries on racks
16	Battery chargers and inverters
17	Engine generators
18	Instruments on racks
19	Temperature sensors
20	Instrumentation and control panels and cabinets
21	Tanks and heat exchangers

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
000278	0	2B21-A003A	SRV AIR ACCUMULATOR	DRYWELL	148	AZ220
000279	0	2B21-A003B	SRV AIR ACCUMULATOR	DRYWELL	148	AZ220
000265	0	2B21-A003D	SRV AIR ACCUMULATOR	DRYWELL	148	AZ133
000280	0	2B21-A003F	SRV AIR ACCUMULATOR	DRYWELL	148	AZ274
000266	0	2B21-A003G	SRV AIR ACCUMULATOR	DRYWELL	148	AZ080
000267	0	2B21-A003H	SRV AIR ACCUMULATOR	DRYWELL	148	AZ118
000281	0	2B21-A003K	SRV AIR ACCUMULATOR	DRYWELL	148	AZ251
000268	0	2B21-A003M	SRV AIR ACCUMULATOR	DRYWELL	148	AZ078
000008	07	2B21-F013A	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ235
000135	07	2B21-F013B	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ235
000006	07	2B21-F013D	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ135
000136	07	2B21-F013F	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ260
000007	07	2B21-F013G	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ095
000137	07	2B21-F013H	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ135
000009	07	2B21-F013K	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ260
000138	07	2B21-F013M	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ090
000129	07	2B21-F022A	INBOARD MSIV	DRYWELL	130	AZ190
000130	07	2B21-F022B	INBOARD MSIV	DRYWELL	130	AZ220
000131	07	2B21-F022C	INBOARD MSIV	DRYWELL	130	AZ160
000132	07	2B21-F022D	INBOARD MSIV	DRYWELL	130	AZ175
000001	07	2B21-F028A	OUTBOARD MSIV	REACTOR	130	RB/R19
000002	07	2B21-F028B	OUTBOARD MSIV	REACTOR	130	RB/R18
000003	07	2B21-F028C	OUTBOARD MSIV	REACTOR	130	RB/R20
000004	07	2B21-F028D	OUTBOARD MSIV	REACTOR	130	RB/R19
000767	18	2B21-N093B	RPV LEVEL 8 LT	REACTOR	158	RH/R17
000179	08A	2B31-F023A	RECIRC PUMP SUCTION ISOLATION	DRYWELL	087	AZ340
000740	08B	2C11-D001-117	PILOT SCRAM SOLENOID	REACTOR	130	HCU
000741	08B	2C11-D001-118	PILOT SCRAM SOLENOID	REACTOR	130	HCU
000746	0	2C11-D001-125	SCRAM ACCUMULATOR	REACTOR	130	HCU
000747	07	2C11-D001-126	SCRAM INLET VALVE	REACTOR	130	HCU

Report Date/Time: 06-12-95 / 16:02:59
Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
000748	07	2C11-D001-127	SCRAM INLET VALVE	REACTOR	130	HCU
000226	08B	2C11-F009	PILOT AIR HEADER SUPPLY	REACTOR	130	RA/R21
000141	07	2C11-F010A	SCRAM DISCH VOL ISOLATION	REACTOR	158	
000142	07	2C11-F010B	SCRAM DISCH VOL ISOLATION	REACTOR	158	
000143	07	2C11-F011	SCRAM DISCH VOL DRAIN	REACTOR	087	RB/R17
000145	07	2C11-F035A	SCRAM DISCH VOL ISOLATION	REACTOR	130	RF/R15
000146	07	2C11-F035B	SCRAM DISCH VOL ISOLATION	REACTOR	130	RF/R24
000147	07	2C11-F037	SCRAM DISCH VOL DRAIN	REACTOR	087	RB/R17
000227	08B	2C11-F040	PILOT AIR HEADER SUPPLY	REACTOR	130	RA/R17
000148	08B	2C11-F110A	BACKUP SCRAM VALVE	REACTOR	130	RA/R21
000144	08B	2C11-F110B	BACKUP SCRAM VALVE	REACTOR	130	RA/R21
000768	18	2C71-N050C	DRYWELL PRESSURE PT	REACTOR	158	RE/R16
000769	18	2C71-N050D	DRYWELL PRESSURE PT	REACTOR	158	RE/R21
000725	20	2C82-P001	REMOTE SHUTDOWN PANEL	REACTOR	130	RA/R16
000172	21	2E11-B001A	RHR HEAT EXCHANGER A	REACTOR	087	RL/R14
000095	21	2E11-B001B	RHR HEAT EXCHANGER B	REACTOR	087	RL/R24
000208	06	2E11-C001A	RHRSW PUMP 2A	INTAKE	111	
000096	06	2E11-C001D	RHRSW PUMP 2D	INTAKE	111	
000160	06	2E11-C002A	RHR PUMP 2A	REACTOR	087	RL/R14
000097	06	2E11-C002B	RHR PUMP 2B	REACTOR	087	RL/R24
000161	08A	2E11-F007A	RHR PUMP MIN FLOW BYPASS	REACTOR	087	RL/R14
000101	08A	2E11-F007B	RHR PUMP 2D MIN FLOW BYPASS VL	REACTOR	087	RL/R24
000171	08A	2E11-F015A	INBOARD INJECTION	REACTOR	130	RJ/R18
000340	08A	2E11-F016A	CONTAINMENT SPRAY OUTBOARD	REACTOR	130	RJ/R21
000376	08A	2E11-F016B	CONT SPRAY DISCHARGE VALVE	REACTOR	158	RF/R23
000170	08A	2E11-F017A	RHR LPCI DISCHARGE VALVE	REACTOR	130	RJ/R18
000367	08A	2E11-F021A	CONTAINMENT SPRAY INBOARD	REACTOR	130	RJ/R21
000406	08A	2E11-F021B	CONTAINMENT SPRAY INBOARD	REACTOR	158	RH/R23
000105	08A	2E11-F024B	RHR TEST LINE VALVE	REACTOR	087	RF/R24
000365	08A	2E11-F027A	TORUS SPRAY INBOARD ISOLATION	REACTOR	087	RF/R14

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
000379	08A	2E11-F027B	SUPP POOL SPRAY VALVE	REACTOR	087	RF/R24
000339	08A	2E11-F028A	TORUS SPRAY OUTBOARD ISOLATION	REACTOR	087	RH/R14
000108	08A	2E11-F028B	RHR INLET TO SUPP POOL	REACTOR	087	RH/R24
000165	08A	2E11-F048A	RHR HX BYPASS	REACTOR	087	RL/R14
000110	08A	2E11-F048B	RHR HX B BYPASS VALVE	REACTOR	087	RL/R24
000210	08A	2E11-F068A	RHRSW HX FLOW CONTROL	REACTOR	087	RL/R14
000113	08A	2E11-F068B	RHR HX B TUBE TO SHELL OUTLET	REACTOR	087	RH/R24
000216	18	2E11-N017A	RHRSW HX INLET PS	REACTOR	087	RH/R14
000122	18	2E11-N017B	RHR HX B INLET PRESS SW	REACTOR	087	RH/R24
000217	18	2E11-N017C	RHRSW HX INLET PS	REACTOR	087	RH/R14
000123	18	2E11-N017D	RHR HX B INLET PRESS SW	REACTOR	087	RH/R24
000770	18	2E11-N094A	DRYWELL PRESSURE PT	REACTOR	158	RE/R16
000771	18	2E11-N094B	DRYWELL PRESSURE PT	REACTOR	158	RH/R21
000772	18	2E11-N094C	DRYWELL PRESSURE PT	REACTOR	158	RE/R16
000773	18	2E11-N094D	DRYWELL PRESSURE PT	REACTOR	158	RH/R21
000083	06	2E21-C001B	CORE SPRAY PUMP 2B	REACTOR	087	RL/R24
000086	08A	2E21-F005B	CORE SPRAY TO RVP ISOL VLV	REACTOR	158	RF/R21
000089	08A	2E21-F031B	CORE SPRAY MIN FLOW BYPASS	REACTOR	087	RL/R24
000015	05	2E41-C001	HPCI PUMP	REACTOR	087	RL/R25
000016	05	2E41-C002	HPCI TURBINE	REACTOR	087	RG/R24
000018	08A	2E41-F001	HPCI TURBINE STEAM SUPPLY VLV	REACTOR	087	RG/R25
000021	08A	2E41-F004	HPCI PUMP SUCTION FROM CST	REACTOR	087	RL/R25
000022	08A	2E41-F006	HPCI PUMP INBOARD DISCH. VALVE	REACTOR	087	RB/R19
000025	08A	2E41-F012	HPCI MINIMUM FLOW BYPASS VALVE	REACTOR	087	RG/R24
000026	08A	2E41-F041	HPCI PUMP SUCTION - SUPP POOL	REACTOR	087	RL/R25
000027	08A	2E41-F042	HPCI PUMP SUCT FROM SUPP POOL	REACTOR	087	RG/R24
000029	08A	2E41-F059	HPCI BAR COND COOLING WTR VLV	REACTOR	087	RG/R24
000057	18	2E41-N062B	SUPPRESSION POOL LEVEL TRANS.	REACTOR	087	RG/R24
000058	18	2E41-N062D	SUPPRESSION POOL LEVEL TRANS.	REACTOR	087	RG/R24
000539	08B	2G11-F003	DRYWELL FL DR PMP ISOL VALVE	REACTOR	087	RE/R19

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
000540	08B	2G11-F004	DRYWELL FL DR PMP ISOL VALVE	REACTOR	087	RE/R19
000541	08B	2G11-F019	DRYWELL EQ DR PMP ISOL VALVE	REACTOR	087	RF/R21
000542	08B	2G11-F020	DRYWELL EQ DR PMP ISOL VALVE	REACTOR	087	RF/R21
000543	08A	2G31-F001	RWCU INBOARD ISOL GATE VALVE	DRYWELL	165	AZ020
000544	08A	2G31-F004	RWCU OUTBOARD ISOL GATE VALVE	REACTOR	158	RF/R19
000632	20	2H11-P601	REAC CNTMT COOL ISO BN BD	CONTROL	164	TE/T12
000633	20	2H11-P602	REAC WTR CLNUP & RECIRC	CONTROL	164	TC/T12
000634	20	2H11-P603	REAC CONTROL BN BD	CONTROL	164	TC/T12
000635	20	2H11-P604	PROCESS RADMON VER BD	CONTROL	164	TE/T13
000636	20	2H11-P605A	CNTMT ATM OIL VERT BD	CONTROL	164	
000637	20	2H11-P605B	CNTMT ATM OIL VERT BD	CONTROL	164	
000638	20	2H11-P606	STARTUP NEUT MON PNL	CONTROL	164	TC/T13
000639	20	2H11-P608	RWR RNGE NEUT MON PNL	CONTROL	164	TC/T13
000640	20	2H11-P609	CH A PRI ISOL & RPS VB	CONTROL	164	TE/T13
000641	20	2H11-P611	CH B PRI ISOL & RPS VB	CONTROL	164	TE/T13
000642	20	2H11-P612	FW AND RECIRC INST PNL	CONTROL	164	TC/T13
000643	20	2H11-P613	PROCESS INST VERT BD	CONTROL	164	TC/T13
000644	20	2H11-P614	NSSS TEMP DET VERT BD	CONTROL	164	TE/T13
000645	20	2H11-P617	CHAN A RHR RELAY VERT BD	CONTROL	164	TC/T12
000646	20	2H11-P618	CHAN B RHR RELAY VERT BD	CONTROL	164	TC/T12
000647	20	2H11-P620	HPCI RELAY VERT BD	CONTROL	164	TC/T12
000648	20	2H11-P622	INBD ISO VLV VERT PNL	CONTROL	164	TC/T13
000649	20	2H11-P623	OUTBD ISO VLV VERT PNL	CONTROL	164	TC/T12
000650	20	2H11-P626	CORE SPRAY CTRL PNL DIV 1	CONTROL	164	TC/T12
000651	20	2H11-P627	CORE SPRAY CTRL PNL DIV 2	CONTROL	164	TC/T12
000652	20	2H11-P628	ADS RELAY PANEL	CONTROL	164	TC/T12
000653	20	2H11-P650	TURB FDWTR 7 COND PNL	CONTROL	164	TC/T12
000654	20	2H11-P652	DSL GEN & EMER STA PNL	CONTROL	164	TC/T11
000655	20	2H11-P654	GAS TREAT VENT VERT BD	CONTROL	164	TC/T12
000656	20	2H11-P656	TURB AUX SYSTEM VERT PNL	CONTROL	164	TC/T11

Report Date/Time: 06-12-95 / 16:02:59
Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
000657	20	2H11-P657	VNT DRYWELL INERT VERT BD	CONTROL	164	TC/T12
000658	20	2H11-P664	MSIV LEAK CTRL SYS PNL	CONTROL	164	TA/T13
000659	20	2H11-P674	START UP XFMR 2C PANEL	CONTROL	164	TA/T12
000660	20	2H11-P675	START UP XFMR 2D PANEL	CONTROL	164	TA/T12
000661	20	2H11-P679	STA SERV XFMR RLY PANEL	CONTROL	164	TA/T12
000662	20	2H11-P691	ANALOG SIG CONV PANEL	CONTROL	164	TB/T11
000663	20	2H11-P700	ANALOG VNT LEAK DETECT PNL	CONTROL	164	TE/T13
000664	20	2H11-P921	RPS TRIP UNIT CABINET	CONTROL	164	TI/T12
000665	20	2H11-P922	RPS TRIP UNIT CABINET	CONTROL	164	TI/T12
000666	20	2H11-P923	RPS TRIP UNIT CABINET	CONTROL	164	TI/T12
000667	20	2H11-P924	RPS TRIP UNIT CABINET	CONTROL	164	TI/T12
000668	20	2H11-P925	ECCS TRIP UNIT CABINET	CONTROL	164	TI/T12
000669	20	2H11-P926	ECCS TRIP UNIT CABINET	CONTROL	164	TI/T12
000670	20	2H11-P927	ECCS TRIP UNIT CABINET	CONTROL	164	TI/T12
000671	20	2H11-P928	ECCS TRIP UNIT CABINET	CONTROL	164	TI/T12
000672	18	2H21-P002	REACTOR WATER CLEANUP PNL	REACTOR	158	RF/R24
000673	18	2H21-P016	MAIN STEAM FLOW INST RACK	REACTOR	087	RL/R24
000674	18	2H21-P018	RHR INST RACK CHANNEL A	REACTOR	087	RL/R14
000675	18	2H21-P036	HPCI SYS LOCAL RACK	REACTOR	130	RJ/R23
000676	20	2H21-P052	HPCI TEST VLV CONTROL PNL	REACTOR	087	RH/R24
000677	20	2H21-P173	SHUTDOWN INSTRUMENT PANEL	REACTOR	130	RA/R16
000460	20	2H21-P198	AMMETER SHUNT BATTERY CHG	DIESEL	130	A02
000500	20	2H21-P199	AMMETER SHUNT BATTERY CHG	DIESEL	130	C02
000678	20	2H21-P200	DIESEL GEN 2A RELAY PANEL	DIESEL	130	A02
000679	20	2H21-P202	DIESFL GEN 2C RELAY PANEL	DIESEL	130	C02
000680	18	2H21-P220	TURBINE BUILDING INST RACK	TURBINE	130	TH/T20
000681	18	2H21-P225	TURBINE BUILDING INST RACK	TURBINE	130	TH/T20
000682	20	2H21-P230	RELAY PANEL 2A-D/G 2A	DIESEL	130	A02
000683	20	2H21-P231	RELAY PANEL 2B-D/G 2B	DIESEL	130	B02
000684	20	2H21-P232	RELAY PANEL 2C-D/G 2C	DIESEL	130	B02

Report Date/Time: 06-12-95 / 16:02:59
Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
000685	20	2H21-P245	600 VOLT BUS 2C CNT PNL	CONTROL	130	TE/T13
000686	20	2H21-P246	600 VOLT BUS 2D CNT PNL	CONTROL	130	TC/T13
000687	20	2H21-P248	250 VOLT DC SWGR 2A CNT PNL	CONTROL	130	TE/T13
000688	20	2H21-P249	250 VOLT DC SWGR 2B CNT PNL	CONTROL	130	TC/T13
000689	20	2H21-P255	DG FUEL PMP & MOV CONT PNL	DIESEL	130	A02
000690	20	2H21-P256	DG FUEL PMP & MOV CONT PNL	DIESEL	130	C02
000691	20	2H21-P257	D/G 2A HT/VEN CONT PNL	DIESEL	130	A01
000692	20	2H21-P259	D/G 2C HT/VEN CONT PNL	DIESEL	130	B01
000693	20	2H21-P260	SWGR 2E RM HT/VEN CONT PNL	DIESEL	130	A02
000694	20	2H21-P262	SWGR 2G RM HT/VEN CONT PNL	DIESEL	130	B02
000695	20	2H21-P266	MOV CONTROL PNL 2A DIV 1	INTAKE	111	
000696	20	2H21-P267	MOV CONTROL PNL 2B DIV 2	INTAKE	111	
000466	20	2H21-P285	SHUNT BOX A	CONTROL	112	TE/T13
000467	20	2H21-P286	SHUNT BOX B	CONTROL	112	TF/T14
000468	20	2H21-P287	SHUNT BOX C	CONTROL	112	TF/T13
000503	20	2H21-P288	BATTERY SHUNT BOX D	CONTROL	112	TC/T13
000504	20	2H21-P289	BATTERY SHUNT BOX E	CONTROL	112	TC/T14
000505	20	2H21-P290	BATTERY SHUNT BOX F	CONTROL	112	TE/T13
000461	20	2H21-P291	BATTERY 2A FUSE BOX	DIESEL	130	B02
000501	20	2H21-P293	BATTERY 2C FUSE BOX	DIESEL	130	C02
000697	20	2H21-P303	DG 2A LOADING TIMER PANEL	DIESEL	130	A03
000698	20	2H21-P305	DG 2C LOADING TIMER PANEL	DIESEL	130	C03
000699	18	2H21-P401	CS INSTRUMENT RACK	REACTOR	087	RL/R14
000700	18	2H21-P404A	RPV LVL/PRESS INSTR RACK	REACTOR	158	RG/R17
000701	18	2H21-P404B	RPV LVL/PRESS INSTR RACK	REACTOR	158	RG/R17
000702	18	2H21-P404C	RPV LVL/PRESS INSTR RACK	REACTOR	158	RG/R17
000703	18	2H21-P404D	RPV LVL/PRESS INSTR RACK	REACTOR	158	RG/R17
000704	18	2H21-P404E	RPV LVL/PRESS INSTR RACK	REACTOR	158	RG/R17
000705	18	2H21-P405A	RPV LVL/PRESS INSTR RACK	REACTOR	158	RH/R23
000706	18	2H21-P405B	RPV LVL/PRESS INSTR RACK	REACTOR	158	RH/R23

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
000707	18	2H21-P405C	RPV LVL/PRESS INSTR RACK	REACTOR	158	RH/R23
000708	18	2H21-P405D	RPV LVL/PRESS INSTR RACK	REACTOR	158	RH/R23
000709	18	2H21-P405E	RPV LVL/PRESS INSTR RACK	REACTOR	158	RH/R23
000710	18	2H21-P409	JET PUMP INSTR RACK	REACTOR	130	RF/R15
000711	18	2H21-P410	JET PUMP INSTR RACK	REACTOR	130	RF/R22
000712	18	2H21-P414A	HPCI INSTR RACK	REACTOR	087	RG/R24
000713	18	2H21-P414B	HPCI INSTR RACK	REACTOR	087	RG/R24
000714	18	2H21-P415A	MAIN STM FLOW INSTRUMENT RACK	REACTOR	130	RF/R15
000715	18	2H21-P415B	MAIN STM FLOW INSTRUMENT RACK	REACTOR	130	RF/R15
000716	18	2H21-P418A	RHR INSTRUMENT RACK	REACTOR	087	RL/R24
000717	18	2H21-P418B	RHR INSTRUMENT RACK	REACTOR	087	RL/R24
000718	18	2H21-P419	CS INSTRUMENT RACK	REACTOR	087	RL/R24
000719	18	2H21-P421A	RHR INSTRUMENT RACK	REACTOR	087	RL/R24
000720	18	2H21-P421B	RHR INSTRUMENT RACK	REACTOR	087	RL/R24
000721	18	2H21-P425A	RHR INSTRUMENT RACK	REACTOR	130	RF/R22
000722	18	2H21-P425B	RHR INSTRUMENT RACK	REACTOR	130	RF/R22
000723	18	2H21-P434	HPCI INSTRUMENT RACK	REACTOR	087	RG/R24
000756	0	2L48-D134	D/G RM 2A FIRE DAMPER	DIESEL	130	2A
000757	0	2L48-D137	D/G RM 2C FIRE DAMPER	DIESEL	130	2C
000255	08A	2N71-F012	CIRC WATER MAKEUP	YARD		
000243	06	2P41-C001A	PLANT SERVICE WATER PUMP A	INTAKE	111	
000228	06	2P41-C001B	PLANT SERVICE WATER PUMP B	INTAKE	111	
000237	07	2P41-F035B	T41B005B CONTROL VALVE	REACTOR	130	RH/R25
000236	07	2P41-F036B	T41B002B CONTROL VALVE	REACTOR	098	RL/R24
000250	07	2P41-F037A	2E11C002A CONTROL VALVE	REACTOR	096	RL/R14
000235	07	2P41-F037B	E11C002B CONTROL VALVE	REACTOR	098	RL/R24
000251	07	2P41-F039A	2T41B003A CONTROL VALVE	REACTOR	120	RL/R15
000248	08A	2P41-F316A	TURBINE BUILDING ISOLATION	YARD		
000229	08A	2P41-F316B	TURBINE BUILDING ISOLATION	YARD		
000246	07	2P41-F339A	DIESEL GENERATOR 2A OUTLET	DIESEL	130	A01

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
000232	07	2P41-F339B	DIESEL GENERATOR 2C OUTLET	DIESEL	130	B01
000252	18	2P41-N303A	PSW DISCHARGE PT	INTAKE	088	
000240	18	2P41-N303B	PSW DISCHARGE PT	INTAKE	088	
000270	0	2P70-A002A	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RB/R23
000271	0	2P70-A002B	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RB/R23
000272	0	2P70-A002C	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RB/R23
000285	07	2P70-F001A	NITROGEN BACKUP SUPPLY	REACTOR	158	RB/R17
000277	0	2P70-F084	EMERGENCY NITROGEN ISOLATION	REACTOR	130	RB/R21
000273	0	2P70-F138A	EMERGENCY NITROGEN CONTROL	REACTOR	130	RB/R23
000274	0	2P70-F138B	EMERGENCY NITROGEN CONTROL	REACTOR	130	RB/R23
000275	0	2P70-F138C	EMERGENCY NITROGEN CONTROL	REACTOR	130	RB/R23
000276	0	2P70-F141	EMERGENCY NITROGEN CONTROL	REACTOR	130	RB/R23
000455	04	2R11-S004	600-120/208 V AC XFMR	DIESEL	130	B03
000493	04	2R11-S006	LTG & MISC POWER XFMR	DIESEL	130	C03
000454	04	2R11-S041	CONT BLDG ESS XFMR 2B	CONTROL	130	TD/T13
000492	04	2R11-S042	CONT BLDG ESS XFMR 2C	CONTROL	130	TC/T13
000463	18	2R20M-P001	FUSE BOX	DIESEL	130	A02
000494	18	2R20M-P002	FUSE BOX	DIESEL	130	C02
000456	18	2R20N-P001	FUSE BOX	CONTROL	130	
000495	18	2R20N-P002	FUSE BOX	CONTROL	130	
000431	03	2R22-S005	4160V STA SVC SWGR 2E	DIESEL	130	A02
000469	03	2R22-S007	4160V STA SVC SWGR 2G	DIESEL	130	C02
000432	02	2R22-S016	250 V DC BATTERY SWGR 2A	CONTROL	130	TEA/T13
000470	02	2R22-S017	250 V DC BATTERY SWGR 2B	CONTROL	130	TB/T13
000433	02	2R23-S003	600 V STA SVC SWGR 2C & XFMR	CONTROL	130	TEA/T14
000471	02	2R23-S004	600 V STA SVC SWGR 2D & XFMR	CONTROL	130	TCA/T14
000434	01	2R24-S009	600/208 V MCC 2A INTAKE STRU	INTAKE	111	
000472	01	2R24-S010	600/208 V MCC 2B INTAKE STRU	INTAKE	111	
000435	01	2R24-S011	600 V MCC 2C ESS DIV 1	REACTOR	130	RF/R14
000436	01	2R24-S011A	600 V MCC ESS DIV 1	REACTOR	158	RA/R17

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
000473	01	2R24-S012	600 V MCC 2B ESS DIV 2	REACTOR	130	RF/R24
000774	01	2R24-S012A	600 V AC MCC	REACTOR	164	RB/R19
000474	01	2R24-S012B	600 V ESS MCC	REACTOR	164	RB/R21
000437	01	2R24-S018A	600 V MCC 2E-A ESS DIV 1	REACTOR	130	RH/R17
000478	01	2R24-S018B	600 V MCC 2E-B ESS DIV 2	REACTOR	130	RJ/R17
000438	01	2R24-S021	125/250 V DC MCC 2A ESS DIV 1	REACTOR	130	RB/R14
000475	01	2R24-S022	125/250 V DC MCC 2B ESS DIV 2	REACTOR	130	RH/R24
000439	01	2R24-S025	600/208 V MCC 2A ESS DIV 1	DIESEL	130	B02
000476	01	2R24-S027	600/208 V MCC 2C ESS DIV 2	DIESEL	130	C02
000440	14	2R25-S001	125 V DC CABINET 2A	CONTROL	130	TDA/T14
000479	14	2R25-S002	125 V DC CABINET 2B	CONTROL	130	TCA/T13
000441	14	2R25-S004	125 V DC CABINET 2D	DIESEL	130	B02
000480	14	2R25-S006	125 V DC CABINET 2F	DIESEL	130	C02
000442	14	2R25-S029	120/208 VAC CABINET 2J	DIESEL	130	B02
000481	14	2R25-S031	120/208 V AC CABINET 2L	DIESEL	130	C02
000443	14	2R25-S036	120/208 V AC ESS CABINET 2A	CONTROL	130	TDA/T13
000482	14	2R25-S037	120/208 V AC ESS CABINET 2B	CONTROL	130	TCA/T13
000445	14	2R25-S064	120/208 V AC CABINET 2A INST	CONTROL	130	TG/T13
000484	14	2R25-S065	120/208 V AC CABINET 2B INST	CONTROL	130	TG/T12
000447	14	2R25-S129	125 V DC DISTR CABINET 2E	CONTROL	130	TEA/T13
000485	14	2R25-S130	125 V DC DISTRIBUTION CAB 2D	CONTROL	130	TDA/T13
000462	18	2R26-M002	2R25-S064 DISCONNECT SWITCH	CONTROL	130	TE/T13
000502	18	2R26-M004	2R25-S025 DISCONNECT SWITCH	CONTROL	130	TC/T13
000464	18	2R26-M031A	125 V DC THROWOVER SWITCH 2A	CONTROL	130	TE/T13
000458	18	2R26-M031B	125 V DC THROWOVER SWITCH 2B	CONTROL	130	TF/T13
000497	18	2R26-M031C	125 V DC THROWOVER SWITCH 2C	CONTROL	130	TB/T13
000498	18	2R26-M031D	125 V DC THROWOVER SWITCH 2D	CONTROL	130	TB/T13
000459	18	2R26-M032A	125 V DC THROWOVER SWITCH	DIESEL	130	C02
000499	18	2R26-M032C	125 V DC THROWOVER SWITCH	DIESEL	130	C02
000733	01	2R27-S093	LOCAL STARTER E11-F006	REACTOR	130	RH/R24

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
000506	01	2R27-S096	LOCAL STARTER E11-F008	REACTOR	130	RH/R21
000776	0	2R34-S005A	SURGE PAK FOR 2P41-C001A	INTAKE	111	
000775	0	2R34-S005B	SURGE PAK FOR 2P41-C001B	INTAKE	111	
000777	0	2R34-S006A	SURGE PAK FOR 2E11-C001A	INTAKE	111	
000778	0	2R34-S006D	SURGE PAK FOR 2E11-C001D	INTAKE	111	
000448	15	2R42-S001A	125/250 V STATION BATTERY 2A	CONTROL	112	TEA/T14
000486	15	2R42-S001B	125/250 V DC STA BATTERY 2B	CONTROL	112	TDA/T14
000449	15	2R42-S002A	125 V DIESEL SYSTEM BATTERY 2A	DIESEL	130	B02
000487	15	2R42-S002C	125 V DIESEL SYS BATTERY 2C	DIESEL	130	C02
000450	16	2R42-S026	125 V BATTERY CHARGER 2A	CONTROL	130	TF/T13
000451	16	2R42-S027	125 V BATTERY CHARGER 2B	CONTROL	130	TF/T13
000488	16	2R42-S029	125 V BATTERY CHARGER 2D	CONTROL	130	TB/T13
000489	16	2R42-S030	125 V BATTERY CHARGER 2E	CONTROL	130	TB/T13
000452	16	2R42-S032A	125 V BATTERY CHARGER 2G	DIESEL	130	B02
000490	16	2R42-S032C	125 V BATTERY CHARGER 2J	DIESEL	130	C02
000412	0	2R43-A005A	DG 2A STARTING AIR RECEIVER	DIESEL	130	B01
000424	0	2R43-A005C	DG 2C STARTING AIR RECEIVER	DIESEL	130	C01
000413	0	2R43-A006A	DG 2A STARTING AIR RECEIVER	DIESEL	130	B01
000425	0	2R43-A006C	DG 2C STARTING AIR RECEIVER	DIESEL	130	C01
000411	18	2R43-N001A	DG 2A DAY TANK LS	DIESEL	130	B01
000423	18	2R43-N003C	DG 2C DAY TANK LS	DIESEL	130	C01
000724	20	2R43-P001A	DIESEL GEN 2A CONT PANEL	DIESEL	130	B02
000726	20	2R43-P001C	DIESEL GEN 2C CONT PANEL	DIESEL	130	
000407	17	2R43-S001A	DIESEL GENERATOR 2A	DIESEL	130	
000419	17	2R43-S001C	DIESEL GENERATOR 2C	DIESEL	130	
000453	16	2R44-S002	STATIC INVERTER	CONTROL	147	TI/T14
000491	16	2R44-S003	STATIC INVERTER	CONTROL	147	TH/T14
000256	10	2T41-B002B	CS/RHR PUMP ROOM COOLER B	REACTOR	087	RL/R24
000262	10	2T41-B003A	CS/RHR PUMP ROOM COOLER A	REACTOR	087	RL/R14
000257	10	2T41-B005B	HPCI PUMP ROOM COOLER B	REACTOR	087	RG/R25

LINE NO.	CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	BUILDING	ELEV	ROOM
(1)	(2)	(3)	(4)	(5)	(6)	(7)
000258	19	2T41-N019B	HPCI PUMP ROOM COOLER TE	REACTOR	087	RH/R25
000259	19	2T41-N020B	CS/RHR PUMP ROOM COOLER TE	REACTOR	087	RL/R24
000263	19	2T41-N021A	CS/RHR PUMP ROOM COOLER TE	REACTOR	087	RL/R14
000282	21	2T48-A001	NITROGEN STORAGE TANK	YARD		
000319	18	2T48-N009A	TORUS WATER TE	REACTOR	087	TORUS
000309	18	2T48-N009B	TORUS WATER TE	REACTOR	087	TORUS
000320	18	2T48-N009C	TORUS WATER TE	REACTOR	087	TORUS
000310	18	2T48-N009D	TORUS WATER TE	REACTOR	087	TORUS
000727	20	2U61-P001	LEAK DETECTION CONTROL PANEL	CONTROL	130	TD/T13
000728	20	2U61-P002	LEAK DETECTION CONTROL PANEL	CONTROL	130	TD/T13
000729	20	2U61-P003	LEAK DETECTION CONTROL PANEL	CONTROL	130	TD/T13
000730	20	2U61-P004	LEAK DETECTION CONTROL PANEL	CONTROL	130	TD/T13
000414	09	2X41-C010A	DG 2A ROOM EXHAUST FAN	DIESEL	130	A02
000426	09	2X41-C010C	DG 2C ROOM EXHAUST FAN	DIESEL	130	B02
000416	0	2X41-C013A	DG 2A ROOM LOUVER	DIESEL	130	
000428	0	2X41-C013B	DG 2C ROOM LOUVER	DIESEL	130	
000534	09	2X41-C014A	SWGR RM 2E FAN	DIESEL	ROOF	
000529	09	2X41-C014E	SWGR RM 2G FAN	DIESEL	ROOF	
000535	0	2X41-C015A	SWGR RM 2E LOUVER	DIESEL	130	SWGR RM 2E
000530	0	2X41-C015C	SWGR RM 2G LOUVER	DIESEL	130	SWGR RM 2G
000417	09	2X41-C016A	DG 2A BATTERY ROOM FAN	DIESEL	130	B02
000429	09	2X41-C016C	DG 2C BATTERY ROOM FAN	DIESEL	130	C02
000758	0	2X41-C024A	D/G BATT RM 2A FIRE DAMPER	DIESEL	130	2A
000759	0	2X41-C024B	D/G BATT RM 2C FIRE DAMPER	DIESEL	130	2C
000760	0	2X41-C024C	D/G BATT RM 2A FIRE DAMPER	DIESEL	130	2A
000761	0	2X41-C024D	D/G BATT RM 2C FIRE DAMPER	DIESEL	130	2C
000418	0	2X41-C028A	DG 2A BATTERY ROOM LOUVER	DIESEL	130	
000430	0	2X41-C028B	DG 2C BATTERY ROOM LOUVER	DIESEL	130	
000762	0	2X41-C030A	D/G RM 2A FIRE DAMPER	DIESEL	130	2A
000763	0	2X41-C030B	D/G RM 2C FIRE DAMPER	DIESEL	130	2C

Report Date/Time: 06-12-95 / 16:02:59
Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO. (1)	CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	BUILDING (5)	ELEV (6)	ROOM (7)
000764	0	2X41-C030C	D/G RM 2A FIRE DAMPER	DIESEL	130	2A
000765	0	2X41-C030D	D/G RM 2C FIRE DAMPER	DIESEL	130	2C
000415	18	2X41-N011A	DG 2A ROOM FAN THERMOSTAT	DIESEL	130	A02
000427	18	2X41-N011B	DG 2C ROOM FAN THERMOSTAT	DIESEL	130	B02
000536	18	2X41-N013A	SWGR RM 2E LVR THERMOSTAT	DIESEL	130	SWGR RM 2E
000531	18	2X41-N013C	SWGR RM 2G LVR THERMOSTAT	DIESEL	130	SWGR RM 2G
000532	18	2X41-N042	FLOW SWITCH FOR FAN 2X41-C010C	DIESEL	ROOF	
000537	18	2X41-N044	FLOW SWITCH FOR FAN 2X41-C010B	DIESEL	ROOF	
000538	18	2X41-N046	FLOW SWITCH FOR FAN 2X41-C010A	DIESEL	ROOF	
000533	18	2X41-N061	FLOW SWITCH FOR FAN 2X41-C010D	DIESEL	ROOF	
000731	20	2X43-P003A	CO2 ZONE 1 CONTROL CABINET	DIESEL	130	A01
000732	20	2X43-P003B	CO2 ZONE 2 CONTROL CABINET	DIESEL	130	B01
000409	21	2Y52-A001A	DG 2A FUEL OIL STORAGE TANK	YARD		
000421	21	2Y52-A001C	DG 2C FUEL OIL STORAGE TANK	YARD		
000408	21	2Y52-A101A	DG 2A FUEL OIL DAY TANK	DIESEL	130	
000420	21	2Y52-A101C	DG 2C FUEL OIL DAY TANK	DIESEL	130	C01
000410	05	2Y52-C001A	DG 2A FUEL OIL PUMP 2A1	YARD		
000422	05	2Y52-C101C	DG 2C FUEL OIL PUMP 2C2	YARD		
000527	09	2Z41-C014	STN BATT RM EMERG EXHAUST FAN	TURBINE	130	TE/T16
000528	09	2Z41-C015	STN BATT RM EMERG EXHAUST FAN	TURBINE	130	TE/T16

APPENDIX C

SCREENING VERIFICATION DATA SHEETS
PLANT HATCH UNIT 1

The Screening Verification Data Sheets (SVDSs) constitute the formal documentation of the screening verification and walkdown for the components contained in the Seismic Review Safe Shutdown Equipment List (SSEL) (Appendix A), and reflect the final judgment of the seismic capability engineers.

The format of the SVDS is provided by the Safe Shutdown Equipment Manager computer program supplied by the Seismic Qualification Utility Group (SQUG). The information on the SVDS is sorted by the component mark number (column 3). The SVDS includes the following information, where applicable and available, for each component.

Column Number	Description
1	Unique line number
2	Equipment class (see table 1 of this appendix)
3	Equipment identification number
4	Equipment description
5	Building or area in which equipment is located
6	Floor elevation from which the equipment can be seen
7	Building grid row/column indicating equipment location
8	Elevation at which the equipment is mounted
9	Indicates whether the equipment is mounted lower in the building than approximately 40 feet above grade: Y - Yes N - No

Column Number	Description
10	Indicates the source of the seismic capacity: BS SQUG bounding spectrum ABS 1.5 x SQUG bounding spectrum N/A Not applicable
11	Indicates the method used to define the seismic demand: GRS Ground response spectra AGS 1.5 x ground response spectra RRS Realistic, median-centered in-structure response spectra N/A Not applicable
12	Indicates whether the capacity of the equipment exceeds the demand: Y Yes - Capacity exceeds demand N No - Capacity does not exceed demand U Unknown whether capacity exceeds demand
13	Indicates whether the equipment is within the scope of the earthquake/testing equipment class and meets the intent of all the caveats for the equipment class: Y Yes - The equipment is in the equipment class, and the intent of all applicable caveats is satisfied. N No - The equipment is not in the equipment class, or the intent of all applicable caveats is not satisfied. U Unknown whether the equipment is in the equipment class or whether the intent of all applicable caveats is satisfied. N/A The earthquake/test equipment class and the caveats are not applicable to this equipment item.

Column Number	Description
14	Indicates whether the equipment anchorage meets the anchorage screening guidelines:

Y Yes - Anchorage capacities equal or exceed seismic demand, and anchorage is free of gross installation defects and has adequate stiffness.

N No - Anchorage capacities do not equal or exceed the seismic demand, anchorage is not without gross installation defects, or anchorage does not have adequate stiffness

U Unknown whether anchorage capacities equal or exceed seismic demand, or whether anchorage is free of gross installation defects or has adequate stiffness.

N/A Anchorage guidelines are not applicable to this equipment (e.g., valves are not evaluated for anchorage).

15 Indicates whether the equipment is free of adverse seismic interaction effects:

Y Yes - The equipment is free of interaction effects, or the interaction effects are acceptable and do not compromise the safe shutdown function of the equipment.

N No - The equipment is not free of adverse interaction effects.

U Unknown whether interaction effects will compromise the safe shutdown function of the equipment.

16 Indicates whether, in the final judgment of the seismic capability engineers, the seismic adequacy of the equipment is verified:

Y Yes - Seismic adequacy has been verified; i.e., code "Y" for all applicable screening guidelines:

1) Seismic capacity is greater than demand.

2) The equipment is in the earthquake/test equipment class and the intent of all the caveats is met (for use with bounding spectrum or GERS only).

Column Number	Description	
16 (continued)		3) Equipment anchorage is adequate.
		4) Seismic interaction effects will not compromise the safe shutdown function of the item of equipment.
	N	No - Seismic adequacy does not meet one or more of the seismic evaluation criteria. Equipment is identified as an outlier requiring further verification effort in accordance with section 5 of the SQUG Generic Implementation Procedure.
17	Explanatory notes:	
	1)	Passive electrical component which requires anchorage evaluation only.
	2)	Passive fire damper requires evaluation of fusible link only.
	3)	Non-USI A-46 component required for Individual Plant Examination of External Events only.

APPENDIX C

TABLE 1
EQUIPMENT CLASS DESIGNATIONS

Equipment Class (Column 2)	Description
01	Motor control centers
02	Low-voltage switchgear
03	Medium-voltage switchgear
04	Transformers
05	Horizontal pumps
06	Vertical pumps
07	Fluid-operated valves
08A	Motor-operated valves
08B	Solenoid-operated valves
09	Fans
10	Air handlers
11	Chillers
12	Air compressors
13	Motor generators
14	Distribution panels
15	Batteries on racks
16	Battery chargers and inverters
17	Engine generators
18	Instruments on racks
19	Temperature sensors
20	Instrumentation and control panels and cabinets
21	Tanks and heat exchangers

EDWIN I. HATCH NUCLEAR PLANT - UNIT 1

SVDS CERTIFICATION

All the information contained on this Screening Verification Data Sheet (SVDS) is, to the best of our knowledge and belief, correct and accurate. "All information" includes each entry and conclusion (whether verified to be seismically adequate or not) for components evaluated by the Seismic Review Team (SRT) as documented in the Seismic Evaluation Work Sheets (SEWS).

Name / Title	Signature / Date
P. W. Garrett / Seismic Capability Engineer	P.W. Garrett BY KDW / 11-9-94
B. R. Goforth / Seismic Capability Engineer	Billy R. Goforth / 11-9-94
T. B. Lantrip / Seismic Capability Engineer	T. B. Lantrip / 11/9/94
D. P. Moore / Seismic Capability Engineer	Donald P. Moore 11/10/94
K. D. Wooten / Seismic Capability Engineer	Keith D. Wooten / 11-9-94

The information provided to the Seismic Capability Engineers regarding systems and operations of the equipment contained on this SVDS is, to the best of our knowledge and belief, correct and accurate.

Name / Title	Signature / Date
W. S. Walker / Mechanical Systems Engineer	William Scott Walker / 11/9/94
J. E. Smith / Electrical Systems Engineer	Jim E Smith

hsign.doc

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0810	20	2H11-P652	ELECT AUX PWR CONTROL CONSOLE	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0770	20	2H21-P231	DIESEL GEN 2B - RELAY PANEL 2B	DIESEL	130	N/A		Y	BS	GRS	Y	N	Y	N	N	
0439	06	2P41-C002	PSW STANDBY PUMP 1B DIESEL	INTAKE	111			Y	BS	GRS	Y	N	Y	Y	N	
0440	07	2P41-F340	PSW ISOL VALVE TO 1B DIESEL	DIESEL	138	D/01		Y	BS	GRS	Y	Y	NA	Y	Y	
0790	03	2R22-S006	4160V SWGR EMERGENCY BUS 2F	DIESEL	130	C/03		Y	BS	GRS	Y	Y	N	Y	N	
0789	01	2R27-S037	LOCAL STARTER FOR 2P41-C002	DIESEL	130	C/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0823	18	2R43B-M01	DIESEL B UNIT 1/2 MODE SWITCH	DIESEL	130			Y	BS	GRS	Y	Y	Y	Y	Y	
0445	0	B21-A003A	AIR ACCUM FOR RELIEF VALVE A	DRYWELL	148	AZ043		Y	BS	GRS	Y	NA	Y	Y	Y	
0446	0	B21-A003B	AIR ACCUM FOR RELIEF VALVE B	DRYWELL	148	AZ077		Y	BS	GRS	Y	NA	Y	Y	Y	
0447	0	B21-A003C	AIR ACCUM FOR RELIEF VALVE C	DRYWELL	148	AZ073		Y	BS	GRS	Y	NA	Y	Y	Y	
0449	0	B21-A003E	AIR ACCUM FOR RELIEF VALVE E	DRYWELL	148	AZ088		Y	BS	GRS	Y	NA	Y	Y	Y	
0530	0	B21-A003F	AIR ACCUM FOR RELIEF VALVE F	DRYWELL	148	AZ295		Y	BS	GRS	Y	NA	Y	Y	Y	
0531	0	B21-A003G	AIR ACCUM FOR RELIEF VALVE G	DRYWELL	148	AZ266		Y	BS	GRS	Y	NA	Y	Y	Y	
0532	0	B21-A003H	AIR ACCUM FOR RELIEF VALVE H	DRYWELL	148	AZ331		Y	BS	GRS	Y	NA	Y	Y	Y	
0533	0	B21-A003J	AIR ACCUM FOR RELIEF VALVE J	DRYWELL	148	AZ320		Y	BS	GRS	Y	NA	Y	Y	Y	
0701	07	B21-F013A	RPV PRESSURE RELIEF VALVE	DRYWELL	148	AZ036		Y	BS	GRS	Y	Y	NA	Y	Y	
0702	07	B21-F013B	RPV SAFETY RELIEF VALVE	DRYWELL	148	AZ054		Y	BS	GRS	Y	Y	NA	Y	Y	
0703	07	B21-F013C	RPV PRESSURE RELIEF VALVE	DRYWELL	148	AZ064		Y	BS	GRS	Y	Y	NA	Y	Y	
0705	07	B21-F013E	RPV SAFETY RELIEF VALVE	DRYWELL	148	AZ056		Y	BS	GRS	Y	Y	NA	Y	Y	
0706	07	B21-F013F	RPV SAFETY RELIEF VALVE	DRYWELL	148	AZ304		Y	BS	GRS	Y	Y	NA	Y	Y	
0707	07	B21-F013G	RPV PRESSURE RELIEF VALVE	DRYWELL	148	AZ293		Y	BS	GRS	Y	Y	NA	Y	Y	
0708	07	B21-F013H	RPV PRESSURE RELIEF VALVE	DRYWELL	148	AZ324		Y	BS	GRS	Y	Y	NA	Y	Y	
0709	07	B21-F013J	RPV SAFETY RELIEF VALVE	DRYWELL	148	AZ306		Y	BS	GRS	Y	Y	NA	Y	Y	
0693	07	B21-F022A	INBOARD MSIV	DRYWELL	130	RB/R07		Y	BS	GRS	Y	Y	NA	Y	Y	
0694	07	B21-F022B	INBOARD MSIV	DRYWELL	130	RB/R07		Y	BS	GRS	Y	Y	NA	Y	Y	
0695	07	B21-F022C	INBOARD MSIV	DRYWELL	130	RB/R07		Y	BS	GRS	Y	Y	NA	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0696	07	B21-F022D	INBOARD MSIV	DRYWELL	130	RB/R07		Y	BS	GRS	Y	Y	NA	Y	Y	
0697	07	B21-F028A	OUTBOARD MSIV	REACTOR	130	RB/R07		Y	BS	GRS	Y	Y	NA	Y	Y	
0698	07	B21-F028B	OUTBOARD MSIV	REACTOR	130	RB/R07		Y	BS	GRS	Y	Y	NA	Y	Y	
0699	07	B21-F028C	OUTBOARD MSIV	REACTOR	130	RB/R07		Y	BS	GRS	Y	Y	NA	Y	Y	
0700	07	B21-F028D	OUTBOARD MSIV	REACTOR	130	RB/R07		Y	BS	GRS	Y	Y	NA	Y	Y	
0726	18	B21-N093A	RPV LEVEL 8 LT	REACTOR	158	RH/R08		Y	BS	GRS	Y	Y	Y	Y	Y	
0730	18	B21-N093B	RPV LEVEL 8 LT	REACTOR	158	RH/R04		Y	BS	GRS	Y	Y	Y	Y	Y	
0460	08A	B31-F023A	RECIRC PUMP C001A SUCTION VALVE	DRYWELL	117	RF/R07		Y	BS	GRS	Y	Y	NA	N	N	
0466	08B	C11-D001-117	PILOT SCRAM SOLENOID	REACTOR	130	HCU		Y	BS	GRS	Y	Y	NA	Y	Y	
0467	08B	C11-D001-118	PILOT SCRAM SOLENOID	REACTOR	130	HCU		Y	BS	GRS	Y	Y	NA	Y	Y	
0468	0	C11-D001-125	SCRAM ACCUMULATOR	REACTOR	130	HCU		Y	NA	NA	Y	NA	Y	Y	Y	
0469	07	C11-D001-126	SCRAM INLET VALVE	REACTOR	130	HCU		Y	BS	GRS	Y	Y	NA	Y	Y	
0470	07	C11-D001-127	SCRAM OUTLET VALVE	REACTOR	130	HCU		Y	BS	GRS	Y	Y	NA	Y	Y	
0462	08B	C11-F009A	SDV VENT & DRAIN PILOT VALVE	REACTOR	130	RA/R05		Y	BS	GRS	Y	Y	NA	Y	Y	
0523	08B	C11-F009B	SDV VENT & DRAIN PILOT VALVE	REACTOR	130	RA/R05		Y	BS	GRS	Y	Y	NA	Y	Y	
0463	07	C11-F010A	SDV VENT VALVE	REACTOR	130	RH/R12		Y	BS	GRS	Y	Y	NA	Y	Y	
0464	07	C11-F010B	SDV VENT VALVE	REACTOR	130	RH/R03		Y	BS	GRS	Y	Y	NA	Y	Y	
0465	07	C11-F011	SDV DRAIN VALVE	REACTOR	130	RB/R03		Y	BS	GRS	Y	Y	NA	Y	Y	
0480	07	C11-F035A	SDV VENT VALVE	REACTOR	130	RH/R12		Y	BS	GRS	Y	Y	NA	Y	Y	
0481	07	C11-F035B	SDV VENT VALVE	REACTOR	130	RH/R12		Y	BS	GRS	Y	Y	NA	Y	Y	
0482	07	C11-F037	SDV DRAIN VALVE	REACTOR	130	RB/R04		Y	BS	GRS	Y	Y	NA	Y	Y	
0479	08B	C11-F040A	SDV VENT & DRAIN PILOT VALVE	REACTOR	130	RA/R05		Y	BS	GRS	Y	Y	NA	Y	Y	
0524	08B	C11-F040B	SDV VENT & DRAIN PILOT VALVE	REACTOR	130	RA/R05		Y	BS	GRS	Y	Y	NA	Y	Y	
0491	08B	C11-F110A	BACKUP SCRAM VALVE	REACTOR	130	RA/R05		Y	BS	GRS	Y	Y	NA	Y	Y	
0492	08B	C11-F110B	BACKUP SCRAM VALVE	REACTOR	130	RA/R05		Y	BS	GRS	Y	Y	NA	Y	Y	
0001	14	C71-P001	RPS POWER DISTRIBUTION PANEL BUS A	CONTROL	130	TG/T12		Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO. (1)	EQUIP CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	<---------- Building (5)	EQUIPMENT Flr.Elv. (6)	LOCATION -----> Rm. or Row/Col. (7)	Base Elev. (8)	<40'? (9)	Capacity Spectrum (10)	Demand Spectrum (11)	Cap. > Demand? (12)	Caveats OK? (13)	Anchor OK? (14)	Inter- act OK? (15)	Equip OK? (16)	Notes (17)
0840	18	C71-P003E	PROTECTION PANEL	CONTROL	130	TF/T12		Y	BS	GRS	Y	Y	Y	Y	Y	
0841	18	C71-P003F	PROTECTION PANEL	CONTROL	130	TF/T12		Y	BS	GRS	Y	Y	Y	Y	Y	
0839	18	C71-S002	LINE VOLTAGE REGULATOR	CONTROL	130			Y	BS	GRS	Y	Y	Y	Y	Y	
0775	20	C82-P002	REMOTE SHUTDOWN PANEL	REACTOR	158	RH/R05		Y	BS	GRS	Y	Y	Y	Y	Y	
0354	21	E11-B001A	RHR HEAT EXCHANGER A	REACTOR	087	RL/R13		Y	NA	NA	Y	NA	Y	NA	Y	
0356	21	E11-B001B	RHR HEAT EXCHANGER B	REACTOR	087	RL/R03		Y	NA	NA	Y	NA	Y	NA	Y	
0290	06	E11-C001A	RHR SERVICE WATER PUMP 1A	INTAKE	111			Y	BS	GRS	Y	N	Y	Y	N	
0357	06	E11-C001D	RHR SERVICE WATER PUMP 1D	INTAKE	111			Y	BS	GRS	Y	N	Y	Y	N	
0292	06	E11-C002A	RHR PUMP 2A	REACTOR	087	RL/R11		Y	BS	GRS	Y	Y	Y	Y	Y	
0358	06	E11-C002D	RHR PUMP 2D	REACTOR	087	RL/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
0335	08A	E11-F007A	RHR PUMP 2A & 2C MIN FLOW BYPASS VLV	REACTOR	087	RL/R11		Y	BS	GRS	Y	Y	NA	Y	Y	
0362	08A	E11-F007B	RHR PUMP 2B & 2D MIN FLOW BYPASS VLV	REACTOR	087	RL/R03		Y	BS	GRS	Y	Y	NA	Y	Y	
0304	08A	E11-F015A	RHR LPCI DISCHARGE VALVE	REACTOR	130	RJ/R08		Y	BS	GRS	Y	Y	NA	Y	Y	
1066	08A	E11-F016A	CONTAINMENT SPRAY DISCH VALVE	REACTOR	158	RH/R08		Y	NA	NA	Y	Y	NA	Y	Y	3
1029	08A	E11-F016B	CONTAINMENT SPRAY DISCH VALVE	REACTOR	158	RH/R08		Y	NA	NA	Y	Y	NA	Y	Y	3
0340	08A	E11-F017A	RHR LPCI DISCHARGE VALVE	REACTOR	130	RJ/R08		Y	BS	GRS	Y	Y	NA	Y	Y	
1067	08A	E11-F021A	CONT SPRAY INBRD GATE MOV	REACTOR	158	RH/R08		Y	NA	NA	Y	Y	NA	Y	Y	3
1030	08A	E11-F021B	CONT SPRAY INBRD GATE MOV	REACTOR	158	RH/R08		Y	NA	NA	Y	Y	NA	Y	Y	3
0366	08A	E11-F024B	RHR TEST LINE TORUS ISO	REACTOR	087	RL/R05		Y	BS	GRS	Y	Y	NA	Y	Y	
1064	08A	E11-F027A	SUPP POOL SPRAY VALVE	REACTOR	087	RJ/R10		Y	BS	GRS	Y	Y	NA	Y	Y	3
1027	08A	E11-F027B	SUPP POOL SPRAY VALVE	REACTOR	087	RL/R05		Y	BS	GRS	Y	Y	NA	Y	Y	3
0369	08A	E11-F028B	RHR INLET TO SUPP POOL VALVE	REACTOR	087	RL/R05		Y	BS	GRS	Y	Y	NA	Y	Y	
0344	08A	E11-F048A	RHR HX A BYPASS VALVE	REACTOR	087	RL/R13		Y	BS	GRS	Y	Y	NA	Y	Y	
0371	08A	E11-F048B	RHR HX B BYPASS VALVE	REACTOR	087	RJ/R02		Y	BS	GRS	Y	Y	NA	Y	Y	
0311	08A	E11-F068A	RHR HX A TUBE TO SHELL OUTLET	REACTOR	087	RH/R13		Y	BS	GRS	Y	Y	NA	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1RS.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0374	08A	E11-F068B	RHR HX B TUBE TO SHELL OUTLET	REACTOR	087	RH/R02		Y	BS	GRS	Y	Y	NA	Y	Y	
0317	18	E11-N002A	RHR HX A TUBE TO SHELL DP TRANS	REACTOR	087	H21-P018		Y	BS	GRS	Y	Y	Y	Y	Y	
0380	18	E11-N002B	RHR HX B TUBE TO SHELL DP TRANS	REACTOR	087	H21-P021		Y	BS	GRS	Y	Y	Y	Y	Y	
0319	18	E11-N007A	RHRSW HX A FLOW TRANSMITTER	REACTOR	087	H21-P018		Y	BS	GRS	Y	Y	Y	Y	Y	
0382	18	E11-N007B	RHRSW HX B FLOW TRANSMITTER	REACTOR	087	RL/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
0322	18	E11-N017A	RHR HX A INLET PRESSURE SWITCH	REACTOR	087	RH/R13		Y	BS	GRS	Y	Y	Y	Y	Y	
0385	18	E11-N017B	RHR HX B INLET PRESSURE SWITCH	REACTOR	087	RL/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
0323	18	E11-N017C	RHR HX A INLET PRESSURE SWITCH	REACTOR	087	RH/R13		Y	BS	GRS	Y	Y	Y	Y	Y	
0386	18	E11-N017D	RHR HX B INLET PRESSURE SWITCH	REACTOR	087	RL/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
1116	18	E11-N055A	RHR PUMP A DISCHARGE PT	REACTOR	087	RJ/R12		Y	BS	GRS	Y	Y	Y	Y	Y	
1117	18	E11-N055B	RHR PUMP B DISCHARGE PT	REACTOR	087	RL/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
1118	18	E11-N055C	RHR PUMP C DISCHARGE PT	REACTOR	087	RJ/R12		Y	BS	GRS	Y	Y	Y	Y	Y	
1119	18	E11-N055D	RHR PUMP D DISCHARGE PT	REACTOR	087	RJ/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
1120	18	E11-N056A	RHR PUMP A DISCHARGE PT	REACTOR	087	RJ/R12		Y	BS	GRS	Y	Y	Y	Y	Y	
1121	18	E11-N056B	RHR PUMP B DISCHARGE PT	REACTOR	087	RL/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
1122	18	E11-N056C	RHR PUMP C DISCHARGE PT	REACTOR	087	RJ/R12		Y	BS	GRS	Y	Y	Y	Y	Y	
1123	18	E11-N056D	RHR PUMP D DISCHARGE PT	REACTOR	087	RJ/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
0324	18	E11-N082A	RHR PUMP A & C FLOW DP TRANS	REACTOR	087	RL/R10		Y	BS	GRS	Y	Y	Y	Y	Y	
0387	18	E11-N082B	RHR PUMP 2B & 2D FLOW DP TRAN	REACTOR	087	RL/R05		Y	BS	GRS	Y	Y	Y	Y	Y	
1124	18	E11-N094A	DRYWELL PT	REACTOR	158	RC/R08		Y	BS	GRS	Y	Y	Y	Y	Y	
1125	18	E11-N094B	DRYWELL PT	REACTOR	158	RJ/R04		Y	BS	GRS	Y	Y	Y	Y	Y	
1126	18	E11-N094C	DRYWELL PT	REACTOR	158	RC/R08		Y	BS	GRS	Y	Y	Y	Y	Y	
1127	18	E11-N094D	DRYWELL PT	REACTOR	158	RJ/R04		Y	BS	GRS	Y	Y	Y	Y	Y	
0394	06	E21-C001B	CORE SPRAY PUMP B	REACTOR	087	RL/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
0397	08A	E21-F005B	CORE SPRAY INLET VALVE	REACTOR	158	RF/R04		N	ABS	RRS	Y	Y	NA	Y	Y	
0400	08A	E21-F031B	MINIMUM FLOW BYPASS VALVE	REACTOR	087	RL/R02		Y	BS	GRS	Y	Y	NA	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0403	18	E21-N051B	CORE SPRAY FLOW TRANSMITTER	REACTOR	087	RL/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
1130	18	E21-N052A	CS PUMP C001A HIGH PRESS	REACTOR	087	RJ/R12		Y	BS	GRS	Y	Y	Y	Y	Y	
1131	18	E21-N052B	CS PUMP C001B HIGH PRESS	REACTOR	087	RL/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
1128	18	E21-N055A	CS PUMP C001A HIGH PRESS	REACTOR	087	RJ/R12		Y	BS	GRS	Y	Y	Y	Y	Y	
1129	18	E21-N055B	CS PUMP C001B HIGH PRESS	REACTOR	087	RL/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
0109	05	E41-C001	HPCI PUMP	REACTOR	087	RG/R01		Y	BS	GRS	Y	Y	Y	Y	Y	
0110	05	E41-C002	HPCI TURBINE	REACTOR	087	RL/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
0111	05	E41-C002-3	HPCI LUBE OIL PUMP	REACTOR	087	RL/R01		Y	BS	GRS	Y	Y	Y	Y	Y	
0112	08A	E41-F001	HPCI TURBINE STEAM SUPPLY VLV	REACTOR	087	RL/R01		Y	BS	GRS	Y	Y	NA	Y	Y	
0115	08A	E41-F004	HPCI PUMP SUCT FROM CST	REACTOR	087	RH/R01		Y	BS	GRS	Y	Y	NA	Y	Y	
0116	08A	E41-F006	HPCI PUMP INBD DISCH VLV	REACTOR	087	RA/R07		Y	BS	GRS	Y	Y	NA	Y	Y	
0119	08A	E41-F012	HPCI MIN FLOW BYPASS VLV	REACTOR	087	RH/R02		Y	BS	GRS	Y	Y	NA	Y	Y	
0122	08	E41-F041	HPCI PUMP SUCT FROM SUPP POOL	REACTOR	087	RH/R01		Y	BS	GRS	Y	Y	NA	Y	Y	
0123	08A	E41-F042	HPCI PUMP SUCT FROM SUPP POOL	REACTOR	087	RH/R02		Y	BS	GRS	Y	Y	NA	Y	Y	
0125	08A	E41-F059	BAR COND COOLING WATER VLV	REACTOR	087	RL/R02		Y	BS	GRS	Y	Y	NA	Y	Y	
1081	07	G11-F003	DRYWELL FL DR PMP ISO VLV	REACTOR	087	RE/R03		Y	BS	GRS	Y	Y	NA	N	N	3
1082	07	G11-F004	DRYWELL FL DR PMP ISO VLV	REACTOR	087	RE/R03		Y	BS	GRS	Y	Y	NA	Y	Y	3
1079	07	G11-F019	DRYWELL EQ DR PMP ISO VLV	REACTOR	087	RB/R05		Y	BS	GRS	Y	Y	NA	Y	Y	3
1080	07	G11-F020	DRYWELL EQ DR PMP ISO VLV	REACTOR	087	RB/R05		Y	BS	GRS	Y	Y	NA	Y	Y	3
1077	08A	G31-F001	RWCU INBOARD ISO GATE VLV	DRYWELL	158	AZ 170		Y	NA	NA	Y	Y	NA	Y	Y	3
1078	08A	G31-F004	RWCU OUTBOARD ISO VLV	REACTOR	158	RH/R07		Y	NA	NA	Y	Y	NA	Y	Y	3
0734	20	H11-P601	REACTOR & CONT COOLING & ISOLA CONTROL PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0735	20	H11-P602	REACTOR WTR CLEAN UP & RECIR CONTROL PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0736	20	H11-P603	REACTOR CONTROL PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0737	20	H11-P605A	DIV 1 ANALOG SIG CONV/ISOLATION	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCHIRS.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0738	20	H11-P605B	DIV 2 ANALOG SIG CONV/ISOLATION	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0739	20	H11-P606	START U RANGE NEUTRON MONTORING	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0807	20	H11-P608	POWER RANGE NEUTRON MONITORING PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0740	20	H11-P609	CHAN A PRI ISO SYS REAC PROT SYS	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0842	20	H11-P610	RPS TEST MON VB	CONTROL	164	TE/T11		Y	BS	GRS	Y	Y	Y	U	N	
0741	[illegible]	H11-P611	CHAN B PRI ISO SYS REAC PROT SYS	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0742	20	H11-P612	FW & RECIR INSTRUMENTATION PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0808	20	H11-P613	PROCESS INSTRUMENTATION PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
1132	20	H11-P614	NUCLEAR STEAM RECORDER VB	CONTROL	164	TE/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0743	20	H11-P617	CHAN A RHR CORE SPRAY RELAY PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0744	20	H11-P618	CHAN B RHR CORE SPRAY RELAY PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0745	20	H11-P620	HPCI RELAY COMPUTER EQUIPMENT PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	N	Y	U	N	
0746	20	H11-P622	INBOARD-PRI CONTROL ISOLATION RELAYS PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0747	20	H11-P623	OUTBOARD-PRI CONTAIN ISOLATION RELAYS PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0748	20	H11-P626	CORE SPRAY CONTR PNL DIV I	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0749	20	H11-P627	CORE SPRAY CONTR PNL DIV II	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0809	20	H11-P628	AUTOMATIC BLOWDOWN RELAY PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0750	20	H11-P650	TURB, FW & COND CONTROL CONSOLE	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0751	20	H11-P651	GEN & STA SER CONTROL CONSOLE	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0752	20	H11-P652	ELEC AUX PWR CONTROL CONSOLE	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0795	20	H11-P654	GAS TREAT & VENT VERTICAL PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	N	Y	U	N	
0796	20	H11-P657	VENT & DRYWELL INERT VERTICAL PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0797	20	H11-P691	ANALOG SIGNAL CONVERSION/ISOLATION PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	N	N	N	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0798	20	H11-P700	ANALYZER/VENT & LEAK DETECTION PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	U	N	
0799	20	H11-P921	ATTS RPS PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0800	20	H11-P922	ATTS RPS PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0801	20	H11-P923	ATTS RPS PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0802	20	H11-P924	ATTS RPS PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0802	20	H11-P925	ATTS ECCS PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	N	N	
0804	20	H11-P926	ATTS ECCS PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	N	N	
0805	20	H11-P927	ATTS ECCS PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0806	20	H11-P928	ATTS ECCS PANEL	CONTROL	164	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0786	18	H21-P016	CS/HPCI LEAK DET RACK	REACTOR	130	RF/R04		Y	BS	GRS	Y	Y	Y	Y	Y	
0772	18	H21-P018	RHR INSTRUMENT RACK-CHANNEL A	REACTOR	087	RL/R13		Y	BS	GRS	Y	Y	Y	Y	Y	
0777	18	H21-P019	CORE SPRAY SYSTEM B RACK	REACTOR	087	RL/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
0776	18	H21-P021	RHR CHANNEL B RACK	REACTOR	087	RL/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
0787	18	H21-P036	HPCI LEAK DET RACK	REACTOR	130	RF/R04		Y	BS	GRS	Y	Y	Y	Y	Y	
0773	20	H21-P173	SHUTDOWN INSTRUMENT PANEL	REACTOR	130	RL/R03		Y	BS	GRS	Y	Y	Y	N	N	
0753	20	H21-P175	HOT SHUTDOWN PUMP CONTROL PANEL	DIESEL	130	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0754	20	H21-P200	DIESEL GEN 1A - RELAY PANEL 1A	DIESEL	130	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0755	20	H21-P201	DIESEL GEN 1B - RELAY PANEL 1A	DIESEL	130	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0756	20	H21-P202	DIESEL GEN 1C - RELAY PANEL 1A	DIESEL	130	N/A		Y	BS	GRS	Y	Y	N	Y	N	
0757	20	H21-P230	DEISEL GEN 1A - RELAY PANEL 1B	DIESEL	130	N/A		Y	BS	GRS	Y	N	Y	N	N	
0758	20	H21-P231	DIESEL GEN 1B - RELAY PANEL 1B	DIESEL	130	N/A		Y	BS	GRS	Y	N	Y	N	N	
07[illegible]	[illegible]	H21-P232	DIESEL GEN 1C - RELAY PANEL 1B	DIESEL	130	N/A		Y	BS	GRS	Y	N	N	N	N	
0760	20	H21-P245	600 VOLT SWGR 1C CONTROL PANEL	CONTROL	130	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0761	20	H21-P246	600 VOLT SWGR 1D CONTROL PANEL	CONTROL	130	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0817	18	H21-P248	250V DC SWITCHGEAR CONTROL PANEL	DIESEL	130	TF/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0818	18	H21-P249	250V DC SWITCHGEAR CONTROL PANEL	DIESEL	130	TB/T11		Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0762	20	H21-P255	MOV & FUEL PUMP CTRL PANEL 1A - DIV I	DIESELDB	130	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0763	20	H21-P256	MOV & FUEL PUMP CTRL PANEL 1B - DIV II	DIESEL	130	N/A		Y	BS	GRS	Y	Y	N	N	N	
0764	20	H21-P257	GEN 1A HEAT & VENT CONTROL PANEL	DIESEL	130	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
1101	20	H21-P258	GEN 1B H&V CONT PNL	DIESEL	130	D01		Y	BS	GRS	Y	Y	Y	Y	Y	
1102	20	H21-P259	GEN 1C H&V CONT PNL	DIESEL	130	E01		Y	BS	GRS	Y	Y	Y	Y	Y	
0765	20	H21-P266	MOV CONTROL PANEL 1A-DIV I	INTAKE	111	N/A		Y	NA	NA	NA	NA	Y	NA	Y	1
0774	20	H21-P267	MOV CONTROL PANEL 1B-DIV II	INTAKE	111	N/A		Y	NA	NA	NA	NA	Y	NA	Y	1
0092	18	H21-P285	125/250V STATION BATT 1A FUSE BOX	CONTROL	112	TF/T11		Y	NA	NA	NA	NA	Y	NA	Y	1
0093	18	H21-P286	125/250V STATION BATT 1A FUSE BOX	CONTROL	112	TE/T11		Y	NA	NA	NA	NA	Y	NA	Y	1
0094	18	H21-P267	125/250V STATION BATT 1A FUSE BOX	CONTROL	112	TF/T11		Y	NA	NA	NA	NA	Y	NA	Y	1
0095	18	H21-P288	125/250V STATION BATT 1B FUSE BOX	CONTROL	112	TC/T11		Y	NA	NA	NA	NA	Y	NA	Y	1
0096	18	H21-P289	125/250V STATION BATT 1B FUSE BOX	CONTROL	112	TE/T11		Y	NA	NA	NA	NA	Y	NA	Y	1
0097	18	H21-P290	125/250V STATION BATT 1B FUSE BOX	CONTROL	112	TD/T10		Y	NA	NA	NA	NA	Y	NA	Y	1
0714	18	H21-P294	TERMINAL BOX	CONTROL	112	TE/T12		Y	NA	NA	NA	NA	Y	NA	Y	1
0715	18	H21-P295	TERMINAL BOX	CONTROL	112	TE/T12		Y	NA	NA	NA	NA	Y	NA	Y	1
0767	20	H21-P303	DIESEL 1A LEAD TIMER PANEL	DIESEL	130	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0768	20	H21-P304	DIESEL 1B LEAD TIMER PANEL	DIESEL	130	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0769	20	H21-P305	DIESEL 1C LEAD TIMER PANEL	DIESEL	130	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0781	18	H21-P404A	INSTRUMENT RACK	REACTOR	158	RH/R08		Y	BS	GRS	Y	Y	Y	Y	Y	
0783	18	H21-P404B	INSTRUMENT RACK	REACTOR	158	RH/R08		Y	BS	GRS	Y	Y	Y	Y	Y	
0782	18	H21-P405A	INSTRUMENT RACK	REACTOR	158	RH/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
0778	18	H21-P405B	INSTRUMENT RACK	REACTOR	158	RH/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
0779	18	H21-P409	INSTRUMENT RACK	REACTOR	130	RH/R10		Y	BS	GRS	Y	Y	Y	Y	Y	
0780	18	H21-P410	INSTRUMENT RACK	REACTOR	130	RF/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
0784	18	H21-P414A	INSTRUMENT RACK	REACTOR	087	RJ/R02		Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0785	18	H21-P414B	INSTRUMENT RACK	REACTOR	087	RJ/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
0788	18	H21-P434	INSTRUMENT RACK	REACTOR	087	RJ/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
1113	20	H21-P530A	FAN 1A PANEL	INTAKE	111	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
1114	20	H21-P530B	FAN 1B PANEL	INTAKE	111	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
1115	20	H21-P530C	FAN 1C PANEL	INTAKE	111	N/A		Y	BS	GRS	Y	Y	Y	Y	Y	
0406	06	P41-C001A	PLANT SERVICE WATER PUMP 1A	INTAKE	111			Y	BS	GRS	Y	N	Y	Y	N	
0426	06	P41-C001B	PLANT SERVICE WATER PUMP 1B	INTAKE	111			Y	BS	GRS	Y	N	Y	Y	N	
0837	07	P4[illegible]035B	HPCI ROOM COOLER INLET VALVE	REACTOR	087	RL/R01		Y	BS	GRS	Y	Y	NA	Y	Y	
0428	07	P41-F036B	RHR/CS ROOM COOLER INLET VALVE	REACTOR	087	RL/R03		Y	BS	GRS	Y	Y	NA	Y	Y	
0407	07	P41-F037A	RHR PUMP COOLER 2A INLET VALVE	REACTOR	087	RL/R13		Y	BS	GRS	Y	Y	NA	Y	Y	
0429	07	P41-F037D	RHR PUMP COOLER 2D INLET VALVE	REACTOR	087	RL/R03		Y	BS	GRS	Y	Y	NA	Y	Y	
0408	07	P41-F039A	RHR/CS ROOM COOLER 2A VALVE	REACTOR	087	RL/R11		Y	BS	GRS	Y	Y	NA	Y	Y	
0897	08B	P41-F123A	Z41-B025 PSW INLET ISOL SV	CONTROL	180	TB/T11		N	ABS	RRS	Y	Y	NA	Y	Y	
0972	08B	P41-F123B	Z41-B025 PSW INLET ISOL SV	CONTROL	180	TB/T[illegible]1		N	ABS	RRS	Y	Y	NA	Y	Y	
0412	08	P41-F310A	PSW TURBINE BLDG ISOL VALVE	YARD				Y	BS	GRS	Y	Y	NA	Y	Y	
0431	08A	P41-F310B	PSW TURBINE BLDG ISOL VALVE	YARD				Y	BS	GRS	Y	Y	NA	Y	Y	
0442	08A	P41-F312	PSW RETURN LINE ISOL VALVE	INTAKE	098			Y	BS	GRS	Y	Y	NA	Y	Y	
0975	08A	P41-F422A	Z41-B008B ISOL GLOBE MOV	CONTROL	180	TB/T11		N	ABS	RRS	Y	Y	NA	Y	Y	
0420	18	P41-N200A	PSW STRAINER DP SWITCH	INTAKE	087			Y	BS	GRS	Y	Y	U	Y	N	
0436	18	P41-N200B	PSW STRAINER DP SWITCH	INTAKE	087			Y	BS	GRS	Y	Y	U	Y	N	
0976	18	P41-N520	PSW CB A/C-1B DI DPS	CONTROL	180	TB/T11		N	ABS	RRS	Y	N	N	Y	N	
0935	18	P41-N521	PSW CD A/C-1B DII DPS	CONTROL	180	TB/T11		N	ABS	RRS	Y	N	N	Y	N	
0525	0	P70-A002A	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RL/R09		Y	NA	NA	Y	NA	Y	Y	Y	
0526	0	P70-A002B	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RL/R09		Y	NA	NA	Y	NA	Y	Y	Y	
0527	0	P70-A002C	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RL/R09		Y	NA	NA	Y	NA	Y	Y	Y	
0450	07	P70-F001A	DW PNEUMATIC NITROGEN SUPPLY VALVE	REACTOR	087	RA/R02		Y	BS	GRS	Y	Y	NA	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1RS.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0528	0	P70-F084	EMERGENCY NITROGEN ISOL VALVE	REACTOR	130	RL/R09		Y	NA	NA	Y	NA	NA	Y	Y	
0529	0	P70-F141	EMERGENCY NITROGEN ISOL VALVE	REACTOR	130	RL/R09		Y	NA	NA	Y	NA	NA	Y	Y	
0454	18	P70-N022A	DRYWELL PNEUMATIC FLOW TRANS	REACTOR	130	RA/R06		Y	BS	GRS	Y	Y	Y	Y	Y	
0455	18	P70-N022B	DRYWELL PNEUMATIC FLOW TRANS	REACTOR	130	RA/R06		Y	BS	GRS	Y	Y	Y	Y	Y	
0010	04	R11-S004	45KVA 600-120/208V PWR XFMR 1D	DIESEL	130	D/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0011	04	R11-S005	45KVA 600-120/208V PWR XFMR 1E	DIESEL	130	E/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0012	04	R11-S006	45KVA 600-120/208V PWR XFMR 1F	DIESEL	130	F/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0013	04	R11-S039	45KVA 600-120/208V TRANSFORMER	REACTOR	130	RF/R13		Y	BS	GRS	Y	Y	Y	Y	Y	
0014	04	R11-S040	45KVA 600-120/208V TRANSFORMER	REACTOR	130	RF/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
0015	04	R11-S041	112.5 KVA 600-120/208V ESSENTIAL XFMR 1B	CONTROL	130	TF/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0016	04	R11-S042	112.5 KVA 600-120/208V ESSENTIAL XFMR 1C	CONTROL	130	TD/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0815	04	R11-S071	MISC POWER TRANSFORMER	DIESEL	130			Y	BS	GRS	Y	Y	Y	Y	Y	
0017	03	R22-S005	4160V SWGR EMERGENCY BUS 1E	DIESEL	130	D/03		Y	BS	GRS	Y	Y	Y	Y	Y	
0018	03	R22-S006	4160V SWGR EMERGENCY BUS 1F	DIESEL	130	E/03		Y	BS	GRS	Y	Y	Y	Y	Y	
0019	03	R22-S007	4160V SWGR EMERGENCY BUS 1G	DIESEL	130	F/03		Y	BS	GRS	Y	Y	Y	Y	Y	
0020	02	R22-S016	250V DC BATTERY DIV 1 SWGR 1A	CONTROL	130	TF/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0021	02	R22-S017	250V DC BATTERY DIV 2 SWGR 1B	CONTROL	130	TB/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0022	02	R23-S003	600V SWGR EMERGENCY BUS 1C & 4160-600V XFMR	CONTROL	130	TE/T10		Y	BS	GRS	Y	Y	Y	Y	Y	
0023	02	R23-S004	600V SWGR EMERGENCY BUS 1D & 4160-600V XFMR	CONTROL	130	TC/T10		Y	BS	GRS	Y	Y	Y	Y	Y	
1103	01	R24-S002	600V/208V MCC 1B	CONTROL	180	TE/T11		N	ABS	RRS	Y	N	N	Y	N	
1104	01	R24-S003	600V/208V MCC 1C	CONTROL	180	TE/T11		N	ABS	RRS	Y	N	N	Y	N	
0024	01	R24-S009	600/208V MCC 1A	INTAKE	111			Y	ABS	RRS	U	Y	U	Y	N	
0025	01	R24-S010	600/208V MCC 1B	INTAKE	111			Y	ABS	RRS	U	Y	U	Y	N	
0026	01	R24-S011	600V ESS DIV 1 MCC 1C	REACTOR	130	RH/R13		Y	ABS	RRS	Y	Y	Y	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCHIR5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO. (1)	EQUIP CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	<---------- Building (5)	EQUIPMENT Flr.Elv. (6)	LOCATION -----> Rm. or Row/Col. (7)	Base Elev. (8)	<40'? (9)	Capacity Spectrum (10)	Demand Spectrum (11)	Cap. > Demand? (12)	Caveats OK? (13)	Anchor OK? (14)	Inter-act OK? (15)	Equip OK? (16)	Notes (17)
0027	01	R24-S012	600V ESS DIV 2 MCC 1B	REACTOR	130	RF/R02		Y	ABS	RRS	Y	Y	Y	Y	Y	
0028	01	R24-S018A	600V ESS DIV 1 MCC 1E-A	REACTOR	130	RL/R08		Y	ABS	RRS	Y	Y	Y	Y	Y	
0029	01	R24-S018B	600V ESS DIV 2 MCC 1E-B	REACTOR	130	RL/R08		Y	ABS	RRS	Y	Y	Y	Y	Y	
0032	01	R24-S022	125/250V DC ESS DIV 2 MCC 1B	REACTOR	130	RF/R02		Y	ABS	RRS	Y	Y	Y	Y	Y	
0033	01	R24-S025	600/208V ESS DIV 1 MCC 1A	DIESEL	130	D/03		Y	ABS	RRS	Y	Y	N	Y	N	
0034	01	R24-S026	600/208V ESS DIV B MCC 1B	DIESEL	130	E/03		Y	ABS	RRS	Y	Y	N	Y	N	
0035	01	R24-S027	600/208V ESS DIV 2 MCC 1C	DIESEL	130	F/03		Y	ABS	RRS	Y	Y	N	Y	N	
1105	01	R24-S029	600V MCC 1E	CONTROL	180	TD/T11		N	ABS	RRS	Y	N	N	Y	N	
1112	01	R24-S031	600V MCC 1G	CONTROL	180	TB/T11		N	ABS	RRS	Y	N	N	Y	N	
0816	01	R24-S048	600/208V MCC 1D	DIESEL	130			Y	ABS	RRS	Y	Y	Y	Y	Y	
0036	14	R25-S001	125V DC DIV 1 CAB 1A	CONTROL	130	TE/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0037	14	R25-S002	125V DC DIV 2 CAB 1B	CONTROL	130	TE/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0038	14	R25-S004	125V DC CAB 1D	DIESEL	130	D/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0039	14	R25-S005	125V DC CAB 1E	DIESEL	130	E/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0040	14	R25-S006	125V DC CAB 1F	DIESEL	130	F/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0519	14	R25-S015	24/48 VDC CABINET 1A	CONTROL	130	TG/T12		Y	BS	GRS	Y	Y	Y	Y	Y	
0520	14	R25-S016	24/48 VDC CABINET 1B	CONTROL	130	TG/T12		Y	BS	GRS	Y	Y	Y	Y	Y	
0041	14	R25-S029	120/208V AC CAB 1J	DIESEL	130	D/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0042	14	R25-S030	120/208V AC CAB 1K	DIESEL	130	E/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0043	14	R25-S031	120/208V AC CAB 1L	DIESEL	130	F/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0862	14	R25-S035	120/208V AC CABINET	DIESEL	130			Y	BS	GRS	Y	Y	Y	Y	Y	
0044	14	R25-S036	120/208V AC ESS CAB 1A	CONTROL	130	TF/T10		Y	BS	GRS	Y	Y	Y	Y	Y	
0045	14	R25-S037	120/208V AC ESS CAB 1B	CONTROL	130	TD/T10		Y	BS	GRS	Y	Y	Y	Y	Y	
0047	14	R25-S064	120/208V AC VITAL CAB 1A INSTR BUS 1A	CONTROL	130	TG/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0048	14	R25-S065	120/208V AC CAB 1B INSTR BUS 1B	CONTROL	130	TG/T12		Y	BS	GRS	Y	Y	Y	Y	Y	
0049	14	R25-S105	125V DC CAB 1D ESS DIV I	CONTROL	130	TE/T11		Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter- act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0050	14	R25-S106	125V DC CAB 1E ESS DIV II	CONTROL	130	TB/T10		Y	BS	GRS	Y	Y	Y	Y	Y	
0051	18	R25-S110	120/208V CAB 1A (R25-S064) FUSE BOX	CONTROL	130			Y	NA	NA	NA	NA	Y	NA	Y	1
0052	18	R25-S111	120/208V CAB 1C (R25-S065) FUSE BOX	CONTROL	130			Y	NA	NA	NA	NA	Y	NA	Y	1
0053	18	R25-S112	120/208V MCC-1A (R24-S025) FUSE BOX	DIESEL	130	D/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0054	18	R25-S113	120/208V MCC-1B (R24-S026) FUSE BOX	DIESEL	130	E/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0055	18	R25-S114	120/208V MCC-1C (R24-S027) FUSE BOX	DIESEL	130	F/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0838	18	R26-M021	RPS DIST CAB THROWOVER SW	CONTROL	130	TG/T12		Y	NA	NA	NA	NA	Y	NA	Y	
0059	18	R26-M031A	125V DC 600A THROWOVER SWITCH 1A	CONTROL	130	TE/T11		Y	NA	NA	NA	NA	Y	NA	Y	1
0060	18	R26-M031B	125V DC 600A THROWOVER SWITCH 1B	CONTROL	130	TE/T11		Y	NA	NA	NA	NA	Y	NA	Y	1
0061	18	R26-M031C	125V DC 600A THROWOVER SWITCH 1C	CONTROL	130	TB/T11		Y	NA	NA	NA	NA	Y	NA	Y	1
0062	18	R26-M031D	125V DC 600A THROWOVER SWITCH 1D	CONTROL	130	TB/T11		Y	NA	NA	NA	NA	Y	NA	Y	1
0063	18	R26-M032A	125V DC THROWOVER SWITCH 1E	DIESEL	130	D/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0064	18	R26-M032B	125V DC THROWOVER SWITCH 1F	DIESEL	130	E/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0065	18	R26-M032C	125V DC THROWOVER SWITCH 1G	DIESEL	130	F/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0513	18	R26-M041A	24 VDC THROWOVER SW 1A	CONTROL	130	TG/T12		Y	NA	NA	NA	NA	Y	NA	Y	1
0514	18	R26-M041B	24 VDC THROWOVER SW 1B	CONTROL	130	TG/T12		Y	NA	NA	NA	NA	Y	NA	Y	1
0515	18	R26-M041C	24 VDC THROWOVER SW 1C	CONTROL	130	TG/T12		Y	NA	NA	NA	NA	Y	NA	Y	1
0516	18	R26-M041D	24 VDC THROWOVER SW 1D	CONTROL	130	TG/T12		Y	NA	NA	NA	NA	Y	NA	Y	1
0521	18	R26-M073	DISCONNECT SW FOR C11-F040A	CONTROL	130	TG/T12		Y	NA	NA	NA	NA	Y	NA	Y	1
0522	18	R26-M074	DISCONNECT SW FOR C11-F040B	CONTROL	138	TG/T12		Y	NA	NA	NA	NA	Y	NA	Y	1
0066	18	R26-M077	600V BREAKER	CONTROL	147	TH/T14		Y	NA	NA	NA	NA	Y	NA	Y	1
0067	18	R26-M078	600V BREAKER	CONTROL	147	TH/T14		Y	NA	NA	NA	NA	Y	NA	Y	1
0107	01	R27-S005	LOCAL STARTER FOR E11-F017A	REACTOR	130	RL/R09		Y	BS	GRS	Y	Y	Y	Y	Y	
0103	01	R27-S035	LOCAL STARTER FOR E41-F006	REACTOR	130	RH/R02		Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1RS.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter- act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0104	01	R27-S036	LOCAL STARTER FOR E41-F007	REACTOR	112	RH/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
0105	01	R27-S037	LOCAL STARTER FOR E41-F008	REACTOR	112	RH/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
0106	01	R27-S066	LOCAL STARTER FOR E41-F002	REACTOR	130	RL/R04		Y	BS	GRS	Y	Y	Y	Y	Y	
0820	0	R34-S004A	NEUTRAL RESISTOR 1A	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	
0821	0	R34-S004B	NEUTRAL RESISTOR 1B	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	
0822	0	R34-S004C	NEUTRAL RESISTOR 1C	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	
0791	0	R34-S005A	SURGE PROT PANEL FOR P41-C001A	INTAKE	111			Y	NA	NA	NA	NA	Y	Y	Y	1
0792	0	R34-S005B	SURGE PROT PANEL FOR P41-C001B	INTAKE	111			Y	NA	NA	NA	NA	Y	Y	Y	1
0793	0	R34-S006A	SURGE PROT PANEL FOR E11-C001A	INTAKE	111			Y	NA	NA	NA	NA	Y	Y	Y	1
0794	0	R34-S006D	SURGE PROT PANEL FOR E11-C002D	INTAKE	111			Y	NA	NA	NA	NA	Y	Y	Y	1
0068	15	R42-S001A	125/250V STATION BATTERY 1A	CONTROL	112	TE/T11		Y	BS	GRS	Y	Y	Y	U	N	
0069	15	R42-S001B	125/250V STATION BATTERY 1B	CONTROL	112	TD/T11		Y	BS	GRS	Y	Y	Y	U	N	
0070	15	R42-S002A	125V DIESEL SYSTEM BATTERY 1A	DIESEL	130	D/02		Y	BS	GRS	Y	Y	U	U	N	
0071	15	R42-S002B	125V DIESEL SYSTEM BATTERY 1B	DIESEL	130	E/02		Y	BS	GRS	Y	Y	U	U	N	
0072	15	R42-S002C	125V DIESEL SYSTEM BATTERY 1C	DIESEL	130	F/02		Y	BS	GRS	Y	Y	U	U	N	
0712	15	R42-S017A	BATTERY 1A 24/48 V	CONTROL	112	TC/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0713	15	R42-S017B	BATTERY 1B 24/48 V	CONTROL	112	TC/T12		Y	BS	GRS	Y	Y	Y	Y	Y	
0073	15	R42-S026	125V BATTERY CHARGER 1A	CONTROL	130	TE/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0074	16	R42-S027	125V BATTERY CHARGER 1B	CONTROL	130	TE/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0076	16	R42-S029	125V BATTERY CHARGER 1D	CONTROL	130	TB/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0077	16	R42-S030	125V BATTERY CHARGER 1E	CONTROL	130	TB/T11		Y	BS	GRS	Y	Y	Y	Y	Y	
0079	16	R42-S032A	125V BATTERY CHARGER 1G	DIESEL	130	D/02		Y	BS	GRS	Y	N	Y	Y	N	
0080	16	R42-S032B	125V BATTERY CHARGER 1H	DIESEL	130	E/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0081	16	R42-S032C	125V BATTERY CHARGER 1J	DIESEL	130	F/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0507	16	R42-S051	BATTERY CHARGER 1A	CONTROL	130	TG/T12		Y	BS	GRS	Y	Y	Y	Y	Y	
0509	16	R42-S052	BATTERY CHARGER 1B	CONTROL	130	TG/T12		Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0510	16	R42-S053	BATTERY CHARGER 1C	CONTROL	130	TG/T12		Y	BS	GRS	Y	Y	Y	Y	Y	
0512	16	R42-S054	BATTERY CHARGER 1D	CONTROL	130	TG/T12		Y	BS	GRS	Y	Y	Y	Y	Y	
0629	21	R43-A001A	FUEL DAY TANK 1A	DIESEL	130	D/01		Y	NA	NA	Y	NA	Y	NA	Y	1
0660	21	R43-A001B	FUEL DAY TANK 1B	DIESEL	130	E/01		Y	NA	NA	Y	NA	Y	NA	Y	1
0691	21	R43-A001C	FUEL DAY TANK 1C	DIESEL	130	F/01		Y	NA	NA	Y	NA	Y	NA	Y	1
0630	21	R43-A002A	FUEL STORAGE TANK 1A	YARD				Y	NA	NA	Y	NA	Y	NA	Y	
0661	21	R43-A002B	FUEL STORAGE TANK 1B	YARD				Y	NA	NA	Y	NA	Y	NA	Y	
0692	21	R43-A002C	FUEL STORAGE TANK 1C	YARD				Y	NA	NA	Y	NA	Y	NA	Y	
0627	21	R43-A003A	AIR RECEIVER	DIESEL	130	D/01		Y	NA	NA	Y	NA	Y	NA	Y	1
0658	21	R43-A003B	AIR RECEIVER	DIESEL	130	E/01		Y	NA	NA	Y	NA	Y	NA	Y	1
0689	21	R43-A003C	AIR RECEIVER	DIESEL	130	F/01		Y	NA	NA	Y	NA	Y	NA	Y	1
0628	21	R43-A007A	AIR RECEIVER	DIESEL	130	D/01		Y	NA	NA	Y	NA	Y	NA	Y	1
0659	21	R43-A007B	AIR RECEIVER	DIESEL	130	E/01		Y	NA	NA	Y	NA	Y	NA	Y	1
0690	21	R43-A007C	AIR RECEIVER	DIESEL	130	F/01		Y	NA	NA	Y	NA	Y	NA	Y	1
0622	20	R43-P001A	DSL GEN 1A CONT PNL	DIESEL	130	D/02		Y	BS	GRS	Y	Y	Y	U	N	
0653	20	R43-P001B	DSL GEN 1B CONT PNL	DIESEL	130	E/02		Y	BS	GRS	Y	Y	Y	U	N	
0684	20	R43-P001C	DSL GEN 1C CONT PNL	DIESEL	130	F/02		Y	BS	GRS	Y	Y	Y	U	N	
0099	17	R43-S001A	DIESEL GENERATOR 1A	DIESEL	130	C/02		Y	BS	GRS	Y	U	Y	Y	N	
0100	17	R43-S001B	DIESEL GENERATOR 1B	DIESEL	130	D/02		Y	BS	GRS	Y	U	Y	Y	N	
0101	17	R43-S001C	DIESEL GENERATOR 1C	DIESEL	130	E/02		Y	BS	GRS	Y	U	Y	Y	N	
0085	16	R44-S002	DC/AC INVERTER FOR MCC 1E-A	CONTROL	147	TI/T14		Y	BS	GRS	Y	Y	Y	Y	Y	
0086	16	R44-S003	DC/AC INVERTER FOR MCC 1E-B	CONTROL	147	TH/T14		Y	BS	GRS	Y	Y	Y	U	N	
0088	04	S11-S009	4160/600V STA SERV XFMR 1F1	DIESEL	130	E/03		Y	BS	GRS	Y	Y	Y	Y	Y	
0819	04	S11-S012	STA SERV XFMR 1F2 4160/600V	DIESEL	130			Y	ABS	RRS	Y	Y	N	Y	N	
0424	10	T41-B002A	RHR/CS PUMP ROOM COOLER	REACTOR	087	RL/R11		Y	BS	GRS	Y	Y	N	Y	N	
0437	10	T41-B003B	RHR/CS PUMP ROOM COOLER	REACTOR	087	RL/R03		Y	BS	GRS	Y	N	N	Y	N	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1RS.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0836	10	T41-B005B	HPCI PUMP ROOM COOLER	REACTOR	087	RL/R03		Y	BS	GRS	Y	Y	Y	Y	Y	
0456	21	T48-A001	UNIT 1 NITROGEN STORAGE TANK	YARD				Y	DOC	ARS	Y	NA	Y	NA	Y	
0539	18	T48-N010A	TORUS WATER LEVEL TRANS	REACTOR	087	RF/R13		Y	BS	GRS	Y	Y	Y	Y	Y	
0545	18	T48-N010B	TORUS WATER LEVEL TRANS	REACTOR	087	RF/R02		Y	BS	GRS	Y	Y	Y	Y	Y	
0517	18	TB1-139	TERMINAL BOX	CONTROL	130	TG/T12		Y	NA	NA	NA	NA	Y	NA	Y	1
0518	18	TB1-140	TERMINAL BOX	CONTROL	130	TG/T12		Y	NA	NA	NA	NA	Y	NA	Y	1
0089	18	TB1-211	125V DC BATTERY 1A FUSE BOX	DIESEL	130	D/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0090	18	TB1-212	125V DC BATTERY 1B FUSE BOX	DIESEL	130	E/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0091	18	TB1-213	125V DC BATTERY 1C FUSE BOX	DIESEL	130	F/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0716	18	TB1-229-7	TERMINAL BOX	DIESEL	130	D/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0717	18	TB1-230-8	TERMINAL BOX	DIESEL	130	E/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0718	18	TB1-231-8	TERMINAL BOX	DIESEL	130	F/02		Y	NA	NA	NA	NA	Y	NA	Y	1
0827	09	X41-C002A	DG ROOM 1A FAN	DIESEL	150			Y	BS	GRS	Y	Y	Y	Y	Y	
0828	09	X41-C002C	DG ROOM 1B FAN	DIESEL	150			Y	BS	GRS	Y	Y	Y	Y	Y	
0829	09	X41-C002E	DG ROOM 1C FAN	DIESEL	150			Y	BS	GRS	Y	Y	Y	Y	Y	
0860	0	X41-C005A	DG ROOM 1A LOUVER	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	
0831	0	X41-C005B	DG ROOM 1B LOUVER	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	
0832	0	X41-C005C	DG ROOM 1C LOUVER	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	
0824	09	X41-C006A	DG SWITCHGEAR ROOM 1E FAN	DIESEL	150			Y	BS	GRS	Y	Y	Y	Y	Y	
0825	09	X41-C006C	DG SWITCHGEAR ROOM 1F FAN	DIESEL	150			Y	BS	GRS	Y	Y	Y	Y	Y	
0826	09	X41-C006E	DG SWITCHGEAR ROOM 1G FAN	DIESEL	150			Y	BS	GRS	Y	Y	Y	Y	Y	
0833	0	X41-C007A	DG SWITCHGEAR ROOM 1E LOUVER	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	
0834	0	X41-C007B	DG SWITCHGEAR ROOM 1F LOUVER	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	
0835	0	X41-C007C	DG SWITCHGEAR ROOM 1G LOUVER	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	
1089	09	X41-C008A	BATTERY ROOM 1A FAN	DIESEL	150			Y	BS	GRS	Y	Y	Y	Y	Y	
1085	09	X41-C008C	BATTERY ROOM 1B FAN	DIESEL	150			Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
1086	09	X41-C008E	BATTERY ROOM 1C FAN	DIESEL	150			Y	BS	GRS	Y	Y	Y	Y	Y	
0985	09	X41-C009A	INTAKE STRUCTURE VENT FAN 1A	INTAKE	150			Y	BS	GRS	Y	Y	Y	Y	Y	
0977	09	X41-C009B	INTAKE STRUCTURE VENT FAN 1B	INTAKE	150			Y	BS	GRS	Y	Y	Y	Y	Y	
0978	09	X41-C009C	INTAKE STRUCTURE VENT FAN 1C	INTAKE	150			Y	BS	GRS	Y	Y	Y	Y	Y	
0882	0	X41-C017A	DG ROOM 1A ROLL-UP FIRE DOOR	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	2
0883	0	X41-C017B	DG ROOM 1B ROLL-UP FIRE DOOR	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	2
0884	0	X41-C017C	DG ROOM 1C ROLL-UP FIRE DOOR	DIESEL	130			Y	NA	NA	Y	NA	Y	Y	Y	2
1090	0	X41-C027A	BATTERY ROOM 1A LOUVER	DIESEL	130	C/01		Y	NA	NA	Y	NA	Y	Y	Y	
1087	0	X41-C027B	BATTERY ROOM 1B LOUVER	DIESEL	130	D/01		Y	NA	NA	Y	NA	Y	Y	Y	
1088	0	X41-C027C	BATTERY ROOM 1C LOUVER	DIESEL	130	E/01		Y	NA	NA	Y	NA	Y	Y	Y	
0987	18	X41-N002A	INTK STRUC VENT FAN 1A THERMOSTAT	INTAKE	111	PUMP RM		Y	BS	GRS	Y	Y	Y	Y	Y	
0979	18	X41-N002B	INTK STRUC VENT FAN 1B THERMOSTAT	INTAKE	111	PUMP RM		Y	BS	GRS	Y	Y	Y	Y	Y	
0980	18	X41-N002C	INTK STRUC VENT FAN 1C THERMOSTAT	INTAKE	111	PUMP RM		Y	BS	GRS	Y	Y	Y	Y	Y	
0885	18	X41-N004A	D/G RM 1A FAN THERMOSTAT	DIESEL	130	C/01		Y	BS	GRS	Y	Y	Y	Y	Y	
0886	18	X41-N004B	D/G RM 1B FAN THERMOSTAT	DIESEL	130	D/01		Y	BS	GRS	Y	Y	Y	Y	Y	
0887	18	X41-N004C	D/G RM 1C FAN THERMOSTAT	DIESEL	130	E/01		Y	BS	GRS	Y	Y	Y	Y	Y	
0888	18	X41-N005A	D/G RM 1A FAN THERMOSTAT	DIESEL	130	D/02		Y	BS	GRS	Y	Y	Y	Y	Y	
0889	18	X41-N005B	D/G RM 1B FAN THERMOSTAT	DIESEL	130	E/01		Y	BS	GRS	Y	Y	Y	Y	Y	
0890	18	X41-N005C	D/G RM 1C FAN THERMOSTAT	DIESEL	130	F/02		Y	BS	GRS	Y	Y	Y	Y	Y	
1109	18	X41-N006A	SWGR RM 1E FAN THERMOSTAT	DIESEL	130	C/03		Y	BS	GRS	Y	Y	Y	Y	Y	
1110	18	X41-N006B	SWGR RM 1F FAN THERMOSTAT	DIESEL	130	D/03		Y	BS	GRS	Y	Y	Y	Y	Y	
1111	18	X41-N006C	SWGR RM 1G FAN THERMOSTAT	DIESEL	130	E/03		Y	BS	GRS	Y	Y	Y	Y	Y	
1106	20	X43-P006A	PNEU-ELECTRO RELAY CAB	DIESEL	130	C/01		Y	BS	GRS	Y	Y	Y	Y	Y	
1107	20	X43-P006B	PNEU-ELECTRO RELAY CAB	DIESEL	130	D/01		Y	BS	GRS	Y	Y	Y	N	N	
1108	20	X43-P006C	PNEU-ELECTRO RELAY CAB	DIESEL	130	E/01		Y	BS	GRS	Y	Y	Y	Y	Y	
0625	06	Y52-C001A	DSL 1A FUEL OIL PUMP 1A1	YARD				Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO. (1)	EQUIP CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	<--------- Building (5)	EQUIPMENT Flr.Elv. (6)	LOCATION -----> Rm. or Row/Col. (7)	Base Elev. (8)	<40'? (9)	Capacity Spectrum (10)	Demand Spectrum (11)	Cap. > Demand? (12)	Caveats OK? (13)	Anchor OK? (14)	Inter-act OK? (15)	Equip OK? (16)	Notes (17)
0656	06	Y52-C101B	DSL 1B FUEL OIL PUMP 1B2	YARD				Y	BS	GRS	Y	Y	Y	Y	Y	
0688	06	Y52-C101C	DSL 1C FUEL OIL PUMP 1C2	YARD				Y	BS	GRS	Y	Y	Y	Y	Y	
0898	10	Z41-B003A	CONTROL ROOM AIR HANDLING UNIT	CONTROL	180	TD/T12		N	ABS	RRS	Y	N	N	Y	N	
0939	10	Z41-B003B	CONTROL ROOM AIR HANDLING UNIT	CONTROL	180	TE/T12		N	ABS	RRS	Y	N	N	Y	N	
0901	10	Z41-B003C	CONTROL ROOM AIR HANDLING UNIT	CONTROL	180	TF/T12		N	ABS	RRS	Y	N	N	Y	N	
0899	11	Z41-B008A	B003A CONDENSING UNIT	CONTROL	180	TB/T11		N	ABS	RRS	Y	N	N	Y	N	
0940	11	Z41-B008B	B003B CONDENSING UNIT	CONTROL	180	TB/T11		N	ABS	RRS	Y	N	N	N	N	
0937	11	Z41-B008C	B003C CONDENSING UNIT	CONTROL	180	TB/T12		N	ABS	RRS	Y	N	N	Y	N	
0903	09	Z41-C012A	D004A BOOSTER FAN	CONTROL	180	TF/T13		N	ABS	RRS	Y	Y	Y	Y	Y	
0941	09	Z41-C012B	D004B BOOSTER FAN	CONTROL	180	TG/T12		N	ABS	RRS	Y	Y	Y	Y	Y	
1097	09	Z41-C014	BATT ROOM EMERGENCY EXH	CONTROL	112	TD/T12		Y	BS	GRS	Y	Y	N	Y	N	
1091	09	Z41-C015	BATT ROOM EMERGENCY EXH	CONTROL	112	TD/T11		Y	BS	GRS	Y	Y	N	Y	N	
0900	0	Z41-D004A	CONTROL ROOM FILTER TRAIN	CONTROL	180	TG/T13		N	NA	NA	Y	NA	Y	Y	Y	
0938	0	Z41-D004B	CONTROL ROOM FILTER TRAIN	CONTROL	180	TG/T13		N	N/A	N/A	Y	N/A	Y	Y	Y	
0904	07	Z41-F007A	AIR OPERATED DAMPER B003A IN	CONTROL	180	TD/T12		N	ABS	RRS	Y	Y	NA	Y	Y	
0905	07	Z41-F007C	AIR OPERATED DAMPER B003C IN	CONTROL	180	TF/T12		N	ABS	RRS	Y	Y	NA	Y	Y	
0943	07	Z41-F008B	AIR OPERATED DAMPER B003B IN	CONTROL	180	TE/T12		N	ABS	RRS	Y	Y	NA	Y	Y	
0944	07	Z41-F008C	AIR OPERATED DAMPER B003C IN	CONTROL	180	TF/T12		N	ABS	RRS	Y	Y	NA	Y	Y	
0945	07	Z41-F009B	AIR OPERATED DAMPER B003B IN	CONTROL	180	TE/T12		N	ABS	RRS	Y	N	NA	Y	N	
0947	07	Z41-F010A	AIR OPERATED DAMPER B003B IN	CONTROL	180	TE/T04		N	ABS	RRS	Y	Y	NA	Y	Y	
0909	07	Z41-F011	AIR OPER DMPR D004A/B BYPASS	CONTROL	180	TG/T12		N	ABS	RRS	Y	Y	NA	Y	Y	
0949	07	Z41-F012	AIR OPER DMPR D004A/B BYPASS	CONTROL	180	TF/T12		N	ABS	RRS	Y	Y	NA	Y	Y	
0910	07	Z41-F013A	AIR OPERATED DAMPER D004A IN	CONTROL	180	TH/T13		N	ABS	RRS	Y	Y	NA	Y	Y	
0957	07	Z41-F013B	AIR OPERATED DAMPER D004B IN	CONTROL	180	TH/T12		N	ABS	RRS	Y	Y	NA	Y	Y	
0912	07	Z41-F014A	AIR OPERATED DAMPER D004A IN	CONTROL	180	TH/T12		N	ABS	RRS	Y	Y	NA	Y	Y	
0953	07	Z41-F014B	AIR OPERATED DAMPER D004B IN	CONTROL	180	TH/T12		N	ABS	RRS	Y	Y	NA	Y	Y	

APPENDIX C
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1RS.DBF / 10/20/95 / 07:17:44
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM 2.2

LINE NO. (1)	EQUIP CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	<---------- Building (5)	EQUIPMENT Flr.Elv. (6)	LOCATION -----> Rm. or Row/Col. (7)	Base Elev. (8)	<40'? (9)	Capacity Spectrum (10)	Demand Spectrum (11)	Cap. > Demand? (12)	Caveats OK? (13)	Anchor OK? (14)	Inter-act OK? (15)	Equip OK? (16)	Notes (17)
0914	07	Z41-F015	AIR OPERATED DAMPER D003 BYPASS	CONTROL	180	TH/T12		N	ABS	RRS	Y	Y	NA	Y	Y	
0956	07	Z41-F018A	AIR OPERATED DAMPER C011A IN	CONTROL	180	TE/T12		N	ABS	RRS	Y	Y	NA	Y	Y	
0951	07	Z41-F018B	AIR OPERATED DAMPER C011B IN	CONTROL	180	TE/T12		N	ABS	RRS	Y	Y	NA	Y	Y	
0918	07	Z41-F019	AIR OPERATED DAMPER RESTROOM	CONTROL	180	TH/T13		N	ABS	RRS	Y	Y	NA	Y	Y	
0919	07	Z41-F020	AIR OPERATED DAMPER RESTROOM	CONTROL	180	TH/T13		N	ABS	RRS	Y	Y	NA	Y	Y	
0960	07	Z41-F028A	AIR OPERATED DAMPER BG10B OUT	CONTROL	180	TE/T13		N	ABS	RRS	Y	Y	NA	Y	Y	
1099	0	Z41-FD-F004	FIRE DMPR STN BATTERY 1A	CONTROL	112	TE/T11		Y	NA	NA	Y	NA	Y	Y	Y	2
1093	0	Z41-FD-F005	FIRE DMPR STN BATTERY 1B	CONTROL	112	TD/T11		Y	NA	NA	Y	NA	Y	Y	Y	2
1092	0	Z41-FD-F006	FIRE DMPR STN BATTERY 1B	CONTROL	112	TC/T11		Y	NA	NA	Y	NA	Y	Y	Y	2
1098	0	Z41-FD-F020	FIRE DMPR STN BATTERY 1A	CONTROL	112	TF/T11		Y	NA	NA	Y	NA	Y	Y	Y	2
0925	18	Z41-N003A	B003A DISCHARGE FS	CONTROL	180	TE/T13		N	ABS	RRS	Y	Y	Y	Y	Y	
0963	18	Z41-N003B	B003B DISCHARGE FS	CONTROL	180	TE/T13		N	ABS	RRS	Y	Y	Y	Y	Y	
0966	18	Z41-N003C	B003C DISCHARGE FS	CONTROL	180	TF/T13		N	ABS	RRS	Y	Y	Y	Y	Y	
0923	18	Z41-N005A	C012A DISCHARGE FS	CONTROL	180	TF/T13		N	ABS	RRS	Y	Y	Y	Y	Y	
0964	18	Z41-N005B	C012B DISCHARGE FS	CONTROL	180	TF/T12		N	ABS	RRS	Y	Y	Y	Y	Y	
0924	18	Z41-N015A	CONTROL RM OUTSIDE AIR INLET RE	CONTROL	180	TH/T13		N	ABS	RRS	Y	Y	Y	Y	Y	
0965	18	Z41-N015B	CONTROL RM OUTSIDE AIR INLET RE	CONTROL	180	TH/T13		N	ABS	RRS	Y	Y	Y	Y	Y	
0926	18	Z41-N600A	B003A COMPRESSOR TIS	CONTROL	180	TD/T12		N	ABS	RRS	Y	Y	Y	Y	Y	
0968	18	Z41-N600B	B003B COMPRESSOR TIS	CONTROL	180	TE/T12		N	ABS	RRS	Y	Y	Y	Y	Y	
0967	18	Z41-N600C	B003C COMPRESSOR TIS	CONTROL	180	TF/T12		N	ABS	RRS	Y	Y	Y	Y	Y	

APPENDIX D

SCREENING VERIFICATION DATA SHEETS
PLANT HATCH UNIT 2

The Screening Verification Data Sheets (SVDSs) constitute the formal documentation of the screening verification and walkdown for the components contained in the Seismic Review Safe Shutdown Equipment List (SSEL) (Appendix B), and reflect the final judgment of the seismic capability engineers.

The format of the SVDS is provided by the Safe Shutdown Equipment Manager computer program supplied by the Seismic Qualification Utility Group (SQUG). The information on the SVDS is sorted by the component mark number (column 3). The SVDS includes the following information, where applicable and available, for each component.

Column Number	Description
1	Unique line number
2	Equipment class (see table 1 of this appendix)
3	Equipment identification number
4	Equipment description
5	Building or area in which equipment is located
6	Floor elevation from which the equipment can be seen
7	Building grid row/column indicating equipment location
8	Elevation at which the equipment is mounted
9	Indicates whether the equipment is mounted lower in the building than approximately 40 feet above grade: Y - Yes N - No

Column Number	Description
10	Indicates the source of the seismic capacity:
	BS SQUG bounding spectrum
	ABS 1.5 x SQUG bounding spectrum
	N/A Not applicable
11	Indicates the method used to define the seismic demand:
	GRS Ground response spectra
	AGS 1.5 x ground response spectra
	RRS Realistic, median-centered in-structure response spectra
	N/A Not applicable
12	Indicates whether the capacity of the equipment exceeds the demand:
	Y Yes - Capacity exceeds demand
	N No - Capacity does not exceed demand
	U Unknown whether capacity exceeds demand
13	Indicates whether the equipment is within the scope of the earthquake/testing equipment class and meets the intent of all the caveats for the equipment class:
	Y Yes - The equipment is in the equipment class, and the intent of all applicable caveats is satisfied.
	N No - The equipment is not in the equipment class, or the intent of all applicable caveats is not satisfied.
	U Unknown whether the equipment is in the equipment class or whether the intent of all applicable caveats is satisfied.
	N/A The earthquake/test equipment class and the caveats are not applicable to this equipment item.

Column Number	Description
14	Indicates whether the equipment anchorage meets the anchorage screening guidelines:

Y — Yes - Anchorage capacities equal or exceed seismic demand, and anchorage is free of gross installation defects and has adequate stiffness.

N — No - Anchorage capacities do not equal or exceed the seismic demand, anchorage is not without gross installation defects, or anchorage does not have adequate stiffness.

U — Unknown whether anchorage capacities equal or exceed seismic demand, or whether anchorage is free of gross installation defects or has adequate stiffness.

N/A — Anchorage guidelines are not applicable to this equipment (e.g., valves are not evaluated for anchorage).

15	Indicates whether the equipment is free of adverse seismic interaction effects:

Y — Yes - The equipment is free of interaction effects, or the interaction effects are acceptable and do not compromise the safe shutdown function of the equipment.

N — No - The equipment is not free of adverse interaction effects.

U — Unknown whether interaction effects will compromise the safe shutdown function of the equipment.

16	Indicates whether, in the final judgment of the seismic capability engineers, the seismic adequacy of the equipment is verified:

Y — Yes - Seismic adequacy has been verified; i.e., code "Y" for all applicable screening guidelines:

1) Seismic capacity is greater than demand.

2) The equipment is in the earthquake/test equipment class and the intent of all the caveats is met (for use with bounding spectrum or GERS only).

Column Number	Description	
16 (continued)		3) Equipment anchorage is adequate.
		4) Seismic interaction effects will not compromise the safe shutdown function of the item of equipment.
	N	No - Seismic adequacy does not meet one or more of the seismic evaluation criteria. Equipment is identified as an outlier requiring further verification effort in accordance with section 5 of the SQUG Generic Implementation Procedure.
17	Explanatory notes:	
	1)	Passive electrical component which requires anchorage evaluation only.
	2)	Passive fire damper requires evaluation of fusible link only.
	3)	Non-USI A-46 component required for Individual Plant Examination of External Events only.

APPENDIX D

TABLE 1
EQUIPMENT CLASS DESIGNATIONS

Equipment Class (Column 2)	Description
01	Motor control centers
02	Low-voltage switchgear
03	Medium-voltage switchgear
04	Transformers
05	Horizontal pumps
06	Vertical pumps
07	Fluid-operated valves
08A	Motor-operated valves
08B	Solenoid-operated valves
09	Fans
10	Air handlers
11	Chillers
12	Air compressors
13	Motor generators
14	Distribution panels
15	Batteries on racks
16	Battery chargers and inverters
17	Engine generators
18	Instruments on racks
19	Temperature sensors
20	Instrumentation and control panels and cabinets
21	Tanks and heat exchangers

EDWIN I. HATCH NUCLEAR PLANT - UNIT 2

SVDS CERTIFICATION

All the information contained on this Screening Verification Data Sheet (SVDS) is, to the best of our knowledge and belief, correct and accurate. "All information" includes each entry and conclusion (whether verified to be seismically adequate or not) for components evaluated by the Seismic Review Team (SRT) as documented in the Seismic Evaluation Work Sheets (SEWS).

Name / Title	Signature / Date
P. W. Garrett / Seismic Capability Engineer	P.W. Garrett BY HSU / 11-9-94
B. R. Goforth / Seismic Capability Engineer	Billy R Goforth Jr. / 11-9-94
P. Hsu / Seismic Capability Engineer	Ping Hsu
T. B. Lantrip / Seismic Capability Engineer	T.B. Lantrip / 11/9/94
D. P. Moore / Seismic Capability Engineer	Donald P. Moore 11/10/94
C. K. Toner / Seismic Capability Engineer	Carl K. Toner / 11-9-94
K. D. Wooten / Seismic Capability Engineer	Keith D. Wooten / 11-9-94

The information provided to the Seismic Capability Engineers regarding systems and operations of the equipment contained on this SVDS is, to the best of our knowledge and belief, correct and accurate.

Name / Title	Signature / Date
W. S. Walker / Mechanical Systems Engineer	William Scott Walker / 11-8-94
J. E. Smith / Electrical Systems Engineer	James E Smith

Report Date/Time: 06-12-95 / 16:06:04

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO. (1)	EQUIP CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	<---------- Building (5)	EQUIPMENT Flr.Elv. (6)	LOCATION -----> Rm. or Row/Col. (7)	Base Elev. (8)	<40'? (9)	Capacity Spectrum (10)	Demand Spectrum (11)	Cap. > Demand? (12)	Caveats OK? (13)	Anchor OK? (14)	Inter-act OK? (15)	Equip OK? (16)	Notes (17)
000278	0	2B21-A003A	SRV AIR ACCUMULATOR	DRYWELL	148	AZ220	156	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000279	0	2B21-A003B	SRV AIR ACCUMULATOR	DRYWELL	148	AZ220	156	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000265	0	2B21-A003D	SRV AIR ACCUMULATOR	DRYWELL	148	AZ133	156	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000280	0	2B21-A003F	SRV AIR ACCUMULATOR	DRYWELL	148	AZ274	156	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000266	0	2B21-A003G	SRV AIR ACCUMULATOR	DRYWELL	148	AZ080	156	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000267	0	2B21-A003H	SRV AIR ACCUMULATOR	DRYWELL	148	AZ118	156	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000281	0	2B21-A003K	SRV AIR ACCUMULATOR	DRYWELL	148	AZ251	153	Y	N/A	N/A	Y	Y	Y	Y	Y	
000268	0	2B21-A003M	SRV AIR ACCUMULATOR	DRYWELL	148	AZ078	156	Y	N/A	N/A	Y	Y	Y	Y	Y	
000008	07	2B21-F013A	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ235	153	Y	BS	GRS	Y	Y	N/A	Y	Y	
000135	07	2B21-F013B	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ235	153	Y	BS	GRS	Y	Y	N/A	Y	Y	
000006	07	2B21-F013D	RPV SAFETY/RELIEF VALVE	DRYWELL	148	[illegible]	153	Y	BS	GRS	Y	Y	N/A	Y	Y	
000136	07	2B21-F013F	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ260	153	Y	BS	GRS	Y	Y	N/A	Y	Y	
000007	07	2B21-F013G	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ095	153	Y	BS	GRS	Y	Y	N/A	Y	Y	
000137	07	2B21-F013H	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ135	153	Y	BS	GRS	Y	Y	N/A	Y	Y	
000009	07	2B21-F013K	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ260	153	Y	BS	GRS	Y	Y	N/A	Y	Y	
000138	07	2B21-F013M	RPV SAFETY/RELIEF VALVE	DRYWELL	148	AZ090	153	Y	BS	GRS	Y	Y	N/A	Y	Y	
000129	07	2B21-F022A	INBOARD MSIV	DRYWELL	130	AZ190	140	Y	BS	GRS	Y	Y	N/A	Y	Y	
000130	07	2B21-F022B	INBOARD MSIV	DRYWELL	130	AZ220	136	Y	BS	GRS	Y	Y	N/A	Y	Y	
000131	07	2B21-F022C	INBOARD MSIV	DRYWELL	130	AZ160	140	Y	BS	GRS	Y	Y	N/A	Y	Y	
000132	07	2B21-F022D	INBOARD MSIV	DRYWELL	130	AZ175	140	Y	BS	GRS	Y	Y	N/A	Y	Y	
000001	07	2B21-F028A	OUTBOARD MSIV	REACTOR	130	RB/R19	140	Y	BS	GRS	Y	Y	N/A	Y	Y	
000002	07	2B21-F028B	OUTBOARD MSIV	REACTOR	130	RB/R18	140	Y	BS	GRS	Y	Y	N/A	Y	Y	
000003	07	2B21-F028C	OUTBOARD MSIV	REACTOR	130	RB/R20	140	Y	BS	GRS	Y	Y	N/A	Y	Y	
000004	07	2B21-F028D	OUTBOARD MSIV	REACTOR	130	RB/R19	140	Y	BS	GRS	Y	Y	N/A	Y	Y	
000767	18	2B21-N093B	RPV LEVEL 8 LT	REACTOR	158	RH/R17	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000179	08A	2B31-F023A	RECIRC PUMP SUCTION ISOLATION	DRYWELL	087	AZ3[illegible]	114	Y	BS	GRS	Y	Y	N/A	Y	Y	

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
000740	08B	2C11-D001-117	PILOT SCRAM SOLENOID	REACTOR	130	HCU	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000741	08B	2C11-D001-118	PILOT SCRAM SOLENOID	REACTOR	130	HCU	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000746	0	2C11-D001-125	SCRAM ACCUMULATOR	REACTOR	130	HCU	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000747	07	2C11-D001-126	SCRAM INLET VALVE	REACTOR	130	HCU	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000748	07	2C11-D001-127	SCRAM INLET VALVE	REACTOR	130	HCU	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000226	08B	2C11-F009	PILOT AIR HEADER SUPPLY	REACTOR	130	RA/R21	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000141	07	2C11-F010A	SCRAM DISCH VOL ISOLATION	REACTOR	158		158	Y	BS	GRS	Y	Y	N/A	Y	Y	
000142	07	2C11-F010B	SCRAM DISCH VOL ISOLATION	REACTOR	158		158	Y	BS	GRS	Y	Y	N/A	Y	Y	
000143	07	2C11-F011	SCRAM DISCH VOL DRAIN	REACTOR	087	RB/R17	120	Y	BS	GRS	Y	Y	N/A	Y	Y	
000145	07	2C11-F035A	SCRAM DISCH VOL ISOLATION	REACTOR	130	RF/R15	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000146	07	2C11-F035B	SCRAM DISCH VOL ISOLATION	REACTOR	130	RF/R24	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000147	07	2C11-F037	SCRAM DISCH VOL DRAIN	REACTOR	087	RB/R17	120	Y	BS	GRS	Y	Y	N/A	Y	Y	
000227	08B	2C11-F040	PILOT AIR HEADER SUPPLY	REACTOR	130	RA/R17	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000148	08B	2C11-F110A	BACKUP SCRAM VALVE	REACTOR	130	RA/R21	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000144	08B	2C11-F110B	BACKUP SCRAM VALVE	REACTOR	130	RA/R21	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000768	18	2C71-N050C	DRYWELL PRESSURE PT	REACTOR	158	RE/R16	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000769	18	2C71-N050D	DRYWELL PRESSURE PT	REACTOR	158	RE/R21	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000725	20	2C82-P001	REMOTE SHUTDOWN PANEL	REACTOR	130	RA/R16	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000172	21	2E11-B001A	RHR HEAT EXCHANGER A	REACTOR	087	RL/R14	108	Y	N/A	N/A	N/A	N/A	Y	Y	Y	
000095	21	2E11-B001B	RHR HEAT EXCHANGER B	REACTOR	087	RL/R24	108	Y	N/A	N/A	N/A	N/A	Y	Y	Y	
000208	06	2E11-C001A	RHRSW PUMP 2A	INTAKE	111		111	Y	BS	GRS	Y	Y	Y	Y	Y	
000096	06	2E11-C001D	RHRSW PUMP 2D	INTAKE	111		111	Y	BS	GRS	Y	Y	Y	Y	Y	
000160	06	2E11-C002A	RHR PUMP 2A	REACTOR	087	RL/R14	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000097	06	2E11-C002B	RHR PUMP 2B	REACTOR	087	RL/R24	087	Y	BS	GRS	Y	Y	Y	N	N	
000161	08A	2E11-F007A	RHR PUMP MIN FLOW BYPASS	REACTOR	087	RL/R14	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000101	08A	2E11-F007B	RHR PUMP 2D MIN FLOW BYPASS VL	REACTOR	087	RL/R24	100	Y	BS	GRS	Y	Y	N/A	Y	Y	

Report Date/Time: 06-12-95 / 16:06:04

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO. (1)	EQUIP CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	<---------- Building (5)	EQUIPMENT Flr.Elv. (6)	LOCATION -----> Rm. or Row/Col. (7)	Base Elev. (8)	<40'? (9)	Capacity Spectrum (10)	Demand Spectrum (11)	Cap. > Demand? (12)	Caveats OK? (13)	Anchor OK? (14)	Inter-act OK? (15)	Equip OK? (16)	Notes (17)
000171	08A	2E11-F015A	INBOARD INJECTION	REACTOR	130	RJ/R18	140	Y	BS	GRS	Y	Y	N/A	Y	Y	
000340	08A	2E11-F016A	CONTAINMENT SPRAY OUTBOARD	REACTOR	130	RJ/R21	135	Y	BS	GRS	Y	Y	N/A	Y	Y	3
000376	08A	2E11-F016B	CONT SPRAY DISCHARGE VALVE	REACTOR	158	RF/R23	165	Y	BS	GRS	Y	Y	N/A	Y	Y	3
000170	08A	2E11-F017A	RHR LPCI DISCHARGE VALVE	REACTOR	130	RJ/R18	140	Y	BS	GRS	Y	Y	N/A	Y	Y	
000367	08A	2E11-F021A	CONTAINMENT SPRAY INBOARD	REACTOR	130	RJ/R21	145	Y	BS	GRS	Y	Y	N/A	Y	Y	3
000406	08A	2E11-F021B	CONTAINMENT SPRAY INBOARD	REACTOR	158	RH/R23	165	Y	BS	GRS	Y	Y	N/A	Y	Y	3
000105	08A	2E11-F024B	RHR TEST LINE VALVE	REACTOR	087	RF/R24	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000365	08A	2E11-F027A	TORUS SPRAY INBOARD ISOLATION	REACTOR	087	RF/R14	120	Y	BS	GRS	Y	Y	N/A	Y	Y	3
000379	08A	2E11-F027B	SUPP POOL SPRAY VALVE	REACTOR	087	RF/R24	120	Y	BS	GRS	Y	Y	N/A	Y	Y	3
000339	08A	2E11-F028A	TORUS SPRAY OUTBOARD ISOLATION	REACTOR	087	RH/R14	120	Y	BS	GRS	Y	N	N/A	Y	N	3
000108	08A	2E11-F028B	RHR INLET TO SUPP POOL	REACTOR	087	RH/R24	130	Y	BS	GRS	Y	N	N/A	Y	N	
000165	08A	2E11-F048A	RHR HX BYPASS	REACTOR	087	RL/R14	120	Y	BS	GRS	Y	Y	N/A	Y	Y	
000110	08A	2E11-F048B	RHR HX B BYPASS VALVE	REACTOR	087	RL/R24	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000210	08A	2E11-F068A	RHRSW HX FLOW CONTROL	REACTOR	087	RL/R14	087	Y	BS	GRS	Y	Y	N/A	Y	Y	
000113	08A	2E11-F068B	RHR HX B TUBE TO SHELL OUTLET	REACTOR	087	RH/R24	110	Y	BS	GRS	Y	Y	N/A	Y	Y	
000216	18	2E11-N017A	RHRSW HX INLET PS	REACTOR	087	RH/R14	120	Y	BS	GRS	Y	Y	Y	Y	Y	
000122	18	2E11-N017B	RHR HX B INLET PRESS SW	REACTOR	087	RH/R24	110	Y	BS	GRS	Y	Y	Y	Y	Y	
000217	18	2E11-N017C	RHRSW HX INLET PS	REACTOR	087	RH/R14	120	Y	BS	GRS	Y	Y	Y	Y	Y	
000123	18	2E11-N017D	RHR HX B INLET PRESS SW	REACTOR	087	RH/R24	110	Y	BS	GRS	Y	Y	Y	Y	Y	
000770	18	2E11-N094A	DRYWELL PRESSURE PT	REACTOR	158	RE/R16	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000771	18	2E11-N094B	DRYWELL PRESSURE PT	REACTOR	158	RH/R21	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000772	18	2E11-N094C	DRYWELL PRESSURE PT	REACTOR	158	RE/R16	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000773	18	2E11-N094D	DRYWELL PRESSURE PT	REACTOR	158	RH/R21	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000083	06	2E21-C001B	CORE SPRAY PUMP 2B	REACTOR	087	RL/R24	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000086	08A	2E21-F005B	CORE SPRAY TO RVP ISOL VLV	REACTOR	158	RF/R21	165	Y	BS	GRS	Y	Y	N/A	Y	Y	
000089	08A	2E21-F031B	CORE SPRAY MIN FLOW BYPASS	REACTOR	087	RL/R24	092	Y	BS	GRS	Y	Y	N/A	Y	Y	

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
000015	05	2E41-C001	HPCI PUMP	REACTOR	087	RL/R25	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000016	05	2E41-C002	HPCI TURBINE	REACTOR	087	RG/R24	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000018	08A	2E41-F001	HPCI TURBINE STEAM SUPPLY VLV	REACTOR	087	RG/R25	096	Y	BS	GRS	Y	Y	N/A	Y	Y	
000021	08A	2E41-F004	HPCI PUMP SUCTION FROM CST	REACTOR	087	RL/R25	097	Y	BS	GRS	Y	Y	N/A	Y	Y	
000022	08A	2E41-F006	HPCI PUMP INBOARD DISCH. VALVE	REACTOR	087	RB/R13	120	Y	BS	GRS	Y	Y	N/A	Y	Y	
000025	08A	2E41-F012	HPCI MINIMUM FLOW BYPASS VALVE	REACTOR	087	RG/R24	090	Y	BS	GRS	Y	Y	N/A	Y	Y	
000026	08A	2E41-F041	HPCI PUMP SUCTION - SUPP POOL	REACTOR	087	RL/R25	096	Y	BS	GRS	Y	Y	N/A	Y	Y	
000027	08A	2E41-F042	HPCI PUMP SUCT FROM SUPP POOL	REACTOR	087	RG/R24	090	Y	BS	GRS	Y	Y	N/A	Y	Y	
000029	08A	2E41-F059	HPCI BAR COND COOLING WTR VLV	REACTOR	087	RG/R24	090	Y	BS	GRS	Y	Y	N/A	Y	Y	
000057	18	2E41-N062B	SUPPRESSION POOL LEVEL TRANS.	REACTOR	087	RG/R24	120	Y	BS	GRS	Y	Y	Y	Y	Y	
000058	18	2E41-N062D	SUPPRESSION POOL LEVEL TRANS.	REACTOR	087	RG/R24	118	Y	BS	GRS	Y	Y	Y	Y	Y	
000539	08B	2G11-F003	DRYWELL FL DR PMP ISOL VALVE	REACTOR	087	RE/R19	120	Y	N/A	N/A	Y	Y	N/A	Y	Y	3
000540	08B	2G11-F004	DRYWELL FL DR PMP ISOL VALVE	REACTOR	087	RE/R19	120	Y	N/A	N/A	Y	Y	N/A	Y	Y	3
000541	08B	2G11-F019	DRYWELL EQ DR PMP ISOL VALVE	REACTOR	087	RF/R21	120	Y	N/A	N/A	Y	Y	N/A	Y	Y	3
000542	08B	2G11-F020	DRYWELL EQ DR PMP ISOL VALVE	REACTOR	087	RF/R21	120	Y	N/A	N/A	Y	Y	N/A	Y	Y	3
000543	08A	2G31-F001	RWCU INBOARD ISOL GATE VALVE	DRYWELL	165	AZ020	165	Y	N/A	N/A	Y	Y	N/A	Y	Y	3
000544	08A	2G31-F004	RWCU OUTBOARD ISOL GATE VALVE	REACTOR	158	RF/R19	158	Y	N/A	N/A	Y	Y	N/A	Y	Y	3
000632	20	2H11-P601	REAC CNTMT COOL ISO BN BD	CONTROL	164	TE/T12	164	Y	BS	GRS	Y	Y	Y	N	N	
000633	20	2H11-P602	REAC WTR CLNUP & RECIRC	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	N	N	
000634	20	2H11-P603	REAC CONTROL BN BD	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	N	N	
000635	20	2H11-P604	PROCESS RADMON VER BD	CONTROL	164	TE/T13	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000636	20	2H11-P605A	CNTMT ATM OIL VERT BD	CONTROL	164		164	Y	BS	GRS	Y	Y	Y	Y	Y	
000637	20	2H11-P605B	CNTMT ATM OIL VERT BD	CONTROL	164		164	Y	BS	GRS	Y	Y	Y	Y	Y	
000638	20	2H11-P606	STARTUP NEUT MON PNL	CONTROL	164	TC/T13	164	Y	BS	GRS	Y	Y	Y	N	N	
000639	20	2H11-P608	RWR RNGE NEUT MON PNL	CONTROL	164	TC/T13	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000640	20	2H11-P609	CH A PRI ISOL & RPS VB	CONTROL	164	TE/T13	164	Y	BS	GRS	Y	Y	Y	N	N	

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
000641	20	2H11-P611	CH B PRI ISOL & RPS VB	CONTROL	164	TE/T13	164	Y	BS	GRS	Y	N	Y	Y	N	
000642	20	2H11-P612	FW AND RECIRC INST PNL	CONTROL	164	TC/T13	164	Y	BS	GRS	Y	N	Y	Y	N	
000643	20	2H11-P613	PROCESS INST VERT BD	CONTROL	164	TC/T13	164	Y	BS	GRS	Y	N	Y	Y	N	
000644	20	2H11-P614	NSSS TEMP DET VERT BD	CONTROL	164	TE/T13	164	Y	BS	GRS	Y	Y	Y	N	N	
000645	20	2H11-P617	CHAN A RHR RELAY VERT BD	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000646	20	2H11-P618	CHAN B RHR RELAY VERT BD	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000647	20	2H11-P620	HPCI RELAY VERT BD	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000648	20	2H11-P622	INBD ISO VLV VERT PNL	CONTROL	164	TC/T13	164	Y	BS	GRS	Y	N	Y	N	N	
000649	20	2H11-P623	OUTBD ISO VLV VERT PNL	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000650	20	2H11-P626	CORE SPRAY CTRL PNL DIV 1	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000651	20	2H11-P627	CORE SPRAY CTRL PNL DIV 2	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000652	20	2H11-P628	ADS RELAY PANEL	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000653	20	2H11-P650	TURB FDWTR 7 COND PNL	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	N	N	
000654	20	2H11-P652	DSL GEN & EMER STA PNL	CONTROL	164	TC/T11	164	Y	ABS	RRS	Y	Y	Y	N	N	
000655	20	2H11-P654	GAS TREAT VENT VERT BD	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000656	20	2H11-P656	TURB AUX SYSTEM VERT PNL	CONTROL	164	TC/T11	164	Y	BS	GRS	Y	Y	Y	N	N	
000657	20	2H11-P657	VNT DRYWELL INERT VERT BD	CONTROL	164	TC/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000658	20	2H11-P664	MSIV LEAK CTRL SYS PNL	CONTROL	164	TA/T13	164	Y	BS	GRS	Y	N	N	N	N	
000659	20	2H11-P674	START UP XFMR 2C PANEL	CONTROL	164	TA/T12	164	Y	ABS	RRS	Y	N	N	Y	N	
000660	20	2H11-P675	START UP XFMR 2D PANEL	CONTROL	164	TA/T12	164	Y	ABS	RRS	Y	N	N	Y	N	
000661	20	2H11-P679	STA SERV XFMR RLY PANEL	CONTROL	164	TA/T12	164	Y	ABS	RRS	Y	N	N	Y	N	
000662	20	2H11-P691	ANALOG SIG CONV PANEL	CONTROL	164	TB/T11	164	Y	BS	GRS	Y	N	N	Y	N	
000663	20	2H11-P700	ANALOG VNT LEAK DETECT PNL	CONTROL	164	TE/T13	164	Y	BS	GRS	Y	U	U	Y	U	
000664	20	2H11-P921	RPS TRIP UNIT CABINET	CONTROL	164	TI/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000665	20	2H11-P922	RPS TRIP UNIT CABINET	CONTROL	164	TI/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000666	20	2H11-P923	RPS TRIP UNIT CABINET	CONTROL	164	TI/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
000667	20	2H11-P924	RPS TRIP UNIT CABINET	CONTROL	164	TI/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000668	20	2H11-P925	ECCS TRIP UNIT CABINET	CONTROL	164	TI/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000669	20	2H11-P926	ECCS TRIP UNIT CABINET	CONTROL	164	TI/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000670	20	2H11-P927	ECCS TRIP UNIT CABINET	CONTROL	164	TI/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000671	20	2H11-P928	ECCS TRIP UNIT CABINET	CONTROL	164	TI/T12	164	Y	BS	GRS	Y	Y	Y	Y	Y	
000672	18	2H21-P002	REACTOR WATER CLEANUP PNL	REACTOR	158	RF/R24	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000673	18	2H21-P016	MAIN STEAM FLOW INST RACK	REACTOR	087	RL/R24	092	Y	BS	GRS	Y	Y	Y	Y	Y	
000674	18	2H21-P018	RHR INST RACK CHANNEL A	REACTOR	087	RL/R14	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000675	18	2H21-P036	HPCI SYS LOCAL RACK	REACTOR	130	RJ/R23	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000676	20	2H21-P052	HPCI TEST VLV CONTROL PNL	REACTOR	087	RH/R24	110	Y	BS	GRS	Y	Y	Y	Y	Y	
000677	20	2H21-P173	SHUTDOWN INSTRUMENT PANEL	REACTOR	130	RA/R16	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000460	20	2H21-P198	AMMETER SHUNT BATTERY CHG	DIESEL	130	A02	140	Y	BS	GRS	Y	Y	Y	Y	Y	
000500	20	2H21-P199	AMMETER SHUNT BATTERY CHG	DIESEL	130	C02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000678	20	2H21-P200	DIESEL GEN 2A RELAY PANEL	DIESEL	130	A02	130	Y	BS	GRS	Y	N	Y	N	N	
000679	20	2H21-P202	DIESEL GEN 2C RELAY PANEL	DIESEL	130	C02	130	Y	BS	GRS	Y	N	Y	N	N	
000680	18	2H21-P220	TURBINE BUILDING INST RACK	TURBINE	130	TH/T20	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000681	18	2H21-P225	TURBINE BUILDING INST RACK	TURBINE	130	TH/T20	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000682	20	2H21-P230	RELAY PANEL 2A-D/G 2A	DIESEL	130	A02	130	Y	BS	GRS	Y	N	Y	N	N	
000683	20	2H21-P231	RELAY PANEL 2B-D/G 2B	DIESEL	130	B02	130	Y	BS	GRS	Y	N	Y	Y	N	
000684	20	2H21-P232	RELAY PANEL 2C-D/G 2C	DIESEL	130	B02	130	Y	BS	GRS	Y	N	Y	Y	N	
000685	20	2H21-P245	600 VOLT BUS 2C CNT PNL	CONTROL	130	TE/T13	[illegible]	Y	BS	GRS	Y	Y	Y	Y	Y	
000686	20	2H21-P246	600 VOLT BUS 2D CNT PNL	CONTROL	130	T[illegible]/T13	130	Y	BS	GRS	[illegible]	Y	Y	Y	Y	
000687	20	2H21-P248	250 VOLT DC SWGR 2A CNT PNL	CONTROL	130	TE/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000688	20	2H21-P249	250 VOLT DC SWGR 2B CNT PNL	CONTROL	130	TC/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000689	20	2H21-P255	DG FUEL PMP & MOV CONT PNL	DIESEL	130	A02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000690	20	2H21-P256	DG FUEL PMP & MOV CONT PNL	DIESEL	130	C02	135	Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
000691	20	2H21-P257	D/G 2A HT/VEN CONT PNL	DIESEL	130	A01	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000692	20	2H21-P259	D/G 2C HT/VEN CONT PNL	DIESEL	130	B01	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000693	20	2H21-P260	SWGR 2E RM HT/VEN CONT PNL	DIESEL	130	A02	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000694	20	2H21-P262	SWGR 2G RM HT/VEN CONT PNL	DIESEL	130	B02	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000695	20	2H21-P266	MOV CONTROL PNL 2A DIV 1	INTAKE	111		128	Y	BS	GRS	Y	Y	Y	Y	Y	1
000696	20	2H21-P267	MOV CONTROL PNL 2B DIV 2	INTAKE	111		111	Y	BS	GRS	Y	Y	Y	Y	Y	1
000466	20	2H21-P285	SHUNT BOX A	CONTROL	112	TE/T13		Y	BS	GRS	Y	Y	Y	Y	Y	1
000467	20	2H21-P286	SHUNT BOX B	CONTROL	112	TF/T14		Y	ABS	RRS	Y	Y	Y	Y	Y	1
000468	20	2H21-P287	SHUNT BOX C	CONTROL	112	TF/T13		Y	BS	GRS	Y	Y	Y	Y	Y	1
000503	20	2H21-P288	BATTERY SHUNT BOX D	CONTROL	112	TC/T13		Y	ABS	RRS	Y	Y	Y	Y	Y	1
000504	20	2H21-P289	BATTERY SHUNT BOX E	CONTROL	112	TC/T14		Y	BS	GRS	Y	Y	Y	Y	Y	1
000505	20	2H21-P290	BATTERY SHUNT BOX F	CONTROL	112	TE/T13		Y	ABS	RRS	Y	Y	Y	Y	Y	1
000461	20	2H21-P291	BATTERY 2A FUSE BOX	DIESEL	130	B02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000501	20	2H21-P293	BATTERY 2C FUSE BOX	DIESEL	130	C02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000697	20	2H21-P303	DG 2A LOADING TIMER PANEL	DIESEL	130	A03	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000698	20	2H21-P305	DG 2C LOADING TIMER PANEL	DIESEL	130	C03	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000699	18	2H21-P401	CS INSTRUMENT RACK	REACTOR	087	RL/R14	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000700	18	2H21-P404A	RPV LVL/PRESS INSTR RACK	REACTOR	158	RG/R17	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000701	18	2H21-P404B	RPV LVL/PRESS INSTR RACK	REACTOR	158	RG/R17	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000702	18	2H21-P404C	RPV LVL/PRESS INSTR RACK	REACTOR	158	RG/R17	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000703	18	2H21-P404D	RPV LVL/PRESS INSTR RACK	REACTOR	158	RG/R17	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000704	18	2H21-P404E	RPV LVL/PRESS INSTR RACK	REACTOR	158	RG/R17	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000705	18	2H21-P405A	RPV LVL/PRESS INSTR RACK	REACTOR	158	RH/R23	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000706	18	2H21-P405B	RPV LVL/PRESS INSTR RACK	REACTOR	158	RH/R23	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000707	18	2H21-P405C	RPV LVL/PRESS INSTR RACK	REACTOR	158	RH/R23	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000708	18	2H21-P405D	RPV LVL/PRESS INSTR RACK	REACTOR	158	RH/R23	158	Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO. (1)	EQUIP CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	<--------- Building (5)	EQUIPMENT Flr.Elv. (6)	LOCATION -----> Rm. or Row/Col. (7)	Base Elev. (8)	<40'? (9)	Capacity Spectrum (10)	Demand Spectrum (11)	Cap. > Demand? (12)	Caveats OK? (13)	Anchor OK? (14)	Inter-act OK? (15)	Equip OK? (16)	Notes (17)
000709	18	2H21-P405E	RPV LVL/PRESS INSTR RACK	REACTOR	158	RH/R23	158	Y	BS	GRS	Y	Y	Y	Y	Y	
000710	18	2H21-P409	JET PUMP INSTR RACK	REACTOR	130	RF/R15	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000711	18	2H21-P410	JET PUMP INSTR RACK	REACTOR	130	RF/R22	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000712	18	2H21-P414A	HPCI INSTR RACK	REACTOR	087	RG/R24	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000713	18	2H21-P414B	HPCI INSTR RACK	REACTOR	087	RG/R24	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000714	18	2H21-P415A	MAIN STM FLOW INSTRUMENT RACK	REACTOR	130	RF/R15	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000715	18	2H21-P415B	MAIN STM FLOW INSTRUMENT RACK	REACTOR	130	RF/R15	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000716	18	2H21-P418A	RHR INSTRUMENT RACK	REACTOR	087	RL/R24	108	Y	BS	GRS	Y	Y	Y	Y	Y	
000717	18	2H21-P418B	RHR INSTRUMENT RACK	REACTOR	087	RL/R24	108	Y	BS	GRS	Y	Y	Y	Y	Y	
000718	18	2H21-P419	CS INSTRUMENT RACK	REACTOR	087	RL/R24	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000719	18	2H21-P421A	RHR INSTRUMENT RACK	REACTOR	087	RL/R24	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000720	18	2H21-P421B	RHR INSTRUMENT RACK	REACTOR	087	RL/R24	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000721	18	2H21-P425A	RHR INSTRUMENT RACK	REACTOR	130	RF/R22	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000722	18	2H21-P425B	RHR INSTRUMENT RACK	REACTOR	130	RF/R22	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000723	18	2H21-P434	HPCI INSTRUMENT RACK	REACTOR	087	RG/R24	093	Y	BS	GRS	Y	Y	Y	Y	Y	
000756	0	2L48-D134	D/G RM 2A FIRE DAMPER	DIESEL	130	2A	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000757	0	2L48-D137	D/G RM 2C FIRE DAMPER	DIESEL	130	2C	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000255	08A	2N71-F012	CIRC WATER MAKEUP	YARD			122	Y	BS	GRS	Y	Y	N/A	Y	Y	
000243	06	2P41-C001A	PLANT SERVICE WATER PUMP A	INTAKE	111		111	Y	BS	GRS	Y	Y	Y	Y	Y	
000228	06	2P41-C001B	PLANT SERVICE WATER PUMP B	INTAKE	111		111	Y	BS	GRS	Y	Y	Y	Y	Y	
000237	07	2P41-F035B	T41B005B CONTROL VALVE	REACTOR	130	RH/R25	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000236	07	2P41-F036B	T41B002B CONTROL VALVE	REACTOR	098	RL/R24	118	Y	BS	GRS	Y	Y	N/A	Y	Y	
000250	07	2P41-F037A	2E11C002A CONTROL VALVE	REACTOR	096	RL/R14	100	Y	BS	GRS	Y	Y	N/A	Y	Y	
000235	07	2P41-F037B	E11C002B CONTROL VALVE	REACTOR	098	RL/R24	098	Y	BS	GRS	Y	Y	N/A	Y	Y	
000251	07	2P41-F039A	2T41B003A CONTROL VALVE	REACTOR	120	RL/R15	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000248	08A	2P41-F316A	TURBINE BUILDING ISOLATION	YARD			120	Y	BS	GRS	Y	Y	N/A	Y	Y	

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO. (1)	EQUIP CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	<--------- EQUIPMENT Building (5)	Flr.Elv. (6)	LOCATION -----> Rm. or Row/Col. (7)	Base Elev. (8)	<40'? (9)	Capacity Spectrum (10)	Demand Spectrum (11)	Cap. > Demand? (12)	Caveats OK? (13)	Anchor OK? (14)	Inter-act OK? (15)	Equip OK? (16)	Notes (17)
000229	08A	2P41-F316B	TURBINE BUILDING ISOLATION	YARD			120	Y	BS	GRS	Y	Y	N/A	Y	Y	
000246	07	2P41-F339A	DIESEL GENERATOR 2A OUTLET	DIESEL	130	A01	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000232	07	2P41-F339B	DIESEL GENERATOR 2C OUTLET	DIESEL	130	B01	130	Y	BS	GRS	Y	Y	N/A	Y	Y	
000252	18	2P41-N303A	PSW DISCHARGE PT	INTAKE	088		088	Y	BS	GRS	Y	N	N	Y	N	
000240	18	2P41-N303B	PSW DISCHARGE PT	INTAKE	088		088	Y	BS	GRS	Y	N	N	Y	N	
000270	0	2P70-A002A	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RB/R23	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000271	0	2P70-A002B	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RB/R23	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000272	0	2P70-A002C	EMERGENCY NITROGEN BOTTLE	REACTOR	130	RB/R23	130	Y	N/A	N/A	Y	Y	N/A	Y	Y	
000285	07	2P70-F001A	NITROGEN BACKUP SUPPLY	REACTOR	158	RB/R17	158	Y	BS	GRS	Y	Y	N/A	Y	Y	
000277	0	2P70-F084	EMERGENCY NITROGEN ISOLATION	REACTOR	130	RB/R21	130	Y	N/A	N/A	Y	Y	N/A	Y	Y	
000273	0	2P70-F138A	EMERGENCY NITROGEN CONTROL	REACTOR	130	RB/R23	130	Y	N/A	N/A	Y	Y	N/A	Y	Y	
000274	0	2P70-F138B	EMERGENCY NITROGEN CONTROL	REACTOR	130	RB/R23	130	Y	N/A	N/A	Y	Y	N/A	Y	Y	
000275	0	2P70-F138C	EMERGENCY NITROGEN CONTROL	REACTOR	130	RB/R23	130	Y	N/A	N/A	Y	Y	N/A	Y	Y	
000276	0	2P70-F141	EMERGENCY NITROGEN CONTROL	REACTOR	130	RB/R23	130	Y	N/A	N/A	Y	Y	N/A	Y	Y	
000455	04	2R11-S004	600-120/208 V AC XFMR	DIESEL	130	B03	130	Y	ABS	AGS	Y	Y	Y	Y	Y	
000493	04	2R11-S006	LTG & MISC POWER XFMR	DIESEL	130	C03	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000454	04	2R11-S041	CONT BLDG ESS XFMR 2B	CONTROL	130	TD/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000492	04	2R11-S042	CONT BLDG ESS XFMR 2C	CONTROL	130	TC/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000463	18	2R20M-P001	FUSE BOX	DIESEL	130	A02	136	Y	BS	GRS	Y	Y	Y	Y	Y	
000494	18	2R20M-P002	FUSE BOX	DIESEL	130	C02	136	Y	BS	GRS	Y	Y	Y	Y	Y	
000456	18	2R20N-P001	FUSE BOX	CONTROL	130		130	Y	BS	GRS	Y	Y	Y	Y	Y	
000495	18	2R20N-P002	FUSE BOX	CONTROL	130		136	Y	BS	GRS	Y	Y	Y	Y	Y	
000431	03	2R22-S005	4160V STA SVC SWGR 2E	DIESEL	130	A02	130	Y	BS	GRS	Y	N	N	N	N	
000469	03	2R22-S007	4160V STA SVC SWGR 2G	DIESEL	130	C02	130	Y	BS	GRS	Y	N	N	Y	N	
000432	02	2R22-S016	250 V DC BATTERY SWGR 2A	CONTROL	130	TEA/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000470	02	2R22-S017	250 V DC BATTERY SWGR 2B	CONTROL	130	TB/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
000433	02	2R23-S003	600 V STA SVC SWGR 2C & XFMR	CONTROL	130	TEA/T14	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000471	02	2R23-S004	600 V STA SVC SWGR 2D & XFMR	CONTROL	130	TCA/T14	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000434	01	2R24-S009	600/208 V MCC 2A INTAKE STRU	INTAKE	111		112	Y	BS	GRS	U	N	N	Y	N	
000472	01	2R24-S010	600/208 V MCC 2B INTAKE STRU	INTAKE	111		111	Y	ABS	RRS	Y	Y	Y	Y	Y	
000435	01	2R24-S011	600 V MCC 2C ESS DIV 1	REACTOR	130	RF/R14	130	Y	ABS	RRS	Y	Y	Y	Y	Y	
000436	01	2R24-S011A	600 V MCC ESS DIV 1	REACTOR	158	RA/R17	158	Y	ABS	RRS	Y	Y	Y	Y	Y	
000473	01	2R24-S012	600 V MCC 2B ESS DIV 2	REACTOR	130	RF/R24	130	Y	ABS	RRS	Y	Y	Y	Y	Y	
000774	01	2R24-S012A	600 V AC MCC	REACTOR	164	RB/R19	164	Y	ABS	RRS	Y	N	Y	N	N	
000474	01	2R24-S012B	600 V ESS MCC	REACTOR	164	RB/R21	164	Y	ABS	RRS	Y	N	Y	N	N	
000437	01	2R24-S018A	600 V MCC 2E-A ESS DIV 1	REACTOR	130	RH/R17	130	Y	ABS	RRS	Y	Y	Y	Y	Y	
000478	01	2R24-S018B	600 V MCC 2E-B ESS DIV 2	REACTOR	130	RJ/R17	130	Y	ABS	RRS	Y	Y	Y	Y	Y	
000438	01	2R24-S021	125/250 V DC MCC 2A ESS DIV 1	REACTOR	130	RB/R14	130	Y	ABS	RRS	Y	Y	Y	Y	Y	
000475	01	2R24-S022	125/250 V DC MCC 2B ESS DIV 2	REACTOR	130	RH/R24	130	Y	ABS	RRS	Y	Y	Y	Y	Y	
000439	01	2R24-S025	600/208 V MCC 2A ESS DIV 1	DIESEL	130	B02	130	Y	BS	GRS	N	N	N	Y	N	
000476	01	2R24-S027	600/208 V MCC 2C ESS DIV 2	DIESEL	130	C02	130	Y	BS	GRS	N	N	N	Y	N	
000440	14	2R25-S001	125 V DC CABINET 2A	CONTROL	130	TDA/T14	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000479	14	2R25-S002	125 V DC CABINET 2B	CONTROL	130	TCA/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000441	14	2R25-S004	125 V DC CABINET 2D	DIESEL	130	B02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000480	14	2R25-S006	125 V DC CABINET 2F	DIESEL	130	C02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000442	14	2R25-S029	120/208 VAC CABINET 2J	DIESEL	130	B02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000481	14	2R25-S031	120/208 V AC CABINET 2L	DIESEL	130	C02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000443	14	2R25-S036	120/208 V AC ESS CABINET 2A	CONTROL	130	TDA/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000482	14	2R25-S037	120/208 V AC ESS CABINET 2B	CONTROL	130	TCA/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000445	14	2R25-S064	120/208 V AC CABINET 2A INST	CONTROL	130	TG/T13	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000484	14	2R25-S065	120/208 V AC CABINET 2B INST	CONTROL	130	TG/T12	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000447	14	2R25-S129	125 V DC DISTR CABINET 2E	CONTROL	130	TEA/T13	135	Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEN v0.0

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
000485	14	2R25-S130	125 V DC DISTRIBUTION CAB 2D	CONTROL	130	TDA/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000462	18	2R26-M002	2R25-S064 DISCONNECT SWITCH	CONTROL	130	TE/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000502	18	2R26-M004	2R25-S025 DISCONNECT SWITCH	CONTROL	130	TC/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000464	18	2R26-M031A	125 V DC THROWOVER SWITCH 2A	CONTROL	130	TE/T13	135	Y	BS	GRS	Y	Y	Y	Y	Y	1
000458	18	2R26-M031B	125 V DC THROWOVER SWITCH 2B	CONTROL	130	TF/T13	135	Y	BS	GRS	Y	Y	Y	Y	Y	1
000497	18	2R26-M031C	125 V DC THROWOVER SWITCH 2C	CONTROL	130	TB/T13	135	Y	BS	GRS	Y	Y	Y	Y	Y	1
000498	18	2R26-M031D	125 V DC THROWOVER SWITCH 2D	CONTROL	130	TB/T13	135	Y	BS	GRS	Y	Y	Y	Y	Y	1
000459	18	2R26-M032A	125 V DC THROWOVER SWITCH	DIESEL	130	C02	135	Y	BS	GRS	Y	Y	Y	Y	Y	1
000499	18	2R26-M032C	125 V DC THROWOVER SWITCH	DIESEL	130	C02	135	Y	BS	GRS	Y	Y	Y	Y	Y	1
000733	01	2R27-S093	LOCAL STARTER E11-F006	REACTOR	130	RH/R24	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000506	01	2R27-S096	LOCAL STARTER E11-F008	REACTOR	130	RH/R21	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000776	0	2R34-S005A	SURGE PAK FOR 2P41-C001A	INTAKE	111		111	Y	N/A	N/A	Y	N/A	Y	Y	Y	1
000775	0	2R34-S005B	SURGE PAK FOR 2P41-C001B	INTAKE	111		111	Y	N/A	N/A	Y	N/A	Y	Y	Y	1
000777	0	2R34-S006A	SURGE PAK FOR 2E11-C001A	INTAKE	111		111	Y	N/A	N/A	Y	N/A	Y	Y	Y	1
000778	0	2R34-S006D	SURGE PAK FOR 2E11-C001D	INTAKE	111		111	Y	N/A	N/A	Y	N/A	Y	Y	Y	1
000448	15	2R42-S001A	125/250 V STATION BATTERY 2A	CONTROL	112	TEA/T14	112	Y	BS	GRS	Y	Y	Y	Y	Y	
000486	15	2R42-S001B	125/250 V DC STA BATTERY 2B	CONTROL	112	TDA/T14	112	Y	BS	GRS	Y	Y	Y	Y	Y	
000449	15	2R42-S002A	125 V DIESEL SYSTEM BATTERY 2A	DIESEL	130	B02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000487	15	2R42-S002C	125 V DIESEL SYS BATTERY 2C	DIESEL	130	C02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000450	16	2R42-S026	125 V BATTERY CHARGER 2A	CONTROL	130	TF/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000451	16	2R42-S027	125 V BATTERY CHARGER 2B	CONTROL	130	TF/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000488	16	2R42-S029	125 V BATTERY CHARGER 2D	CONTROL	130	TB/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000489	16	2R42-S030	125 V BATTERY CHARGER 2E	CONTROL	130	TB/T13	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000452	16	2R42-S032A	125 V BATTERY CHARGER 2G	DIESEL	130	B02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000490	16	2R42-S032C	125 V BATTERY CHARGER 2J	DIESEL	130	C02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000412	0	2R43-A005A	DG 2A STARTING AIR RECEIVER	DIESEL	130	B01	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	1

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
000424	0	2R43-A005C	DG 2C STARTING AIR RECEIVER	DIESEL	130	C01	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	1
000413	0	2R43-A006A	DG 2A STARTING AIR RECEIVER	DIESEL	130	B01	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	1
000425	0	2R43-A006C	DG 2C STARTING AIR RECEIVER	DIESEL	130	C01	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	1
000411	18	2R43-N001A	DG 2A DAY TANK LS	DIESEL	130	B01	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000423	18	2R43-N003C	DG 2C DAY TANK LS	DIESEL	130	C01	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000724	20	2R43-P001A	DIESEL GEN 2A CONT PANEL	DIESEL	130	B02	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000726	20	2R43-P001C	DIESEL GEN 2C CONT PANEL	DIESEL	130		130	Y	BS	GRS	Y	Y	Y	Y	Y	
000407	17	2R43-S001A	DIESEL GENERATOR 2A	DIESEL	130		130	Y	BS	GRS	Y	Y	Y	Y	Y	
000419	17	2R43-S001C	DIESEL GENERATOR 2C	DIESEL	130		130	Y	BS	GRS	Y	Y	Y	Y	Y	
000453	16	2R44-S002	STATIC INVERTER	CONTROL	147	TI/T14	147	Y	BS	GRS	Y	Y	Y	Y	Y	
000491	16	2R44-S003	STATIC INVERTER	CONTROL	147	TH/T14	147	Y	BS	GRS	Y	Y	Y	Y	Y	
000256	10	2T41-B002B	CS/RHR PUMP ROOM COOLER B	REACTOR	087	RL/R24	110	Y	BS	GRS	Y	Y	Y	Y	Y	
000262	10	2T41-B003A	CS/RHR PUMP ROOM COOLER A	REACTOR	087	RL/R14	120	Y	BS	GRS	Y	N	N	Y	N	
000257	10	2T41-B005B	HPCI PUMP ROOM COOLER B	REACTOR	087	RG/R25	096	Y	BS	GRS	Y	Y	Y	N	N	
000258	19	2T41-N019B	HPCI PUMP ROOM COOLER TE	REACTOR	087	RH/R25	105	Y	BS	GRS	Y	Y	N/A	Y	Y	
000259	19	2T41-N020B	CS/RHR PUMP ROOM COOLER TE	REACTOR	087	RL/R24	106	Y	BS	GRS	Y	Y	N/A	Y	Y	
000263	19	2T41-N021A	CS/RHR PUMP ROOM COOLER TE	REACTOR	087	RL/R14	119	Y	BS	GRS	Y	Y	N/A	Y	Y	
000282	21	2T48-A001	NITROGEN STORAGE TANK	YARD			130	Y	BS	GRS	Y	Y	Y	N	N	
000319	18	2T48-N009A	TORUS WATER TE	REACTOR	087	TORUS	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000309	18	2T48-N009B	TORUS WATER TE	REACTOR	087	TORUS	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000320	18	2T48-N009C	TORUS WATER TE	REACTOR	087	TORUS	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000310	18	2T48-N009D	TORUS WATER TE	REACTOR	087	TORUS	087	Y	BS	GRS	Y	Y	Y	Y	Y	
000727	20	2U61-P001	LEAK DETECTION CONTROL PANEL	CONTROL	130	TD/T13	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000728	20	2U61-P002	LEAK DETECTION CONTROL PANEL	CONTROL	130	TD/T13	135	Y	BS	GRS	Y	Y	Y	Y	Y	
000729	20	2U61-P003	LEAK DETECTION CONTROL PANEL	CONTROL	130	TD/T13	137	Y	BS	GRS	Y	Y	Y	Y	Y	
000730	20	2U61-P004	LEAK DETECTION CONTROL PANEL	CONTROL	130	TD/T13	135	Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO.	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	Base Elev.	<40'?	Capacity Spectrum	Demand Spectrum	Cap. > Demand?	Caveats OK?	Anchor OK?	Inter-act OK?	Equip OK?	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
000414	09	2X41-C010A	DG 2A ROOM EXHAUST FAN	DIESEL	130	A02	150	Y	BS	GRS	Y	Y	Y	Y	Y	
000426	09	2X41-C010C	DG 2C ROOM EXHAUST FAN	DIESEL	130	B02	150	Y	BS	GRS	Y	Y	Y	Y	Y	
000416	0	2X41-C013A	DG 2A ROOM LOUVER	DIESEL	130		130	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000428	0	2X41-C013B	DG 2C ROOM LOUVER	DIESEL	130		130	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000534	09	2X41-C014A	SWGR RM 2E FAN	DIESEL	ROOF		150	Y	BS	GRS	Y	Y	Y	Y	Y	
000529	09	2X41-C014E	SWGR RM 2G FAN	DIESEL	ROOF		150	Y	BS	GRS	Y	Y	Y	Y	Y	
000535	0	2X41-C015A	SWGR RM 2E LOUVER	DIESEL	130	SWGR RM 2E	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000530	0	2X41-C015C	SWGR RM 2G LOUVER	DIESEL	130	SWGR RM 2G	138	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000417	09	2X41-C016A	DG 2A BATTERY ROOM FAN	DIESEL	130	B02	150	Y	BS	GRS	Y	Y	Y	Y	Y	
000429	09	2X41-C016C	DG 2C BATTERY ROOM FAN	DIESEL	130	C02	150	Y	BS	GRS	Y	Y	Y	Y	Y	
000758	0	2X41-C024A	D/G BATT RM 2A FIRE DAMPER	DIESEL	130	2A	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	2
000759	0	2X41-C024B	D/G BATT RM 2C FIRE DAMPER	DIESEL	130	2C	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	2
000760	0	2X41-C024C	D/G BATT RM 2A FIRE DAMPER	DIESEL	130	2A	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	2
000761	0	2X41-C024D	D/G BATT RM 2C FIRE DAMPER	DIESEL	130	2C	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	2
000418	0	2X41-C028A	DG 2A BATTERY ROOM LOUVER	DIESEL	130		130	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000430	0	2X41-C028B	DG 2C BATTERY ROOM LOUVER	DIESEL	130		130	Y	N/A	N/A	Y	N/A	Y	Y	Y	
000762	0	2X41-C030A	D/G RM 2A FIRE DAMPER	DIESEL	130	2A	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	2
000763	0	2X41-C030B	D/G RM 2C FIRE DAMPER	DIESEL	130	2C	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	2
000764	0	2X41-C030C	D/G RM 2A FIRE DAMPER	DIESEL	130	2A	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	2
000765	0	2X41-C030D	D/G RM 2C FIRE DAMPER	DIESEL	130	2C	130	Y	N/A	N/A	Y	N/A	Y	Y	Y	2
000415	18	2X41-N011A	DG 2A ROOM FAN THERMOSTAT	DIESEL	130	A02	134	Y	BS	GRS	Y	Y	Y	Y	Y	
000427	18	2X41-N011B	DG 2C ROOM FAN THERMOSTAT	DIESEL	130	B02	134	Y	BS	GRS	Y	Y	Y	Y	Y	
000536	18	2X41-N013A	SWGR RM 2E LVR THERMOSTAT	DIESEL	130	SWGR RM 2E	134	Y	BS	GRS	Y	Y	Y	Y	Y	
000531	18	2X41-N013C	SWGR RM 2G LVR THERMOSTAT	DIESEL	130	SWGR RM 2G	134	Y	BS	GRS	Y	Y	Y	Y	Y	
000532	18	2X41-N042	FLOW SWITCH FOR FAN 2X41-C010C	DIESEL	ROOF		ROOF	Y	BS	GRS	Y	Y	Y	Y	Y	
000537	18	2X41-N044	FLOW SWITCH FOR FAN 2X41-C010B	DIESEL	ROOF		ROOF	Y	BS	GRS	Y	Y	Y	Y	Y	

APPENDIX D
SCREENING VERIFICATION DATA SHEET (SVDS)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: ID Number
Filter Criteria: (Eval. Type CONTAINS 'S')
Program File Name & Version: SSEM v0.0

LINE NO. (1)	EQUIP CLASS (2)	MARK NO. (3)	SYSTEM/EQUIPMENT DESCRIPTION (4)	Building (5)	EQUIPMENT Flr.Elv. (6)	LOCATION Rm. or Row/Col. (7)	Base Elev. (8)	<40'? (9)	Capacity Spectrum (10)	Demand Spectrum (11)	Cap. > Demand? (12)	Caveats OK? (13)	Anchor OK? (14)	Inter-act OK? (15)	Equip OK? (16)	Notes (17)
000538	18	2X41-N046	FLOW SWITCH FOR FAN 2X41-C010A	DIESEL	ROOF		ROOF	Y	BS	GRS	Y	Y	Y	Y	Y	
000533	18	2X41-N061	FLOW SWITCH FOR FAN 2X41-C010D	DIESEL	ROOF		ROOF	Y	BS	GRS	Y	Y	Y	Y	Y	
000731	20	2X43-P003A	CO2 ZONE 1 CONTROL CABINET	DIESEL	130	A01	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000732	20	2X43-P003B	CO2 ZONE 2 CONTROL CABINET	DIESEL	130	B01	130	Y	BS	GRS	Y	Y	Y	Y	Y	
000409	21	2Y52-A001A	DG 2A FUEL OIL STORAGE TANK	YARD			YARD	Y	N/A	N/A	N/A	N/A	N/A	Y	Y	
000421	21	2Y52-A001C	DG 2C FUEL OIL STORAGE TANK	YARD			YARD	Y	N/A	N/A	N/A	N/A	N/A	Y	Y	
000408	21	2Y52-A101A	DG 2A FUEL OIL DAY TANK	DIESEL	130		130	Y	N/A	N/A	N/A	N/A	Y	Y	Y	
000420	21	2Y52-A101C	DG 2C FUEL OIL DAY TANK	DIESEL	130	C01	130	Y	N/A	N/A	N/A	N/A	Y	Y	Y	
000410	05	2Y52-C001A	DG 2A FUEL OIL PUMP 2A1	YARD			120	Y	ABS	GRS	Y	Y	Y	Y	Y	
000422	05	2Y52-C101C	DG 2C FUEL OIL PUMP 2C2	YARD			120	Y	ABS	GRS	Y	Y	Y	Y	Y	
000527	09	2Z41-C014	STN BATT RM EMERG EXHAUST FAN	TURBINE	130	TE/T16	135	Y	ABS	RRS	Y	Y	Y	Y	Y	
000528	09	2Z41-C015	STN BATT RM EMERG EXHAUST FAN	TURBINE	130	TE/T16	135	Y	ABS	RRS	Y	Y	Y	Y	Y	

APPENDIX E
COMPOSITE SAFE SHUTDOWN EQUIPMENT LIST
PLANT HATCH UNIT 1

The Composite Safe Shutdown Equipment List (SSEL) includes all systems/equipment required for safe shutdown of the plant.

The components on the Composite SSEL are sorted by train (path), system, and mark number. The system and path for a particular component can be determined from the alphanumeric designator in column 2 (see table 1).

The evaluation required (seismic and/or A-46 relay) for each system/equipment component is listed in column 10. Although some components occur several times in the Composite SSEL, their evaluation type is listed only once. Therefore, each component appears only once on the Seismic Review SSEL (Appendix A) which is generated from this Composite SSEL database.

The format of the Composite SSEL corresponds to Appendix A of the Seismic Qualification Utility Group Generic Implementation Procedure. Following is a list of column numbers with a description of each.

Column Number	Description
1	Unique line number
2	System and path designator (see table 1)
3	Equipment class (see table 2)
4	Equipment identification number
5	Equipment description
6	Schematic drawing number
7	Building or area in which equipment is located
8	Floor elevation from which equipment can be seen
9	Building grid row/column (or panel number) indicating equipment location
10	Type of evaluation required; that is, seismic (S), relay (R), or rule-of-the-box (Box)

Column Number	Description
11	Not used
12	Normal state of the equipment during normal plant operation:
	OPEN Equipment is normally open.
	CLSD Equipment is normally closed.
	OP/CL Equipment normally changes state from open to closed or from closed to open.
	ON Equipment is on and is normally operating.
	OFF Equipment is off and is normally not operating.
	N/A Not applicable
13	Desired operating state of the equipment to accomplish its safe shutdown function:
	OPEN Equipment should be open.
	CLSD Equipment should be closed.
	OP/CL Equipment should change state from open to closed or from closed to open.
	ON Equipment should be on and operating.
	OFF Equipment should be off and not operating.
	N/A Not applicable
14	Is an external power source required?
	Y - Yes N - No
15	Support system drawing number
16	Not used
17	Not used

APPENDIX E

TABLE 1
SYSTEM/EQUIPMENT AND PATH DESIGNATIONS

1. **Primary Path (P)**

Train Designator (Column 2)	**System/Equipment Description (Column 5)**
P01	Miscellaneous panels
P02	CRD and reactivity monitoring
P03	Nuclear boiler system
P04	HPCI system
P05	Core spray system
P06	RHR-suppression pool cooling
P08	Plant service water system and emergency room coolers
P09	Drywell air
P10	RPV instrumentation
P11	Suppression pool level and temperature
P12	Diesel generators
P13	Power sources
P14	HVAC systems (control building, intake structure, and diesel building)

2. **Alternate Path (A)**

Train Designator (Column 2)	**System/Equipment Description (Column 5)**
A01	Miscellaneous panels
A02	CRD and reactivity monitoring
A03	Nuclear boiler system
A05	RHR-LPCI mode
A06	RHR-shutdown cooling mode
A08	Plant service water system and emergency room coolers
A09	Drywell air
A10	RPV instrumentation
A11	Suppression pool level and temperature
A12	Diesel generators
A13	Power sources
A14	HVAC systems (control building, intake structure, and diesel building)

APPENDIX E

TABLE 2
EQUIPMENT CLASS DESIGNATIONS

Equipment Class (Column 3)	Description
01	Motor control centers
02	Low-voltage switchgear
03	Medium-voltage switchgear
04	Transformers
05	Horizontal pumps
06	Vertical pumps
07	Fluid-operated valves
08A	Motor-operated valves
08B	Solenoid-operated valves
09	Fans
10	Air handlers
11	Chillers
12	Air compressors
13	Motor generators
14	Distribution panels
15	Batteries on racks
16	Battery chargers and inverters
17	Engine generators
18	Instruments on racks
19	Temperature sensors
20	Instrumentation and control panels and cabinets
21	Tanks and heat exchangers

EDWIN I. HATCH NUCLEAR PLANT - UNIT 1

SSEL CERTIFICATION

All the information contained on this Safe Shutdown Equipment List (SSEL) identifying equipment required to bring the plant to a safe shutdown condition is, to the best of our knowledge and belief, correct and accurate.

W. S. Walker / Mechanical Systems Engineer	William Scott Walker / 11-8-94
Name / Title	Signature / Date
J. E. Smith / Electrical Systems Engineer	James E Smith
Name / Title	Signature / Date

Independent Review:

W. T. Barr / Mechanical Systems Engineer	W. T. Barr 11/9/94
Name / Title	Signature / Date
L. D. McWhorter / Electrical Systems Engineer	Larry D. McWhorter 11-9-94
Name / Title	Signature / Date

hsign.doc

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0855	A01	20	C82-P002	REMOTE SHUTDOWN PANEL	N/A	REACTOR	158	RH/R05			N/A	N/A	N/A	N/A		
0734	A01	20	H11-P601	REACTOR & CONT COOLING & ISOLA CONTROL PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0735	A01	20	H11-P602	REACTOR WTR CLEAN UP & RECIR CONTROL PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0736	A01	20	H11-P603	REACTOR CONTROL PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0737	A01	20	H11-P605A	DIV 1 ANALOG SIG CONV/ISOLATION	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0738	A01	20	H11-P605B	DIV 2 ANALOG SIG CONV/ISOLATION	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0739	A01	20	H11-P606	START U RANGE NEUTRON MONTORING	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0807	A01	20	H11-P608	POWER RANGE NEUTRON MONITORING PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0740	A01	20	H11-P609	CHAN A PRI ISO SYS REAC PROT SYS	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0842	A01	20	H11-P610	RPS TEST MON VB	N/A	CONTROL	164	TE/T11	S		N/A	N/A	N/A	N/A		
0808	A01	20	H11-P613	PROCESS INSTRUMENTATION PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0743	A01	20	H11-P617	CHAN A RHR CORE SPRAY RELAY PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0746	A01	20	H11-P622	INBOARD-PRI CONTROL ISOLATION RELAYS PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0748	A01	20	H11-P626	CORE SPRAY CONTR PNL DIV I	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0809	A01	20	H11-P628	AUTOMATIC BLOWDOWN RELAY PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0750	A01	20	H11-P650	TURB, FW & COND CONTROL CONSOLE	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0752	A01	20	H11-P652	ELEC AUX PWR CONTROL CONSOLE	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0796	A01	20	H11-P657	VENT & DRYWELL INERT VERTICAL PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0797	A01	20	H11-P691	ANALOG SIGNAL CONVERSION/ISOLATION PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0798	A01	20	H11-P700	ANALYZER/VENT & LEAK DETECTION PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0799	A01	20	H11-P921	ATTS RPS PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0801	A01	20	H11-P923	ATTS RPS PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0802	A01	20	H11-P925	ATTS ECCS PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0805	A01	20	H11-P927	ATTS ECCS PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0772	A01	18	H21-P018	RHR INSTRUMENT RACK-CHANNEL A	N/A	REACTOR	087	RL/R13	S		N/A	N/A	N/A	N/A		
0773	A01	20	H21-P173	SHUTDOWN INSTRUMENT PANEL	N/A	REACTOR	130	RL/R03	S		N/A	N/A	N/A	N/A		
0754	A01	20	H21-P200	DIESEL GEN 1A - RELAY PANEL 1A	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		
0757	A01	20	H21-P230	DEISEL GEN 1A - RELAY PANEL 1B	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		
0760	A01	20	H21-P245	600 VOLT SWGR 1C CONTROL PANEL	N/A	CONTROL	130	N/A	S		N/A	N/A	N/A	N/A		
0762	A01	20	H21-P255	MOV & FUEL PUMP CTRL PANEL 1A - DIV I	N/A	DIESELDB	130	N/A	S		N/A	N/A	N/A	N/A		
0765	A01	20	H21-P266	MOV CONTROL PANEL 1A-DIV I	N/A	INTAKE	111	N/A	S		N/A	N/A	N/A	N/A		
0767	A01	20	H21-P303	DIESEL 1A LEAD TIMER PANEL	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		
0781	A01	18	H21-P404A	INSTRUMENT RACK	N/A	REACTOR	158	RH/R08	S		N/A	N/A	N/A	N/A		
0783	A01	18	H21-P404B	INSTRUMENT RACK	N/A	REACTOR	158	RH/R08	S		N/A	N/A	N/A	N/A		
0779	A01	18	H21-P409	INSTRUMENT RACK	N/A	REACTOR	130	RH/R10	S		N/A	N/A	N/A	N/A		
0466	A02	08B	C11-D001-117	PILOT SCRAM SOLENOID	H16064	REACTOR	130	HCU	SR		ENERGZ	D-ENERG	N	H17793		
0467	A02	08B	C11-D001-118	PILOT SCRAM SOLENOID	H16064	REACTOR	130	HCU	SR		ENERGZ	D-ENERG	N	H17793		
0471	A02	08B	C11-D001-120	ROD POSITION CONTROL VALVE	H16064	REACTOR	130	HCU			CLOSED	CLOSED	N			
0472	A02	08B	C11-D001-121	ROD POSITION CONTROL VALVE	H16064	REACTOR	130	HCU			CLOSED	CLOSED	N			
0473	A02	08B	C11-D001-122	ROD POSITION CONTROL VALVE	H16064	REACTOR	130	HCU			CLOSED	CLOSED	N			
0474	A02	08B	C11-D001-123	ROD POSITION CONTROL VALVE	H16064	REACTOR	130	HCU			CLOSED	CLOSED	N			
0468	A02	0	C11-D001-125	SCRAM ACCUMULATOR	H16064	REACTOR	130	HCU	S		N/A	N/A	N			
0469	A02	07	C11-D001-126	SCRAM INLET VALVE	H16064	REACTOR	130	HCU	S		CLOSED	OPEN	N			
0470	A02	07	C11-D001-127	SCRAM OUTLET VALVE	H16064	REACTOR	130	HCU	S		CLOSED	OPEN	N			
0462	A02	08B	C11-F009A	SDV VENT & DRAIN PILOT VALVE	H16065	REACTOR	130	RA/R05	SR		ENERGZ	D-ENERG	N	H17793		
0523	A02	08B	C11-F009B	SDV VENT & DRAIN PILOT VALVE	H16065	REACTOR	130	RA/R05	SR		ENERGZ	D-ENERG	N	H17793		
0463	A02	07	C11-F010A	SDV VENT VALVE	H16065	REACTOR	130	RH/R12	S		OPEN	CLOSED	N	H17794		
0464	A02	07	C11-F010B	SDV VENT VALVE	H16065	REACTOR	130	RH/R03	S		OPEN	CLOSED	N	H17794		
0465	A02	07	C11-F011	SDV DRAIN VALVE	H16065	REACTOR	130	RB/R03	S		OPEN	CLOSED	N	H17794		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0476	A02	20	C71-S3A	MANUAL SCRAM SWITCH	N/A	CONTROL	164	H11-P603	R		N/A	N/A	Y	H17791		
0477	A02	20	C71-S3B	MANUAL SCRAM SWITCH	N/A	CONTROL	164	H11-P603	R		N/A	N/A	Y	H17791		
0861	A03	08B	B21-F004	RPV HEAD VENT	H16062	DRYWELL	193	RF/R07	R		CLOSED	CLOSED	N	H17813		
0706	A03	07	B21-F013F	RPV SAFETY RELIEF VALVE	H16062	DRYWELL	148	AZ304	SR		CLOSED	OP/CL	Y	H17753		
0707	A03	07	B21-F013G	RPV PRESSURE RELIEF VALVE	H16062	DRYWELL	148	AZ293	SR		CLOSED	OP/CL	Y	H17753		
0708	A03	07	B21-F013H	RPV PRESSURE RELIEF VALVE	H16062	DRYWELL	148	AZ324	SR		CLOSED	OP/CL	Y	H17753		
0709	A03	07	B21-F013J	RPV SAFETY RELIEF VALVE	H16062	DRYWELL	148	AZ306	SR		CLOSED	OP/CL	Y	H17753		
0697	A03	07	B21-F028A	OUTBOARD MSIV	H16062	REACTOR	130	RB/R07	SR		OPEN	CLOSED	N	H17816		
0698	A03	07	B21-F028B	OUTBOARD MSIV	H16062	REACTOR	130	RB/R07	SR		OPEN	CLOSED	N	H17816		
0699	A03	07	B21-F028C	OUTBOARD MSIV	H16062	REACTOR	130	RB/R07	SR		OPEN	CLOSED	N	H17816		
0700	A03	07	B21-F028D	OUTBOARD MSIV	H16062	REACTOR	130	RB/R07	SR		OPEN	CLOSED	N	H17816		
0495	A05	08A	B31-F023A	RECIRC PUMP C001A SUCTION VALVE	H16066	DRYWELL	117	RF/R07			OPEN	CLOSED	Y	H17865		
0354	A05	21	E11-B001A	RHR HEAT EXCHANGER A	H16330	REACTOR	087	RL/R13	S		N/A	N/A	N	N/A		
0331	A05	06	E11-C002A	RHR PUMP 2A	H16330	REACTOR	087	RH/R11			OFF	ON	Y	H17782		
0332	A05	08A	E11-F003A	RHR HX A DISCH VALVE	H16330	REACTOR	087	RH/R13			OPEN	OPEN	N	H17772		
0333	A05	08A	E11-F004A	RHR PUMP 2A SUCTION VALVE	H16330	REACTOR	087	RH/R13			OPEN	OPEN	N	H17772		
0334	A05	08A	E11-F006A	RHR SDC SUCT ISOL VALVE	H16330	REACTOR	087	RH/R11			CLOSED	CLOSED	N	H17772		
0335	A05	08A	E11-F007A	RHR PUMP 2A & 2C MIN FLOW BYPASS VLV	H16330	REACTOR	087	RL/R11	SR		OPEN	CLOSED	Y	H17773		
0336	A05	08A	E11-F010	RHR HX HDR BYPASS VALVE	H16330	REACTOR	087	RL/R07			CLOSED	CLOSED	N	H17772		
0337	A05	08A	E11-F011A	RHR HX A DRN TO SUPP POOL VLV	H16330	REACTOR	087	RH/R11			CLOSED	CLOSED	N	H17772		
0338	A05	08A	E11-F015A	RHR LPCI DISCHARGE VALVE	H16330	REACTOR	130	RJ/R08			CLOSED	OPEN	Y	H17775		
0339	A05	08A	E11-F016A	CONTAINMENT SPRAY DISCH VALVE	H16330	REACTOR	158	RH/R08			CLOSED	CLOSED	N	H17772		
0340	A05	08A	E11-F017A	RHR LPCI DISCHARGE VALVE	H16330	REACTOR	130	RJ/R08	SR		OPEN	CONTROL	Y	H17772		
0341	A05	08A	E11-F026A	RHR HX A TO RCIC VALVE	H16330	REACTOR	087	RJ/R11			CLOSED	CLOSED	N	H17772		
0342	A05	08A	E11-F028A	RHR INLET TO SUPP POOL VLV	H16330	REACTOR	087	RH/R09			CLOSED	CLOSED	N	H17772		
0343	A05	08A	E11-F047A	RHR HX A INLET VALVE	H16330	REACTOR	087	RL/R13			OPEN	OPEN	N	H17772		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0344	A05	08A	E11-F048A	RHR HX A BYPASS VALVE	H16330	REACTOR	087	RL/R13	SR		OPEN	OPEN	N	H17772		
0345	A05	07	E11-F065A	RHR PUMP 2A SUPP POOL SUCT VALVE	H16330	REACTOR	087	RH/R11	R		OPEN	OPEN	N	H17778		
0355	A05	08B	E11-F079A	RHR PROCESS SAMPLE VALVE	H16330	REACTOR	087	RL/R13			CLOSED	CLOSED	N	H17806		
0346	A05	08A	E11-F091A	HPCI DISCH TO RHR HX A VLV	H16330	REACTOR	087	RH/R11			CLOSED	CLOSED	N	H17776		
0347	A05	08A	E11-F104A	RHR HX A VENT VALVE	H16330	REACTOR	087	RH/R13			CLOSED	CLOSED	N	H17773		
0348	A05	18	E11-K600A	RHR HDR FLOW SQUARE ROOT CONV	H16330	CONTROL	164	H11-P613			N/A	N/A	Y	H17771		
0349	A05	0	E11-N014A	RHR PUMP A & C DISCH FLOW ELEMENT	H16330	REACTOR	087	RH/R09			N/A	N/A	N	N/A		
0350	A05	18	E11-N015A	RHR HX A DISCH HDR FLOW TRANS	H16330	REACTOR	087	H21-P018			N/A	N/A	Y	H17760		
0351	A05	18	E11-N082A	RHR PUMP A & C FLOW DP TRANS	H16330	REACTOR	087	RL/R10			N/A	N/A	Y	H19823		
0352	A05	18	E11-N682A	RHR PUMP A & C FLOW DP TRIP UNIT	H16330	CONTROL	164	H11-P925			N/A	N/A	Y	H19823		
0353	A05	20	E11-R603A	RHR HX A DISCH HDR FLOW IND	H16330	CONTROL	164	H11-P601			N/A	N/A	Y	H17769		
0460	A06	08A	B31-F023A	RECIRC PUMP C001A SUCTION VALVE	H16066	DRYWELL	117	RF/R07	SR		OPEN	CLOSED	Y	H17865		
0289	A06	21	E11-B001A	RHR HEAT EXCHANGER A	H16330	REACTOR	087	RL/R13			N/A	N/A	N	N/A		
0290	A06	06	E11-C001A	RHR SERVICE WATER PUMP 1A	D11004	INTAKE	111		SR		OFF	ON	Y	H17781		
0292	A06	06	E11-C002A	RHR PUMP 2A	H16330	REACTOR	087	RL/R11	SR		OFF	ON	Y	H17782		
0293	A06	08A	E11-F003A	RHR HX A DISCH VALVE	H16330	REACTOR	087	RH/R13	R		OPEN	OPEN	N	H17772		
0294	A06	08A	E11-F004A	RHR PUMP 2A SUCTION VALVE	H16330	REACTOR	087	RH/R13	R		OPEN	OPEN	N	H17772		
0295	A06	08A	E11-F006A	RHR SDC SUCT ISOL VALVE	H16330	REACTOR	087	RH/R11	R		CLOSED	CLOSED	N	H17772		
0297	A06	08A	E11-F006C	RHR SDC SUCT ISOL VALVE	H16330	REACTOR	087	RL/R11	R		CLOSED	CLOSED	N	H17772		
0299	A06	08A	E11-F007A	RHR PUMP 2A & 2C MIN FLOW BYPASS VLV	H16330	REACTOR	087	RL/R11			OPEN	CLOSED	Y	H17773		
0302	A06	08A	E11-F010	RHR HX HDR BYPASS VALVE	H16330	REACTOR	087	RL/R07	R		CLOSED	CLOSED	N	H17772		
0303	A06	08A	E11-F011A	RHR HX A DRN TO SUPP POOL VLV	H16330	REACTOR	087	RH/R11	R		CLOSED	CLOSED	N	H17772		
0304	A06	08A	E11-F015A	RHR LPCI DISCHARGE VALVE	H16330	REACTOR	130	RJ/R08	SR		CLOSED	OPEN	Y	H17775		
0305	A06	08A	E11-F016A	CONTAINMENT SPRAY DISCH VALVE	H16330	REACTOR	158	RH/R08	R		CLOSED	CLOSED	N	H17772		
0306	A06	08A	E11-F017A	RHR LPCI DISCHARGE VALVE	H16330	REACTOR	130	RJ/R08			OPEN	OP/CL	N	H17772		
0307	A06	08A	E11-F026A	RHR HX A TO RCIC VALVE	H16330	REACTOR	087	RJ/R11	R		CLOSED	CLOSED	N	H17772		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEN 2.2

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	Building (7)	EQUIPMENT Flr.Elv. (8)	LOCATION Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	OP. ST. Normal (12)	OP. ST. Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)	REG. ISSUE (17)
0308	A06	08A	E11-F028A	RHR INLET TO SUPP POOL VLV	H16330	REACTOR	087	RH/R09	R		CLOSED	CLOSED	N	H17772		
0309	A06	08A	E11-F047A	RHR HX A INLET VALVE	H16330	REACTOR	087	RL/R13	R		OPEN	OPEN	N	H17772		
0310	A06	08A	E11-F048A	RHR HX A BYPASS VALVE	H16330	REACTOR	087	RL/R13			CONTROL	CONTROL	Y	H17772		
0301	A06	07	E11-F065A	RHR PUMP 2A SUPP POOL SUCT VALVE	H16330	REACTOR	087	RH/R11			OPEN	OPEN	N	H17778		
0311	A06	08A	E11-F068A	RHR HX A TUBE TO SHELL OUTLET	H16330	REACTOR	087	RH/R13	SR		CLOSED	OP/CL	Y	H17773		
0312	A06	08A	E11-F073A	RHR HX A SERV WATER DISCH VALVE	H16330	REACTOR	087	RL/R07	R		CLOSED	CLOSED	N	H17772		
0330	A06	08B	E11-F079A	RHR PROCESS SAMPLE VALVE	H16330	REACTOR	087	RL/R13	R		CLOSED	CLOSED	N	H17819		
0313	A06	08A	E11-F091A	HPCI DISCH TO RHR HX A VLV	H16330	REACTOR	087	RH/R11	R		CLOSED	CLOSED	N	H17776		
0314	A06	08A	E11-F104A	RHR HX A VENT VALVE	H16330	REACTOR	087	RH/R13	R		CLOSED	CLOSED	N	H17773		
0315	A06	08A	E11-F119A	RHR HX A BYPASS VALVE	H16330	REACTOR	087	RL/R07	R		CLOSED	CLOSED	N	H17778		
0316	A06	18	E11-K600A	RHR HDR FLOW SQUARE ROOT CONV	H16330	CONTROL	164	H11-P613			N/A	N/A	Y	H17771		
0317	A06	18	E11-N002A	RHR HX A TUBE TO SHELL DP TRANS	H16330	REACTOR	087	H21-P018	S		N/A	N/A	Y	H17777		
0318	A06	0	E11-N006A	RHRSW PUMP A & C DISCH FLOW ELEM	H16330	REACTOR	087	RH/R10			N/A	N/A	N	N/A		
0319	A06	18	E11-N007A	RHRSW HX A FLOW TRANSMITTER	H16330	REACTOR	087	H21-P018	S		N/A	N/A	Y	H17769		
0320	A06	0	E11-N014A	RHR PUMP A & C DISCH FLOW ELEMENT	H16330	REACTOR	087	RH/R09			N/A	N/A	N	NA		
0321	A06	18	E11-N015A	RHR HX A DISCH HDR FLOW TRANS	H16330	REACTOR	087	H21-P018			N/A	N/A	Y	H17760		
0322	A06	18	E11-N017A	RHR HX A INLET PRESSURE SWITCH	H16330	REACTOR	087	RH/R13	S		N/A	N/A	Y	H17763		
0323	A06	18	E11-N017C	RHR HX A INLET PRESSURE SWITCH	H16330	REACTOR	087	RH/R13	S		N/A	N/A	Y	H17763		
0324	A06	18	E11-N082A	RHR PUMP A & C FLOW DP TRANS	H16330	REACTOR	087	RL/R10	S		N/A	N/A	Y	H19823		
0325	A06	18	E11-N682A	RHR PUMP A & C FLOW DP TRIP UNIT	H16330	CONTROL	164	H11-P925			N/A	N/A	Y	H19823		
0326	A06	18	E11-R600A	RHR HX A TUBE TO SHELL DP PI CONT	H16330	CONTROL	164	H11-P613			N/A	N/A	Y	H17777		
0327	A06	20	E11-R602A	RHRSW HX A INLET FI	H16330	CONTROL	164	H11-P601			N/A	N/A	Y	H17769		
0328	A06	20	E11-R603A	RHR HX A DISCH HDR FLOW IND	H16330	CONTROL	164	H11-P601			N/A	N/A	Y	H17769		
0329	A06	18	E11-S600A	RHR HX A TUBE TO SHELL POS MOD	H16330	CONTROL	164	H11-P613			N/A	N/A	Y	H17763		
1048	A07	21	E11-B001A	RHR HEAT EXCHANGER A	H16330	REACTOR	087	RL/R13			N/A	N/A	N/A			
1069	A07	06	E11-C001A	RHR SERVICE WATER PUMP 1A	D11004	INTAKE	111				OFF	ON	Y			

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
1043	A07	06	E11-C002A	RHR PUMP 2A	H16330	REACTOR	087	RL/R11			OFF	ON	Y			
1055	A07	08A	E11-F003A	RHR HX A DISCH VALVE	H16330	REACTOR	087	RH/R13			OPEN	OPEN	N			
1041	A07	08A	E11-F004A	RHR PUMP 2A SUCTION VALVE	H16330	REACTOR	087	RH/R13			OPEN	OPEN	N			
1042	A07	08A	E11-F006A	RHR SDC SUCT ISOL VALVE	H16330	REACTOR	087	RH/R11			CLOSED	CLOSED	N			
1044	A07	08A	E11-F007A	RHR PUMP 2A & 2C MIN FLOW BYPASS VLV	H16330	REACTOR	087	RL/R11			OPEN	CLOSED	Y			
1061	A07	08A	E11-F010	RHR HX HDR BYPASS VALVE	H16330	REACTOR	087	RL/R07			CLOSED	CLOSED	N			
1054	A07	08A	E11-F011A	RHR HX A DRN TO SUPP POOL VLV	H16330	REACTOR	087	RH/R11			CLOSED	CLOSED	N			
1065	A07	08A	E11-F015A	RHR LPCI DISCHARGE VALVE	H16330	REACTOR	130	RJ/R08			CLOSED	CLOSED	N			
1066	A07	08A	E11-F016A	CONTAINMENT SPRAY DISCH VALVE	H16330	REACTOR	158	RH/R08	S		CLOSED	OPEN	Y			
1067	A07	08A	E11-F021A	CONT SPRAY INBRD GATE MOV	H16330	REACTOR	158	RH/R08	S		CLOSED	OPEN	Y			
1063	A07	08A	E11-F024A	RHR TEST LINE TORUS ISO	H16330	REACTOR	087	RH/R10			CLOSED	CLOSED	N			
1053	A07	08A	E11-F026A	RHR HX A TO RCIC VALVE	H16330	REACTOR	087	RJ/R11			CLOSED	CLOSED	N			
1064	A07	08A	E11-F027A	SUPP POOL SPRAY VALVE	H16330	REACTOR	087	RJ/R10	S		CLOSED	OPEN	Y			
1062	A07	08A	E11-F028A	RHR INLET TO SUPP POOL VLV	H16330	REACTOR	087	RH/R09			CLOSED	OPEN	Y			
1046	A07	08A	E11-F047A	RHR HX A INLET VALVE	H16330	REACTOR	087	RL/R13			OPEN	OPEN	N			
1045	A07	08A	E11-F048A	RHR HX A BYPASS VALVE	H16330	REACTOR	087	RL/R13			OPEN	CLOSED	Y			
1040	A07	07	E11-F065A	RHR PUMP 2A SUPP POOL SUCT VALVE	H16330	REACTOR	087	RH/R11			OPEN	OPEN	N			
1076	A07	08A	E11-F068A	RHR HX A TUBE TO SHELL OUTLET	H16330	REACTOR	087	RH/R13			CLOSED	OP/CL	Y			
1070	A07	08A	E11-F073A	RHR HX A SERV WATER DISCH VALVE	H16330	REACTOR	087	RL/R07			CLOSED	CLOSED	N			
1047	A07	08A	E11-F091A	HPCI DISCH TO RHR HX A VLV	H16330	REACTOR	087	RH/R11			CLOSED	CLOSED	N			
1049	A07	08A	E11-F104A	RHR HX A VENT VALVE	H16330	REACTOR	087	RH/R13			CLOSED	CLOSED	N			
1071	A07	08A	E11-F119A	RHR HX A BYPASS VALVE	H16330	REACTOR	087	RL/R07			CLOSED	CLOSED	N			
1057	A07	18	E11-K600A	RHR HDR FLOW SQUARE ROOT CONV	H16330	CONTROL	164	H11-P613			N/A	N/A	Y			
1068	A07	20	E11-K603A	POWER SUPPLY	H16330	CONTROL	164	H11-P613			N/A	N/A	Y			
1050	A07	18	E11-N002A	RHR HX A TUBE TO SHELL DP TRANS	H16330	REACTOR	087	H21-P018			N/A	N/A	Y			
1072	A07	18	E11-N007A	RHRSW HX A FLOW TRANSMITTER	H16330	REACTOR	087	H21-P018			N/A	N/A	Y			

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH?95.DBF / 12/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	<---------- Building (7)	EQUIPMENT Flr.Elv. (8)	LOCATION -----> Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	<-- OP. Normal (12)	ST. --> Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)	REG. ISSUE (17)
1056	A07	18	E11-N015A	RHR HX A DISCH HDR FLOW TRANS	H16330	REACTOR	087	H21-P018			N/A	N/A	Y			
1074	A07	18	E11-N017A	RHR HX A INLET PRESSURE SWITCH	H16330	REACTOR	087	RH/R13			N/A	N/A	Y			
1075	A07	18	E11-N017C	RHR HX A INLET PRESSURE SWITCH	H16330	REACTOR	087	RH/R13			N/A	N/A	Y			
1059	A07	18	E11-N082A	RHR PUMP A & C FLOW DP TRANS	H16330	REACTOR	087	RL/R10			N/A	N/A	Y			
1060	A07	20	E11-N682A	RHR PUMP A & C FLOW DP TRIP UNIT	H16330	CONTROL	164	H11-P925			N/A	N/A	Y			
1051	A07	18	E11-R600A	RHR HX A TUBE TO SHELL DP PI CONT	H16330	CONTROL	164	H11-P613			N/A	N/A	Y			
1073	A07	20	E11-R602A	RHRSW HX A INLET FI	H16330	CONTROL	164	H11-P601			N/A	N/A	Y			
1058	A07	20	E11-R603A	RHR HX A DISCH HDR FLOW IND	H16330	CONTROL	164	H11-P601			N/A	N/A	Y			
1052	A07	18	E11-S600A	RHR HX A TUBE TO SHELL POS MOD	H16330	CONTROL	164	H11-P613			N/A	N/A	Y			
0406	A08	06	P41-C001A	PLANT SERVICE WATER PUMP 1A	D11001	INTAKE	111		SR		ON/OFF	ON	Y	H13586		
0407	A08	07	P41-F037A	RHR PUMP COOLER 2A INLET VALVE	H16011	REACTOR	087	RL/R13	SR		CLOSED	OPEN	Y	H17087		
0408	A08	07	P41-F039A	RHR/CS ROOM COOLER 2A VALVE	H16011	REACTOR	087	RL/R11	SR		CLOSED	OPEN	Y	H17042		
0410	A08	07	P41-F067	PSW INLET VALVE	H16011	REACTOR	124	RL/R07	R		OPEN	OPEN	N	H17087		
0411	A08	08	P41-F303A	PSW VALVE	H11024	YARD			R		OPEN	OPEN	N	H13388		
0412	A08	08	P41-F310A	PSW TURBINE BLDG ISOL VALVE	H11600	YARD			SR		OPEN	CLOSED	Y	H13388		
0413	A08	08	P41-F312	PSW RETURN LINE ISOL VALVE	D11001	INTAKE	098				CLOSED	CLOSED	N	H13388		
0414	A08	08A	P41-F313A	PSW STRAINER A ISOL VALVE	D11001	INTAKE	094		R		CLOSED	CLOSED	N	H13388		
0415	A08	08A	P41-F317A	DIV. I PSW TO DIESEL 1A & 1B	H11600	YARD			R		OPEN	OPEN	N	H13388		
0416	A08	08A	P41-F380A	PSW RX BLDG. ISOL VALVE	H11600	YARD			R		OPEN	OPEN	N	H13388		
0417	A08	08A	P41-F401A	PSW 1B DIESEL ISOL VALVE	H11600	DIESEL	140	D/01	R		CLOSED	CLOSED	N	H13388		
0418	A08	08A	P41-F402B	PSW 1B DIESEL ISOL VALVE	H11600	DIESEL	140	D/01	R		CLOSED	CLOSED	N	H13388		
0419	A08	08A	P41-F403A	PSW 1A DIESEL ISOL VALVE	H11600	DIESEL	140	C/01	R		OPEN	OPEN	N	H13388		
0420	A08	18	P41-N200A	PSW STRAINER DP SWITCH	D11001	INTAKE	087		S		N/A	N/A	N	H13388		
0424	A08	10	T41-B002A	RHR/CS PUMP ROOM COOLER	H16011	REACTOR	087	RL/R11	SR		OFF	ON	Y	H17042		
0814	A08	08A	W32-F001	FLUME WASH WATER MOV	H11001	INTAKE	108		R		CLOSED	CLOSED	N	H13388		
0445	A09	0	B21-A003A	AIR ACCUM FOR RELIEF VALVE A	H16299	DRYWELL	148	AZ043	S		N/A	N/A	N	N/A		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1RS.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0446	A09	0	B21-A003B	AIR ACCUM FOR RELIEF VALVE B	H16299	DRYWELL	148	AZ077	S		N/A	N/A	N	N/A		
0447	A09	0	B21-A003C	AIR ACCUM FOR RELIEF VALVE C	H16299	DRYWELL	148	AZ073	S		N/A	N/A	N	N/A		
0449	A09	0	B21-A003E	AIR ACCUM FOR RELIEF VALVE E	H16299	DRYWELL	148	AZ088	S		N/A	N/A	N	N/A		
0450	A09	07	P70-F001A	DW PNEUMATIC NITROGEN SUPPLY VALVE	H16286	REACTOR	087	RA/R02	SR		CLOSED	OPEN	N	H17144		
0451	A09	08A	P70-F004	DW PNEUMATIC HIGH FLOW ISOL VALVE	H16286	REACTOR	145	RA/R06	R		OPEN	OPEN	N	H40217		
0452	A09	08A	P70-F005	DW PNEUMATIC HIGH FLOW ISOL VALVE	H16286	REACTOR	145	RA/R06	R		OPEN	OPEN	N	H40217		
0453	A09	0	P70-N021	DRYWELL PNEUMATIC FLOW ELEMENT	H16286	REACTOR	142	RA/R06			N/A	N/A	N	N/A		
0454	A09	18	P70-N022A	DRYWELL PNEUMATIC FLOW TRANS	H16286	REACTOR	130	RA/R06	S		N/A	N/A	N	H40217		
0455	A09	18	P70-N022B	DRYWELL PNEUMATIC FLOW TRANS	H16286	REACTOR	130	RA/R06	S		N/A	N/A	N	H40217		
0456	A09	21	T48-A001	UNIT 1 NITROGEN STORAGE TANK	H16000	YARD			S		N/A	N/A	N	N/A		
0457	A09	08A	T48-F013A	UNIT 2 ISOLATION VALVE	H16000	REACTOR	130	RE/R13	R		CLOSED	CLOSED	N	H17046		
0458	A09	0	T48-F111	NITROGEN HEADER ISOLATION VALVE	H16000	REACTOR	130	RB/R09			OPEN	CLOSED	N	N/A		
0865	A10	18	B21-N078A	PRV PRESS SCRAM PT	H16063	REACTOR	158	H21-P404C			N/A	N/A	Y	H19809		
0867	A10	18	B21-N078C	PRV PRESS SCRAM PT	H16063	REACTOR	158	H21-P405C			N/A	N/A	Y	H19815		
0869	A10	18	B21-N080A	RPV LEVEL 3 LT	H16063	REACTOR	158	H21-P404C			N/A	N/A	Y	H19809		
0871	A10	18	B21-N080C	RPV LEVEL 3 LT	H16063	REACTOR	158	H21-P405C			N/A	N/A	Y	H19815		
0723	A10	18	B21-N085A	SHROUD LEVEL LT	H16063	REACTOR	158	H21-P409			N/A	N/A	Y	H19823		
0549	A10	18	B21-N090A	RPV LOW PRESSURE PT	H16063	REACTOR	158	H21-P404A			N/A	N/A	Y	H19827		
0719	A10	18	B21-N090C	RPV LOW PRESSURE PT	H16063	REACTOR	158	H21-P409			N/A	N/A	Y	H19827		
0550	A10	18	B21-N091A	RPV LEVEL 2 & 1 LT	H16063	REACTOR	158	H21-P404A			N/A	N/A	Y	H19823		
0721	A10	18	B21-N091C	RPV LEVEL 2 & 1 LT	H16063	REACTOR	158	H21-P404A			N/A	N/A	Y	H19823		
0726	A10	18	B21-N093A	RPV LEVEL 8 LT	H16063	REACTOR	158	RH/R08	S		N/A	N/A	Y	H19823		
0551	A10	18	B21-N095A	RPV LEVEL 3 LT	H16063	REACTOR	158	H21-P404B			N/A	N/A	Y	H19823		
0873	A10	20	B21-N678A	RPV PRESS SCRAM PIS	H16063	CONTROL	164	H11-P921			N/A	N/A	Y	H19809		
0875	A10	20	B21-N678C	RPV PRESS SCRAM PIS	H16063	CONTROL	164	H11-P923			N/A	N/A	Y	H19815		
0877	A10	20	B21-N680A	RPV LEVEL 3 LIS	H16063	CONTROL	164	H11-P921			N/A	N/A	Y	H19809		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 0?:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0879	A10	20	B21-N680C	RPV LEVEL 3 LIS	H16063	CONTROL	164	H11-P923			N/A	N/A	Y	H19815		
0724	A10	20	B21-N685A	RPV LEVEL 0 LIS	H16063	CONTROL	164	H11-P925			N/A	N/A	Y	H19823		
0566	A10	20	B21-N690A	RPV PRESSURE LOW PIS	H16063	CONTROL	164	H11-P927			N/A	N/A	Y	H19827		
0720	A10	20	B21-N690C	RPV PRESSURE LOW PIS	H16063	REACTOR	164	H11-P927			N/A	N/A	Y	H19827		
0567	A10	20	B21-N691A	RPV LEVEL 1 LIS	H16063	CONTROL	164	H11-P925			N/A	N/A	Y	H19823		
0722	A10	20	B21-N691C	RPV LEVEL 1 LIS	H16063	CONTROL	164	H11-P925			N/A	N/A	Y	H19823		
0727	A10	20	B21-N693A	RPV LEVEL 8 LS	H16063	CONTROL	164	H11-P925			N/A	N/A	Y	H19823		
0569	A10	20	B21-N693C	RPV LEVEL 8 LS	H16063	CONTROL	164	H11-P925			N/A	N/A	N	H19823		
0570	A10	20	B21-N695A	RPV LEVEL 3 LIS	H16063	CONTROL	164	H11-P925			N/A	N/A	N	H19823		
0571	A10	20	B21-R604A	RPV LEVEL (HOT LEG) LI	H16063	CONTROL	164	H11-P603			N/A	N/A	Y	H19823,H19835		
0725	A10	20	B21-R615	RPV SHROUD LEVEL LR	H16063	CONTROL	164	H11-P601			N/A	N/A	Y	H19835		
0573	A10	20	B21-R623A	RPV LEVEL/PRESSURE RECORDER	H16063	CONTROL	164	H11-P601			N/A	N/A	Y	H19822,19835		
0535	A11	20	T47-R611	DRYWELL COOLING TEMP RECORDER	H16007	CONTROL	164	H11-P657			N/A	N/A	Y	H19677		
0536	A11	20	T48-K620	TEMP SIGNAL R/V CONVERTER	H16395	CONTROL	164	H11-P691			N/A	N/A	Y			
0537	A11	19	T48-N009A	TORUS WATER TEMP ELEMENT	H16024	REACTOR	087	RA/R07			N/A	N/A	Y	H17083		
0538	A11	19	T48-N009C	TORUS WATER TEMP ELEMENT	H16024	REACTOR	087	RL/R07			N/A	N/A	Y	H17083		
0539	A11	18	T48-N010A	TORUS WATER LEVEL TRANS	H16024	REACTOR	087	RF/R13	S		N/A	N/A	Y	H17083		
0540	A11	20	T48-R622A	TORUS WATER LEVEL IND	H16024	CONTROL	164	H11-P657			N/A	N/A	Y	H17083		
0764	A12	20	H21-P257	GEN 1A HEAT & VENT CONTROL PANEL	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		
0820	A12	0	R34-S004A	NEUTRAL RESISTOR 1A		DIESEL	130		S		N/A	N/A				
0629	A12	21	R43-A001A	FUEL DAY TANK 1A	H11037	DIESEL	130	D/01	S		N/A	N/A	N	N/A		
0630	A12	21	R43-A002A	FUEL STORAGE TANK 1A	H11037	YARD			S		N/A	N/A	N	N/A		
0627	A12	21	R43-A003A	AIR RECEIVER	H11631	DIESEL	130	D/01	S		N/A	N/A	N	N/A		
0628	A12	21	R43-A007A	AIR RECEIVER	H11631	DIESEL	130	D/01	S		N/A	N/A	N	N/A		
0600	A12	08B	R43-F015A	DIESEL AIR START SOLENOID VALVE	H11631	DIESEL	130	C/01	R		CLOSED	OPEN	Y	H13412		
0601	A12	08B	R43-F016A	DIESEL AIR START SOLENOID VALVE	H11631	DIESEL	130	C/01	R		CLOSED	OPEN	Y	H13412		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	<--------- EQUIPMENT Building (7)	Flr.Elv. (8)	LOCATION -----> Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	<-- OP. ST. --> Normal (12)	Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)	REG. ISSUE (17)
0602	A12	08B	R43-F017A	DIESEL AIR VENT SOLENOID VALVE	H11631	DIESEL	130	C/01	R		OPEN	CLOSED	Y	H13412		
0622	A12	20	R43-P001A	DSL GEN 1A CONT PNL	N/A	DIESEL	130	D/02	SR		N/A	N/A	Y	H13412		
0524	A12	17	R43-S001A	DIESEL GENERATOR 1A	H11631	DIESEL	130	C/01			OFF	ON		H13412		
1106	A12	20	X43-P006A	PNEU-ELECTRO RELAY CAB	H13026	DIESEL	130	C/01	S		N/A	N/A	N		X41-C005A	
0625	A12	06	Y52-C001A	DSL 1A FUEL OIL PUMP 1A1	H11637	YARD			SR		OFF	OFF/ON	Y	H13584		
0001	A13	14	C71-P001	RPS POWER DISTRIBUTION PANEL BUS A	N/A	CONTROL	130	TG/T12	SR		N/A	N/A	Y	H13369		
0840	A13	18	C71-P003E	PROTECTION PANEL	N/A	CONTROL	130	TF/T12	SR		N/A	N/A	Y	H13369		
0841	A13	18	C71-P003F	PROTECTION PANEL	N/A	CONTROL	130	TF/T12	SR		N/A	N/A	Y	H13369		
0839	A13	18	C71-S002	LINE VOLTAGE REGULATOR	N/A	CONTROL	130		SR		N/A	N/A	Y	H13369		
0817	A13	18	H21-P248	250V DC SWITCHGEAR CONTROL PANEL	N/A	DIESEL	130	TF/T11	S		N/A	N/A	N	H13592		
0092	A13	18	H21-P285	125/250V STATION BATT 1A FUSE BOX	N/A	CONTROL	112	TF/T11	S		N/A	N/A	Y	H13370		
0093	A13	18	H21-P286	125/250V STATION BATT 1A FUSE BOX	N/A	CONTROL	112	TE/T11	S		N/A	N/A		H13370		
0094	A13	18	H21-P287	125/250V STATION BATT 1A FUSE BOX	N/A	CONTROL	112	TF/T11	S		N/A	N/A	Y	H13370		
0714	A13	18	H21-P294	TERMINAL BOX	N/A	CONTROL	112	TE/T12	S		N/A	N/A	Y	H13635		
0010	A13	04	R11-S004	45KVA 600-120/208V PWR XFMR 1D	N/A	DIESEL	130	D/02	S		N/A	N/A	Y	H13647		
0013	A13	04	R11-S039	45KVA 600-120/208V TRANSFORMER	N/A	REACTOR	130	RF/R13	S		N/A	N/A	Y	H17016		
0015	A13	04	R11-S041	112.5 KVA 600-120/208V ESSENTIAL XFMR 1B	N/A	CONTROL	130	TF/T11	S		N/A	N/A	Y	H13369		
0017	A13	03	R22-S005	4160V SWGR EMERGENCY BUS 1E	N/A	DIESEL	130	D/03	SR		N/A	N/A	Y	H13356		
0020	A13	02	R22-S016	250V DC BATTERY DIV 1 SWGR 1A	N/A	CONTROL	130	TF/T11	SR		N/A	N/A	Y	H13370		
0022	A13	02	R23-S003	600V SWGR EMERGENCY BUS 1C & 4160-600V XFMR	N/A	CONTROL	130	TE/T10	SR		N/A	N/A	Y	H13361		
0024	A13	01	R24-S009	600/208V MCC 1A	N/A	INTAKE	111		S		N/A	N/A	Y	H13389		
0026	A13	01	R24-S011	600V ESS DIV 1 MCC 1C	N/A	REACTOR	130	RH/R13	S		N/A	N/A	Y	H17016,H17010		
0028	A13	01	R24-S018A	600V ESS DIV 1 MCC 1E-A	N/A	REACTOR	130	RL/R08	S		N/A	N/A	Y	H17012		
0033	A13	01	R24-S025	600/208V ESS DIV 1 MCC 1A	N/A	DIESEL	130	D/03	S		N/A	N/A	Y	H13647		
0036	A13	14	R25-S001	125V DC DIV 1 CAB 1A	N/A	CONTROL	130	TE/T11	S		N/A	N/A	Y	H13370		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION ------> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0038	A13	14	R25-S004	125V DC CAB 1D	N/A	DIESEL	130	D/02	S		N/A	N/A	Y	H13371		
0519	A13	14	R25-S015	24/48 VDC CABINET 1A	N/A	CONTROL	130	TG/T12	S		N/A	N/A	Y	H13635		
0041	A13	14	R25-S029	120/208V AC CAB 1J	N/A	DIESEL	130	D/02	S		N/A	N/A	Y	H13362		
0044	A13	14	R25-S036	120/208V AC ESS CAB 1A	N/A	CONTROL	130	TF/T10	S		N/A	N/A	Y	H13369		
0047	A13	14	R25-S064	120/208V AC VITAL CAB 1A INSTR BUS 1A	N/A	CONTROL	130	TG/T11	S		N/A	N/A	Y	H13369		
0049	A13	14	R25-S105	125V DC CAB 1D ESS DIV I	N/A	CONTROL	130	TE/T11	S		N/A	N/A	Y	H19836		
0051	A13	18	R25-S110	120/208V CAB 1A (R25-S064) FUSE BOX	N/A	CONTROL	130		S		N/A	N/A	Y	H13369		
0053	A13	18	R25-S112	120/208V MCC-1A (R24-S025) FUSE BOX	N/A	DIESEL	130	D/02	S		N/A	N/A	Y	H13647		
0056	A13	0	R25-S125	120/208V AC ESS DIV 1 CAB	N/A	REACTOR	130	RF/R13			N/A	N/A	Y	H17016		
0838	A13	18	R26-M021	RPS DIST CAB THROWOVER SW	N/A	CONTROL	130	TG/T12	S		N/A	N/A	Y	H13369		
0059	A13	18	R26-M031A	125V DC 600A THROWOVER SWITCH 1A	N/A	CONTROL	130	TE/T11	S		N/A	N/A	N	H13370		
0060	A13	18	R26-M031B	125V DC 600A THROWOVER SWITCH 1B	N/A	CONTROL	130	TE/T11	S		N/A	N/A	N	H13370		
0063	A13	18	R26-M032A	[illegible] DC THROWOVER SWITCH 1E	N/A	DIESEL	130	D/02	S		N/A	N/A	N	H13371		
0513	A13	18	R26-M041A	24 VDC THROWOVER SW 1A	N/A	CONTROL	130	TG/T12	S		N/A	N/A	N	H13635		
0514	A13	18	R26-M041B	24 VDC THROWOVER SW 1B	N/A	CONTROL	130	TG/T12	S		N/A	N/A	N	H13635		
0066	A13	18	R26-M077	600V BREAKER	N/A	CONTROL	147	TH/T14	S		N/A	N/A	Y	H17012		
0107	A13	01	R27-S005	LOCAL STARTER FOR E11-F017A	N/A	REACTOR	130	RL/R09	S		OFF	ON	Y	H17010		
0791	A13	0	R34-S005A	SURGE PROT PANEL FOR P41-C001A	N/A	INTAKE	111		S		N/A	N/A	Y	SX25148		
0793	A13	0	R34-S006A	SURGE PROT PANEL FOR E11-C001A	N/A	INTAKE	111		S		N/A	N/A	Y	SX25148		
0068	A13	15	R42-S001A	125/250V STATION BATTERY 1A	N/A	CONTROL	112	TE/T11	S		N/A	N/A	Y	H13370		
0070	A13	15	R42-S002A	125V DIESEL SYSTEM BATTERY 1A	N/A	DIESEL	130	D/02	S		N/A	N/A	Y	H13371		
0712	A13	15	R42-S017A	BATTERY 1A 24/48 V	N/A	CONTROL	112	TC/T11	S		N/A	N/A	Y	H13635		
0073	A13	15	R42-S026	125V BATTERY CHARGER 1A	N/A	CONTROL	130	TE/T11	S		N/A	N/A	Y	H13370		
0074	A13	16	R42-S027	125V BATTERY CHARGER 1B	N/A	CONTROL	130	TE/T11	S		N/A	N/A	Y	H13370		
0079	A13	16	R42-S032A	125V BATTERY CHARGER 1G	N/A	DIESEL	130	D/02	S		N/A	N/A	Y	H13371		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION ------> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0507	A13	16	R42-S051	BATTERY CHARGER 1A	N/A	CONTROL	130	TG/T12	S		N/A	N/A	Y	H13635		
0509	A13	16	R42-S052	BATTERY CHARGER 1B	N/A	CONTROL	130	TG/T12	S		N/A	N/A	Y	H13635		
0099	A13	17	R43-S001A	DIESEL GENERATOR 1A	N/A	DIESEL	130	C/02	SR		OFF	ON	Y	H13350		
0085	A13	16	R44-S002	DC/AC INVERTER FOR MCC 1E-A	N/A	CONTROL	147	TI/T14	S		N/A	N/A	Y	H17012		
0517	A13	18	TB1-139	TERMINAL BOX	N/A	CONTROL	130	TG/T12	S		N/A	N/A	N	H13635		
0089	A13	18	TB1-211	125V DC BATTERY 1A FUSE BOX	N/A	DIESEL	130	D/02	S		N/A	N/A	Y	H13371		
0716	A13	18	TB1-229-7	TERMINAL BOX	N/A	DIESEL	130	D/02	S		N/A	N/A	N	H13371		
0991	A14	07	2P41-F066	2P64/2E51 Cooler Isolation	H-26050	REACTOR	120	RB/R19	R		OPEN	OPEN	N	H-27748		
1113	A14	20	H21-P530A	FAN 1A PANEL	H13610	INTAKE	111	N/A	S		N/A	N/A	N		X41-C009A	
1115	A14	20	H21-P530C	FAH 1C PANEL	H13028	INTAKE	111	N/A	S		N/A	N/A	N		X41-C009C	
0972	A14	08B	P41-F123B	Z41-B025 PSW INLET ISOL SV	H11611	CONTROL	180	TB/T11	SR		OPEN	CLOSED	N			
0973	A14	08A	P41-F420A	Z41-B008A ISOL GLOBE MOV	H11609	CONTROL	180	TB/T11	R		OPEN	OPEN	N	H13388		
0974	A14	08A	P41-F421A	Z41-B008B ISOL GLOBE MOV	H11609	CONTROL	180	TB/T11	R		OPEN	OPEN	N	H13388		
0975	A14	08A	P41-F422A	Z41-B008B ISOL GLOBE MOV	H11609	CONTROL	180	TB/T11	RS		OPEN	CLOSED	Y	H13388		
0990	A14	08A	P41-F422B	Z41-B008C ISOL GLOBE MOV	H11609	CONTROL	180	TB/T12			OPEN	OPEN	N	H13388		
0976	A14	18	P41-N520	PSW CB A/C-1B DI DPS	H11609	CONTROL	180	TB/T11	S		N/A	N/A	Y	H13388		
0827	A14	09	X41-C002A	DG ROOM 1A FAN	H12619	DIESEL	150		SR		ON/OFF	ON	Y	H13395		
0860	A14	0	X41-C005A	DG ROOM 1A LOUVER	H12619	DIESEL	130		SR		OP/CL	OPEN	Y	H13395		
0824	A14	09	X41-C006A	DG SWITCHGEAR ROOM 1E FAN	H12619	DIESEL	150		SR		ON/OFF	ON	Y	H13395		
0833	A14	0	X41-C007A	DG SWITCHGEAR ROOM 1E LOUVER	H12619	DIESEL	130		SR		OP/CL	OPEN	Y	H13395		
1089	A14	09	X41-C008A	BATTERY ROOM 1A FAN	H12619	DIESEL	150		SR		ON/OFF	ON	Y	H13395		
0985	A14	09	X41-C009A	INTAKE STRUCTURE VENT FAN 1A	H44073	INTAKE	150		SR		ON/OFF	ON	Y	H13610		
0986	A14	09	X41-C009C	INTAKE STRUCTURE VENT FAN 1C	H44073	INTAKE	150				ON/OFF	ON	Y	H13610		
1090	A14	0	X41-C027A	BATTERY ROOM 1A LOUVER	H12619	DIESEL	130	C/01	SR		OP/CL	OPEN	Y	H13395		
0989	A14	0	X41-C031A	INTAKE STRUCTURE AUTO DAMPER	H44073	INTAKE		WET PIT AREA			OPEN	OPEN	N			
1000	A14	0	X41-C031B	INTAKE STRUCTURE AU'O DAMPER	H44073	INTAKE		WET PIT AREA			OPEN	OPEN	N			

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCHIRS.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- EQUIPMENT Building	Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
1001	A14	0	X41-C032A	INTAKE STRUCTURE AUTO GAMPER	H44073	INTAKE		WET PIT AREA			OPEN	OPEN	N			
1002	A14	0	X41-C032B	INTAKE STRUCTURE AUTO DAMPER	H44073	INTAKE		WET PIT AREA			OPEN	OPEN	N			
0987	A14	18	X41-N002A	INTK STRUC VENT FAN 1A THERMOSTAT	H44073	INTAKE	111	PUMP RM	SR		N/A	N/A	Y	H13610		
0988	A14	18	X41-N002C	INTK STRUC VENT FAN 1C THERMOSTAT	H44073	INTAKE	111	PUMP RM			N/A	N/A	Y	H13610		
0885	A14	18	X41-N004A	D/G RM 1A FAN THERMOSTAT	H12619	DIESEL	130	C/01	S		N/A	N/A	N	H13395		
0888	A14	18	X41-N005A	D/G RM 1A FAN THERMOSTAT	H12619	DIESEL	130	D/02	S		N/A	N/A	N	H13395		
1109	A14	18	X41-N006A	SWGR RM 1E FAN THERMOSTAT	H12619	DIESEL	130	C/03	S		N/A	N/A	Y		X41-C006A	
0891	A14	20	X41-N045A	D/G RM 1A FAN FLOW SWITCH	H12619	DIESEL	130	C/02			N/A	N/A	N	H13395		
0892	A14	20	X41-N045B	D/G RM 1A FAN FLOW SWITCH	H12619	DIESEL	130	D/02			N/A	N/A	N	H13395		
0939	A14	10	Z41-B003B	CONTROL ROOM AIR HANDLING UNIT	H16042	CONTROL	180	TE/T12	SR		OFF	ON	Y	H17068		
0936	A14	10	Z41-B003C	CONTROL ROOM AIR HANDLING UNIT	H16042	CONTROL	180	TF/T12			ON	ON	Y	H17068		
0940	A14	11	Z41-B008B	B003B CONDENSING UNIT	H16042	CONTROL	180	TB/T11	SR		N/A	N/A	Y	H17068		
0937	A14	11	Z41-B008C	B003C CONDENSING UNIT	H16042	CONTROL	180	TB/T12	SR		N/A	N/A	Y	H17068		
0941	A14	09	Z41-C012B	D004B BOOSTER FAN	H16042	CONTROL	180	TG/T12	SR		ON	ON	Y	H17069		
1097	A14	09	Z41-C014	BATT ROOM EMERGENCY EXH	H16040	CONTROL	112	TD/T12	SR		OFF	ON	Y	H17067		
0938	A14	0	Z41-D004B	CONTROL ROOM FILTER TRAIN	H16042	CONTROL	180	TG/T13	S		N/A	N/A	N/A	H17072		
0943	A14	07	Z41-F008B	AIR OPERATED DAMPER B003B IN	H16042	CONTROL	180	TE/T12	SR		CLOSED	OPEN	N	H17070		
0944	A14	07	Z41-F008C	AIR OPERATED DAMPER B003C IN	H16042	CONTROL	180	TF/T12	SR		CLOSED	OPEN	N	H17070		
0945	A14	07	Z41-F009B	AIR OPERATED DAMPER B003B IN	H16042	CONTROL	160	TE/T12	SR		CLOSED	OPEN	N	H17070		
0946	A14	07	Z41-F009C	AIR OPERATED DAMPER B003C IN	H16042	CONTROL	180	TF/T12			OPEN	OPEN	N	H17070		
0947	A14	07	Z41-F010A	AIR OPERATED DAMPER B003B IN	H16042	CONTROL	180	TE/T04	SR		CLOSED	OPEN	N	H17070		
0948	A14	07	Z41-F010B	AIR OPERATED DAMPER B003B IN	H16042	CONTROL	180	TE/T04	R		CLOSED	CLOSED	Y	H17070		
0949	A14	07	Z41-F012	AIR OPER DMPR D004A/B BYPASS	H16042	CONTROL	180	TF/T12	SR		OPEN	CLOSED	N	H17070		
0957	A14	07	Z41-F013B	AIR OPERATED DAMPER D004B IN	H16042	CONTROL	180	TH/T12	SR		CLOSED	OPEN	N	H17071		
0953	A14	07	Z41-F014B	AIR OPERATED DAMPER D004B IN	H16042	CONTROL	180	TH/T12	SR		CLOSED	OPEN	N	H17070		
0955	A14	07	Z41-F016	AIR OPERATED DAMPER D003 ISOL	H16042	CONTROL	180	TH/T12	R		OPEN	OPEN	N	H17071		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0956	A14	07	Z41-F018A	AIR OPERATED DAMPER C011A IN	[illegible]042	CONTROL	180	TE/T12	SR		OPEN	CLOSED	N	H17070		
0951	A14	07	Z41-F018B	AIR OPERATED DAMPER C011B IN	H16[illegible]	CONTROL	180	TE/T12	SR		OPEN	CLOSED	N	H17070		
0960	A14	07	Z41-F028A	AIR OPERATED DAMPER B010B OUT	H16042	CONTROL	180	TE/T13	SR		CLOSED	OPEN	N	H17070		
0962	A14	07	Z41-F030C	AIR OPERATED DAMPER B010C OUT	H16042	CONTROL	180	TF/T13			OPEN	OPEN	N	H17070		
1099	A14	0	Z41-FD-F004	FIRE DMPR STN BATTERY 1A	H16041	CONTROL	112	TE/T11	SR		OPEN	OPEN	N			
1098	A14	0	Z41-FD-F020	FIRE DMPR STN BATTERY 1A	H16041	CONTROL	112	TF/T11	SR		OPEN	OPEN	N			
096[illegible]	A14	18	Z41-N003B	B003B DISCHARGE FS	H16042	CONTROL	180	TE/T13	SR		N/A	N/A	Y	H17073		
0966	A14	18	Z41-N003C	B003C DISCHARGE FS	H16042	CONTROL	180	TF/T13	SR		N/A	N/A	Y	H17073		
0964	A14	18	Z41-N005B	C012B DISCHARGE FS	H16042	CONTROL	180	TF/T12	SR		N/A	N/A	Y	H17073		
0965	A14	18	Z41-N015B	CONTROL RM OUTSIDE AIR INLET RE	H16042	CONTROL	180	TH/T13	SR		N/A	N/A	Y	H17142		
0968	A14	18	Z41-N600B	B003B COMPRESSOR TIS	H16042	CONTROL	180	TE/T12	SR		N/A	N/A	Y	H17068		
0967	A14	18	Z41-N600C	B003C COMPRESSOR TIS	H16042	CONTROL	180	TF/T12	SR		N/A	N/A	Y	H17068		
0970	A14	18	Z41-R615B	CTRL ROOM OUTSIDE AIR INLET RIS	H16042	CONTROL	164	H11-P654	R		N/A	N/A	Y	H17072		
1081	P/A15	07	G11-F003	DRYWELL FL DR PMP ISO VLV	H16176	REACTOR	087	RE/R03	S		OPEN	CLOSED	N			
1082	P/A15	07	G11-F004	DRYWELL FL DR PMP ISO VLV	H16176	REACTOR	087	RE/R03	S		OPEN	CLOSED	N			
1079	P/A15	07	G11-F019	DRYWELL EQ DR PMP ISO VLV	H16176	REACTOR	087	RB/R05	S		OPEN	CLOSED	N			
1080	P/A15	07	G11-F020	DRYWELL EQ DR PMP ISO VLV	H16176	REACTOR	087	RB/R05	S		OPEN	CLOSED	N			
1077	P/A15	08A	G31-F001	RWCU INBOARD ISO GATE VLV	H16188	DRYWELL	158	AZ 170	S		OPEN	CLOSED	Y			
1078	P/A15	08A	G31-F004	RWCU OUTBOARD ISO VLV	H16188	REACTOR	158	RH/R07	S		OPEN	CLOSED	Y			
1083	P/A15	07	T48-F310	TORUS VAC BRK ISO BTRFLY	H16024	REACTOR	121	RE/R10			CLOSED	CLOSED	Y			
1084	P/A15	07	T48-F311	TORUS VAC BRK ISO BTRFLY	H16024	REACTOR	121	RE/R11			CLOSED	CLOSED	Y			
0810	P01	20	2H11-P652	ELECT AUX PWR CONTROL CONSOLE	N/A	CONTROL	164	N/A	SR		N/A	N/A	N/A	N/A		
0770	P01	20	2H21-P231	DIESEL GEN 2B - RELAY PANEL 2B	N/A	DIESEL	130	N/A	SR		N/A	N/A	N/A	N/A		
0775	P01	20	C82-P002	REMOTE SHUTDOWN PANEL	N/A	REACTOR	158	RH/R05	S		N/A	N/A	N/A	N/A		
0844	P01	20	H11-P601	REACTOR & CONT COOLING & ISOLA CONTROL PANEL	N/A	CONTROL	164	N/A			N/A	N/A	N/A	N/A		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	<---------- EQUIPMENT Building (7)	LOCATION Flr.Elv. (8)	-----> Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	<-- OP. ST. Normal (12)	--> Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)	REG. ISSUE (17)
0845	P01	20	H11-P602	REACTOR WTR CLEAN UP & RECIR CONTROL PANEL	N/A	CONTROL	164	N/A			N/A	N/A	N/A	N/A		
0846	P01	20	H11-P603	REACTOR CONTROL PANEL	N/A	CONTROL	164	N/A			N/A	N/A	N/A	N/A		
0741	P01	20	H11-P611	CHAN B PRI ISO SYS REAC PROT SYS	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0742	P01	20	H11-P612	FW & RECIR INSTRUMENTATION PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0744	P01	20	H11-P618	CHAN B RHR CORE SPRAY RELAY PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0745	P01	20	H11-P620	HPCI RELAY COMPUTER EQUIPMENT PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0747	P01	20	H11-P623	OUTBOARD-PRI CONTAIN ISOLATION RELAYS PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0749	P01	20	H11-P627	CORE SPRAY CONTR PNL DIV II	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0849	P01	20	H11-P628	AUTOMATIC BLOWDOWN RELAY PANEL	N/A	CONTROL	164	N/A			N/A	N/A	N/A	N/A		
0850	P01	20	H11-P650	TURB, FW & COND CONTROL CONSOLE	N/A	CONTROL	164	N/A			N/A	N/A	N/A	N/A		
0751	P01	20	H11-P651	GEN & STA SER CONTROL CONSOLE	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0852	P01	20	H11-P652	ELEC AUX PWR CONTROL CONSOLE	N/A	CONTROL	164	N/A			N/A	N/A	N/A	N/A		
0795	P01	20	H11-P654	GAS TREAT & VENT VERTICAL PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0800	P01	20	H11-P922	ATTS RPS PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0802	P01	20	H11-P924	ATTS RPS PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0804	P01	20	H11-P926	ATTS ECCS PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0806	P01	20	H11-P928	ATTS ECCS PANEL	N/A	CONTROL	164	N/A	S		N/A	N/A	N/A	N/A		
0786	P01	18	H21-P016	CS/HPCI LEAK DET RACK	N/A	REACTOR	130	RF/R04	S		N/A	N/A	N/A	N/A		
0777	P01	18	H21-P019	CORE SPRAY SYSTEM B RACK	N/A	REACTOR	087	RL/R03	S		N/A	N/A	N/A	N/A		
0776	P01	18	H21-P021	RHR CHANNEL B RACK	N/A	REACTOR	087	RL/R03	S		N/A	N/A	N/A	N/A		
0787	P01	18	H21-P036	HPCI LEAK DET RACK	N/A	REACTOR	130	RF/R04	S		N/A	N/A	N/A	N/A		
0753	P01	20	H21-P175	HOT SHUTDOWN PUMP CONTROL PANEL	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		
0755	P01	20	H21-P201	DIESEL GEN 1B - RELAY PANEL 1A	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		
0756	P01	20	H21-P202	DIESEL GEN 1C - RELAY PANEL 1A	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		
0758	P01	20	H21-P231	DIESEL GEN 1B - RELAY PANEL 1B	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	<---------- EQUIPMENT Building (7)	Flr.Elv. (8)	LOCATION -----> Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	<-- OP. Normal (12)	ST. --> Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)	REG. ISSUE (17)
0759	P01	20	H21-P232	DIESEL GEN 1C - RELAY PANEL 1B	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		
0761	P01	20	H21-P246	600 VOLT SWGR 1D CONTROL PANEL	N/A	CONTROL	130	N/A	S		N/A	N/A	N/A	N/A		
0763	P01	20	H21-P256	MOV & FUEL PUMP CTRL PANEL 1B - DIV II	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		
0774	P01	20	H21-P267	MOV CONTROL PANEL 1B-DIV II	N/A	INTAKE	111	N/A	S		N/A	N/A	N/A	N/A		
0768	P01	20	H21-P304	DIESEL 1B LEAD TIMER PANEL	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		
0769	P01	20	H21-P305	DIESEL 1C LEAD TIMER PANEL	N/A	DIESEL	130	N/A	S		N/A	N/A	N/A	N/A		
0782	P01	18	H21-P405A	INSTRUMENT RACK	N/A	REACTOR	158	RH/R03	S		N/A	N/A	N/A	N/A		
0778	P01	18	H21-P405B	INSTRUMENT RACK	N/A	REACTOR	158	RH/R03	S		N/A	N/A	N/A	N/A		
0780	P01	18	H21-P410	INSTRUMENT RACK	N/A	REACTOR	130	RF/R03	S		N/A	N/A	N/A	N/A		
0784	P01	18	H21-P414A	INSTRUMENT RACK	N/A	REACTOR	087	RJ/R02	S		N/A	N/A	N/A	N/A		
0785	P01	18	H21-P414B	INSTRUMENT RACK	N/A	REACTOR	087	RJ/R02	S		N/A	N/A	N/A	N/A		
0788	P01	18	H21-P434	INSTRUMENT RACK	N/A	REACTOR	087	RJ/R02	S		N/A	N/A	N/A	N/A		
0486	P02	08B	C11-D001-120	ROD POSITION CONTROL VALVE	H16064	REACTOR	130	HCU			CLOSED	CLOSED	N			
0487	P02	08B	C11-D001-121	ROD POSITION CONTROL VALVE	H16064	REACTOR	130	HCU			CLOSED	CLOSED	N			
0488	P02	08B	C11-D001-122	ROD POSITION CONTROL VALVE	H16064	REACTOR	130	HCU			CLOSED	CLOSED	N			
0489	P02	08B	C11-D001-123	ROD POSITION CONTROL VALVE	H16064	REACTOR	130	HCU			CLOSED	CLOSED	N			
0483	P02	0	C11-D001-125	SCRAM ACCUMULATOR	H16064	REACTOR	130	HCU			N/A	N/A	N			
0484	P02	07	C11-D001-126	SCRAM INLET VALVE	H16064	REACTOR	130	HCU			CLOSED	OPEN	N			
0485	P02	07	C11-D001-127	SCRAM OUTLET VALVE	H16064	REACTOR	130	HCU			CLOSED	OPEN	N			
0480	P02	07	C11-F035A	SDV VENT VALVE	H16065	REACTOR	130	RH/R12	S		OPEN	CLOSED	N	H17794		
0481	P02	07	C11-F035B	SDV VENT VALVE	H16065	REACTOR	130	RH/R12	S		OPEN	CLOSED	N	H17794		
0482	P02	07	C11-F037	SDV DRAIN VALVE	H16065	REACTOR	130	RB/R04	S		OPEN	CLOSED	N	H17794		
0479	P02	08B	C11-F040A	SDV VENT & DRAIN PILOT VALVE	H16065	REACTOR	130	RA/R05	SR		ENERGZ	D-ENERG	N	H41740		
0524	P02	08B	C11-F040B	SDV VENT & DRAIN PILOT VALVE	H16065	REACTOR	130	RA/R05	SR		ENERGZ	D-ENERG	N	H41740		
0491	P02	08B	C11-F110A	BACKUP SCRAM VALVE	H16065	REACTOR	130	RA/R05	SR		D-ENERG	ENERGZ	Y	H17794		
0492	P02	08B	C11-F110B	BACKUP SCRAM VALVE	H16065	REACTOR	130	RA/R05	SR		D-ENERG	ENERGZ	Y	H17794		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0493	P02	20	C71-S3A	MANUAL SCRAM SWITCH	N/A	CONTROL	164	H11-P603			N/A	N/A	Y	H17791		
0494	P02	20	C71-S3B	MANUAL SCRAM SWITCH	N/A	CONTROL	164	H11-P603			N/A	N/A	Y	H17791		
0813	P03	08B	B21-F003	RPV HEAD VENT	H16062	DRYWELL	193	RF/R07	R		CLOSED	CLOSED	N	H17813		
0701	P03	07	B21-F013A	RPV PRESSURE RELIEF VALVE	H16062	DRYWELL	148	AZ036	SR		CLOSED	OP/CL	Y	H17753		
0702	P03	07	B21-F013B	RPV SAFETY RELIEF VALVE	H16062	DRYWELL	148	AZ054	SR		CLOSED	OP/CL	Y	H17753		
0703	P03	07	B21-F013C	RPV PRESSURE RELIEF VALVE	H16062	DRYWELL	148	AZ064	SR		CLOSED	OP/CL	Y	H17753		
0705	P03	07	B21-F013E	RPV SAFETY RELIEF VALVE	H16062	DRYWELL	148	AZ056	SR		CLOSED	OP/CL	Y	H17753		
0693	P03	07	B21-F022A	INBOARD MSIV	H16062	DRYWELL	130	RB/R07	SR		OPEN	CLOSED	N	H17815		
0694	P03	07	B21-F022B	INBOARD MSIV	H16062	DRYWELL	130	RB/R07	SR		OPEN	CLOSED	N	H17815		
0695	P03	07	B21-F022C	INBOARD MSIV	H16062	DRYWELL	130	RB/R07	SR		OPEN	CLOSED	N	H17815		
0696	P03	07	B21-F022D	INBOARD [illegible]	H11062	DRYWELL	130	RB/R07	SR		OPEN	CLOSED	N	H17815		
1116	P04	18	E11-N055A	RHR PUMP A DISCHARGE PT	H16330	REACTOR	087	RJ/R12	SR		N/A	N/A	Y			
1117	P04	18	E11-N055B	RHR PUMP B DISCHARGE PT	H16329	REACTOR	087	RL/R03	SR		N/A	N/A	Y			
1118	P04	18	E11-N055C	RHR PUMP C DISCHARGE PT	H16330	REACTOR	087	RJ/R12	SR		N/A	N/A	Y			
1119	P04	18	E11-N055D	RHR PUMP D DISCHARGE PT	H16329	REACTOR	087	RJ/R03	SR		N/A	N/A	Y			
1120	P04	18	E11-N056A	RHR PUMP A DISCHARGE PT	H16330	REACTOR	087	RJ/R12	SR		N/A	N/A	Y			
1121	P04	18	E11-N056B	RHR PUMP B DISCHARGE PT	H16329	REACTOR	087	RL/R03	SR		N/A	N/A	Y			
1122	P04	18	E11-N056C	RHR PUMP C DISCHARGE PT	H16330	REACTOR	087	RJ/R12	SR		N/A	N/A	Y			
1123	P04	18	E11-N056D	RHR PUMP D DISCHARGE PT	H16329	REACTOR	087	RJ/R03	SR		N/A	N/A	Y			
1124	P04	18	E11-N094A	DRYWELL PT	H16330	REACTOR	158	RC/R08	SR		N/A	N/A	Y			
1125	P04	18	E11-N094B	DRYWELL PT	H16329	REACTOR	158	RJ/R04	SR		N/A	N/A	Y			
1126	P04	18	E11-N094C	DRYWELL PT	H16329	REACTOR	158	RC/R08	SR		N/A	N/A	Y			
1127	P04	18	E11-N094D	DRYWELL PT	H16329	REACTOR	158	RJ/R04	SR		N/A	N/A	Y			
1130	P04	18	E21-N052A	CS PUMP C001A HIGH PRESS	H16331	REACTOR	087	RJ/R12	SR		N/A	N/A	Y			
1131	P04	18	E21-N052B	CS PUMP C001B HIGH PRESS	H16331	REACTOR	087	RL/R03	SR		N/A	N/A	Y			
1128	P04	18	E21-N055A	CS PUMP C001A HIGH PRESS	H16331	REACTOR	087	RJ/R12	SR		N/A	N/A	Y			

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	<---------- Building (7)	EQUIPMENT Flr.Elv. (8)	LOCATION -----> Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	<-- OP. Normal (12)	ST. --> Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)	REG. ISSUE (17)
1129	P04	18	E21-N055B	CS PUMP C001B HIGH PRESS	H16331	REACTOR	087	RL/R03	SR		N/A	N/A	Y			
0109	P04	05	E41-C001	HPCI PUMP	H16333	REACTOR	087	RG/R01	S		OFF	ON	N	N/A		
0110	P04	05	E41-C002	HPCI TURBINE	H16333	REACTOR	087	RL/R02	S		OFF	RUN	N	H17159		
0111	P04	05	E41-C002-3	HPCI LUBE OIL PUMP	N/A	REACTOR	087	RL/R01	SR		OFF	RUN	Y	H17162		
0112	P04	08A	E41-F001	HPCI TURBINE STEAM SUPPLY VLV	H16332	REACTOR	087	RL/R01	SR		CLOSED	OPEN	Y	H17163		
0113	P04	08A	E41-F002	STEAM SUPPLY INBOARD ISOLATION	H16332	REACTOR	130	RH/R07	R		OPEN	OPEN	N	H17163		
0114	P04	08A	E41-F003	STEAM SUPPLY OUTBOARD ISOLATION	H16332	REACTOR	130	RH/R07	R		OPEN	OPEN	N	H17163		
0115	P04	08A	E41-F004	HPCI PUMP SUCT FROM CST	H16332	REACTOR	087	RH/R01	SR		OPEN	OP/CL	Y	H17164, H19586		
0116	P04	08A	E41-F006	HPCI PUMP INBD DISCH VLV	H16332	REACTOR	087	RA/R07	SR		CLOSED	OPEN	Y	H17164, H19586		
0117	P04	08A	E41-F007	HPCI PUMP OUTBD DISCH VLV	H16332	REACTOR	087	RH/R02	R		OPEN	OPEN	N	H17160, H19586		
0118	P04	08A	E41-F008	BYPASS TEST VALVE TO CST	H16332	REACTOR	087	RH/R02	R		CLOSED	CLOSED	N	H17160		
0119	P04	08A	E41-F012	HPCI MIN FLOW BYPASS VLV	H16332	REACTOR	087	RH/R02	SR		CLOSED	OPEN	Y	H17163		
0120	P04	08A	E41-F026	BAR COND & LUBE OIL COOLER DRN	H16333	REACTOR	087	RG/R01	R		CLOSED	CLOSED	N	H17164		
0121	P04	0	E41-F035	HPCI PCV FOR BAR COND & OIL COOL	H16333	REACTOR	087	RH/R02			OPEN	OPEN	N	N/A		
0122	P04	08	E41-F041	HPCI PUMP SUCT FROM SUPP POOL	H16332	REACTOR	087	RH/R01	SR		CLOSED	OPEN	Y	H17163		
0123	P04	08A	E41-F042	HPCI PUMP SUCT FROM SUPP POOL	H16332	REACTOR	087	RH/R02	SR		CLOSED	OPEN	Y	H17163		
0124	P04	07	E41-F051	SUPP POOL SUCT ISOL VLV	H16332	REACTOR	087	RF/R02	R		OPEN	OPEN	N	H17164		
0125	P04	08A	E41-F059	BAR COND COOLING WATER VLV	H16333	REACTOR	087	RL/R02	SR		CLOSED	OPEN	Y	H17163		
0812	P04	08A	E41-F104	VACUUM BREAKER LINE MOV	H16332	REACTOR	087	RF/R02	R		OPEN	OPEN	N	H17157		
0811	P04	08A	E41-F111	VACUUM BREAKER LINE MOV	H16332	REACTOR	087	RF/R02	R		OPEN	OPEN	N	H19586		
0126	P04	08B	E41-F124	HPCI REMOTE MANUAL TRIP SOLENOID	H16333	REACTOR	087	RL/R02	BOX		CLOSED	CLOSED	Y	H17163		
0127	P04	07	E41-F3052	HPCI TURBINE CONTROL VALVE	H16333	REACTOR	087	RL/R02	BOX		CLOSED	CONTROL	N	H17162		
0128	P04	07	E41-F3053	HPCI TURBINE STOP VALVE	H16333	REACTOR	087	RL/R02	BOX		CLOSED	OPEN	N	H17162		
0129	P04	18	E41-K600	TRANSMITTER POWER SUPPLY H11-P612	H16332	CONTROL	164	H11-P612			N/A	N/A	Y	H17158		
0130	P04	18	E41-K601	DISCH. FLOW SQ. ROOT CONVERTER	H16332	CONTROL	164	H11-P612			N/A	N/A	Y	H17158		
0131	P04	18	E41-K603	DC/AC INVERTER	H16332	CONTROL	164	H11-P612			N/A	N/A	Y	H17158		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION ------> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0132	P04	01	E41-K615	HPCI DISCHARGE CONTROLLER	H16332	CONTROL	164	H11-P612			N/A	N/A	Y	H17158		
0133	P04	18	E41-K616	HPCI DISCHARGE CONTROLLER AMP	H16332	CONTROL	164	H11-P612			N/A	N/A	Y	H17158		
0134	P04	18	E41-N002	CST LOW LEVEL SWITCH	H16332	CST	130	B/03			N/A	N/A	Y	H17159		
0135	P04	18	E41-N003	CST LOW LEVEL SWITCH	H16332	CST	130	D/02			N/A	N/A	Y	H17159		
0136	P04	0	E41-N007	HPCI SYSTEM FLOW ELEMENT	H16332	REACTOR	087	RG/R01			N/A	N/A	N	H17157		
0137	P04	18	E41-N008	HPCI FLOW TRANSMITTER H21-P414A	H16332	REACTOR	087	H21-P414A			N/A	N/A	Y	H17162		
0138	P04	18	E41-N050	HPCI PRESS TRANSMITTER H21-P414B	H16332	REACTOR	087	H21-P414B			N/A	N/A	Y	H19824		
0139	P04	18	E41-N051	HI/LOW FLOW TRANSMITTER H21-P414A	H16332	REACTOR	087	H21-P414A			N/A	N/A	Y	H19824		
0149	P04	18	E41-N053	LOW SUCT PRESS TRANSMITTER	H16332	REACTOR	087	H21-P414B			N/A	N/A	Y	H19824		
0150	P04	18	E41-N055A	TURB EXH VENT PRESS TRANSMITTER	H16333	REACTOR	087	H21-P434			N/A	N/A	N	H19822		
0151	P04	18	E41-N055B	TURB EXH VENT PRESS TRANSMITTER	H16333	REACTOR	087	H21-P414A			N/A	N/A	N	H19825		
0152	P04	18	E41-N055C	TURB EXH VENT PRESS TRANSMITTER	H16333	REACTOR	087	H21-P434			N/A	N/A	N	H19822		
0153	P04	18	E41-N055D	TURB EXH VENT PRESS TRANSMITTER	H16333	REACTOR	087	H21-P414A			N/A	N/A	N	H19825		
0154	P04	18	E41-N056B	TURB EXH PRESS TRANSMITTER	H16333	REACTOR	087	H21-P414B			N/A	N/A	N	H19824		
0155	P04	18	E41-N056D	TURB EXH PRESS TRANSMITTER	H16333	REACTOR	087	H21-P414B			N/A	N/A	N	H19824		
0156	P04	18	E41-N057A	STEAM LINE HIGH DP TRANS	H16332	REACTOR	130	H21-P016			N/A	N/A	N	H19822		
0157	P04	18	E41-N057B	STEAM LINE HIGH DP TRANS	H16332	REACTOR	130	H21-P036			N/A	N/A	N	H19825		
0158	P04	18	E41-N058A	STM HDR LOW PRESS TRANS	H16332	REACTOR	130	H21-P016			N/A	N/A	N	H19822		
0159	P04	18	E41-N058B	STM HDR LOW PRESS TRANS	H16332	REACTOR	130	H21-P036			N/A	N/A	N	H19825		
0160	P04	18	E41-N058C	STM HDR LOW PRESS TRANS	H16332	REACTOR	130	H21-P016			N/A	N/A	N	H19822		
0161	P04	18	E41-N058D	STM HDR LOW PRESS TRANS	H16332	REACTOR	130	H21-P03[illegible]			N/A	N/A	N	H19825		
0140	P04	18	E41-N062B	SUPP POOL HIGH LEVEL TRANSMITTER	H16332	REACTOR	118	H21-P414B			N/A	N/A	N	H19825		
0141	P04	18	E41-N062D	SUPP POOL HIGH LEVEL TRANSMITTER	H16332	REACTOR	118	H21-P434			N/A	N/A	N	H19825		
0142	P04	18	E41-N074	HPCI STOP VLV POS. SWITCH E41-C002	H16333	REACTOR	087	H21-P414A			N/A	N/A	Y	H17159		
0143	P04	18	E41-N650	DISCH PRESS IND. SWITCH	H16332	CONTROL	164	H11-P926	R		N/A	N/A	Y	H19824		
0144	P04	18	E41-N651	FLOW DP INDICATING SWITCH	H16332	CONTROL	164	H11-P926	R		N/A	N/A	Y	H19824		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION ------> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0166	P04	18	E41-N653	LOW SUCT PRESS IND. SWITCH	H16332	CONTROL	164	H11-P926	R		N/A	N/A	Y	H19824		
0167	P04	18	E41-N655A	EXH VENT PRESS IND. SWITCH	H16333	CONTROL	164	H11-P925	R		N/A	N/A	N	H19822		
0168	P04	18	E41-N655B	EXH VENT PRESS IND. SWITCH	H16333	CONTROL	164	H11-P926	R		N/A	N/A	N	H19825		
0169	P04	18	E41-N655C	EXH VENT PRESS IND. SWITCH	H16333	CONTROL	164	H11-P925	R		N/A	N/A	N	H19822		
0170	P04	18	E41-N655D	EXH VENT PRESS IND. SWITCH	H16333	CONTROL	164	H11-P926	R		N/A	N/A	N	H19825		
0171	P04	18	E41-N656B	TURB EXH PRESS IND. SWITCH	H16333	CONTROL	164	H11-P926	R		N/A	N/A	N	H19824		
0172	P04	18	E41-N656D	TURB EXH PRESS IND. SWITCH	H16333	CONTROL	164	H11-P926	R		N/A	N/A	N	H19824		
0173	P04	18	E41-N657A	STM LINE HIGH DP IND SWITCH	H16332	CONTROL	164	H11-P925	R		N/A	N/A	N	H19822		
0174	P04	18	E41-N657B	STM LINE HIGH DP IND SWITCH	H16332	CONTROL	164	H11-P926	R		N/A	N/A	N	H19825		
0175	P04	18	E41-N658A	STM LINE LOW PRESS IND SWITCH	H16332	CONTROL	164	H11-P925	R		N/A	N/A	N	H19822		
0176	P04	18	E41-N658B	STM LINE LOW PRESS IND SWITCH	H16332	CONTROL	164	H11-P926	R		N/A	N/A	N	H19825		
0177	P04	18	E41-N658C	STM LINE LOW PRESS IND SWITCH	H16332	CONTROL	164	H11-P925	R		N/A	N/A	N	H19822		
0178	P04	18	E41-N658D	STM LINE LOW PRESS IND SWITCH	H16332	CONTROL	164	H11-P926	R		N/A	N/A	N	H19825		
0179	P04	18	E41-N660A	STM LINE HIGH DP IND SWITCH	H16332	CONTROL	164	H11-P925	R		N/A	N/A	N	H19822		
0180	P04	18	E41-N660B	STM LINE HIGH DP IND SWITCH	H16332	CONTROL	164	H11-P926	R		N/A	N/A	N	H19825		
0145	P04	18	E41-N662B	SUPP POOL HI LEVEL IND. SWITCH	H16332	CONTROL	164	H21-P414B	R		N/A	N/A	N	H19825		
0146	P04	18	E41-N662D	SUPP POOL HI LEVEL IND. SWITCH	H16332	CONTROL	164	H21-P414B	R		N/A	N/A	N	H19825		
0147	P04	18	E41-R612	HPCI FLOW CONTROLLER	H16332	CONTROL	164	H11-P601	R		N/A	N/A	Y	H17162		
0148	P04	18	E41-R613	HPCI FLOW INDICATOR	H16332	CONTROL	164	H11-P601	R		N/A	N/A	Y	H17162		
1132	P04	20	H11-P614	NUCLEAR STEAM RECORDER VB	H17763	CONTROL	164	TE/T11	S		N/A	N/A	N/A			
0394	P05	06	E21-C001B	CORE SPRAY PUMP B	H16331	REACTOR	087	RL/R02	SR		OFF	ON	Y	H17112		
0395	P05	08A	E21-F001B	TORUS SUCTION VALVE	H16331	REACTOR	087	RH/R02	R		OPEN	OPEN	N	H17111		
0396	P05	08A	E21-F004B	CORE SPRAY OUTBOARD VALVE	H16331	REACTOR	158	RF/R04	R		OPEN	OPEN	N	H17111		
0397	P05	08A	E21-F005B	CORE SPRAY INLET VALVE	H16331	REACTOR	158	RF/R04	SR		CLOSED	OPEN	Y	H17111		
0398	P05	08A	E21-F015B	CORE SPRAY TEST BYPASS VALVE	H16331	REACTOR	123	RL/R02	R		CLOSED	CLOSED	N	H17111		
0399	P05	07	E21-F019B	TORUS SUCTION VALVE	H16331	REACTOR	087	RJ/R03	R		OPEN	OPEN	N	H17110		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0400	P05	08A	E21-F031B	MINIMUM FLOW BYPASS VALVE	H16331	REACTOR	087	RL/R02	SR		OPEN	CLOSED	Y	H17111		
0401	P05	0	E21-N002B	CORE SPRAY FLOW ELEMENT	H16331	REACTOR	087	RL/R02			N/A	N/A	N	H17108		
0402	P05	18	E21-N003B	CORE SPRAY FLOW TRANSMITTER	H16331	REACTOR	087	H21-P019			N/A	N/A	Y	H17110		
0403	P05	18	E21-N051B	CORE SPRAY FLOW TRANSMITTER	H16331	REACTOR	087	RL/R02	S		N/A	N/A	Y	H19831		
0404	P05	18	E21-N651B	CORE SPRAY FLOW TRIP UNIT	H16331	CONTROL	164	H11-P928			N/A	N/A	Y	H19831		
0405	P05	20	E21-R601B	CORE SPRAY FLOW INDICATOR	H16331	CONTROL	164	H11-P601			N/A	N/A	Y	H17110		
0356	P06	21	E11-B001B	RHR HEAT EXCHANGER B	H16329	REACTOR	087	RL/R03	S		N/A	N/A	N	N/A		
0357	P06	06	E11-C001D	RHR SERVICE WATER PUMP 1D	D11004	INTAKE	111		SR		OFF	ON	Y	H17781		
0358	P06	06	E11-C002D	RHR PUMP 2D	H16329	REACTOR	087	RL/R02	SR		OFF	ON	Y	H17782		
0359	P06	08A	E11-F003B	RHR HX B DISCHARGE VALVE	H16329	REACTOR	087	RJ/R03	R		OPEN	OPEN	N	H17774		
0360	P06	08A	E11-F004D	RHR PUMP 2D SUCTION VALVE	H16329	REACTOR	087	RL/R03	R		OPEN	OPEN	N	H17774		
0361	P06	08A	E11-F006D	RHR SDC SUCT ISOL VALVE	H16329	REACTOR	087	RL/R03			CLOSED	CLOSED	N	H17774		
0362	P06	08A	E11-F007B	RHR PUMP 2B & 2D MIN FLOW BYPASS VLV	H16329	REACTOR	087	RL/R03	SR		OPEN	CLOSED	Y	H17775		
0363	P06	08A	E11-F010	RHR HX HDR BYPASS VALVE	H16330	REACTOR	087	RL/R07			CLOSED	CLOSED	N	H17772		
0364	P06	08A	E11-F011B	RHR HX B DRAIN TO SUPP POOL	H16329	REACTOR	087	RL/R03	R		CLOSED	CLOSED	N	H17774		
0393	P06	08A	E11-F015B	RHR LPCI DISCHARGE VALVE	H16329	REACTOR	130	RH/R06	R		CLOSED	CLOSED	N	H17774		
0365	P06	08A	E11-F016B	CONTAINMENT SPRAY DISCH VALVE	H16329	REACTOR	158	RH/R08	R		CLOSED	CLOSED	N	H17774		
0366	P06	08A	E11-F024B	RHR TEST LINE TORUS ISO	H16329	REACTOR	087	RL/R05	SR		CLOSED	OPEN	Y	H17774		
0367	P06	08A	E11-F026B	RHR HX B TO RCIC VALVE	H16329	REACTOR	087	RJ/R03	R		CLOSED	CLOSED	N	H17774		
0368	P06	08A	E11-F027B	SUPP POOL SPRAY VALVE	H16329	REACTOR	087	RL/R05	R		CLOSED	CLOSED	N	H17774		
0369	P06	08A	E11-F028B	RHR INLET TO SUPP POOL VALVE	H16329	REACTOR	087	RL/R05	SR		CLOSED	OPEN	Y	H17774		
0370	P06	08A	E11-F047B	RHR HX B TO INLET VALVE	H16329	REACTOR	087	RJ/R02	R		OPEN	OPEN	N	H17774		
0371	P06	08A	E11-F048B	RHR HX B BYPASS VALVE	H16329	REACTOR	087	RJ/R02	SR		OPEN	CLOSED	Y	H17774		
0372	P06	08A	E11-F049	RHR RADWASTE DISCH ISOL VALVE	H16329	REACTOR	087	RL/R05	R		CLOSED	CLOSED	N	H17817		
0373	P06	07	E11-F065D	RHR PUMP 2D SUPP POOL SUCT VLV	H16329	REACTOR	087	RL/R07	R		OPEN	OPEN	N	H17778		
0374	P06	08A	E11-F068B	RHR HX B TUBE TO SHELL OUTLET	H16329	REACTOR	087	RH/R02	SR		CLOSED	OP/CL	Y	H17775		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0375	P06	08A	E11-F073B	RHR HX B SERVICE WATER DISCH	H16329	REACTOR	087	RL/R07	R		CLOSED	CLOSED	N	H17778		
0376	P06	08A	E11-F091B	HPCI DISCH TO HX B VALVE	H16329	REACTOR	087	RH/R03	R		CLOSEC	CLOSED	N	H17776		
0377	P06	08A	E11-F104B	RHR HX B VENT VALVE	H16329	REACTOR	087	RJ/R02	R		CLOSED	CLOSED	N	H17775		
0378	P06	08A	E11-F119B	RHR HX B BYPASS VALVE	H16329	REACTOR	087	RL/R07	R		CLOSED	CLOSED	N	H17778		
0379	P06	18	E11-K600B	RHR HDR FLOW SQUARE ROOT CONV	H16329	CONTROL	164	H11-P612			N/A	N/A	Y	H17771		
0380	P06	18	E11-N002B	RHR HX B TUBE TO SHELL DP TRANS	H16329	REACTOR	087	H21-P021	S		N/A	N/A	Y	H17777		
0381	P06	0	E11-N006B	RHRSW PUMP B & D DISCH FLOW ELEM	H16329	REACTOR	087	RH/R07			N/A	N/A	N	N/A		
0382	P06	18	E11-N007B	RHRSW HX B FLOW TRANSMITTER	H16329	REACTOR	087	RL/R03	S		N/A	N/A	Y	H17769		
0383	P06	0	E11-N014B	RHR PUMP B & D DISCH FLOW ELEMENT	H16329	REACTOR	087	RH/R05			N/A	N/A	N	H17769		
0384	P06	18	E11-N015B	RHR HX B DISCH HDR FLOW TRANS	H16329	REACTOR	087	H21-P021			N/A	N/A	Y	H17769		
0385	P06	18	E11-N017B	RHR HX B INLET PRESSURE SWITCH	H16329	REACTOR	087	RL/R02	S		N/A	N/A	Y	H17766		
0386	P06	18	E11-N017D	RHR HX B INLET PRESSURE SWITCH	H16329	REACTOR	087	RL/R02	S		N/A	N/A	Y	H17766		
0387	P06	18	E11-N082B	RHR PUMP 2B & 2D FLOW DP TRAN	H16329	REACTOR	087	RL/R05	S		N/A	N/A	Y	H19826		
0388	P06	18	E11-N682B	RHR PUMP 2B & 2D FLOW DP TRIP UNIT	H16329	CONTROL	164	H11-P926			N/A	N/A	Y	H19826		
0389	P06	18	E11-R600B	RHR HX B TUBE TO SHELL DP CONT	H16329	CONTROL	164	H11-P613			N/A	N/A	Y	H17777		
0390	P06	20	E11-R602B	RHRSW HX B INLET FI	H16329	CONTROL	164	H11-P601			N/A	N/A	Y	H17769		
0391	P06	20	E11-R603B	RHR HX B DISCH HDR FLOW IND	H16329	CONTROL	164	H11-P601			N/A	N/A	Y	H17769		
0392	P06	18	E11-S600B	RHR HX B TUBE TO SHELL POS MOD	H16329	CONTROL	164	H11-P612			N/A	N/A	Y	H17760		
1011	P07	21	E11-B001B	RHR HEAT EXCHANGER B	H16329	REACTOR	087	RL/R03			N/A	N/A	N/A			
1032	P07	06	E11-C001D	RHR SERVICE WATER PUMP 1D	D11004	INTAKE	111				OFF	ON	Y			
1006	P07	06	E11-C002D	RHR PUMP 2D	H16329	REACTOR	087	RL/R02			OFF	ON	Y			
1018	P07	08A	E11-F003B	RHR HX B DISCHARGE VALVE	H16329	REACTOR	087	RJ/R03			OPEN	OPEN	N			
1004	P07	08A	E11-F004D	RHR PUMP 2D SUCTION VALVE	H16329	REACTOR	087	RL/R03			OPEN	OPEN	N			
1005	P07	08A	E11-F006D	RHR SDC SUCT ISOL VALVE	H16329	REACTOR	087	RL/R03			CLOSED	CLOSED	N			
1007	P07	08A	E11-F007B	RHR PUMP 2B & 2D MIN FLOW BYPASS VLV	H16329	REACTOR	087	RL/R03			OPEN	CLOSED	Y			
1024	P07	08A	E11-F010	RHR HX HDR BYPASS VALVE	H16330	REACTOR	087	RL/R07			CLOSED	CLOSED	N			

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
1017	P07	08A	E11-F011B	RHR HX B DRAIN TO SUPP POOL	H16329	REACTOR	087	RL/R03			CLOSED	CLOSED	N			
1028	P07	08A	E11-F015B	RHR LPCI DISCHARGE VALVE	H16329	REACTOR	130	RH/R06			CLOSED	CLOSED	N			
1029	P07	08A	E11-F016B	CONTAINMENT SPRAY DISCH VALVE	H16329	REACTOR	158	RH/R08	S		CLOSED	OPEN	Y			
1030	P07	08A	E11-F021B	CONT SPRAY INBRD GATE MOV	H16329	REACTOR	158	RH/R08	S		CLOSED	OPEN	Y			
1026	P07	08A	E11-F024B	RHR TEST LINE TORUS ISO	H16329	REACTOR	087	RL/R05			CLOSED	CLOSED	N			
1016	P07	08A	E11-F026B	RHR HX B TO RCIC VALVE	H16329	REACTOR	087	RJ/R03			CLOSED	CLOSED	N			
1027	P07	08A	E11-F027B	SUPP POOL SPRAY VALVE	H16329	REACTOR	087	RL/R05	S		CLOSED	OPEN	Y			
1025	P07	08A	E11-F028B	RHR INLET TO SUPP POOL VALVE	H16329	REACTOR	087	RL/R05			CLOSED	OPEN	Y			
1009	P07	08A	E11-F047B	RHR HX B TO INLET VALVE	H16329	REACTOR	087	RJ/R02			OPEN	OPEN	N			
1008	P07	08A	E11-F048B	RHR HX B BYPASS VALVE	H16329	REACTOR	087	RJ/R02			OPEN	CLOSED	Y			
1003	P07	07	E11-F065D	RHR PUMP 2D SUPP POOL SUCT VLV	H16329	REACTOR	087	RL/R07			OPEN	OPEN	N			
1039	P07	08A	E11-F068B	RHR HX B TUBE TO SHELL OUTLET	H16329	REACTOR	087	RH/R02			CLOSED	OP/CL	Y			
1033	P07	08A	E11-F073B	RHR HX B SERVICE WATER DISCH	H16329	REACTOR	087	RL/R07			CLOSED	CLOSED	N			
1010	P07	08A	E11-F091B	HPCI DISCH TO HX B VALVE	H16329	REACTOR	087	RH/R03			CLOSED	CLOSED	N			
1012	P07	08A	E11-F104B	RHR HX B VENT VALVE	H16329	REACTOR	087	RJ/R02			CLOSED	CLOSED	N			
1034	P07	08A	E11-F119B	RHR HX B BYPASS VALVE	H16329	REACTOR	087	RL/R07			CLOSED	CLOSED	N			
1020	P07	18	E11-K600B	RHR HDR FLOW SQUART ROOT CONV	H16329	CONTROL	164	H11-P612			N/A	N/A	Y			
1031	P07	20	E11-K603B	POWER SUPPLY	H16330	CONTROL	164	H11-P612			N/A	N/A	Y			
1013	P07	18	E11-N002B	RHR HX B TUBE TO SHELL DP TRANS	H16329	REACTOR	087	H21-P021			N/A	N/A	Y			
1035	P07	18	E11-N007B	RHRSW HX B FLOW TRANSMITTER	H16329	REACTOR	087	RL/R03			N/A	N/A	Y			
1019	P07	18	E11-N015B	RHR HX B DISCH HDR FLOW TRANS	H16329	REACTOR	087	H21-P021			N/A	N/A	Y			
1037	P07	18	E11-N017B	RHR HX B INLET PRESSURE SWITCH	H16329	REACTOR	087	RL/R02			N/A	N/A	Y			
1038	P07	18	E11-N017D	RHR HX B INLET PRESSURE SWITCH	H16329	REACTOR	087	RL/R02			N/A	N/A	Y			
1022	P07	18	E11-N082B	RHR PUMP 2B & 2D FLOW DP TRAN	H16329	REACTOR	087	RL/R05			N/A	N/A	Y			
1023	P07	20	E11-N682B	RHR PUMP 2B & 2D FLOW DP TRIP UNIT	H16329	CONTROL	164	H11-P926			N/A	N/A	Y			
1014	P07	18	E11-R600B	RHR HX B TUBE TO SHELL DP CONT	H16329	CONTROL	164	H11-P613			N/A	N/A	Y			

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
1036	P07	20	E11-R602B	RHRSW HX B INLET FI	H16329	CONTROL	164	H11-P601			N/A	N/A	Y			
1021	P07	20	E11-R603B	RHR HX B DISCH HDR FLOW IND	H16329	CONTROL	164	H11-P601			N/A	N/A	Y			
1015	P07	18	E11-S600B	RHR HX B TUBE TO SHELL POS MOD	H16329	CONTROL	164	H11-P612			N/A	N/A	Y			
0439	P08	06	2P41-C002	PSW STANDBY PUMP 1B DIESEL	H21033	INTAKE	111		SR		OFF	ON	Y	H23779		
0440	P08	07	2P41-F340	PSW ISOL VALVE TO 1B DIESEL	H11600	DIESEL	138	D/01	SR		CLOSED	OPEN	Y	H23779		
0426	P08	06	P41-C001B	PLANT SERVICE WATER PUMP 1B	D11001	INTAKE	111		SR		ON/OFF	ON	Y	H13586		
0837	P08	07	P41-F035B	HPCI ROOM COOLER INLET VALVE	H16011	REACTOR	087	RL/R01	SR		CLOSED	OPEN	Y	H17044		
0428	P08	07	P41-F036B	RHR/CS ROOM COOLER INLET VALVE	H16011	REACTOR	087	RL/R03	SR		CLOSED	OPEN	Y	H17043		
0429	P08	07	P41-F037D	RHR PUMP COOLER 2D INLET VALVE	H16011	REACTOR	087	RL/R03	SR		CLOSED	OPEN	Y	H17087		
0430	P08	07	P41-F066	PSW INLET VALVE	H16011	REACTOR	124	RL/R07	R		OPEN	OPEN	N	H17087		
0441	P08	08A	P41-F303A	PSW VALVE	H11024	YARD					OPEN	OPEN	N	H13388		
0431	P08	08A	P41-F310B	PSW TURBINE BLDG ISOL VALVE	H11600	YARD			SR		OPEN	CLOSED	Y	H13388		
0442	P08	08A	P41-F312	PSW RETURN LINE ISOL VALVE	D11001	INTAKE	098		SR		CLOSED	CLOSED	N	H13388		
0432	P08	08A	P41-F313B	PSW STRAINER B ISOL VALVE	D11001	INTAKE	094		R		CLOSED	CLOSED	N	H13388		
0433	P08	08A	P41-F317B	DIV. II PSW TO DIESEL 1B & 1C	H11600	YARD			R		OPEN	OPEN	N	H13388		
0434	P08	08A	P41-F380B	PSW RX BLDG. ISOL VALVE	H11600	YARD			R		OPEN	OPEN	N	H13388		
0435	P08	08A	P41-F401B	PSW 1C DIESEL ISOL VALVE	H11600	DIESEL	140	E/01	R		OPEN	OPEN	N	H13388		
0443	P08	08A	P41-F402A	PSW 1B DIESEL ISOL VALVE	H11600	DIESEL	140	D/01	R		CLOSED	CLOSED	N	H13388		
0444	P08	08A	P41-F403B	PSW 1B DIESEL ISOL VALVE	H11600	DIESEL	140	D/01	R		CLOSED	CLOSED	N	H13388		
0436	P08	18	P41-N200B	PSW STRAINER DP SWITCH	D11001	INTAKE	087		S		N/A	N/A	N	H13388		
0437	P08	10	T41-B003B	RHR/CS PUMP ROOM COOLER	H16011	REACTOR	087	RL/R03	SR		OFF	ON	Y	H17043		
0836	P08	10	T41-B005B	HPCI PUMP ROOM COOLER	H16011	REACTOR	087	RL/R03	SR		OFF	ON	Y	H17044		
0530	P09	0	B21-A003F	AIR ACCUM FOR RELIEF VALVE F	H16299	DRYWELL	148	AZ295	S		N/A	N/A	N	N/A		
0531	P09	0	B21-A003G	AIR ACCUM FOR RELIEF VALVE G	H16299	DRYWELL	148	AZ266	S		N/A	N/A	N	N/A		
0532	P09	0	B21-A003H	AIR ACCUM FOR RELIEF VALVE H	H16299	DRYWELL	148	AZ331	S		N/A	N/A	N	N/A		
0533	P09	0	B21-A003J	AIR ACCUM FOR RELIEF VALVE J	H16299	DRYWELL	148	AZ320	S		N/A	N/A	N	N/A		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1RS.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	Building (7)	EQUIPMENT Flr.Elv. (8)	LOCATION Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	OP. ST. Normal (12)	OP. ST. Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)	REG. ISSUE (17)
0525	P09	0	P70-A002A	EMERGENCY NITROGEN BOTTLE	H16286	REACTOR	130	RL/R09	S		N/A	N/A	N	N/A		
0526	P09	0	P70-A002B	EMERGENCY NITROGEN BOTTLE	H16286	REACTOR	130	RL/R09	S		N/A	N/A	N	N/A		
0527	P09	0	P70-A002C	EMERGENCY NITROGEN BOTTLE	H16286	REACTOR	130	RL/R09	S		N/A	N/A	N	N/A		
0528	P09	0	P70-F084	EMERGENCY NITROGEN ISOL VALVE	H16286	REACTOR	130	RL/R09	S		CLOSED	OPEN	N	N/A		
0529	P09	0	P70-F141	EMERGENCY NITROGEN ISOL VALVE	H16286	REACTOR	130	RL/R09	S		CLOSED	OPEN	N	N/A		
0574	P10	18	B21-N027	REFUELING LEVEL LT	H16063	REACTOR	158	H21-P405B			N/A	N/A	Y	H17752		
0866	P10	18	B21-N078B	RPV PRESS SCRAM PT	H16063	REACTOR	158	H21-P404D			N/A	N/A	Y	H19812		
0868	P10	18	B21-N078D	RPV PRESS SCRAM PT	H16063	REACTOR	158	H21-P405D			N/A	N/A	Y	H19818		
0870	P10	18	B21-N080B	RPV LEVEL 3 LT	H16063	REACTOR	158	H21-P404D			N/A	N/A	Y	H19812		
0872	P10	18	B21-N080D	RPV LEVEL 3 LT	H16063	REACTOR	158	H21-P405D			N/A	N/A	Y	H19818		
0548	P10	18	B21-N085B	SHROUD LEVEL LT	H16063	REACTOR	130	H21-P410			N/A	N/A	Y	H19826		
0732	P10	18	B21-N090B	RPV LOW PRESSURE PT	H16063	REACTOR	130	H21-P410			N/A	N/A	Y	H19830		
0576	P10	18	B21-N090D	RPV LOW PRESSURE PT	H16063	REACTOR	158	H21-P405A			N/A	N/A	Y	H19830		
0577	P10	18	B21-N091B	RPV LEVEL 2 & 1 LT	H16063	REACTOR	158	H21-P405A			N/A	N/A	Y	H19826		
0728	P10	18	B21-N091D	RPV LEVEL 2 & 1 LT	H16063	REACTOR	158	H21-P405A			N/A	N/A	Y	H19826		
0730	P10	18	B21-N093B	RPV LEVEL 8 LT	H16063	REACTOR	158	RH/R04	S		N/A	N/A	Y	H19826		
0578	P10	18	B21-N095B	RPV LEVEL 3 LT	H16063	REACTOR	158	H21-P405B			N/A	N/A	Y	H19826		
0874	P10	20	B21-N678B	RPV PRESS SCRAM PIS	H16063	CONTROL	164	H11-P922			N/A	N/A	Y	H19812		
0876	P10	20	B21-N678D	RPV PRESS SCRAM PIS	H16063	CONTROL	164	H11-P924			N/A	N/A	Y	H19818		
0878	P10	20	B21-N680B	RPV LEVEL 3 LIS	H16063	CONTROL	164	H11-P922			N/A	N/A	Y	H19812		
0880	P10	20	B21-N680D	RPV LEVEL 3 LIS	H16063	CONTROL	164	H11-P924			N/A	N/A	Y	H19818		
0565	P10	20	B21-N685B	RPV LEVEL 8 LIS	H16063	CONTROL	164	H11-P926			N/A	N/A	Y	H19826		
0733	P10	20	B21-N690B	RPV PRESSURE LOW PIS	H16063	CONTROL	164	H11-P928			N/A	N/A	Y	H19830		
0592	P10	20	B21-N690D	RPV PRESSURE LOW PIS	H16063	CONTROL	164	H11-P928			N/A	N/A	Y	H19830		
0593	P10	20	B21-N691B	RPV LEVEL 1 LIS	H16063	CONTROL	164	H11-P926			N/A	N/A	Y	H19826		
0729	P10	20	B21-N691D	RPV LEVEL 1 LIS	H16063	CONTROL	164	H11-P926			N/A	N/A	Y	H19826		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0863	P10	20	B21-N692B	RPV LEVEL 2 LS	H16063	CONTROL	164	H11-P926			N/A	N/A	Y	H19826		
0864	P10	20	321-N692D	RPV LEVEL 2 LS	H16063	CONTROL	164	H11-P926			N/A	N/A	Y	H19826		
0731	P10	20	B21-N693B	RPV LEVEL 8 LS	H16063	CONTROL	164	H11-P926			N/A	N/A	Y	H19826		
0595	P10	20	B21-N693D	RPV LEVEL 8 LS	H16063	CONTROL	164	H11-P926			N/A	N/A	Y	H19826		
0596	P10	20	B21-N695B	RPV LEVEL 3 LIS	H16063	CONTHOL	164	H11-P926			N/A	N/A	Y	H19826		
0597	P10	20	B21-R604B	RPV LEVEL (HOT LEG) LI	H16063	CONTROL	164	H11-P603			N/A	N/A	Y	H19826,H19835		
0598	P10	20	B21-R605	RPV REFUELING LEVEL LI	H16063	CONTROL	164	H11-P602			N/A	N/A	Y	H17110,H17752		
0572	P10	20	B21-R610	RPV SHROUD LEVEL LI	H16063	CONTROL	164	H11-P601			N/A	N/A	Y	H19826,H19835		
0599	P10	20	B21-R623B	RPV LEVEL/PRESSURE RECORDER	H16063	CONTROL	164	H11-P601			N/A	N/A	Y	H19830,H19835		
0542	P11	20	T47-K600	TEMP SIGNAL R/V CONVERTER	H16394	CONTROL	164	H11-P691			N/A	N/A	Y			
0541	P11	20	T47-R612	DRYWELL COOLING TEMP RECORDER	H16007	CONTROL	164	H11-P654	R		N/A	N/A	Y	H19667		
0543	P11	19	T48-N009B	TORUS WATER TEMP ELEMENT	H16024	REACTOR	087	RF/R02			N/A	N/A	Y	H17083		
0544	P11	19	T48-N009D	TORUS WATER TEMP ELEMENT	H16024	REACTOR	087	RF/R13			N/A	N/A	Y	H17083		
0545	P11	18	T48-N010B	TORUS WATER LEVEL TRANS	H16024	REACTOR	087	RF/R02	S		N/A	N/A	Y	H17083		
0546	P11	20	T48-R622B	TORUS WATER LEVEL IND	H16024	CONTROL	164	H11-P654			N/A	N/A	Y	H17083		
0823	P12	18	2R43B-M01	DIESEL B UNIT 1/2 MODE SWITCH	N/A	DIESEL	130		SR		UNIT1	UNIT1	Y	H23775		
1101	P12	20	H21-P258	GEN 1B H&V CONT PNL	H-13396	DIESEL	130	D01	S		N/A	N/A	N			
1102	P12	20	H21-P259	GEN 1C H&V CONT PNL	H-13397	DIESEL	130	E01	S		N/A	N/A	N			
0821	P12	0	R34-S004B	NEUTRAL RESISTOR 1B		DIESEL	130		S		N/A	N/A	N/A			
0822	P12	0	R34-S004C	NEUTRAL RESISTOR 1C		DIESEL	130		S		N/A	N/A	N/A			
0660	P12	21	R43-A001B	FUEL DAY TANK 1B	H11037	DIESEL	130	E/01	S		N/A	N/A	N	N/A		
0691	P12	21	R43-A001C	FUEL DAY TANK 1C	H11037	DIESEL	130	F/01	S		N/A	N/A	N	N/A		
0661	P12	21	R43-A002B	FUEL STORAGE TANK 1B	H11037	YARD			S		N/A	N/A	N	N/A		
0692	P12	21	R43-A002C	FUEL STORAGE TANK 1C	H11037	YARD			S		N/A	N/A	N	N/A		
0658	P12	21	R43-A003B	AIR RECEIVER	H11631	DIESEL	130	E/01	S		N/A	N/A	N	N/A		
0689	P12	21	R43-A003C	AIR RECEIVER	H11631	DIESEL	130	F/01	S		N/A	N/A	N	N/A		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1RS.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Rcw/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0659	P12	21	R43-A007B	AIR RECEIVER	H11631	DIESEL	130	E/01	S		N/A	N/A	N	N/A		
0690	P12	21	R43-A007C	AIR RECEIVER	H11631	DIESEL	130	F/01	S		N/A	N/A	N	N/A		
0631	P12	08B	R43-F015B	DIESEL AIR START SOLENOID VALVE	H11631	DIESEL	130	D/01	R		CLOSED	OPEN	Y	H23775		
0662	P12	08B	R43-F015C	DIESEL AIR START SOLENOID VALVE	H11631	DIESEL	130	E/01	R		CLOSED	OPEN	Y	H13414		
0632	P12	08B	R43-F016B	DIESEL AIR START SOLENOID VALVE	H11631	DIESEL	130	D/01	R		CLOSED	OPEN	Y	H23775		
0663	P12	08B	R43-F016C	DIESEL AIR START SOLENOID VALVE	H11631	DIESEL	130	E/01	R		CLOSED	OPEN	Y	H13414		
0633	P12	08B	R43-F017B	DIESEL AIR VENT SOLENOID VALVE	H11631	DIESEL	130	D/01	R		OPEN	CLOSED	Y	H13413,H23775		
0664	P12	08B	R43-F017C	DIESEL AIR VENT SOLENOID VALVE	H11631	DIESEL	130	E/01	R		OPEN	CLOSED	Y	H13414		
0653	P12	20	R43-P001B	DSL GEN 1B CONT PNL	N/A	DIESEL	130	E/02	SR		N/A	N/A	Y	H13413,H23775		
0684	P12	20	R43-P001C	DSL GEN 1C CONT PNL	N/A	DIESEL	130	F/02	SR		N/A	N/A	Y	H13414		
0655	P12	17	R43-S001B	DIESEL GENERATOR 1B	H11631	DIESEL	130	E/01			OFF	ON		H13413		
0686	P12	17	R43-S001C	DIESEL GENERATOR 1C	H11631	DIESEL	130	E/01			OFF	ON		H13414		
1107	P12	20	X43-P006B	PNEU-ELECTRO RELAY CAB	H13025	DIESEL	130	D/01	S		N/A	N/A	N		X41-C005B	
1108	P12	20	X43-P006C	PNEU-ELECTRO RELAY CAB	H13025	DIESEL	130	E/01	S		N/A	N/A	N		X41-C005C	
0656	P12	06	Y52-C101B	DSL 1B FUEL OIL PUMP 1B2	H11037	YARD			SR		OFF	OFF/ON	Y	H13584		
0688	P12	06	Y52-C101C	DSL 1C FUEL OIL PUMP 1C2	H11037	YARD			SR		OFF	OFF/ON	Y	H13584		
0790	P13	03	2R22-S006	4160V SWGR EMERGENCY BUS 2F	N/A	DIESEL	130	C/03	SR		ON	ON	N	H23357		
0789	P13	01	2R27-S037	LOCAL STARTER FOR 2P41-C002	N/A	DIESEL	130	C/02	SR		ON	ON	Y	H13648		
0102	P13	14	C71-P001	RPS POWER DISTRIBUTION PANEL BUS B	N/A	CONTROL	130	TG/T12			N/A	N/A	N	H13369		
0818	P13	18	H21-P249	250V DC SWITCHGEAR CONTROL PANEL	N/A	DIESEL	130	TB/T11	S		N/A	N/A	N	H13592		
0095	P13	18	H21-P288	125/250V STATION BATT 1B FUSE BOX	N/A	CONTROL	112	TC/T11	S		N/A	N/A	Y	H13370		
0096	P13	18	H21-P289	125/250V STATION BATT 1B FUSE BOX	N/A	CONTROL	112	TE/T11	S		N/A	N/A	Y	H13370		
0097	P13	18	H21-P290	125/250V STATION BATT 1B FUSE BOX	N/A	CONTROL	112	TD/T10	S		N/A	N/A	Y	H13370		
0715	P13	18	H21-P295	TERMINAL BOX	N/A	CONTROL	112	TE/T12	S		N/A	N/A	Y	H13635		
0011	P13	04	R11-S005	45KVA 600-120/208V PWR XFMR 1E	N/A	DIESEL	130	E/02	S		N/A	N/A	Y	H13648		
0012	P13	04	R11-S006	45KVA 600-120/208V PWR XFMR 1F	N/A	DIESEL	130	F/02	S		N/A	N/A	Y	H13649		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEF 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<--------- EQUIPMENT Building	Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- GP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0014	P13	04	R11-S040	45KVA 600-120/208V TRANSFORMER	N/A	REACTOR	130	RF/R02	S		N/A	N/A	Y	H17017		
0016	P13	04	R11-S042	112.5 KVA 600-120/208V ESSENTIAL XFMR 1C	N/A	CONTROL	130	TD/T11	S		N/A	N/A	Y	H13369		
0815	P13	04	R11-S071	MISC POWER TRANSFORMER	N/A	DIESEL	130		S		N/A	N/A	Y	H13648		
0018	P13	03	R22-S006	4160V SWGR EMERGENCY BUS 1F	N/A	DIESEL	130	E/03	SR		N/A	N/A	Y	H13356		
0019	P13	03	R22-S007	4160V SWGR EMERGENCY BUS 1G	N/A	DIESEL	130	F/03	SR		N/A	N/A	Y	H13357		
0021	P13	02	R22-S017	250V DC BATTERY DIV 2 SWGR 1B	N/A	CONTROL	130	TB/T11	SR		N/A	N/A	Y	H13370		
0023	P13	02	R23-S004	600V SWGR EMERGENCY BUS 1D & 4160-600V XFMR	N/A	CONTROL	130	TC/T10	SR		N/A	N/A	Y	H13361		
1103	P13	01	R24-S002	600V/208V MCC 1B	H13365	CONTROL	180	TE/T11	S		N/A	N/A	Y	H13365		
1104	P13	01	R24-S003	600V/208V MCC 1C	N/A	CONTROL	180	TE/T11	S		N/A	N/A	Y	H13365		
0025	P13	01	R24-S010	600/208V MCC 1B	N/A	INTAKE	111		S		N/A	N/A	Y	H13389		
0027	P13	01	R24-S012	600V ESS DIV 2 MCC 1B	N/A	REACTOR	130	RF/R02	S		N/A	N/A	Y	H17017,H17011		
0029	P13	01	R24-S018B	600V ESS DIV 2 MCC 1E-B	N/A	REACTOR	130	RL/R08	S		N/A	N/A	Y	H17012		
0032	P13	01	R24-S022	125/250V DC ESS DIV 2 MCC 1B	N/A	REACTOR	130	RF/R02	S		N/A	N/A	Y	H17023		
0034	P13	01	R24-S026	600/208V ESS DIV B MCC 1B	N/A	DIESEL	130	E/03	SR		N/A	N/A	Y	H13648		
0035	P13	01	R24-S027	600/208V ESS DIV 2 MCC 1C	N/A	DIESEL	130	F/03	S		N/A	N/A	Y	H13649		
1105	P13	01	R24-S029	600V MCC 1E	N/A	CONTROL	180	TD/T11	S		N/A	N/A	Y	H13364		
1112	P13	01	R24-S031	600V MCC 1G	N/A	CONTROL	180	TB/T11	S		N/A	N/A	Y			
0816	P13	01	R24-S048	600/208V MCC 1D	N/A	DIESEL	130		S		N/A	N/A	Y	H13648		
0037	P13	14	R25-S002	125V DC DIV 2 CAB 1B	N/A	CONTROL	130	TE/T11	S		N/A	N/A	Y	H13370		
0039	P13	14	R25-S005	125V DC CAB 1E	N/A	DIESEL	130	E/02	S		N/A	N/A	Y	H13371		
0040	P13	14	R25-S006	125V DC CAB 1F	N/A	DIESEL	130	F/02	S		N/A	N/A	Y	H13371		
0520	P13	14	R25-S016	24/48 VDC CABINET 1B	N/A	CONTROL	130	TG/T12	S		N/A	N/A	Y	H13635		
0042	P13	14	R25-S030	120/208V AC CAB 1K	N/A	DIESEL	130	E/02	S		N/A	N/A	Y	H13362		
0043	P13	14	R25-S031	120/208V AC CAB 1L	N/A	DIESEL	130	F/02	S		N/A	N/A	Y	H13362		
0862	P13	14	R25-S035	120/208V AC CABINET	N/A	DIESEL	130		S		N/A	N/A	Y	H13648		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCHIR5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- EQUIPMENT Building	Flr.Elv.	LOCATION ------> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0045	P13	14	R25-S037	120/208V AC ESS CAB 1B	N/A	CONTROL	130	TD/T10	S		N/A	N/A	Y	H13369		
0048	P13	14	R25-S065	120/208V AC CAB 1B INSTR BUS 1B	N/A	CONTROL	130	TG/T12	S		N/A	N/A	Y	H13369		
0050	P13	14	R25-S106	125V DC CAB 1E ESS DIV II	N/A	CONTROL	130	TB/T10	S		N/A	N/A	Y	H19836		
0052	P13	18	R25-S111	120/208V CAB 1C (R25-S065) FUSE BOX	N/A	CONTROL	130		S		N/A	N/A	Y	H13369		
0054	P13	18	R25-S113	120/208V MCC-1B (R24-S026) FUSE BOX	N/A	DIESEL	130	E/02	S		N/A	N/A	Y	H13648		
0055	P13	18	R25-S114	120/208V MCC-1C (R24-S027) FUSE BOX	N/A	DIESEL	130	F/02	S		N/A	N/A	Y	H13649		
0057	P13	0	R25-S116	120/208V AC ESS DIV 2 CAB	N/A	REACTOR	130	RF/R02			N/A	N/A	Y	H17017		
0061	P13	18	R26-M031C	125V DC 600A THROWOVER SWITCH 1C	N/A	CONTROL	130	TB/T11	S		N/A	N/A	N	H13370		
0062	P13	18	R26-M031D	125V DC 600A THROWOVER SWITCH 1D	N/A	CONTROL	130	TB/T11	S		N/A	N/A	N	H13370		
0064	P13	18	R26-M032B	125V DC THROWOVER SWITCH 1F	N/A	DIESEL	130	E/02	S		N/A	N/A	N	H13371		
0065	P13	18	R26-M032C	125V DC THROWOVER SWITCH 1G	N/A	DIESEL	130	F/02	S		N/A	N/A	N	H13371		
0515	P13	18	R26-M041C	24 VDC THROWOVER SW 1C	N/A	CONTROL	130	TG/T12	S		N/A	N/A	N	H13635		
0516	P13	18	R26-M041D	24 VDC THROWOVER SW 1D	N/A	CONTROL	130	TG/T12	S		N/A	N/A	N	H13635		
0521	P13	18	R26-M073	DISCONNECT SW FOR C11-F040A	N/A	CONTROL	130	TG/T12	S		N/A	N/A	N	H41740		
0522	P13	18	R26-M074	DISCONNECT SW FOR C11-F040B	N/A	CONTROL	138	TG/T12	S		N/A	N/A	N	H41740		
0067	P13	18	R26-M078	600V BREAKER	N/A	CONTROL	147	TH/T14	S		N/A	N/A	Y	H17012		
0103	P13	01	R27-S035	LOCAL STARTER FOR E41-F006	N/A	REACTOR	130	RH/R02	S		OFF	ON	Y	H19586,H17164		
0104	P13	01	R27-S036	LOCAL STARTER FOR E41-F007	N/A	REACTOR	112	RH/R02	S		OFF	OFF	N	H19586,H17160		
0105	P13	01	R27-S037	LOCAL STARTER FOR E41-F008	N/A	REACTOR	112	RH/R02	S		OFF	OFF	N	H19586,H17160		
0106	P13	01	R27-S066	LOCAL STARTER FOR E41-F002	N/A	REACTOR	130	RL/R04	S		OFF	OFF	N	H17163		
0792	P13	0	R34-S005B	SURGE PROT PANEL FOR P41-C001B	N/A	INTAKE	111		S		N/A	N/A	Y	SX25148		
0794	P13	0	R34-S006D	SURGE PROT PANEL FOR E11-C002D	N/A	INTAKE	111		S		N/A	N/A	Y	SX25148		
0069	P13	15	R42-S001B	125/250V STATION BATTERY 1B	N/A	CONTROL	112	TD/T11	S		N/A	N/A	Y	H13370		
0071	P13	15	R42-S002B	125V DIESEL SYSTEM BATTERY 1B	N/A	DIESEL	130	E/02	S		N/A	N/A	Y	H13371		
0072	P13	15	R42-S002C	125V DIESEL SYSTEM BATTERY 1C	N/A	DIESEL	130	F/02	S		N/A	N/A	Y	H13371		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0713	P13	15	R42-S017B	BATTERY 1B 24/48 V	N/A	CONTROL	112	TC/T12	S		N/A	N/A	Y	H13635		
0076	P13	16	R42-S029	125V BATTERY CHARGER 1D	N/A	CONTROL	130	TB/T11	S		N/A	N/A	Y	H13370		
0077	P13	16	R42-S030	125V BATTERY CHARGER 1E	N/A	CONTROL	130	TB/T11	S		N/A	N/A	Y	H13370		
0080	P13	16	R42-S032B	125V BATTERY CHARGER 1H	N/A	DIESEL	130	E/02	S		N/A	N/A	Y	H13371		
0081	P13	16	R42-S032C	125V BATTERY CHARGER 1J	N/A	DIESEL	130	F/02	S		N/A	N/A	Y	H13371		
0510	P13	16	R42-S053	BATTERY CHARGER 1C	N/A	CONTROL	130	TG/T12	S		N/A	N/A	Y	H13635		
0512	P13	16	R42-S054	BATTERY CHARGER 1D	N/A	CONTROL	130	TG/T12	S		N/A	N/A	Y	H13635		
0100	P13	17	R43-S001B	DIESEL GENERATOR 1B	N/A	DIESEL	130	D/02	SR		OFF	ON	Y	H13350		
0101	P13	17	R43-S001C	DIESEL GENERATOR 1C	N/A	DIESEL	130	E/02	SR		OFF	ON	Y	H13350		
0086	P13	16	R44-S003	DC/AC INVERTER FOR MCC 1E-B	N/A	CONTROL	147	TH/T14	S		N/A	N/A	Y	H17012		
0088	P13	04	S11-S009	4160/600V STA SERV XFMR 1F1	N/A	DIESEL	130	E/03	S		N/A	N/A	Y	H13648		
0819	P13	04	S11-S012	STA SERV XFMR 1F2 4160/600V	N/A	DIESEL	130		S		N/A	N/A		H13648		
0518	P13	18	TB1-140	TERMINAL BOX	N/A	CONTROL	130	TG/T12	S		N/A	N/A	N	H13635		
0090	P13	18	TB1-212	125V DC BATTERY 1B FUSE BOX	N/A	DIESEL	130	E/02	S		N/A	N/A	Y	H13371		
0091	P13	18	TB1-213	125V DC BATTERY 1C FUSE BOX	N/A	DIESEL	130	F/02	S		N/A	N/A	Y	H13371		
0717	P13	18	TB1-230-8	TERMINAL BOX	N/A	DIESEL	130	E/02	S		N/A	N/A	N	H13371		
0718	P13	18	TB1-231-8	TERMINAL BOX	N/A	DIESEL	130	F/02	S		N/A	N/A	N	H13371		
0992	P14	07	2P41-F067	PSW Div. I-II	H-26051	REACTOR	125	RB/R19	R		OPEN	OPEN	N	H-27748		
1114	P14	20	H21-P530B	FAN 1B PANEL	H13610	INTAKE	111	N/A	S		N/A	N/A	N		X41-C009B	
0897	P14	08B	P41-F123A	Z41-B025 PSW INLET ISOL SV	H11611	CONTROL	180	TB/T11	SR		OPEN	CLOSED	N	H23325		
0932	P14	08A	P41-F420A	Z41-B008A ISOL GLOBE MOV	H11609	CONTROL	180	TB/T11			OPEN	OPEN	N	H13388		
0931	P14	08A	P41-F421A	Z41-B008B ISOL GLOBE MOV	H11609	CONTROL	180	TB/T11			OPEN	OPEN	N	H13388		
0933	P14	08A	P41-F422A	Z41-B008B ISOL GLOBE MOV	H11609	CONTROL	180	TB/T11			OPEN	OPEN	N	H13388		
0934	P14	08A	P41-F422B	Z41-B008C ISOL GLOBE MOV	H11609	CONTROL	180	TB/T12	R		OPEN	OPEN	N	H13388		
0935	P14	18	P41-N521	PSW CD A/C-1B DII DPS	H11609	CONTROL	180	TB/T11	SR		N/A	N/A	Y	H13388		
0828	P14	09	X41-C002C	DG ROOM 1B FAN	H12619	DIESEL	150		SR		ON/OFF	ON	Y	H13396		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0829	P14	09	X41-C002E	DG ROOM 1C FAN	H12619	DIESEL	150		SR		ON/OFF	ON	Y	H13397		
0831	P14	0	X41-C005B	DG ROOM 1B LOUVER	H12619	DIESEL	130		SR		OP/CL	OPEN	Y	H13396		
0832	P14	0	X41-C005C	DG ROOM 1C LOUVER	H12619	DIESEL	130		SR		OP/CL	OPEN	Y	H13397		
0825	P14	09	X41-C006C	DG SWITCHGEAR ROOM 1F FAN	H12619	DIESEL	150		SR		ON/OFF	ON	Y	H13396		
0826	P14	09	X41-C006E	DG SWITCHGEAR ROOM 1G FAN	H12619	DIESEL	150		SR		ON/OFF	ON	Y	H13397		
0834	P14	0	X41-C007B	DG SWITCHGEAR ROOM 1F LOUVER	H12619	DIESEL	130		SR		OP/CL	OPEN	Y	H13396		
0835	P14	0	X41-C007C	DG SWITCHGEAR ROOM 1G LOUVER	H12619	DIESEL	130		SR		OP/CL	OPEN	Y	H13397		
1085	P14	09	X41-C008C	BATTERY ROOM 1B FAN	H12619	DIESEL	150		SR		ON/OFF	ON	Y	H13396		
1086	P14	09	X41-C008E	BATTERY ROOM 1C FAN	H12619	DIESEL	150		SR		ON/OFF	ON	Y	H13397		
0977	P14	09	X41-C009B	INTAKE STRUCTURE VENT FAN 1B	H44073	INTAKE	150		SR		ON/OFF	ON	Y	H13610		
0978	P14	09	X41-C009C	INTAKE STRUCTURE VENT FAN 1C	H44073	INTAKE	150		SR		ON/OFF	ON	Y	H13610		
0882	P14	0	X41-C017A	DG ROOM 1A ROLL-UP FIRE DOOR	H12619	DIESEL	130		SR		OPEN	OPEN	N			
0883	P14	0	X41-C017B	DG ROOM 1B ROLL-UP FIRE DOOR	H12619	DIESEL	130		SR		OPEN	OPEN	N			
0884	P14	0	X41-C017C	DG ROOM 1C ROLL-UP FIRE DOOR	H12619	DIESEL	130		SR		OPEN	OPEN	N			
1087	P14	0	X41-C027B	BATTERY ROOM 1B LOUVER	H12619	DIESEL	130	D/01	SR		OP/CL	OPEN	Y	H13396		
1088	P14	0	X41-C027C	BATTERY ROOM 1C LOUVER	H12619	DIESEL	130	E/01	SR		OP/CL	OPEN	Y	H13397		
0981	P14	0	X41-C031A	INTAKE STRUCTURE AUTO DAMPER	H44073	INTAKE		WET PIT AREA			OPEN	OPEN	N			
0982	P14	0	X41-C031B	INTAKE STRUCTURE AUTO DAMPER	H44073	INTAKE		WET PIT AREA			OPEN	OPEN	N			
0983	P14	0	X41-C032A	INTAKE STRUCTURE AUTO DAMPER	H44073	INTAKE		WET PIT AREA			OPEN	OPEN	N			
0984	P14	0	X41-C032B	INTAKE STRUCTURE AUTO DAMPER	H44073	INTAKE		WET PIT AREA			OPEN	OPEN	N			
0979	P14	18	X41-N002B	INTK STRUC VENT FAN 1B THERMOSTAT	H44073	INTAKE	111	PUMP RM	SR		N/A	N/A	Y	H13610		
0980	P14	18	X41-N002C	INTK STRUC VENT FAN 1C THERMOSTAT	H44073	INTAKE	111	PUMP RM	SR		N/A	N/A	Y	H13610		
0886	P14	18	X41-N004B	D/G RM 1B FAN THERMOSTAT	H12619	DIESEL	130	D/01	S		N/A	N/A	N	H13396		
0887	P14	18	X41-N004C	D/G RM 1C FAN THERMOSTAT	H12619	DIESEL	130	E/01	S		N/A	N/A	N	H13397		
0889	P14	18	X41-N005B	D/G RM 1B FAN THERMOSTAT	H12619	DIESEL	130	E/01	S		N/A	N/A	N	H13396		
0890	P14	18	X41-N005C	D/G RM 1C FAN THERMOSTAT	H12619	DIESEL	130	F/02	S		N/A	N/A	N	H13397		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1RS.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
1110	P14	18	X41-N006B	SWGR RM 1F FAN THERMOSTAT	H12619	DIESEL	130	D/03	S		N/A	N/A	Y		X41-C006C	
1111	P14	18	X41-N006C	SWGR RM 1G FAN THERMOSTAT	H12619	DIESEL	130	E/03	S		N/A	N/A	Y		X41-C006E	
0893	P14	20	X41-N045C	D/G RM 1B FAN FLOW SWITCH	H12619	DIESEL	130	C/02			N/A	N/A	N			
0894	P14	20	X41-N045D	D/G RM 1B FAN FLOW SWITCH	H12619	DIESEL	130	D/02			N/A	N/A	N			
0895	P14	20	X41-N045E	D/G RM 1C FAN FLOW SWITCH	H12619	DIESEL	130	E/02			N/A	N/A	N			
0896	P14	20	X41-N045E	D/G RM 1C FAN FLOW SWITCH	H12619	DIESEL	130	D/02			N/A	N/A	N			
0898	P14	10	Z41-B003A	CONTROL ROOM AIR HANDLING UNIT	H16042	CONTROL	180	TD/T12	SR		ON	ON	Y	H17068		
0901	P14	10	Z41-B003C	CONTROL ROOM AIR HANDLING UNIT	H16042	CONTROL	180	TF/T12	SR		ON	ON	Y	H17068		
0899	P14	11	Z41-B008A	B003A CONDENSING UNIT	H16042	CONTROL	180	TB/T11	SR		N/A	N/A	Y	H17068		
0902	P14	11	Z41-B008C	B003C CONDENSING UNIT	H16042	CONTROL	180	TB/T12			N/A	N/A	Y	H17068		
0903	P14	09	Z41-C012A	D004A BOOSTER FAN	H16042	CONTROL	180	TF/T13	SR		ON	ON	Y	H17069		
1091	P14	09	Z41-C015	BATT ROOM EMERGENCY EXH	H16040	CONTROL	112	TD/T11	SR		OFF	ON	Y	H17067		
0900	P14	0	Z41-D004A	CONTROL ROOM FILTER TRAIN	H16042	CONTROL	180	TG/T13	S		N/A	N/A	N/A	H17072		
0904	P14	07	Z41-F007A	AIR OPERATED DAMPER B003A IN	H16042	CONTROL	180	TD/T12	SR		CLOSED	OPEN	N	H17071		
0905	P14	07	Z41-F007C	AIR OPERATED DAMPER B003C IN	H16042	CONTROL	180	TF/T12	SR		CLOSED	OPEN	N	H17071		
0906	P14	07	Z41-F009A	AIR OPERATED DAMPER B003A IN	H16042	CONTROL	180	TE/T12	R		OPEN	OPEN	N	H17070		
0907	P14	07	Z41-F009C	AIR OPERATED DAMPER B003C IN	H16042	CONTROL	180	TF/T12	R		OPEN	OPEN	N	H17070		
0909	P14	07	Z41-F011	AIR OPER DMPR D004A/B BYPASS	H16042	CONTROL	180	TG/T12	SR		OPEN	CLOSED	N	H17070		
0910	P14	07	Z41-F013A	AIR OPERATED DAMPER D004A IN	H16042	CONTROL	180	TH/T13	SR		CLOSED	OPEN	N	H17071		
0912	P14	07	Z41-F014A	AIR OPERATED DAMPER D004A IN	H16042	CONTROL	180	TH/T12	SR		CLOSED	OPEN	N	H17071		
0914	P14	07	Z41-F015	AIR OPERATED DAMPER D003 BYPASS	H16042	CONTROL	180	TH/T12	SR		CLOSED	OPEN	N	H17071		
0916	P14	07	Z41-F017A	AIR OPER INLET VANE C011A IN	H16042	CONTROL	180	TD/T11	R		CLOSED	CLOSED	N	H17068		
0917	P14	07	Z41-F017B	AIR OPER INLET VANE C011B IN	H16042	CONTROL	180	TE/T11	R		CLOSED	CLOSED	N	H17070		
0918	P14	07	Z41-F019	AIR OPERATED DAMPER RESTROOM	H16042	CONTROL	180	TH/T13	SR		OPEN	CLOSED	N	H17071		
0919	P14	07	Z41-F020	AIR OPERATED DAMPER RESTROOM	H16042	CONTROL	180	TH/T13	SR		OPEN	CLOSED	N	H17071		
0920	P14	07	Z41-F030A	AIR OPERATED DAMPER B010A OUT	H16042	CONTROL	180	TF/T13	R		OPEN	OPEN	N	H17070		

APPENDIX E
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 1

Data Base File Name/Date/Time: HATCH1R5.DBF / 10/20/95 / 07:17:44
Sort Criteria: Train,ID Number
Filter Criteria: <none>
Program File Name & Version: SSEM 2.2

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION ------> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS	REG. ISSUE
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)
0921	P14	07	Z41-F030C	AIR OPERATED DAMPER B010C OUT	H16042	CONTROL	180	TF/T13	R		OPEN	OPEN	N	H17070		
1093	P14	0	Z41-FD-F005	FIRE DMPR STN BATTERY 1B	H16041	CONTROL	112	TD/T11	SR		OPEN	OPEN	N			
1092	P14	0	Z41-FD-F006	FIRE DMPR STN BATTERY 1B	H16041	CONTROL	112	TC/T11	SR		OPEN	OPEN	N			
0925	P14	18	Z41-N003A	B003A DISCHARGE FS	H16042	CONTROL	180	TE/T13	SR		N/A	N/A	Y	H17073		
0922	P14	18	Z41-N003C	B003C DISCHARGE FS	H16042	CONTROL	180	TF/T13			N/A	N/A	Y	H17073		
0923	P14	18	Z41-N005A	C012A DISCHARGE FS	H16042	CONTROL	180	TF/T13	SR		N/A	N/A	Y	H17073		
0924	P14	18	Z41-N015A	CONTROL RM OUTSIDE AIR INLET RE	H16042	CONTROL	180	TH/T13	SR		N/A	N/A	Y	H17142		
0926	P14	18	Z41-N600A	B003A COMPRESSOR TIS	H16042	CONTROL	180	TD/T12	SR		N/A	N/A	Y	H17068		
0927	P14	18	Z41-N600C	B003C COMPRESSOR TIS	H16042	CONTROL	180	TF/T12			N/A	N/A	Y	H17068		
0929	P14	20	Z41-R615A	CTRL ROOM OUTSIDE AIR INLET RIS	H16042	CONTROL	164	H11-P657	R		N/A	N/A	Y	H17072		

APPENDIX F
COMPOSITE SAFE SHUTDOWN EQUIPMENT LIST
PLANT HATCH UNIT 2

The Composite Safe Shutdown Equipment List (SSEL) includes all systems/equipment required for safe shutdown of the plant.

The components on the Composite SSEL are sorted by train (path), system, and mark number. The system and path for a particular component can be determined from the alphanumeric designator in column 2 (see table 1).

The evaluation required (seismic and/or A-46 relay) for each system/equipment component is listed in column 10. Although some components occur several times in the Composite SSEL, their evaluation type is listed only once. Therefore, each component appears only once on the Seismic Review SSEL (Appendix B) which is generated from this Composite SSEL database.

The format of the Composite SSEL corresponds to Appendix A of the Seismic Qualification Utility Group Generic Implementation Procedure. Following is a list of column numbers with a description of each.

<u>Column Number</u>	<u>Description</u>
1	Unique line number
2	System and path designator (see table 1)
3	Equipment class (see table 2)
4	Equipment identification number
5	Equipment description
6	Schematic drawing number
7	Building or area in which equipment is located
8	Floor elevation from which equipment can be seen
9	Building grid row/column (or panel number) indicating equipment location
10	Type of evaluation required; that is, seismic (S), relay (R), or rule-of-the-box (Box)

Column Number	Description
11	Not used
12	Normal state of the equipment during normal plant operation:
	OPEN Equipment is normally open.
	CLSD Equipment is normally closed.
	OP/CL Equipment normally changes state from open to closed or from closed to open.
	ON Equipment is on and is normally operating.
	OFF Equipment is off and is normally not operating.
	N/A Not applicable
13	Desired operating state of the equipment to accomplish its safe shutdown function:
	OPEN Equipment should be open.
	CLSD Equipment should be closed.
	OP/CL Equipment should change state from open to closed or from closed to open.
	ON Equipment should be on and operating.
	OFF Equipment should be off and not operating.
	N/A Not applicable
14	Is an external power source required?
	Y - Yes N - No
15	Support system drawing number
16	Not used
17	Not used

APPENDIX F

TABLE 1
SYSTEM/EQUIPMENT AND PATH DESIGNATIONS

1. Primary Path (P)

Train Designator (Column 2)	**System/Equipment Description (Column 5)**
P01	Miscellaneous panels
P02	CRD and reactivity monitoring
P03	Nuclear boiler system
P04	HPCI system
P05	Core spray system
P06	RHR-suppression pool cooling
P08	Plant service water system and emergency room coolers
P09	Drywell air
P10	RPV instrumentation
P11	Suppression pool level and temperature
P12	Diesel generators
P13	Power sources
P14	HVAC systems (control building, intake structure, and diesel building)

2. Alternate Path (A)

Train Designator (Column 2)	**System/Equipment Description (Column 5)**
A01	Miscellaneous panels
A02	CRD and reactivity monitoring
A03	Nuclear boiler system
A05	RHR-LPCI mode
A06	RHR-shutdown cooling mode
A08	Plant service water system and emergency room coolers
A09	Drywell air
A10	RPV instrumentation
A11	Suppression pool level and temperature
A12	Diesel generators
A13	Power sources
A14	HVAC systems (control building, intake structure, and diesel building)

APPENDIX F

TABLE 2
EQUIPMENT CLASS DESIGNATIONS

Equipment Class (Column 3)	Description
01	Motor control centers
02	Low-voltage switchgear
03	Medium-voltage switchgear
04	Transformers
05	Horizontal pumps
06	Vertical pumps
07	Fluid-operated valves
08A	Motor-operated valves
08B	Solenoid-operated valves
09	Fans
10	Air handlers
11	Chillers
12	Air compressors
13	Motor generators
14	Distribution panels
15	Batteries on racks
16	Battery chargers and inverters
17	Engine generators
18	Instruments on racks
19	Temperature sensors
20	Instrumentation and control panels and cabinets
21	Tanks and heat exchangers

EDWIN I. HATCH NUCLEAR PLANT - UNIT 2

SSEL CERTIFICATION

All the information contained on this Safe Shutdown Equipment List (SSEL) identifying equipment required to bring the plant to a safe shutdown condition is, to the best of our knowledge and belief, correct and accurate.

Name / Title	Signature / Date
W. S. Walker / Mechanical Systems Engineer	William Scott Walker / 11-8-94
J. E. Smith / Electrical Systems Engineer	James E Smith

Independent Review:

Name / Title	Signature / Date
W. T. Barr / Mechanical Systems Engineer	W. T. Barr 11/9/94
L. D. McWhorter / Electrical Systems Engineer	Larry D. McWhorter 11-9-94

hsign.doc

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEL v0.0

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	Building (7)	EQUIPMENT Flr.Elv. (8)	LOCATION Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	OP. ST. Normal (12)	OP. ST. Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)
000781	A01	20	2C82-P001	REMOTE SHUTDOWN PANEL	H-27284	REACTOR	130	RA/R16			N/A	N/A	N		
000783	A01	20	2H11-P601	REAC CNTMT COOL ISO BN BD	H-13138	CONTROL	164	TE/T12			N/A	N/A	N		
000633	A01	20	2H11-P602	REAC WTR CLNUP & RECIRC	H-23259	CONTROL	164	TC/T12	S		N/A	N/A	N		
000784	A01	20	2H11-P603	REAC CONTROL BN BD	H-23258	CONTROL	164	TC/T12			N/A	N/A	N		
000636	A01	20	2H11-P605A	CNTMT ATM OIL VERT BD	H-23733	CONTROL	164		S		N/A	N/A	N		
000637	A01	20	2H11-P605B	CNTMT ATM OIL VERT BD	H-23733	CONTROL	164		S		N/A	N/A	N		
000638	A01	20	2H11-P606	STARTUP NEUT MON PNL	H-23259	CONTROL	164	TC/T13	S		N/A	N/A	N		
000639	A01	20	2H11-P608	RNR RNGE NEUT MON PNL	H-23259	CONTROL	164	TC/T13	S		N/A	N/A	N		
000640	A01	20	2H11-P609	CH A PRI ISOL & RPS VB	H-23259	CONTROL	164	TE/T13	S		N/A	N/A	N		
000788	A01	20	2H11-P611	CH B PRI ISOL & RPS VB	H-23247	CONTROL	164	TE/T13			N/A	N/A	N		
000643	A01	20	2H11-P613	PROCESS INST VERT BD	H-23258	CONTROL	164	TC/T13	S		N/A	N/A	N		
000645	A01	20	2H11-P617	CHAN A RHR RELAY VERT BD	H-23258	CONTROL	164	TC/T12	S		N/A	N/A	N		
000648	A01	20	2H11-P622	I[illegible]D ISO VLV VERT PNL	H-13138	CONTROL	164	TC/T13	S		N/A	N/A	N		
000650	A01	20	2H11-P626	CORE SPRAY CTRL PNL DIV 1	H-13138	CONTROL	164	TC/T12	S		N/A	N/A	N		
000789	A01	20	2H11-P628	ADS RELAY PANEL	H-23258	CONTROL	164	TC/T12			N/A	N/A	N		
000790	A01	20	2H11-P650	TURB FDWTR 7 COND PNL	H-23258	CONTROL	164	TC/T12			N/A	N/A	N		
000791	A01	20	2H11-P652	DSL GEN & EMER STA PNL		CONTROL	164	TC/T11			N/A	N/A	N		
000792	A01	20	2H11-P656	TURB AUX SYSTEM VERT PNL	H-23257	CONTROL	164	TC/T11			N/A	N/A	N		
000657	A01	20	2H11-P657	VNT DRYWELL INERT VERT BD	H-23258	CONTROL	164	TC/T12	S		N/A	N/A	N		
000793	A01	20	2H11-P664	MSIV LEAK CTRL SYS PNL		CONTROL	164	TA/T13			N/A	N/A	N		
000794	A01	20	2H11-P674	START UP XFMR 2C PANEL		CONTROL	164	TA/T12			N/A	N/A	N		
000660	A01	20	2H11-P675	START UP XFMR 2D PANEL	H-23666	CONTROL	164	TA/T12	S		N/A	N/A	N		
000795	A01	20	2H11-P700	ANALOG VNT LEAK DETECT PNL	H-23246	CONTROL	164	TE/T13			N/A	N/A	N		
000664	A01	20	2H11-P921	RPS TRIP UNIT CABINET	H-16249	CONTROL	164	TI/T12	S		N/A	N/A	N		
000666	A01	20	2H11-P923	RPS TRIP UNIT CABINET	H-16249	CONTROL	164	TI/T12	S		N/A	N/A	N		
000668	A01	20	2H11-P925	ECCS TRIP UNIT CABINET	H-16249	CONTROL	164	TI/T12	S		N/A	N/A	N		

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000670	A01	20	2H11-P927	ECCS TRIP UNIT CABINET	H-16249	CONTROL	164	TI/T12	S		N/A	N/A	N		
000782	A01	18	2H21-P002	REACTOR WATER CLEANUP PNL	H-26100	REACTOR	158	RF/R24			N/A	N/A	N		
000674	A01	18	2H21-P018	RHR INST RACK CHANNEL A	H-27267	REACTOR	087	RL/R14	S		N/A	N/A	N		
000678	A01	20	2H21-P200	DIESEL GEN 2A RELAY PANEL	H-23071	DIESEL	130	A02	S		N/A	N/A	N		
000680	A01	18	2H21-P220	TURBINE BUILDING INST RACK	H-23067	TURBINE	130	TH/T20	S		N/A	N/A	N		
000682	A01	20	2H21-P230	RELAY PANEL 2A-D/G 2A	H-23071	DIESEL	130	A02	S		N/A	N/A	N		
000683	A01	20	2H21-P231	RELAY PANEL 2B-D/G 2B	H-23071	DIESEL	130	B02	S		N/A	N/A	N		
000685	A01	20	2H21-P245	600 VOLT BUS 2C CNT PNL	H-23240	CONTROL	130	TE/T13	S		N/A	N/A	N		
000689	A01	20	2H21-P255	DG FUEL PMP & MOV CONT PNL	H-23071	DIESEL	130	A02	S		N/A	N/A	N		
000691	A01	20	2H21-P257	D/G 2A HT/VEN CONT PNL	H-23071	DIESEL	130	A01	S		N/A	N/A	N		
000693	A01	20	2H21-P260	SWGR 2E RM HT/VEN CONT PNL	H-23071	DIESEL	130	A02	S		N/A	N/A	N		
000695	A01	20	2H21-P266	MOV CONTROL PNL 2A DIV 1	H-23027	INTAKE	111		S		N/A	N/A	N		
000697	A01	20	2H21-P303	DG 2A LOADING TIMER PANEL	H-23340	DIESEL	130	A03	S		N/A	N/A	N		
000700	A01	18	2H21-P404A	RPV LVL/PRESS INSTR RACK	H-26100	REACTOR	158	RG/R17	S		N/A	N/A	N		
000701	A01	18	2H21-P404B	RPV LVL/PRESS INSTR RACK	H-26100	REACTOR	158	RG/R17	S		N/A	N/A	N		
000702	A01	18	2H21-P404C	RPV LVL/PRESS INSTR RACK	H-26100	REACTOR	158	RG/R17	S		N/A	N/A	N		
000704	A01	18	2H21-P404E	RPV LVL/PRESS INSTR RACK	H-26100	REACTOR	158	RG/R17	S		N/A	N/A	N		
000707	A01	18	2H21-P405C	RPV LVL/PRESS INSTR RACK	H-26100	REACTOR	158	RH/R23	S		N/A	N/A	N		
000710	A01	18	2H21-P409	JET PUMP INSTR RACK	H-26098	REACTOR	130	RF/R15	S		N/A	N/A	N		
000714	A01	18	2H21-P415A	MAIN STM FLOW INSTRUMENT RACK	H-26098	REACTOR	130	RF/R15	S		N/A	N/A	N		
000716	A01	18	2H21-P418A	RHR INSTRUMENT RACK	H-26096	REACTOR	087	RL/R24	S		N/A	N/A	N		
000717	A01	18	2H21-P418B	RHR INSTRUMENT RACK	H-26096	REACTOR	087	RL/R24	S		N/A	N/A	N		
000720	A01	18	2H21-P421B	RHR INSTRUMENT RACK	H-26096	REACTOR	087	RL/R24	S		N/A	N/A	N		
000721	A01	18	2H21-P425A	RHR INSTRUMENT RACK	H-26096	REACTOR	130	RF/R22	S		N/A	N/A	N		
000728	A01	20	2U61-P002	LEAK DETECTION CONTROL PANEL	H-23240	CONTROL	130	TD/T13	S		N/A	N/A	N		
000730	A01	20	2U61-P004	LEAK DETECTION CONTROL PANEL	H-23240	CONTROL	130	TD/T13	S		N/A	N/A	N		

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION ------> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000749	A02	08B	2C11-D001-120	ROD POSITION CONTROL VALVE	H-26006	REACTOR	130	HCU			CLSD	CLSD	N		
000750	A02	08B	2C11-D001-121	ROD POSITION CONTROL VALVE	H-26006	REACTOR	130	HCU			CLSD	CLSD	N		
000751	A02	08B	2C11-D001-122	ROD POSITION CONTROL VALVE	H-26006	REACTOR	130	HCU			CLSD	CLSD	N		
000752	A02	08B	2C11-D001-123	ROD POSITION CONTROL VALVE	H-26006	REACTOR	130	HCU			CLSD	CLSD	N		
000753	A02	0	2C11-D001-125	SCRAM ACCUMULATOR	H-26006	REACTOR	130	HCU			N/A	N/A	N		
000754	A02	07	2C11-D001-126	SCRAM INLET VALVE	H-26006	REACTOR	130	HCU			CLSD	OPEN	N		
000755	A02	07	2C11-D001-127	SCRAM INLET VALVE	H-26006	REACTOR	130	HCU			CLSD	OPEN	N		
000145	A02	07	2C11-F035A	SCRAM DISCH VOL ISOLATION	H-26007	REACTOR	130	RF/R15	RS		OPEN	CLSD	N	H-27615	
000146	A02	07	2C11-F035B	SCRAM DISCH VOL ISOLATION	H-26007	REACTOR	130	RF/R24	RS		OPEN	CLSD	N	H-27615	
000147	A02	07	2C11-F037	SCRAM DISCH VOL DRAIN	H-26007	REACTOR	08	RB/R17	RS		OPEN	CLSD	N	H-27615	
000227	A02	08B	2C11-F040	PILOT AIR HEADER SUPPLY	H-26007	REACTOR	130	RA/R17	RS		ENRG	D-ENRG	N	H-27614	
000148	A02	08B	2C11-F110A	BACKUP SCRAM VALVE	H-26007	REACTOR	130	RA/R21	RS		D-ENRG	ENRG	Y	H-27615	
000144	A02	08B	2C11-F110B	BACKUP SCRAM VALVE	H-26007	REACTOR	130	RA/R21	RS		D-ENRG	ENRG	Y	H-27615	
000149	A02	0	2C71-S3A	MANUAL SCRAM SWITCH		CONTROL	164	2H11-P603			N/A	N/A	N	H-27607	
000150	A02	0	2C71-S3B	MANUAL SCRAM SWITCH		CONTROL	164	2H11-P603			N/A	N/A	N	H-27607	
000151	A02	0	2C71-S3C	MANUAL SCRAM SWITCH		CONTROL	164	2H11-P603			N/A	N/A	N	H-27607	
000152	A02	0	2C71-S3D	MANUAL SCRAM SWITCH		CONTROL	164	2H11-P603			N/A	N/A	N	H-27607	
000133	A03	07	2B21-F003	RPV HEAD VENT VALVE	H-26000	DRYWELL	148	AZ060	R		CLSD	CLSD	N	H-27458	
000008	A03	07	2B21-F013A	RPV SAFETY/RELIEF VALVE	H-26000	DRYWELL	148	AZ235	RS		CLSD	OP/CL	Y	H-27472	
000135	A03	07	2B21-F013B	RPV SAFETY/RELIEF VALVE	H-26000	DRYWELL	148	AZ235	RS		CLSD	OP/CL	Y	H-27937	
000136	A03	07	2B21-F013F	RPV SAFETY/RELIEF VALVE	H-26000	DRYWELL	148	AZ260	RS		CLSD	OP/CL	Y	H-27472	
000009	A03	07	2B21-F013K	RPV SAFETY/RELIEF VALVE	H-26000	DRYWELL	148	AZ260	RS		CLSD	OP/CL	Y	H-27472	
000134	A03	08A	2B21-F016	STEAM DRAIN ISOLATION	H-26000	DRYWELL	127	AZ170	R		CLSD	CLSD	N	H-27450	
000129	A03	07	2B21-F022A	INBOARD MSIV	H-26000	DRYWELL	130	AZ190	RS		OPEN	CLSD	Y	H-27460	
000130	A03	07	2B21-F022B	INBOARD MSIV	H-26000	DRYWELL	130	AZ220	RS		OPEN	CLSD	Y	H-27460	
000131	A03	07	2B21-F022C	INBOARD MSIV	H-26000	DRYWELL	130	AZ160	RS		OPEN	CLSD	Y	H-27989	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	Building	EQUIPMENT Flr.Elv.	LOCATION Rm. or Row/Col.	SORT	NOTES	OP. ST. Normal	OP. ST. Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000132	A03	07	2B21-F022D	INBOARD MSIV	H-26000	DRYWELL	130	AZ175	RS		OPEN	CLSD	Y	H-27989	
000804	A03	18	2E11-N094A	DRYWELL PRESSURE PT	H-26015	REACTOR	158	RE/R16			N/A	N/A	Y	H-24427	
000805	A03	18	2E11-N094C	DRYWELL PRESSURE PT	H-26015	REACTOR	158	RE/R16			N/A	N/A	Y	H-24427	
000179	A05	08A	2B31-F023A	RECIRC PUMP SUCTION ISOLATION	H-26003	DRYWELL	087	AZ340	RS		OPEN	CLSD	Y	H-27488	
000194	A05	21	2E11-B001A	RHR HEAT EXCHANGER A	H-26015	REACTOR	087	RL/R14			N/A	N/A	N/A		
000182	A05	06	2E11-C002A	RHR PUMP 2A	H-26015	REACTOR	087	RL/R14			OFF	ON	Y	H-27657	
000180	A05	08A	2E11-F004A	TORUS SUCTION ISOLATION	H-26015	REACTOR	087	RL/R14			OPEN	OPEN	N	H-27647	
000181	A05	08A	2E11-F006A	SHUTDOWN COOLING SUCTION	H-26015	REACTOR	087	RL/R14			CLSD	CLSD	N	H-27647	
000183	A05	08A	2E11-F007A	RHR PUMP MIN FLOW BYPASS	H-26015	REACTOR	087	RL/R14			OPEN	OP/CL	Y	H-27648	
000189	A05	08A	2E11-F010	RHR HX HDR CROSSTIE VALVE	H-26015	REACTOR	087	RH/R19			CLSD	CLSD	N	H-27647	
000193	A05	08A	2E11-F015A	INBOARD INJECTION	H-26015	REACTOR	130	RJ/R18			CLSD	OPEN	Y	H-27648	
000191	A05	08A	2E11-F016A	CONTAINMENT SPRAY OUTBOARD	H-26015	REACTOR	130	RJ/R21			CLSD	CLSD	N	H-27647	
000192	A05	08A	2E11-F017A	RHR LPCI DISCHARGE VALVE	H-26015	REACTOR	130	RJ/R18			OPEN	OP/CL	Y	H-27648	
000184	A05	08A	2E11-F026A	RHR HX DRAIN TO RCIC	H-26015	REACTOR	087	RL/R14	R		CLSD	CLSD	N	H-27647	
000190	A05	08A	2E11-F028A	TORUS SPRAY OUTBOARD ISOLATION	H-26015	REACTOR	087	RH/R14	R		CLSD	CLSD	N	H-27647	
000187	A05	08A	2E11-F048A	RHR HX BYPASS	H-26015	REACTOR	087	RL/R14			OPEN	OPEN	N	H-27647	
000201	A05	07	2E11-F065A	RHR PUMP 2A SUCTION	H-26015	REACTOR	087	RH/R14	R		OPEN	OPEN	N	H-27653	
000186	A05	08B	2E11-F079A	RHR HX SAMPLE SOLENOID VALVE	H-26015	REACTOR	087	RJ/R23	R		CLSD	CLSD	N	H-27464	
000188	A05	08A	2E11-F091A	HPCI STEAM TO RHR	H-26015	REACTOR	087	RH/R15	R		CLSD	CLSD	N	H-27651	
000185	A05	08A	2E11-F104A	RHR HX VENT	H-26015	REACTOR	087	RL/R14	R		CLSD	CLSD	N	H-27648	
000196	A05	18	2E11-K600A	RHR DISCH HEADER FT SQRT	H-26015	CONTROL	164	2H11-P613	R		N/A	N/A	Y	H-27644	
000200	A05	18	2E11-K603A	POWER SUPPLY	H-26015	CONTROL	164	2H11-P613	R		N/A	N/A	Y	H-27644	
000195	A05	18	2E11-N015A	RHR DISCH HEADER FT	H-26015	REACTOR	087	2H21-P418A	R		N/A	N/A	Y	H-27644	
000198	A05	18	2E11-N082A	RHR PUMP FLOW DPT	H-26015	REACTOR	087	2H21-P418A	R		N/A	N/A	Y	H-24423	
000796	A05	18	2E11-N094A	DRYWELL PRESSURE PT	H-26015	REACTOR	158	RE/R16			N/A	N/A	Y	H-24427	
000797	A05	18	2E11-N094C	DRYWELL PRESSURE PT	H-26015	REACTOR	158	RE/R16			N/A	N/A	Y	H-24427	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000199	A05	18	2E11-N682A	RHR PUMP FLOW DPIS	H-26015	CONTROL	164	2H11-P925	R		N/A	N/A	Y	H-24423	
000197	A05	18	2E11-R603A	RHR DISCH HEADER FI	H-26015	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-27644	
000157	A06	08A	2B31-F023A	RECIRC PUMP SUCTION ISOLATION	H-26003	DRYWELL	087	AZ340			OPEN	CLSD	Y	H-27488	
000172	A06	21	2E11-B001A	RHR HEAT EXCHANGER A	H-26015	REACTOR	087	RL/R14	S		N/A	N/A	N/A		
000208	A06	06	2E11-C001A	RHRSW PUMP 2A	H-21039	INTAKE	111		RS		OFF	ON	Y	H-27656	
000160	A06	06	2E11-C002A	RHR PUMP 2A	H-26015	REACTOR	087	RL/R14	RS		OFF	ON	Y	H-27657	
000221	A06	08A	2E11-F003A	RHR HX OUTLET	H-26015	REACTOR	087	RL/R14	R		OPEN	OPEN	N	H-27647	
000158	A06	08A	2E11-F004A	TORUS SUCTION ISOLATION	H-26015	REACTOR	087	RL/R14	R		OPEN	OPEN	N	H-27647	
000159	A06	08A	2E11-F006A	SHUTDOWN COOLING SUCTION	H-26015	REACTOR	087	RL/R14	R		CLSD	CLSD	N	H-27647	
000206	A06	08A	2E11-F006C	SHUTDOWN COOLING SUCTION	H-26015	REACTOR	087	RL/R14	R		CLSD	CLSD	N	H-27647	
000161	A06	08A	2E11-F007A	RHR PUMP MIN FLOW BYPASS	H-26015	REACTOR	087	RL/R14	RS		OPEN	OP/CL	Y	H-27648	
000167	A06	08A	2E11-F010	RHR HX HDR CROSSTIE VALVE	H-26015	REACTOR	087	RH/R19			CLSD	CLSD	N	H-27647	
000507	A06	08A	2E11-F011A	RHR HX A DRAIN TO SUPP POOL	H-26014	REACTOR	087	RL/R14	R		CLSD	CLSD	N	H-27647	
000171	A06	08A	2E11-F015A	INBOARD INJECTION	H-26015	REACTOR	130	RJ/R18	RS		CLSD	OPEN	Y	H-27648	
000169	A06	08A	2E11-F016A	CONTAINMENT SPRAY OUTBOARD	H-26015	REACTOR	130	RJ/R21	R		CLSD	CLSD	N	H-27647	
000170	A06	08A	2E11-F017A	RHR LPCI DISCHARGE VALVE	H-26015	REACTOR	130	RJ/R18	RS		OPEN	OP/CL	Y	H-27648	
000162	A06	08A	2E11-F026A	RHR HX DRAIN TO RCIC	H-26015	REACTOR	087	RL/R14			CLSD	CLSD	N	H-27647	
000168	A06	08A	2E11-F028A	TORUS SPRAY OUTBOARD ISOLATION	H-26015	REACTOR	087	RH/R14			CLSD	CLSD	N	H-27647	
000207	A06	08A	2E11-F047A	RHR HX INLET	H-26015	REACTOR	087	RL/R14	R		OPEN	OPEN	N	H-27647	
000165	A06	08A	2E11-F048A	RHR HX BYPASS	H-26015	REACTOR	087	RL/R14	RS		OPEN	OP/CL	Y	H-27647	
000205	A06	07	2E11-F065A	RHR PUMP 2A SUCTION	H-26015	REACTOR	087	RH/R14			OPEN	OPEN	N	H-27653	
000210	A06	08A	2E11-F068A	RHRSW HX FLOW CONTROL	H-21039	REACTOR	087	RL/R14	RS		CLSD	OP/CL	Y	H-27648	
000211	A06	08A	2E11-F073A	RHRSW TO RHR CROSSTIE	H-26015	REACTOR	087	RH/R14	R		CLSD	CLSD	N	H-27647	
000164	A06	08B	2E11-F079A	RHR HX SAMPLE SOLENOID VALVE	H-26015	REACTOR	087	RJ/R23			CLSD	CLSD	Y	H-27464	
000166	A06	08A	2E11-F091A	HPCI STEAM TO RHR	H-26015	REACTOR	087	RH/R15			CLSD	CLSD	N	H-27651	
000163	A06	08A	2E11-F104A	RHR HX VENT	H-26015	REACTOR	087	RL/R14			CLSD	CLSD	N	H-27648	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	<---------- Building (7)	EQUIPMENT Flr.Elv. (8)	LOCATION ------> Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	<-- OP. Normal (12)	ST. --> Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)
000209	A06	08A	2E11-F119A	RHRSW SYSTEM CROSSTIE	H-21039	REACTOR	087	RL/R21	R		CLSD	CLSD	N	H-27537	
000174	A06	18	2E11-K600A	RHR DISCH HEADER FT SQRT	H-26015	CONTROL	164	2H11-P613			N/A	N/A	Y	H-27644	
000178	A06	18	2E11-K603A	POWER SUPPLY	H-26015	CONTROL	164	2H11-P613			N/A	N/A	Y	H-27644	
000212	A06	20	2E11-K613A	CONTROL AMPLIFIER	H-26015	CONTROL	164	2H11-P613	R		N/A	N/A	Y	H-27652	
000213	A06	18	2E11-N002A	RHR HX TUBE TO SHELL D[illegible]	H-26015	REACTOR	087	2H21-P418A	R		N/A	N/A	Y	H-27652	
000218	A06	18	2E11-N007A	RHRSW HX INLET FT	H-21039	REACTOR	087	2H21-P418A	R		N/A	N/A	Y	H-27644	
000173	A06	18	2E11-N015A	RHR DISCH HEADER FT	H-26015	REACTOR	087	2H21-P418A			N/A	N/A	Y	H-27644	
000216	A06	18	2E11-N017A	RHRSW HX INLET PS	H-21039	REACTOR	087	RH/R14	RS		N/A	N/A	Y	H-27638	
000217	A06	18	2E11-N017C	RHRSW HX INLET PS	H-21039	REACTOR	087	RH/R14	RS		N/A	N/A	Y	H-27638	
000176	A06	18	2E11-N082A	RHR PUMP FLOW DPT	H-26015	REACTOR	087	2H21-P418A			N/A	N/A	Y	H-24423	
000800	A06	18	2E11-N094A	DRYWELL PRESSURE PT	H-26015	REACTOR	158	RE/R16			N/A	N/A	Y	H-24427	
000801	A06	18	2E11-N094C	DRYWELL PRESSURE PT	H-26015	REACTOR	158	RE/R16			N/A	N/A	Y	H-24427	
000177	A06	18	2E11-N682A	RHR PUMP FLOW DPIS	H-26015	CONTROL	164	2H11-P925			N/A	N/A	Y	H-24423	
000214	A06	18	2E11-R600A	RHR HX TUBE TO SHELL DPI	H-26015	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-27652	
000219	A06	18	2E11-R602A	RHRSW HX INLET FI	H-21039	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-27644	
000175	A06	18	2E11-R603A	RHR DISCH HEADER FI	H-26015	CONTROL	164	2H11-P601			N/A	N/A	Y	H-27644	
000220	A06	18	2E11-R628A	RHRSW CONTROLLER	H-26015	CONTROL	164	2H11-P613	R		N/A	N/A	Y	H-27635	
000215	A06	18	2E11-S600A	F068A POSITION MODULATOR	H-26015	CONTROL	164	2H11-P613	R		N/A	N/A	Y	H-27640	
000777	A06	0	2R34-S006A	SURGE PAK FOR 2E11-C001A	H-27656	INTAKE	111		S		N/A	N/A	Y		
000342	A07	21	2E11-B001A	RHR HEAT EXCHANGER A	H-26015	REACTOR	087	RL/R14			N/A	N/A	N/A		
000350	A07	06	2E11-C001A	RHRSW PUMP 2A	H-21039	INTAKE	111	+			OFF	ON	Y	H-27656	
000331	A07	06	2E11-C002A	RHR PUMP 2A	H-26015	REACTOR	087	RL/R14			OFF	ON	Y	H-27657	
000363	A07	08A	2E11-F003A	RHR HX OUTLET	H-26015	REACTOR	087	RL/R14			OPEN	OPEN	N	H-27647	
000329	A07	08A	2E11-F004A	TORUS SUCTION ISOLATION	H-26015	REACTOR	087	RL/R14			OPEN	OPEN	N	H-27647	
000330	A07	08A	2E11-F006A	SHUTDOWN COOLING SUCTION	H-26015	REACTOR	087	RL/R14			CLSD	CLSD	N	H-27647	
000332	A07	08A	2E11-F007A	RHR PUMP MIN FLOW BYPASS	H-26015	REACTOR	087	RL/R14			OPEN	OP/CL	Y	H-27648	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000338	A07	08A	2E11-F010	RHR HX HDR CROSSTIE VALVE	H-26015	REACTOR	087	RH/R19			CLSD	CLSD	N	H-27647	
000341	A07	08A	2E11-F015A	INBOARD INJECTION	H-26015	REACTOR	130	RJ/R18			CLSD	CLSD	N	H-27648	
000340	A07	08A	2E11-F016A	CONTAINMENT SPRAY OUTBOARD	H-26015	REACTOR	130	RJ/R21	S		CLSD	OPEN	Y	H-27647	
000367	A07	08A	2E11-F021A	CONTAINMENT SPRAY INBOARD	H-26015	REACTOR	130	RJ/R21	S		CLSD	OPEN	Y	H-27647	
000366	A07	08A	2E11-F024A	TEST LINE TO TORUS	H-26015	REACTOR	087	RF/R14			CLSD	CLSD	N	H-27647	
000333	A07	08A	2E11-F026A	RHR HX DRAIN TO RCIC	H-26015	REACTOR	087	RL/R14			CLSD	CLSD	N	H-27647	
000365	A07	08A	2E11-F027A	TORUS SPRAY INBOARD ISOLATION	H-26015	REACTOR	087	RF/R14	S		CLSD	OPEN	Y	H-27647	
000339	A07	08A	2E11-F028A	TORUS SPRAY OUTBOARD ISOLATION	H-26015	REACTOR	087	RH/R14	S		CLSD	OPEN	Y	H-27647	
000349	A07	08A	2E11-F047A	RHR HX INLET	H-26015	REACTOR	087	RL/R14			OPEN	OPEN	N	H-27647	
000336	A07	08A	2E11-F048A	RHR HX BYPASS	H-26015	REACTOR	087	RL/R14			OPEN	OP/CL	Y	H-27647	
000364	A07	07	2E11-F065A	RHR PUMP 2A SUCTION	H-26015	REACTOR	087	RH/R14			OPEN	OPEN	N	H-27653	
000352	A07	08A	2E11-F068A	RHRSW HX FLOW CONTROL	H-21039	REACTOR	087	RL/R14			CLSD	OP/CL	Y	H-27648	
000353	A07	08A	2E11-F073A	RHRSW TO RHR CROSSTIE	H-26015	REACTOR	087	RH/R14			CLSD	CLSD	N	H-27647	
000335	A07	08B	2E11-F079A	RHR HX SAMPLE SOLENOID VALVE	H-26015	REACTOR	087	RJ/R23			CLSD	CLSD	Y	H-27464	
000337	A07	08A	2E11-F091A	HPCI STEAM TO RHR	H-26015	REACTOR	087	RH/R15			CLSD	CLSD	N	H-27651	
000334	A07	08A	2E11-F104A	RHR HX VENT	H-26015	REACTOR	087	RL/R14			CLSD	CLSD	N	H-27648	
000351	A07	08A	2E11-F119A	RHRSW SYSTEM CROSSTIE	H-21039	REACTOR	087	RL/R21			CLSD	CLSD	N	H-27537	
000344	A07	18	2E11-K600A	RHR DISCH HEADER FT SQRT	H-26015	CONTROL	164	2H11-P613			N/A	N/A	Y	H-27644	
000348	A07	18	2E11-K603A	POWER SUPPLY	H-26015	CONTROL	164	2H11-P613			N/A	N/A	Y	H-27644	
000354	A07	20	2E11-K613A	CONTROL AMPLIFIER	H-26015	CONTROL	164	2H11-P613			N/A	N/A	Y	H-27652	
000355	A07	18	2E11-N002A	RHR HX TUBE TO SHELL DPT	H-26015	REACTOR	087	2H21-P418A			N/A	N/A	Y	H-27652	
000360	A07	18	2E11-N007A	RHRSW HX INLET FT	H-21039	REACTOR	087	2H21-P418A			N/A	N/A	Y	H-27644	
000343	A07	18	2E11-N015A	RHR DISCH HEADER FT	H-26015	REACTOR	087	2H21-P418A			N/A	N/A	Y	H-27644	
000358	A07	18	2E11-N017A	RHRSW HX INLET PS	H-21039	REACTOR	087	RH/R14			N/A	N/A	Y	H-27638	
000359	A07	18	2E11-N017C	RHRSW HX INLET PS	H-21039	REACTOR	087	RH/R14			N/A	N/A	Y	H-27638	
000346	A07	18	2E11-N082A	RHR PUMP FLOW DPT	H-26015	REACTOR	087	2H21-P418A			N/A	N/A	Y	H-24423	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000347	A07	18	2E11-N682A	RHR PUMP FLOW DPIS	H-26015	CONTROL	164	2H11-P925			N/A	N/A	Y	H-24423	
000356	A07	18	2E11-R600A	RHR HX TUBE TO SHELL DPI	H-26015	CONTROL	164	2H11-P601			N/A	N/A	Y	H-27652	
000361	A07	18	2E11-R602A	RHRSW HX INLET FI	H-21039	CONTROL	164	2H11-P601			N/A	N/A	Y	H-27644	
000345	A07	18	2E11-R603A	RHR DISCH HEADER FI	H-26015	CONTROL	164	2H11-P601			N/A	N/A	Y	H-27644	
000362	A07	18	2E11-R628A	RHRSW CONTROLLER	H-26015	CONTROL	164	2H11-P613			N/A	N/A	Y	H-27635	
000357	A07	18	2E11-S600A	F068A POSITION MODULATOR	H-26015	CONTROL	164	2H11-P613			N/A	N/A	Y	H-27640	
000779	A07	0	2R34-S006A	SURGE PAK FOR 2E11-C001A	H-27656	INTAKE	111				N/A	N/A	Y		
000244	A08	08A	1P41-F313D	UNIT 2 PSW ISOLATION	D-11001	YARD			R		CLSD	CLSD	Y	H-23689	
000255	A08	08A	2N71-F012	CIRC WATER MAKEUP	H-21026	YARD			RS		CLSD	OPEN	Y	H-23686	
000243	A08	06	2P41-C001A	PLANT SERVICE WATER PUMP A	H-21033	INTAKE	111		RS		ON/OFF	ON	Y	H-23698	
000250	A08	07	2P41-F037A	2E11C002A CONTROL VALVE	H-26050	REACTOR	096	RL/R14	RS		CLSD	OPEN	Y	H-27748	
000251	A08	07	2P41-F039A	2T41B003A CONTROL VALVE	H-26050	REACTOR	120	RL/R15	RS		CLSD	OPEN	Y	H-27756	
000249	A08	07	2P41-F066	2P64/2E51 COOLER ISOLATION	H-26050	REACTOR	120	RB/R19	R		OPEN	OPEN	N	H-27748	
000245	A08	08A	2P41-F312A	DIESEL GENERATOR 2A ISOL	H-21033	YARD			R		OPEN	OPEN	N	H-23695	
000247	A08	08A	2P41-F315A	REACTOR BUILDING ISOLATION	H-21033	YARD			R		OPEN	OPEN	N	H-23695	
000248	A08	08A	2P41-F316A	TURBINE BUILDING ISOLATION	H-21033	YARD			RS		OPEN	CLSD	Y	H-23695	
000246	A08	07	2P41-F339A	DIESEL GENERATOR 2A OUTLET	H-21033	DIESEL	130	A01	RS		CLSD	OPEN	Y	H-23697	
000254	A08	18	2P41-K601A	ELECTRICAL SUPPLY	H-21033	CONTROL	164	2H11-P656	R		N/A	N/A	Y	H-23633	
000252	A08	18	2P41-N303A	PSW DISCHARGE PT	H-21033	INTAKE	088		RS		N/A	N/A	Y	H-23633	
000253	A08	18	2P41-R601A	PSW DIV. I PI	H-21033	CONTROL	164	2H11-P650	R		N/A	N/A	Y	H-23633	
000776	A08	0	2R34-S005A	SURGE PAK FOR 2P41-C001A	H-23698	INTAKE	111		S		N/A	N/A	Y		
000262	A08	10	2T41-B003A	CS/RHR PUMP ROOM COOLER A	H-26071	REACTOR	087	RL/R14	RS		OFF	ON	Y	H-27758	
000263	A08	19	2T41-N021A	CS/RHR PUMP ROOM COOLER TE	H-26071	REACTOR	087	RL/R14	RS		N/A	N/A	Y	H-27234	
000264	A08	18	2T41-R611A	CS/RHR PUMP ROOM COOLER TIS	H-26071	CONTROL	164	2H11-P700	R		N/A	N/A	Y	H-27234	
000278	A09	0	2B21-A003A	SRV AIR ACCUMULATOR	H-28023	DRYWELL	148	AZ220	S		N/A	N/A	N		
000279	A09	0	2B21-A003B	SRV AIR ACCUMULATOR	H-28023	DRYWELL	148	AZ220	S		N/A	N/A	N		

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- EQUIPMENT Building	Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000280	A09	0	2B21-A003F	SRV AIR ACCUMULATOR	H-28023	DRYWELL	148	AZ274	S		N/A	N/A	N		
000281	A09	0	2B21-A003K	SRV AIR ACCUMULATOR	H-28023	DRYWELL	148	AZ251	S		N/A	N/A	N		
000768	A09	18	2C71-N050C	DRYWELL PRESSURE PT	H-28001	REACTOR	158	RE/R16	RS		N/A	N/A	Y	H-24401	
000769	A09	18	2C71-N050D	DRYWELL PRESSURE PT	H-28001	REACTOR	158	RE/R21	RS		N/A	N/A	Y	H-24401	
000285	A09	07	2P70-F001A	NITROGEN BACKUP SUPPLY	H-26066	REACTOR	158	RB/R17	RS		CLSD	OPEN	Y	H-27744	
000286	A09	08B	2P70-F004	DRYWELL PNEUMATIC HEADER	H-26066	REACTOR	167	RE/R16	R		OPEN	OPEN	N	H-27233	
000287	A09	08B	2P70-F005	DRYWELL PNEUMATIC HEADER	H-26066	REACTOR	167	RE/R16	R		OPEN	OPEN	N	H-27233	
000282	A09	21	2T48-A001	NITROGEN STORAGE TANK	H-26083	YARD			S		N/A	N/A	N	H-27779	
000284	A09	07	2T48-F104	NITROGEN SUPPLY	H-26083	REACTOR	087	RB/R19	R		CLSD	CLSD	N	H-27599	
000283	A09	07	2T48-F112A	NITROGEN FLOW CONTROL	H-26083	REACTOR	087	RE/R23	R		CLSD	CLSD	N	H-27773	
000288	A09	07	2T48-F321	DRYWELL NITROGEN SUPPLY	H-26083	REACTOR	087	RE/R19	R		CLSD	CLSD	N	H-27773	
000289	A09	07	2T48-F325	DRYWELL/TORUS NITROGEN SUPPLY	H-26083	REACTOR	087	RF/R23	R		CLSD	CLSD	N	H-27773	
000307	A10	18	2B21-N085A	RPV WATER LEVEL LT	H-26001	REACTOR	130	2H21-P409	R		N/A	N/A	Y	H-24423	
000304	A10	18	2B21-N090A	RPV PRESSURE PT	H-26001	REACTOR	158	2H21-P404A	R		N/A	N/A	Y	H-24427	
000301	A10	18	2B21-N091A	RPV WATER LEVEL LT	H-26001	REACTOR	158	2H21-P404A	R		N/A	N/A	Y	H-24423	
000306	A10	20	2B21-N685A	RPV WATER LEVEL LIS	H-26001	CONTROL	164	2H11-P925	R		N/A	N/A	Y	H-24423	
000303	A10	20	2B21-N690A	RPV PRESSURE PIS	H-26001	CONTROL	164	2H11-P927	R		N/A	N/A	Y	H-24427	
000300	A10	20	2B21-N691A	RPV WATER LEVEL LIS	H-26001	CONTROL	164	2H11-P925	R		N/A	N/A	Y	H-24423	
000299	A10	20	2B21-R604A	RPV WATER LEVEL	H-26001	CONTROL	164	2H11-P603	R		N/A	N/A	Y	H-24435	
000305	A10	20	2B21-R615	RPV WATER LEVEL	H-26001	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-24435	
000302	A10	20	2B21-R623A	RPV PRESSURE	H-26001	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-24435	
000318	A11	18	2T47-R626	TORUS TEMPERATURE RECORDER	H-26074	CONTROL	164	2H11-P657	R		N/A	N/A	Y	H-24548	
000327	A11	07	2T48-F361A	TORUS LEVEL INST. ISOLATION	H-26084	REACTOR	087	RF/R14	R		OPEN	OPEN	N	H-27776	
000328	A11	07	2T48-F362A	TORUS LEVEL INST. ISOLATION	H-26084	REACTOR	087	RB/R16	R		OPEN	OPEN	N	H-27776	
000325	A11	18	2T48-K604A	INSTRUMENT DISTRIBUTION	H-26284	CONTROL	164	2H11-P691	R		N/A	N/A	Y		
000326	A11	18	2T48-K621A	TORUS WATER LEVEL INST. I/V	H-26285	CONTROL	164	2H11-P691	R		N/A	N/A	Y	H-27778	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000321	A11	18	2T48-K622	TORUS TEMPERATURE INST. R/V	H-26285	CONTROL	164	2H11-P605A	R		N/A	N/A	Y	h-27546	
000319	A11	18	2T48-N009A	TORUS WATER TE	H-26084	REACTOR	087	TORUS	RS		N/A	N/A	Y	H-27778	
000320	A11	18	2T48-N009C	TORUS WATER TE	H-26084	REACTOR	087	TORUS	RS		N/A	N/A	Y	H-27778	
000323	A11	18	2T48-N010A	TORUS WATER LT	H-26084	REACTOR	087	RE/R14	R		N/A	N/A	Y	H-27778	
000324	A11	18	2T48-R622A	TORUS WATER LEVEL INDICATOR	H-26084	CONTROL	164	2H11-P657	R		N/A	N/A	Y	H-27778	
000322	A11	18	2X75-K651A	POWER DISTRIBUTION	H-26285	CONTROL	164	P605A	R		ON	ON	Y		
000770	A12	18	2E11-N094A	DRYWELL PRESSURE PT	H-26015	REACTOR	158	RE/R16	RS		N/A	N/A	Y	H-24427	
000772	A12	18	2E11-N094C	DRYWELL PRESSURE PT	H-26015	REACTOR	158	RE/R16	RS		N/A	N/A	Y	H-24427	
000412	A12	0	2R43-A005A	DG 2A STARTING AIR RECEIVER	H-21074	DIESEL	130	B01	S		N/A	N/A	N		
000413	A12	0	2R43-A006A	DG 2A STARTING AIR RECEIVER	H-21074	DIESEL	130	B01	S		N/A	N/A	N		
000411	A12	18	2R43-N001A	DG 2A DAY TANK LS	H-21074	DIESEL	130	B01	RS		N/A	N/A	Y	H-23801	
000724	A12	20	2R43-P001A	DIESEL GEN 2A CONT PANEL	H-23022	DIESEL	130	B02	S		N/A	N/A	N		
000407	A12	17	2R43-S001A	DIESEL GENERATOR 2A	H-21074	DIESEL	130		RS		OFF	ON	Y	H-23807	
000409	A12	21	2Y52-A001A	DG 2A FUEL OIL STORAGE TANK	H-21074	YARD			S		N/A	N/A	N		
000408	A12	21	2Y52-A101A	DG 2A FUEL OIL DAY TANK	H-21074	DIESEL	130		S		N/A	N/A	N		
000410	A12	05	2Y52-C001A	DG 2A FUEL OIL PUMP 2A1	H-21074	YARD			RS		OFF	ON	Y	H-23807	
000460	A13	20	2H21-P198	AMMETER SHUNT BATTERY CHG	H-23025	DIESEL	130	A02	S		ON	ON	N	H-23371	
000687	A13	20	2H21-P248	250 VOLT DC SWGR 2A CNT PNL	H-23240	CONTROL	130	TE/T13	S		N/A	N/A	N		
000466	A13	20	2H21-P285	SHUNT BOX A	H-23235	CONTROL	112	TE/T13	S		ON	ON	N	H-23390	
000467	A13	20	2H21-P286	SHUNT BOX B	H-23235	CONTROL	112	TF/T14	S		ON	ON	N	H-23390	
000468	A13	20	2H21-P287	SHUNT BOX C	H-23235	CONTROL	112	TF/T13	S		ON	ON	N	H-23390	
000461	A13	20	2H21-P291	BATTERY 2A FUSE BOX	H-23022	DIESEL	130	B02	S		ON	ON	N	H-23371	
000455	A13	04	2R11-S004	600-120/208 V AC XFMR	H-23025	DIESEL	130	B03	S		N/A	N/A	Y	H-23315,H-23023	
000457	A13	04	2R11-S011	600/208 V XFMR	H-23027	YARD					N/A	N/A	N	H-23380	
000465	A13	04	2R11-S031	600/208 V XFMR	H-27279	REACTOR	130	2R24-S011			N/A	N/A	Y	H-27012	
000454	A13	04	2R11-S041	CONT BLDG ESS XFMR 2B	H-23240	CONTROL	130	TD/T13	S		N/A	N/A	Y	H-23369	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000463	A13	18	2R20M-P001	FUSE BOX	H-23023	DIESEL	130	A02	S		N/A	N/A	N	H-23315	
000456	A13	18	2R20N-P001	FUSE BOX	H-23240	CONTROL	130		S		N/A	N/A	N	H-23369	
000431	A13	03	2R22-S005	4160V STA SVC SWGR 2E	H-23023	DIESEL	130	A02	SR		N/A	N/A	Y	H-23357	
000432	A13	02	2R22-S016	250 V DC BATTERY SWGR 2A	H-23239	CONTROL	130	TEA/T13	SR		N/A	N/A	Y	H-23390	
000433	A13	02	2R23-S003	600 V STA SVC SWGR 2C & XFMR	H-23240	CONTROL	130	TEA/T14	SR		N/A	N/A	Y	H-23362	
000434	A13	01	2R24-S009	600/208 V MCC 2A INTAKE STRU	H-23027	INTAKE	111		S		N/A	N/A	N	H-23380	
000435	A13	01	2R24-S011	600 V MCC 2C ESS DIV 1	H-27279	REACTOR	130	RF/R14	RS		N/A	N/A	Y	H-27011,H-27012	
000436	A13	01	2R24-S011A	600 V MCC ESS DIV 1	H-27296	REACTOR	158	RA/R17	RS		N/A	N/A	Y	H-27009	
000437	A13	01	2R24-S018A	600 V MCC 2E-A ESS DIV 1	H-27279	REACTOR	130	RH/R17	RS		N/A	N/A	Y	H-27021	
000438	A13	01	2R24-S021	125/250 V DC MCC 2A ESS DIV 1	H-27284	REACTOR	130	RB/R14	RS		N/A	N/A	Y	H-27023	
000439	A13	01	2R24-S025	600/208 V MCC 2A ESS DIV 1	H-23023	DIESEL	130	B02	RS		N/A	N/A	Y	H-23315	
000440	A13	14	2R25-S001	125 V DC CABINET 2A	H-23239	CONTROL	130	TDA/T14	S		N/A	N/A	Y	H-23390	
000441	A13	14	2R25-S004	125 V DC CABINET 2D	H-23025	DIESEL	130	B02	S		N/A	N/A	Y	H-23371	
000442	A13	14	2R25-S029	120/208 VAC CABINET 2J	H-23025	DIESEL	130	B02	S		N/A	N/A	Y	H-23315	
000443	A13	14	2R25-S036	120/208 V AC ESS CABINET 2A	H-23240	CONTROL	130	TDA/T13	S		N/A	N/A	Y	H-23369	
000445	A13	14	2R25-S064	120/208 V AC CABINET 2A INST	H-23240	CONTROL	130	TG/T13	S		N/A	N/A	Y	H-23369	
000446	A13	14	2R25-S101	120/208 V ESS MISC POWER CAB	H-27279	REACTOR	130	2R24-S011			N/A	N/A	Y	H-27012	
000447	A13	14	2R25-S129	125 V DC DISTR CABINET 2E	H-23239	CONTROL	130	TEA/T13	S		N/A	N/A	Y	H-24436	
000462	A13	18	2R26-M002	2R25-S064 DISCONNECT SWITCH	H-23240	CONTROL	130	TE/T13	S		N/A	N/A	N	H-23369	
000464	A13	18	2R26-M031A	125 V DC THROWOVER SWITCH 2A	H-23240	CONTROL	130	TE/T13	S		N/A	N/A	N	H-23390	
000458	A13	18	2R26-M031B	125 V DC THROWOVER SWITCH 2B	H-23240	CONTROL	130	TF/T13	S		N/A	N/A	N	H-23390	
000459	A13	18	2R26-M032A	125 V DC THROWOVER SWITCH	H-23025	DIESEL	130	C02	S		N/A	N/A	N	H-23371	
000448	A13	15	2R42-S001A	125/250 V STATION BATTERY 2A	H-23220	CONTROL	112	TEA/T14	S		N/A	N/A	Y	H-23390	
000449	A13	15	2R42-S002A	125 V DIESEL SYSTEM BATTERY 2A	H-23022	DIESEL	130	B02	S		N/A	N/A	Y	H-23371	
000450	A13	16	2R42-S026	125 V BATTERY CHARGER 2A	H-23240	CONTROL	130	TF/T13	S		N/A	N/A	Y	H-23390	
000451	A13	16	2R42-S027	125 V BATTERY CHARGER 2B	H-23240	CONTROL	130	TF/T13	S		N/A	N/A	Y	H-23390	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	Building (7)	EQUIPMENT Flr.Elv. (8)	LOCATION Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	OP. ST. Normal (12)	OP. ST. Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)
000452	A13	16	2R42-S032A	125 V BATTERY CHARGER 2G	H-23025	DIESEL	130	B02	S		N/A	N/A	Y	H-23371	
000453	A13	16	2R44-S002	STATIC INVERTER	H-13131	CONTROL	147	TI/T14	S		ON	ON	Y	H-23390,H-27021	
000756	A14	0	2L48-D134	D/G RM 2A FIRE DAMPER	H-23395	DIESEL	130	2A	RS		OPEN	OPEN	N		
000414	A14	09	2X41-C010A	DG 2A ROOM EXHAUST FAN	H-12619	DIESEL	130	A02	RS		OFF	ON	Y	H-23395	
000416	A14	0	2X41-C013A	DG 2A ROOM LOUVER	H-12619	DIESEL	130		RS		CLSD	OPEN	N	H-23395	
000534	A14	09	2X41-C014A	SWGR RM 2E FAN	H-12619	DIESEL	ROOF		RS		OFF	ON	Y	H-23397	
000535	A14	0	2X41-C015A	SWGR RM 2E LOUVER	H-12619	DIESEL	130	SWGR RM 2E	RS		CLSD	OPEN	Y	H-23397	
000417	A14	09	2X41-C016A	DG 2A BATTERY ROOM FAN	H-12619	DIESEL	130	B02	RS		OFF	ON	Y	H-23397	
000758	A14	0	2X41-C024A	D/G BATT RM 2A FIRE DAMPER	H-23395	DIESEL	130	2A	RS		OPEN	OPEN	N		
000760	A14	0	2X41-C024C	D/G BATT RM 2A FIRE DAMPER	H-23395	DIESEL	130	2A	RS		OPEN	OPEN	N		
000418	A14	0	2X41-C028A	DG 2A BATTERY ROOM LOUVER	H-12619	DIESEL	130		RS		CLSD	OPEN	N	H-23397	
000762	A14	0	2X41-C030A	D/G RM 2A FIRE DAMPER	H-23395	DIESEL	130	2A	RS		OPEN	OPEN	N		
000764	A14	0	2X41-C030C	D/G RM 2A FIRE DAMPER	H-23395	DIESEL	130	2A	RS		OPEN	OPEN	N		
000415	A14	18	2X41-N011A	DG 2A ROOM FAN THERMOSTAT	H-12619	DIESEL	130	A02	RS		N/A	N/A	Y	H-23395	
000536	A14	18	2X41-N013A	SWGR RM 2E LVR THERMOSTAT	H-12619	DIESEL	130	SWGR RM 2E	RS		N/A	N/A	Y	H-23397	
000537	A14	18	2X41-N044	FLOW SWITCH FOR FAN 2X41-C010B	H-12619	DIESEL	ROOF		RS		N/A	N/A	Y	H-23395	
000538	A14	18	2X41-N046	FLOW SWITCH FOR FAN 2X41-C010A	H-12619	DIESEL	ROOF		RS		N/A	N/A	Y	H-23395	
000731	A14	20	2X43-P003A	CO2 ZONE 1 CONTROL CABINET	H-23747	DIESEL	130	A01	S		N/A	N/A	N		
000527	A14	09	2Z41-C014	STN BATT RM EMERG EXHAUST FAN	H-26093	TURBINE	130	TE/T16	RS		OFF	ON	Y	H-27800	
000543	A15	08A	2G31-F001	RWCU INBOARD ISOL GATE VALVE	H-26036	DRYWELL	165	AZ020	S		OPEN	CLSD	Y	H-27462, H27991	
000545	A15	08B	2T48-F310	TORUS VAC BRKR ISOL BUTTERFLY VLV	H-26084	REACTOR	087	RF/R23			CLSD	CLSD	Y	H-27777	
000539	P/A15	08B	2G11-F003	DRYWELL FL DR PMP ISOL VALVE	H-26026	REACTOR	087	RE/R19	S		OPEN	CLSD	N	H-27457	
000540	P/A15	08B	2G11-F004	DRYWELL FL DR PMP ISOL VALVE	H-26026	REACTOR	087	RE/R19	S		OPEN	CLSD	N	H-27457	
000541	P/A15	08B	2G11-F019	DRYWELL EQ DR PMP ISOL VALVE	H-26026	REACTOR	087	RF/R21	S		OPEN	CLSD	N	H-27457	
000542	P/A15	08B	2G11-F020	DRYWELL EQ DR PMP ISOL VALVE	H-26026	REACTOR	087	RF/R21	S		OPEN	CLSD	N	H-27457	
000725	P01	20	2C82-P001	REMOTE SHUTDOWN PANEL	H-27284	REACTOR	130	RA/R16	S		N/A	N/A	N		

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	<--- EQUIPMENT Building (7)	Flr.Elv. (8)	LOCATION ---> Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	<-- OP. ST. --> Normal (12)	Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)
000632	P01	20	2H11-P601	REAC CNTMT COOL ISO BN BD	H-13138	CONTROL	164	TE/T12	S		N/A	N/A	N		
000634	P01	20	2H11-P603	REAC CONTROL BN BD	H-23258	CONTROL	164	TC/T12	S		N/A	N/A	N		
000635	P01	20	2H11-P604	PROCESS RADMON VER BD	H-23259	CONTROL	164	TE/T13	S		N/A	N/A	N		
000785	P01	20	2H11-P606	STARTUP NEUT MON PNL	H-23259	CONTROL	164	TC/T13			N/A	N/A	N		
000786	P01	20	2H11-P608	RWR RNGE NEUT MON PNL	H-23259	CONTROL	164	TC/T13			N/A	N/A	N		
000787	P01	20	2H11-P609	CH A PRI ISOL & RPS VB	H-23259	CONTROL	164	TE/T13			N/A	N/A	N		
000641	P01	20	2H11-P611	CH B PRI ISOL & RPS VB	H-23247	CONTROL	164	TE/T13	S		N/A	N/A	N		
000642	P01	20	2H11-P612	FW AND RECIRC INST PNL	H-23247	CONTROL	164	TC/T13	S		N/A	N/A	N		
000644	P01	20	2H11-P614	NSSS TEMP DET VERT BD	H-23256	CONTROL	164	TE/T13	S		N/A	N/A	N		
000646	P01	20	2H11-P618	CHAN B RHR RELAY VERT BD		CONTROL	164	TC/T12	S		N/A	N/A	N		
000647	P01	20	2H11-P620	HPCI RELAY VERT BD	H-13138	CONTROL	164	TC/T12	S		N/A	N/A	N		
000649	P01	20	2H11-P623	OUTBD ISO VLV VERT PNL	H-23257	CONTROL	164	TC/T12	S		N/A	N/A	N		
000651	P01	20	2H11-P627	CORE SPRAY CTRL PNL DIV 2	H-13138	CONTROL	164	TC/T12	S		N/A	N/A	N		
000652	P01	20	2H11-P628	ADS RELAY PANEL	H-23258	CONTROL	164	TC/T12	S		N/A	N/A	N		
000653	P01	20	2H11-P650	TURB FDWTR 7 COND PNL	H-23258	CONTROL	164	TC/T12	S		N/A	N/A	N		
000654	P01	20	2H11-P652	DSL GEN & EMER STA PNL		CONTROL	164	TC/T11	S		N/A	N/A	N		
000655	P01	20	2H11-P654	GAS TREAT VENT VERT BD	H-23257	CONTROL	164	TC/T12	S		N/A	N/A	N		
000656	P01	20	2H11-P656	TURB AUX SYSTEM VERT PNL	H-23257	CONTROL	164	TC/T11	S		N/A	N/A	N		
000658	P01	20	2H11-P664	MSIV LEAK CTRL SYS PNL		CONTROL	164	TA/T13	S		N/A	N/A	N		
000659	P01	20	2H11-P674	START UP XFMR 2C PANEL		CONTROL	164	TA/T12	S		N/A	N/A	N		
000661	P01	20	2H11-P679	STA SERV XFMR RLY PANEL	H-23257	CONTROL	164	TA/T12	S		N/A	N/A	N		
000662	P01	20	2H11-P691	ANALOG SIG CONV PANEL		CONTROL	164	TB/T11	S		N/A	N/A	N		
000663	P01	20	2H11-P700	ANALOG VNT LEAK DETECT PNL	H-23246	CONTROL	164	TE/T13	S		N/A	N/A	N		
000665	P01	20	2H11-P922	RPS TRIP UNIT CABINET	H-16249	CONTROL	164	TI/T12	S		N/A	N/A	N		
000667	P01	20	2H11-P924	RPS TRIP UNIT CABINET	H-16249	CONTROL	164	TI/T12	S		N/A	N/A	N		
000669	P01	20	2H11-P926	ECCS TRIP UNIT CABINET	H-16249	CONTROL	164	TI/T12	S		N/A	N/A	N		

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- EQUIPMENT Building	Flr.Elv.	LOCATION ------> Rm. or Row/Col.	SORT	NOTES	<-- OP. ST. --> Normal	Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000671	P01	20	2H11-P928	ECCS TRIP UNIT CABINET	H-16249	CONTROL	164	TI/T12	S		N/A	N/A	N		
000672	P01	18	2H21-P002	REACTOR WATER CLEANUP PNL	H-26100	REACTOR	158	RF/R24	S		N/A	N/A	N		
000673	P01	18	2H21-P016	MAIN STEAM FLOW INST RACK	H-27279	REACTOR	087	RL/R24	S		N/A	N/A	N		
000675	P01	18	2H21-P036	HPCI SYS LOCAL RACK	H-27281	REACTOR	130	RJ/R23	S		N/A	N/A	N		
000676	P01	20	2H21-P052	HPCI TEST VLV CONTROL PNL	H-27271	REACTOR	087	RH/R24	S		N/A	N/A	N		
000677	P01	20	2H21-P173	SHUTDOWN INSTRUMENT PANEL	H-27284	REACTOR	130	RA/R16	S		N/A	N/A	N		
000679	P01	20	2H21-P202	DIESEL GEN 2C RELAY PANEL	H-23071	DIESEL	130	C02	S		N/A	N/A	N		
000681	P01	18	2H21-P225	TURBINE BUILDING INST RACK	H-23067	TURBINE	130	TH/T20	S		N/A	N/A	N		
000684	P01	20	2H21-P232	RELAY PANEL 2C-D/G 2C	H-23023	DIESEL	130	B02	S		N/A	N/A	N		
000686	P01	20	2H21-P246	600 VOLT BUS 2D CNT PNL	H-23240	CONTROL	130	TC/T13	S		N/A	N/A	N		
000690	P01	20	2H21-P256	DG FUEL PMP & MOV CONT PNL	H-23071	DIESEL	130	C02	S		N/A	N/A	N		
000692	P01	20	2H21-P259	D/G 2C HT/VEN CONT PNL	H-23071	DIESEL	130	B01	S		N/A	N/A	N		
000694	P01	20	2H21-P262	SWGR 2G RM HT/VEN CONT PNL	H-23023	DIESEL	130	B02	S		N/A	N/A	N		
000696	P01	20	2H21-P267	MOV CONTROL PNL 2B DIV 2	H-23027	INTAKE	111		S		N/A	N/A	N		
000698	P01	20	2H21-P305	DG 2C LOADING TIMER PANEL	H-23071	DIESEL	130	C03	S		N/A	N/A	N		
000699	P01	18	2H21-P401	CS INSTRUMENT RACK	H-26096	REACTOR	087	RL/R14	S		N/A	N/A	N		
000703	P01	18	2H21-P404D	RPV LVL/PRESS INSTR RACK	H-26100	REACTOR	158	RG/R17	S		N/A	N/A	N		
000705	P01	18	2H21-P405A	RPV LVL/PRESS INSTR RACK	H-26100	REACTOR	158	RH/R23	S		N/A	N/A	N		
000706	P01	18	2H21-P405B	RPV LVL/PRESS INSTR RACK	H-26100	REACTOR	158	RH/R23	S		N/A	N/A	N		
000708	P01	18	2H21-P405D	RPV LVL/PRESS INSTR RACK	H-26100	REACTOR	158	RH/R23	S		N/A	N/A	N		
000709	P01	18	2H21-P405E	RPV LVL/PRESS INSTR RACK	H-26100	REACTOR	158	RH/R23	S		N/A	N/A	N		
000711	P01	18	2H21-P410	JET PUMP INSTR RACK	H-26098	REACTOR	130	RF/R22	S		N/A	N/A	N		
000712	P01	18	2H21-P414A	HPCI INSTR RACK	H-26096	REACTOR	087	RG/R24	S		N/A	N/A	N		
000713	P01	18	2H21-P414B	HPCI INSTR RACK	H-26096	REACTOR	087	RG/R24	S		N/A	N/A	N		
000715	P01	18	2H21-P415B	MAIN STM FLOW INSTRUMENT RACK	H-26098	REACTOR	130	RF/R15	S		N/A	N/A	N		
000718	P01	18	2H21-P419	CS INSTRUMENT RACK	H-26096	REACTOR	087	RL/R24	S		N/A	N/A	N		

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000719	P01	18	2H21-P421A	RHR INSTRUMENT RACK	H-26096	REACTOR	087	RL/R24	S		N/A	N/A	N		
000722	P01	18	2H21-P425B	RHR INSTRUMENT RACK	H-26098	REACTOR	130	RF/R22	S		N/A	N/A	N		
000723	P01	18	2H21-P434	HPCI INSTRUMENT RACK	H-26098	REACTOR	087	RG/R24	S		N/A	N/A	N		
000774	P01	01	2R24-S012A	600 V AC MCC		REACTOR	164	RB/R19	RS		N/A	N/A	Y		
000727	P01	20	2U61-P001	LEAK DETECTION CONTROL PANEL	H-23240	CONTROL	130	TD/T13	S		N/A	N/A	N		
000729	P01	20	2U61-P003	LEAK DETECTION CONTROL PANEL	H-23240	CONTROL	130	TD/T13	S		N/A	N/A	N		
000740	P02	088	2C11-D001-117	PILOT SCRAM SOLENOID	H-26006	REACTOR	130	HCU	S		ENRG	D-ENRG	N		
000741	P02	088	2C11-D001-118	PILOT SCRAM SOLENOID	H-26006	REACTOR	130	HCU	S		ENRG	D-ENRG	N		
000742	P02	088	2C11-D001-120	ROD POSITION CONTROL VALVE	H-26006	REACTOR	130	HCU			CLSD	CLSD	N		
000743	P02	088	2C11-D001-121	ROD POSITION CONTROL VALVE	H-26006	REACTOR	130	HCU			CLSD	CLSD	N		
000744	P02	088	2C11-D001-122	ROD POSITION CONTROL VALVE	H-26006	REACTOR	130	HCU			CLSD	CLSD	N		
000745	P02	088	2C11-D001-123	ROD POSITION CONTROL VALVE	H-26006	REACTOR	130	HCU			CLSD	CLSD	N		
000746	P02	0	2C11-D001-125	SCRAM ACCUMULATOR	H-26006	REACTOR	130	HCU	S		N/A	N/A	N		
000747	P02	07	2C11-D001-126	SCRAM INLET VALVE	H-26006	REACTOR	130	HCU	S		CLSD	OPEN	N		
000748	P02	07	2C11-D001-127	SCRAM INLET VALVE	H-26006	REACTOR	130	HCU	S		CLSD	OPEN	N		
000226	P02	088	2C11-F009	PILOT AIR HEADER SUPPLY	H-26007	REACTOR	130	RA/R21	RS		ENRG	D-ENRG	N	H-27614	
000141	P02	07	2C11-F010A	SCRAM DISCH VOL ISOLATION	H-26007	REACTOR	158		RS		OPEN	CLSD	N	H-27605	
000142	P02	07	2C11-F010B	SCRAM DISCH VOL ISOLATION	H-26007	REACTOR	158		RS		OPEN	CLSD	N	H-27605	
000143	P02	07	2C11-F011	SCRAM DISCH VOL DRAIN	H-26007	REACTOR	087	RB/R17	RS		OPEN	CLSD	N	H-27605	
000153	P02	0	2C71-S3A	MANUAL SCRAM SWITCH		CONTROL	164	2H11-P603	R		N/A	N/A	N	H-27607	
000154	P02	0	2C71-S3B	MANUAL SCRAM SWITCH		CONTROL	164	2H11-P603	R		N/A	N/A	N	H-27607	
000155	P02	0	2C71-S3C	MANUAL SCRAM SWITCH		CONTROL	164	2H11-P603	R		N/A	N/A	N	H-27607	
000156	P02	0	2C71-S3D	MANUAL SCRAM SWITCH		CONTROL	164	2H11-P603	R		N/A	N/A	N	H-27607	
000005	P03	07	2B21-F004	RPV HEAD VENT	H-26000	DRYWELL	148	AZ060	R		CLSD	CLSD	N	H-27458	
000006	P03	07	2B21-F013D	RPV SAFETY/RELIEF VALVE	H-26000	DRYWELL	148	AZ135	RS		CLSD	OP/CL	Y	H-27473	
000007	P03	07	2B21-F013G	RPV SAFETY/RELIEF VALVE	H-26000	DRYWELL	148	AZ095	RS		CLSD	OP/CL	Y	H-27472	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	<---------- EQUIPMENT Building (7)	Flr.Elv. (8)	LOCATION -----> Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	<-- OP. Normal (12)	ST. --> Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)
000137	P03	07	2B21-F013H	RPV SAFETY/RELIEF VALVE	H-26000	DRYWELL	148	AZ135	RS		CLSD	OP/CL	Y	H-27473	
000138	P03	07	2B21-F013M	RPV SAFETY/RELIEF VALVE	H-26000	DRYWELL	148	AZ090	RS		CLSD	OP/CL	Y	H-27470	
000010	P03	08A	2B21-F019	STEAM DRAIN ISOLATION	H-26000	REACTOR	130	RB/R19	R		CLSD	CLSD	N	H-27458	
000001	P03	07	2B21-F028A	OUTBOARD MSIV	H-26000	REACTOR	130	RB/R19	RS		OPEN	CLSD	Y	H-27450	
000002	P03	07	2B21-F028B	OUTBOARD MSIV	H-26000	REACTOR	130	RB/R18	RS		OPEN	CLSD	Y	H-27450	
000003	P03	07	2B21-F028C	OUTBOARD MSIV	H-26000	REACTOR	130	RB/R20	RS		OPEN	CLSD	Y	H-27990	
000004	P03	07	2B21-F028D	OUTBOARD MSIV	H-26000	REACTOR	130	RB/R19	RS		OPEN	CLSD	Y	H-27990	
000802	P03	18	2E11-N094B	DRYWELL PRESSURE PT	H-26014	REACTOR	158	RH/R21			N/A	N/A	Y	H-24430	
000803	P03	18	2E11-N094D	DRYWELL PRESSURE PT	H-26014	REACTOR	158	RH/R21			N/A	N/A	Y	H-24430	
000011	P03	08A	2E32-F001B	MSIV LEAKAGE CONTROL	H-26022	REACTOR	130	RA/R19	R		CLSD	CLSD	N	H-27413	
000012	P03	08A	2E32-F001F	MSIV LEAKAGE CONTROL	H-26022	REACTOR	130	RA/R19	R		CLSD	CLSD	N	H-27413	
000013	P03	08A	2E32-F001K	MSIV LEAKAGE CONTROL	H-26022	REACTOR	130	RA/R19	R		CLSD	CLSD	N	H-27413	
000014	P03	08A	2E32-F001P	MSIV LEAKAGE CONTROL	H-26022	REACTOR	130	RA/R19	R		CLSD	CLSD	N	H-27413	
000081	P04	08A	2E11-F091B	HPCI DISCH TO HX B VALVE	H-26014	REACTOR	087	RH/R23	R		CLSD	CLSD	N	H-27635	
000082	P04	08A	2E11-F140A	HPCI STEAM TO RHR HX A	H-26015	REACTOR	087	RH/R16	R		CLSD	CLSD	N	H-27635	
000798	P04	18	2E11-N094B	DRYWELL PRESSURE PT	H-26014	REACTOR	158	RH/R21			N/A	N/A	Y	H-24430	
000799	P04	18	2E11-N094D	DRYWELL PRESSURE PT	H-26014	REACTOR	158	RH/R21			N/A	N/A	Y	H-24430	
000015	P04	05	2E41-C001	HPCI PUMP	H-26021	REACTOR	087	RL/R25	S		OFF	ON	Y	H-27664	
000016	P04	05	2E41-C002	HPCI TURBINE	H-26021	REACTOR	087	RG/R24	RS		OFF	ON	Y	H-27664	
000017	P04	05	2E41-C002-3	HPCI LUBE OIL PUMP	H-51165	REACTOR	087	RG/R24	BOX		OFF	ON	Y	H-27670	
000018	P04	08A	2E41-F001	HPCI TURBINE STEAM SUPPLY VLV	H-26020	REACTOR	087	RG/R25	RS		CLSD	OPEN	Y	H-27671	
000019	P04	08A	2E41-F002	HPCI STEAM SUPPLY INBOARD ISOL	H-26020	DRYWELL	127	AZ020	R		OPEN	OPEN	N	H-27671	
000020	P04	08A	2E41-F003	HPCI STEAM SUPPLY OUTBOARD ISO	H-26020	REACTOR	130	RH/R19	R		OPEN	OPEN	N	H-27671	
000021	P04	08A	2E41-F004	HPCI PUMP SUCTION FROM CST	H-26020	REACTOR	087	RL/R25	RS		OPEN	CLSD	Y	H-27671	
000022	P04	08A	2E41-F006	HPCI PUMP INBOARD DISCH. VALVE	H-26020	REACTOR	087	RB/R19	RS		CLSD	OPEN	Y	H-27671	
000023	P04	08A	2E41-F007	HPCI PUMP OUTBOARD DISCH. VLV	H-26020	REACTOR	087	RG/R25	R		OPEN	OPEN	N	H-27671	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<--------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000024	P04	08A	2E41-F008	HPCI BYPASS TEST VALVE TO CST	H-26020	REACTOR	087	RG/R24	R		CLSD	CLSD	N	H-27671	
000025	P04	08A	2E41-F012	HPCI MINIMUM FLOW BYPASS VALVE	H-26020	REACTOR	087	RG/R24	RS		CLSD	OP/CL	Y	H-27671	
000035	P04	07	2E41-F026	BAR COND & LUBE OIL COOLER DRN	H-26021	REACTOR	087	RL/R25	R		CLSD	CLSD	Y	H-27672	
000026	P04	08A	2E41-F041	HPCI PUMP SUCTION - SUPP POOL	H-26020	REACTOR	087	RL/R25	RS		CLSD	OPEN	Y	H-27671	
000027	P04	08A	2E41-F042	HPCI PUMP SUCT FROM SUPP POOL	H-26020	REACTOR	087	RG/R24	RS		CLSD	OPEN	Y	H-27671	
000028	P04	07	2E41-F051	HPCI SUPP POOL SUCT ISOL VLV	H-26020	REACTOR	087	RF/R24	R		OPEN	OPEN	N	H-27672	
000029	P04	08A	2E41-F059	HPCI BAR COND COOLING WTR VLV	H-26021	REACTOR	087	RG/R24	RS		CLSD	OPEN	Y	H-27671	
000030	P04	08A	2E41-F104	HPCI VACUUM BREAKER LINE MOV	H-26020	REACTOR	087	RF/R23	R		OPEN	OPEN	N	H-27672	
000031	P04	08A	2E41-F111	HPCI VACUUM BREAKER LINE MOV	H-26020	REACTOR	087	RF/R23	R		OPEN	OPEN	N	H-27672	
000032	P04	08B	2E41-F124	HPCI REMOTE MAN. TRIP SOLENOID	H-26021	REACTOR	087	RG/R25	BOX		CLSD	OP/CL	Y	H-27668	
000033	P04	0	2E41-F3052	HPCI TURBINE CONTROL VALVE	H-26021	REACTOR	087	RG/R25	BOX		CLSD	OP/CL	Y	H-27672	
000034	P04	0	2E41-F3053	HPCI TURBINE STOP VALVE	H-26021	REACTOR	087	RG/R25	BOX		CLSD	OPEN	Y	H-27672	
000036	P04	18	2E41-K600	FLOW TRANSMITTER POWER SUPPLY	H-26020	CONTROL	164	2H11-P612	R		N/A	N/A	Y	H-27670	
000037	P04	18	2E41-K601	DISCH. FLOW SQ. ROOT CONVERTER	H-26020	CONTROL	164	2H11-P612	R		N/A	N/A	Y	H-27670	
000038	P04	18	2E41-K603	DC/AC INVERTER	H-26020	CONTROL	164	2H11-P612	R		N/A	N/A	Y	H-27666	
000039	P04	18	2E41-K615	HPCI DISCHARGE FLOW CONTROLLER	H-26020	CONTROL	164	2H11-P612	R		N/A	N/A	Y	H-27670	
000040	P04	18	2E41-K616	HPCI DISCHARGE CONTROL AMP	H-26020	CONTROL	164	2H11-P612	R		N/A	N/A	Y	H-27670	
000041	P04	18	2E41-N008	HPCI DISCH. FLOW TRANSMITTER	H-26020	REACTOR	087	2H21-P414A	R		N/A	N/A	Y	H-27670	
000042	P04	18	2E41-N050	HPCI DISCH. PRESS TRANSMITTER	H-26020	REACTOR	087	2H21-P414B	R		N/A	N/A	Y	H-24424	
000043	P04	18	2E41-N051	HPCI DISCHARGE DP TRANSMITTER	H-26020	REACTOR	087	2H21-P414A	R		N/A	N/A	Y	H-24424	
000044	P04	18	2E41-N053	HPCI PUMP SUCT PRESS TRANS	H-26021	REACTOR	087	2H21-P414B	R		N/A	N/A	N	H-24424	
000045	P04	18	2E41-N055A	HPCI TURB EXH RUPT. DSK PT	H-26021	REACTOR	087	2H21-P434	R		N/A	N/A	N	H-24422	
000046	P04	18	2E41-N055B	HPCI TURB EXH RUPT. DSK PT	H-26021	REACTOR	087	2H21-P414A	R		N/A	N/A	N	H-24425	
000047	P04	18	2E41-N055C	HPCI TURB EXH RUPT. DSK PT	H-26021	REACTOR	087	2H21-P434	R		N/A	N/A	N	H-24422	
000048	P04	18	2E41-N055D	HPCI TURB EXH RUPT. DSK PT	H-26021	REACTOR	087	2H21-P414A	R		N/A	N/A	N	H-24425	
000049	P04	18	2E41-N056B	HPCI TURBINE EXHAUST PT	H-26021	REACTOR	087	2H21-P414B	R		N/A	N/A	N	H-24424	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000050	P04	18	2E41-N056D	HPCI TURBINE EXHAUST PT	H-26021	REACTOR	087	2H21-P414B	R		N/A	N/A	N	H-24424	
000051	P04	18	2E41-N057A	HPCI STEAM LINE DP TRANSMITTER	H-26020	REACTOR	087	2H21-P016	R		N/A	N/A	N	H-24422	
000052	P04	18	2E41-N057B	HPCI STEAM LINE DP TRANSMITTER	H-26020	REACTOR	130	2H21-P036	R		N/A	N/A	N	H-24425	
000053	P04	18	2E41-N058A	HPCI STEAM LINE PT	H-26020	REACTOR	087	2H21-P016	R		N/A	N/A	N	H-24422	
000054	P04	18	2E41-N058B	HPCI STEAM LINE PT	H-26020	REACTOR	087	2H21-P036	R		N/A	NA/	N	H-24425	
000055	P04	18	2E41-N058C	HPCI STEAM LINE PT	H-26020	REACTOR	087	2H21-P016	R		N/A	N/A	N	H-24422	
000056	P04	18	2E41-N058D	HPCI STEAM LINE PT	H-26020	REACTOR	130	2H21-P036	R		N/A	N/A	N	H-24425	
000057	P04	18	2E41-N062B	SUPPRESSION POOL LEVEL TRANS.	H-26020	REACTOR	087	RG/R24	RS		N/A	N/A	N	H-24432	
000058	P04	18	2E41-N062D	SUPPRESSION POOL LEVEL TRANS.	H-26020	REACTOR	087	RG/R24	RS		N/A	N/A	N	H-24432	
000059	P04	18	2E41-N074	HPCI TURBINE STOP VLV POS. SW.	H-26021	REACTOR	087	2E41-C002	BOX		N/A	N/A	Y	H-27667	
000060	P04	18	2E41-N650	HPCI DISCH PRESS IND. SWITCH	H-26020	CONTROL	164	2H11-P926	R		N/A	N/A	Y	H-24424	
000061	P04	18	2E41-N651	HPCI DISCH DP IND SWITCH	H-26020	CONTROL	164	2H11-P926	R		N/A	N/A	Y	H-24424	
000062	P04	16	2E41-N653	HPCI PUMP SUCT PRESS IND SW.	H-26021	CONTROL	164	2H11-P926	R		N/A	N/A	N	H-24424	
000063	P04	18	2E41-N655A	TURB EXH RUPT DIAPHRAM PIS	H-26021	CONTROL	164	2H11-P925	R		N/A	N/A	N	H-24422	
000064	P04	18	2E41-N655B	TURB EXH RUPT DIAPHRAM PIS	H-26021	CONTROL	164	2H11-P926	R		N/A	N/A	N	H-24424	
000065	P04	18	2E41-N655C	TURB EXH RUPT DIAPHRAM PIS	H-26021	CONTROL	164	2H11-P925	R		N/A	N/A	N	H-24422	
000066	P04	18	2E41-N655D	TURB EXH RUPT DIAPHRAM PIS	H-26021	CONTROL	164	2H11-P926	R		N/A	N/A	N	H-24424	
000067	P04	18	2E41-N656B	HPCI TURB EXH PRESS IND SWITCH	H-26021	CONTROL	164	2H11-P926	R		N/A	N/A	N	H-24424	
000068	P04	18	2E41-N656D	HPCI TURB EXH PRESS IND SWITCH	H-26021	CONTROL	164	2H11-P926	R		N/A	N/A	N	H-24424	
000069	P04	18	2E41-N657A	HPCI STM LINE DP IND. SWITCH	H-26020	CONTROL	164	2H11-P925	R		N/A	N/A	N	H-24422	
000070	P04	18	2E41-N657B	HPCI STM LINE DP IND. SWITCH	H-26020	CONTROL	164	2H11-P926	R		N/A	N/A	N	H-24425	
000071	P04	18	2E41-N658A	HPCI STM LINE PRESS IND SWITCH	H-26020	CONTROL	164	2H11-P925	R		N/A	N/A	N	H-24422	
000072	P04	18	2E41-N658B	HPCI STM LINE PRESS IND SWITCH	H-26020	CONTROL	164	2H11-P926	R		N/A	N/A	N	H-24425	
000073	P04	18	2E41-N658C	HPCI STM LINE PRESS IND SWITCH	H-26020	CONTROL	164	2H11-P925	R		N/A	N/A	N	H-24422	
000074	P04	18	2E41-N658D	HPCI STM LINE PRESS IND SWITCH	H-26020	CONTROL	164	2H11-P926	R		N/A	N/A	N	H-24425	
000075	P04	18	2E41-N660A	HPCI STM LINE DP IND SWITCH	H-26020	CONTROL	164	2H11-P925	R		N/A	N/A	N	H-24422	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000076	P04	18	2E41-N660B	HPCI STM LINE DP IND SWITCH	H-26020	CONTROL	164	2H11-P926	R		N/A	N/A	N	H-24425	
000077	P04	18	2E41-N662B	SUPP POOL LEVEL IND SWITCH	H-26020	CONTROL	164	2H11-P928	R		N/A	N/A	N	H-24432	
000078	P04	18	2E41-N662D	SUPP POOL LEVEL IND SWITCH	H-26020	CONTROL	164	2H11-P928	R		N/A	N/A	N	H-24432	
000079	P04	18	2E41-R612	HPCI FLOW CONTROLLER	H-26020	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-27670	
000080	P04	18	2E41-R613	HPCI FLOW INDICATOR	H-26020	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-27670	
000083	P05	06	2E21-C001B	CORE SPRAY PUMP 2B	H-26018	REACTOR	087	RL/R24	RS		OFF	ON	Y	H-27663	
000084	P05	08A	2E21-F001B	CORE SPRAY TORUS SUCTION VLV	H-26018	REACTOR	087	RH/R24	R		OPEN	OPEN	N	H-27662	
000085	P05	08A	2E21-F004B	CORE SPRAY TO RVP OUTBOARD VLV	H-26018	REACTOR	158	RF/R21	R		OPEN	OPEN	N	H-27662	
000086	P05	08A	2E21-F005B	CORE SPRAY TO RVP ISOL VLV	H-26018	REACTOR	158	RF/R21	RS		CLSD	OPEN	Y	H-27662	
000087	P05	08A	2E21-F015B	CORE SPRAY ¡EST BYPASS VLV	H-26018	REACTOR	087	RL/R24	R		CLSD	CLSD	N	H-27662	
000088	P05	07	2E21-F019B	CORE SPRAY TORUS SUCTION VLV	H-26018	REACTOR	087	RH/R24	R		OPEN	OPEN	N	H-27661	
000089	P05	08A	2E21-F031B	CORE SPRAY MIN FLOW BYPASS	H-26018	REACTOR	087	RL/R24	RS		OPEN	OP/CL	Y	H-27662	
000090	P05	18	2E21-K600B	CORE SPRAY FLOW INST POWER SUP	H-26018	CONTROL	164	2H11-P612	R		ON	ON	Y	H-27661	
000091	P05	18	2E21-N003B	CORE SPRAY FLOW TRANSMITTER	H-26018	REACTOR	087	2H21-P419	R		ON	ON	Y	H-27661	
000092	P05	18	2E21-N051B	CORE SPRAY FLOW DPT	H-26018	REACTOR	087	2H21-P419	R		ON	ON	Y	H-24431	
000093	P05	18	2E21-N651B	CORE SPRAY FLOW DPIS	H-26018	CONTROL	164	2H11-P928	R		CLSD	OP/CL	Y	H-24428	
000094	P05	20	2E21-R601B	CORE SPRAY FLOW INDICATOR	H-26018	CONTROL	164	2H11-P601	R		ON	ON	Y	H-27661	
000095	P06	21	2E11-B001B	RHR HEAT EXCHANGER B	H-26014	REACTOR	087	RL/R24	S		N/A	N/A	N/A		
000096	P06	06	2E11-C001D	RHRSW PUMP 2D	H-21039	INTAKE	111		RS		OFF	ON	Y	H-27657	
000097	P06	06	2E11-C002B	RHR PUMP 2B	H-26014	REACTOR	087	RL/R24	RS		OFF	ON	Y	H-27657	
000098	P06	08A	2E11-F003B	RHR HX B DISCHARGE VALVE	H-26014	REACTOR	087	RL/R24	R		OPEN	OPEN	N	H-27649	
000099	P06	08A	2E11-F004B	RHR PUMP 2B SUCTION VALVE	H-26014	REACTOR	087	RL/R24	R		OPEN	OPEN	N	H-27649	
000100	P06	08A	2E11-F006B	RHR SDC SUCTION ISOLATION VLV	H-26014	REACTOR	087	RL/R24	R		CLSD	CLSD	N	H-27649	
000101	P06	08A	2E11-F007B	RHR PUMP 2D MIN FLOW BYPASS VL	H-26014	REACTOR	087	RL/R24	RS		OPEN	OP/CL	Y	H-27650	
000102	P06	08A	2E11-F010	RHR HX HDR CROSSTIE VALVE	H-26014	REACTOR	087	RH/R19	R		CLSD	CLSD	N	H-27647	
000103	P06	08A	2E11-F011B	RHR HX B DRAIN TO SUPP POOL	H-26014	REACTOR	087	RL/R24	R		CLSD	CLSD	N	H-27649	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	<---------- Building (7)	EQUIPMENT Flr.Elv. (8)	LOCATION -----> Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	<-- OP. Normal (12)	ST. --> Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)
000104	P06	08A	2E11-F016B	CONT SPRAY DISCHARGE VALVE	H-26014	REACTOR	158	RF/R23	R		CLSD	CLSD	N	H-27649	
000105	P06	08A	2E11-F024B	RHR TEST LINE VALVE	H-26014	REACTOR	087	RF/R24	RS		CLSD	OPEN	Y	H-27649	
000106	P06	08A	2E11-F026B	RHR HX TO RCIC VALVE	H-26014	REACTOR	087	RF/R24	R		CLSD	CLSD	N	H-27649	
000107	P06	08A	2E11-F027B	SUPP POOL SPRAY VALVE	H-26014	REACTOR	087	RF/R24	R		CLSD	CLSD	N	H-27649	
000108	P06	08A	2E11-F028B	RHR INLET TO SUPP POOL	H-26014	REACTOR	087	RH/R24	RS		CLSD	OPEN	Y	H-27649	
000109	P06	08A	2E11-F047B	RHR HX B INLET VALVE	H-26014	REACTOR	087	RL/R24	R		OPEN	OPEN	N	H-27649	
000110	P06	08A	2E11-F048B	RHR HX B BYPASS VALVE	H-26014	REACTOR	087	RL/R24	RS		OPEN	CLSD	Y	H-27649	
000111	P06	08A	2E11-F049	RHR RADWASTE DISCH ISOL VLV	H-26014	REACTOR	087	RJ/R21	R		CLSD	CLSD	N	H-27450	
000112	P06	07	2E11-F065B	RHR PUMP 2B SUPP POOL SUCT VLV	H-26014	REACTOR	087	RL/R19	R		OPEN	OPEN	N	H-27653	
000113	P06	08A	2E11-F068B	RHR HX B TUBE TO SHELL OUTLET	H-21039	REACTOR	087	RH/R24	RS		CLSD	OP/CL	Y	H-27650	
000114	P06	08A	2E11-F073B	RHR HX B SERVICE WATER DISCH	H-26014	REACTOR	087	RH/R24	R		CLSD	CLSD	N	H-27653	
000115	P06	08A	2E11-F091B	HPCI DISCH TO HX B VALVE	H-26014	REACTOR	087	RH/R23			CLSD	CLSD	N	H-27651	
000116	P06	08A	2E11-F104B	RHR HX B VENT VALVE	H-26014	REACTOR	087	RL/R24	R		CLSD	CLSD	N	H-27650	
000117	P06	08A	2E11-F119B	RHR HX B BYPASS VALVE	H-21039	REACTOR	087	RL/R24	R		CLSD	CLSD	N	H-27537	
000118	P06	18	2E11-K600B	RHR HDR FLOW SQ ROOT CONVERTER	H-26014	CONTROL	164	2H11-P612	R		N/A	N/A	Y	H-27644	
000223	P06	20	2E11-K603B	POWER SUPPLY	H-26015	CONTROL	164	2H11-P612	R		N/A	N/A	Y	H-27652	
000225	P06	20	2E11-K613B	AMPLIFIER (K628B)	H-26014	CONTROL	164	2H11-P612	R		N/A	N/A	Y	H-27652	
000119	P06	18	2E11-N002B	RHR HX B TUBE TO SHELL DP TRAN	H-26014	REACTOR	087	2H21-P421A	R		N/A	N/A	Y	H-27652	
000120	P06	18	2E11-N007B	RHRSW FLOW TRANSMITTER	H-21039	REACTOR	087	2H21-P421A	R		N/A	N/A	Y	H-27644	
000121	P06	18	2E11-N015B	RHR HX B DISCH HDR FLOW TRANS	H-26014	REACTOR	087	2H21-P421A	R		N/A	N/A	Y	H-27644	
000122	P06	18	2E11-N017B	RHR HX B INLET PRESS SW	H-21039	REACTOR	087	RH/R24	RS		N/A	N/A	Y	H-27641	
000123	P06	18	2E11-N017D	RHR HX B INLET PRESS SW	H-21039	REACTOR	087	RH/R24	RS		N/A	N/A	Y	H-27641	
000124	P06	18	2E11-N082B	RHR PUMP 2B & 2D FLOW DP TRANS	H-26014	REACTOR	087	2H21-P421A	R		N/A	N/A	Y	H-24426	
000125	P06	18	2E11-N682B	RHR PUMP 2B&2D FLOW DP TRIP UN	H-26014	CONTROL	164	2H11-P926	R		N/A	N/A	Y	H-24426	
000126	P06	18	2E11-R600B	RHR HX B TUBE TO SHELL DP CONT	H-26014	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-27652	
000222	P06	18	2E11-R602B	RHRSW HEAT EXCHANGER B INLET	H-21039	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-27644	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000127	P06	18	2E11-R603B	RHR HX B DISCH HDR FLOW IND	H-26014	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-27644	
000224	P06	18	2E11-R628B	RHR SERVICE WATER DISCHARGE C	H-26014	CONTROL	164	2H11-P612	R		N/A	N/A	Y	H-27635	
000128	P06	18	2E11-S600B	RHR HX B TUBE TO SHELL POS MOD	H-26014	CONTROL	164	2H11-P612	R		N/A	N/A	Y	H-27643	
000778	P06	0	2R34-S006D	SURGE PAK FOR 2E11-C001D	H-27656	INTAKE	111		S		N/A	N/A	Y		
000368	P07	21	2E11-B001B	RHR HEAT EXCHANGER B	H-26014	REACTOR	087	RL/R24			N/A	N/A	N/A		
000369	P07	06	2E11-C001D	RHRSW PUMP 2D	H-21039	INTAKE	111	+			OFF	ON	Y	H-27657	
000370	P07	06	2E11-C002B	RHR PUMP 2B	H-26014	REACTOR	087	RL/R24			OFF	ON	Y	H-27657	
000371	P07	08A	2E11-F003B	RHR HX B DISCHARGE VALVE	H-26014	REACTOR	087	RL/R24			OPEN	OPEN	N	H-27649	
000372	P07	08A	2E11-F004B	RHR PUMP 2B SUCTION VALVE	H-26014	REACTOR	087	RL/R24			OPEN	OPEN	N	H-27649	
000373	P07	08A	2E11-F006B	RHR SDC SUCTION ISOLATION VLV	H-26014	REACTOR	087	RL/R24			CLSD	CLSD	N	H-27649	
000374	P07	08A	2E11-F007B	RHR PUMP 2D MIN FLOW BYPASS VL	H-26014	REACTOR	087	RL/R24			OPEN	OP/CL	Y	H-27650	
000375	P07	08A	2E11-F010	RHR HX HDR CROSSTIE VALVE	H-26014	REACTOR	087	RH/R19			CLSD	CLSD	N	H-27647	
000405	P07	08A	2E11-F015B	INBOARD INJECTION ISOLATION	H-26014	REACTOR	130	RJ/R21			CLSD	CLSD	N	H-27650	
000376	P07	08A	2E11-F016B	CONT SPRAY DISCHARGE VALVE	H-26014	REACTOR	158	RF/R23	S		CLSD	OPEN	Y	H-27649	
000406	P07	08A	2E11-F021B	CONTAINMENT SPRAY INBOARD	H-26014	REACTOR	158	RH/R23	S		CLSD	OPEN	Y	H-27649	
000377	P07	08A	2E11-F024B	RHR TEST LINE VALVE	H-26014	REACTOR	087	RF/R24			CLSD	CLSD	N	H-27649	
000378	P07	08A	2E11-F026B	RHR HX TO RCIC VALVE	H-26014	REACTOR	087	RF/R24			CLSD	CLSD	N	H-27649	
000379	P07	08A	2E11-F027B	SUPP POOL SPRAY VALVE	H-26014	REACTOR	087	RF/R24	S		CLSD	OPEN	Y	H-27649	
000380	P07	08A	2E11-F028B	RHR INLET TO SUPP POOL	H-26014	REACTOR	087	RH/R24			CLSD	OPEN	Y	H-27649	
000381	P07	08A	2E11-F047B	RHR HX B INLET VALVE	H-26014	REACTOR	087	RL/R24			OPEN	OPEN	N	H-27649	
000382	P07	08A	2E11-F048B	RHR HX B BYPASS VALVE	H-26014	REACTOR	087	RL/R24			OPEN	CLSD	Y	H-27649	
000383	P07	08A	2E11-F049	RHR RADWASTE DISCH ISOL VLV	H-26014	REACTOR	087	RJ/R21			CLSD	CLSD	N	H-27450	
000384	P07	07	2E11-F065B	RHR PUMP 2B SUPP POOL SUCT VLV	H-26014	REACTOR	087	RL/R19			OPEN	OPEN	N	H-27653	
000385	P07	08A	2E11-F068B	RHR HX B TUBE TO SHELL OUTLET	H-21039	REACTOR	087	RH/R24			CLSD	OP/CL	Y	H-27650	
000386	P07	08A	2E11-F073B	RHR HX B SERVICE WATER DISCH	H-26014	REACTOR	087	RH/R24			CLSD	CLSD	N	H-27654	
000387	P07	08A	2E11-F091B	HPCI DISCH TO HX B VALVE	H-26014	REACTOR	087	RH/R23			CLSD	CLSD	N	H-27651	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000388	P07	08A	2E11-F104B	RHR HX B VENT VALVE	H-26014	REACTOR	087	RL/R24			CLSD	CLSD	N	H-27650	
000389	P07	08A	2E11-F119B	RHR HX B BYPASS VALVE	H-21039	REACTOR	087	RL/R24			CLSD	CLSD	N	H-27537	
000390	P07	18	2E11-K600B	RHR HDR FLOW SQ ROOT CONVERTER	H-26014	CONTROL	164	2H11-P612			N/A	N/A	Y	H-27644	
000402	P07	20	2E11-K603B	POWER SUPPLY	H-26015	CONTROL	164	2H11-P612			N/A	N/A	Y	H-27652	
000404	P07	20	2E11-K613B	AMPLIFIER (K628B)	H-26014	CONTROL	164	2H11-P612			N/A	N/A	Y	H-27652	
000391	P07	18	2E11-N002B	RHR HX B TUBE TO SHELL DP TRAN	H-26014	REACTOR	087	2H21-P421A			N/A	N/A	Y	H-27652	
000392	P07	18	2E11-N007B	RHRSW FLOW TRANSMITTER	H-21039	REACTOR	087	2H21-P421A			N/A	N/A	Y	H-27644	
000393	P07	18	2E11-N015B	RHR HX B DISCH HDR FLOW TRANS	H-26014	REACTOR	087	2H21-P421A			N/A	N/A	Y	H-27644	
000394	P07	18	2E11-N017B	RHR HX B INLET PRESS SW	H-21039	REACTOR	087	RH/R24			N/A	N/A	Y	H-27641	
000395	P07	18	2E11-N017D	RHR HX B INLET PRESS SW	H-21039	REACTOR	087	RH/R24			N/A	N/A	Y	H-27641	
000396	P07	18	2E11-N082B	RHR PUMP 2B & 2D FLOW DP TRANS	H-26014	REACTOR	087	2H21-P421A			N/A	N/A	Y	H-24426	
000397	P07	18	2E11-N682B	RHR PUMP 2B&2D FLOW DP TRIP UN	H-26014	CONTROL	164	2H11-P926			N/A	N/A	Y	H-24426	
000398	P07	18	2E11-R600B	RHR HX B TUBE TO SHELL DP CONT	H-26014	CONTROL	164	2H11-P601			N/A	N/A	Y	H-27652	
000401	P07	18	2E11-R602B	RHRSW HEAT EXCHANGER B INLET	H-21039	CONTROL	164	2H11-P601			N/A	N/A	Y	H-27644	
000399	P07	18	2E11-R603B	RHR HX B DISCH HDR FLOW IND	H-26014	CONTROL	164	2H11-P601			N/A	N/A	Y	H-27644	
000403	P07	18	2E11-R628B	RHR SERVICE WATER DISCHARGE C	H-26014	CONTROL	164	2H11-P612			N/A	N/A	Y	H-27635	
000400	P07	18	2E11-S600B	RHR HX B TUBE TO SHELL POS MOD	H-26014	CONTROL	164	2H11-P612			N/A	N/A	Y	H-27643	
000780	P07	0	2R34-S006D	SURGE PAK FOR 2E11-C001D	H-27656	INTAKE	111				N/A	N/A	Y		
000234	P08	08A	1P41-F313C	UNIT 2 PSW ISOLATION	D-11001	INTAKE			R		CLSD	CLSD	Y	H-23696	
000238	P08	08A	2N71-F013	CIRC WATER BLOWDOWN	H-21026	YARD			R		CLSD	CLSC	N	H-23686	
000228	P08	06	2P41-C001B	PLANT SERVICE WATER PUMP B	H-21033	INTAKE	111		RS		ON/OFF	ON	Y	H-23699	
000237	P08	07	2P41-F035B	T41B005B CONTROL VALVE	H-26051	REACTOR	130	RH/R25	RS		CLSD	OPEN	Y	H-27759	
000236	P08	07	2P41-F036B	T41B002B CONTROL VALVE	H-26051	REACTOR	098	RL/R24	RS		CLSD	OPEN	Y	H-27757	
000235	P08	07	2P41-F037B	E11C002B CONTROL VALVE	H-26051	REACTOR	098	RL/R24	RS		CLSD	OPEN	Y	H-27748	
000242	P08	07	2P41-F067	PSW DIV. I-II	H-26054	REACTOR	125	RB/R19	R		OPEN	OPEN	N	H-27748	
000233	P08	08A	2P41-F310	RADWASTE DILUTION	H-21033	YARD			R		CLSD	CLSD	N	H-23697	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000231	P08	08A	2P41-F312B	DIESEL GENERATOR 2C ISOL	H-21033	YARD			R		OPEN	OPEN	N	H-23695	
000230	P08	08A	2P41-F315B	REACTOR BUILDING ISOLATION	H-21033	YARD			R		OPEN	OPEN	N	H-23695	
000229	P08	08A	2P41-F316B	TURBINE BUILDING ISOLATION	H-21033	YARD			RS		OPEN	CLSD	Y	H-23695	
000232	P08	07	2P41-F339B	DIESEL GENERATOR 2C OUTLET	H-21033	DIESEL	130	B01	RS		CLSD	OPEN	Y	H-23697	
000239	P08	18	2P41-K601B	ELECTRICAL SUPPLY	H-21033	CONTROL	164	2H11-P656	R		N/A	N/A	Y	H-23633	
000240	P08	18	2P41-N303B	PSW DISCHARGE PT	H-21033	INTAKE	088		RS		N/A	N/A	Y	H-23633	
000241	P08	18	2P41-R601B	PSW DIV. 2 PI	H-21033	CONTROL	164	2H11-P650	R		N/A	N/A	Y	H-23633	
000775	P08	0	2R34-S005B	SURGE PAK FOR 2P41-C001B	H-23699	INTAKE	111		S		N/A	N/A	Y		
000256	P08	10	2T41-B002B	CS/RHR PUMP ROOM COOLER B	H-26071	REACTOR	087	RL/R24	RS		OFF	ON	Y	H-27757	
000257	P08	10	2T41-B005B	HPCI PUMP ROOM COOLER B	H-26071	REACTOR	087	RG/R25	RS		OFF	ON	Y	H-27759	
000258	P08	19	2T41-N019B	HPCI PUMP ROOM COOLER TE	H-26071	REACTOR	087	RH/R25	RS		N/A	N/A	Y	H-27234	
000259	P08	19	2T41-N020B	CS/RHR PUMP ROOM COOLER TE	H-26071	REACTOR	087	RL/R24	RS		N/A	N/A	Y	H-27234	
000260	P08	18	2T41-R609B	HPCI PUMP ROOM COOLER TIS	H-26071	CONTROL	164	2H11-P700	R		N/A	N/A	Y	H-27234	
000261	P08	18	2T41-R610B	CS/RHR PUMP ROOM COOLER TIS	H-26071	CONTROL	164	2H11-P700	R		N/A	N/A	Y	H-27234	
000265	P09	0	2B21-A003D	SRV AIR ACCUMULATOR	H-26000	DRYWELL	148	AZ133	S		N/A	N/A	N		
000266	P09	0	2B21-A003G	SRV AIR ACCUMULATOR	H-26000	DRYWELL	148	AZ080	S		N/A	N/A	N		
000267	P09	0	2B21-A003H	SRV AIR ACCUMULATOR	H-26000	DRYWELL	148	AZ118	S		N/A	N/A	N		
000268	P09	0	2B21-A003M	SRV AIR ACCUMULATOR	H-26000	DRYWELL	148	AZ078	S		N/A	N/A	N		
000270	P09	0	2P70-A002A	EMERGENCY NITROGEN BOTTLE	H-26066	REACTOR	130	RB/R23	S		OFF	ON	N		
000271	P09	0	2P70-A002B	EMERGENCY NITROGEN BOTTLE	H-26066	REACTOR	130	RB/R23	S		OFF	ON	N		
000272	P09	0	2P70-A002C	EMERGENCY NITROGEN BOTTLE	H-26066	REACTOR	130	RB/R23	S		OFF	ON	N		
000277	P09	0	2P70-F084	EMERGENCY NITROGEN ISOLATION	H-26066	REACTOR	130	RB/R21	RS		CLSD	OPEN	N		
000273	P09	0	2P70-F138A	EMERGENCY NITROGEN CONTROL	H-26066	REACTOR	130	RB/R23	RS		CLSD	OPEN	N		
000274	P09	0	2P70-F138B	EMERGENCY NITROGEN CONTROL	H-26066	REACTOR	130	RB/R23	RS		CLSD	OPEN	N		
000275	P09	0	2P70-F138C	EMERGENCY NITROGEN CONTROL	H-26066	REACTOR	130	RB/R23	RS		CLSD	OPEN	N		
000276	P09	0	2P70-F141	EMERGENCY NITROGEN CONTROL	H-26066	REACTOR	130	RB/R23	RS		CLSD	OPEN	N		

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000269	P09	07	2T48-F112B	NITROGEN FLOW CONTROL	H-26083	REACTOR	087	RE/R19	R		CLSD	CLSD	N	H-27773	
000298	P10	18	2B21-N085B	RPV WATER LEVEL LT	H-26001	REACTOR	130	2H21-P410	R		N/A	N/A	Y	H-24426	
000295	P10	18	2B21-N090D	RPV PRESSURE PT	H-26001	REACTOR	158	2H21-P405A	R		N/A	N/A	Y	H-24430	
000292	P10	18	2B21-N091B	RPV WATER LEVEL LT	H-26001	REACTOR	158	2H21-P405A	R		N/A	N/A	Y	H-24426	
000767	P10	18	2B21-N093B	RPV LEVEL 8 LT	H-26001	REACTOR	158	RH/R17	RS		N/A	N/A	Y	H-24426	
000297	P10	20	2B21-N685B	RPV WATER LEVEL LIS	H-26001	CONTROL	164	2H11-P926	R		N/A	N/A	Y	H-24426	
000294	P10	20	2B21-N690D	RPV PRESSURE PIS	H-26001	CONTROL	164	2H11-P928	R		N/A	N/A	Y	H-24430	
000291	P10	20	2B21-N691B	RPV WATER LEVEL LIS	H-26001	CONTROL	164	2H11-P926	R		N/A	N/A	Y	H-24426	
000290	P10	20	2B21-R604B	RPV WATER LEVEL	H-26001	CONTROL	164	2H11-P603	R		N/A	N/A	Y	H-24435, H27297	
000296	P10	20	2B21-R610	RPV WATER LEVEL LI	H-26001	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-24435, H13138	
000293	P10	20	2B21-R623B	RPV PRESSURE	H-26001	CONTROL	164	2H11-P601	R		N/A	N/A	Y	H-24435	
000311	P11	18	2T47-K600	SIGNAL CONVERTER	H-26284	CONTROL	164	2H11-P691	R		N/A	N/A	Y	H-24547	
000308	P11	18	2T47-R627	TORUS TEMPERATURE RECORDER	H-26074	CONTROL	164	2H11-P650	R		N/A	N/A	Y	H-24548	
000316	P11	07	2T48-F361B	TORUS LEVEL INST. ISOLATION	H-26084	REACTOR	087	RF/R24	R		OPEN	OPEN	N	H-27776	
000317	P11	07	2T48-F362B	TORUS LEVEL INST. ISOLATION	H-26084	REACTOR	087	RB/R20	R		OPEN	OPEN	N	H-27776	
000312	P11	18	2T48-K604B	INSTRUMENT DISTRIBUTION	H-26284	CONTROL	164	2H11-P691	R		N/A	N/A	Y		
000315	P11	18	2T48-K621B	TORUS LEVEL INST I/V	H-26284	CONTROL	164	2H11-P691	R		N/A	N/A	Y	H-27778	
000309	P11	18	2T48-N009B	TORUS WATER TE	H-26084	REACTOR	087	TORUS	RS		N/A	N/A	Y	h-24547	
000310	P11	18	2T48-N009D	TORUS WATER TE	H-26084	REACTOR	087	TORUS	RS		N/A	N/A	Y	H-24547	
000313	P11	18	2T48-N010B	TORUS WATER LT	H-26084	REACTOR	087	RE/R24	R		N/A	N/A	Y	H-27778	
000314	P11	18	2T48-R622B	TORUS WATER LEVEL INDICATOR	H-26084	CONTROL	164	2H11-P654	R		N/A	N/A	Y	H-27778	
000771	P12	18	2E11-N094B	DRYWELL PRESSURE PT	H-26014	REACTOR	158	RH/R21	RS		N/A	N/A	Y	H-24430	
000773	P12	18	2E11-N094D	DRYWELL PRESSURE PT	H-26014	REACTOR	158	RH/R21	RS		N/A	N/A	Y	H-24430	
000424	P12	0	2R43-A005C	DG 2C STARTING AIR RECEIVER	H-21074	DIESEL	130	C01	S		N/A	N/A	N		
000425	P12	0	2R43-A006C	DG 2C STARTING AIR RECEIVER	H-21074	DIESEL	130	C01	S		N/A	N/A	N		
000423	P12	18	2R43-N003C	DG 2C DAY TANK LS	H-21074	DIESEL	130	C01	RS		N/A	N/A	Y	H-23801	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000726	P12	20	2R43-P001C	DIESEL GEN 2C CONT PANEL		DIESEL	130		S		N/A	N/A	N		
000419	P12	17	2R43-S001C	DIESEL GENERATOR 2C	H-21074	DIESEL	130		RS		OFF	ON	Y	H-23801	
000421	P12	21	2Y52-A001C	DG 2C FUEL OIL STORAGE TANK	H-21074	YARD			S		N/A	N/A	N		
000420	P12	21	2Y52-A101C	DG 2C FUEL OIL DAY TANK	H-21074	DIESEL	130	C01	S		N/A	N/A	N		
000422	P12	05	2Y52-C101C	DG 2C FUEL OIL PUMP 2C2	H-21074	YARD			RS		OFF	ON	Y	H-23801	
000500	P13	20	2H21-P199	AMMETER SHUNT BATTERY CHG	H-23025	DIESEL	130	C02	S		ON	ON	N	H-23371	
000688	P13	20	2H21-P249	250 VOLT DC SWGR 2B CNT PNL	H-23240	CONTROL	130	TC/T13	S		N/A	N/A	N		
000503	P13	20	2H21-P288	BATTERY SHUNT BOX D	H-23235	CONTROL	112	TC/T13	S		ON	ON	N	H-23390	
000504	P13	20	2H21-P289	BATTERY SHUNT BOX E	H-23235	CONTROL	112	TC/T14	S		ON	ON	N	H-23390	
000505	P13	20	2H21-P290	BATTERY SHUNT BOX F	H-23235	CONTROL	112	TE/T13	S		ON	ON	N	H-23390	
000501	P13	20	2H21-P293	BATTERY 2C FUSE BOX	H-23022	DIESEL	130	C02	S		ON	ON	N	H-23371	
000493	P13	04	2R11-S006	LTG & MISC POWER XFMR	H-23023	DIESEL	130	C03	S		N/A	N/A	Y	H-23317	
000496	P13	04	2R11-S012	600/208 V XFMR	H-23027	YARD					N/A	N/A	N	H-23380	
000492	P13	04	2R11-S042	CONT BLDG ESS XFMR 2C	H-23240	CONTROL	130	TC/T13	S		N/A	N/A	Y	H-23369	
000494	P13	18	2R20M-P002	FUSE BOX	H-23023	DIESEL	130	C02	S		N/A	N/A	N	H-23317	
000495	P13	18	2R20N-P002	FUSE BOX	H-23240	CONTROL	130		S		N/A	N/A	N	H-23369	
000469	P13	03	2R22-S007	4160V STA SVC SWGR 2G	H-23023	DIESEL	130	C02	SR		N/A	N/A	Y	H-23358	
000470	P13	02	2R22-S017	250 V DC BATTERY SWGR 2G	H-23239	CONTROL	130	TB/T13	SR		N/A	N/A	Y	H-23390	
000471	P13	02	2R23-S004	600 V STA SVC SWGR 2D & XFMR	H-23240	CONTROL	130	TCA/T14	SR		N/A	N/A	Y	H-23362	
000472	P13	01	2R24-S010	600/208 V MCC 2B INTAKE STRU	H-23027	INTAKE	111		S		N/A	N/A	N	H-23380	
000473	P13	01	2R24-S012	600 V MCC 2B ESS DIV 2	H-27281	REACTOR	130	RF/R24	RS		N/A	N/A	Y	H-27013,H-27014	
000474	P13	01	2R24-S012B	600 V ESS MCC	H-27298	REACTOR	164	RB/R21	RS		N/A	N/A	Y	H-27010	
000478	P13	01	2R24-S018B	600 V MCC 2E-B ESS DIV 2	H-27279	REACTOR	130	RJ/R17	RS		N/A	N/A	Y	H-27021	
000475	P13	01	2R24-S022	125/250 V DC MCC 2B ESS DIV 2	H-27281	REACTOR	130	RH/R24	RS		N/A	N/A	Y	H-27024	
000476	P13	01	2R24-S027	600/208 V MCC 2C ESS DIV 2	H-23023	DIESEL	130	C02	RS		N/A	N/A	Y	H-23317	
000479	P13	14	2R25-S002	125 V DC CABINET 2B	H-23240	CONTROL	130	TCA/T13	S		N/A	N/A	Y	H-23390	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DGF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM v0.0

LINE NO. (1)	TRAIN (2)	EQUIP CLASS (3)	MARK NO. (4)	SYSTEM/EQUIPMENT DESCRIPTION (5)	Dwg. No./Rev./Zone (6)	<---------- Building (7)	EQUIPMENT Flr.Elv. (8)	LOCATION -----> Rm. or Row/Col. (9)	SORT (10)	NOTES (11)	<-- OP. Normal (12)	ST. --> Desired (13)	POWER REQD? (14)	SUPPORTING SYS. DWG. NO./REV. (15)	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS (16)
000480	P13	14	2R25-S006	125 V DC CABINET 2F	H-23023	DIESEL	130	C02	S		N/A	N/A	Y	H-23371	
000481	P13	14	2R25-S031	120/208 V AC CABINET 2L	H-23023	DIESEL	130	C02	S		N/A	N/A	Y	H-23317	
000482	P13	14	2R25-S037	120/208 V AC ESS CABINET 2B	H-23240	CONTROL	130	TCA/T13	S		N/A	N/A	Y	H-23369	
000484	P13	14	2R25-S065	120/208 V AC CABINET 2B INST	H-23240	CONTROL	130	TG/T12	S		N/A	N/A	Y	H-23369	
000485	P13	14	2R25-S130	125 V DC DISTRIBUTION CAB 2D	H-23239	CONTROL	130	TDA/T13	S		N/A	N/A	Y	H-24436	
000502	P13	18	2R26-M004	2R25-S025 DISCONNECT SWITCH	H-23240	CONTROL	130	TC/T13	S		N/A	N/A	N	H-23369	
000497	P13	18	2R26-M031C	125 V DC THROWOVER SWITCH 2C	H-23240	CONTROL	130	TB/T13	S		N/A	N/A	N	H-23390	
000498	P13	18	2R26-M031D	125 V DC THROWOVER SWITCH 2D	H-23240	CONTROL	130	TB/T13	S		N/A	N/A	N	H-23390	
000499	P13	18	2R26-M032C	125 V DC THROWOVER SWITCH	H-23025	DIESEL	130	C02	S		N/A	N/A	N	H-23371	
000733	P13	01	2R27-S093	LOCAL STARTER E11-F006	H-27991	REACTOR	130	RH/R24	SR		N/A	N/A	Y	H-51689	
000506	P13	01	2R27-S096	LOCAL STARTER E11-F008	H-27281	REACTOR	130	RH/R21	SR		N/A	N/A	Y	H-27991	
000486	P13	15	2R42-S001B	125/250 V DC STA BATTERY 2B	H-23235	CONTROL	112	TDA/T14	S		N/A	N/A	Y	H-23390	
000487	P13	15	2R42-S002C	125 V DIESEL SYS BATTERY 2C	H-23022	DIESEL	130	C02	S		N/A	N/A	Y	H-23371	
000488	P13	16	2R42-S029	125 V BATTERY CHARGER 2D	H-23240	CONTROL	130	TB/T13	S		N/A	N/A	Y	H-23390	
000489	P13	16	2R42-S030	125 V BATTERY CHARGER 2E	H-23240	CONTROL	130	TB/T13	S		N/A	N/A	Y	H-23390	
000490	P13	16	2R42-S032C	125 V BATTERY CHARGER 2J	H-23025	DIESEL	130	C02	S		N/A	N/A	Y	H-23371	
000491	P13	16	2R44-S003	STATIC INVERTER	H-13131	CONTROL	147	TH/T14	S		ON	ON	Y	H-23390, H27021	
000757	P14	0	2L48-D137	D/G RM 2C FIRE DAMPER	H-23395	DIESEL	130	2C	RS		OPEN	OPEN	N		
000426	P14	09	2X41-C010C	DG 2C ROOM EXHAUST FAN	H-12619	DIESEL	130	B02	RS		OFF	ON	Y	H-23395	
000428	P14	0	2X41-C013B	DG 2C ROOM LOUVER	H-12619	DIESEL	130		RS		CLSD	OPEN	N	H-23395	
000529	P14	09	2X41-C014E	SWGR RM 2G FAN	H-12619	DIESEL	ROOF		RS		OFF	ON	Y	H-23397	
000530	P14	0	2X41-C015C	SWGR RM 2G LOUVER	H-12619	DIESEL	130	SWGR RM 2G	RS		CLSD	OPEN	Y	H-23397	
000429	P14	09	2X41-C016C	DG 2C BATTERY ROOM FAN	H-12619	DIESEL	130	C02	RS		OFF	ON	Y	H-23397	
000759	P14	0	2X41-C024B	D/G BATT RM 2C FIRE DAMPER	H-23395	DIESEL	130	2C	RS		OPEN	OPEN	N		
000761	P14	0	2X41-C024D	D/G BATT RM 2C FIRE DAMPER	H-23395	DIESEL	130	2C	RS		OPEN	OPEN	N		
000430	P14	0	2X41-C028B	DG 2C BATTERY ROOM LOUVER	H-12619	DIESEL	130		RS		CLSD	OPEN	N	H-23397	

APPENDIX F
SAFE SHUTDOWN EQUIPMENT LIST (SSEL)
E. I. HATCH NUCLEAR PLANT - UNIT 2

Data Base File Name/Date/Time: HATCH2R4.DBF / 05/19/95 / 14:12:34
Sort Criteria: Train,ID Number
Filter Criteria:
Program File Name & Version: SSEM [illegible].0

LINE NO.	TRAIN	EQUIP CLASS	MARK NO.	SYSTEM/EQUIPMENT DESCRIPTION	Dwg. No./Rev./Zone	<---------- Building	EQUIPMENT Flr.Elv.	LOCATION -----> Rm. or Row/Col.	SORT	NOTES	<-- OP. Normal	ST. --> Desired	POWER REQD?	SUPPORTING SYS. DWG. NO./REV.	REQ'D INTERCONNECTIONS & SUPPORTING COMPONENTS
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
000763	P14	0	2X41-C030B	D/G RM 2C FIRE DAMPER	H-23395	DIESEL	130	2C	RS		OPEN	OPEN	N		
000765	P14	0	2X41-C030D	D/G RM 2C FIRE DAMPER	H-23395	DIESEL	130	2C	RS		OPEN	OPEN	N		
000427	P14	18	2X41-N011B	DG 2C ROOM FAN THERMOSTAT	H-12619	DIESEL	130	B02	RS		N/A	N/A	Y	H-23395	
000531	P14	18	2X41-N013C	SWGR RM 2G LVR THERMOSTAT	H-12619	DIESEL	130	SWGR RM 2G	RS		N/A	N/A	Y	H-23397	
000532	P14	18	2X41-N042	FLOW SWITCH FOR FAN 2X41-C010C	H-12619	DIESEL	ROOF		RS		N/A	N/A	Y	H-23395	
000533	P14	18	2X41-N061	FLOW SWITCH FOR FAN 2X41-C010D	H-12619	DIESEL	ROOF		RS		N/A	N/A	Y	H-23395	
000732	P14	20	2X43-P003B	CO2 ZONE 2 CONTROL CABINET	H-23747	DIESEL	130	B01	S		N/A	N/A	N		
000528	P14	09	2Z41-C015	STN BATT RM EMERG EXHAUST FAN	H-26093	TURBINE	130	TE/T16	RS		OFF	ON	Y	H-27800	
000544	P15	08A	2G31-F004	RWCU OUTBOARD ISOL GATE VALVE	H-26036	REACTOR	158	RF/R19	S		OPEN	CLSD	Y	H-27462, H27965	
000546	P15	08B	2T48-F311	TORUS VAC BRKR ISOL BUTTERFLY VALVE	H-26084	REACTOR	087	RF/R23			CLSD	CLSD	Y	H-27777	

APPENDIX G

SUMMARY OF REQUIRED OPERATOR ACTIONS

UNIT 1

Unless noted, all actions are performed in the MCR.

PRIMARY PATH SHUTDOWN

High Pressure Coolant Injection System Operation (HPCI) - HPCI initiation will occur automatically. If required, HPCI suction will be aligned to the suppression pool as follows:

Close valve E41-F004.
Open valve E41-F041.
Open valve E41-F042.

Depressurization - Depressurization is controlled by the operators by manually opening SRVs 2B21-F013A, B, C, and E.

LPCS Operation - LPCS is manually initiated as follows:

Start pump E21-C001B.
Open valve E21-F005B.
Throttle E21-F005B to control level.

Suppression Pool Cooling Operation - Suppression pool cooling is required for the alternate shutdown cooling function and may be required to maintain pool temperature within limits during HPCI operation. It is manually operated as follows:

Start residual heat removal service water (RHRSW) pump E11-C001D.
Adjust flow controller E11-R600B.
Open E11-F028B.
Start residual heat removal (RHR) pump E11-C002D.
Confirm or start room cooler T41-B003B.
Throttle open E11-F024B.
Close E11-F048B.
Adjust controller E11-R600B.

Alternate Shutdown Cooling Operation - Alternate shutdown cooling is manually placed in service by operators as follows:

Open one SRV 2B21-F013A, B, C, or E.

Raise level with LPCS and establish flow through the open SRV.

APPENDIX G

SUMMARY OF REQUIRED OPERATOR ACTIONS

ALTERNATE PATH SHUTDOWN

Trip core spray pump E21-C001A if it starts as a result of relay chatter.

Low Pressure Coolant Injection (LPCI) Operation - LPCI is manually initiated at approximately 1330 seconds following the reactor scram as follows:

Start RHR pump E11-C002A.
Confirm or start room cooler T41-B002A.
Open valve E11-F017A.
Depressurize by manually opening SRVs 2B21-F013F, G, H and J.
Open valve E11-F015A.
Close valve B31-F023A.
Throttle valve E11-F017A.

Shutdown Cooling Operation - Shutdown cooling is manually initiated at approximately 42 minutes into the event, after level is restored and pressure is within the shutdown cooling operation limit as follows:

Start RHRSW pump E11-C001A.
Throttle flow at controller E11-R600A.
Close valve E11-F017A.
Open valve E11-F015A.
Trip RHR pump E11-C002A, if running.
Close valve E11-F004A.
Open valve E11-F006A.
Close valve E11-F007A.
Rack out breaker for valve E11-F007A at motor control center (MCC) R24-S018A.
Rack out breaker for valve E11-F028A at MCC R24-S011.
Rack out breaker for valve E11-F016A at MCC R24-S011.
Manually open valve E11-F008.
Open valve E11-F009.
Throttle open valve E11-F017A.
Start RHR pump E11-C002A.
Adjust valve E11-F017A.
Adjust flow at controller E11-R600A.

APPENDIX G

SUMMARY OF REQUIRED OPERATOR ACTIONS

SUPPORT SYSTEMS MANUAL ACTIONS

Plant Service Water (PSW) - If it is not running:

Start PSW pump P41-C001A or B.

Start standby PSW pump 2P41-C002 if it is not running for primary path shutdown.

Isolate the turbine building by closing valve P431-F310A or B if not already closed.

Drywell Pneumatic System

Primary Path	Manually open P70-F141. Manually open valve P70-F138A, B, or C. Manually open valve P70-F084.
Alternate Path	Open P70-F001.

Diesel Generators (DGs) - If DGs do not auto-start because of relay chatter:

Primary Path

Reset generator 1B differential lockout relay 87-D1BX.
Reset generator 1B shutdown logic.
Reset generator 1C differential lockout relay 87-D1CX.
Reset generator 1C shutdown logic.

Alternate Path

Reset generator 1A differential lockout relay 87-D1AX.
Reset generator 1A shutdown logic.

To ensure the operation of switchgear and battery room fans:

Primary Path -

Verify or place control switch X41B-S18 on panel H21-P261 for fan X41-C006C in PRIMARY.
Verify or place control switch X41C-S18 on Panel H21-P262 for fan X41-C006E in PRIMARY.
Verify or place control switch X41B-S16 for fan X41-C008C in PRIMARY.
Verify or place control switch X41C-S16 for fan X41-C008E in PRIMARY.

Alternate Path -

Verify or place control switch X41A-S18 on panel H21-P260 for fan X41-C006A in PRIMARY.
Verify or place control switch X41A-S16 for fan X41-C008A in PRIMARY.

APPENDIX G

SUMMARY OF REQUIRED OPERATOR ACTIONS

Power Sources -

Reset 600-V transformer 1C differential lockout relay 87-S1CX.

Operate control switch for 4160-V switchgear bus 1E, frame 10, breaker 135717, transformer 1C (R23-S003).

Operate control switch for 600-V switchgear bus 1C, frame 2, breaker 135811, transformer 1C (R23-S003).

Reset 600-V bus 1C undervoltage logic.

Operate control switch for 600-V switchgear bus 1C, frame 6T, battery charger (R24-S026)

Operate control switch for 600 V switchgear bus 1C, frame 6M, battery charger (R24-S027)

Reset 600-V transformer 1D differential lockout relay 87-S1DX

Operate control switch for 4160-V switchgear bus 1G, frame 9, breaker 135719, transformer 1D (R23-S004).

Operate control switch for 600-V switchgear bus 1D, frame 9, breaker 135814, transformer 1D (R23-S004).

Reset 600-V bus 1D undervoltage logic.

Operate control switch for 600-V switchgear bus 1D, frame 8T, battery charger (R42-S029).

Operate control switch for 600-V switchgear bus 1D, frame 8M, battery charger (R42-S030).

Align transfer switch R26-M021 to alternate-alternate feed (R25-S036)

Operate control switch C71-S1 to the ALT-A position for reactor protection bus A (C71-P001).

APPENDIX G

SUMMARY OF REQUIRED OPERATOR ACTIONS

UNIT 2

Unless noted, all actions are performed in the MCR.

PRIMARY PATH SHUTDOWN

HPCI System Operation - HPCI initiation will occur automatically. If required, HPCI suction will be aligned to the suppression pool as follows:

Close valve 2E41-F004.
Open valve 2E41-F041.
Open valve 2E41-F042.

Depressurization - Depressurization is controlled by the operators by manually opening SRVs 2B21-F013D, G, H, and M.

LPCS Operation - LPCS is manually initiated as follows:

Reset overcurrent relay targets and lockout relay for pump 2E21-C001B.
Start pump 2E21-C001B.
Open valve 2E21-F005B.
Throttle 2E21-F005B to control level.

Suppression Pool Cooling Operation - Suppression pool cooling is required for the alternate shutdown cooling function and may be required to maintain pool temperature within limits during HPCI operation. It is manually operated as follows:

Reset overcurrent relay targets on 2E11-C001D.
Start RHRSW pump 2E11-C001D.
Adjust flow controller 2E11-R600B.
Open 2E11-F028B.
Reset overcurrent relay targets and lockout relay for pump 2E11-C002B.
Start RHR pump 2E11-C002B.
Confirm or start room cooler 2T41-B002B.
Throttle open 2E11-F024B.
Close 2E11-F048B.
Adjust controller 2E11-R600B.

Alternate Shutdown Cooling Operation - Alternate shutdown cooling is manually placed in service by operators as follows:

Open one SRV 2B21-F013D, G, H, or M.

Raise level with LPCS and establish flow through the open SRV.

APPENDIX G

SUMMARY OF REQUIRED OPERATOR ACTIONS

ALTERNATE PATH SHUTDOWN

Trip core spray pump 2E21-C001A if it starts as a result of relay chatter.

LPCI Operation - LPCI is manually initiated at approximately 1330 seconds following the reactor scram as follows:

Reset targets on overcurrent relays and overcurrent lockout relay.
Start RHR pump 2E11-C002A.
Confirm or start room cooler 2T41-B003A.
Open valve 2E11-F017A.
Depressurize by manually opening SRVs 2B21-F013A, B, F and K.
Open valve 2E11-F015A.
Close valve 2B31-F023A.
Throttle valve 2E11-F017A.

Shutdown Cooling Operation - Shutdown cooling is manually initiated at approximately 42 minutes into the event, after level is restored and pressure is within the shutdown cooling operation limit as follows:

Reset targets on overcurrent relays for pump 2E11-C001A.
Start RHRSW pump 2E11-C001A.
Throttle flow at controller 2E11-R600A.
Close valve 2E11-F017A.
Open valve 2E11-F015A.
Trip RHR pump 2E11-C002A, if running.
Close valve 2E11-F004A.
Open valve 2E11-F006A.
Close valve 2E11-F007A.
Rack out breaker for valve 2E11-F007A at MCC 2R24-S018A.
Rack out breaker for valve 2E11-F028A at MCC 2R24-S011.
Rack out breaker for valve 2E11-F016A at MCC 2R24-S011.
Manually open valve 2E11-F008.
Open valve 2E11-F009.
Throttle open valve 2E11-F017A.
Reset targets on overcurrent relays and reset lockout relay for pump 2E11-C002A.
Start RHR pump 2E11-C002A.
Adjust valve 2E11-F017A.
Adjust flow at controller 2E11-R600A.

APPENDIX G

SUMMARY OF REQUIRED OPERATOR ACTIONS

SUPPORT SYSTEMS MANUAL ACTIONS

Plant Service Water - If it is not running:

Reset overcurrent relay targets and overcurrent lockout relay for pump 2P41-C001A or B

Start PSW pump 2P41-C001A or B.

Isolate the turbine building by closing valve 2P431-F316A or B if not already closed.

Drywell Pneumatic System

Primary Path	Manually open 2P70-F141. Manually open valve 2P70-F138A, B, or C. Manually open valve 2P70-F084.
Alternate Path	Open 2P70-F001.

Diesel Generators - If DGs do not auto-start because of relay chatter:

Primary Path

Reset targets on generator differential relays.
Reset generator 2C differential lockout relay 87-D2CX.
Reset generator 2C shutdown logic.

Alternate Path

Reset targets on generator differential relays.
Reset generator 2A differential lockout relay 87-D2AX.
Reset generator 2A shutdown logic.

To ensure the operation of switchgear and battery room fans:

Primary Path -
Verify or place control switch 2X41-S18C in panel 2H21-P262 for fan 2X41-C014E in PRIMARY.
Verify or place control switch 2X41-S16C in panel 2H21-P259 for fan 2X41-C016C in PRIMARY.

Alternate Path -
Verify or place control switch 2X41-S18A in panel 2H21-P260 for fan 2X41-C014A in PRIMARY.
Verify or place control switch 2X41-S16A in panel 2H21-P657 for fan 2X41-C016A in PRIMARY.

APPENDIX G

SUMMARY OF REQUIRED OPERATOR ACTIONS

<u>Power Sources</u> -

Reset 600-V transformer 2C differential lockout relay 87-S2CX.

Operate control switch for 4160-V switchgear bus 2E, frame 2, breaker 135536, transformer 2C (2R23-S003).

Operate control switch for 600-V switchgear bus 2C, frame 2, breaker 135674, transformer 2C (2R23-S003).

Reset 600-V bus 2C undervoltage logic.

Operate control switch for 600-V switchgear bus 2C, frame 7T, battery charger (2R24-S026).

Operate control switch for 600-V switchgear bus 2C, frame 7M, battery charger 2R24-S027).

Reset 600-V transformer 2D differential lockout relay 87-S2DX.

Operator control switch for 4160-V switchgear bus 2G, frame 2, breaker 135556, transformer 2D (2R23-S004).

Operate control switch for 600-V switchgear bus 2D, frame 8, breaker 135684, transformer 2D (2R23-S004).

Reset 600-V bus 2D undervoltage logic.

Operate control switch for 600-V switchgear bus 2D, frame 3T, battery charger 2R42-S029).

Operate control switch for 600-V switchgear bus 2D, frame 3M, battery charger 2R42-S030).

APPENDIX H

SUMMARY OF SEISMIC RESPONSE ANALYSES PERFORMED FOR THE HATCH SMA

SSI analyses were performed to calculate median-centered in-structure response spectra, maximum relative displacements, and maximum absolute accelerations at each floor level in the control building, reactor building, diesel generator building, and river intake structure. For all structures, SSI analyses were performed for three soil property cases: an intermediate case; a lower bound case for which the soil shear modulus profile was equal to the intermediate case profile factored by 0.60; and an upper bound profile for which the soil shear modulus profile was equal to the intermediate case profile factored by 1.60. Also for all structures, the free-field motions used were those provided for the Hatch SMA seismic response analyses. Unless specified otherwise, the in-structure response spectrum used in the seismic margin assessment is the envelope of the three in-structure response spectra associated with the lower bound, the intermediate, and the upper bound soil modulus profiles with the in-structure response spectra for the intermediate soil modulus broadened by ±15% in the frequency domain. This follows the procedures discussed in Reference 1. A summary of the analyses for each of these structures is contained in the following paragraphs.

Control Building

The structural model for the control building, which is a Unit 1 and 2 shared structure, was a three-dimensional model containing 17 beam elements connecting 18 nodal points. Nodal mass points were defined for each floor level in the structure. Typically, at each floor level, a nodal mass point would be located at the center of mass for that floor slab and two other nodal points would be located at the centers of rigidity of the wall systems above and below the floor. Figure H-1 shows a sketch of the structural model. Eigenvalue extraction analyses were performed to obtain 25 modes. The mode shapes and frequencies were compared with the data received from SCS and good agreement was obtained. Table H-1 summarizes the fixed-base modes of the structural model. SSI response was obtained at all 7 mass points.

The basemat of the control building is located at el 105 ft, approximately 25 ft below grade; however, embedment contact with the side soil occurs on only one side, the other sides being adjacent to other structures of equal or greater embedment. Impedance functions and foundation input motions (wave scattering functions) were therefore calculated for a flat foundation on a truncated soil column extending to a maximum elevation of 105 ft using the CLASSI computer programs GLAY and CLAN. In addition, no variation in free-field motion over the 25 ft from soil free surface at grade to the bottom of the foundation was considered; i.e., the foundation input motions are taken to be equal to the free-field motion (or scattering is unity). This adds a degree of conservatism to the results.

SSI analyses were performed for the three soil property cases described above and maximum absolute accelerations, maximum relative displacements, and response spectra at 3%, 5% and 10% spectral damping were obtained at all mass points.

Reactor Building

This analysis was performed for the Unit 1 reactor building; however, the results are applicable to the Unit 2 reactor building since it is essentially a mirror image of Unit 1. The Unit 1 reactor building was modeled for two assumptions of connectivity of the roof structure with the vestibule. The first or base case assumed that the connection is fully capable of transferring all loads between the roof and vestibule. The second case assumed that the connection does not exist and that the roof structure is completely released from the vestibule. SSI response analyses were carried out for both assumptions.

The base case of the reactor building consisted of three uncoupled models (E-W, N-S, and vertical). These models are similar to those used for a previous study (Reference 2) performed for the NRC. Each horizontal model contained 34 beam elements connecting 30 nodal points. The vertical model was of similar refinement. Nodal mass points were defined for each floor level in the structure, as well as appropriate locations for the drywell, the reactor pedestal and sacrificial shield, and the reactor pressure vessel.

Typically, each model would allow displacements only in the direction for which it was modeled, assuming motions in the other directions to be uncoupled because of the symmetry of the structure. Figure H-2 shows a sketch of the structural model for the N-S and E-W direction. Eigenvalue extraction analyses were performed on each model to obtain a total of 25 modes. Table H-2 summarizes the fixed-base modes of the structural models. SSI response was obtained at all 22 mass points.

The bottom of the reactor building basemat is located at el 75 ft, 55 ft below grade. As with the control building, the reactor building foundation is only partially embedded on three sides, being bordered by other foundations of varying basemat elevations. Therefore, a sensitivity study was conducted to evaluate the effect of embedment and side-soil bonding on the response of the reactor building. For this study, impedance and scattering functions were calculated for two embedment assumptions.

- Foundation Case 1 assumed the foundation to be fully embedded with its sidewalls fully bonded to the soil. For this case, impedance and scattering functions were calculated for an equivalent embedded cylinder using the computer program SUPERALUSH.

- Foundation Case 2 assumed that there was no contact between the foundation sidewalls and the soil. Impedance functions for a flat square foundation were calculated by programs GLAY and CLAN using the soil properties below the base of the foundation. Scattering functions were calculated by using SUPERALUSH assuming an embedded circular cylinder with no sidewall contact with the soil.

For both cases, the horizontal variation of soil properties resulting from the backfill next to the building above the base of the foundation was included and the impedance and scattering functions were calculated for the intermediate soil properties.

SSI analyses were performed for each foundation case and response spectra were obtained on the basemat and in the structure. Comparison of the results showed that the Case 2 soil model produced response at somewhat lower frequencies than the Case 1 model. The Case 2 soil model was judged to better represent the actual configuration of the reactor building foundation. Case 2 was therefore used for the final analyses and the uncertainty of effective embedment was accounted for by broadening the spectra +10% in the frequency domain for subsystems whose fundamental natural frequency is between approximately 2 and 3 Hz. This is done to account for a possible shift of the major spectral peak of the in-structure response spectra caused by a stiffer foundation resulting from side soil contact on a portion of the embedded reactor building. This additional broadening is applied to the envelope of the three spectra with the ±15% broadening of the intermediate soil shear modulus profile spectra previously discussed.

SSI analyses were performed for both structural model cases (connected and released roof) and for the three soil property cases using Foundation Case 2, as described above. Maximum absolute accelerations, maximum relative displacements, and response spectra at 3%, 5%, and 10% spectral damping were obtained at all mass points.

Diesel Generator Building

The diesel generator building, which is a Unit 1 and 2 shared structure, was modeled by three uncoupled models (E-W, N-S, and vertical). These models are very similar to those used for the previous study made for the NRC. Each model contained one beam element connecting two nodal points. Nodal mass points were defined for each floor level. Typically, each model would allow displacements only in the direction which it modeled, assuming motions in the other directions to be uncoupled because of the symmetry of the structure. Figure H-3 shows a sketch of the structural model. Eigenvalue extraction analyses were performed on each model to obtain a total of three modes. The mode shapes and frequencies were compared with hand calculations and good agreement was seen. Table H-3 summarizes the fixed-base modes of the structural models. SSI response was obtained at all mass points.

The bottom of the basemat of the diesel generator building is located at el 125 ft, 5 ft below grade. Because of the shallow embedment, impedance functions were calculated for a flat foundation on a truncated soil column extending to a maximum elevation of 125 ft and the foundation input motions were taken to be equal to the free-field motions; i.e., scattering was unity. The CLASSI computer programs GLAY and CLAN were used for this calculation.

SSI analyses were performed for the three soil property cases, as described above. Maximum absolute accelerations, maximum relative displacements, and response spectra at 3%, 5%, and 10% spectral damping were obtained at all mass points.

River Intake Structure

The structural model for the intake structure, which is a Unit 1 and 2 shared structure, was a three-dimensional model containing 13 beam elements connecting 14 nodal points. Nodal mass points were defined for each floor level in the structure. Typically, at each floor level, a nodal mass point would be located at the center of mass for that floor slab and two other nodal points would be located at the centers of rigidity of the wall systems above and below the floor. Because of the symmetry of the intake structure about its plane perpendicular to the riverbed, modeling eccentricities extended only in that plane. Figure H-4 shows a sketch of the structural model. Eigenvalue extraction analyses were performed to obtain 9 modes. The mode shapes and frequencies were compared with the data received from SCS and good agreement was obtained. Table H-4 summarizes the fixed-base modes of the structural model. SSI response was obtained at all five mass points.

The foundation conditions for the intake structure are different from those of any other Plant Hatch structure; the structure is fully embedded at one end by a berm at el 110 ft, has no embedment at the other end where its basemat meets the riverbed at el 52 ft, and both sides have embedment conditions which vary from full embedment to no embedment. To approximate these conditions, impedance functions were calculated assuming a flat foundation having the shape of the intake structure's basemat and augmenting them with corrections obtained as the difference between a circular cylinder embedded in a uniform halfspace having properties of the side soil and a flat foundation with the same properties.

The uncertainty in the foundation input motion (FIM) was addressed by conducting a sensitivity study of the effect of control point location on in-structure response. The effect of locating the control point for the free-field motion on the berm as opposed to locating it on the riverbed was studied by first calculating scattering functions for the longitudinal cross-section of the intake structure (perpendicular to the riverbed) relative to both locations and then using the two sets of scattering functions, along with the impedance functions described above, in SSI analyses of the intake structure. The scattering functions were calculated using program SUPERFLUSH with a model having the berm (el 110 ft) on one side of the rigid massless foundation and the riverbed (el 52 ft) on the other side. Full bonding between soil and foundation was assumed. Scattering functions were obtained for two cases:

- Case 1 related foundation input motion to free-field motion on the berm.

- Case 2 related foundation input motion to free-field motion on the riverbed.

SSI analyses were performed for each case and response spectra were obtained on the basemat and in the structure. Comparison of the results showed that the response for Case 1 was generally higher than for Case 2. Case 1 scattering functions were therefore used for the final SSI analyses.

Scattering functions for components in the transverse direction (parallel to the riverbank) were taken to be unity; i.e., foundation input motion equals free-field motion.

SSI analyses were performed for the three soil property cases and maximum absolute accelerations, maximum relative displacements, and response spectra at 3%, 5%, and 10% spectral damping were obtained at all mass points.

Perspective on SSI Response of the Reactor Building

The in-structure response spectra calculated for the reactor building demonstrate low frequency amplification; i.e., soil/structure system frequencies in the range of 1 to 2 Hz depending on the assumed soil profile. The Hatch reactor building is characterized by a relatively stiff structure founded on relatively soft effective soil properties. The principal horizontal frequencies of the Hatch reactor building are approximately 7 Hz in each direction. Hence, the resulting soil/structure system frequency of 1 to 2 Hz is a direct result of the soil stiffness properties.

Hatch is not unique in this respect. Appendix A of Reference 3 tabulates information for all U.S. commercial nuclear power plants including site conditions by general category including stratigraphy and stiffness. Many are listed with general site descriptions similar to that of Hatch. A subset of these has been reanalyzed recently for SSE level or greater excitations; i.e., in the context of a PRA. Two, in particular, which demonstrate low frequency soil/structure system frequencies are Surry and La Salle. Surry structures were analyzed for 1SSE (0.15g), 2SSE (0.30g), and 3SSE (0.45g) level earthquakes. The Surry reactor building/internal structure demonstrates low frequency (1 to 2 Hz) response for the 1SSE case with lower system frequencies for the 2SSE and 3SSE cases. These Surry structures are relatively stiff and their overall response is governed by the behavior of the soil/foundation. Not that Surry is a PWR. The La Salle nuclear power station presents a second example where a relatively stiff structural system is supported on a relatively soft soil leading to low frequency soil/structure system modes. La Salle is a Mark II BWR with a very large foundation.

In addition to calculated response, a limited amount of recorded data from past earthquakes has been obtained for EPRI (Reference 4). Figures H-5 and H-6 (Reference 5) show two cases of recorded response on the structure's basemat and operating floor. Figure H-5 is a BWR Mark II containment situated on a relatively uniform soil profile of low strain shear wave velocity of 1500 ft/s. The earthquake event was very low level; i.e., approximately 5 gals. ZPA on the basemat and slightly less than 10 gals. ZPA on the operating floor. These 5% damped spectra demonstrate a low frequency soil/structure system frequency of approximately 3 Hz. This very low level event is unlikely to have induced significant strains in the soil; consequently, no significant softening of the soil would be expected. If, as in the cases of Hatch, Surry, and La Salle, a reduction in shear wave velocity of 30 to 50% were anticipated for an earthquake of 0.30 g pga and the structures were assumed to behave rigidly, a reduction in frequency by a factor of 2 or more could be expected, bringing the soil/structure system frequency into the same range as that calculated for Hatch. Figure H-6 shows the same type of data for a different site and structure. The site is reported to be an average soil site which translates to a shear wave velocity of approximately 1500 ft/s as above. The structure is a PWR containment/internal structure and the responses shown in Figure H-6 are for the basemat and operating floor. As above, the principal frequency is at 3 Hz for this very low level event. If significant softening of the soil is expected for a 0.30 g pga earthquake, a frequency reduction factor of 2 or greater could be expected, as discussed above.

The analytical and recorded dat~ support the calculated frequencies of the Hatch reactor building/soil system given the hypothesized occurrence of an earthquake with pga of 0.30 g.

REFERENCES

1. ASCE Standard - Seismic Analysis of Safety Related Nuclear Structures. American Society of Civil Engineers, ASCE 4-86. September 1986.

2. J.. Johnson, O.R. Maslenikov, D.J. Doyle, Review of Seismic Analysis of Hatch Units 1 and 2: In-Structure Response Spectra. Lawrence Livermore National Laboratory, Livermore, CA, UCID-21015, 1987.

3. J.J. Johnson, E.C. Schewe, and O.R. Maslenikov, SSI Response of a Typical Shear Wall Structure. Lawrence Livermore National Laboratory, Livermore, CA, UCID-20122, Vols. 1 and 2, 1984.

4. D.P. Jhaveri, R.M. Czarnecki, R.P. Kassawara, and A. Singh, Seismic Demand Evaluation Based on Actual Earthquake Records. Presented at Current Issues Related to Nuclear Power Structures, Equipment and Piping, Orlando, FL, December 1988.

5. Personal communication, D.P. Jhaveri to J.J. Johnson, January 18, 1989.

Table H-1

Control Building Fixed Base Modal Characteristics

Effective Modal Masses

x, y, z = fraction of static mass

Mode No	Freq (Hz)	Dampg Ratio	x	y	z
1	0.70	0.070	0.000	2.298	0.000
2	2.00	0.070	0.000	3.972	0.000
3	2.24	0.070	6.727	0.000	0.000
4	2.58	0.070	0.000	0.000	0.000
5	3.45	0.070	0.000	0.488	0.000
6	6.78	0.070	0.018	0.000	0.000
7	7.98	0.070	7.274	0.001	0.001
8	9.35	0.070	68.675	0.094	0.009
9	10.75	0.070	0.110	83.362	0.209
10	12.81	0.070	0.005	0.116	13.535
11	14.30	0.070	1.356	0.164	0.000
12	18.26	0.070	8.205	0.000	0.002
13	22.06	0.070	0.026	1.657	55.214
14	23.86	0.070	0.000	0.000	0.002
15	26.58	0.070	0.890	2.334	4.682
16	27.09	0.070	0.743	2.809	10.460
17	31.72	0.070	4.723	0.012	0.164
18	32.09	0.070	0.090	0.531	5.063
19	33.18	0.070	0.250	0.130	0.049
20	38.88	0.070	0.005	1.688	0.130
21	39.26	0.070	0.067	0.023	0.071
22	41.22	0.070	0.604	0.018	0.003
23	46.44	0.070	0.000	0.009	3.025
24	47.95	0.070	0.201	0.001	0.006
25	49.16	0.070	0.000	0.000	0.355
Total Pct Mass			99.971	99.707	92.977

Table H-2

Control Building Fixed Base Modal Characteristics

x (East-West) Model

Effective Modal Masses

Mode No	Freq (Hz)	Dampg Ratio	x	y	z
1	2.86	0.100	5.997	0.000	0.000
2	6.27	0.100	0.585	0.000	0.000
3	6.99	0.070	73.891	0.000	0.000
4	8.42	0.070	0.769	0.000	0.000
5	15.64	0.070	0.913	0.000	0.000
6	19.45	0.070	9.376	0.000	0.000
7	21.85	0.070	0.636	0.000	0.000
8	27.09	0.070	1.828	0.000	0.000
9	34.04	0.070	0.924	0.000	0.000
Total Pct Mass			94.920	0.000	0.000

y (North-South) Model

Mode No	Freq (Hz)	Dampg Ratio	x	y	z
1	1.48	0.070	0.000	0.342	0.000
2	3.26	0.100	0.000	5.981	0.000
3	6.87	0.070	0.000	64.572	0.000
4	7.47	0.070	0.000	10.211	0.000
5	8.43	0.070	0.000	0.858	0.000
6	15.64	0.070	0.000	0.791	0.000
7	21.36	0.070	0.000	9.815	0.000
8	22.01	0.070	0.000	0.444	0.000
9	31.42	0.070	0.000	1.813	0.000
10	34.07	0.070	0.000	0.655	0.000
Total Pct Mass			0.000	95.481	0.000

z (Vertical) Model

Mode No	Freq (Hz)	Dampg Ratio	x	y	z
1	3.94	0.070	0.000	0.000	0.246
2	14.67	0.070	0.000	0.000	78.684
3	22.60	0.070	0.000	0.000	3.341
4	23.67	0.070	0.000	0.000	1.605
5	31.24	0.070	0.000	0.000	0.001
6	36.65	0.070	0.000	0.000	0.680
Total Pct Mass			0.000	0.000	84.557

Table H-3

Diesel Generator Building Fixed Base Modal Characteristics

x (East-West) Model

Effective Modal Masses and Heights

x, y, z = fraction of static mass

Mode No	Freq (Hz)	Dampg Ratio	x	y	z
1	27.34	0.070	99.737	0.000	0.000

y (North-South) Model

Mode No	Freq (Hz)	Dampg Ratio	x	y	z
1	31.43	0.070	0.000	99.840	0.000

z (Vertical) Model

Mode No	Freq (Hz)	Dampg Ratio	x	y	z
1	66.02	0.070	0.000	0.000	99.944

Table H-4

River Intake Structure Fixed Base Modal Characteristics

Effective Modal Masses and Heights

x, y, z = fraction of static mass

Mode No	Freq (Hz)	Dampg Ratio	x	y	z
1	10.20	0.070	0.000	52.920	0.000
2	10.61	0.070	0.000	24.504	0.000
3	17.16	0.070	80.549	0.000	0.000
4	27.44	0.070	0.000	18.006	0.000
5	31.21	0.070	0.000	0.544	0.000
6	35.65	0.070	0.068	0.000	88.623
7	38.93	0.070	13.507	0.000	0.624
8	42.32	0.070	0.000	0.025	0.000
9	46.76	0.070	0.000	0.458	0.000
Total Pct Mass			94.124	96.458	89.247

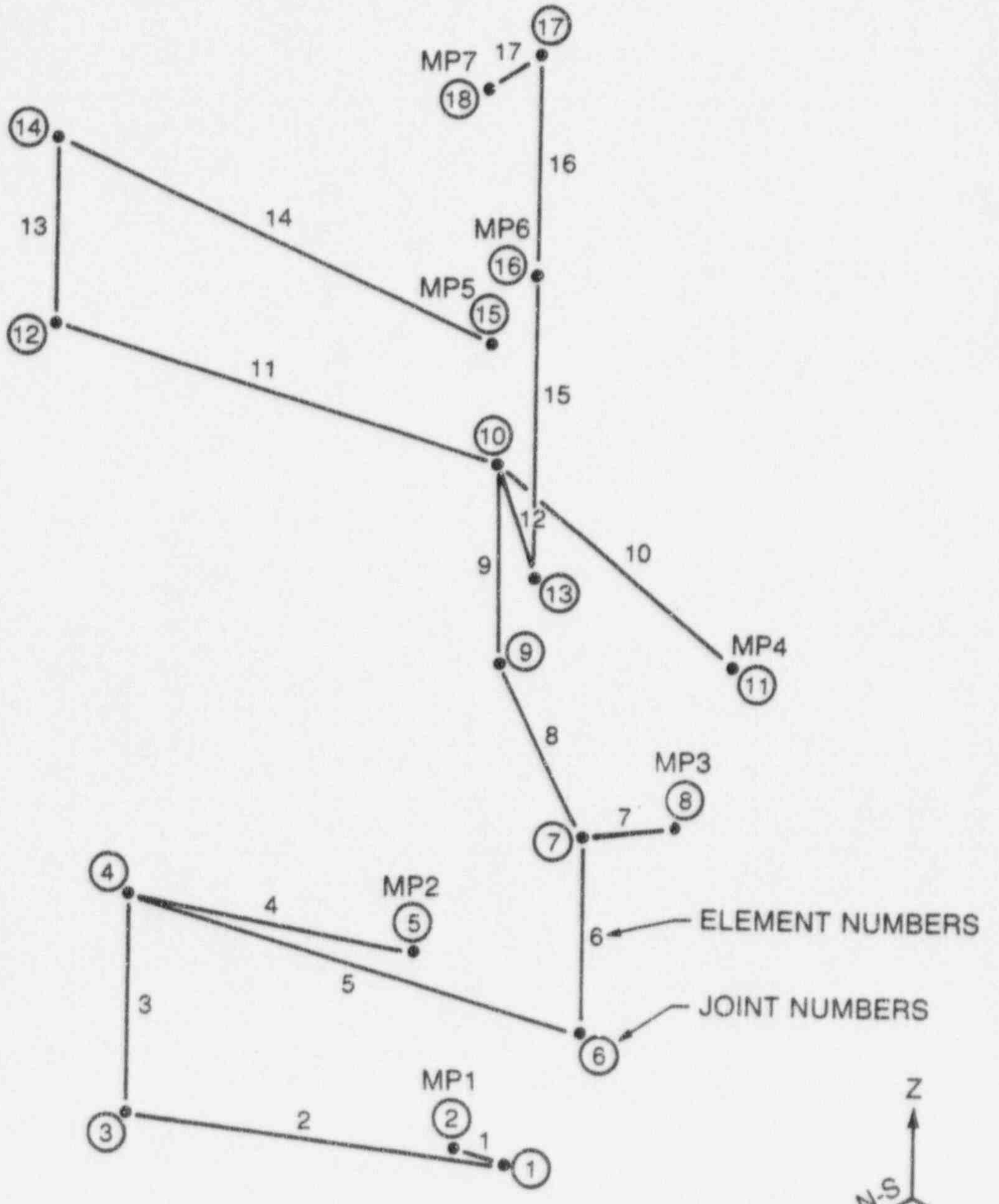


Figure H-1 Control Building Stick Model

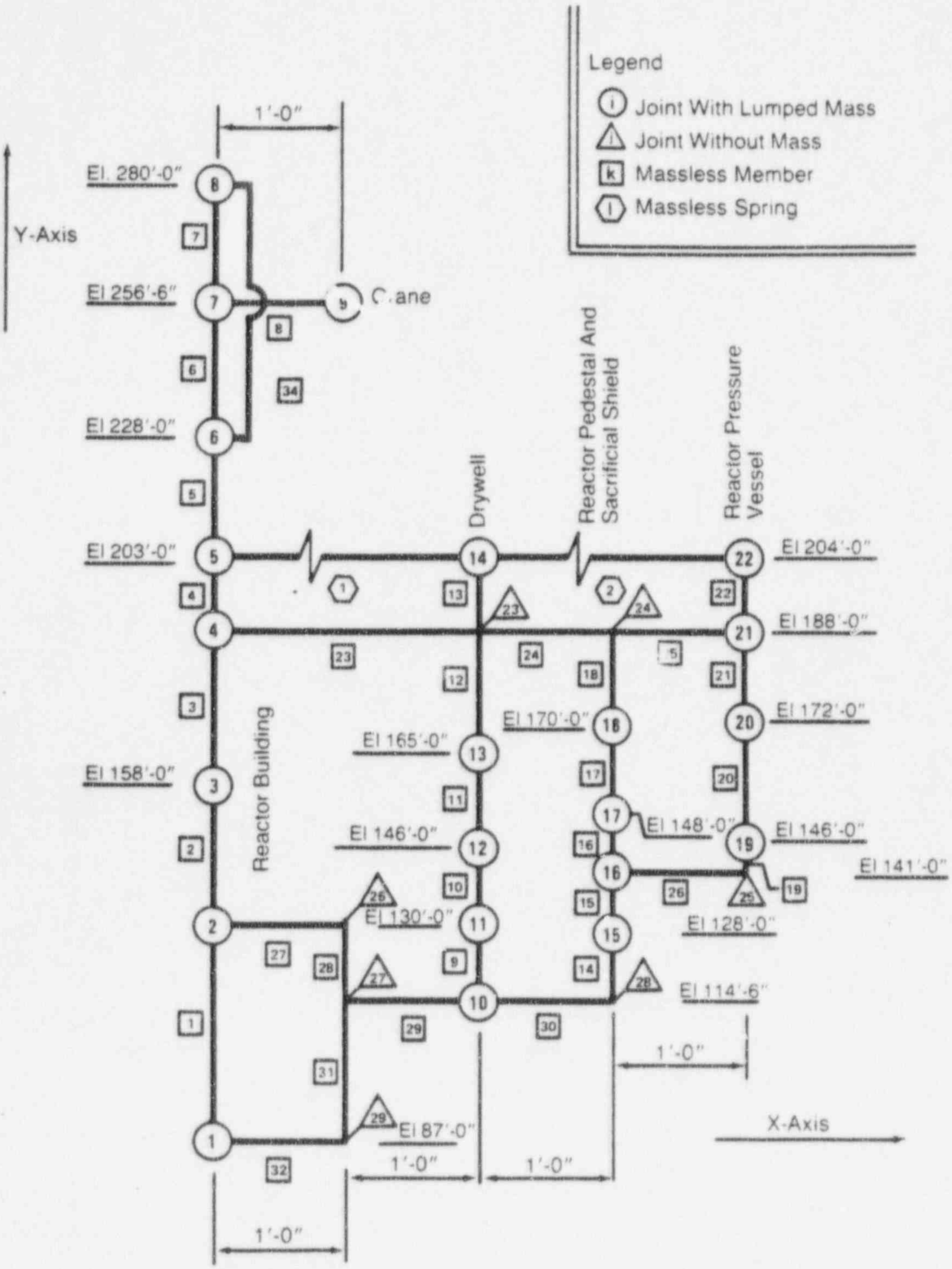


Figure H-2 Unit 1 Reactor Building Seismic Model, N-S and E-W Roof Connected to Vestibule

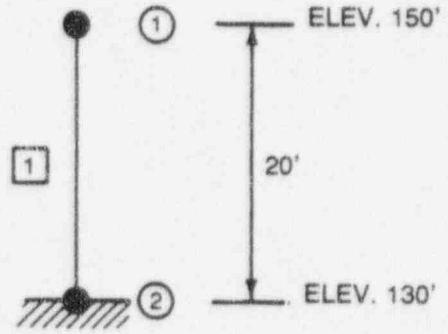


Figure H-3 Diesel Generator Building Unit 2 Model

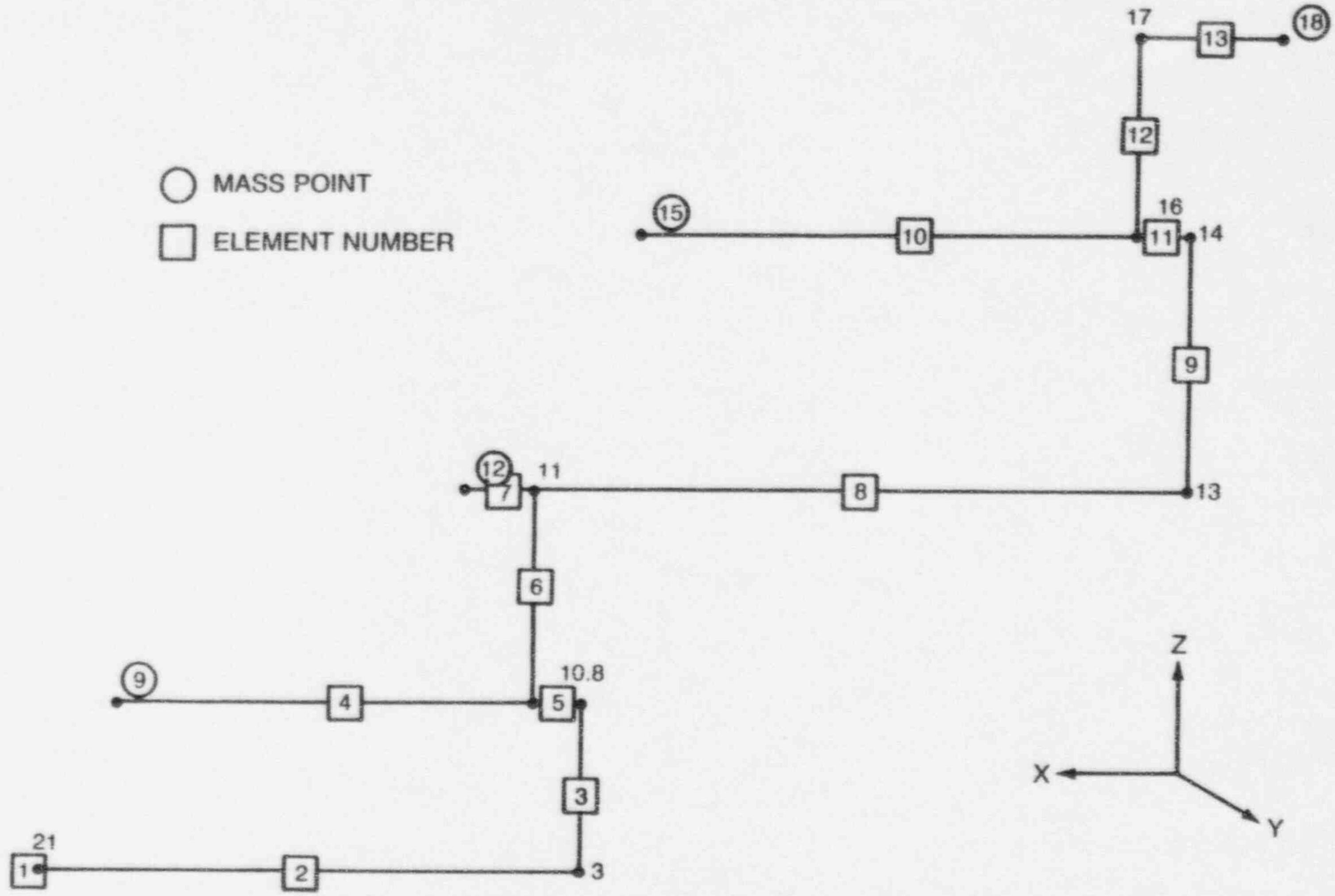


Figure H-4 Isometric Sketch of Intake Structure Dynamic Model
(Not To Scale)

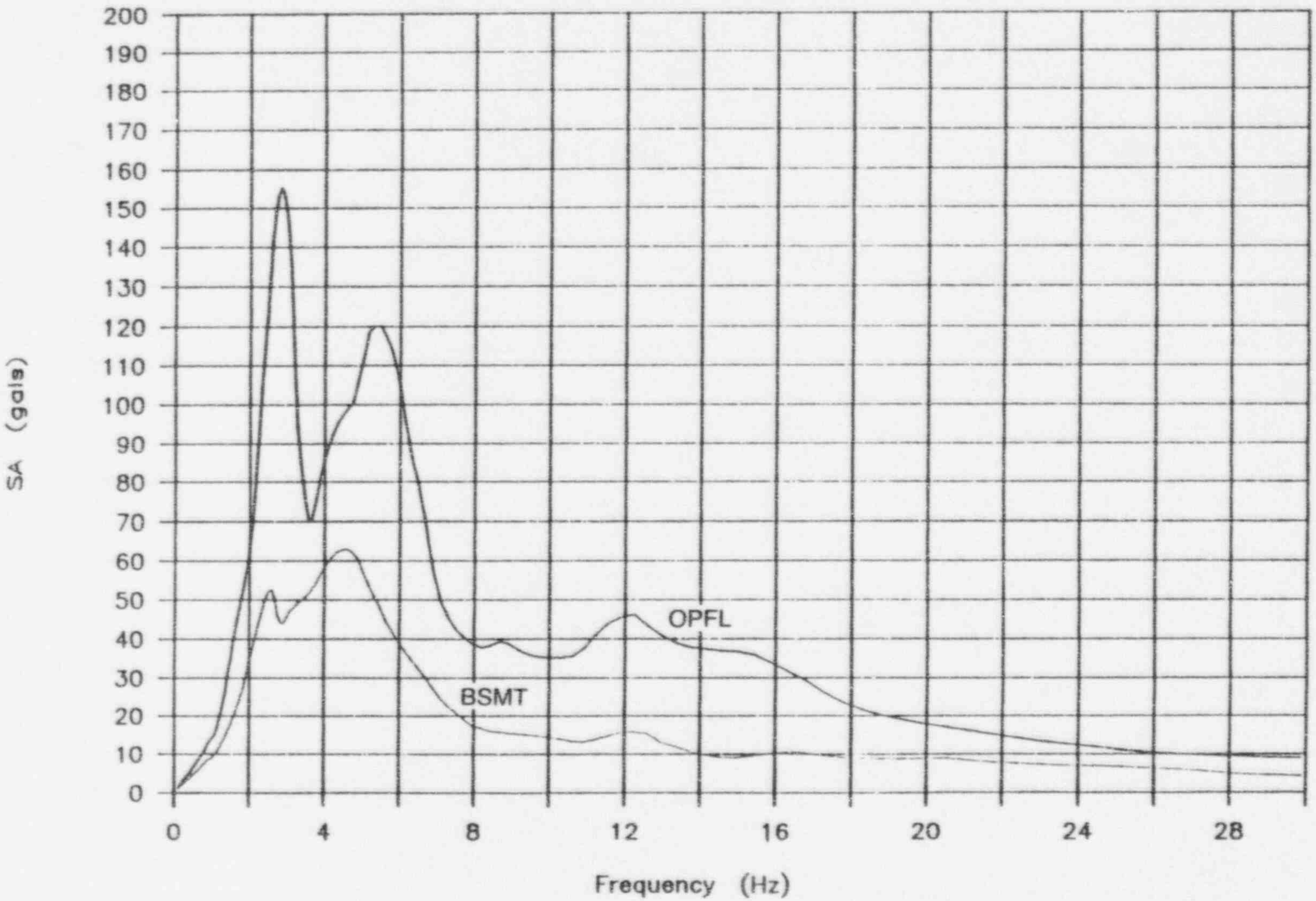


Figure H-5 BWR4 Reactor Building Recorded Spectra EQ8

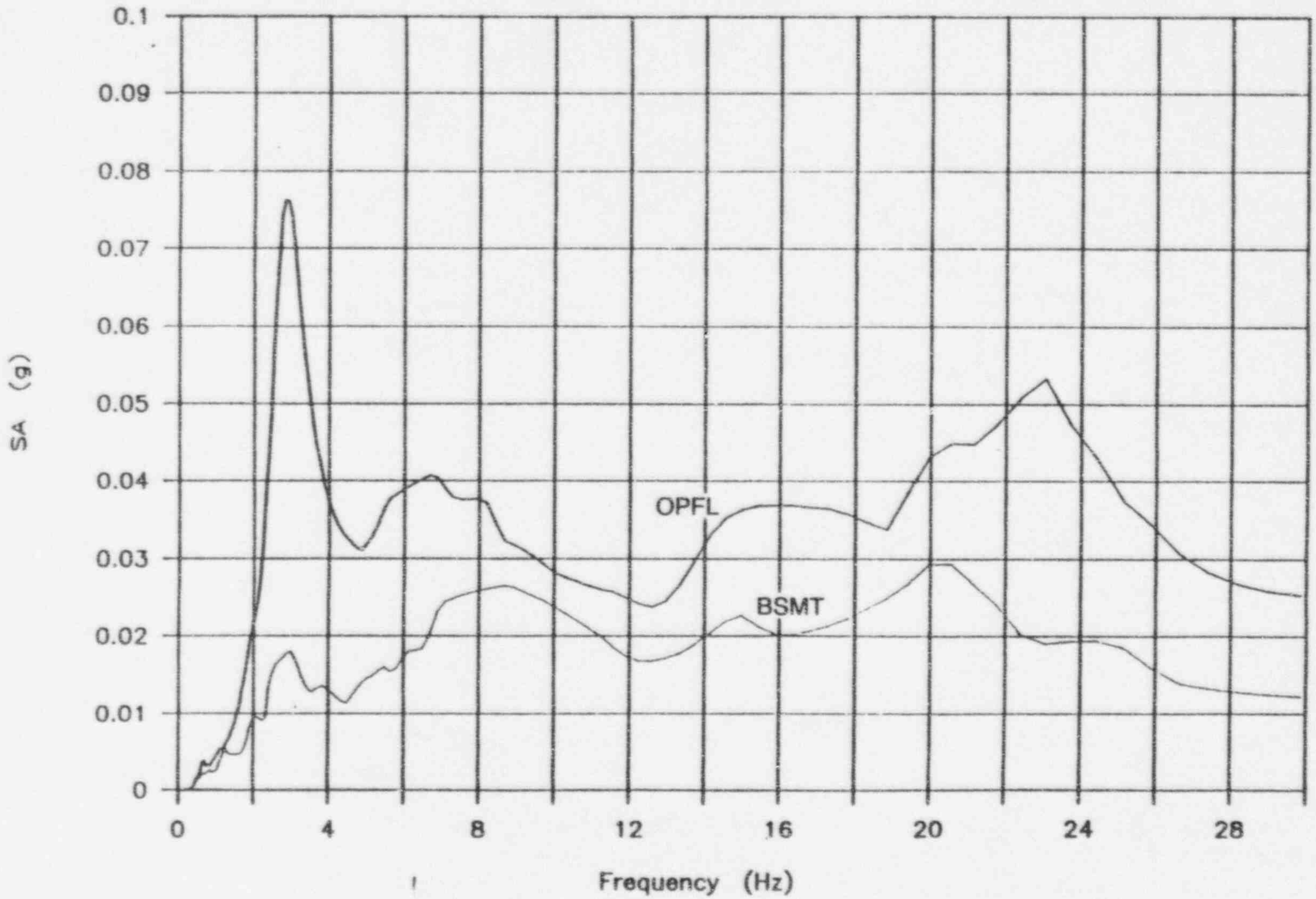


Figure H-6 PWR1 Int. Cont. Recorded EQ Spectra

PLANT HATCH UNIT 1

REACTOR BUILDING

SUMMARY OF MAXIMUM ACCELERATIONS AND DISPLACEMENTS

AND

5% DAMPED SME IN-STRUCTURE RESPONSE SPECTRA [1]

(ROOF CONNECTED TO VESTIBULE)

[1] Spectra are raw curves that have not been broadened.

REACTOR BUILDING
SUMMARY OF MAXIMUM ABSOLUTE ACCELERATIONS (g) (Page 1 of 3)

	Lower Bound Soil Properties	Intermediate Soil Properties	Upper Bound Soil Properties
Node 1, Elevation 87.0			
E-W dir	0.2563	0.1978	0.2088
N-S dir	0.1907	0.2265	0.2877
Vertical	0.1698	0.2028	0.2205
Node 2, Elevation 130.0			
E-W dir	0.1446	0.1446	0.1817
N-S dir	0.1469	0.1615	0.2096
Vertical	0.1712	0.2031	0.2228
Node 3, Elevation 158.0			
E-W dir	0.1193	0.1462	0.1774
N-S dir	0.1437	0.1759	0.1839
Vertical	0.1743	0.2032	0.2257
Node 4, Elevation 188.0			
E-W dir	0.1609	0.1755	0.2137
N-S dir	0.1583	0.2221	0.2533
Vertical	0.1766	0.2035	0.2278
Node 5, Elevation 203.0			
E-W dir	0.1967	0.1932	0.2430
N-S dir	0.1800	0.2472	0.2901
Vertical	0.1775	0.2040	0.2286
Node 6, Elevation 228.0			
E-W dir	0.2824	0.2534	0.2949
N-S dir	0.2279	0.2872	0.3490
Vertical	0.1783	0.2044	0.2294
Node 7, Elevation 256.5			
E-W dir	0.8766	0.7739	0.6440
N-S dir	0.5943	0.6497	0.5840
Vertical	0.1818	0.2076	0.2372
Node 8, Elevation 280.0			
E-W dir	0.8413	0.6356	0.5405
N-S dir	0.8477	0.9287	0.9388
Vertical	0.1838	0.2123	0.2415
Node 9, Elevation 256.5			
E-W dir	0.8766	0.7739	0.6440
N-S dir	0.5920	0.9128	1.2110
Vertical	0.2224	0.3231	0.5584

REACTOR BUILDING
SUMMARY OF MAXIMUM ABSOLUTE ACCELERATIONS (g) (Page 2 of 3)

	Lower Bound Soil Properties	Intermediate Soil Properties	Upper Bound Soil Properties
Node 10, Elevation 114.5			
E-W dir	0.1734	0.1553	0.1909
N-S dir	0.1558	0.1854	0.2404
Vertical	0.1704	0.2030	0.2214
Node 11, Elevation 130.0			
E-W dir	0.1455	0.1434	0.1804
N-S dir	0.1447	0.1606	0.2120
Vertical	0.1706	0.2033	0.2218
Node 12, Elevation 146.0			
E-W dir	0.1264	0.1421	0.1735
N-S dir	0.1429	0.1556	0.1844
Vertical	0.1708	0.2038	0.2221
Node 13, Elevation 165.0			
E-W dir	0.1235	0.1482	0.1773
N-S dir	0.1417	0.1806	0.1917
Vertical	0.1710	0.2043	0.2226
Node 14, Elevation 204.0			
E-W dir	0.1983	0.1935	0.2405
N-S dir	0.1824	0.2474	0.2923
Vertical	0.1713	0.2053	0.2235
Node 15, Elevation 128.0			
E-W dir	0.1503	0.1474	0.1847
N-S dir	0.1459	0.1678	0.2203
Vertical	0.1707	0.2044	0.2227
Node 16, Elevation 141.0			
E-W dir	0.1377	0.1465	0.1796
N-S dir	0.1464	0.1526	0.2002
Vertical	0.1713	0.2056	0.2242
Node 17, Elevation 148.0			
E-W dir	0.1325	0.1498	0.1775
N-S dir	0.1494	0.1633	0.1940
Vertical	0.1720	0.2068	0.2253
Node 18, Elevation 170.0			
E-W dir	0.1377	0.1527	0.1937
N-S dir	0.1534	0.1997	0.2122
Vertical	0.1737	0.2106	0.2280

REACTOR BUILDING
SUMMARY OF MAXIMUM ABSOLUTE ACCELERATIONS (g) (Page 3 of 3)

	Lower Bound Soil Properties	Intermediate Soil Properties	Upper Bound Soil Properties
Node 19, Elevation 146.0			
E-W dir	0.1357	0.1478	0.1789
N-S dir	0.1501	0.1610	0.1904
Vertical	0.1723	0.2075	0.2258
Node 20, Elevation 172.0			
E-W dir	0.1339	0.1571	0.1843
N-S dir	0.1499	0.1994	0.2343
Vertical	0.1736	0.2103	0.2279
Node 21, Elevation 188.0			
E-W dir	0.1648	0.1807	0.2130
N-S dir	0.1666	0.2294	0.2787
Vertical	0.1739	0.2111	0.2285
Node 22, Elevation 204.0			
E-W dir	0.2264	0.2088	0.2702
N-S dir	0.1977	0.2664	0.3252
Vertical	0.1740	0.2113	0.2287

REACTOR BUILDING

SUMMARY OF MAXIMUM RELATIVE DISPLACEMENTS (in.) (Page 1 of 3)

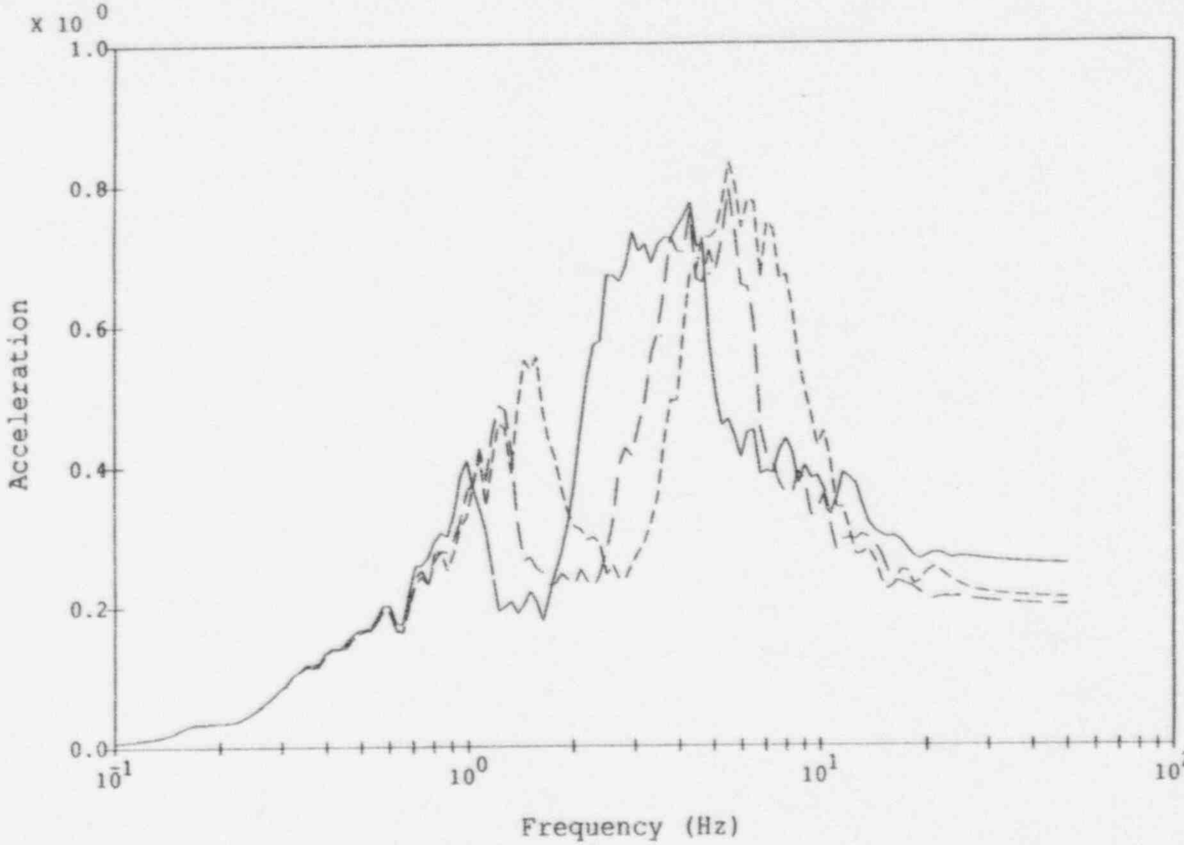
	Lower Bound Soil Properties	Intermediate Soil Properties	Upper Bound Soil Properties
Node 1, Elevation 87.0			
E-W dir	0.0000	0.0000	0.0000
N-S dir	0.0000	0.0000	0.0000
Vertical	0.0000	0.0000	0.0000
Node 2, Elevation 130.0			
E-W dir	0.0141	0.0155	0.0184
N-S dir	0.0144	0.0200	0.0202
Vertical	0.0029	0.0034	0.0038
Node 3, Elevation 158.0			
E-W dir	0.0257	0.0274	0.0326
N-S dir	0.0303	0.0401	0.0412
Vertical	0.0062	0.0072	0.0080
Node 4, Elevation 188.0			
E-W dir	0.0385	0.0393	0.0474
N-S dir	0.0418	0.0537	0.0556
Vertical	0.0084	0.0098	0.0109
Node 5, Elevation 203.0			
E-W dir	0.0430	0.0429	0.0519
N-S dir	0.0455	0.0577	0.0595
Vertical	0.0093	0.0107	0.0120
Node 6, Elevation 228.0			
E-W dir	0.0524	0.0487	0.0588
N-S dir	0.0505	0.0628	0.0636
Vertical	0.0100	0.0115	0.0129
Node 7, Elevation 256.5			
E-W dir	1.0881	0.8953	0.7322
N-S dir	0.5437	0.6256	0.5421
Vertical	0.0124	0.0144	0.0164
Node 8, Elevation 280.0			
E-W dir	0.9880	0.7863	0.6437
N-S dir	0.8346	0.9417	0.8733
Vertical	0.0135	0.0156	0.0179
Node 9, Elevation 256.5			
E-W dir	1.0881	0.8953	0.7322
N-S dir	2.4672	4.2427	5.3478
Vertical	0.1515	0.2168	0.3662

REACTOR BUILDING
SUMMARY OF MAXIMUM RELATIVE DISPLACEMENTS (in.) (Page 2 of 3)

	Lower Bound Soil Properties	Intermediate Soil Properties	Upper Bound Soil Properties
Node 10, Elevation 114.5			
E-W dir	0.0028	0.0031	0.0037
N-S dir	0.0028	0.0040	0.0040
Vertical	0.0014	0.0017	0.0019
Node 11, Elevation 130.0			
E-W dir	0.0073	0.0080	0.0095
N-S dir	0.0077	0.0106	0.0108
Vertical	0.0018	0.0021	0.0024
Node 12, Elevation 146.0			
E-W dir	0.0140	0.0150	0.0178
N-S dir	0.0150	0.0200	0.0205
Vertical	0.0023	0.0027	0.0030
Node 13, Elevation 165.0			
E-W dir	0.0238	0.0251	0.0299
N-S dir	0.0257	0.0338	0.0348
Vertical	0.0029	0.0034	0.0037
Node 14, Elevation 204.0			
E-W dir	0.0404	0.0417	0.0502
N-S dir	0.0437	0.0563	0.0589
Vertical	0.0038	0.0046	0.0050
Node 15, Elevation 128.0			
E-W dir	0.0059	0.0067	0.0082
N-S dir	0.0064	0.0084	0.0088
Vertical	0.0024	0.0028	0.0031
Node 16, Elevation 141.0			
E-W dir	0.0087	0.0100	0.0122
N-S dir	0.0097	0.0124	0.0131
Vertical	0.0033	0.0039	0.0042
Node 17, Elevation 148.0			
E-W dir	0.0123	0.0139	0.0170
N-S dir	0.0134	0.0175	0.0185
Vertical	0.0039	0.0047	0.0051
Node 18, Elevation 170.0			
E-W dir	0.0262	0.0289	0.0345
N-S dir	0.0273	0.0373	0.0403
Vertical	0.0055	0.0066	0.0072

REACTOR BUILDING
SUMMARY OF MAXIMUM RELATIVE DISPLACEMENTS (in.) (Page 3 of 3)

	Lower Bound Soil Properties	Intermediate Soil Properties	Upper Bound Soil Properties
Node 19, Elevation 146.0			
E-W dir	0.0124	0.0144	0.0173
N-S dir	0.0137	0.0180	0.0195
Vertical	0.0042	0.0050	0.0054
Node 20, Elevation 172.0			
E-W dir	0.0315	0.0357	0.0415
N-S dir	0.0317	0.0456	0.0515
Vertical	0.0053	0.0064	0.0070
Node 21, Elevation 188.0			
E-W dir	0.0446	0.0497	0.0579
N-S dir	0.0451	0.0639	0.0723
Vertical	0.0057	0.0068	0.0074
Node 22, Elevation 204.0			
E-W dir	0.0573	0.0632	0.0740
N-S dir	0.0580	0.0816	0.0925
Vertical	0.0058	0.0069	0.0075



Legend:

Lower Bound ————

Intermediate — — — —

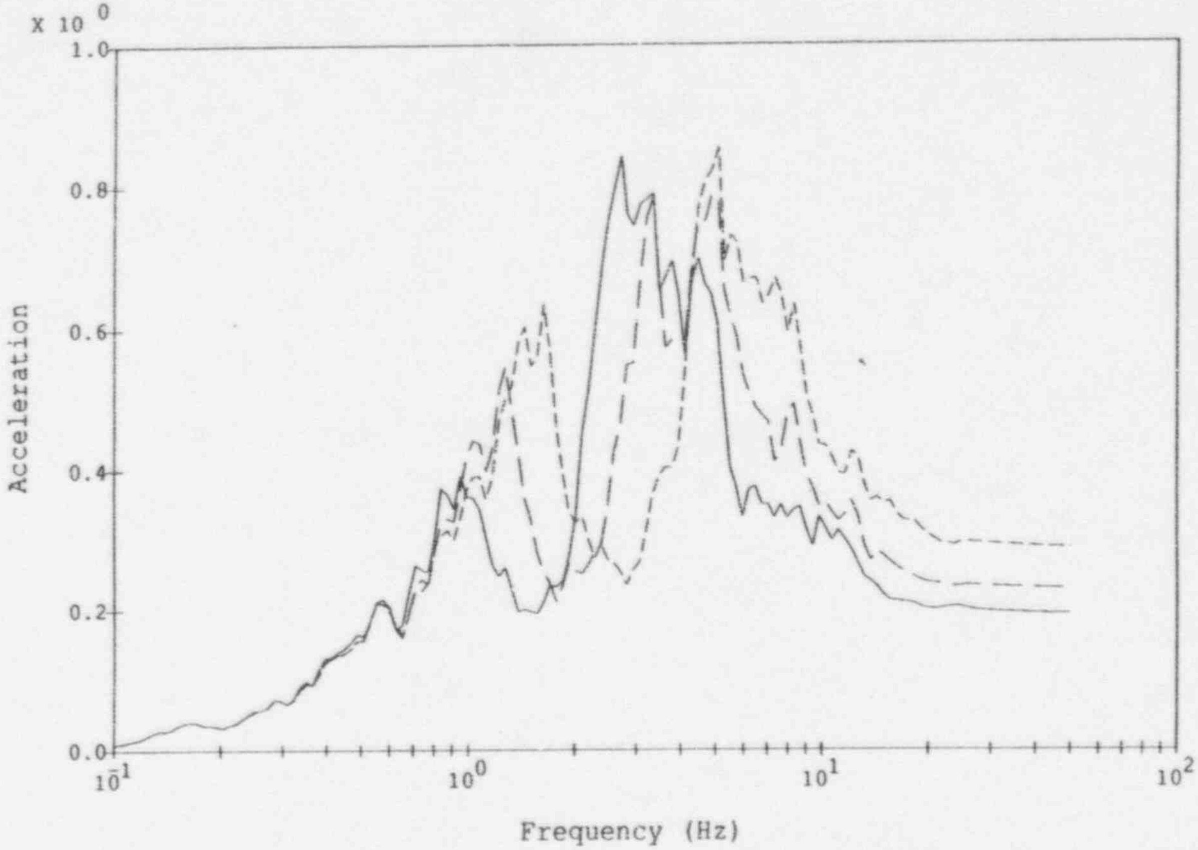
Upper Bound - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 1, Elevation 87.0 ft, East-West Direction



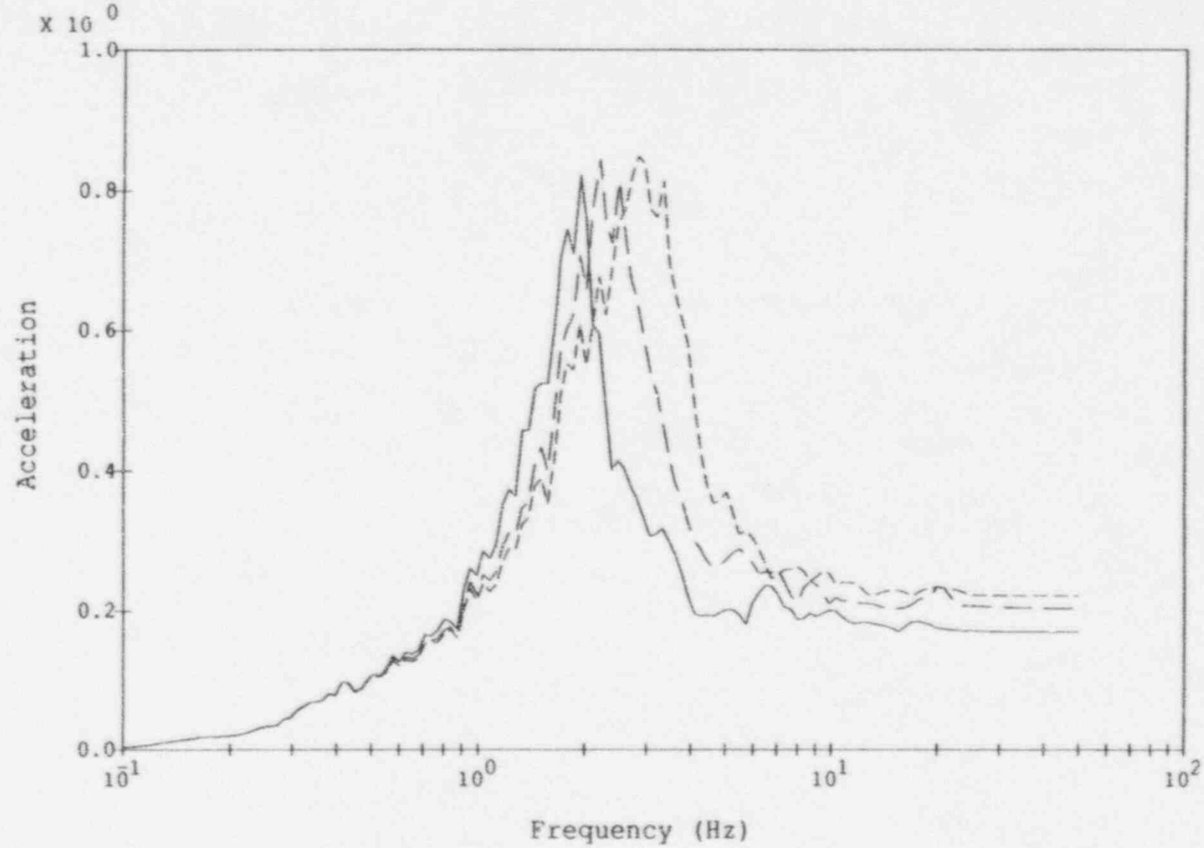
Legend:

Lower Bound
Intermediate
Upper Bound

Notes:

Accelerations in g's
5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 1, Elevation 87.0 ft, North-South Direction



Legend:

Lower Bound

Intermediate

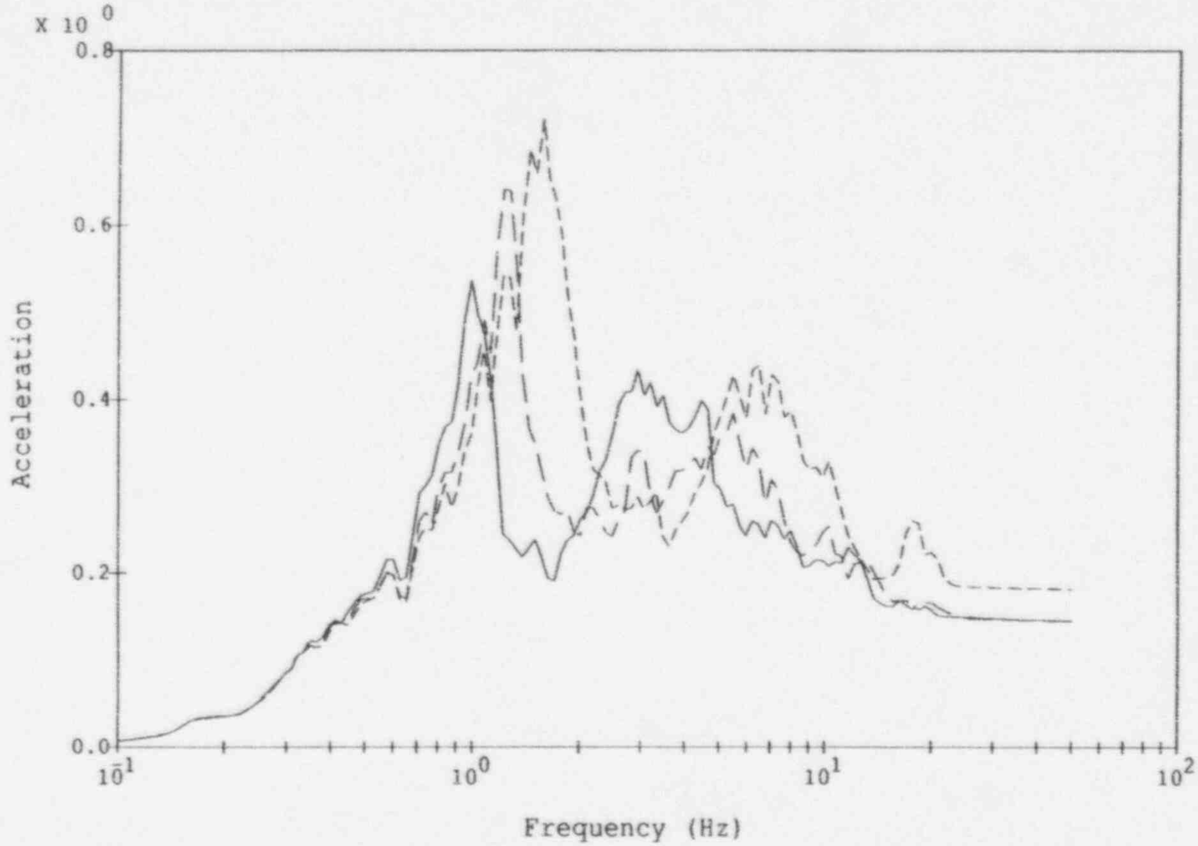
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 1, Elevation 87.0 ft, Vertical Direction



Legend:

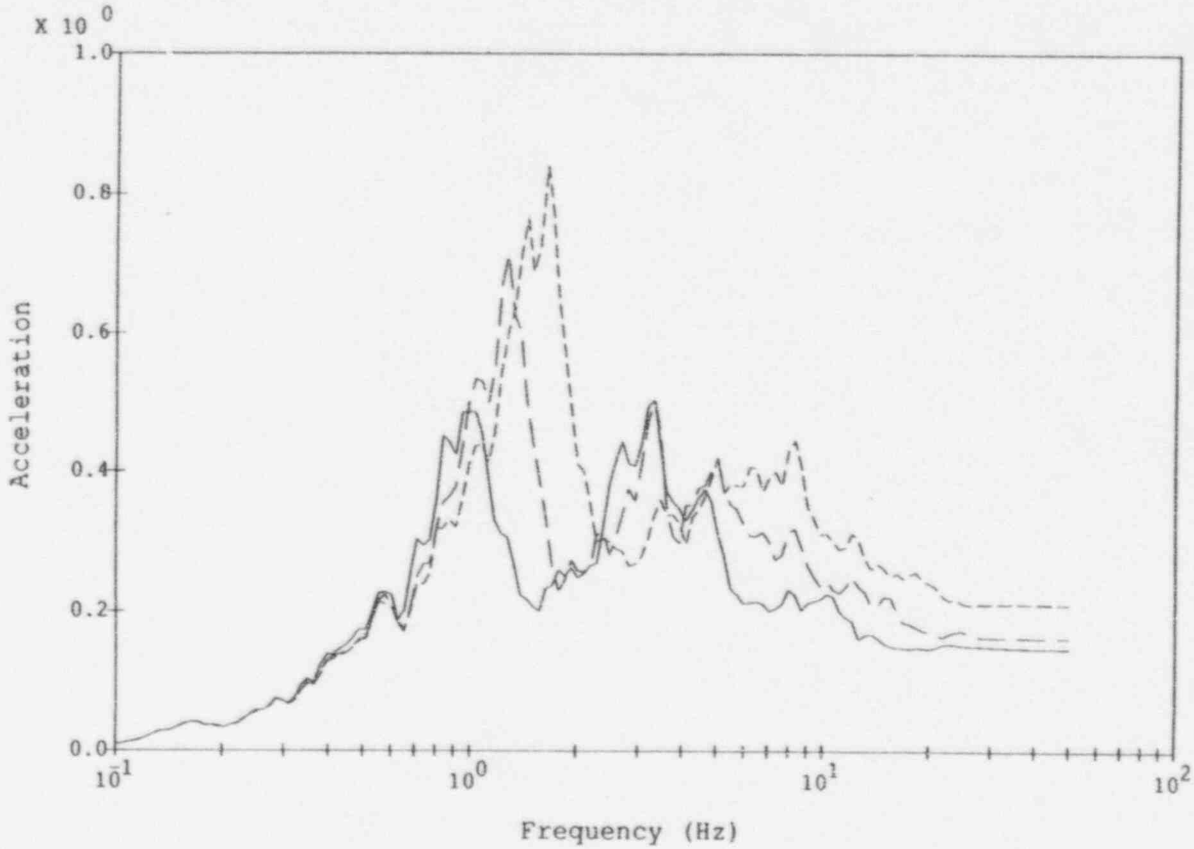
Lower Bound	————————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 2, Elevation 130.0 ft, East-West Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

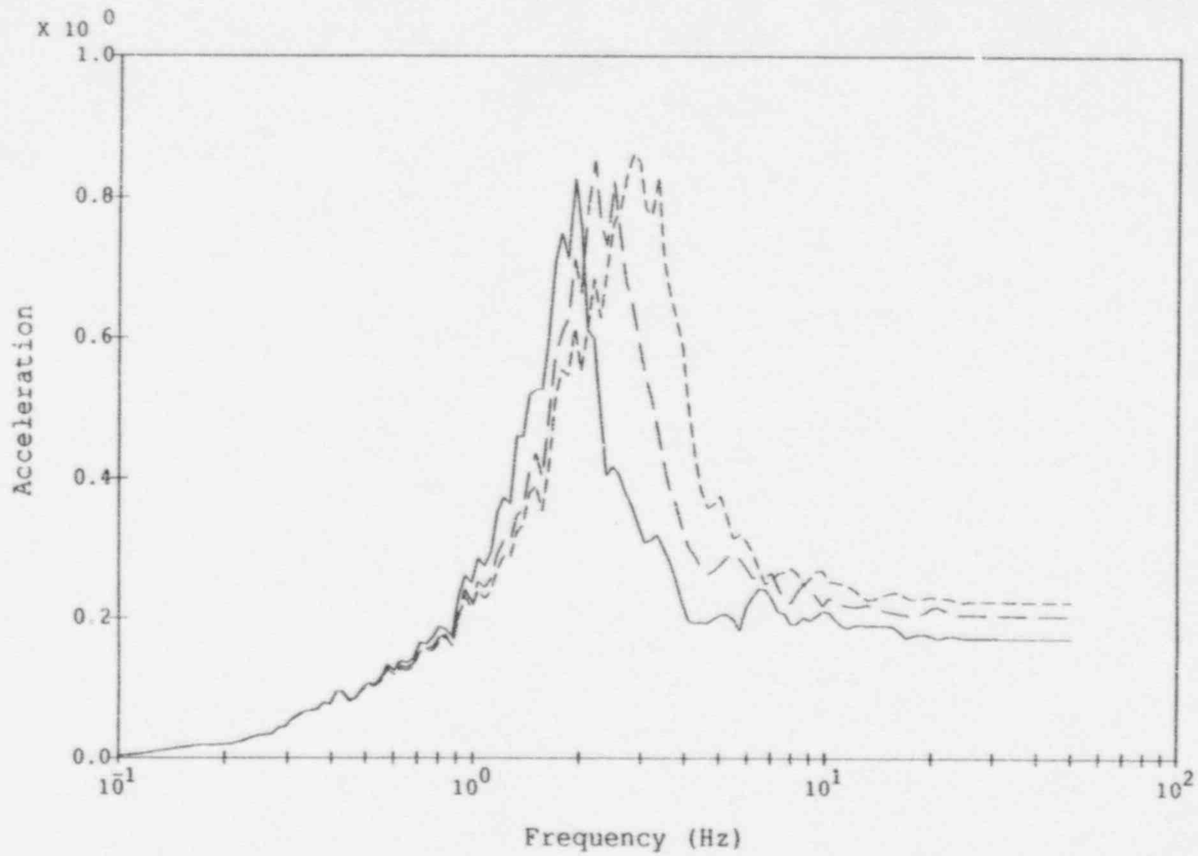
Upper Bound ---------

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 2, Elevation 130.0 ft, North-South Direction



Legend:

Lower Bound ————

Intermediate — — — —

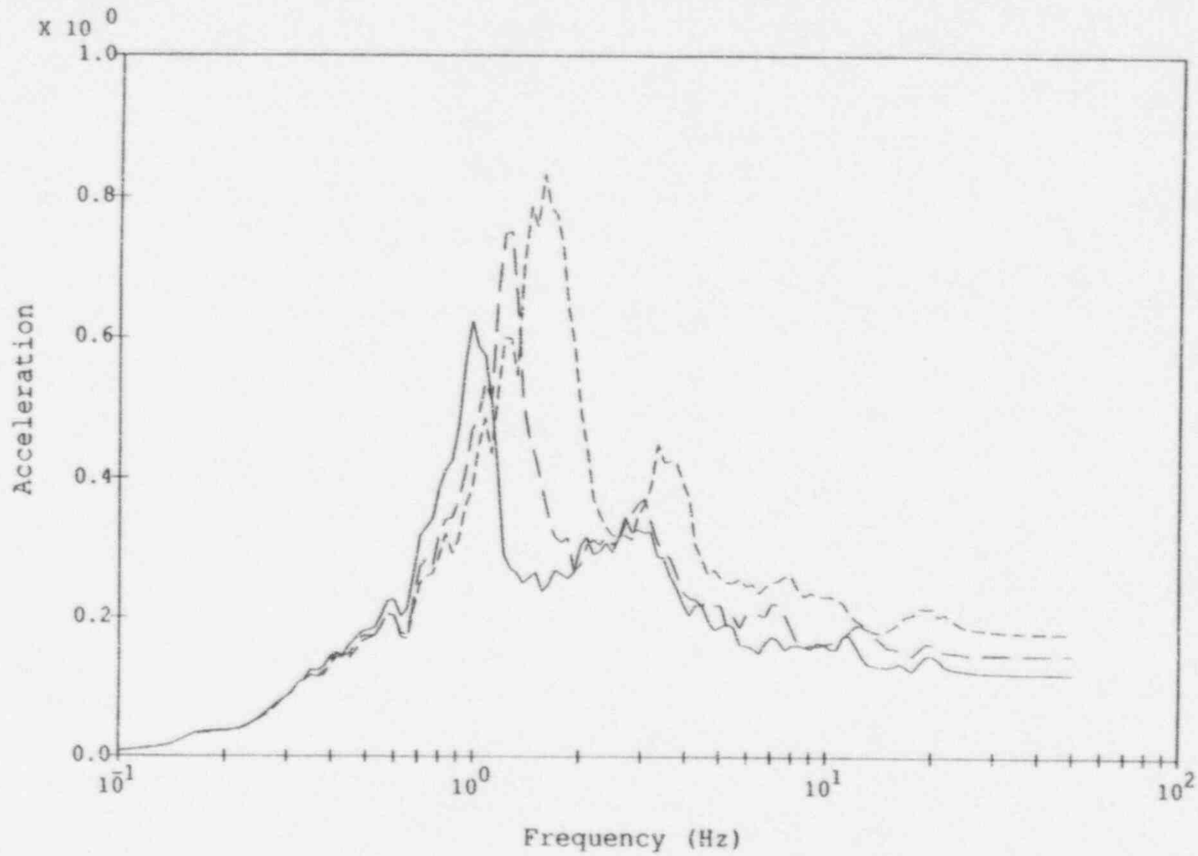
Upper Bound - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 2, Elevation 130.0 ft, Vertical Direction



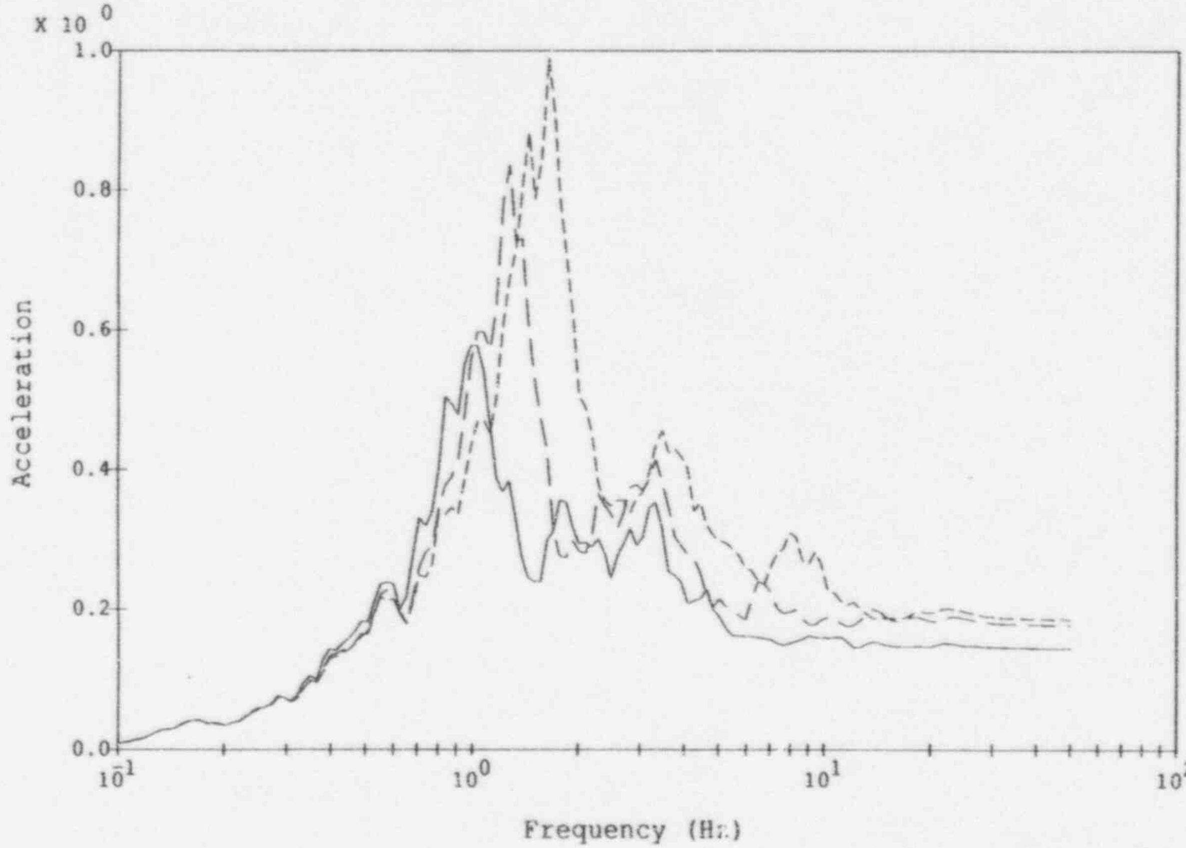
Legend:

Lower Bound	————————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's
5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 3, Elevation 158.0 ft, East-West Direction



Legend:

Lower Bound

Intermediate

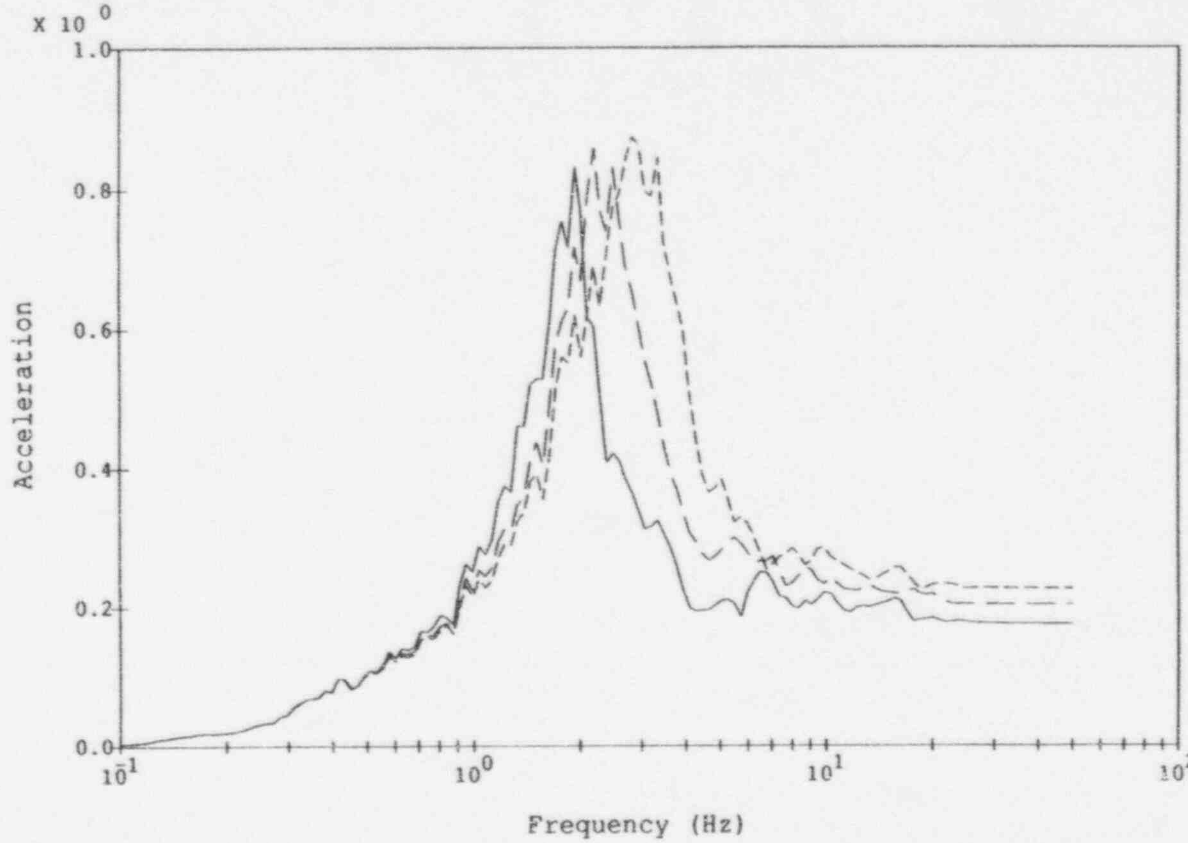
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 3, Elevation 158.0 ft, North-South Direction



Legend:

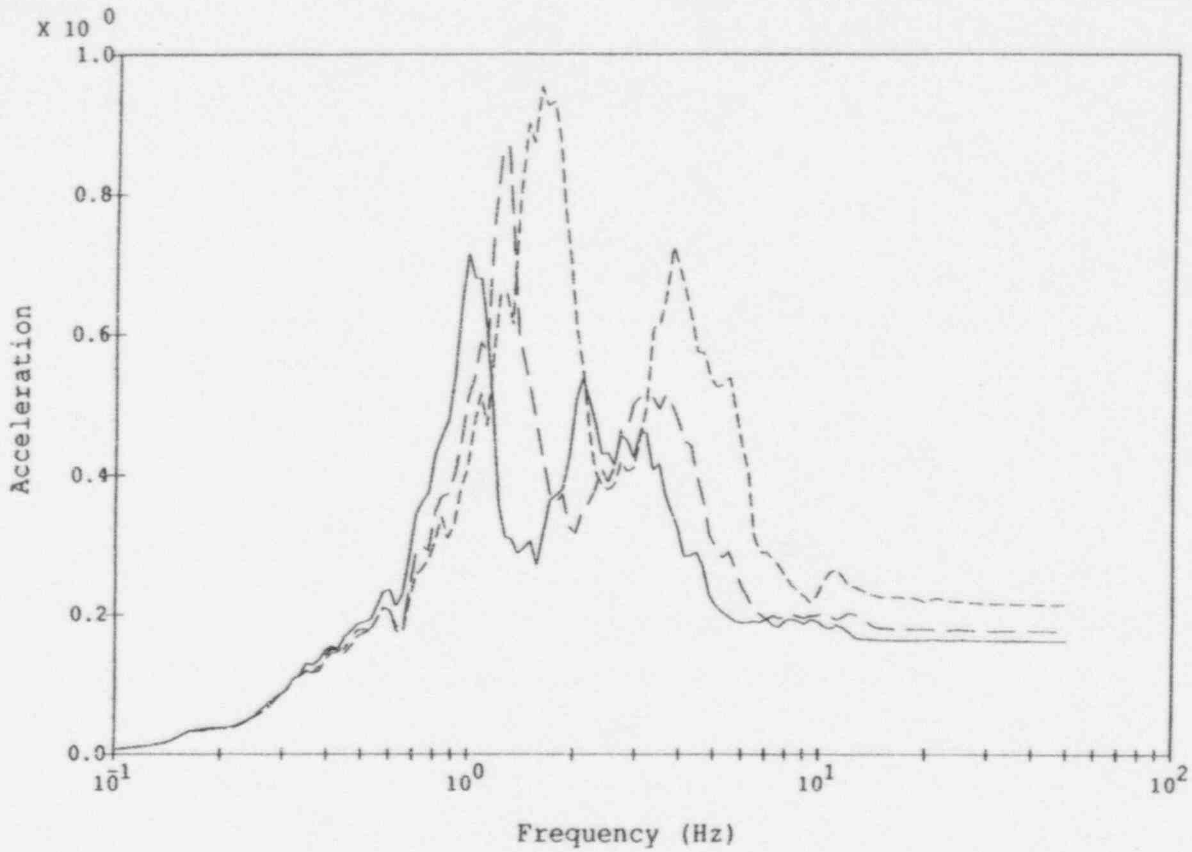
Lower Bound	————————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 3, Elevation 158.0 ft, Vertical Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

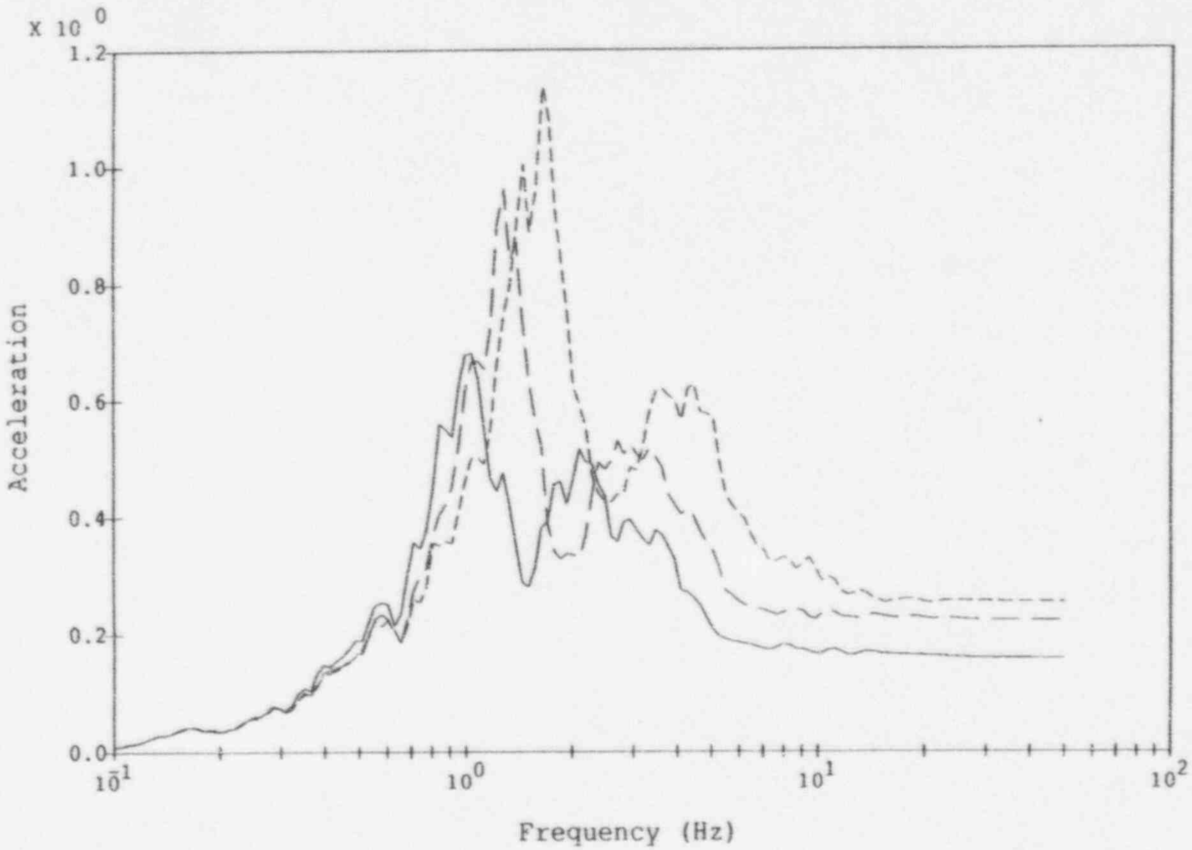
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 4, Elevation 188.0 ft, East-West Direction



Legend:

Lower Bound

Intermediate

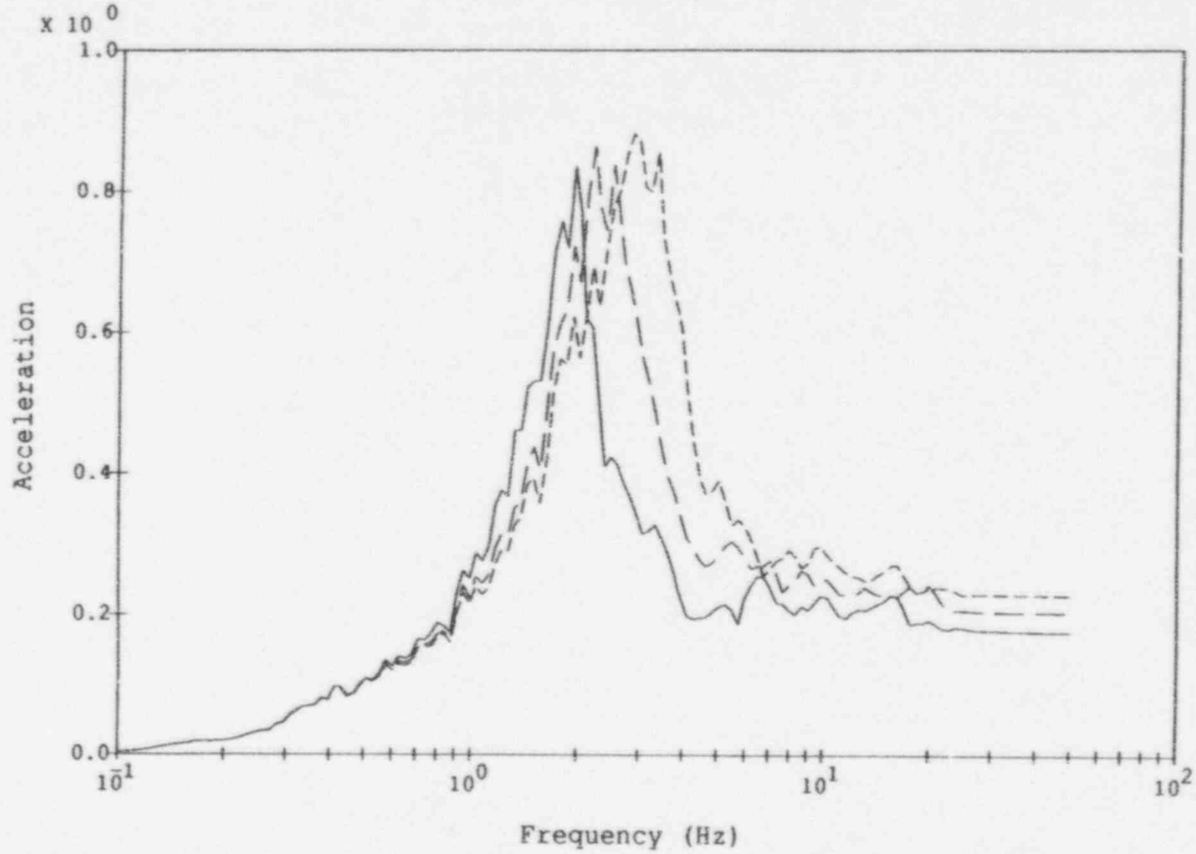
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 4, Elevation 188.0 ft, North-South Direction



Legend:

Lower Bound ————

Intermediate — — — —

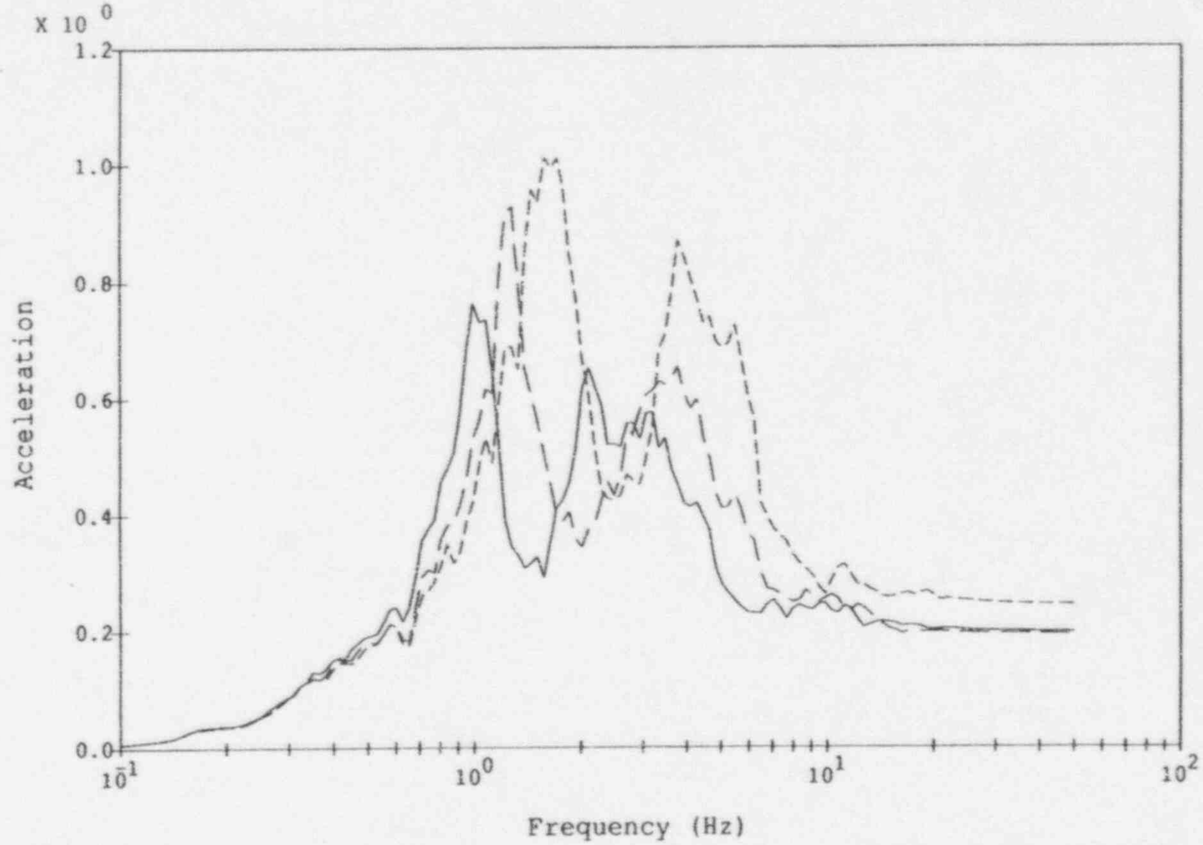
Upper Bound - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 4, Elevation 185.0 ft, Vertical Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

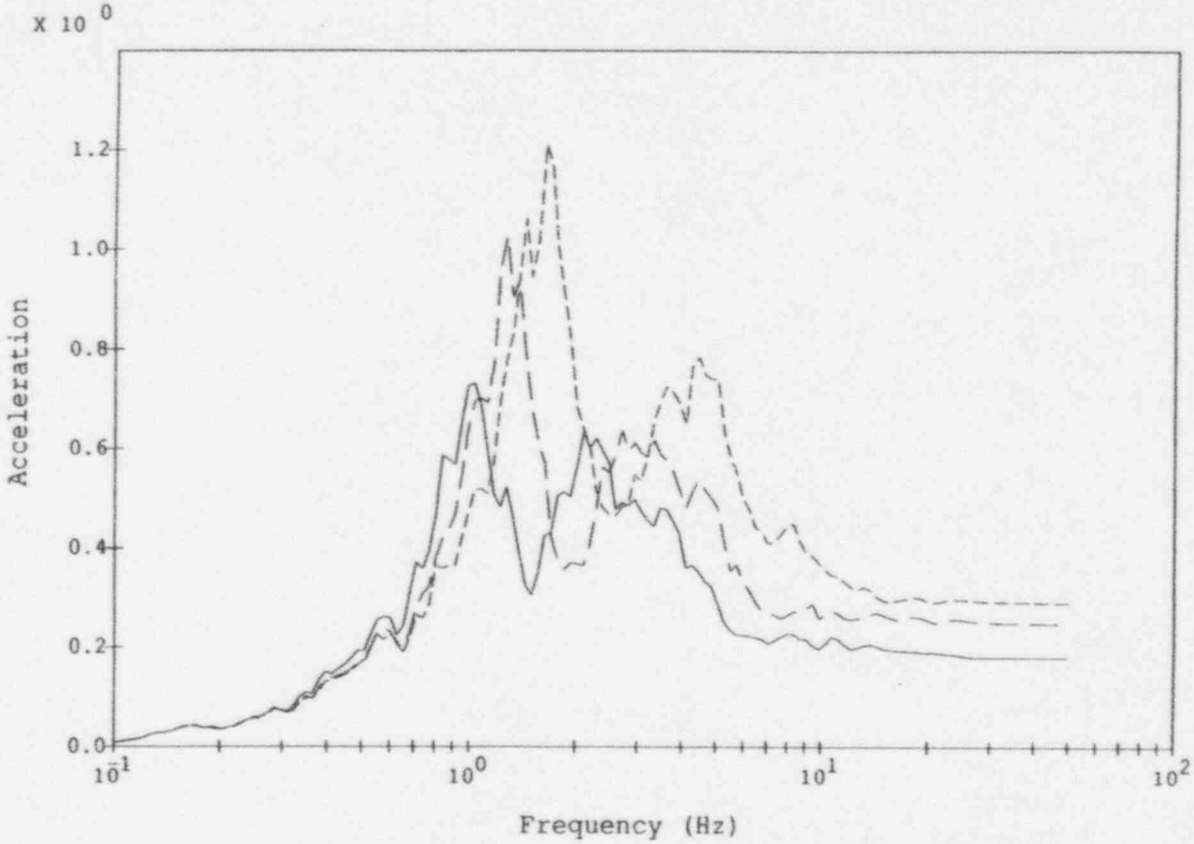
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 5, Elevation 203.0 ft, East-West Direction



Legend:

Lower Bound ————

Intermediate — — — —

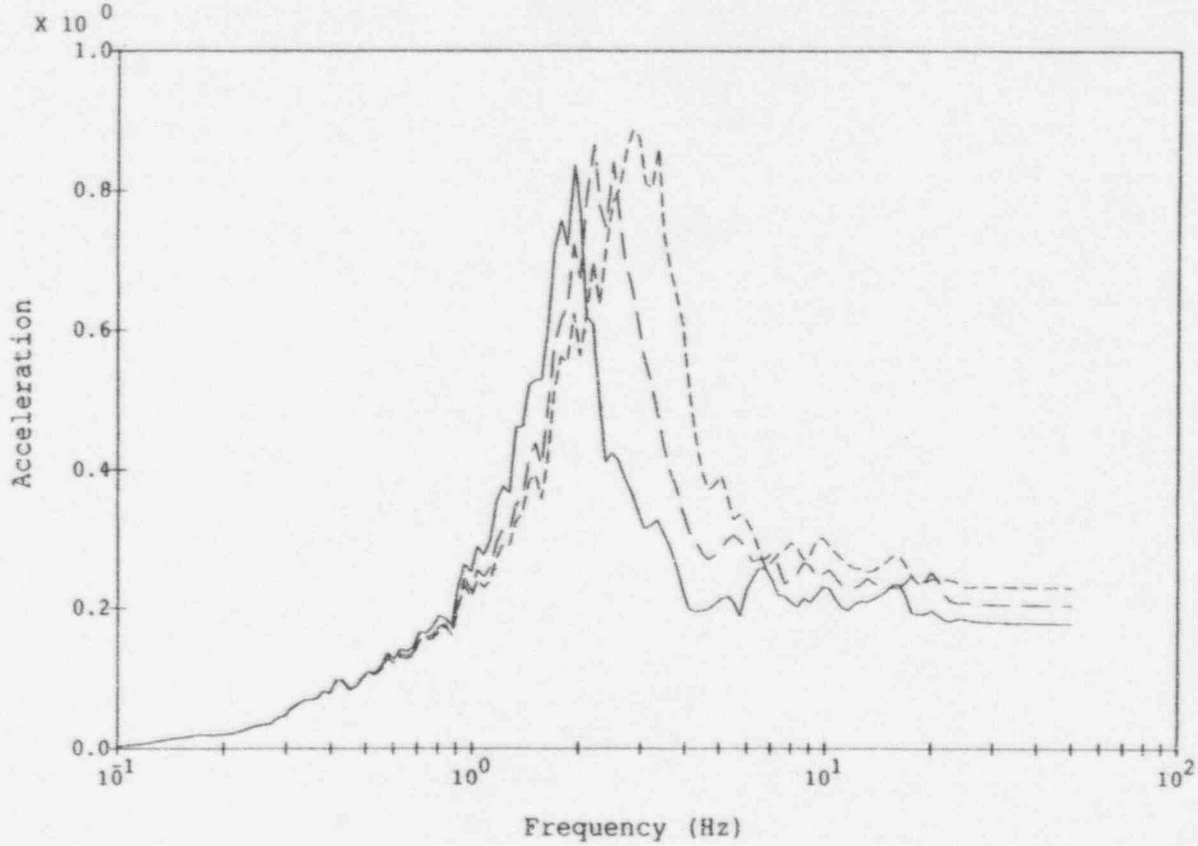
Upper Bound - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 5, Elevation 203.0 ft, North-South Direction



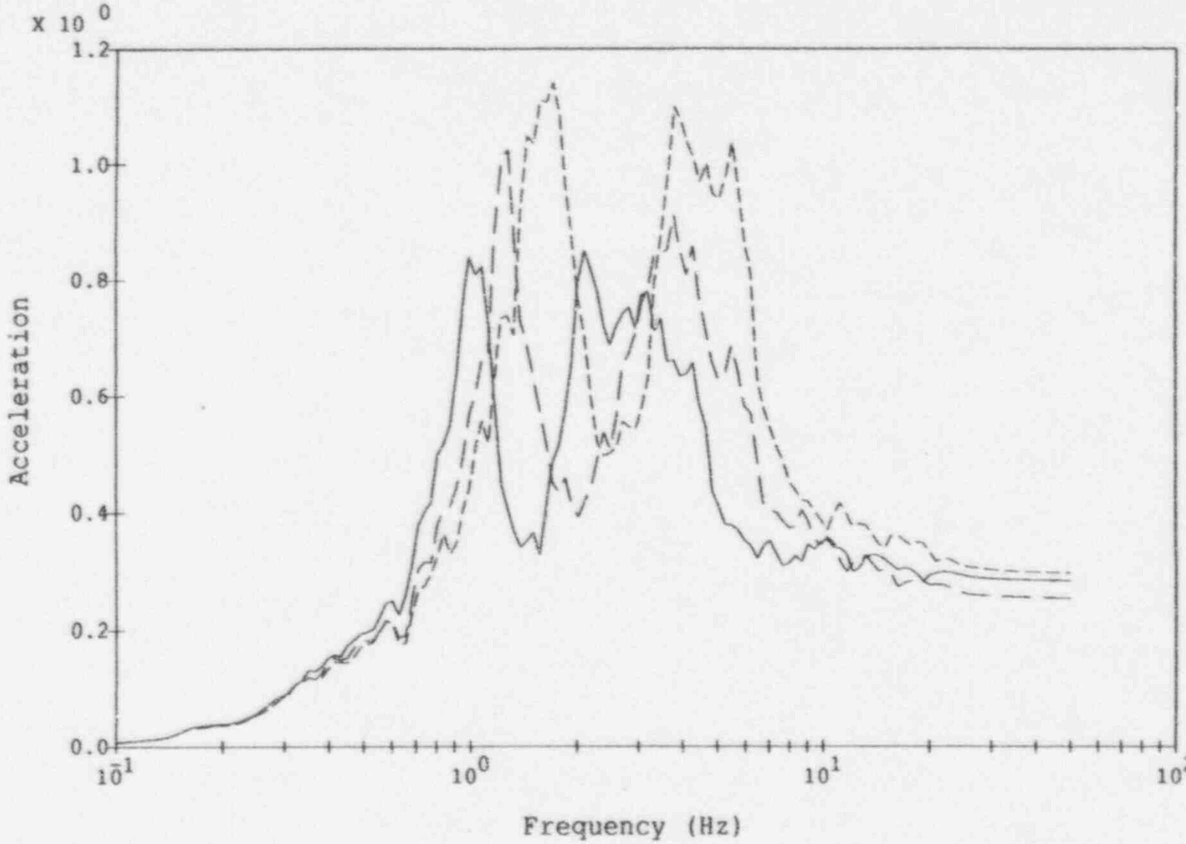
Legend:

Lower Bound ―――――――
Intermediate — — — — —
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's
5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 5, Elevation 203.0 ft, Vertical Direction



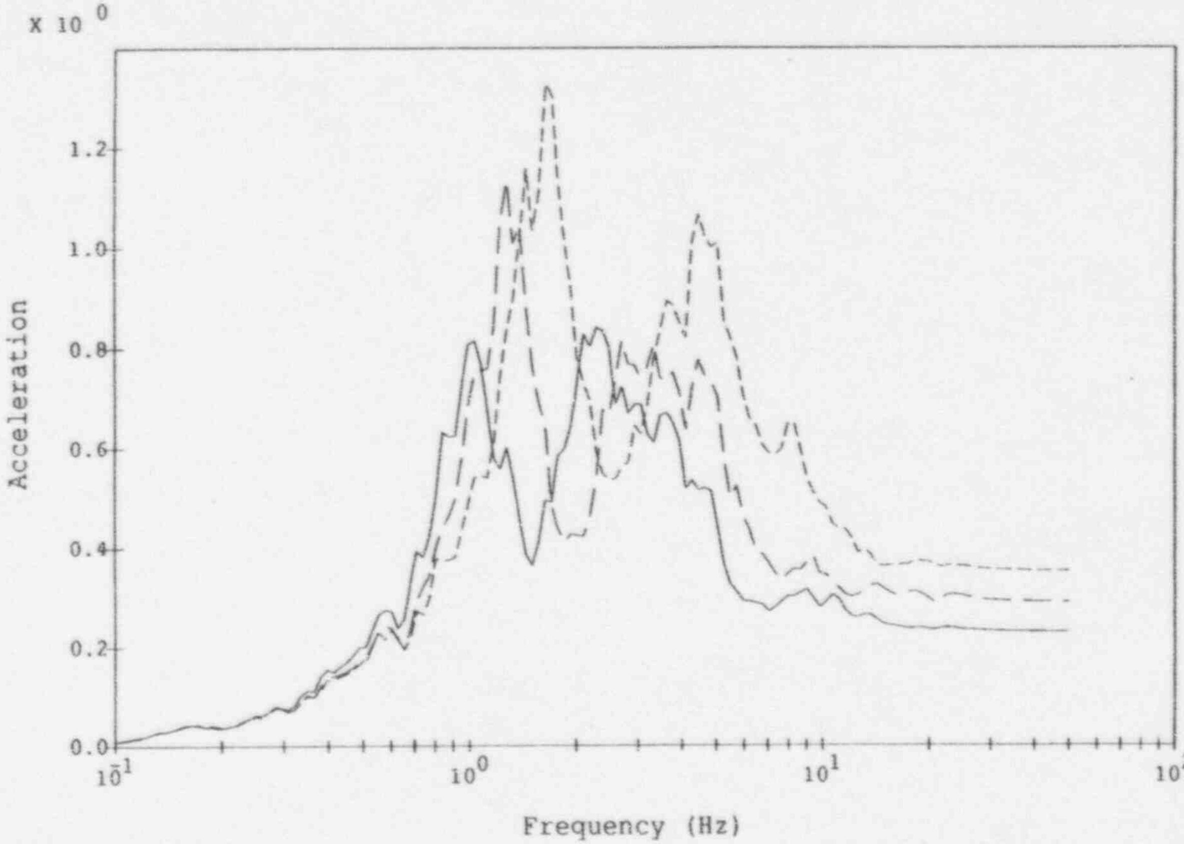
Legend:

Lower Bound	————————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's
5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 6, Elevation 228.0 ft, East-West Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

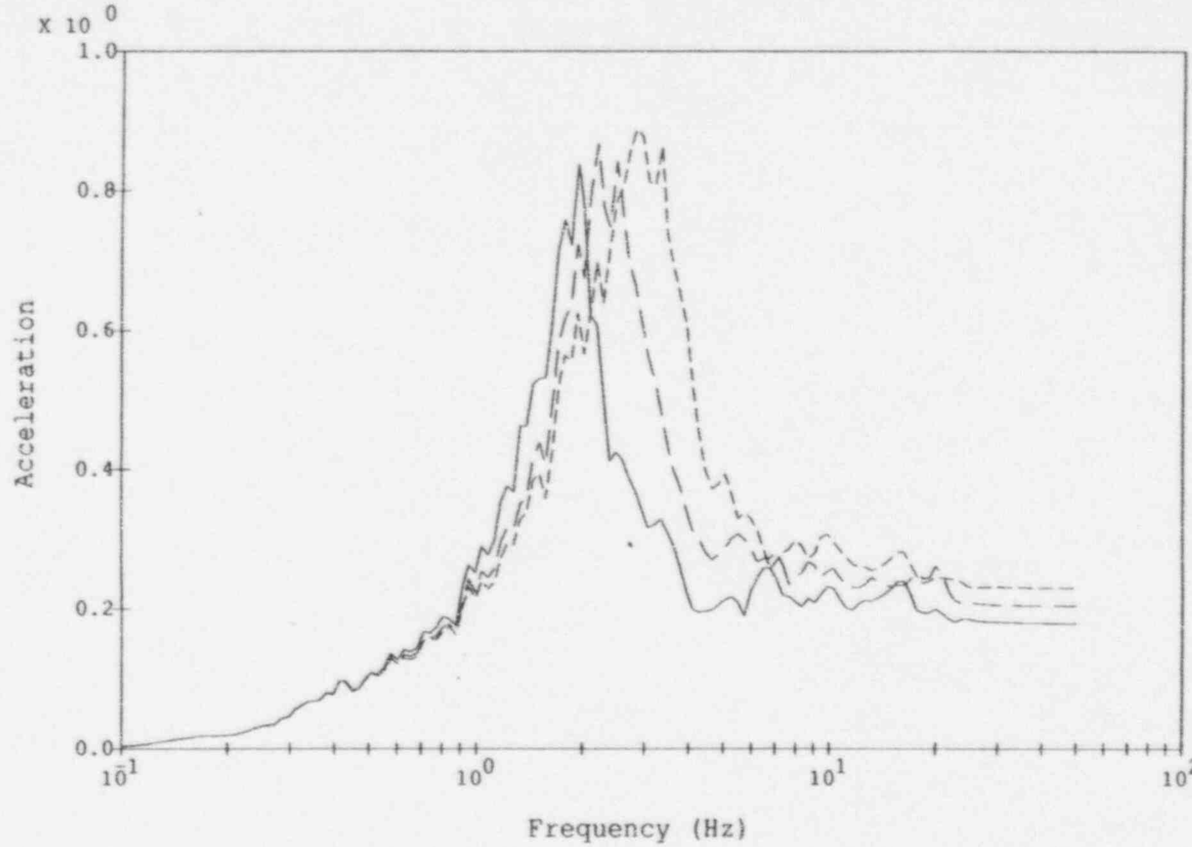
Upper Bound ---------

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 6, Elevation 228.0 ft, North-South Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

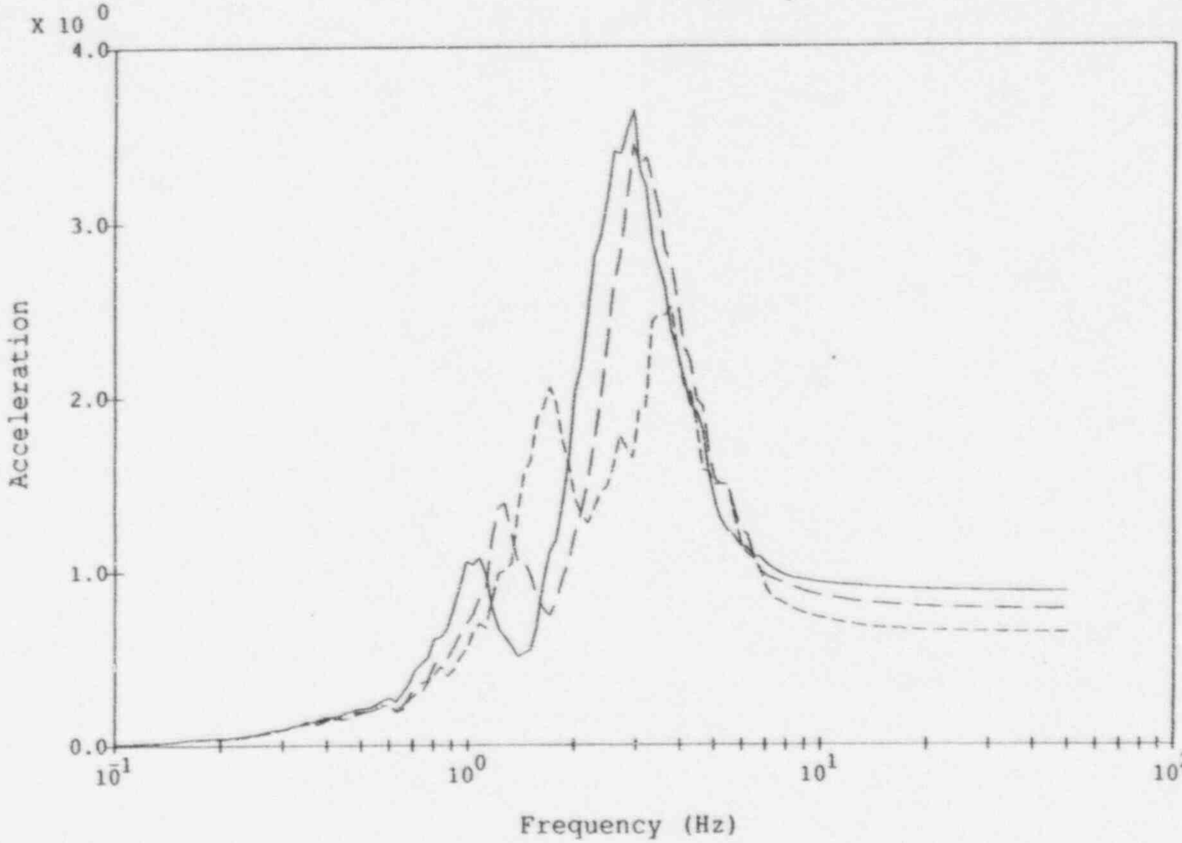
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 6, Elevation 228.0 ft, Vertical Direction



Legend:

Lower Bound

Intermediate

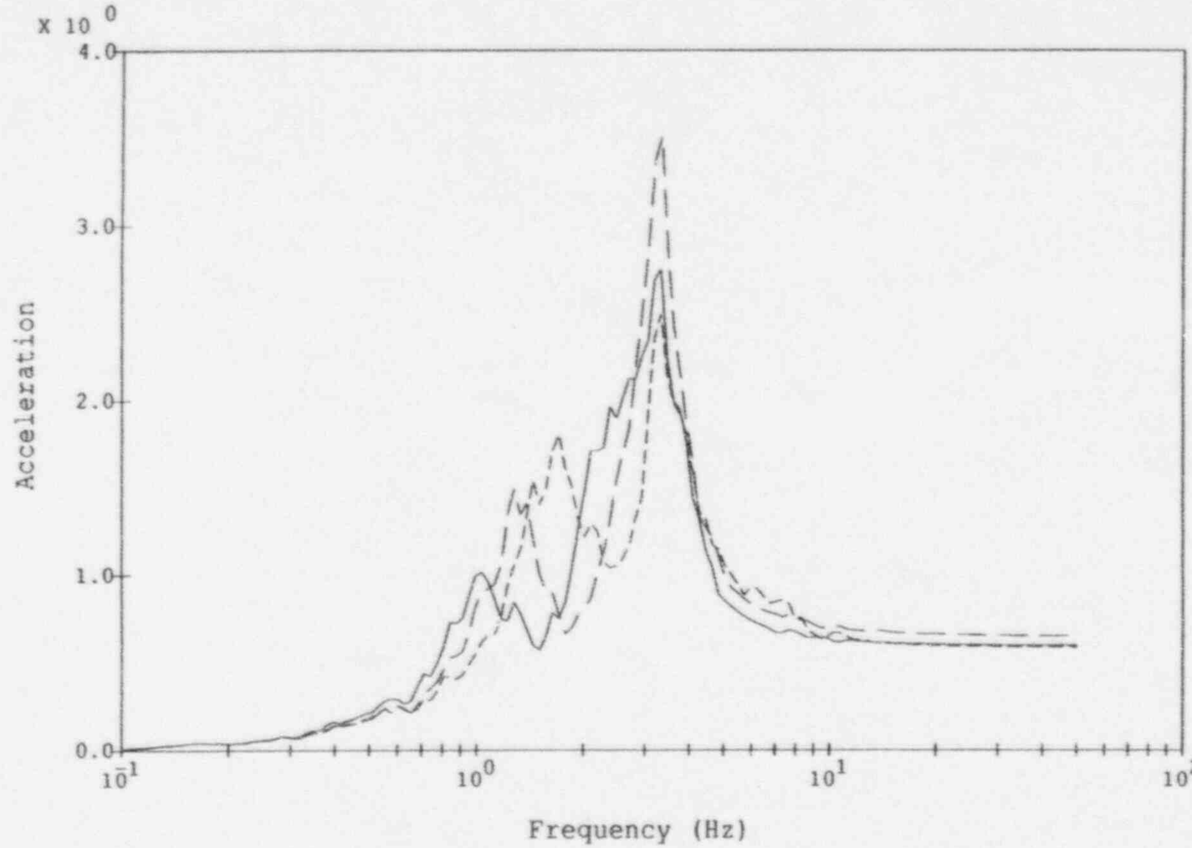
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 7, Elevation 256.5 ft, East-West Direction



Legend:

Lower Bound

Intermediate

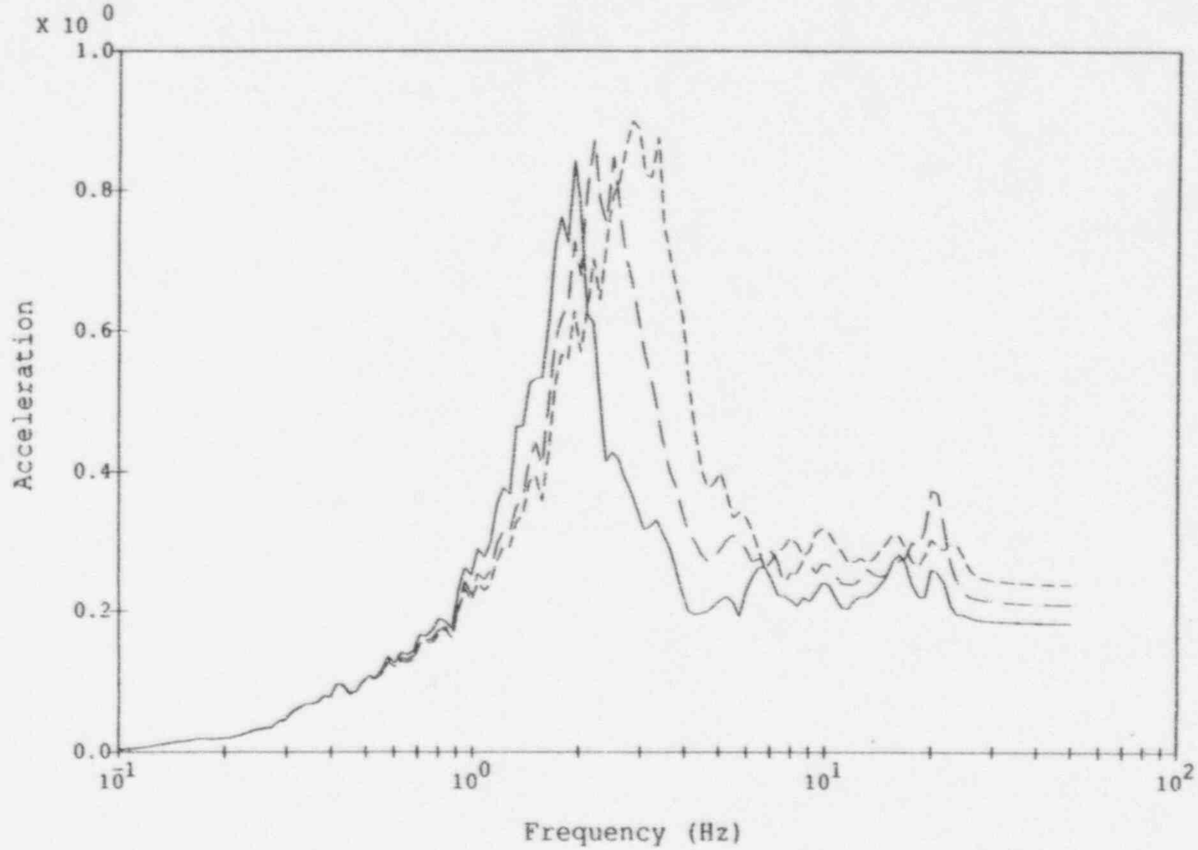
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 7, Elevation 256.5 ft, North-South Direction



Legend:

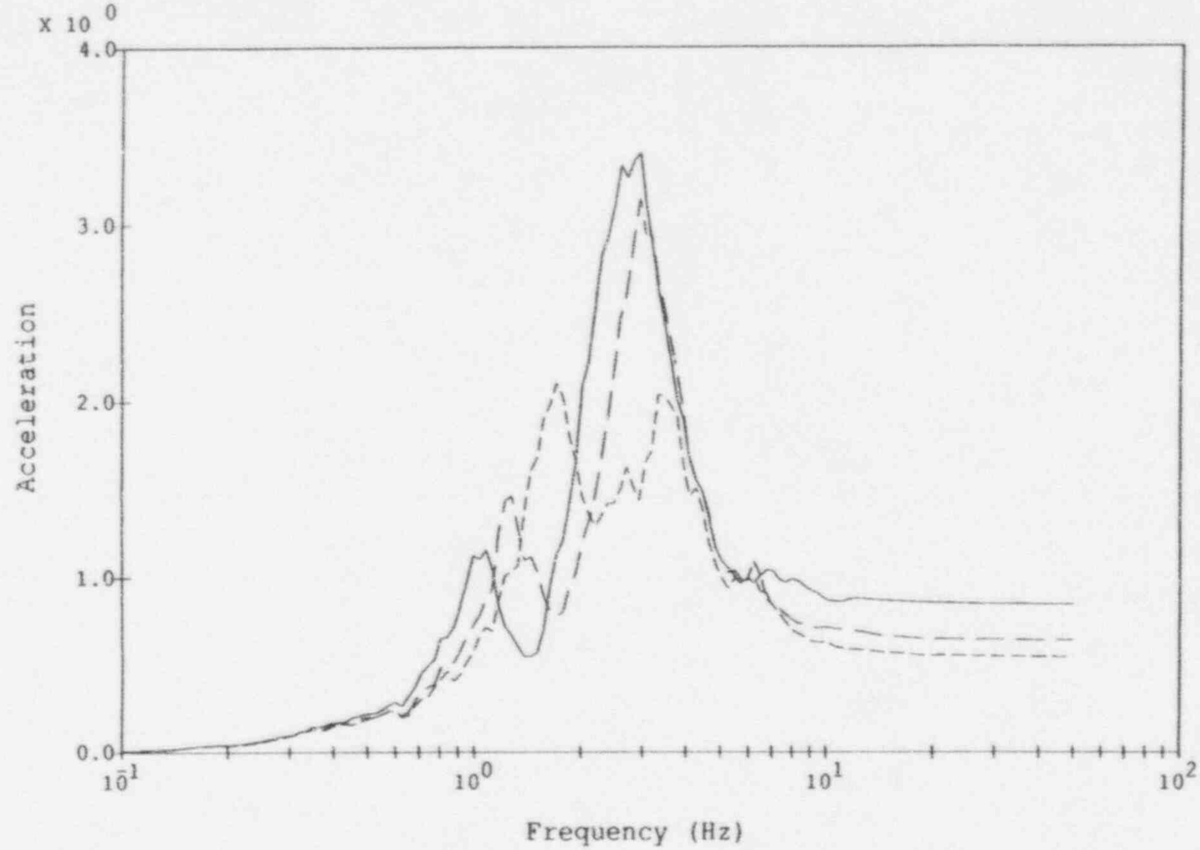
Lower Bound	———————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 7, Elevation 256.5 ft, Vertical Direction



Legend:

Lower Bound ————

Intermediate — — — —

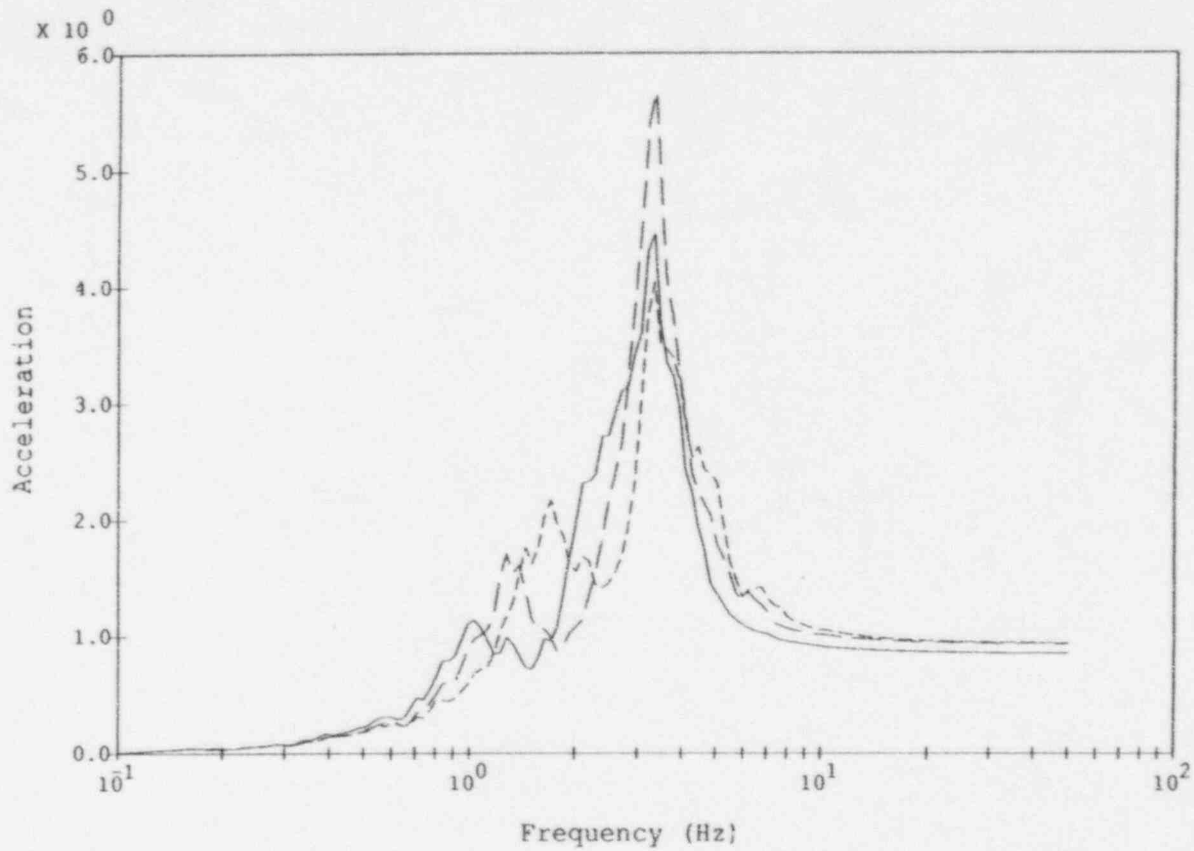
Upper Bound - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 8, Elevation 280.0 ft, East-West Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

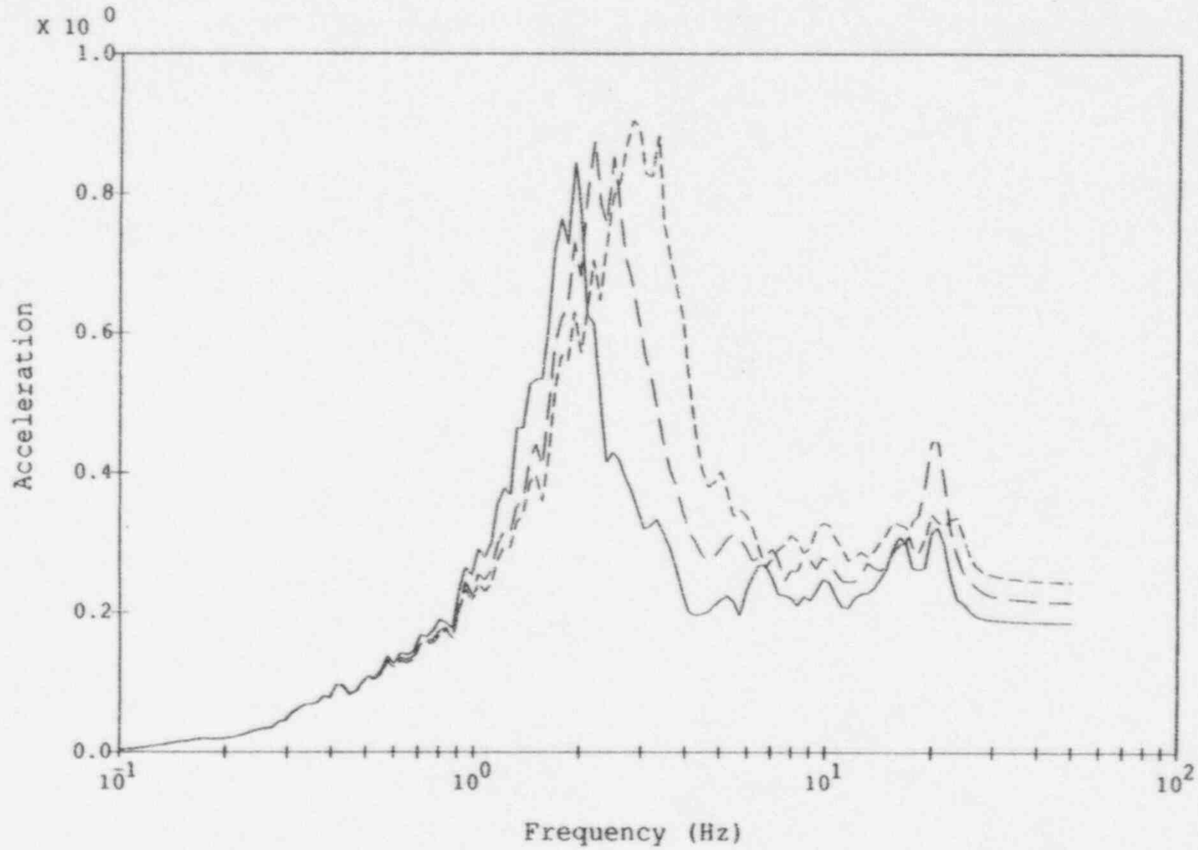
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 8, Elevation 280.0 ft, North-South Direction



Legend:

Lower Bound ————

Intermediate — — — —

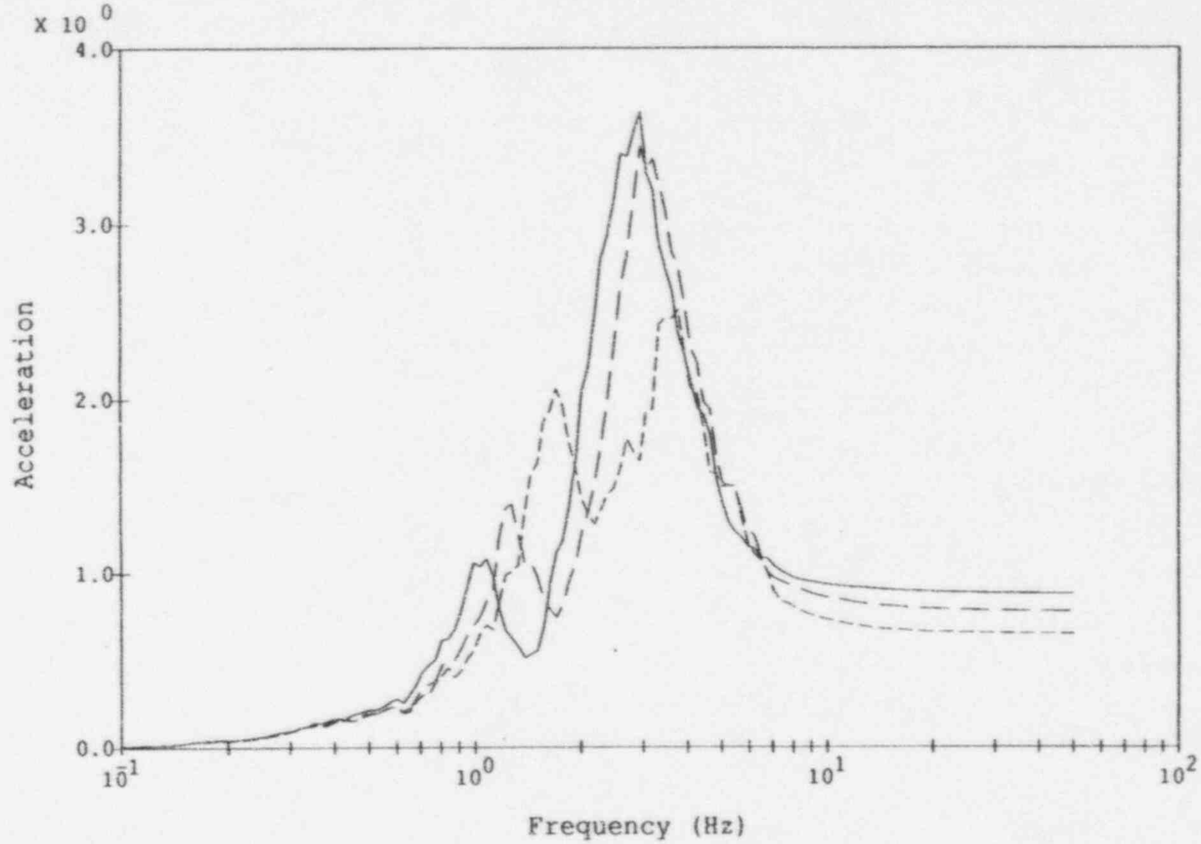
Upper Bound - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 8, Elevation 280.0 ft, Vertical Direction



Legend:

Lower Bound ____________

Intermediate — — — — —

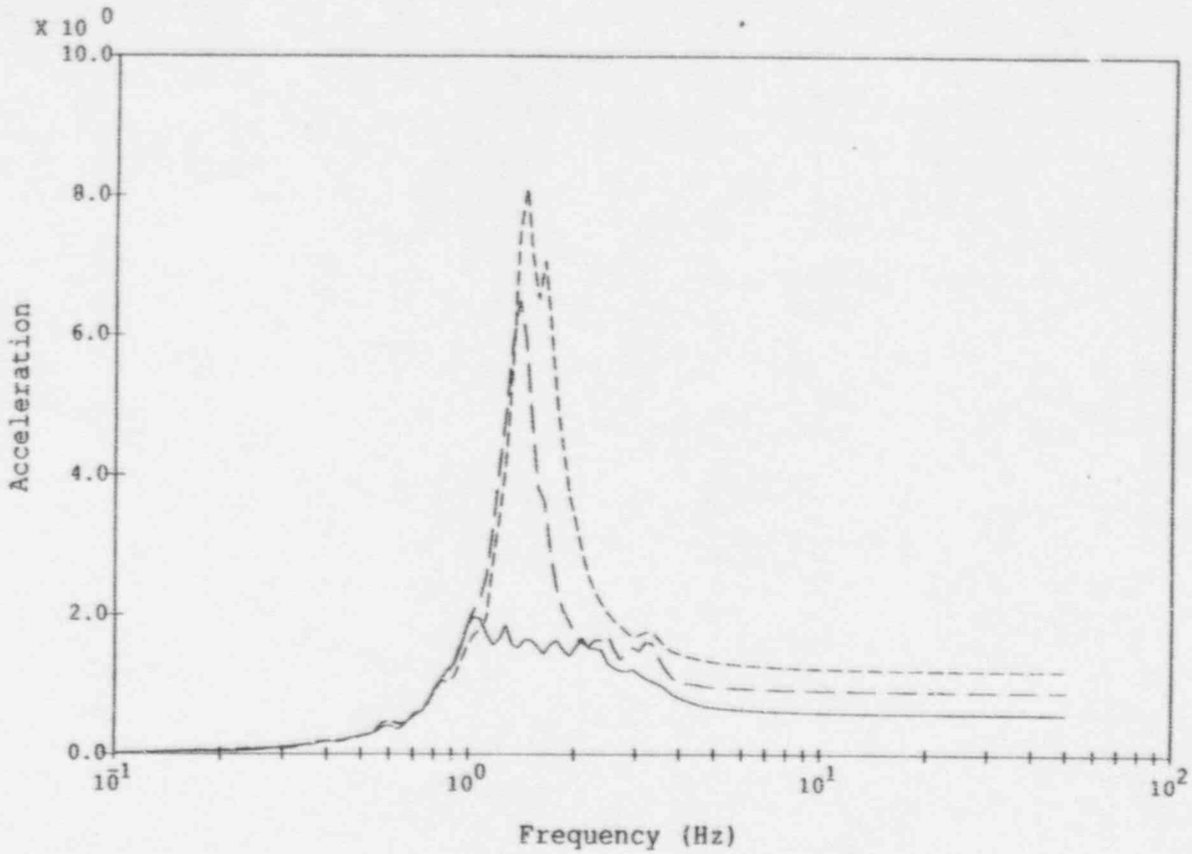
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 9, Elevation 256.5 ft, East-West Direction



Legend:

Lower Bound

Intermediate

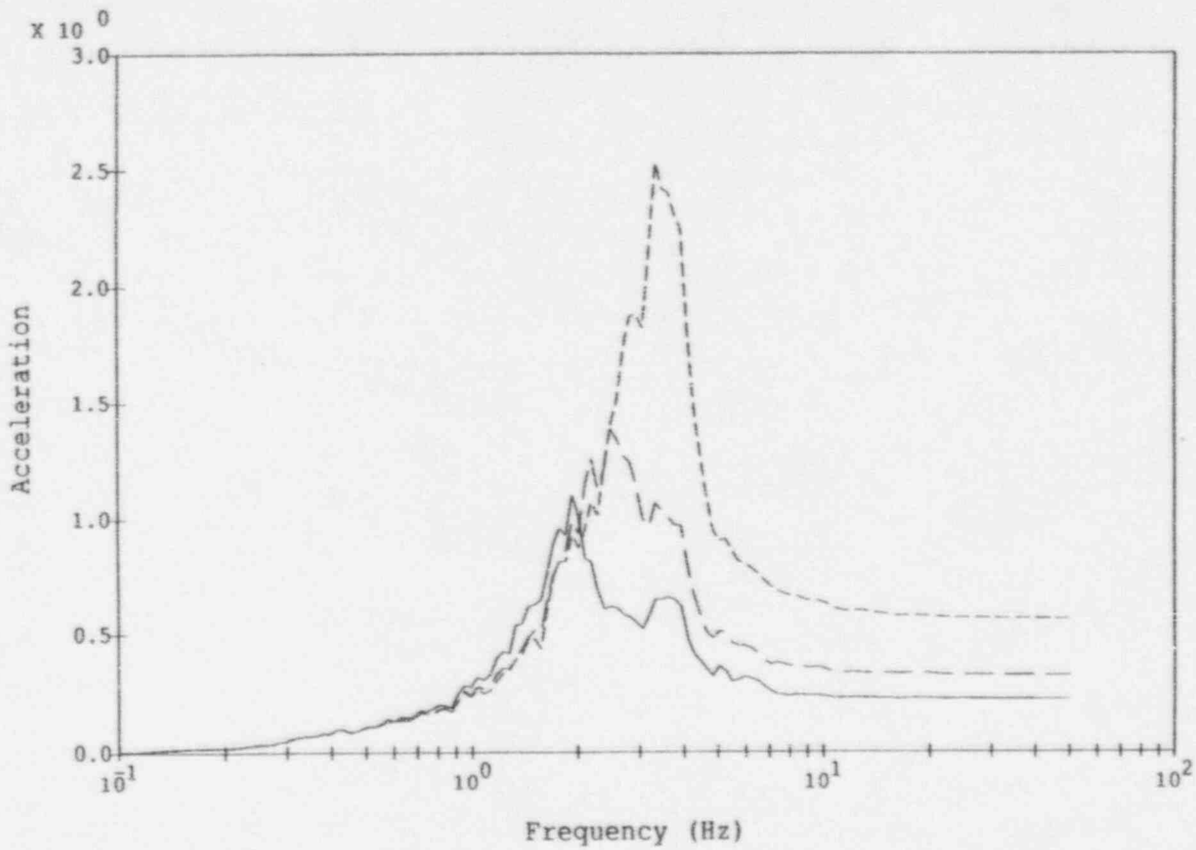
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 9, Elevation 256.5 ft, North-South Direction



Legend:

Lower Bound ————

Intermediate — — — —

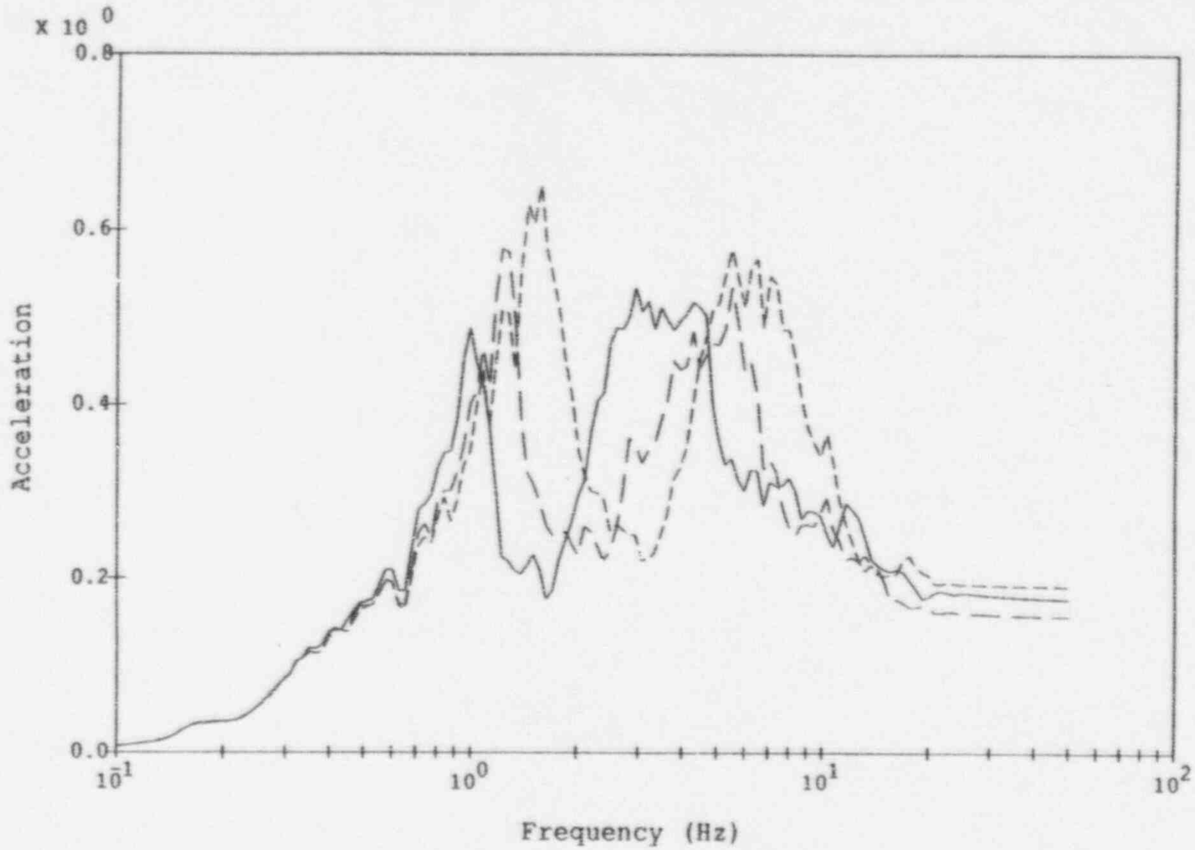
Upper Bound - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 9, Elevation 257.5 ft, Vertical Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

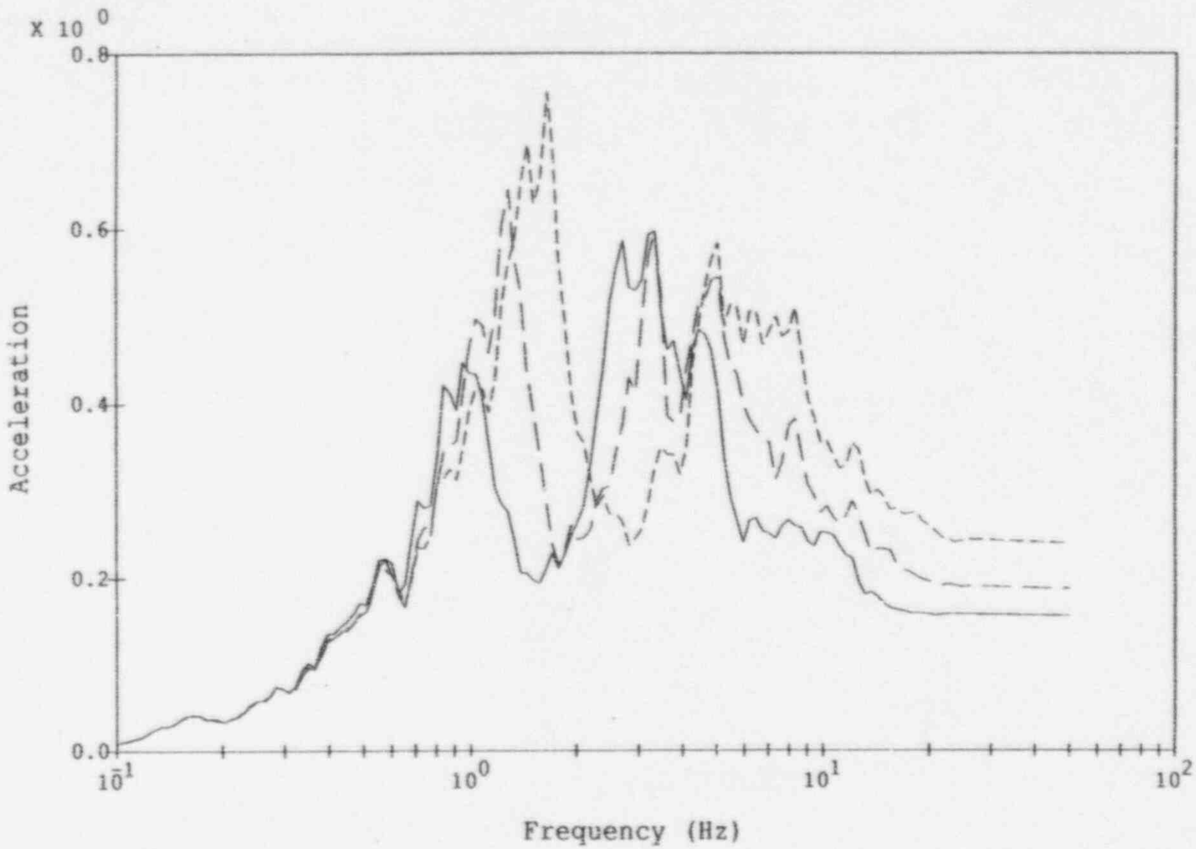
Upper Bound - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 10, Elevation 114.5 ft, East-West Direction



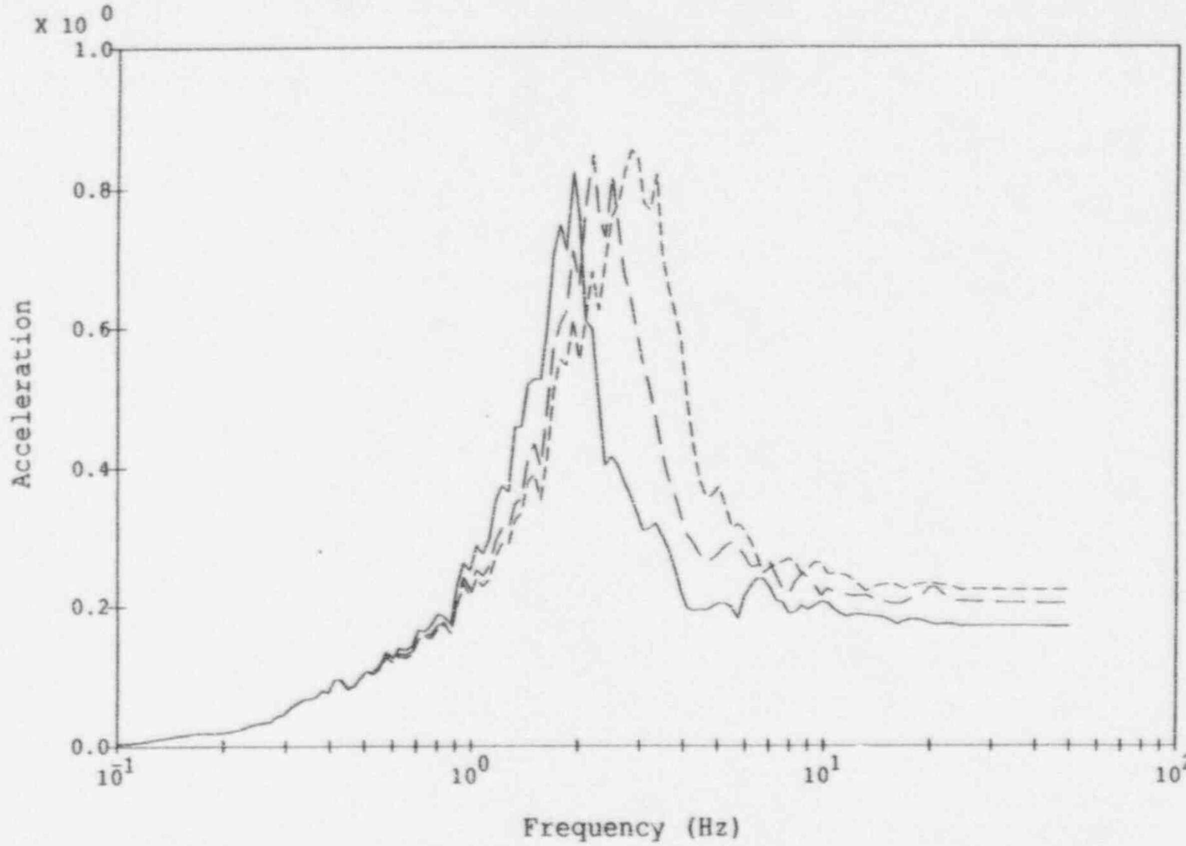
Legend:

Lower Bound	————————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's
5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 10, Elevation 114.5 ft, North-South Direction



Legend:

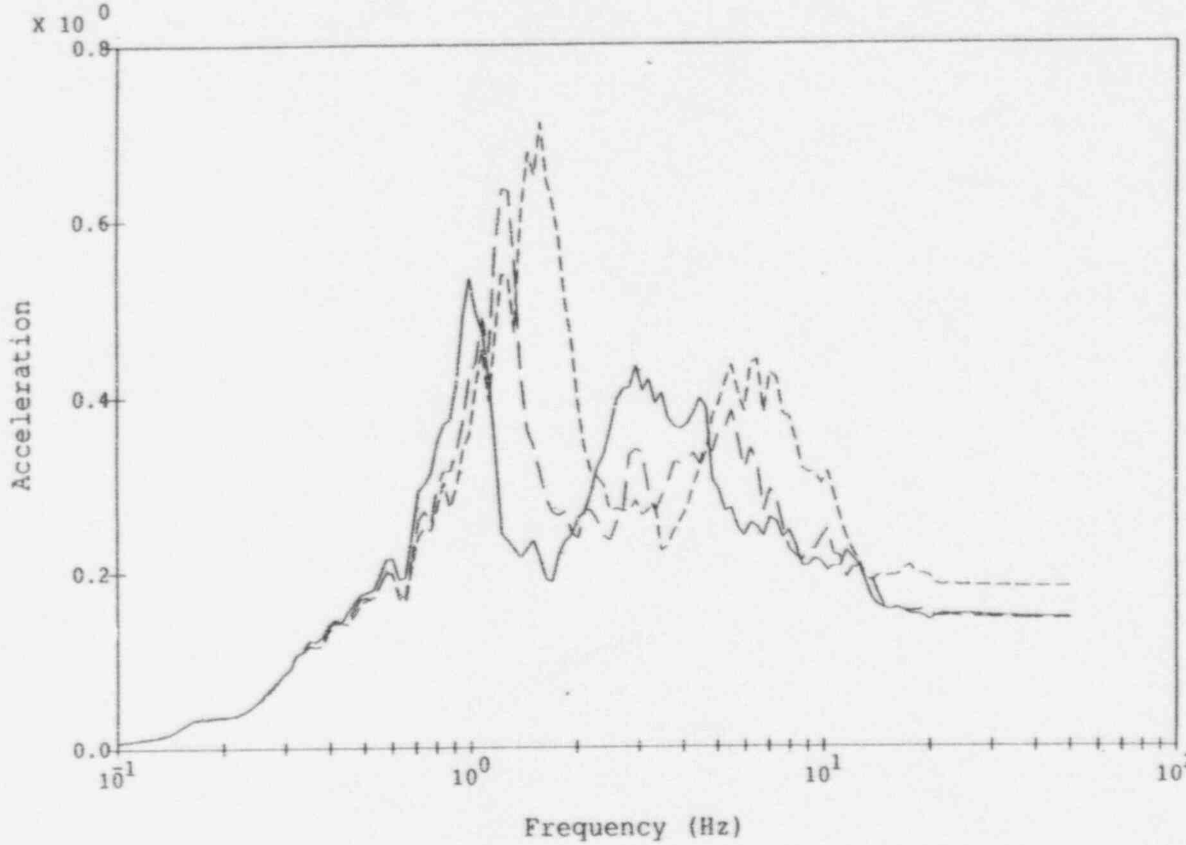
Lower Bound	————————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 10, Elevation 114.5 ft, Vertical Direction



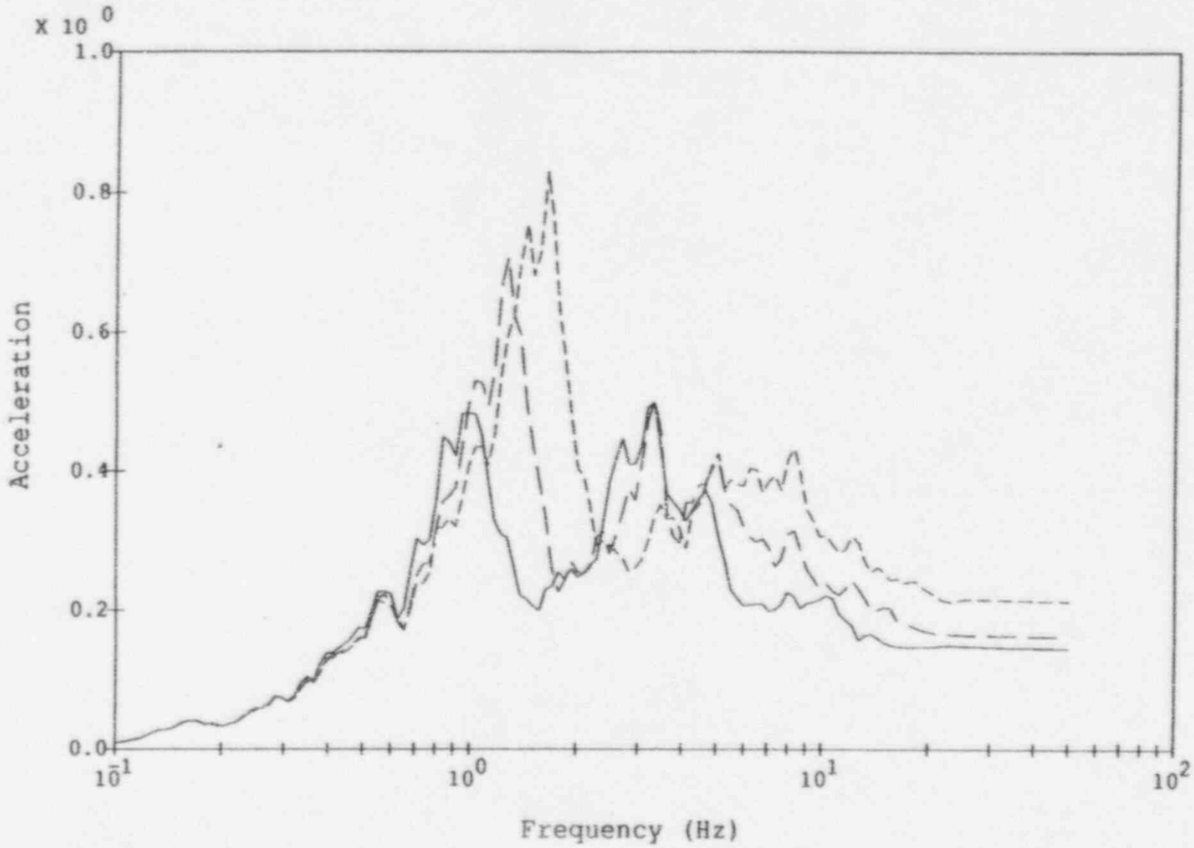
Legend:

Lower Bound	————————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's
5% Spectral Damping

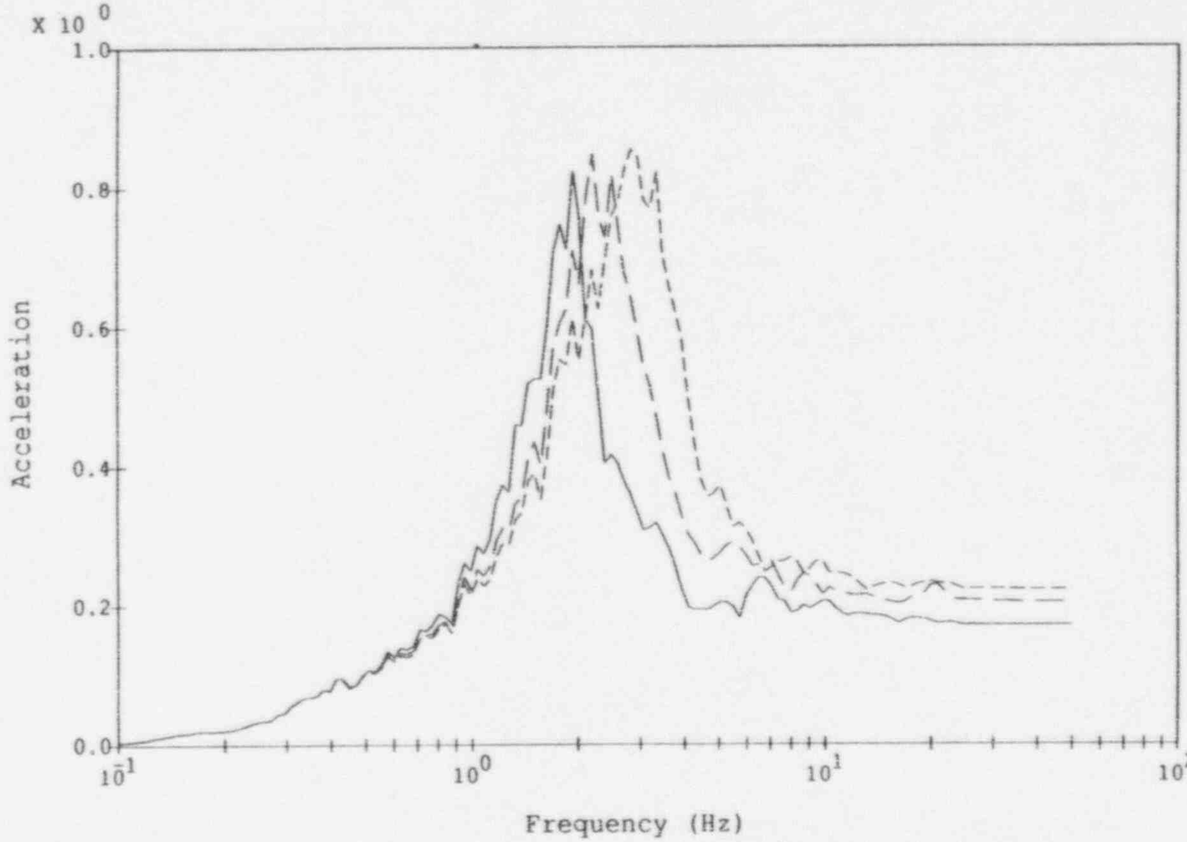
Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 11, Elevation 130.0 ft, East-West Direction



Legend:
Lower Bound
Intermediate
Upper Bound

Notes:
Accelerations in g's
5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 11, Elevation 130.0 ft, North-South Direction



Legend:

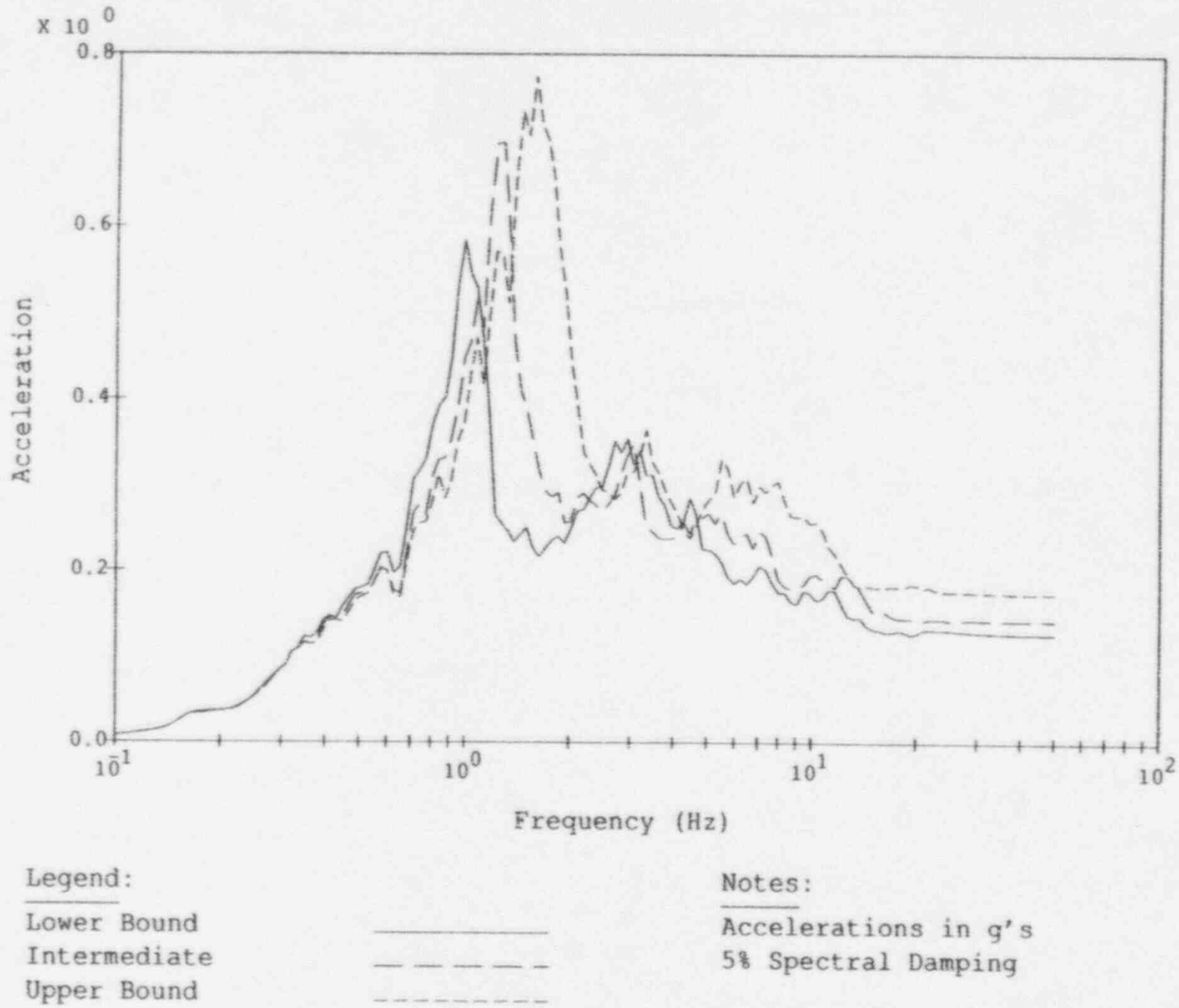
Lower Bound	———————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

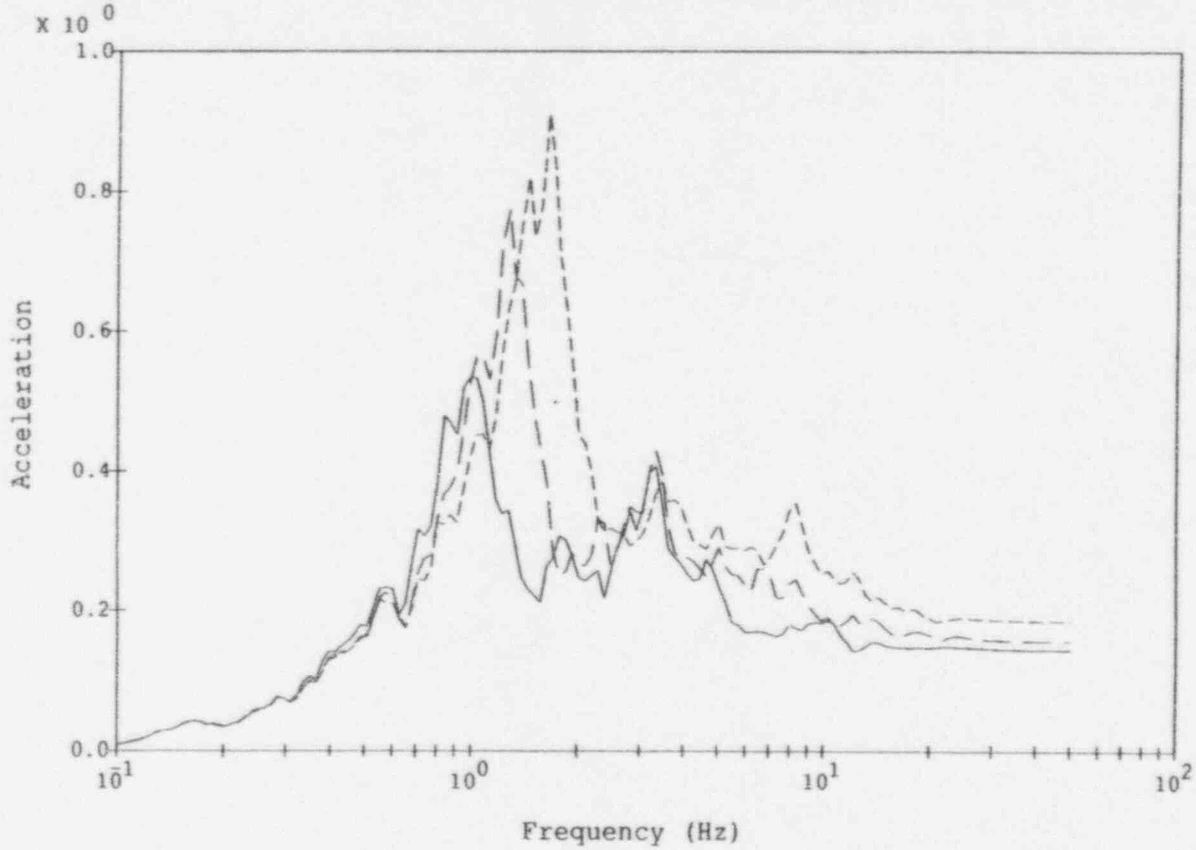
Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 11, Elevation 130.0 ft, Vertical Direction



Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 12, Elevation 146.0 ft, East-West Direction



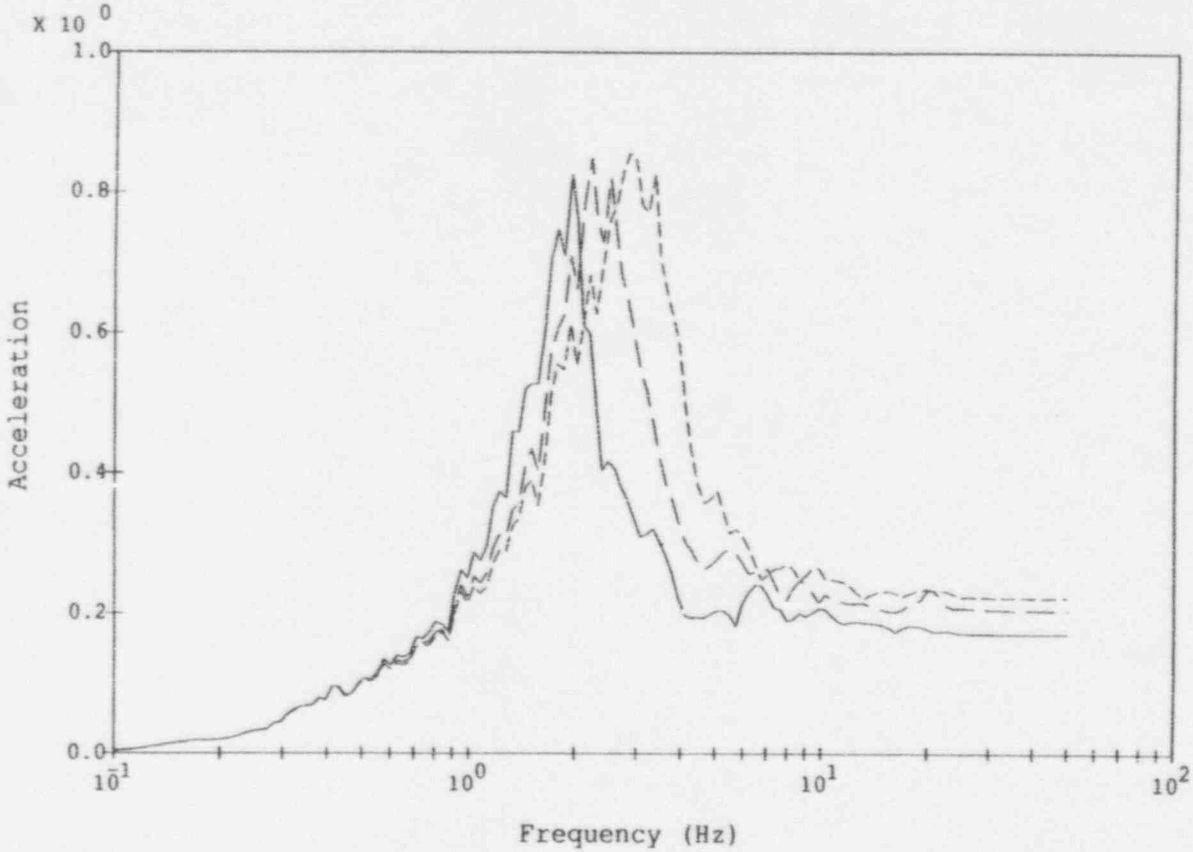
Legend:

Lower Bound ————————
Intermediate — — — — —
Upper Bound ----------

Notes:

Accelerations in g's
5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 12, Elevation 146.0 ft, North-South Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

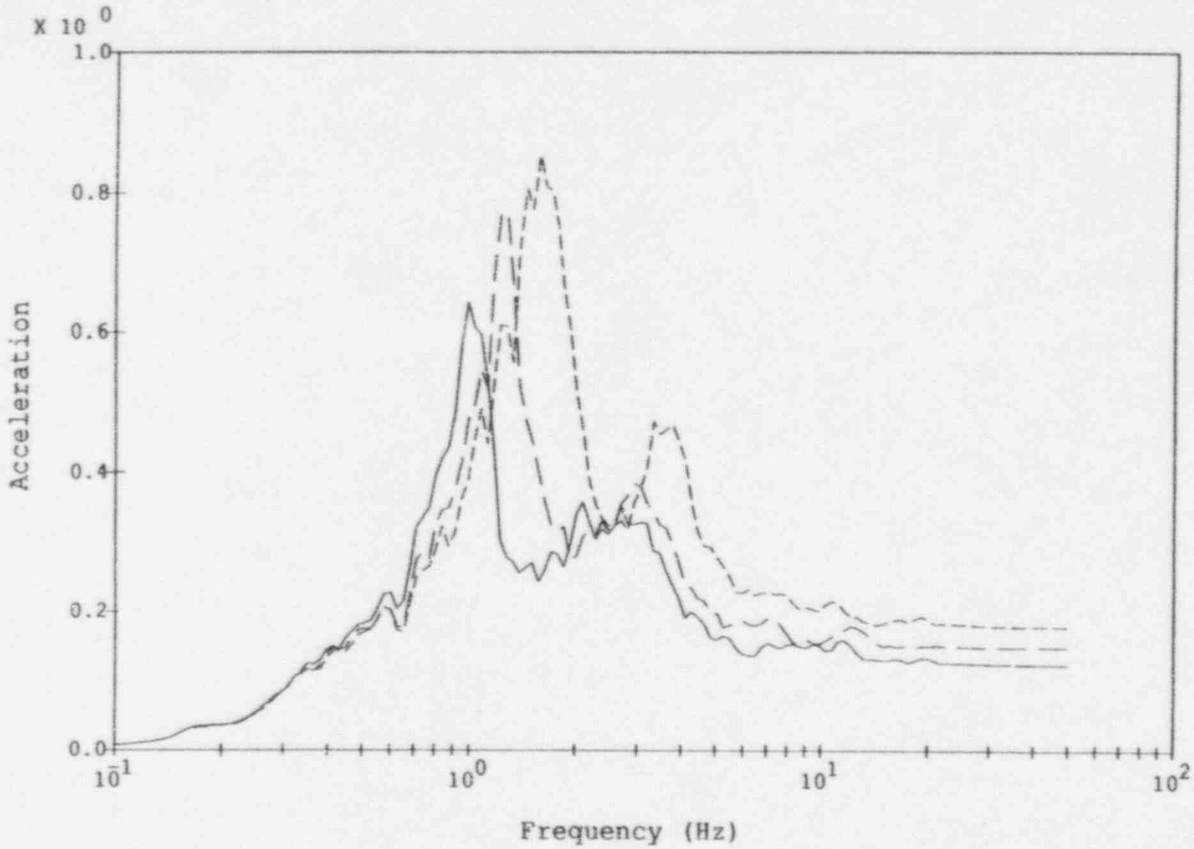
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 12, Elevation 146.0 ft, Vertical Direction



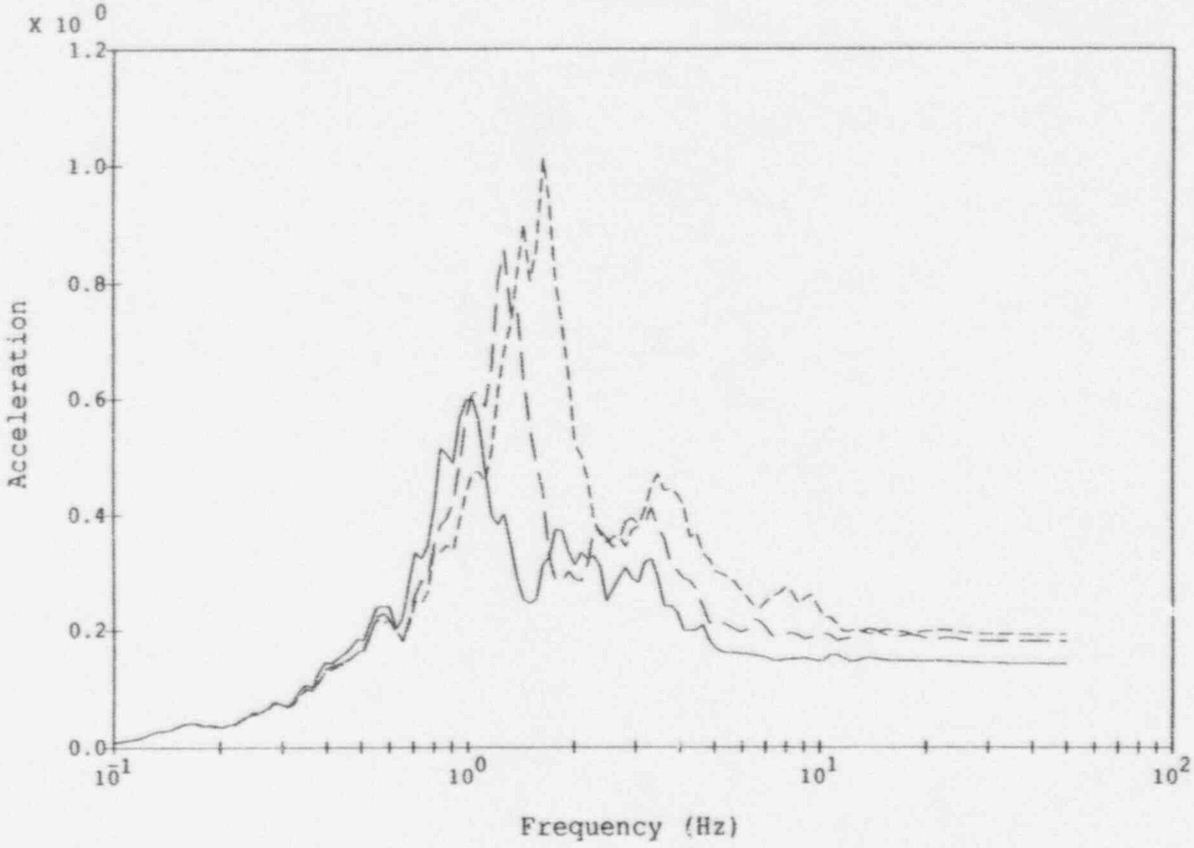
Legend:

Lower Bound	————————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's
5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 13, Elevation 165.0 ft, East-West Direction



Legend:

Lower Bound

Intermediate

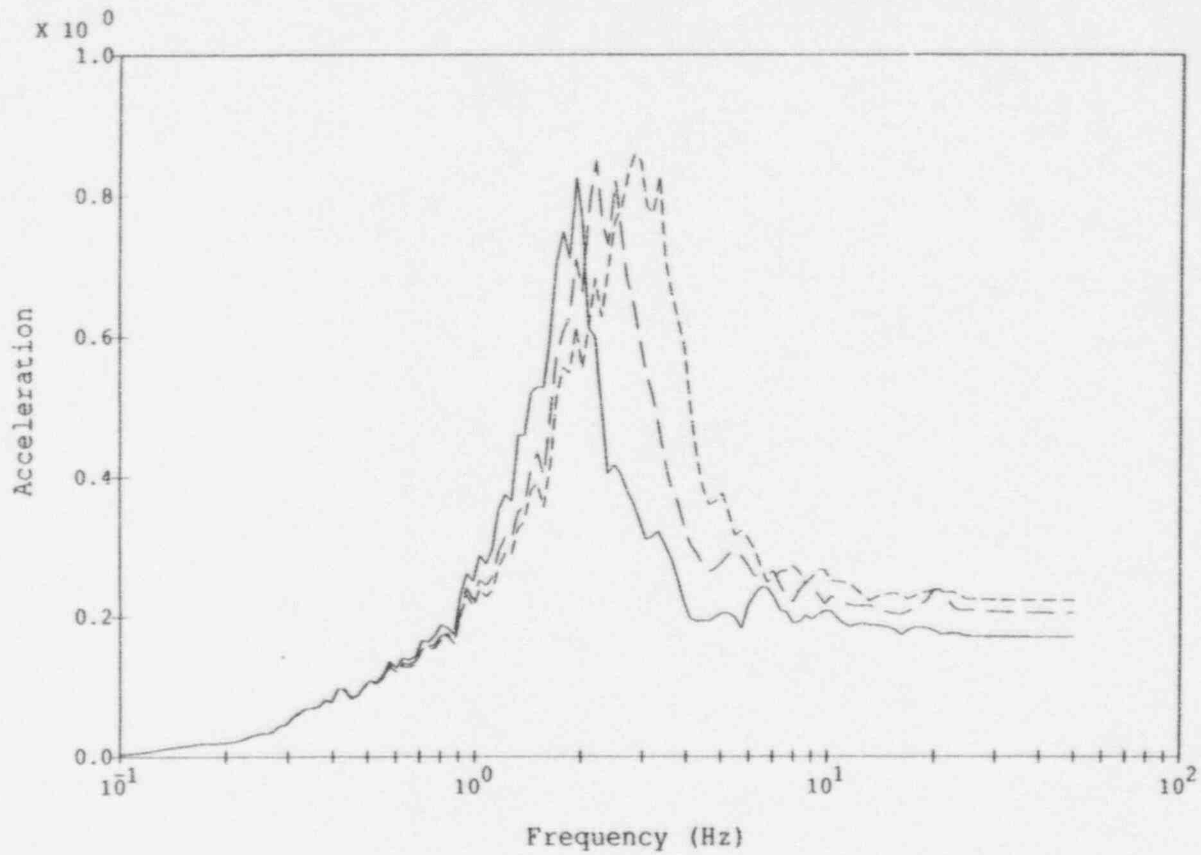
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 13, Elevation 165.0 ft, North-South Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

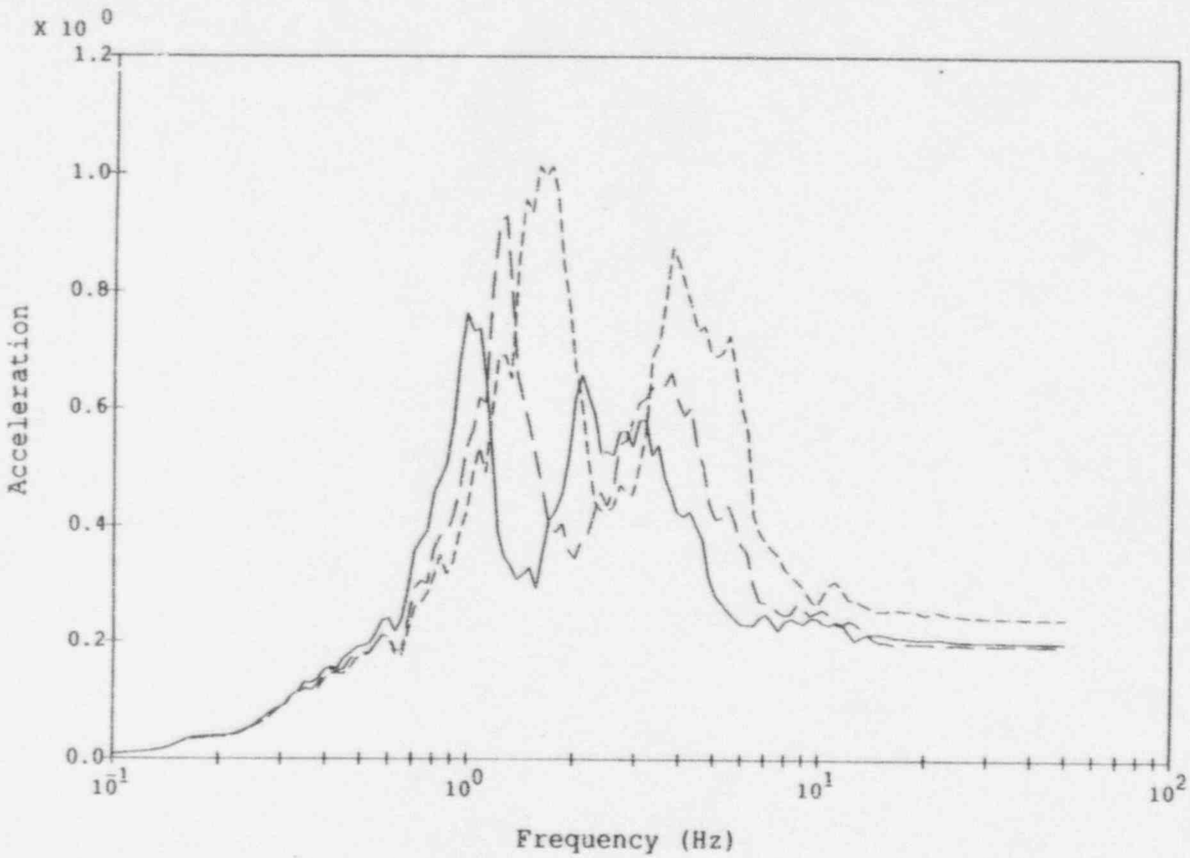
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 13, Elevation 165.0 ft, Vertical Direction



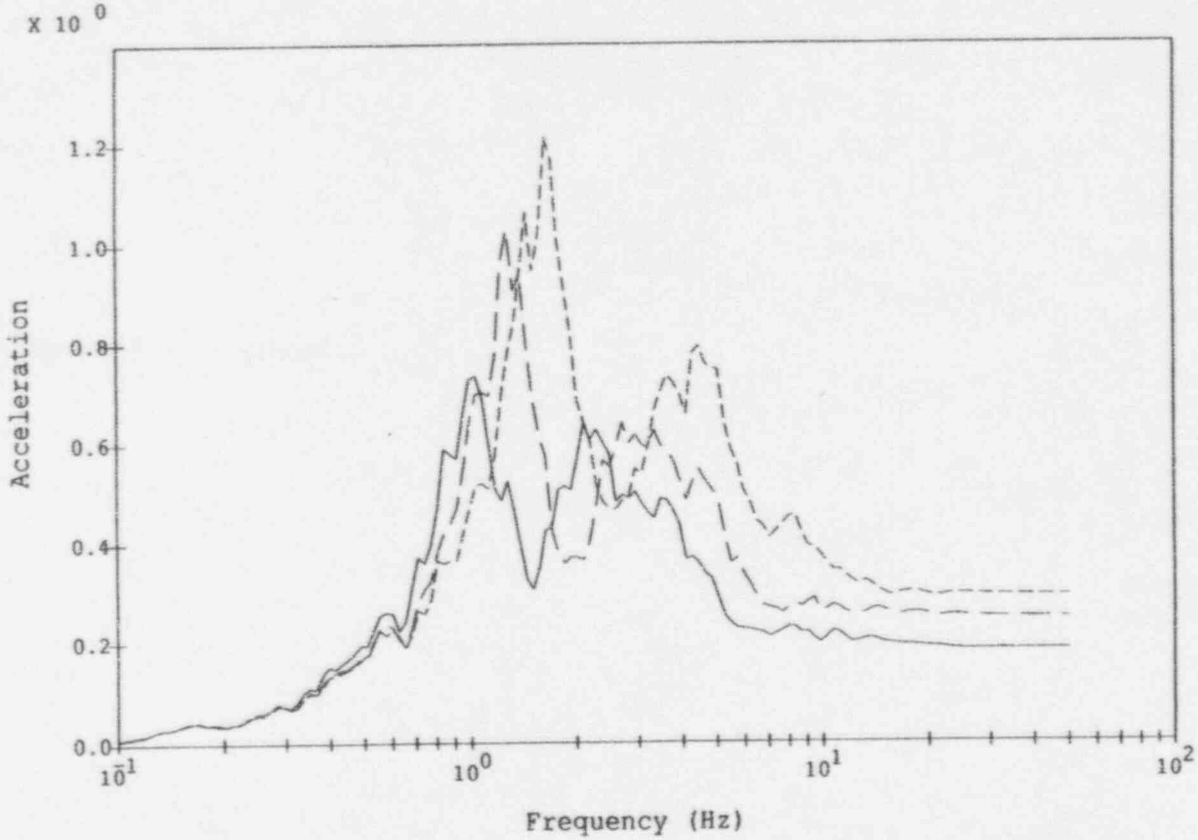
Legend:

Lower Bound
Intermediate
Upper Bound

Notes:

Accelerations in g's
5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 14, Elevation 204.0 ft, East-West Direction



Legend:

Lower Bound ————

Intermediate — — — —

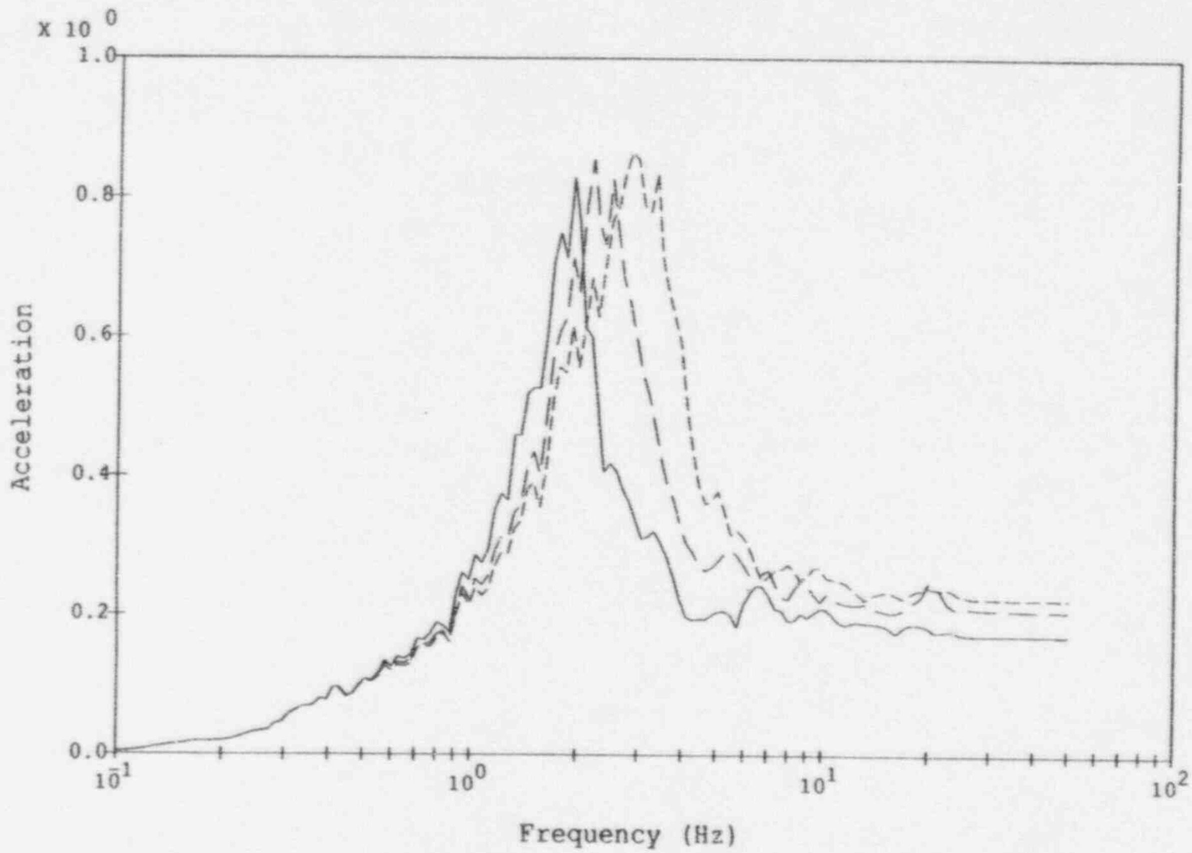
Upper Bound - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 14, Elevation 204.0 ft, North-South Direction



Legend:

Lower Bound

Intermediate

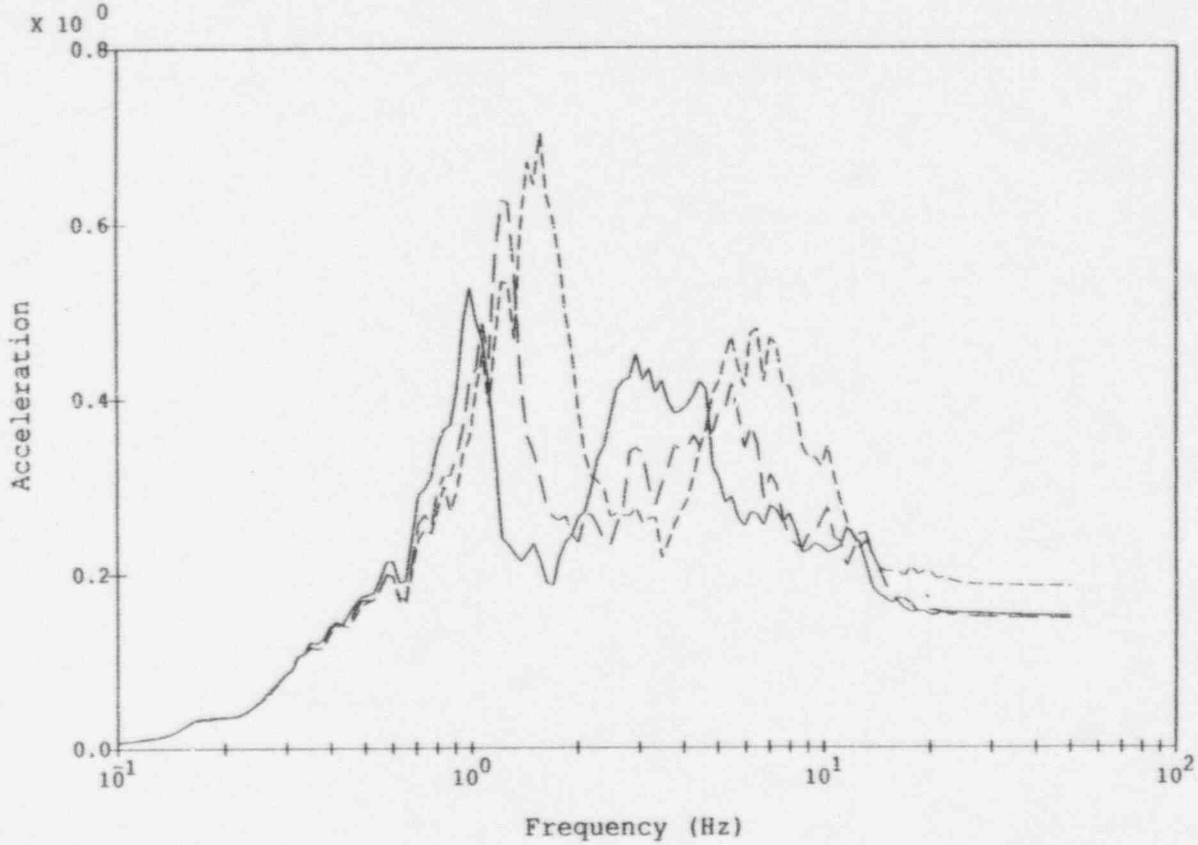
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 14, Elevation 204.0 ft, Vertical Direction



Legend:

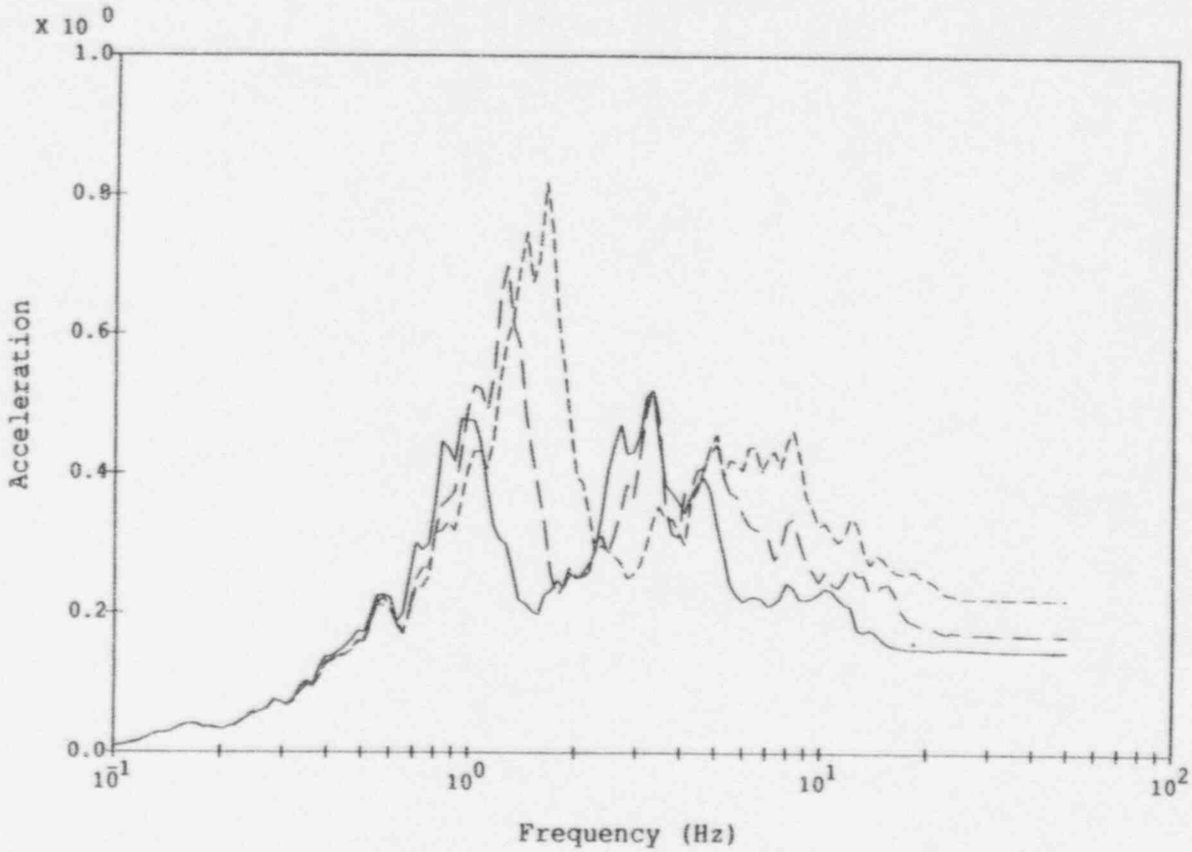
Lower Bound	————————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 15, Elevation 128.0 ft, East-West Direction



Legend:

Lower Bound ————————

Intermediate — — — — — —

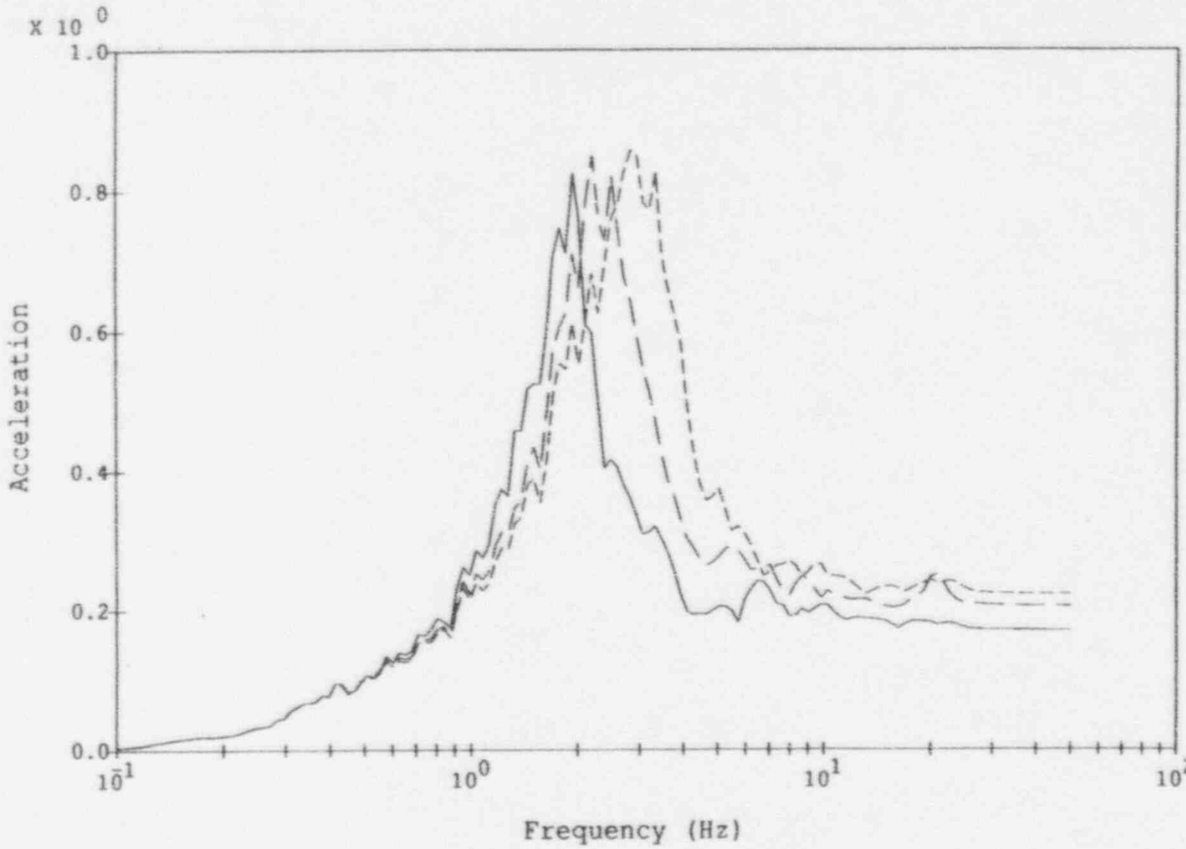
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 15, Elevation 128.0 ft, North-South Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

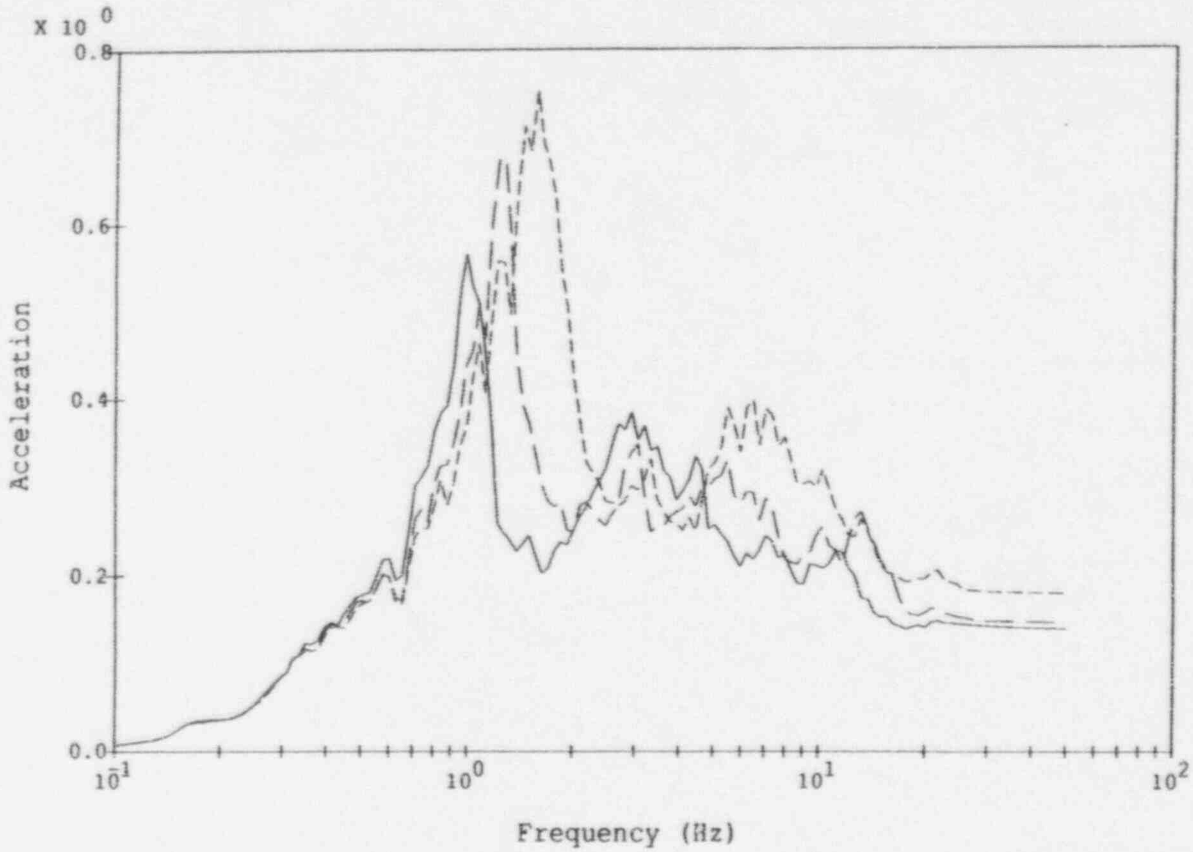
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 15, Elevation 128.0 ft, Vertical Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

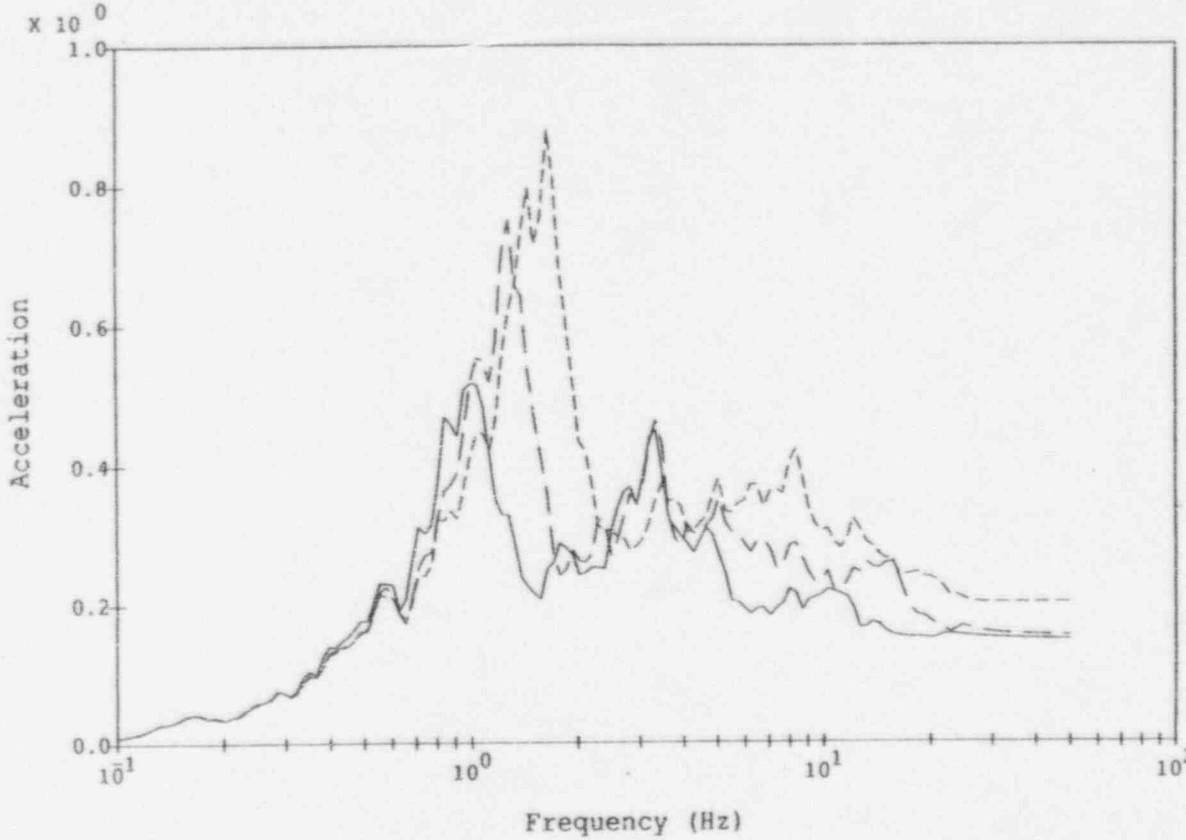
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 16, Elevation 141.0 ft, East-West Direction



Legend:

Lower Bound

Intermediate

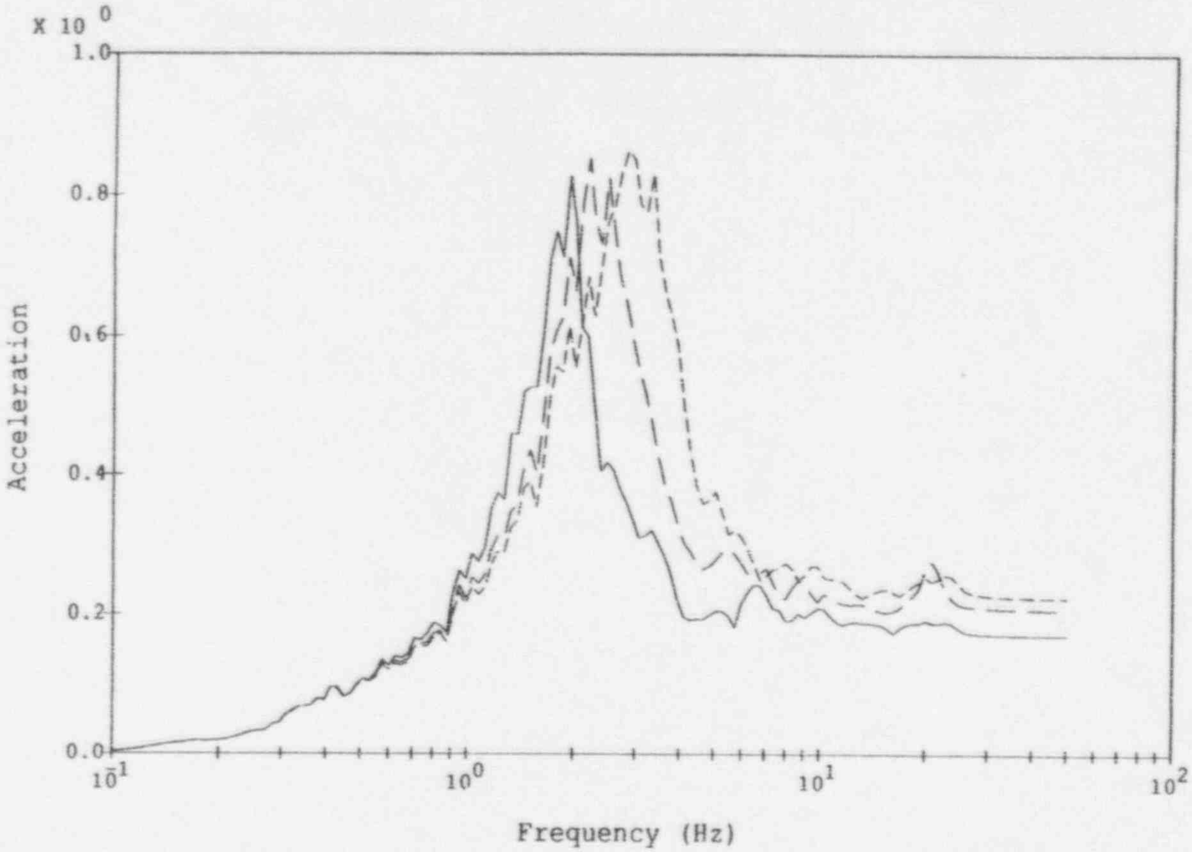
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 16, Elevation 141.0 ft, North-South Direction



Legend:

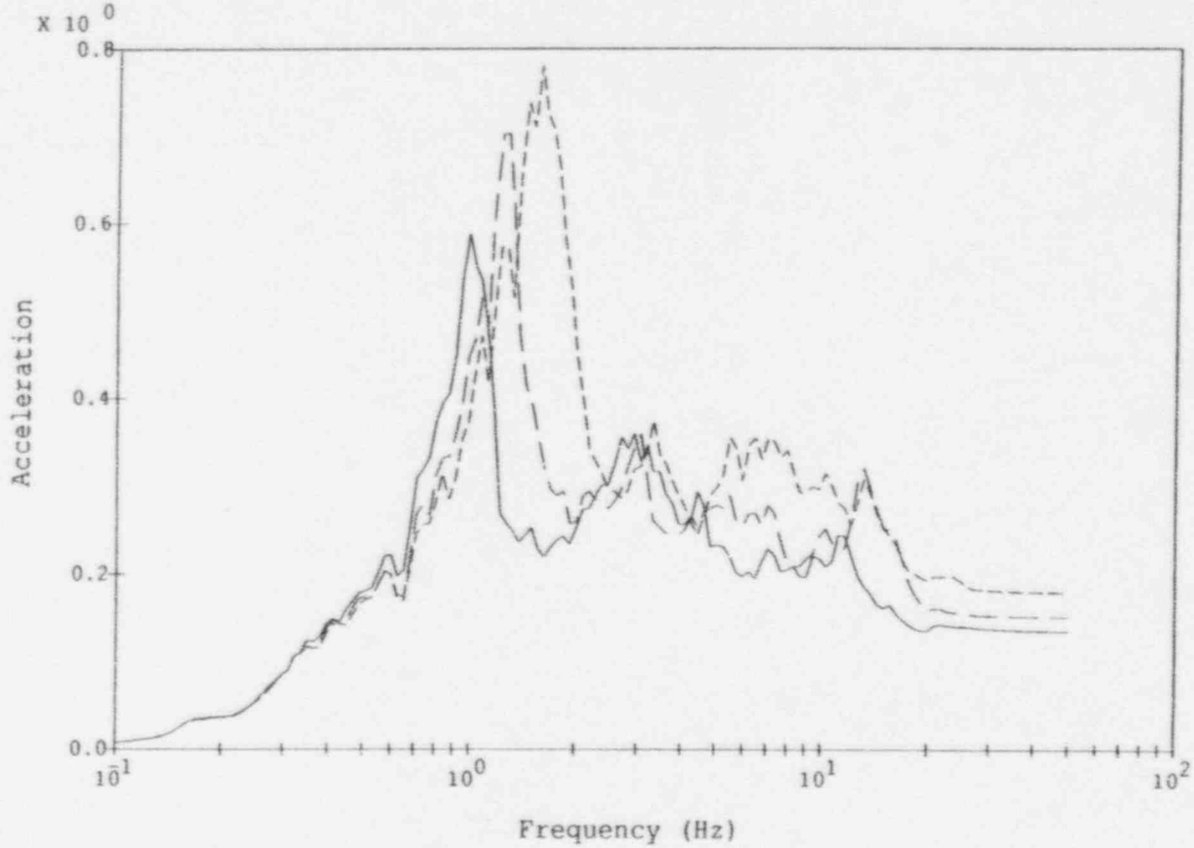
Lower Bound	————————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 16, Elevation 141.0 ft, Vertical Direction



Legend:

Lower Bound

Intermediate

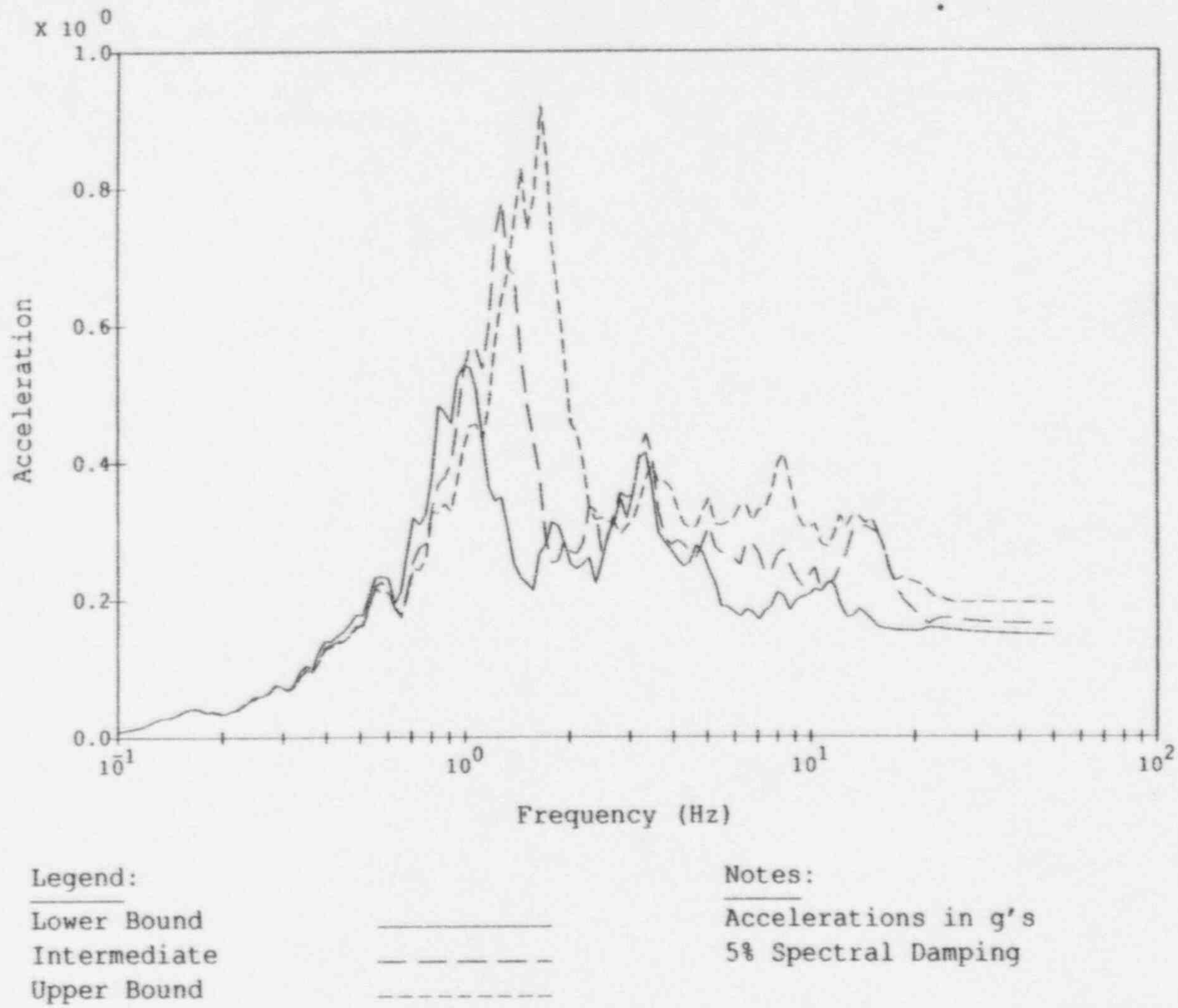
Upper Bound

Notes:

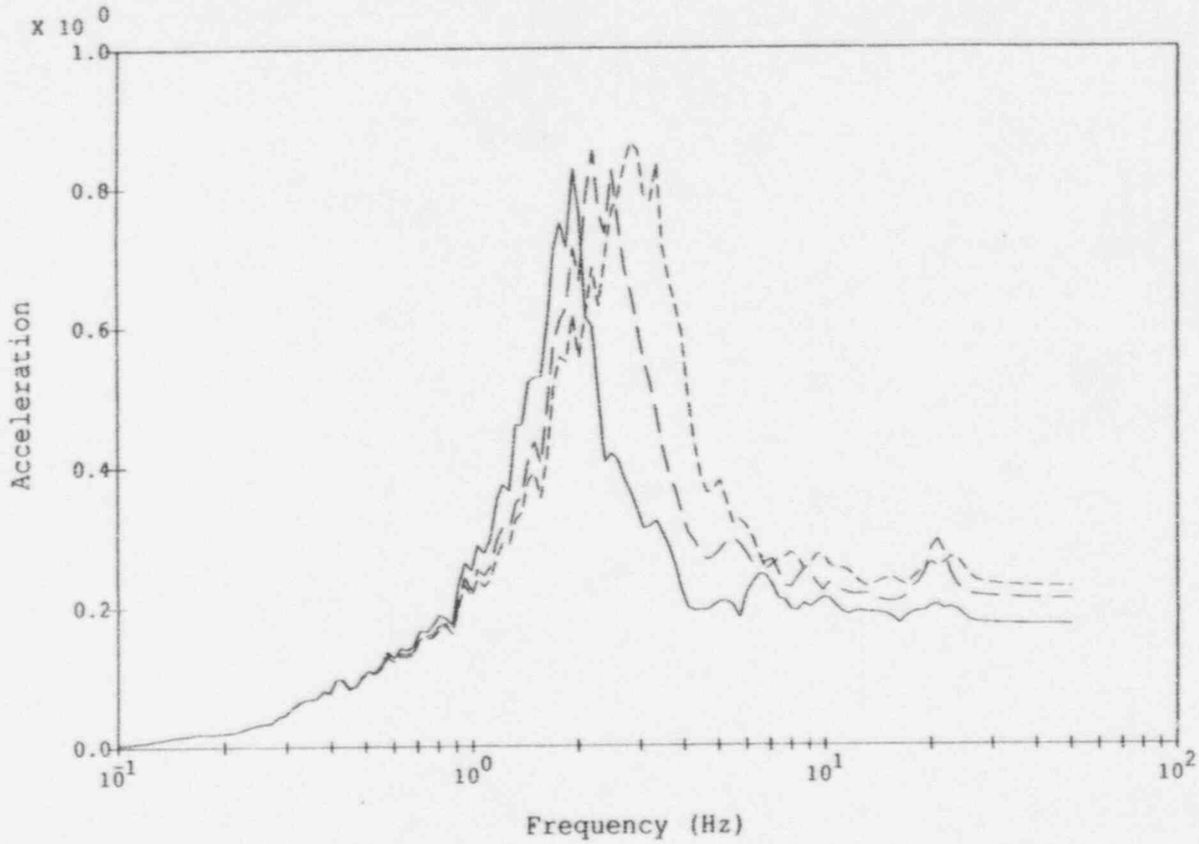
Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 17, Elevation 148.0 ft, East-West Direction



Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 17, Elevation 148.0 ft, North-South Direction



Legend:

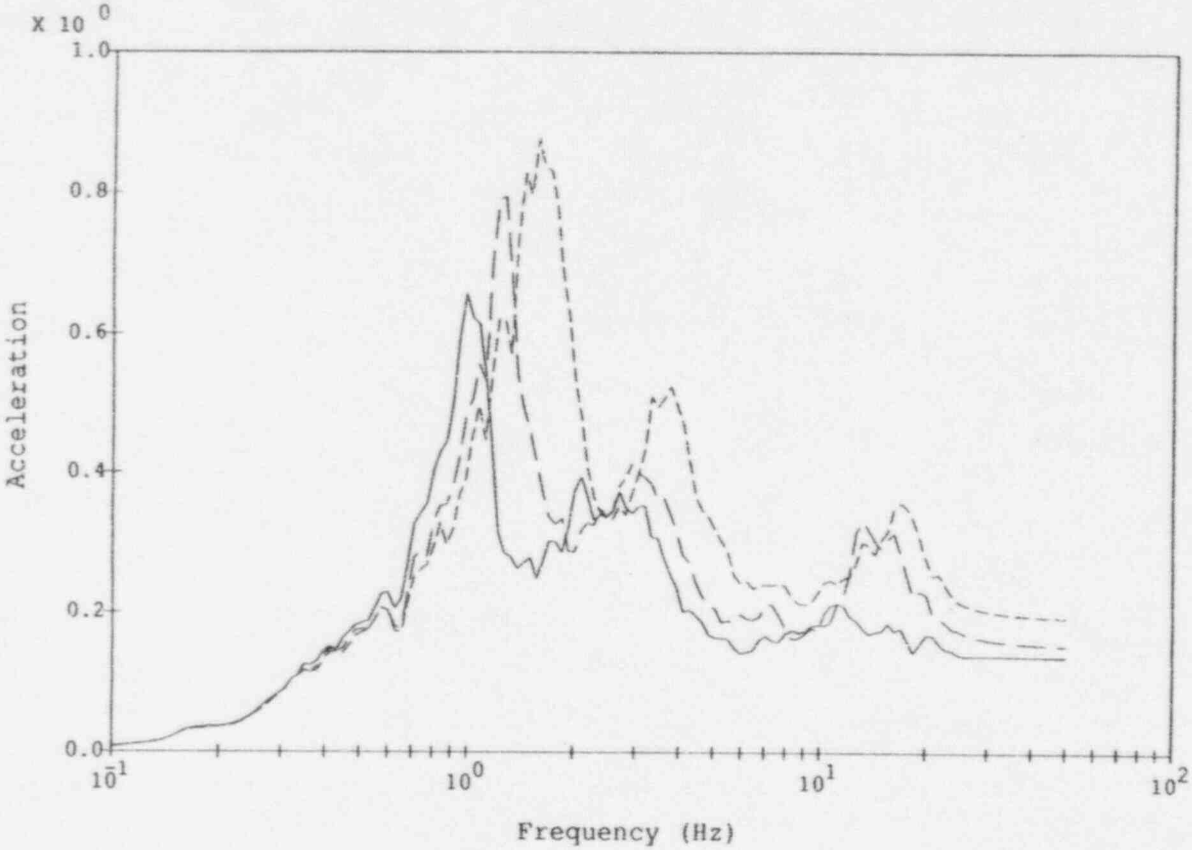
Lower Bound	————————
Intermediate	— — — — —
Upper Bound	- - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 17, Elevation 148.0 ft, Vertical Direction



Legend:

Lower Bound ————————

Intermediate — — — — — —

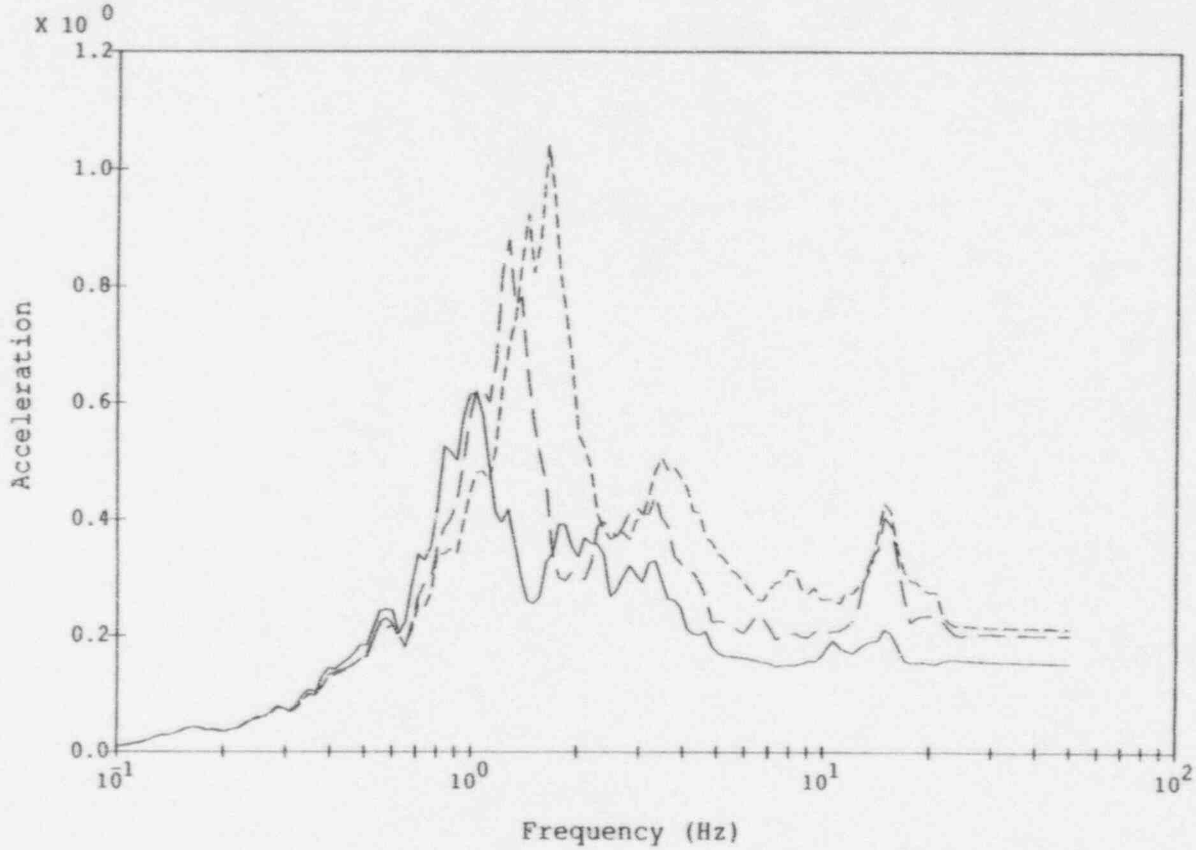
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 18, Elevation 170.0 ft, East-West Direction



Legend:

Lower Bound ——————

Intermediate — — — — —

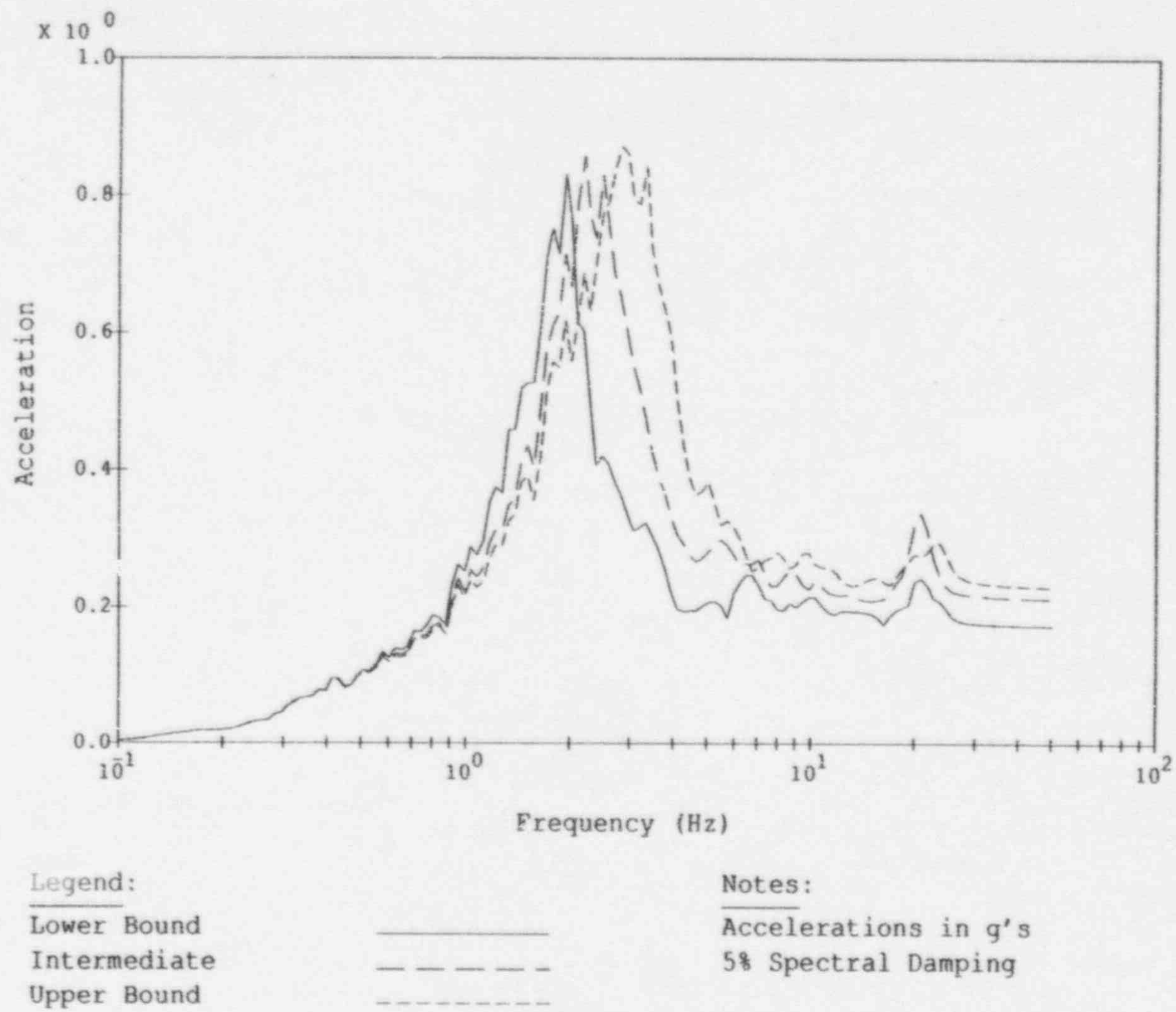
Upper Bound - - - - - - - - -

Notes:

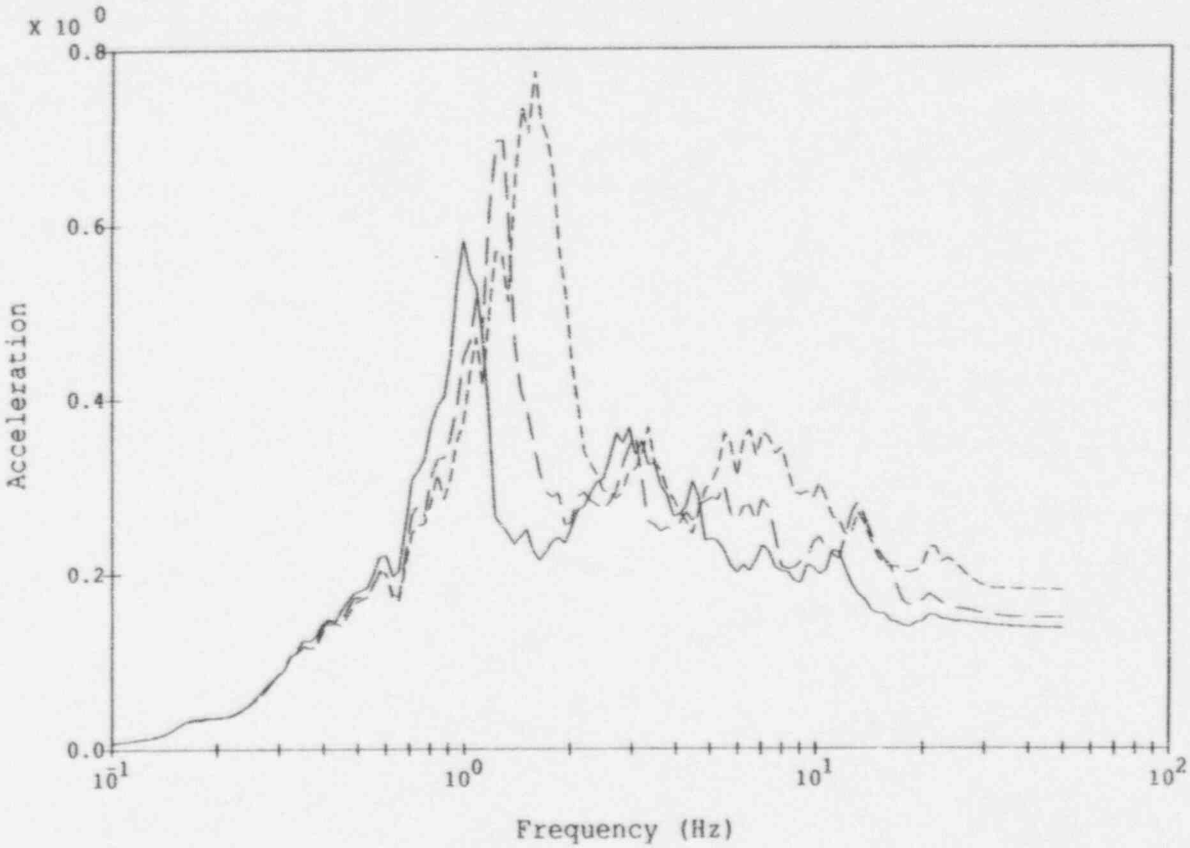
Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 18, Elevation 170.0 ft, North-South Direction



Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 18, Elevation 170.0 ft, Vertical Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

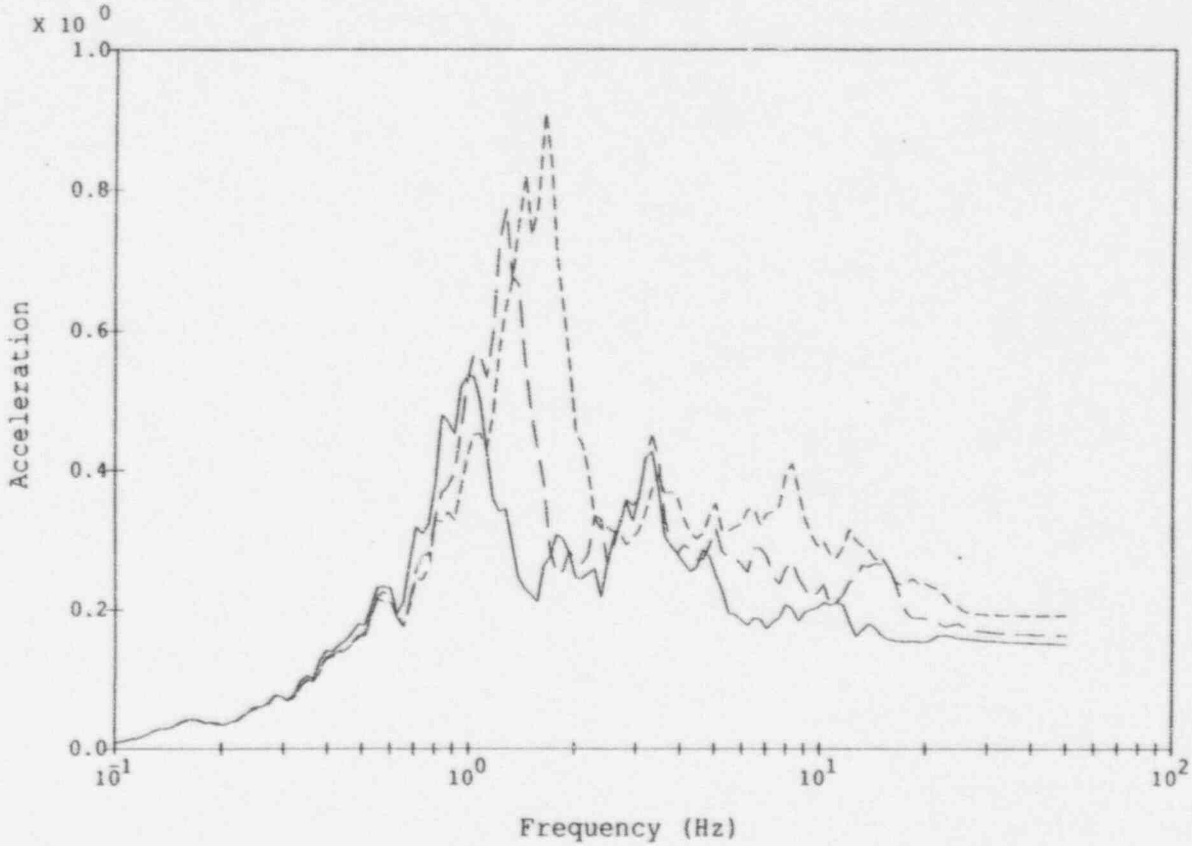
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 19, Elevation 146.0 ft, East-West Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

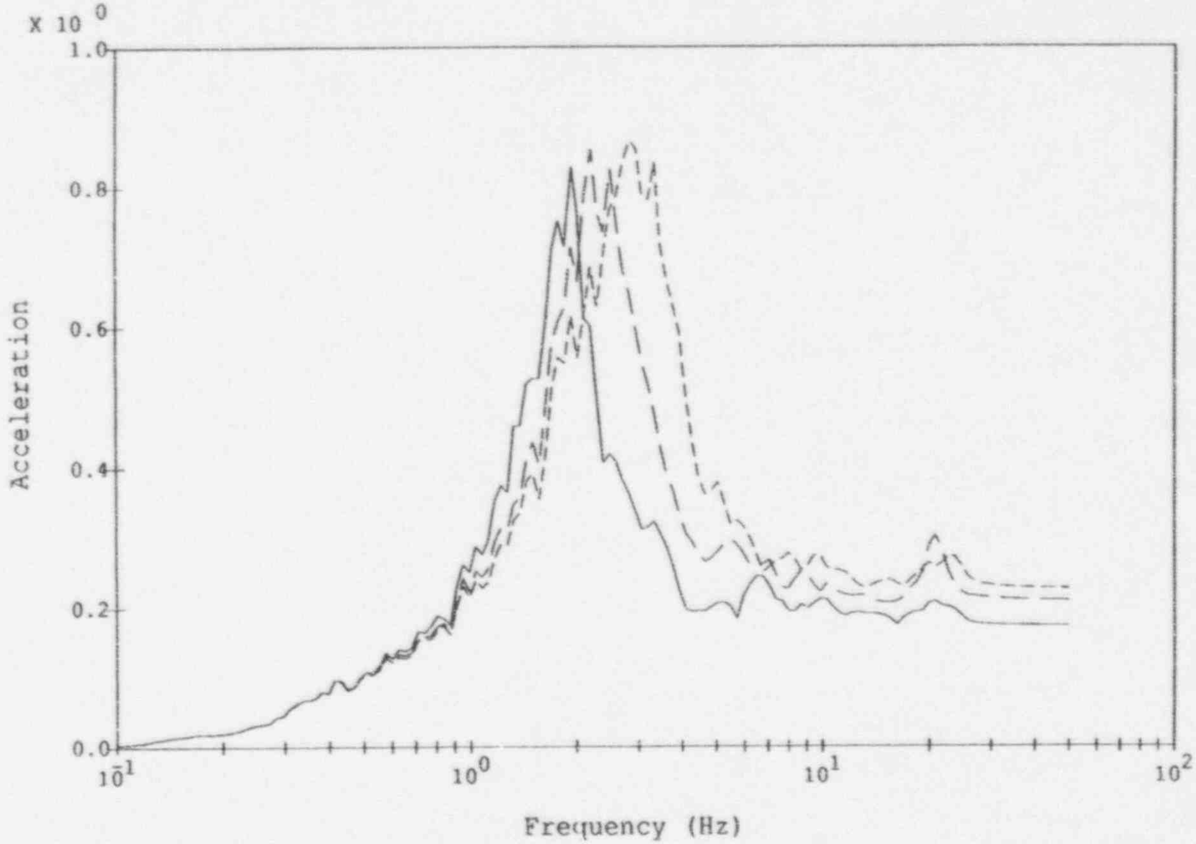
Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 19, Elevation 146.0 ft, North-South Direction



Legend:

Lower Bound

Intermediate

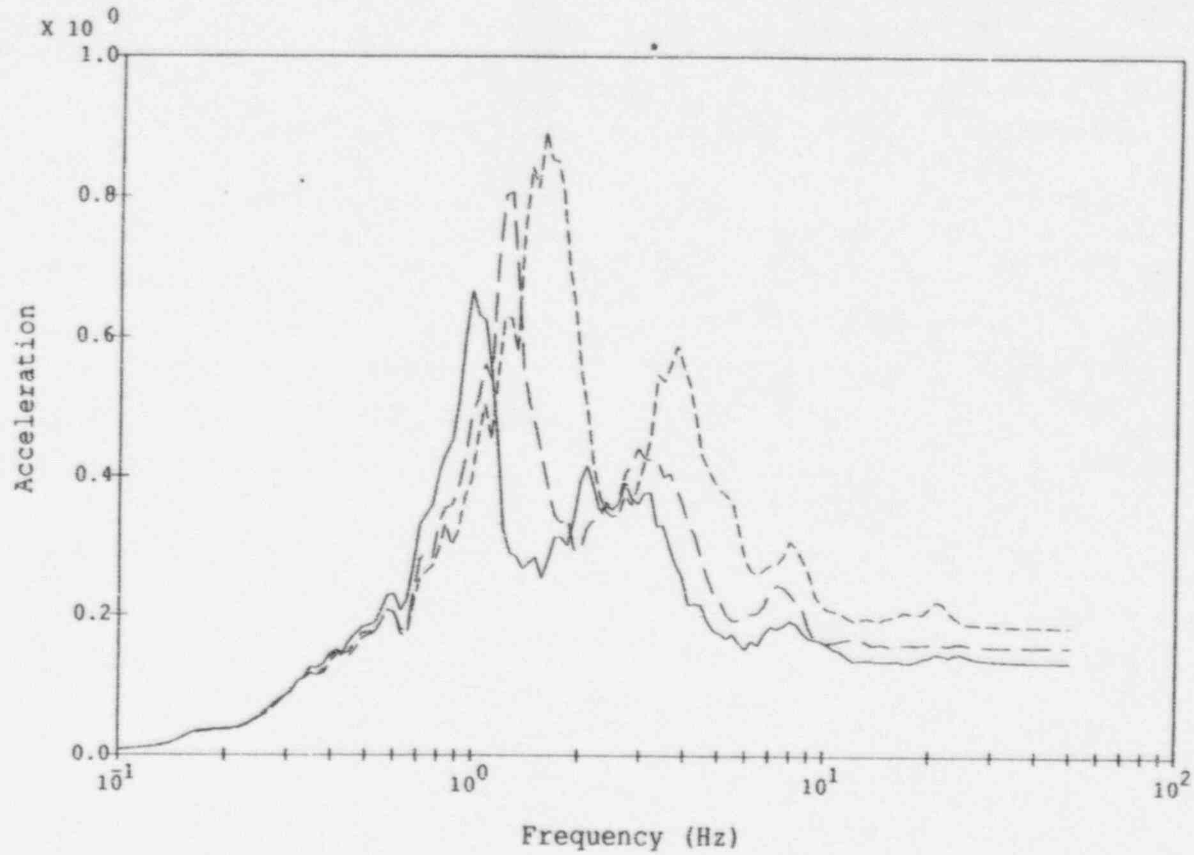
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 19, Elevation 146.0 ft, Vertical Direction



Legend:

Lower Bound

Intermediate

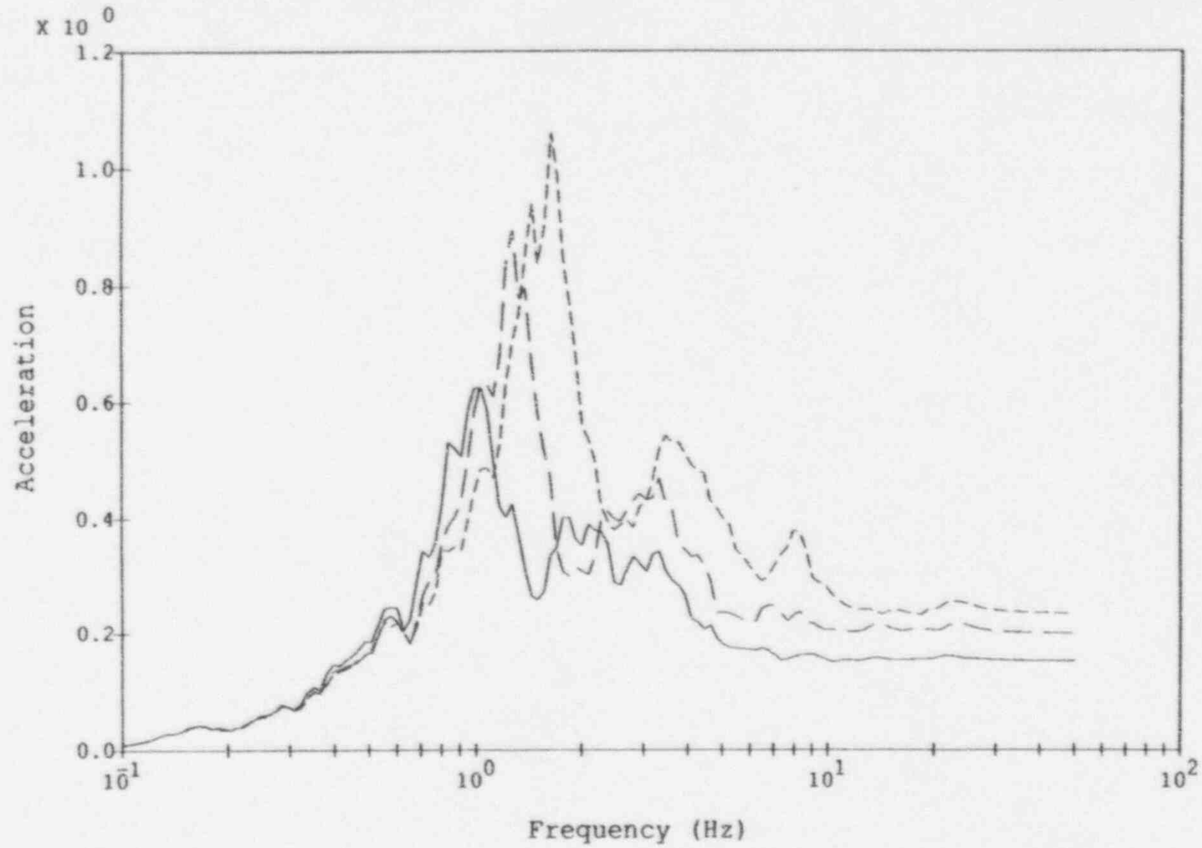
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 20, Elevation 172.0 ft, East-West Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

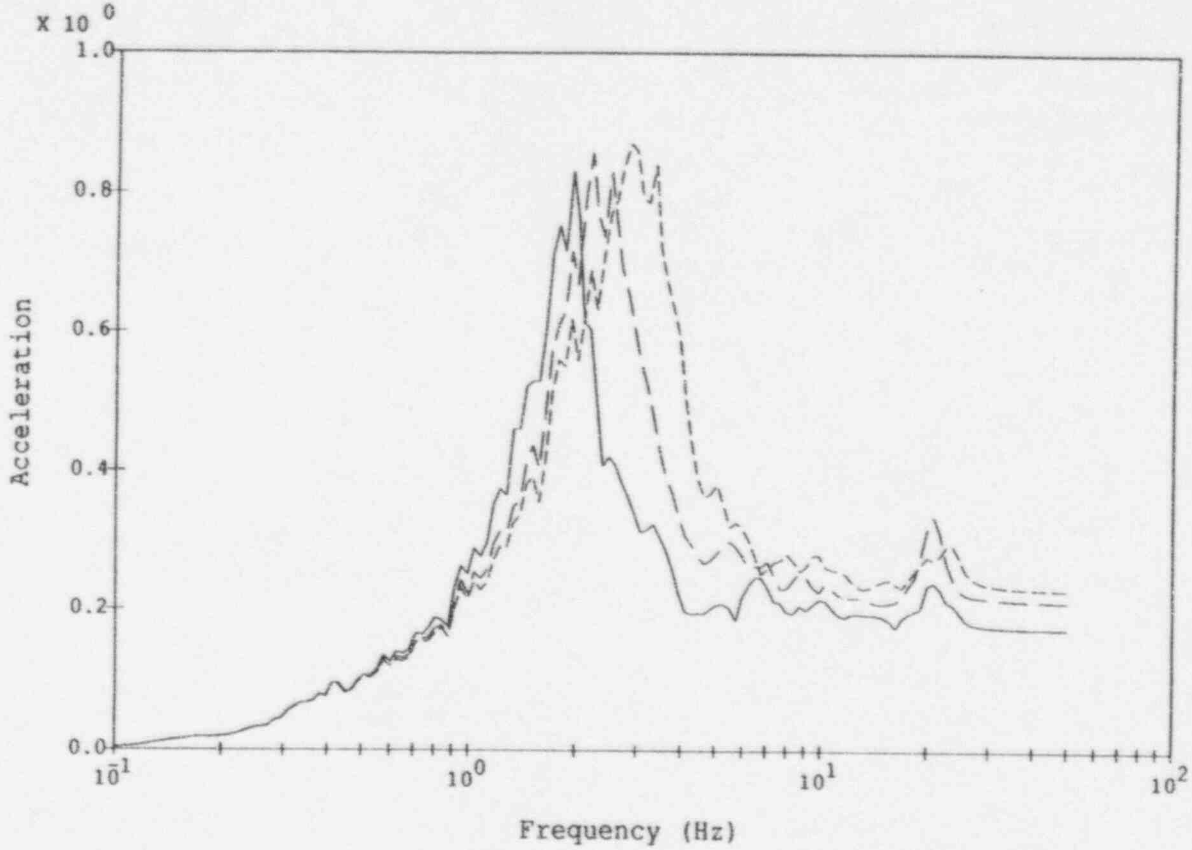
Upper Bound ---------

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 20, Elevation 172.0 ft, North-South Direction



Legend:

Lower Bound

Intermediate

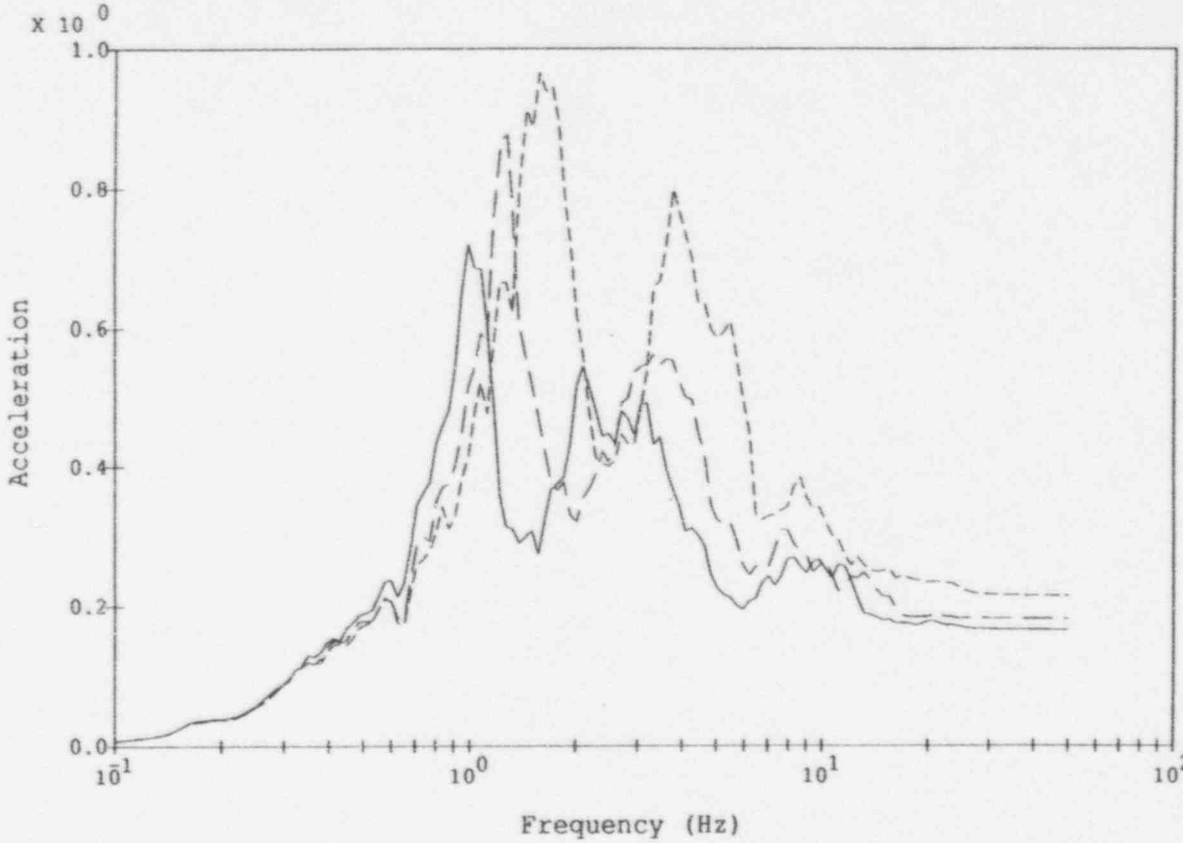
Upper Bound

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 20, Elevation 172.0 ft, Vertical Direction



Legend:

Lower Bound ————

Intermediate — — — —

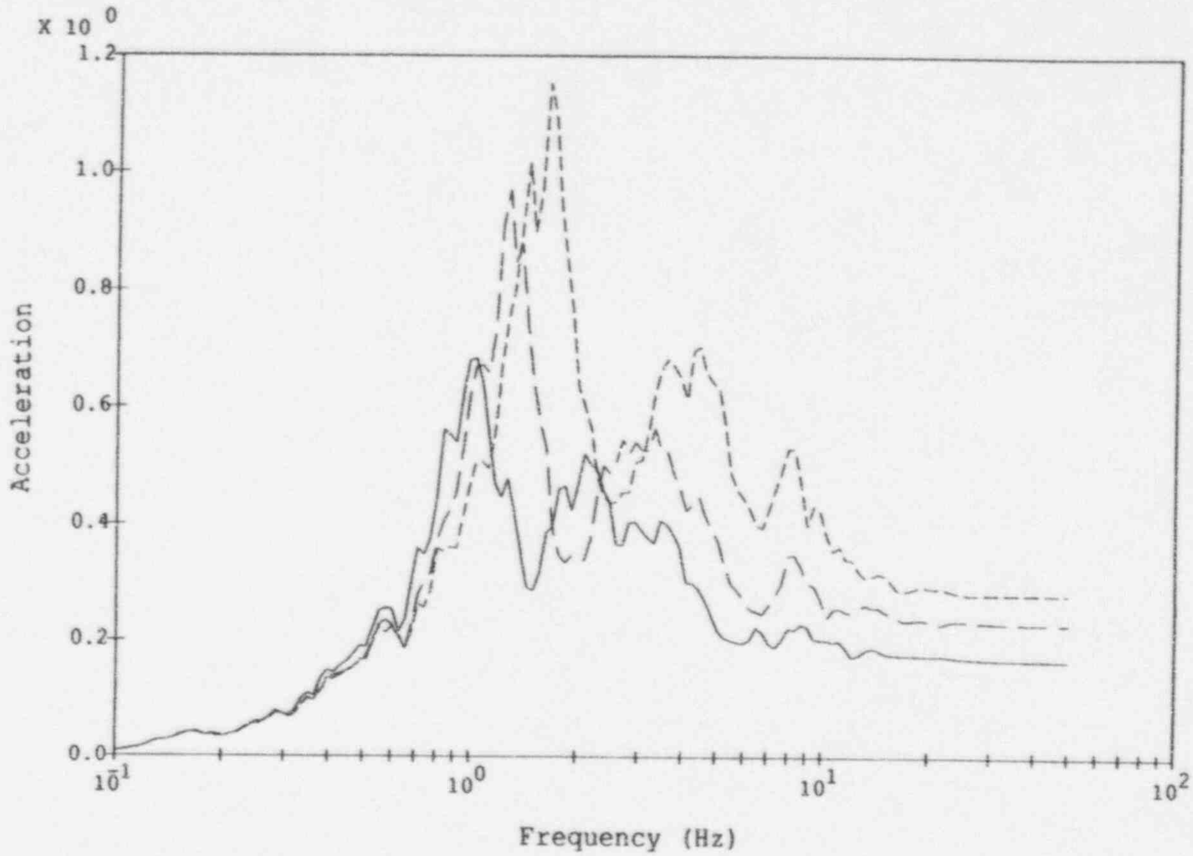
Upper Bound - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 21, Elevation 188.0 ft, East-West Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

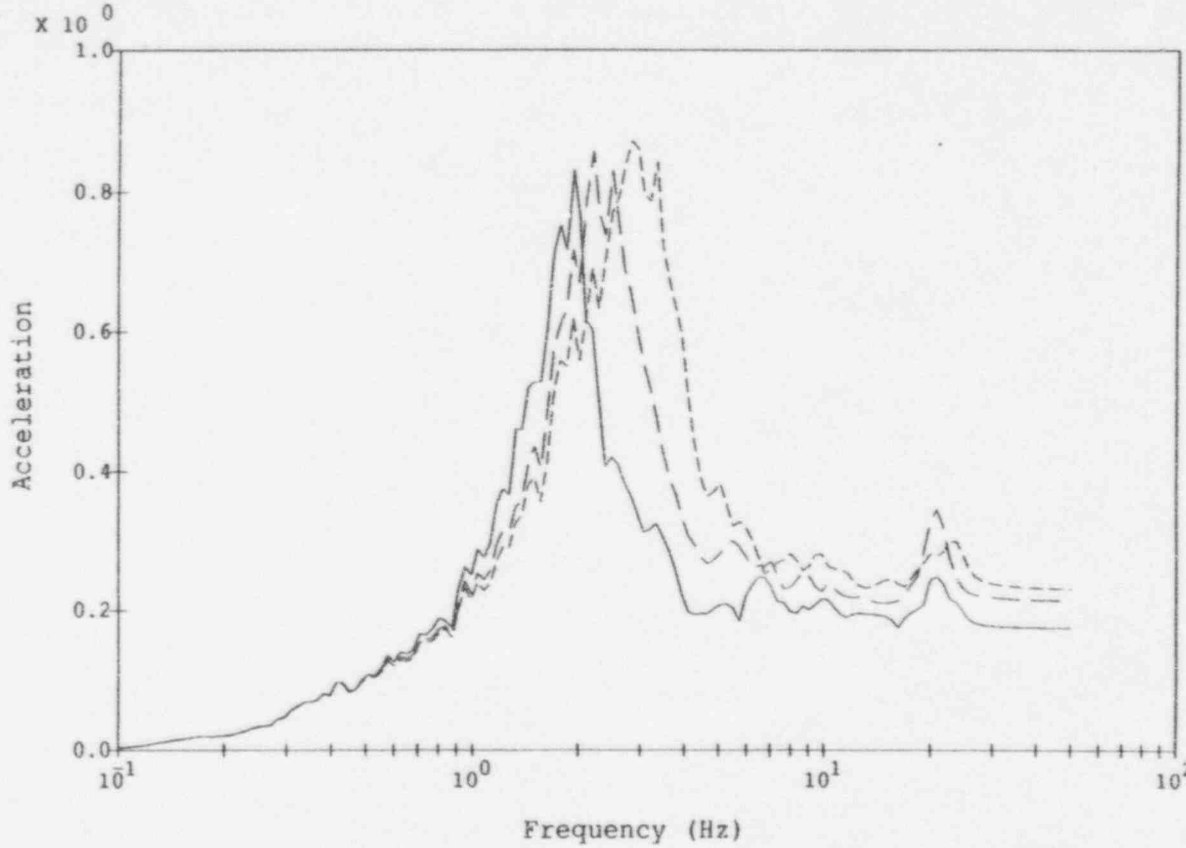
Upper Bound ---------

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 21, Elevation 188.0 ft, North-South Direction



Legend:

Lower Bound ————

Intermediate — — — —

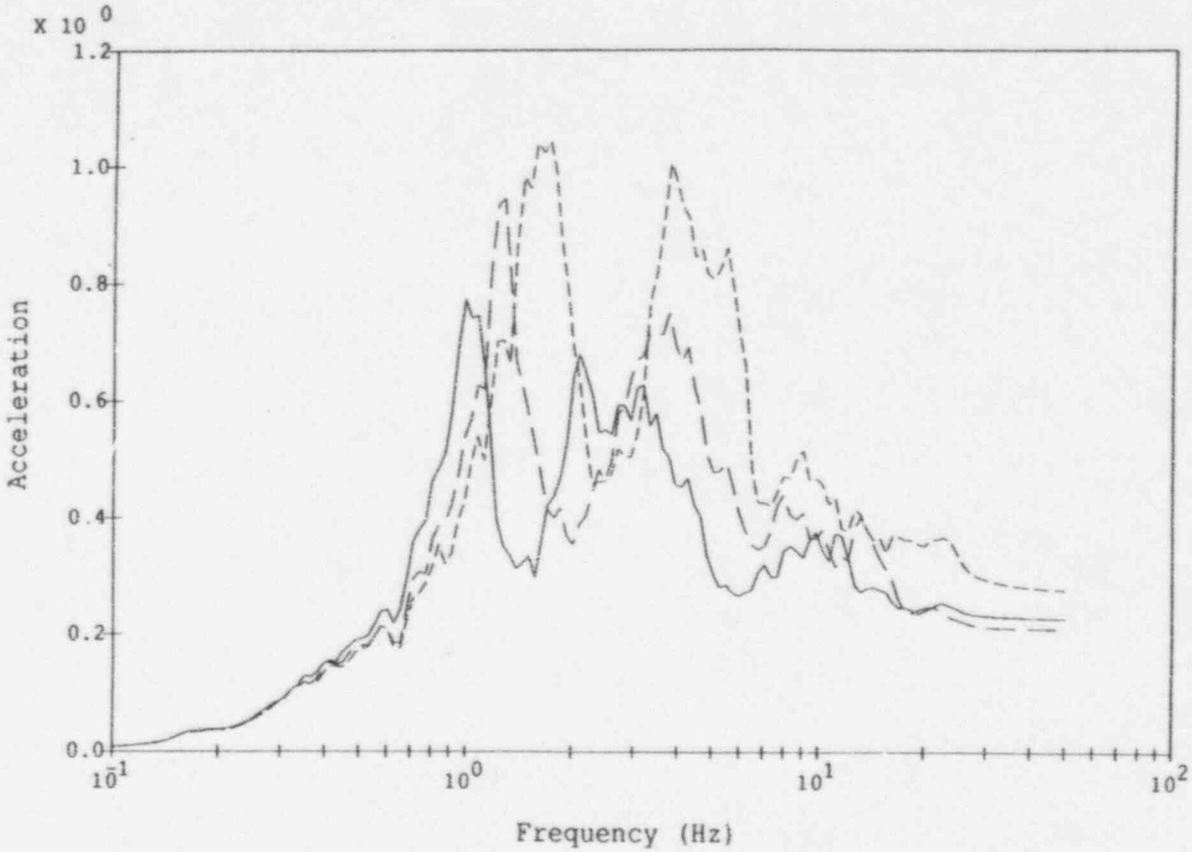
Upper Bound - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 21, Elevation 188.0 ft, Vertical Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

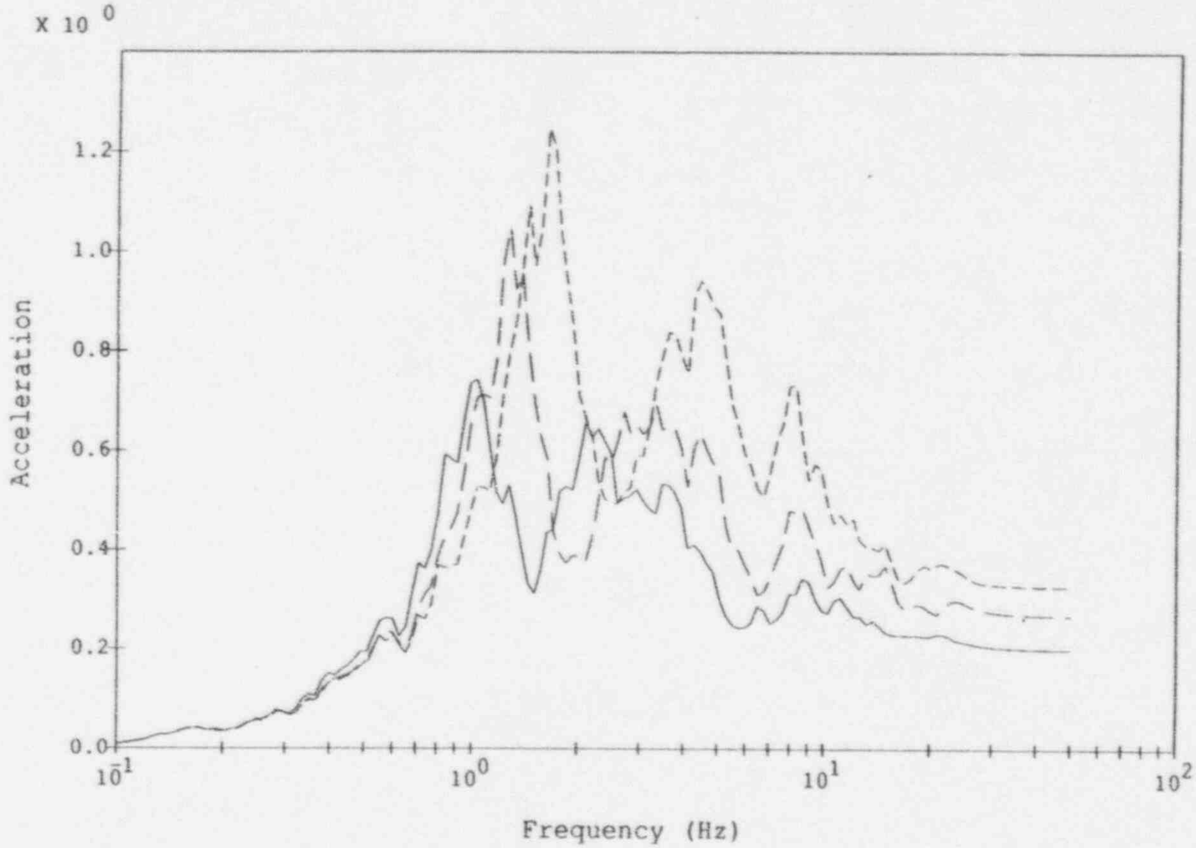
Upper Bound ---------

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 22, Elevation 204.0 ft, East-West Direction



Legend:

Lower Bound ————————

Intermediate — — — — —

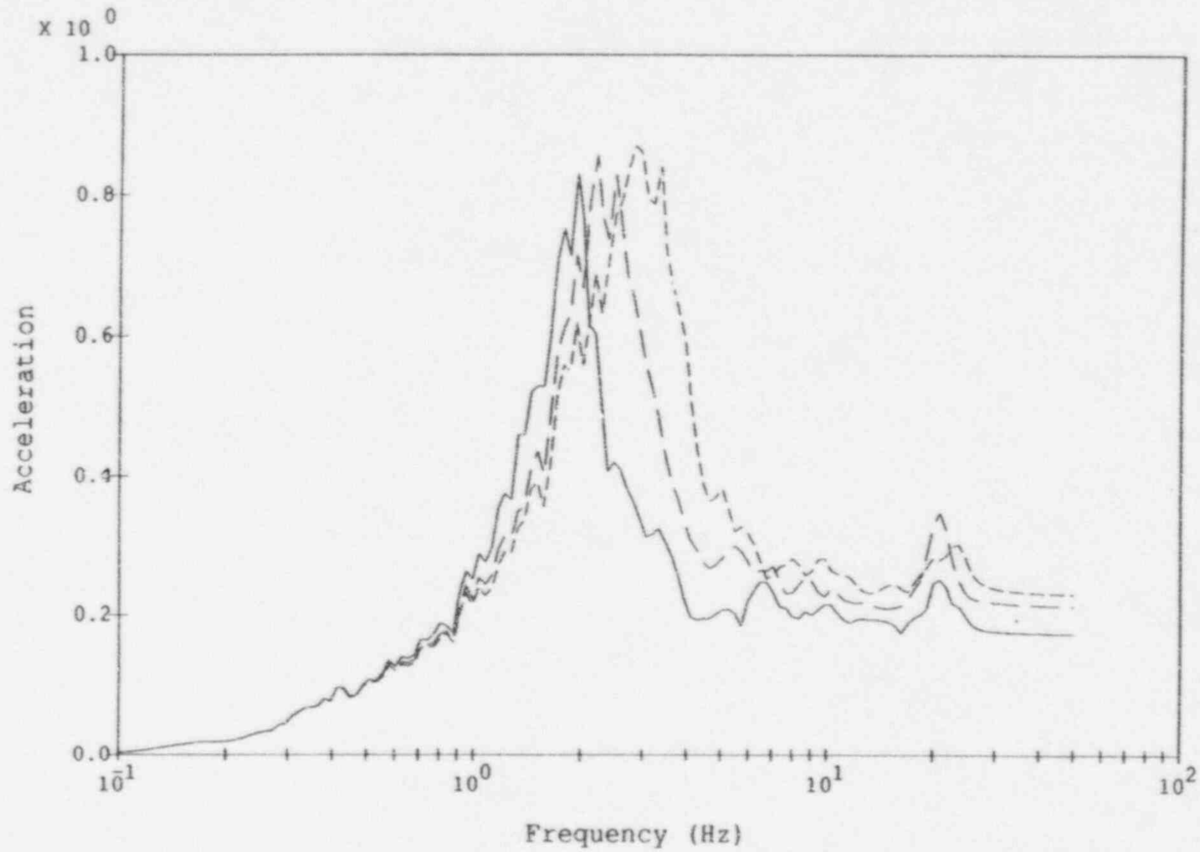
Upper Bound ---------

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 22, Elevation 204.0 ft, North-South Direction



Legend:

Lower Bound ————————

Intermediate — — — — — —

Upper Bound - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Reactor Building SSI Analysis In-Structure Response Spectra
At Mass Point 22, Elevation 204.0 ft, Vertical Direction

PLANT HATCH

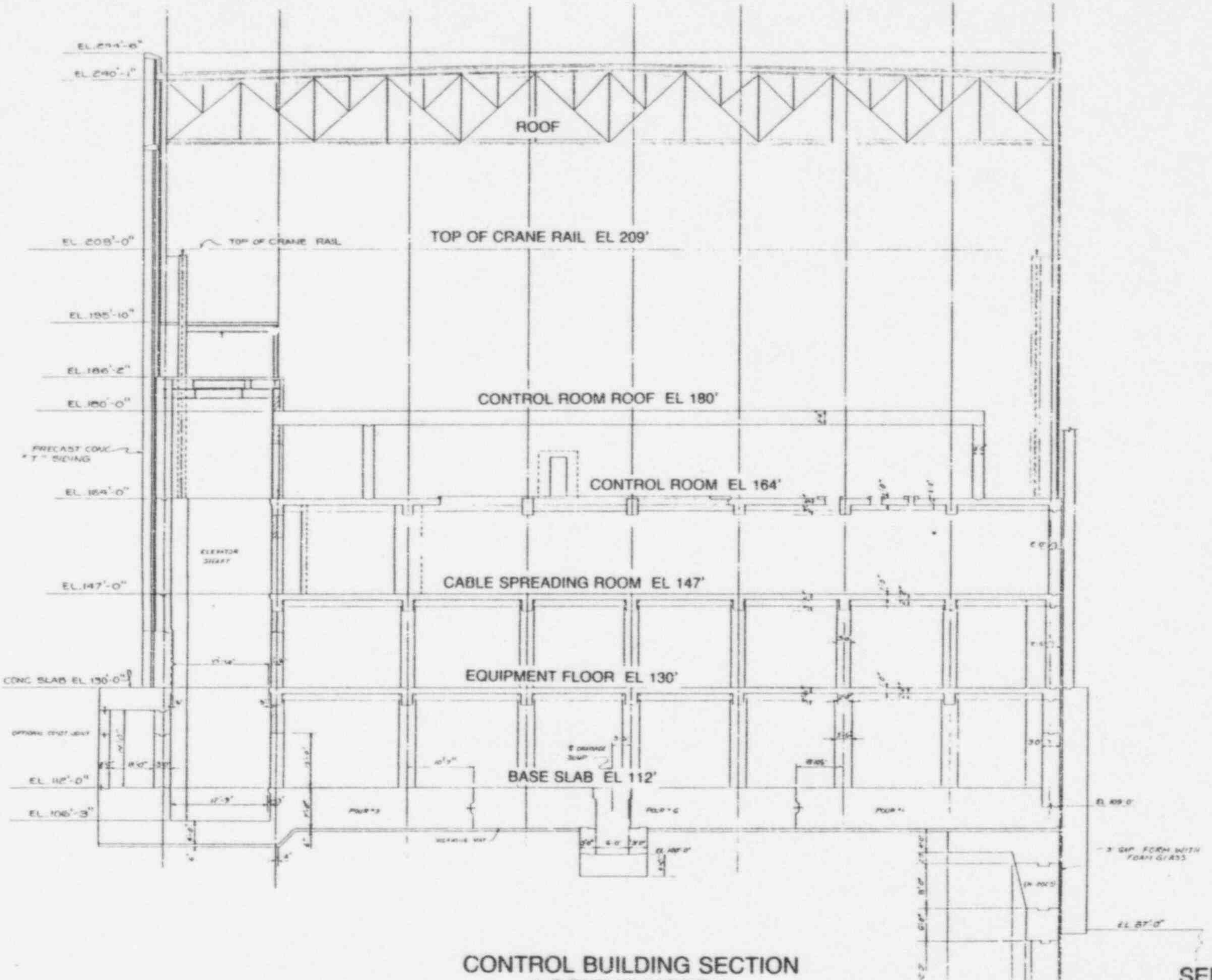
CONTROL BUILDING

SUMMARY OF MAXIMUM ACCELERATIONS AND DISPLACEMENTS

AND

5% DAMPED SME IN-STRUCTURE RESPONSE SPECTRA (1)

(1) Spectra are raw curves that have not been broadened.



CONTROL BUILDING SECTION
LOOKING NORTH

SMA
SEISMIC MODEL
MASS POINTS

CONTROL BUILDING

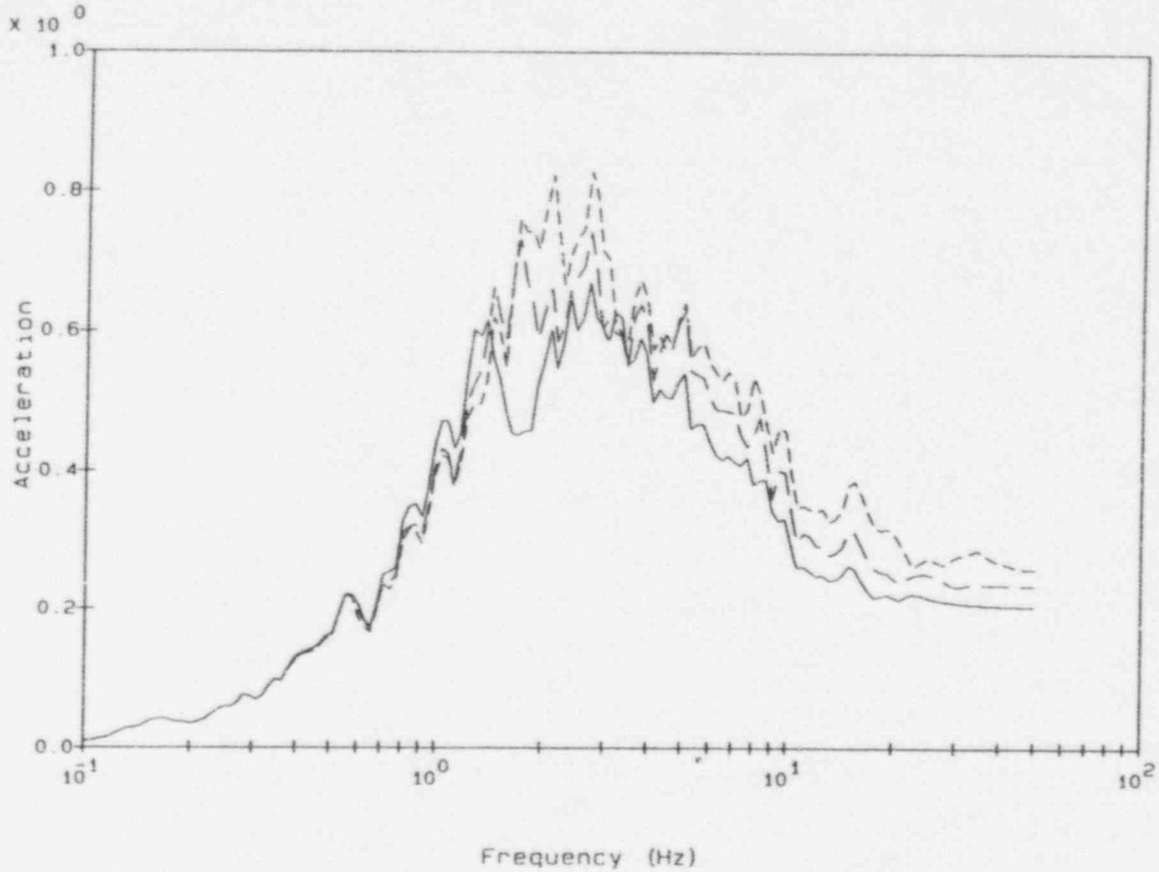
SUMMARY OF MAXIMUM ABSOLUTE ACCELERATIONS (g)

	Lower bound soil properties	Intermediate soil properties	Upper bound soil properties
Node 2			
N-S dir	0.204	0.234	0.257
E-W dir	0.206	0.223	0.250
Vert.	0.164	0.183	0.186
Node 5			
N-S dir	0.192	0.237	0.273
E-W dir	0.200	0.242	0.287
Vert.	0.172	0.187	0.195
Node 8			
N-S dir	0.185	0.243	0.279
E-W dir	0.194	0.261	0.312
Vert.	0.177	0.176	0.201
Node 11			
N-S dir	0.204	0.257	0.312
E-W dir	0.204	0.280	0.333
Vert.	0.179	0.176	0.204
Node 15			
N-S dir	0.228	0.286	0.365
E-W dir	0.222	0.298	0.349
Vert.	0.185	0.184	0.211
Node 16			
N-S dir	0.400	0.604	0.622
E-W dir	0.501	0.595	0.530
Vert.	0.207	0.196	0.215
Node 18			
N-S dir	0.718	1.139	1.365
E-W dir	0.358	0.449	0.404
Vert.	0.227	0.237	0.250

CONTROL BUILDING

SUMMARY OF MAXIMUM RELATIVE DISPLACEMENT (in.)

	Lower bound soil properties	Intermediate soil properties	Upper bound soil properties
Node 2			
N-S dir	0.000	0.000	0.000
E-W dir	0.000	0.000	0.000
Vert.	0.000	0.000	0.000
Node 5			
N-S dir	0.011	0.015	0.017
E-W dir	0.010	0.013	0.016
Vert.	0.002	0.002	0.002
Node 8			
N-S dir	0.021	0.027	0.031
E-W dir	0.015	0.020	0.024
Vert.	0.003	0.003	0.003
Node 11			
N-S dir	0.028	0.034	0.041
E-W dir	0.019	0.026	0.030
Vert.	0.004	0.004	0.004
Node 15			
N-S dir	0.037	0.044	0.055
E-W dir	0.023	0.030	0.035
Vert.	0.005	0.004	0.005
Node 16			
N-S dir	0.642	1.037	1.190
E-W dir	1.211	1.389	0.270
Vert.	0.012	0.012	0.013
Node 18			
N-S dir	1.400	2.238	2.627
E-W dir	2.000	1.949	1.852
Vert.	0.018	0.016	0.018



Legend:

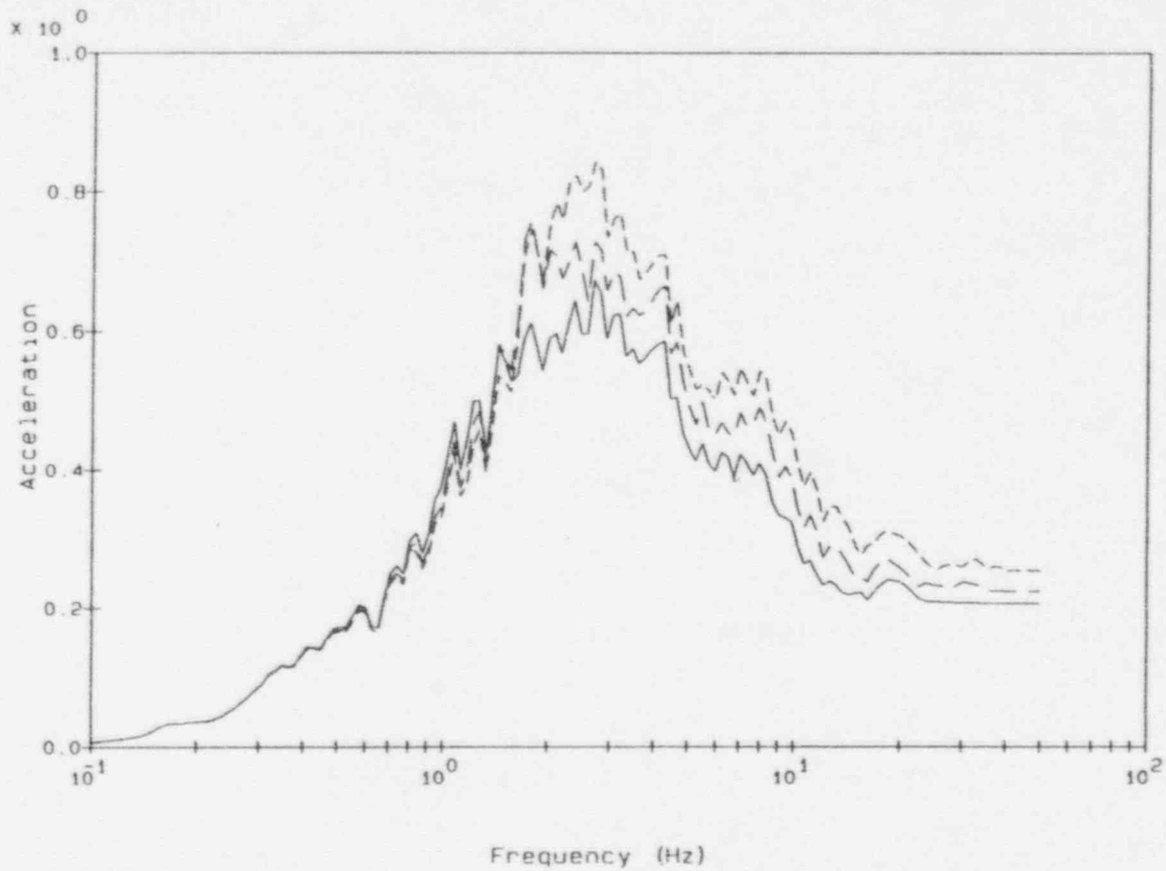
0.6 x Soil Modulus	————————
1.0 x Soil Modulus	— — — — -
1.6 x Soil Modulus	- - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Node 2. North-South Response   CONTROL BUILDING el 112 ft



Legend:

0.6 x Soil Modulus ————————

1.0 x Soil Modulus — — — — .

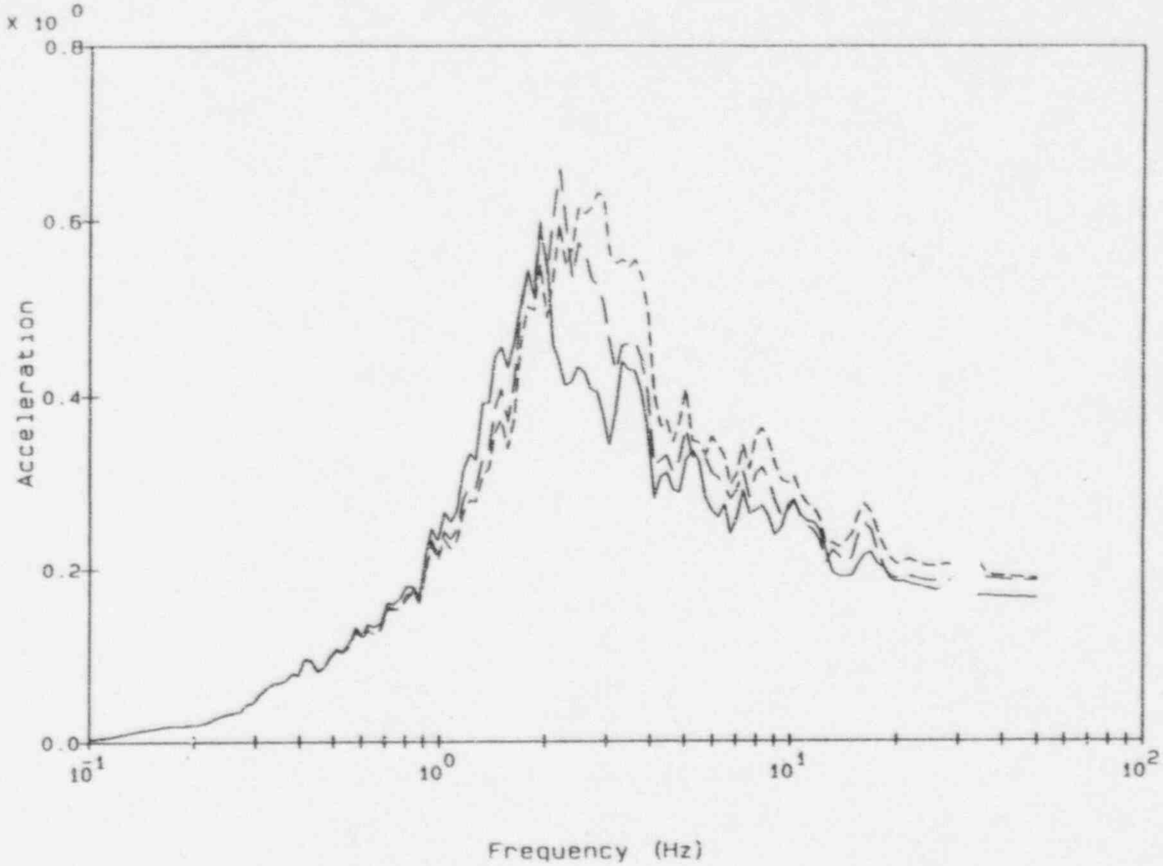
1.6 x Soil Modulus - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Node 2, East-West Response CONTROL BUILDING el 112 ft



Legend:

0.6 x Soil Modulus

1.0 x Soil Modulus

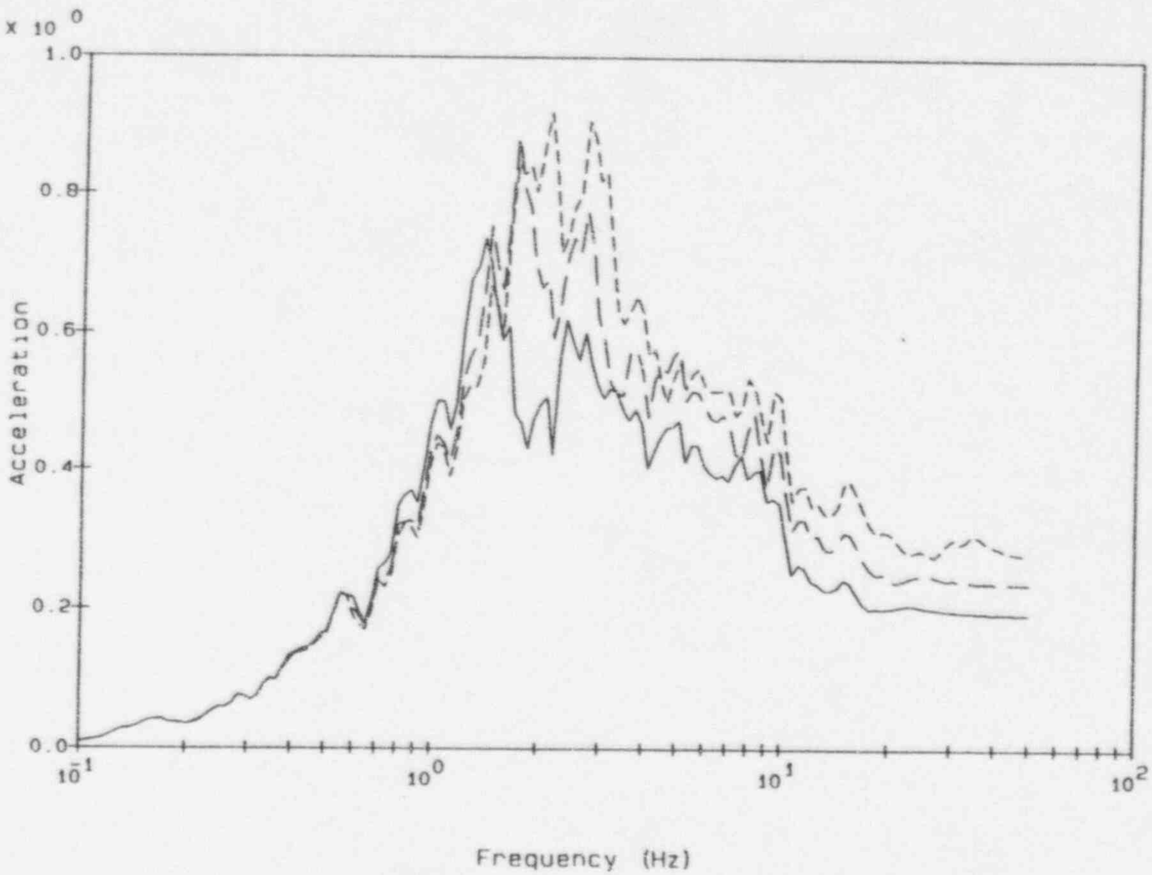
1.6 x Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

Node 2, Vertical Response CONTROL BUILDING el 112 ft



Legend:

0.6 x Soil Modulus

1.0 x Soil Modulus

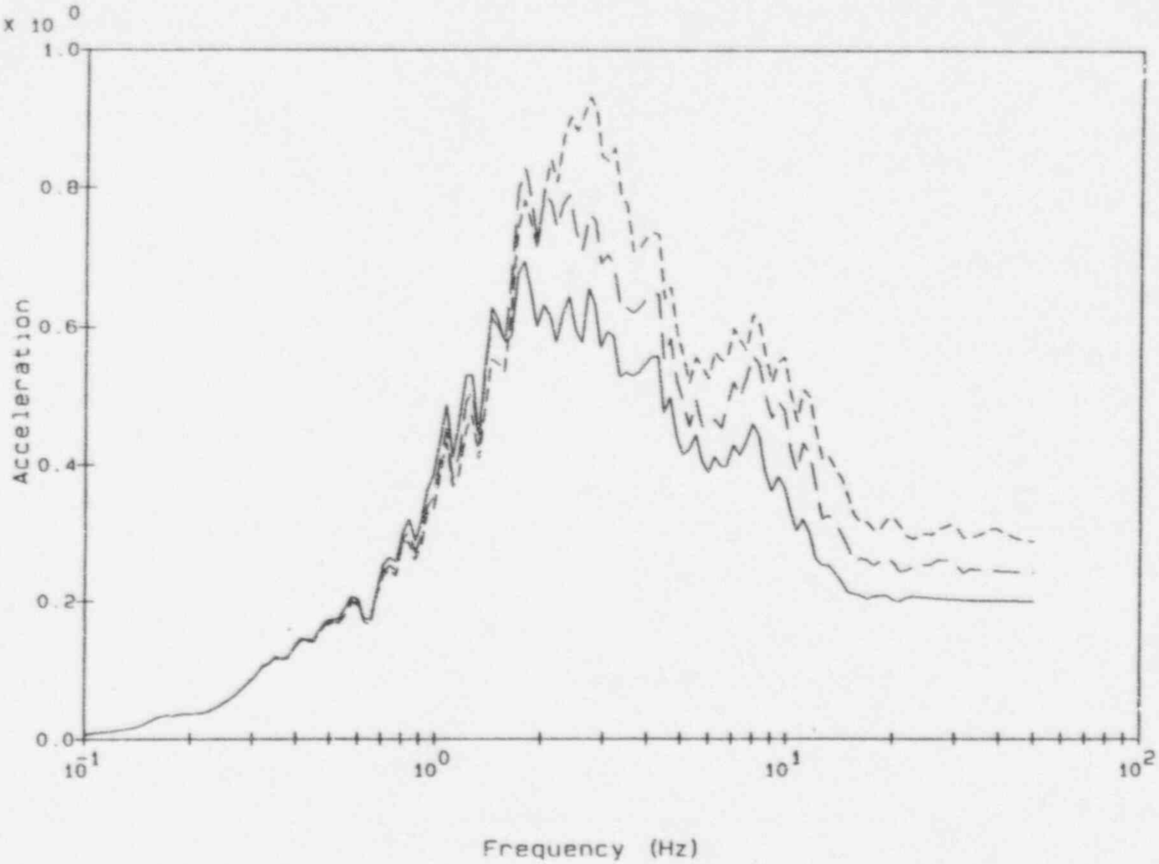
1.6 x Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

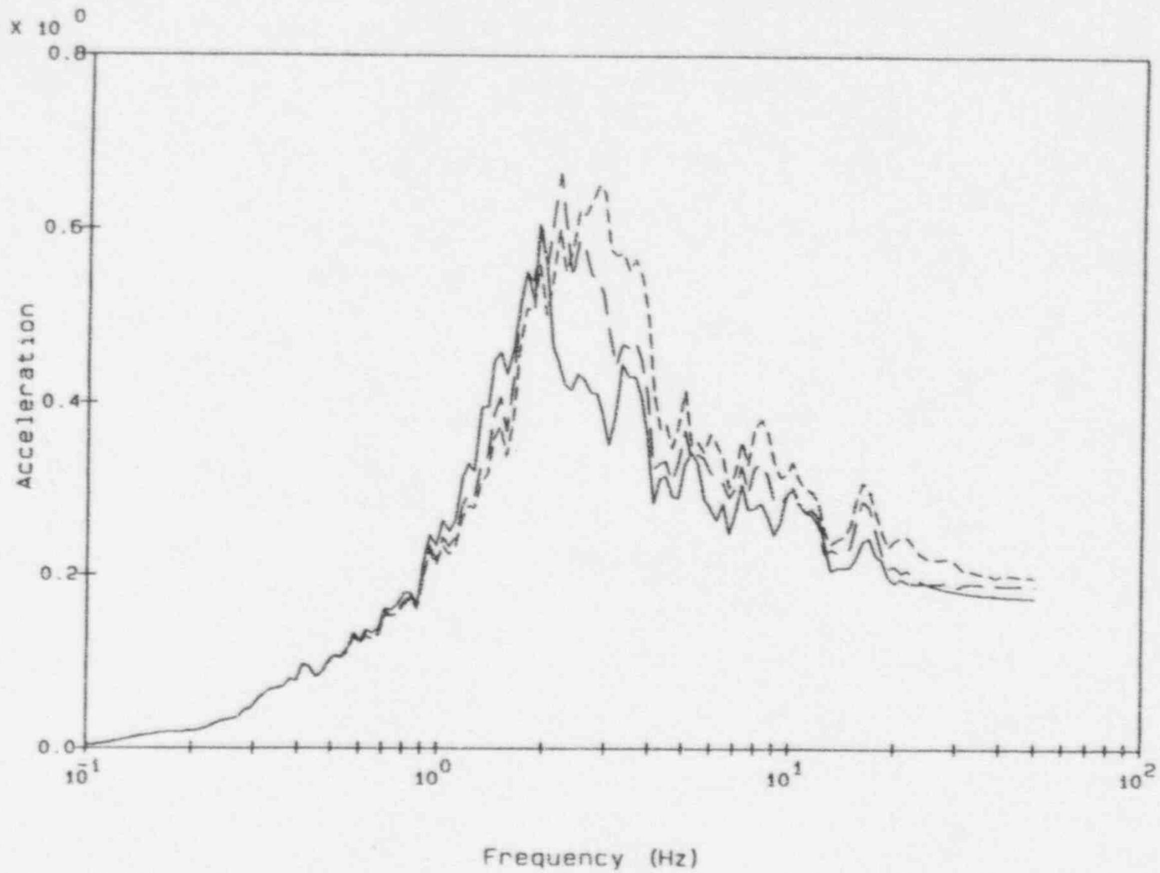
Node 5, North-South Response CONTROL BUILDING el 130 ft



Legend:
0.6 x Soil Modulus
1.0 x Soil Modulus
1.6 x Soil Modulus

Notes:
Accelerations in g's
5% Spectral Damping

Node 5. East-West Response CONTROL BUILDING el 130 ft



Legend:

0.6 x Soil Modulus

1.0 x Soil Modulus

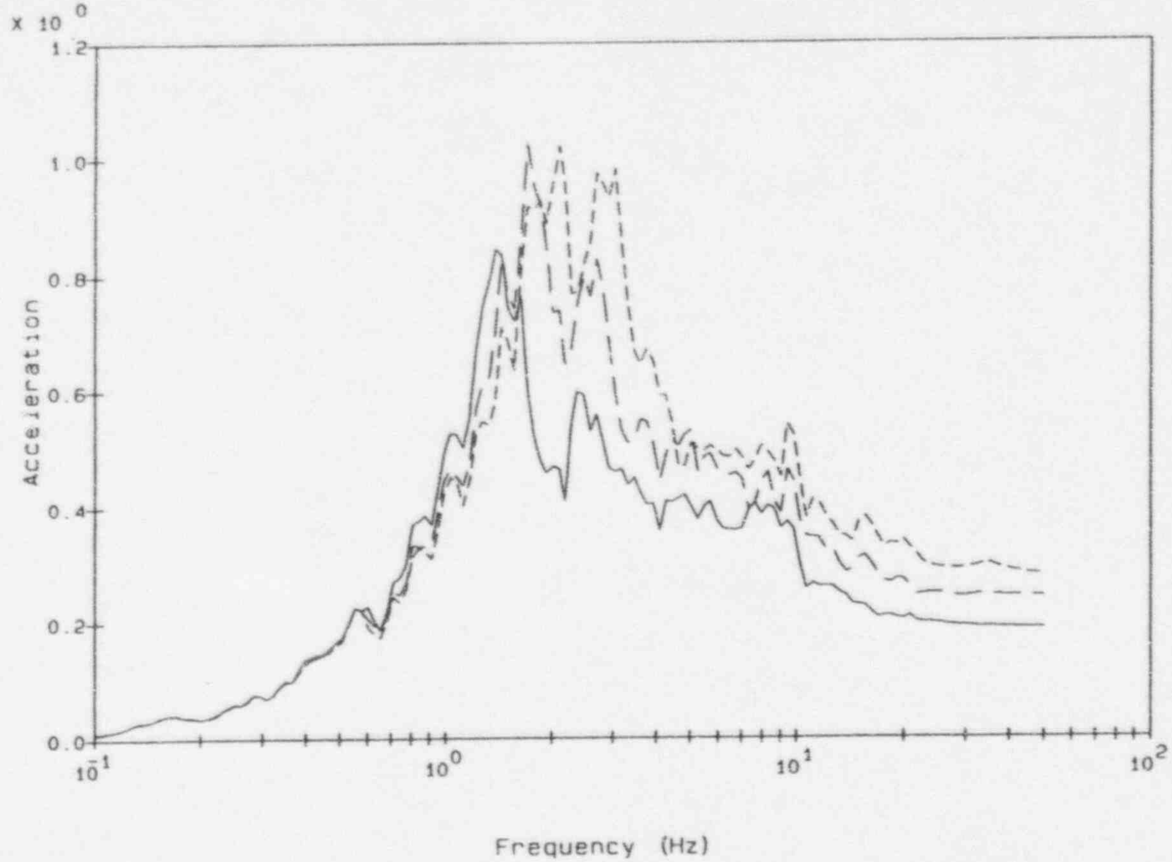
1.6 x Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

Node 5. Vertical Response CONTROL BUILDING el 130 ft



Legend:

0.6 x Soil Modulus ————

1.0 x Soil Modulus — — — —

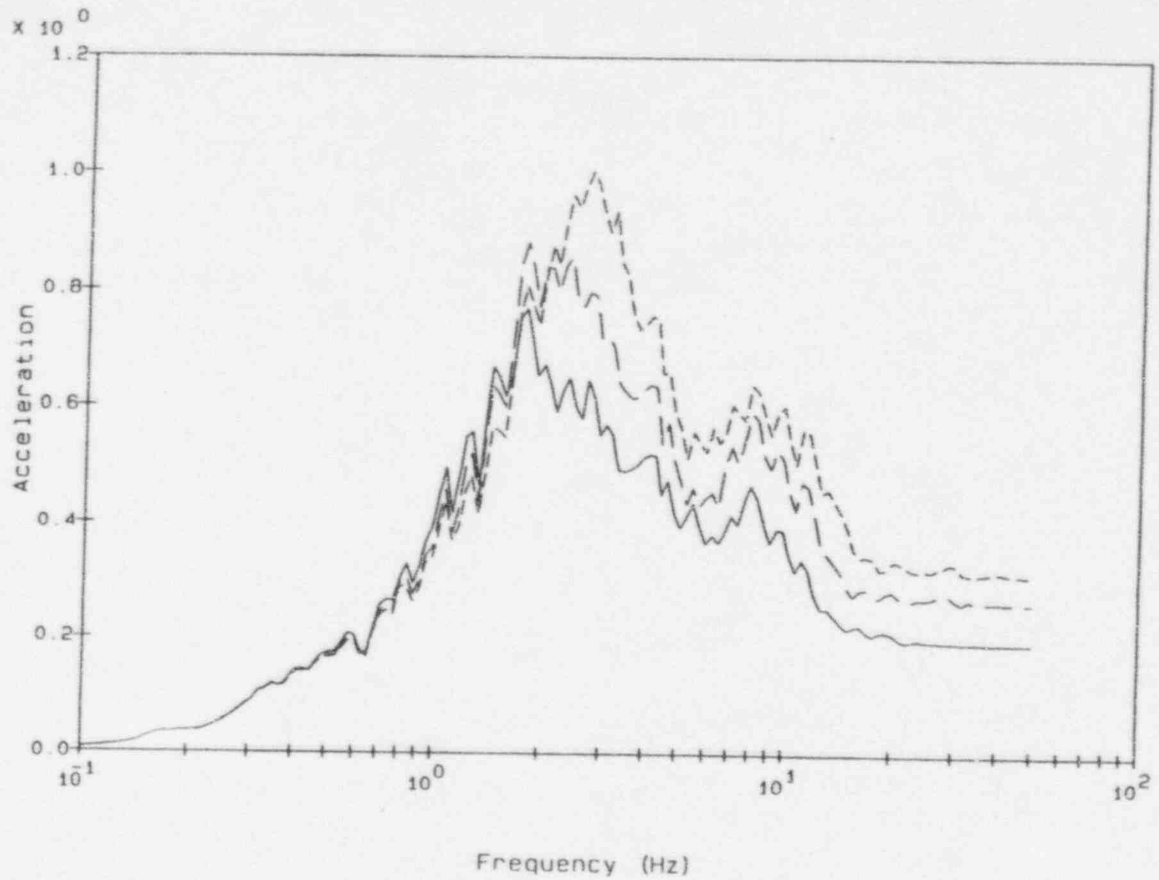
1.6 x Soil Modulus - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Node 8, North-South Response CONTROL BUILDING el 147 ft



Legend:

0.6 x Soil Modulus ————

1.0 x Soil Modulus — — — —

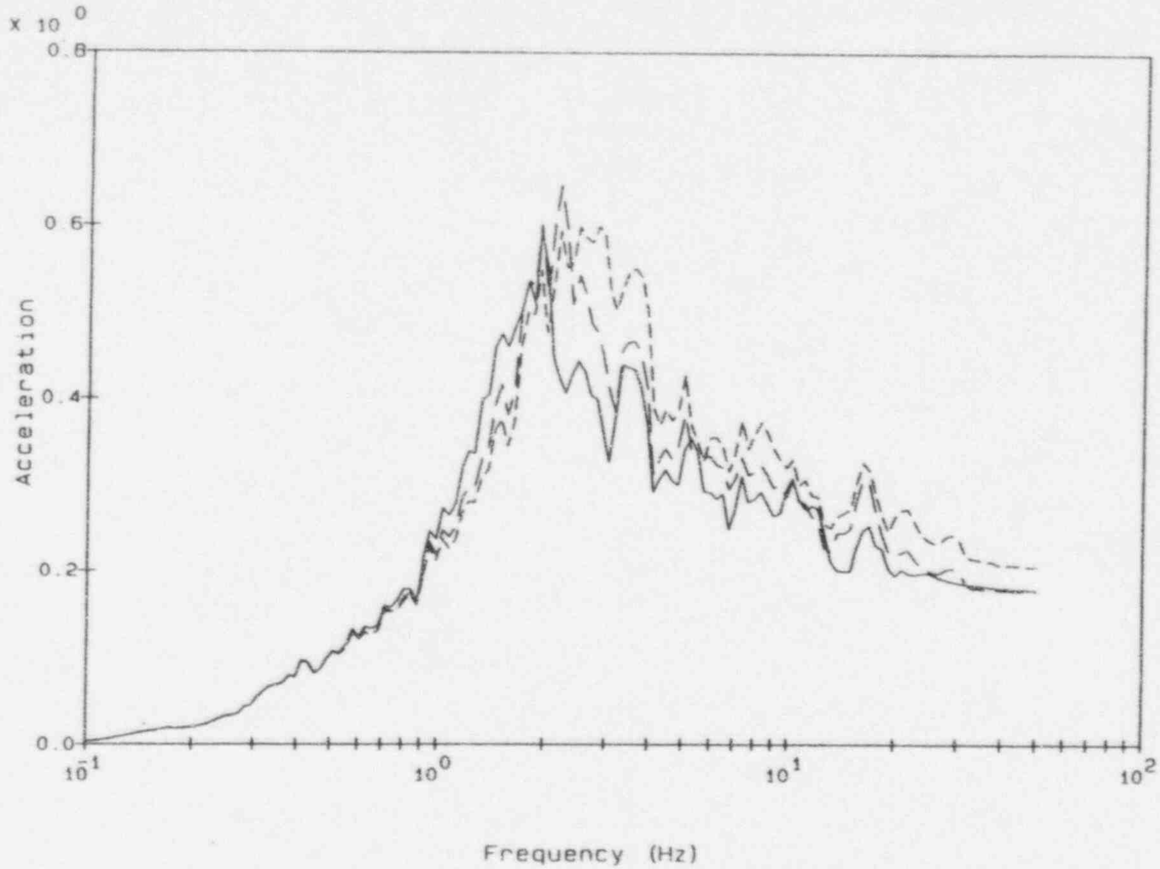
1.6 x Soil Modulus - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Node 8. East-West Response   CONTROL BUILDING el 147 ft



Legend:

0.6 x Soil Modulus ————————

1.0 x Soil Modulus — — — — —

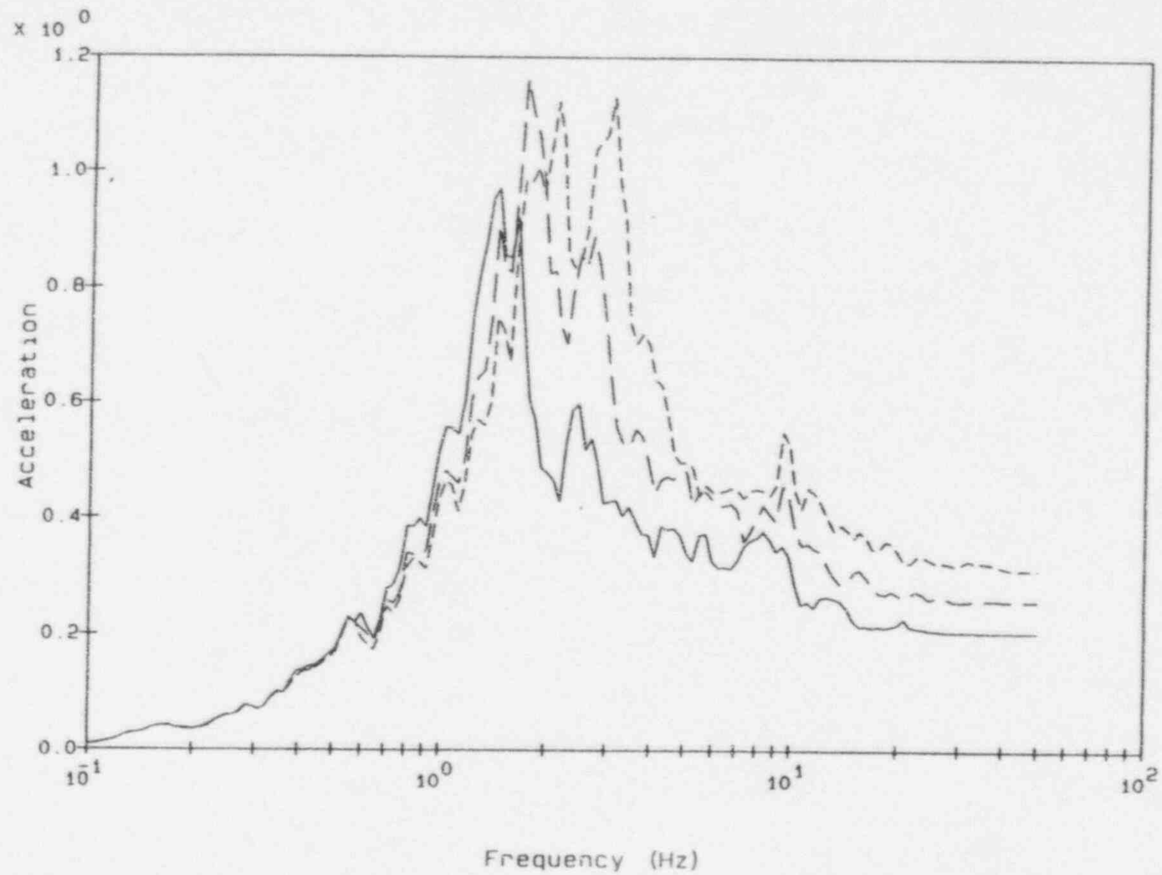
1.6 x Soil Modulus - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Node 8, Vertical Response CONTROL BUILDING el 147 ft



Legend:

0.6 x Soil Modulus ————

1.0 x Soil Modulus — — — —

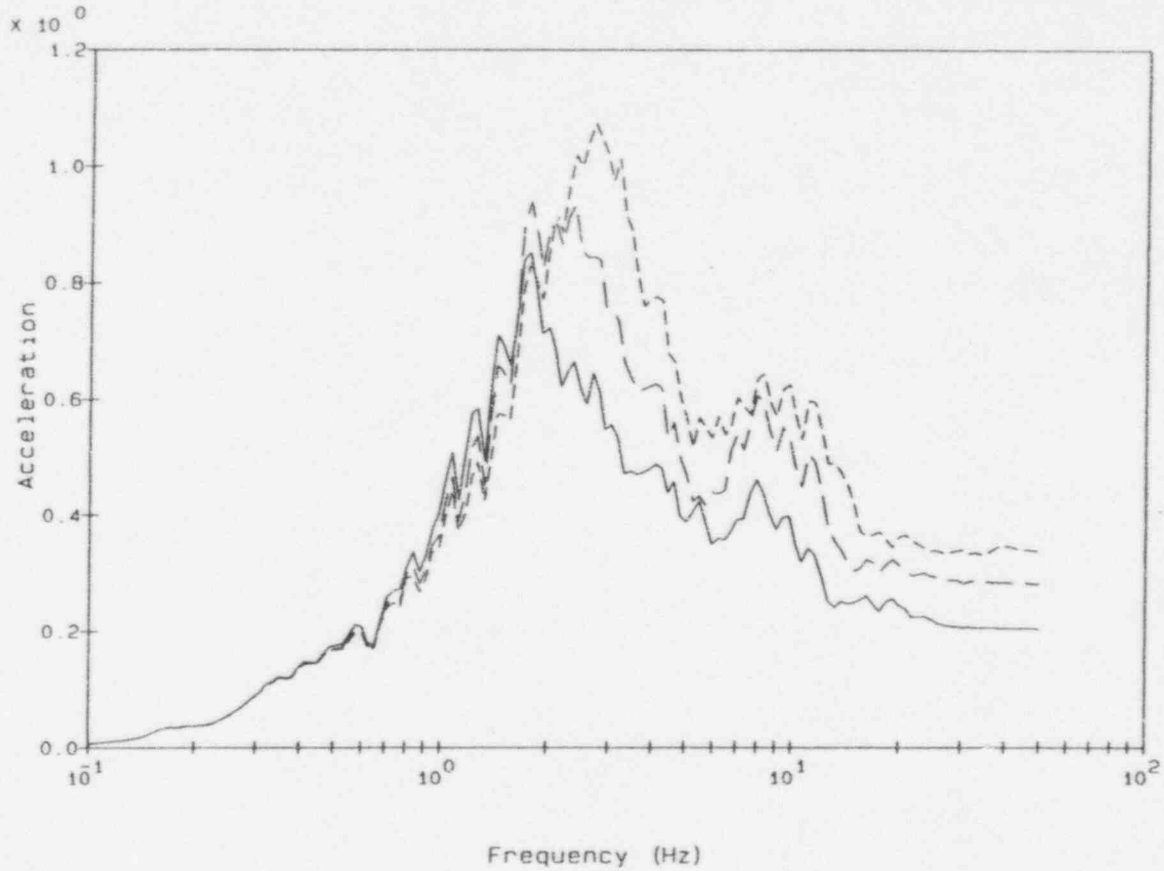
1.6 x Soil Modulus - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Node 11. North-South Response CONTROL BUILDING el 164 ft



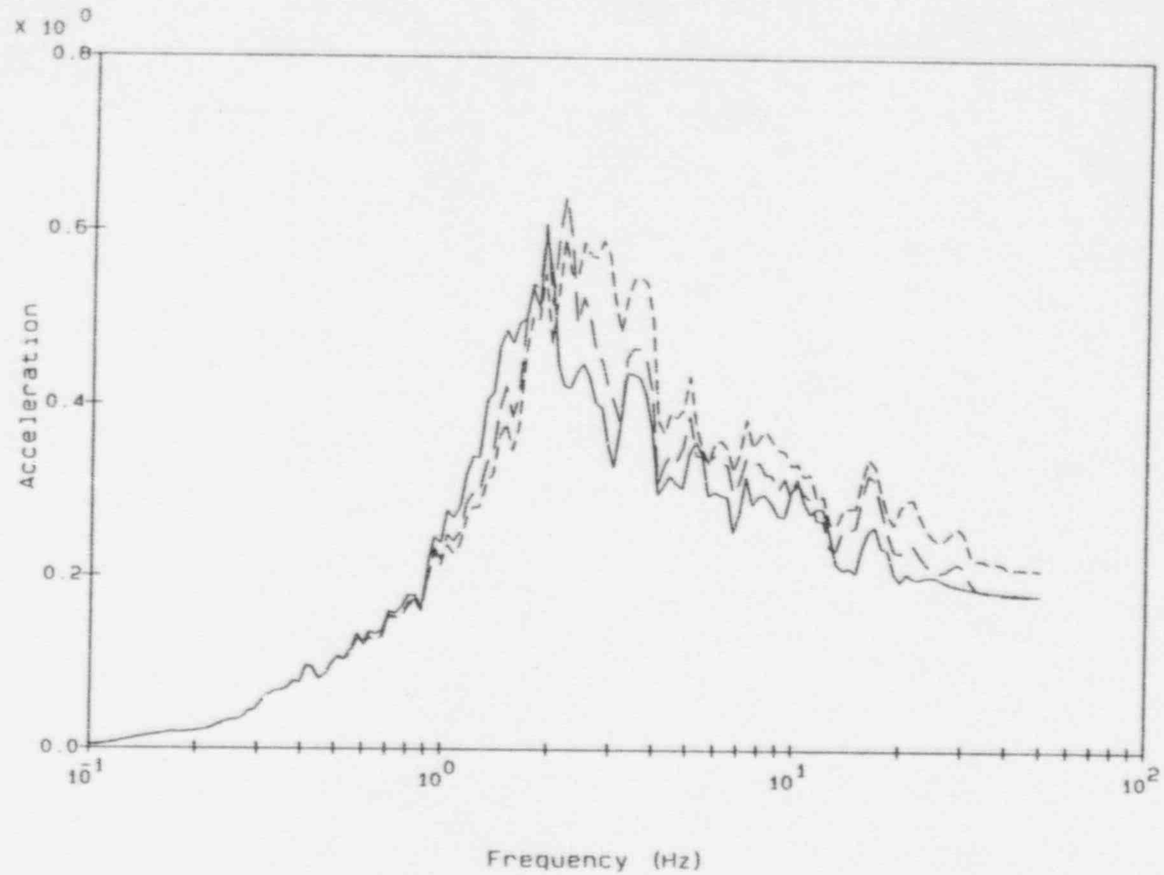
Legend:

0.6 x Soil Modulus ————————
1.0 x Soil Modulus — — — — —
1.6 x Soil Modulus - - - - - - - - -

Notes:

Accelerations in g's
5% Spectral Damping

Node 11. East-West Response CONTROL BUILDING el 164 ft



Legend:

0.6 x Soil Modulus ————

1.0 x Soil Modulus — — — —

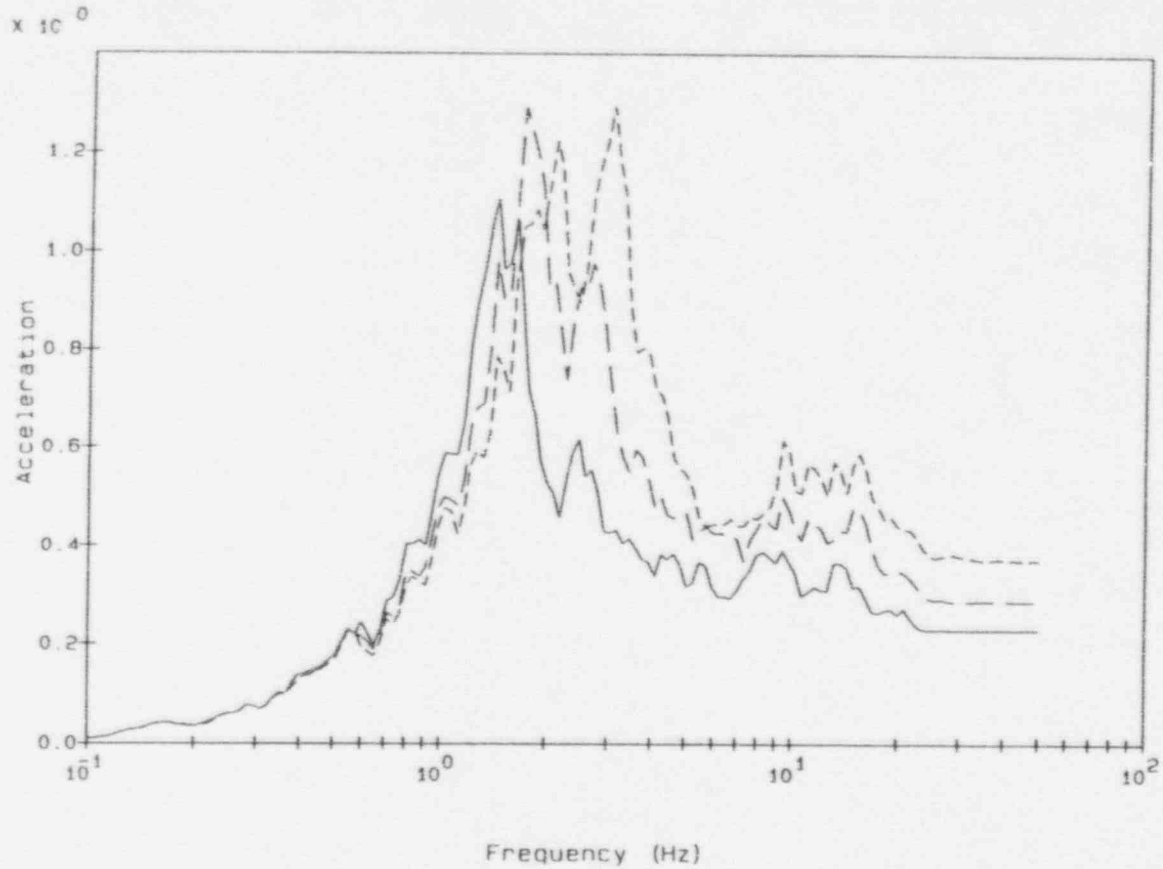
1.6 x Soil Modulus - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Node 11. Vertical Response CONTROL BUILDING el 164 ft



Legend:

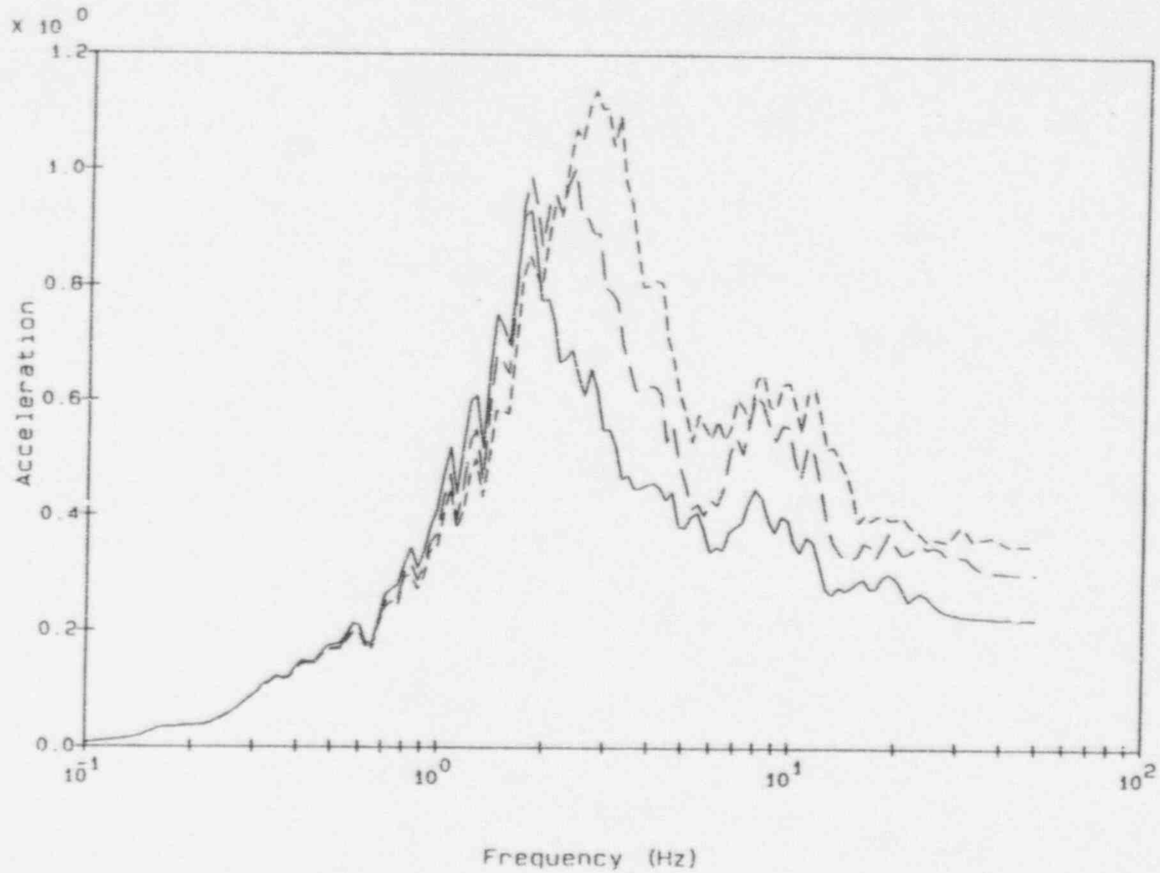
0.6 x Soil Modulus	————————
1.0 x Soil Modulus	— — — — —
1.6 x Soil Modulus	- - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Node 15, North-South Response   CONTROL BUILDING el 180 ft



Legend:

0.6 x Soil Modulus ————————

1.0 x Soil Modulus — — — — —

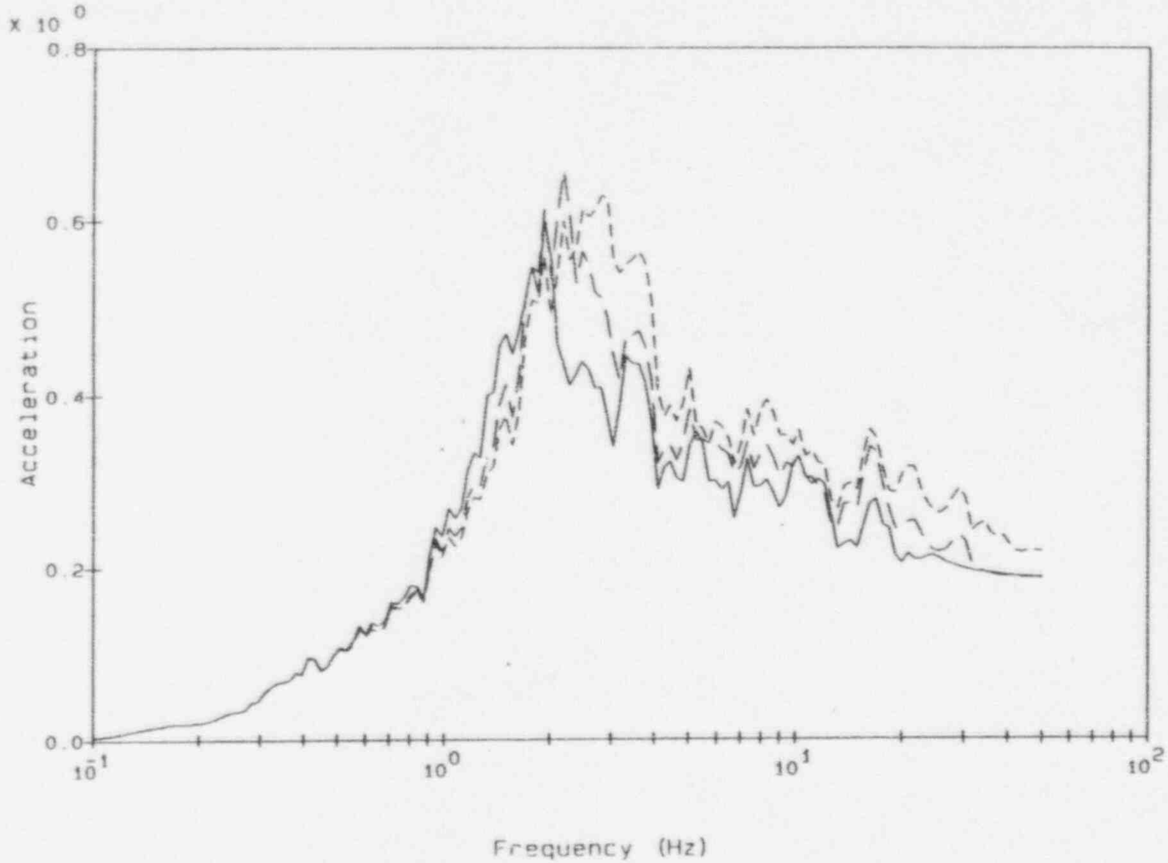
1.6 x Soil Modulus - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Node 15. East-West Response CONTROL BUILDING el 180 ft



Legend:

0.6 x Soil Modulus

1.0 x Soil Modulus

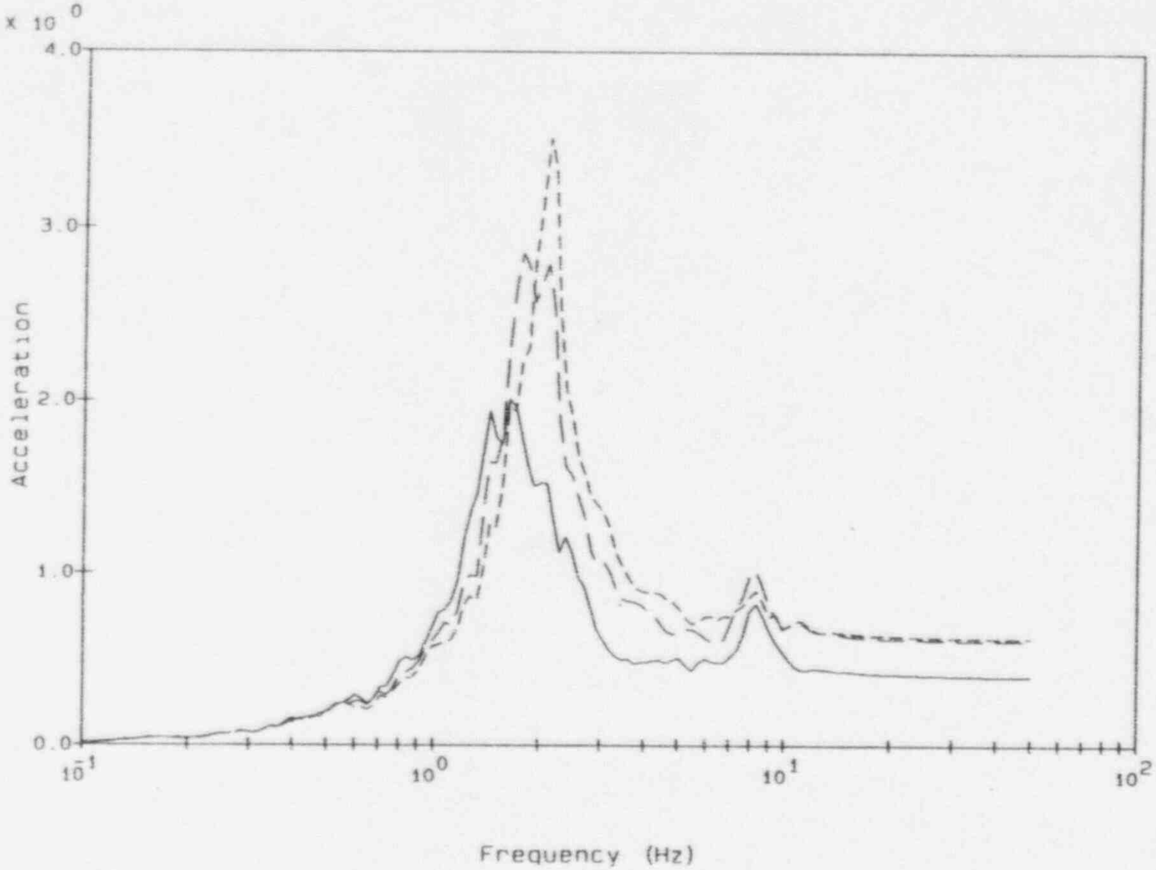
1.6 x Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

Node 15, Vertical Response CONTROL BUILDING el 180 ft



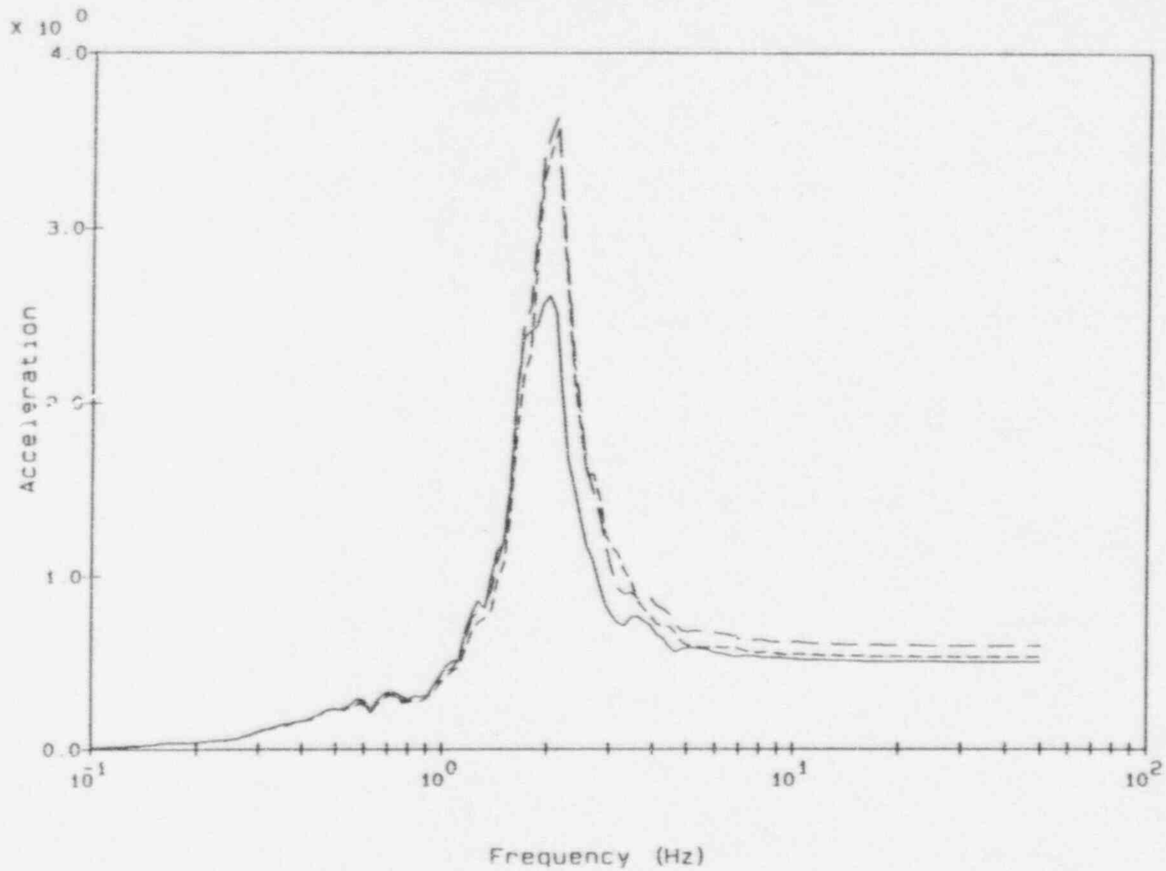
Legend:

0.6 x Soil Modulus
1.0 x Soil Modulus
1.6 x Soil Modulus

Notes:

Accelerations in g's
5% Spectral Damping

Node 16, North-South Response CONTROL BUILDING el 209 ft



Legend:

0.6 x Soil Modulus

1.0 x Soil Modulus

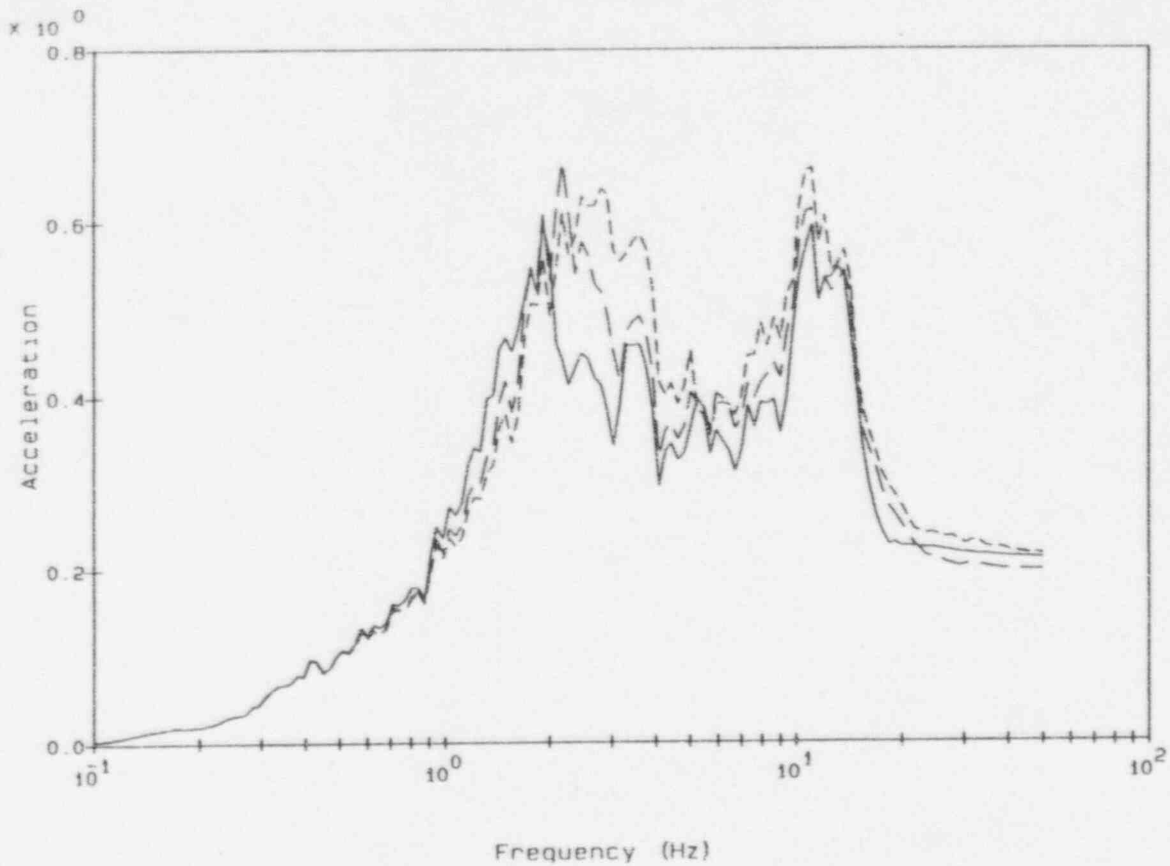
1.6 x Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

Node 16, East-West Response   CONTROL BUILDING el 209 ft



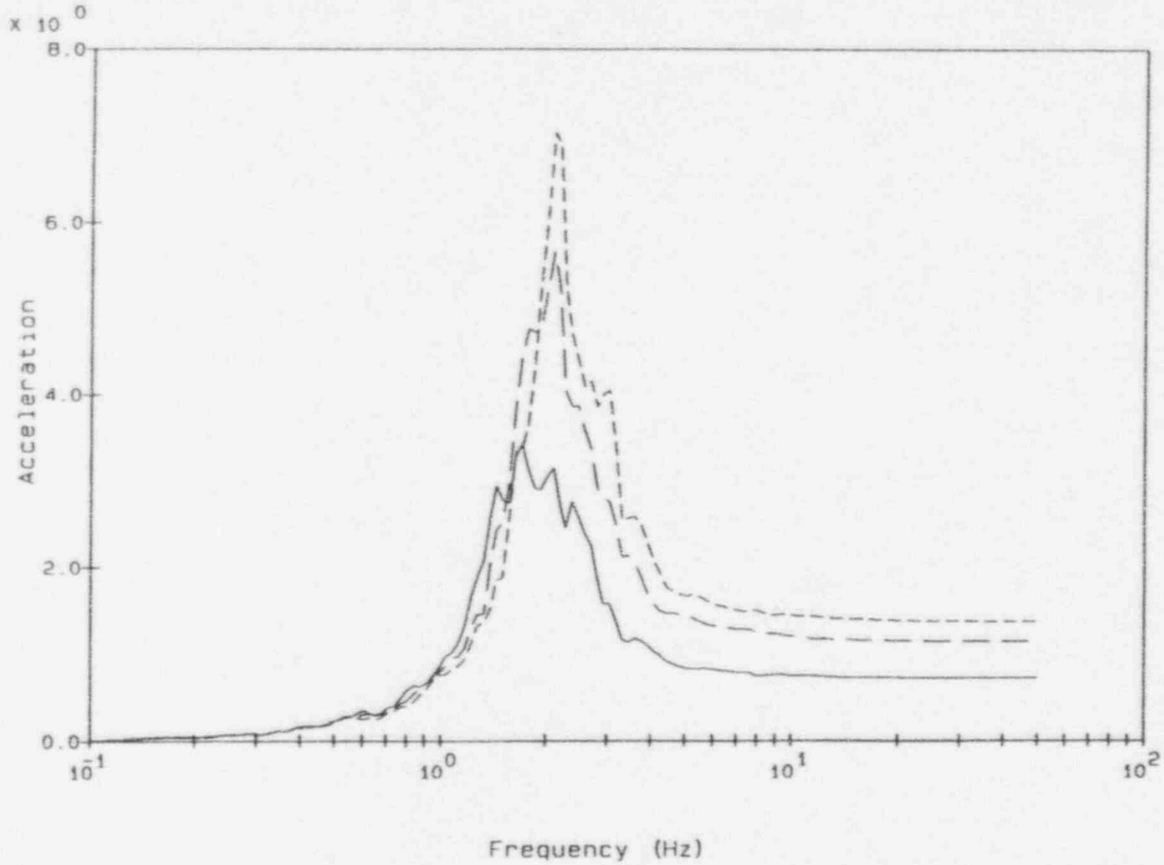
Legend:

0.6 x Soil Modulus
1.0 x Soil Modulus
1.6 x Soil Modulus

Notes:

Accelerations in g's
5% Spectral Damping

Node 16. Vertical Response CONTROL BUILDING el 209 ft



Legend:

0.6 x Soil Modulus

1.0 x Soil Modulus

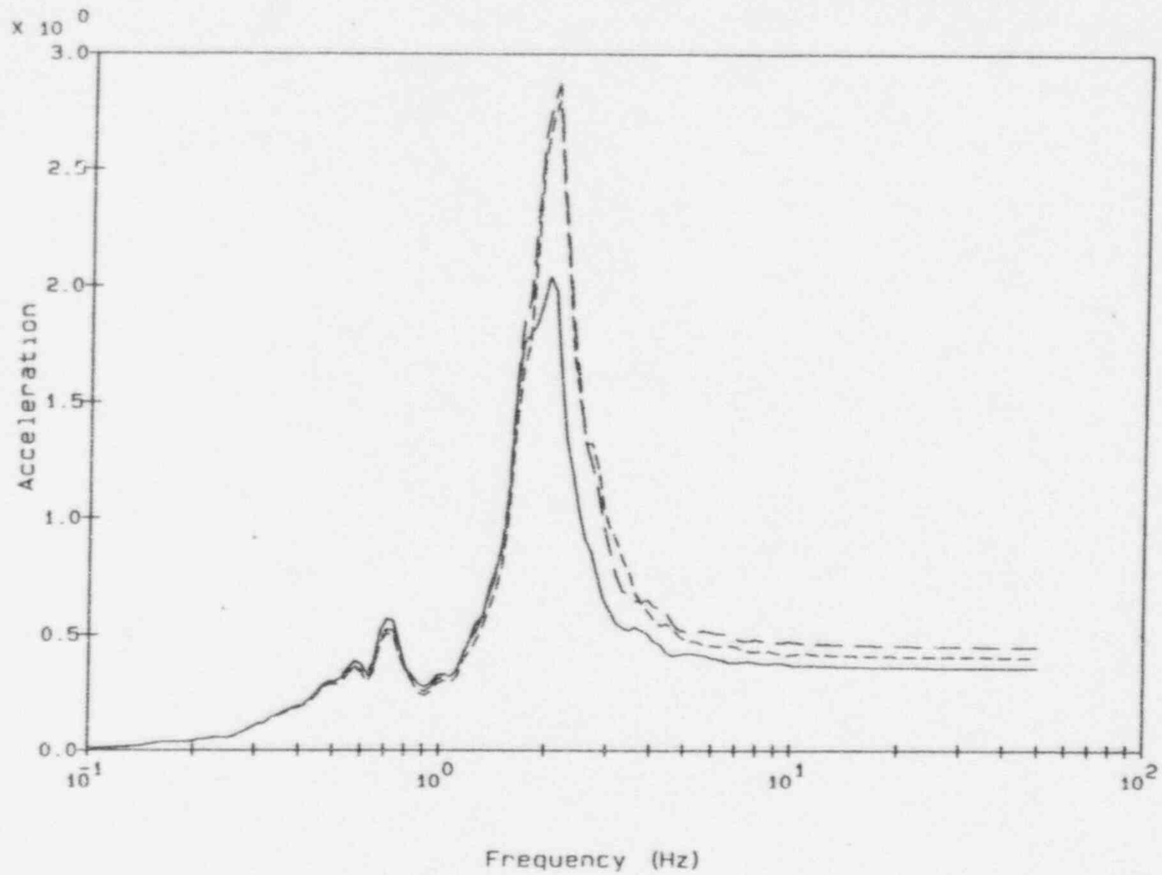
1.6 x Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

Node 18, North-South Response CONTROL BUILDING el 241 ft



Legend:

0.6 x Soil Modulus ————

1.0 x Soil Modulus — — — —

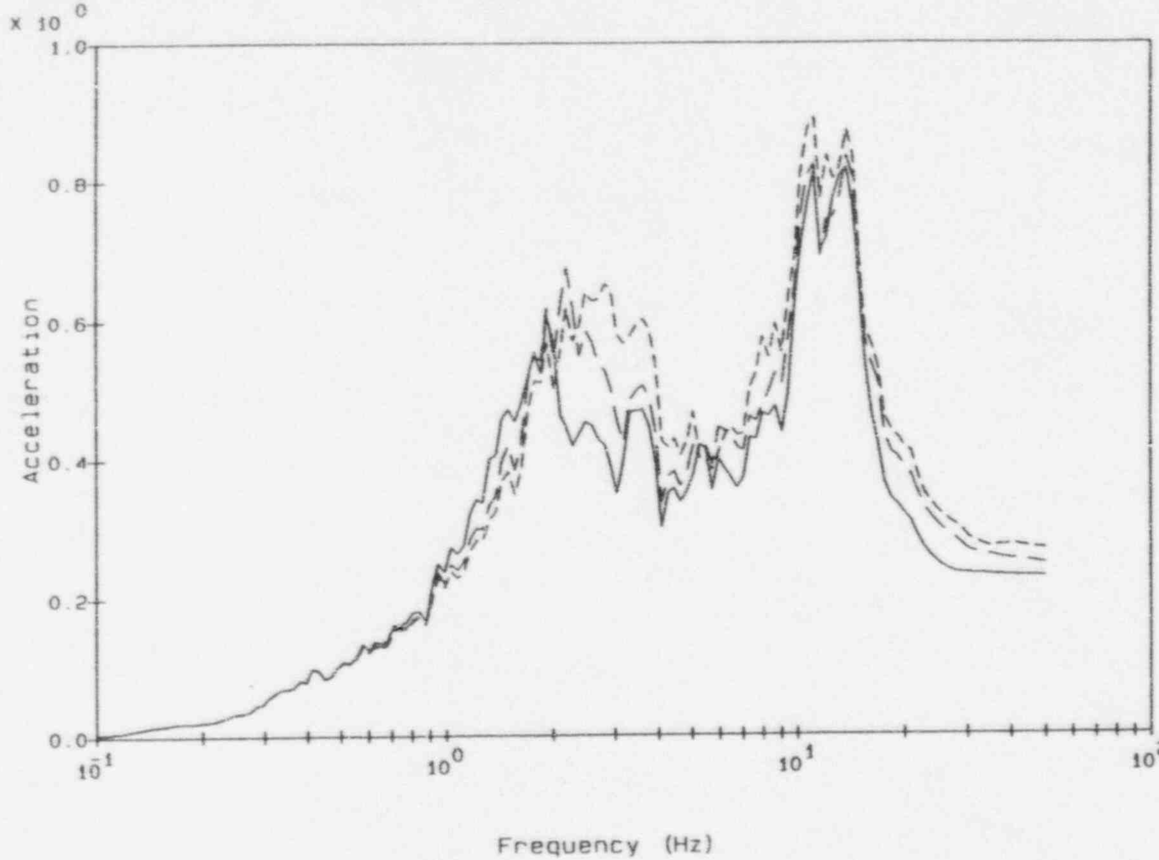
1.6 x Soil Modulus - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Node 18, East-West Response   CONTROL BUILDING el 241 ft



Legend:

0.6 x Soil Modulus

1.0 x Soil Modulus

1.6 x Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

Node 18, Vertical Response   CONTROL BUILDING el 241 ft

PLANT HATCH

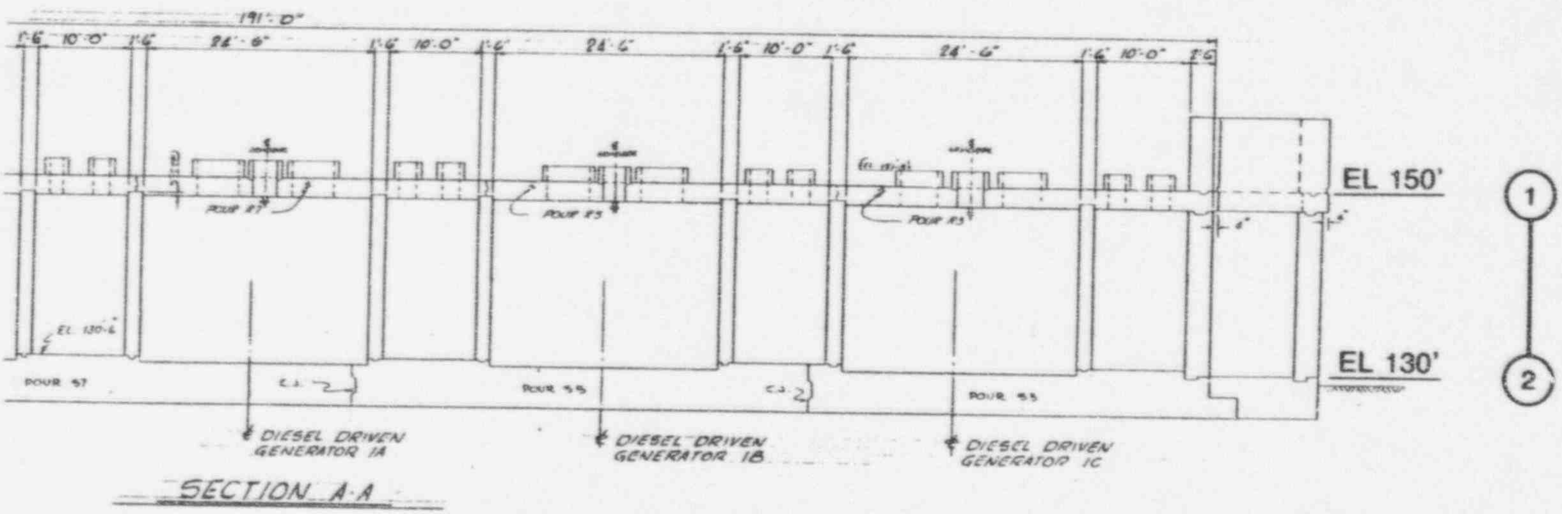
DIESEL GENERATOR BUILDING

SUMMARY OF MAXIMUM ACCELERATIONS AND DISPLACEMENTS

AND

5% DAMPED SME IN-STRUCTURE RESPONSE SPECTRA (1)

(1) Spectra are raw curves that have not been broadened.



PARTIAL SECTION THROUGH DIESEL GENERATOR BUILDING
LOOKING EAST

SMA
SEISMIC MODEL

DIESEL GENERATOR BUILDING

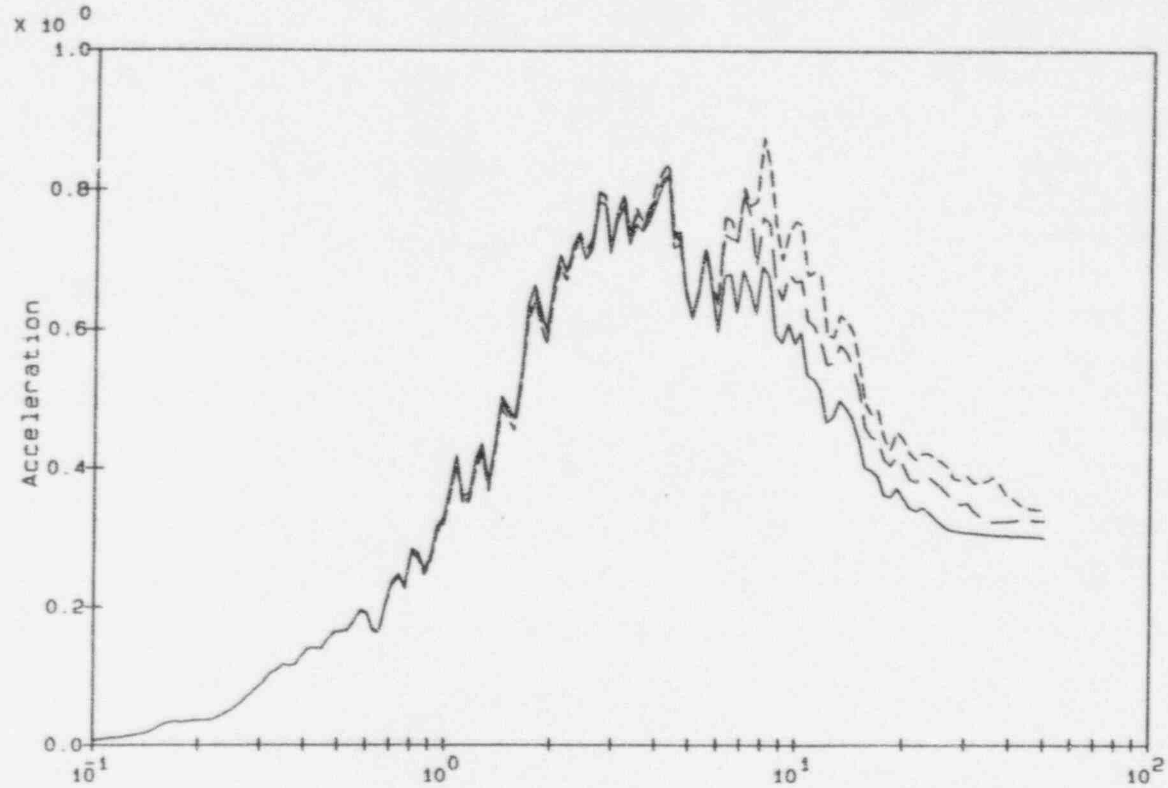
SUMMARY OF MAXIMUM ACCELERATIONS AND DISPLACEMENTS

Absolute Accelerations (g)

	Lower bound soil properties	Intermediate soil properties	Upper bound soil properties
Node 1			
E-W dir	0.3000	0.3235	0.3381
N-S dir	0.2785	0.2960	0.3142
Vert.	0.1986	0.1952	0.2000
Node 2			
E-W dir	0.2925	0.3101	0.3251
N-S dir	0.2679	0.2861	0.2868
Vert.	0.1956	0.1932	0.1979

Relative Displacements (in.)

	Lower bound soil properties	Intermediate soil properties	Upper bound soil properties
Node 1			
E-W dir	0.0039	0.0042	0.0044
N-S dir	0.0028	0.0029	0.0031
Vert.	0.0004	0.0004	0.0004
Node 2			
E-W dir	0.0000	0.0000	0.0000
N-S dir	0.0000	0.0000	0.0000
Vert.	0.0000	0.0000	0.0000



Legend:

0.6 x Soil Modulus

1.0 x Soil Modulus

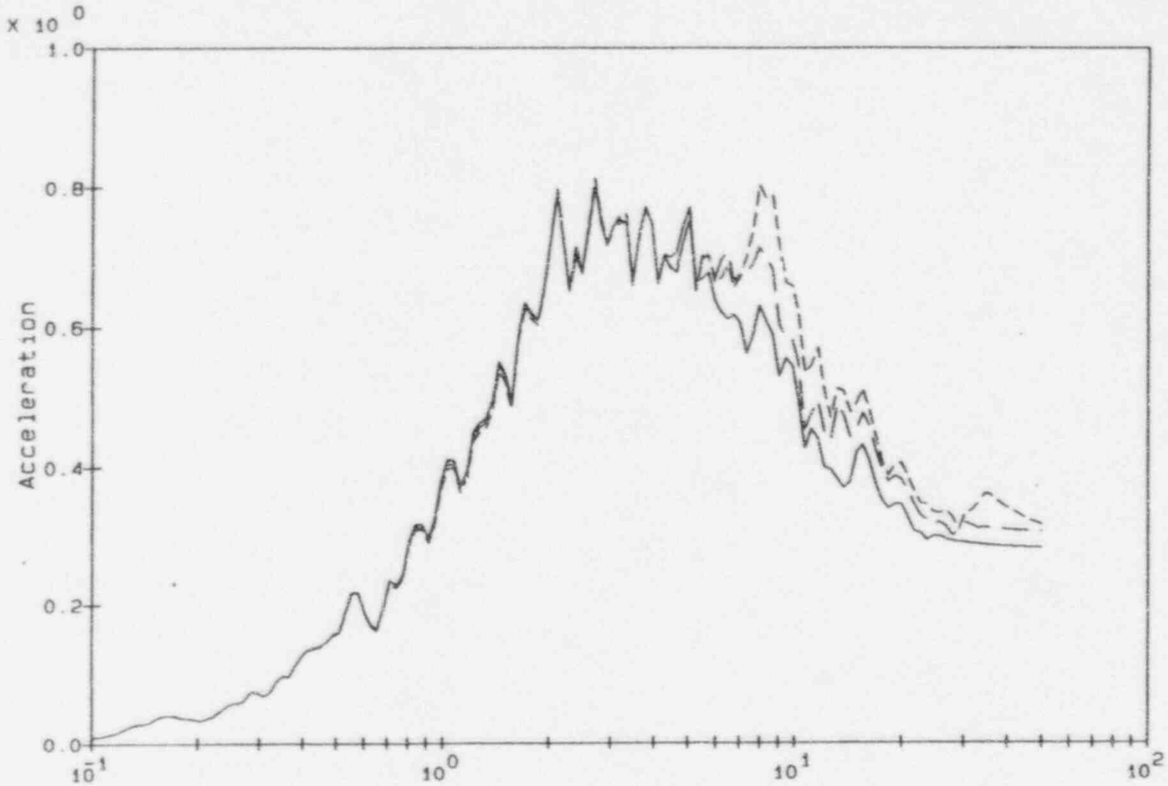
1.6 x Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

Node 1, East-West Response   DIESEL GENERATOR BUILDING el 150 ft



Frequency (Hz)

Legend:

0.6 x Soil Modulus

1.0 x Soil Modulus

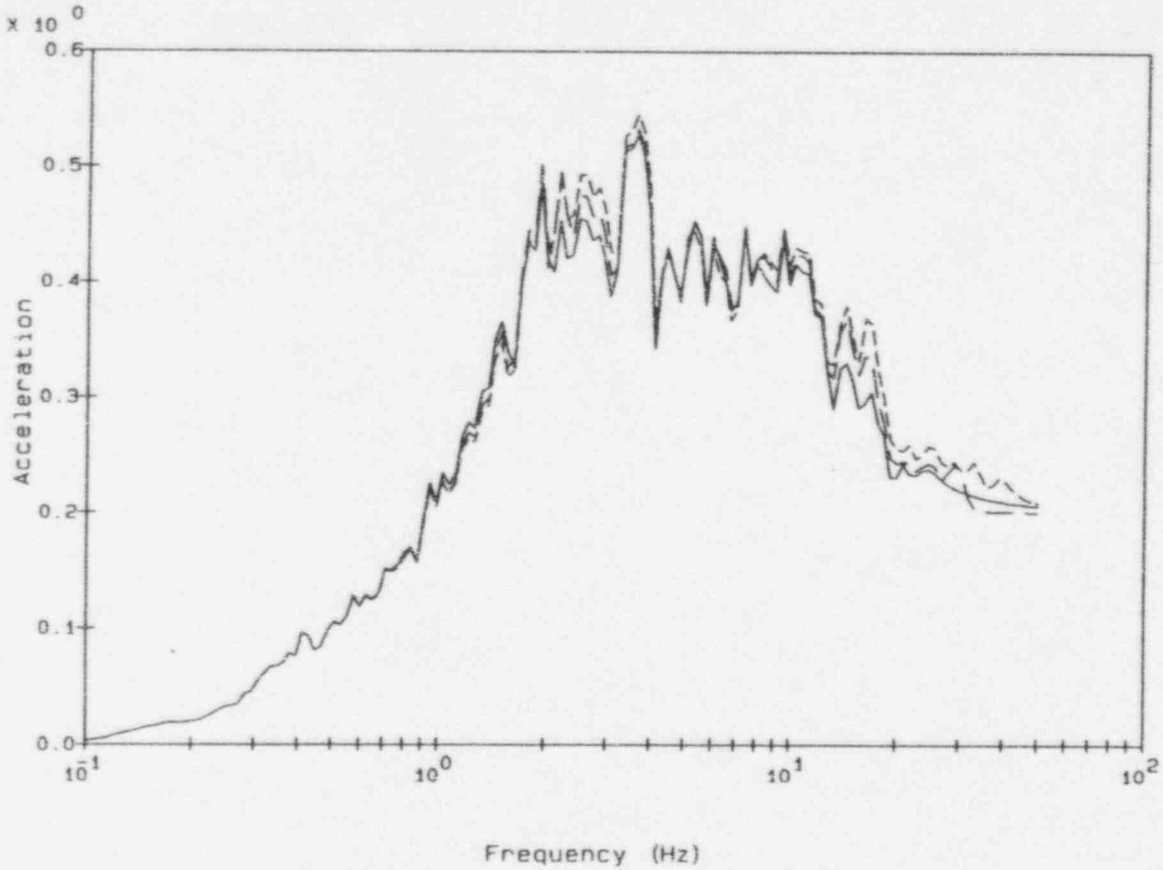
1.6 x Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

Node 1. North-South Response DIESEL GENERATOR BUILDING el 150 ft



Legend:

0.6 x Soil Modulus

1.0 x Soil Modulus

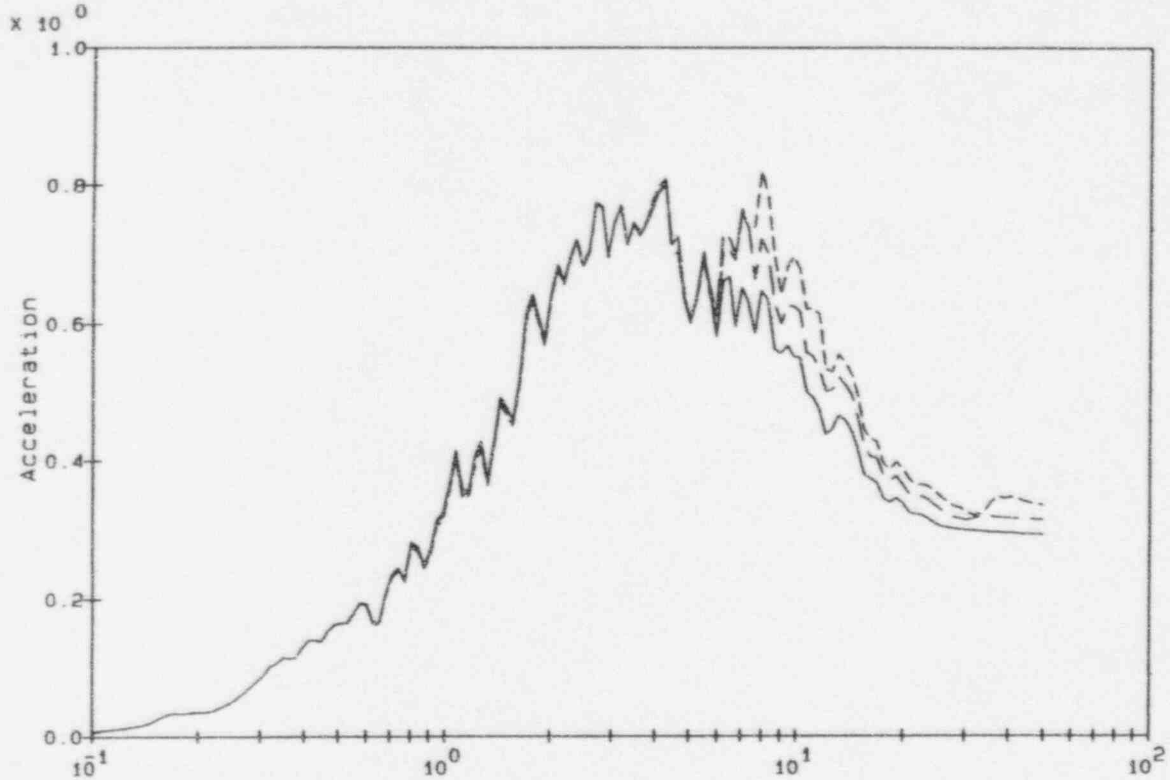
1.6 x Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

Node 1. Vertical Response DIESEL GENERATOR BUILDING el 150 ft



Legend:

0.6 x Soil Modulus

1.0 x Soil Modulus

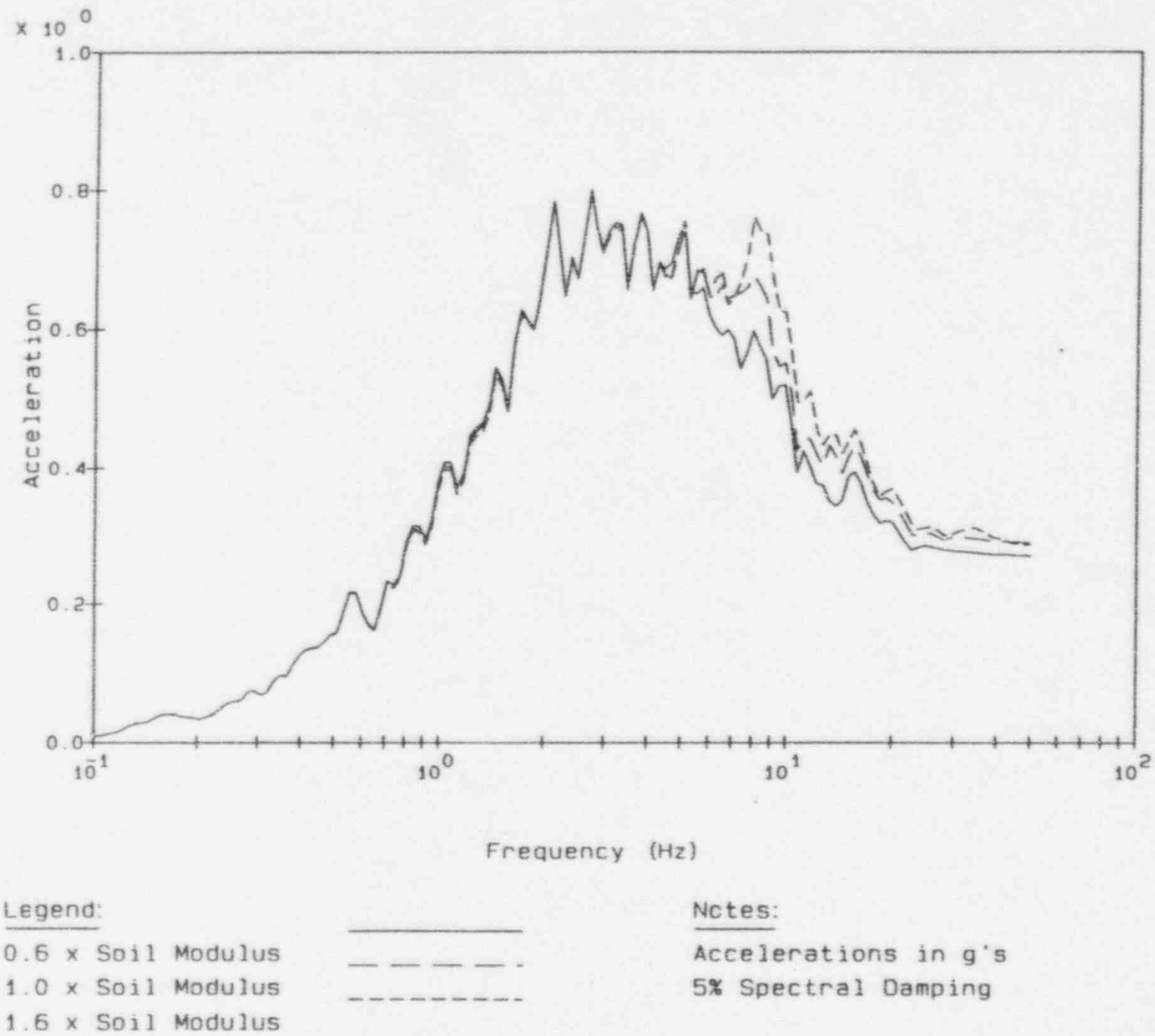
1.6 x Soil Modulus

Notes:

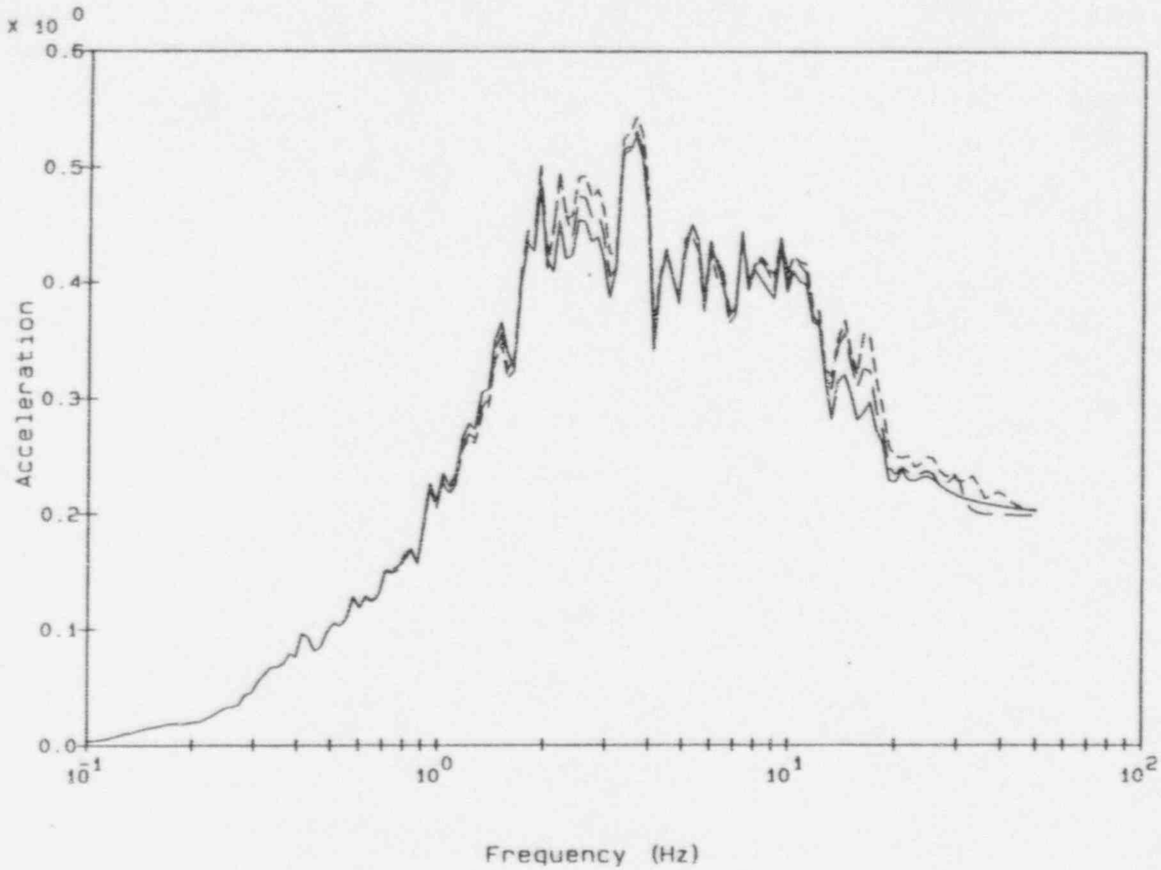
Accelerations in g's

5% Spectral Damping

Node 2. East-West Response DIESEL GENERATOR BUILDING el 130 ft



Node 2, North-South Response   DIESEL GENERATOR BUILDING el 130 ft



Legend:

0.6 x Soil Modulus ————

1.0 x Soil Modulus — — — — -

1.6 x Soil Modulus - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Node 2, Vertical Response   DIESEL GENERATOR BUILDING el 130 ft

PLANT HATCH

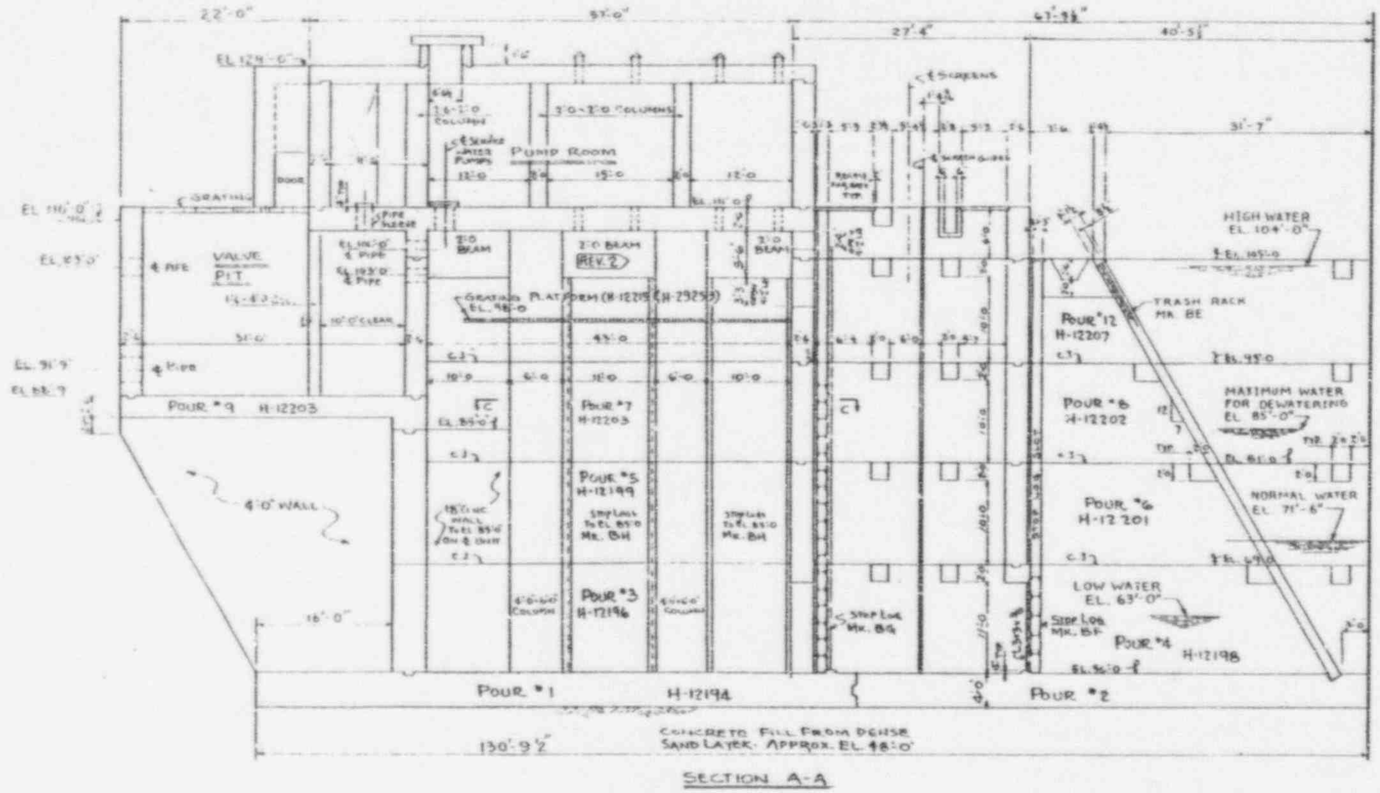
INTAKE STRUCTURE

SUMMARY OF MAXIMUM ACCELERATIONS AND DISPLACEMENTS

AND

5% DAMPED SME IN-STRUCTURE RESPONSE SPECTRA [1]

[1] Spectra are raw curves that have not been broadened.



INTAKE STRUCTURE SECTION
LOOKING WEST

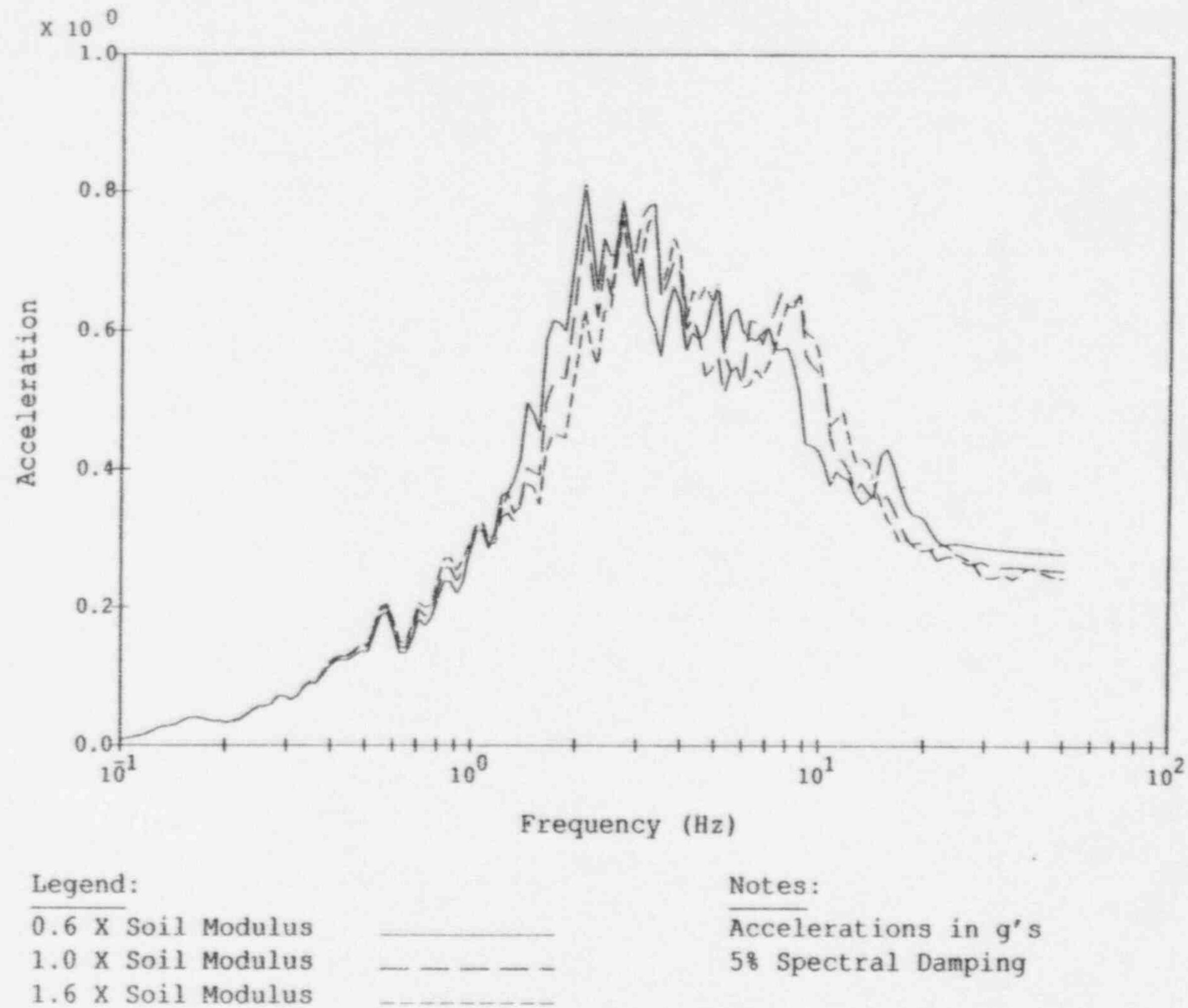
SMA
SEISMIC MODEL
MASS POINTS

INTAKE STRUCTURE
SUMMARY OF MAXIMUM ABSOLUTE ACCELERATIONS (g)

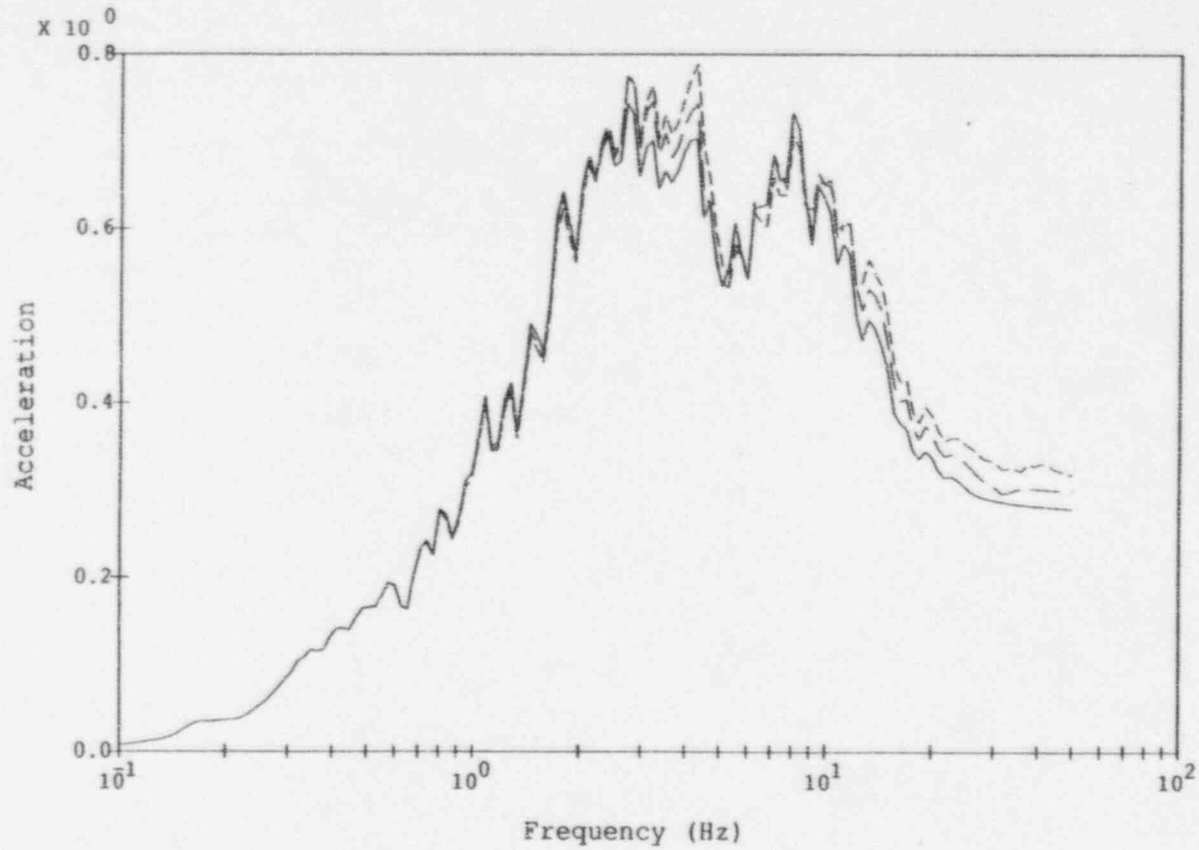
		Lower Bound Soil Properties	Intermediate Soil Properties	Upper Bound Soil Properties
Node 2,	Elevation 54.00			
	N-S dir	0.2768	0.2533	0.2417
	E-W dir	0.2777	0.2980	0.3148
	Vertical	0.2065	0.1953	0.2165
Node 9,	Elevation 69.00			
	N-S dir	0.2632	0.2564	0.2492
	E-W dir	0.2731	0.3029	0.3161
	Vertical	0.2093	0.1998	0.2188
Node 12,	Elevation 87.50			
	N-S dir	0.2963	0.2852	0.2950
	E-W dir	0.2866	0.3299	0.3319
	Vertical	0.2126	0.2048	0.2156
Node 15,	Elevation 109.75			
	N-S dir	0.3557	0.3240	0.3501
	E-W dir	0.3606	0.3978	0.3991
	Vertical	0.2149	0.2087	0.2236
Node 18,	Elevation 127.00			
	N-S dir	0.4006	0.3536	0.3922
	E-W dir	0.4320	0.4643	0.4551
	Vertical	0.2173	0.2214	0.2486

INTAKE STRUCTURE
SUMMARY OF MAXIMUM RELATIVE DISPLACEMENTS (in)

		Lower Bound Soil Properties	Intermediate Soil Properties	Upper Bound Soil Properties
Node 2,	Elevation 54.00			
	N-S dir	0.0000	0.0000	0.0000
	E-W dir	0.0000	0.0000	0.0000
	Vertical	0.0000	0.0000	0.0000
Node 9,	Elevation 69.00			
	N-S dir	0.0039	0.0037	0.0039
	E-W dir	0.0092	0.0104	0.0106
	Vertical	0.0008	0.0007	0.0009
Node 12,	Elevation 87.50			
	N-S dir	0.0084	0.0079	0.0084
	E-W dir	0.0209	0.0234	0.0237
	Vertical	0.0014	0.0013	0.0014
Node 15,	Elevation 109.75			
	N-S dir	0.0129	0.0118	0.0127
	E-W dir	0.0354	0.0392	0.0388
	Vertical	0.0019	0.0019	0.0020
Node 18,	Elevation 127.00			
	N-S dir	0.0160	0.0144	0.0157
	E-W dir	0.0442	0.0484	0.0475
	Vertical	0.0022	0.0022	0.0026



Hatch Intake Structure Seismic Response
Node 2 (el 54.0 ft), North-South Direction



Legend:

0.6 X Soil Modulus

1.0 X Soil Modulus

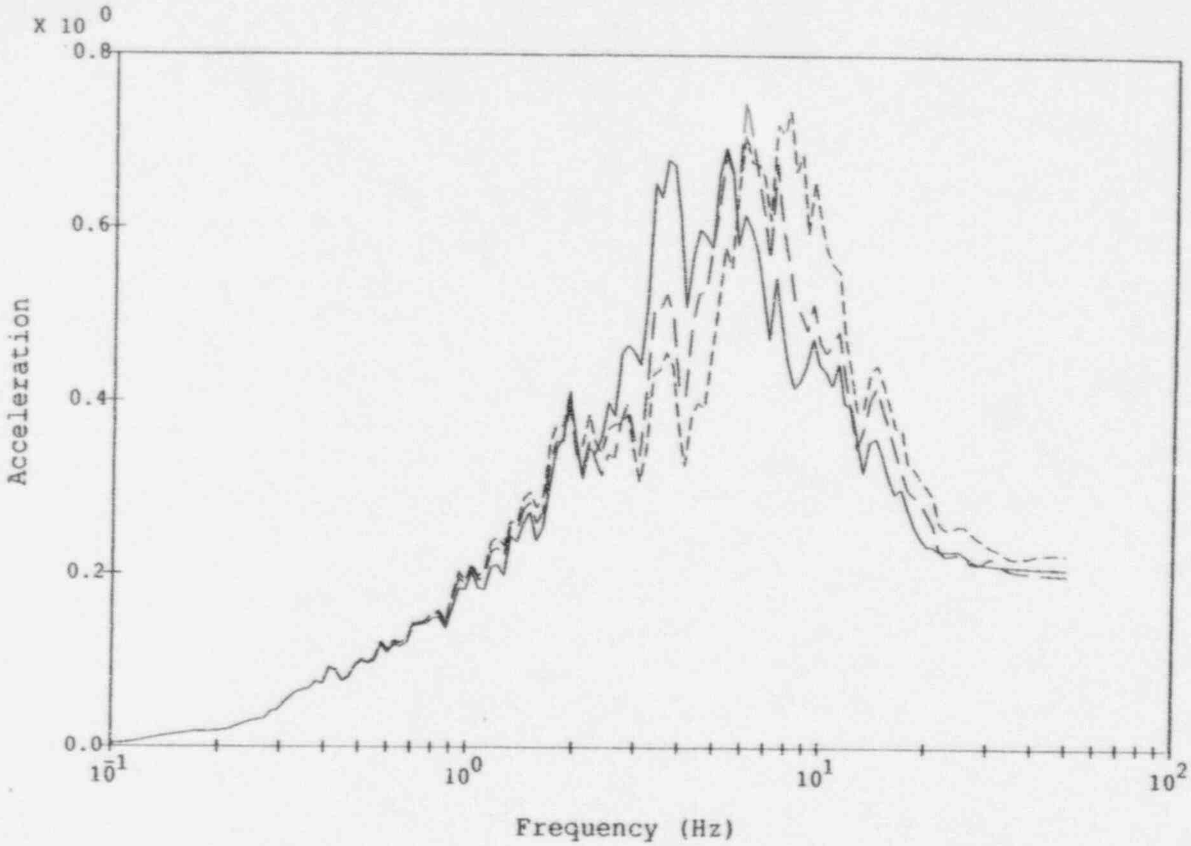
1.6 X Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 2 (el 54.0 ft), East-West Direction



Legend:

0.6 X Soil Modulus ————————

1.0 X Soil Modulus — — — — —

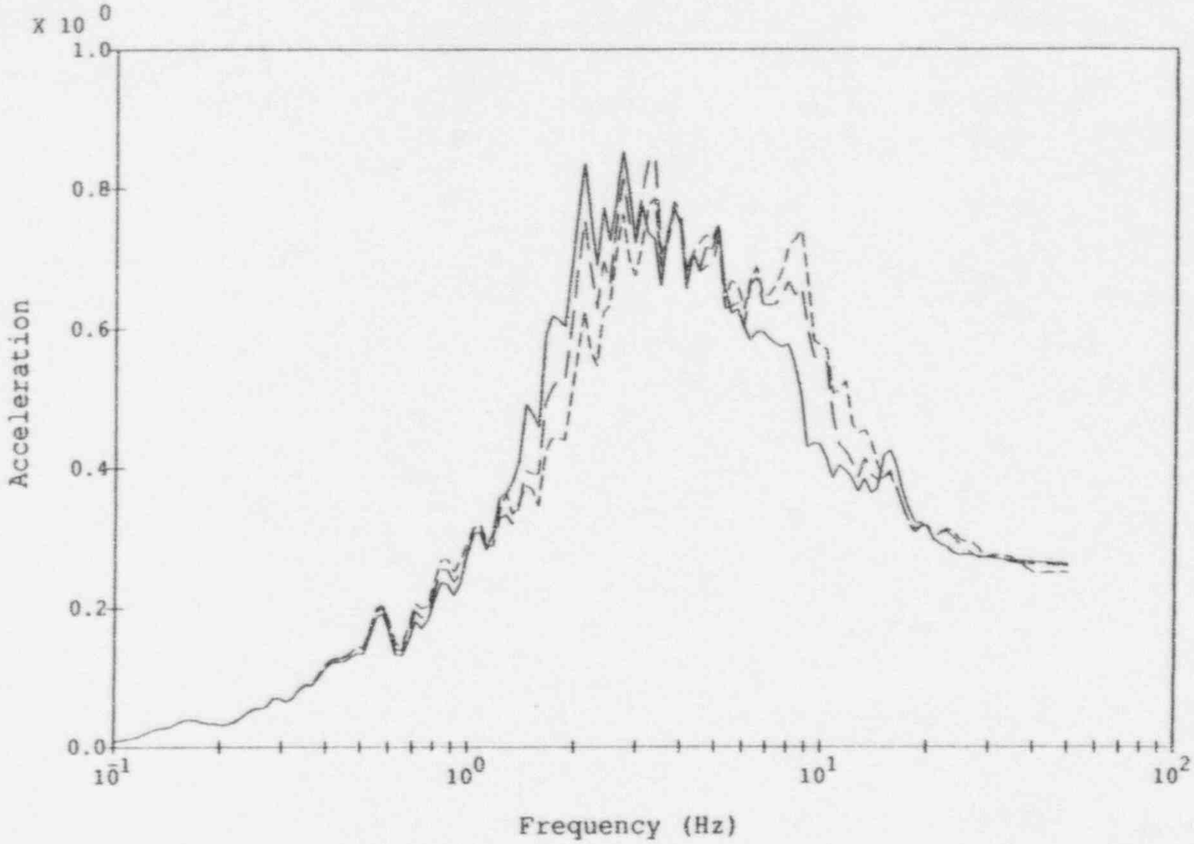
1.6 X Soil Modulus - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 2 (el 54.0 ft), Vertical Direction



Legend:

0.6 X Soil Modulus ————————

1.0 X Soil Modulus — — — — —

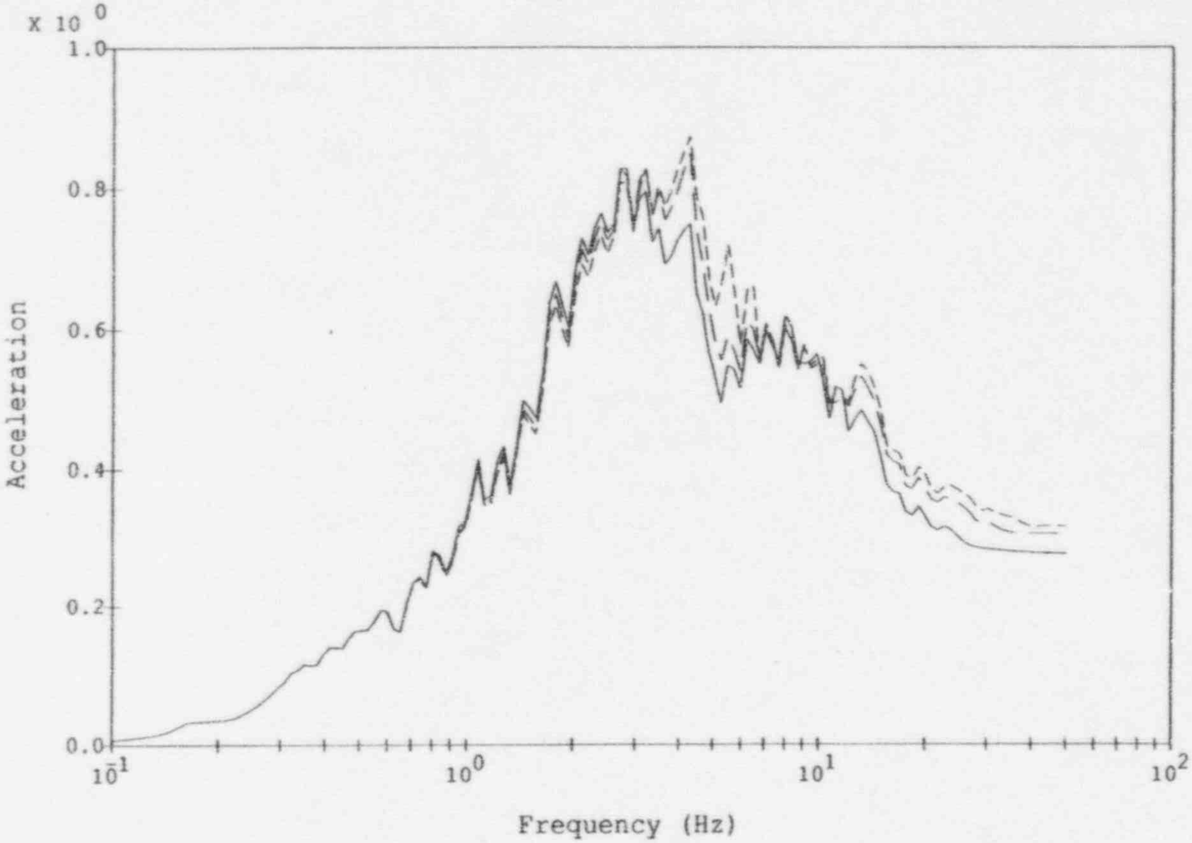
1.6 X Soil Modulus - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 9 (el 69.0 ft), North-South Direction



Legend:

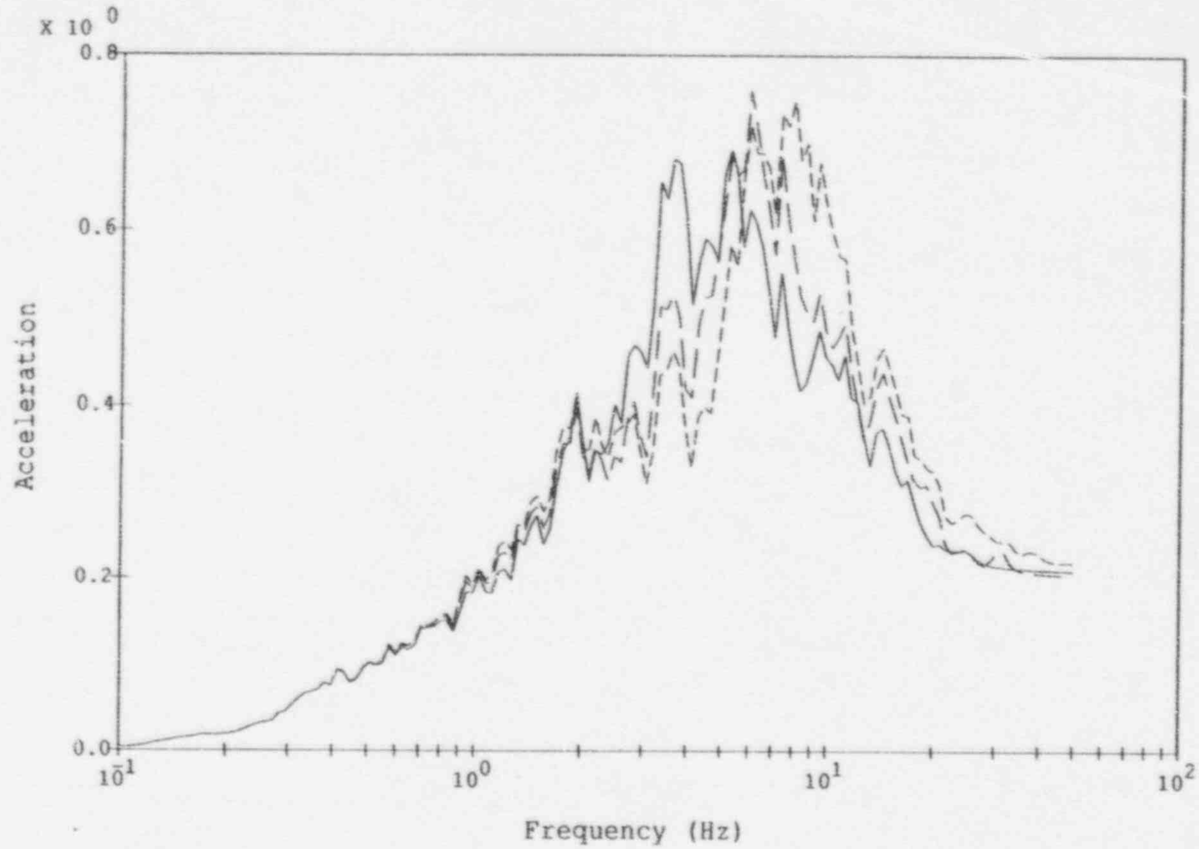
0.6 X Soil Modulus	————
1.0 X Soil Modulus	— — — —
1.6 X Soil Modulus	- - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 9 (el 69.0 ft), East-West Direction



Legend:

0.6 X Soil Modulus ————

1.0 X Soil Modulus — — — —

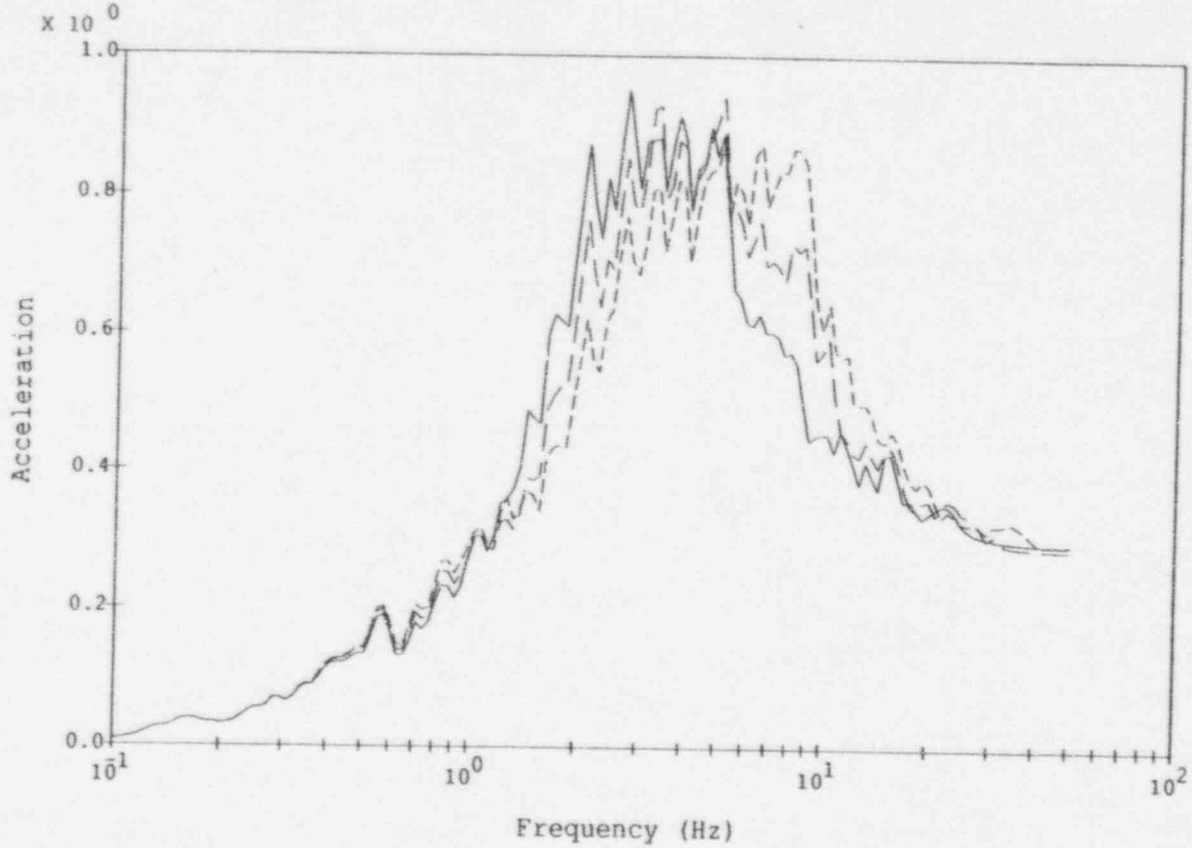
1.6 X Soil Modulus - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 9 (el 69.0 ft), Vertical Direction



Legend:

0.6 X Soil Modulus ————

1.0 X Soil Modulus — — — —

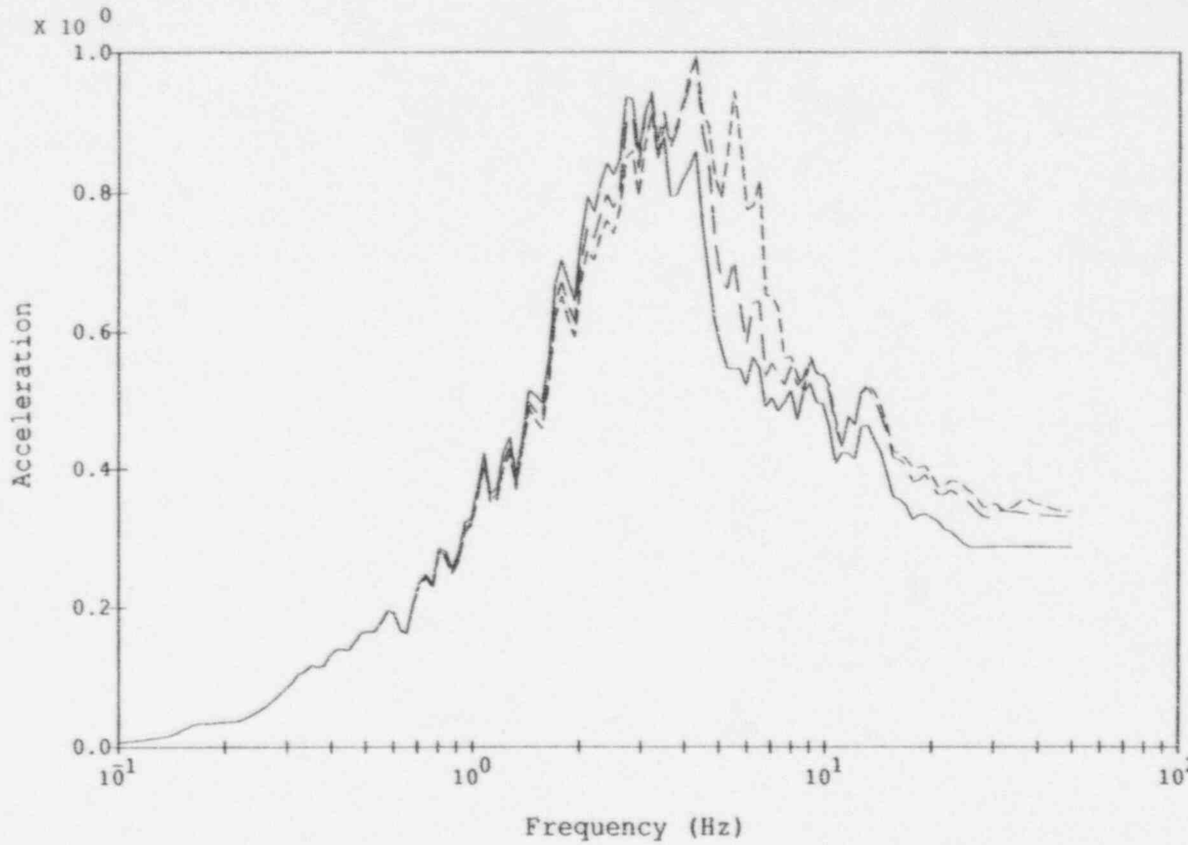
1.6 X Soil Modulus - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 12 (el 87.5 ft), North-South Direction



Legend:

0.6 X Soil Modulus ————————

1.0 X Soil Modulus — — — — —

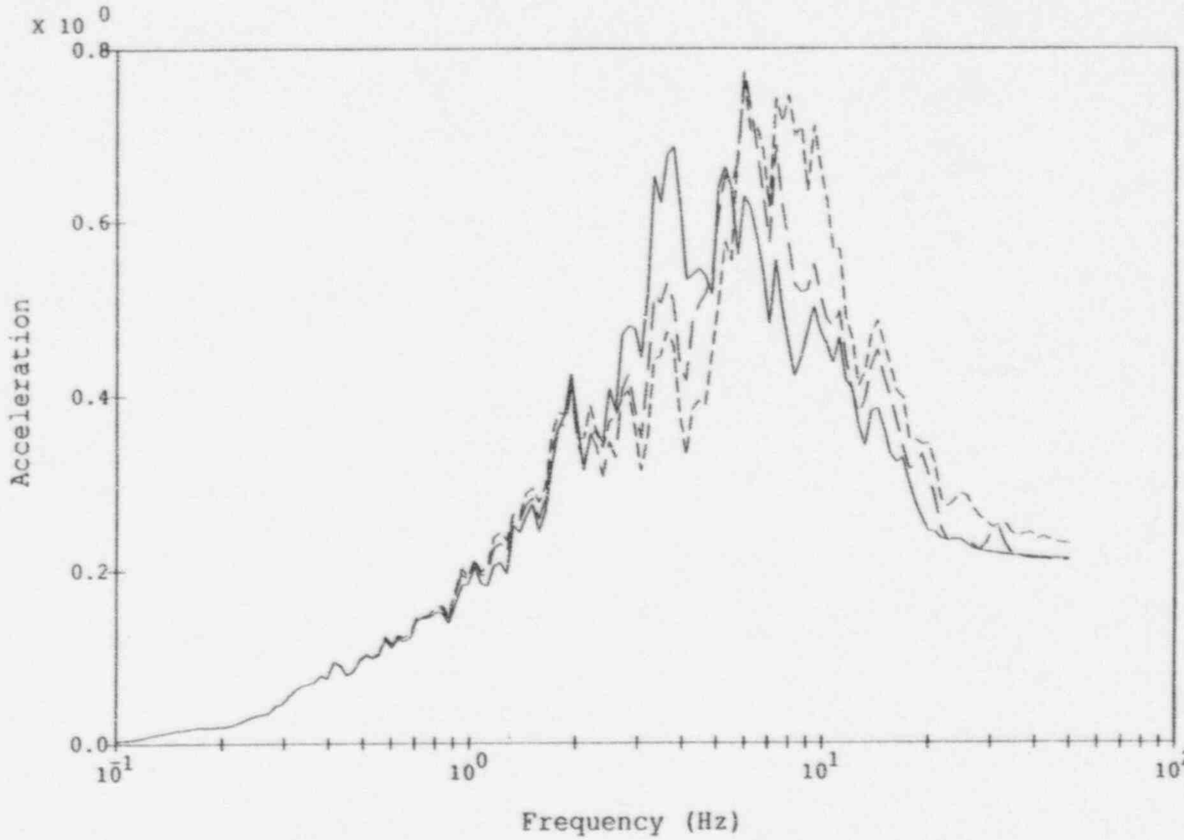
1.6 X Soil Modulus - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 12 (el 87.5 ft), East-West Direction



Legend:

0.6 X Soil Modulus ————————

1.0 X Soil Modulus — — — — —

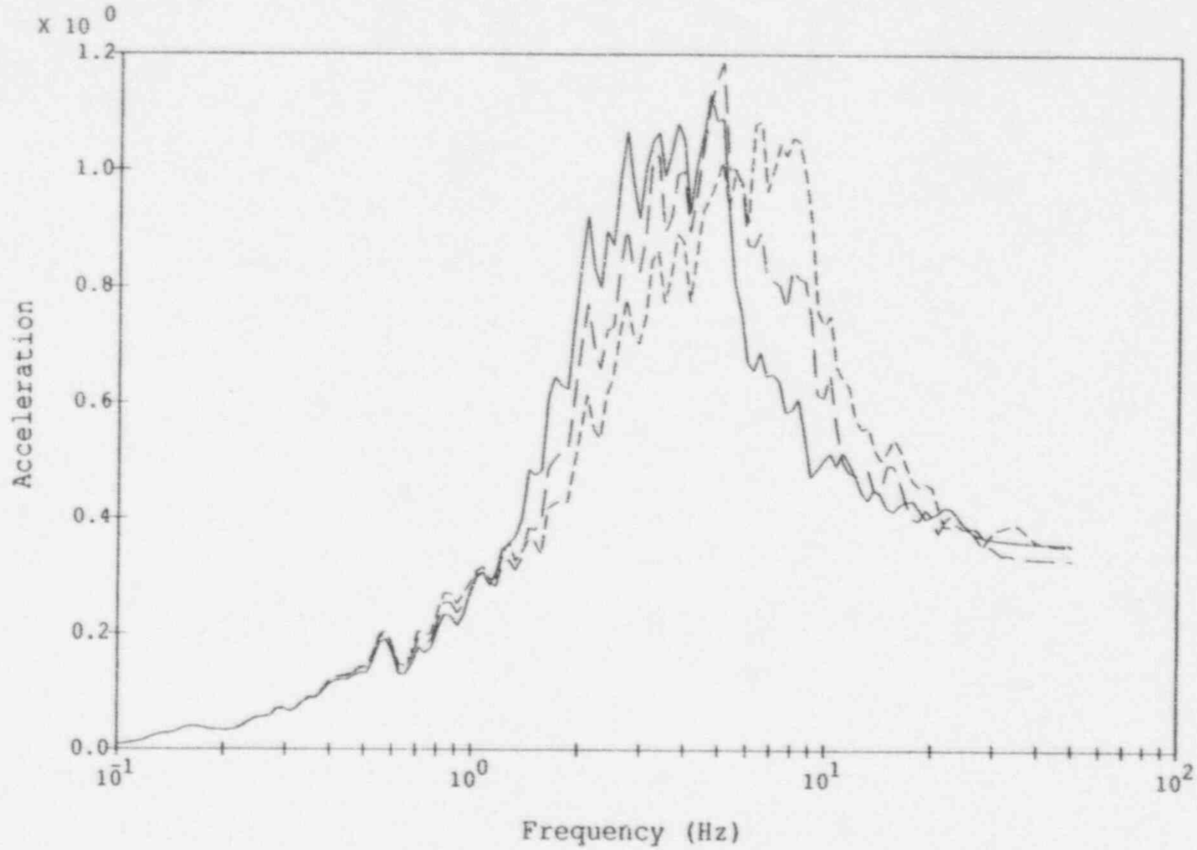
1.6 X Soil Modulus - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 12 (el 87.5 ft), Vertical Direction



Legend:

0.6 X Soil Modulus ————————

1.0 X Soil Modulus — — — — —

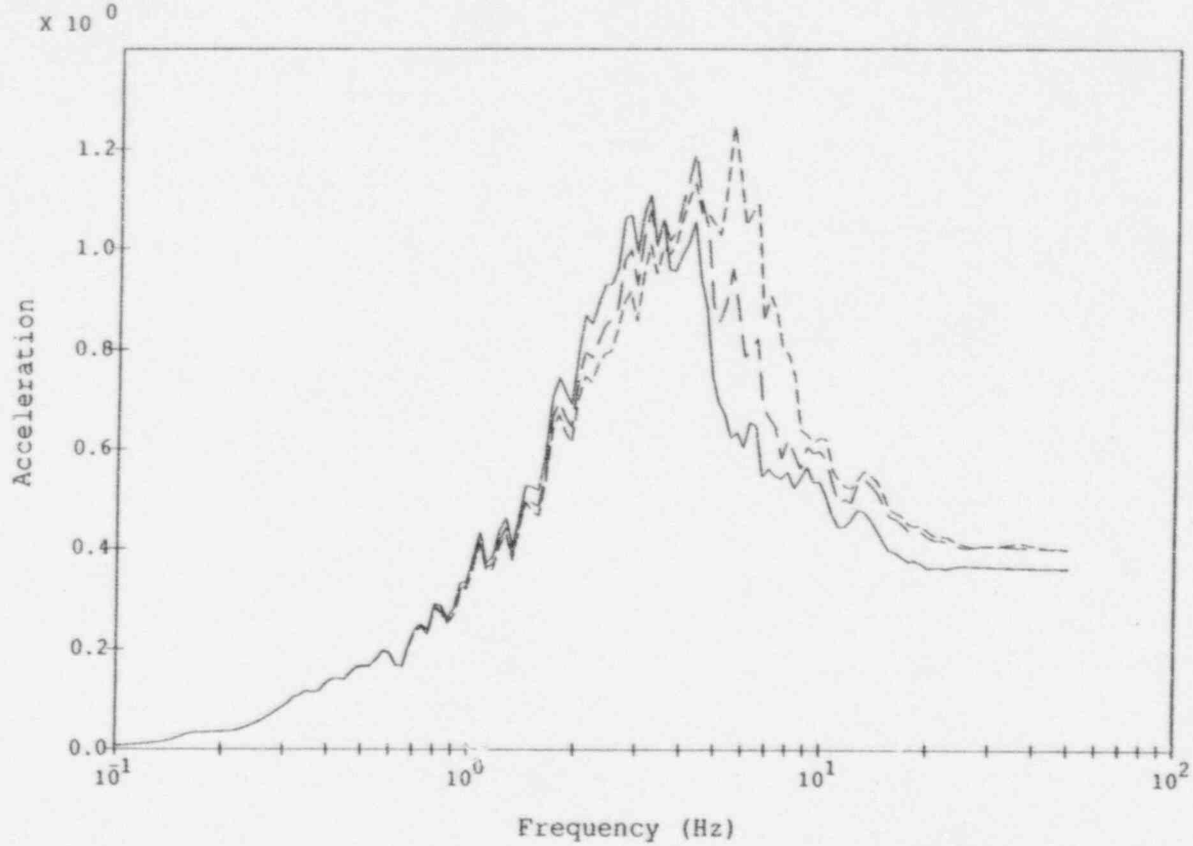
1.6 X Soil Modulus - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 15 (el 109.75 ft), North-South Direction



Legend:

0.6 X Soil Modulus ——————

1.0 X Soil Modulus — — — — —

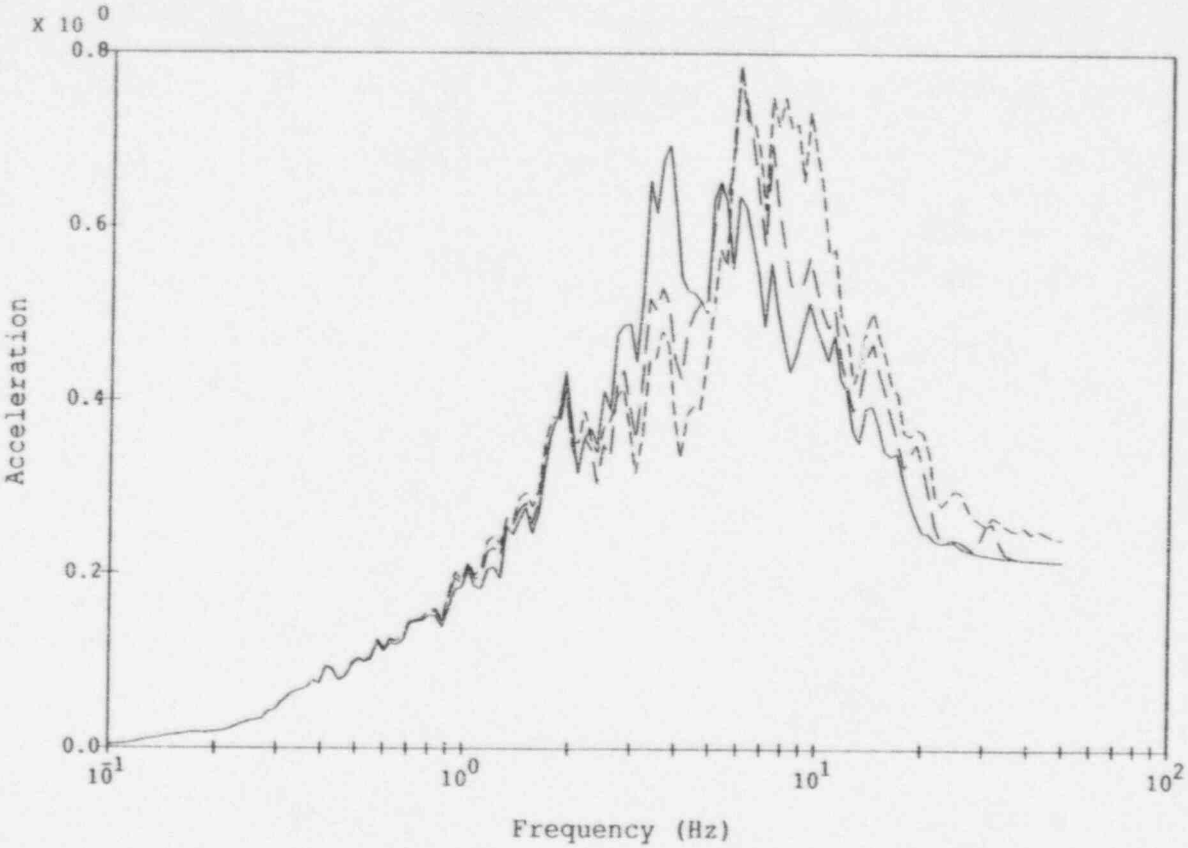
1.6 X Soil Modulus - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 15 (el 109.75), East-West Direction



Legend:

0.6 X Soil Modulus

1.0 X Soil Modulus

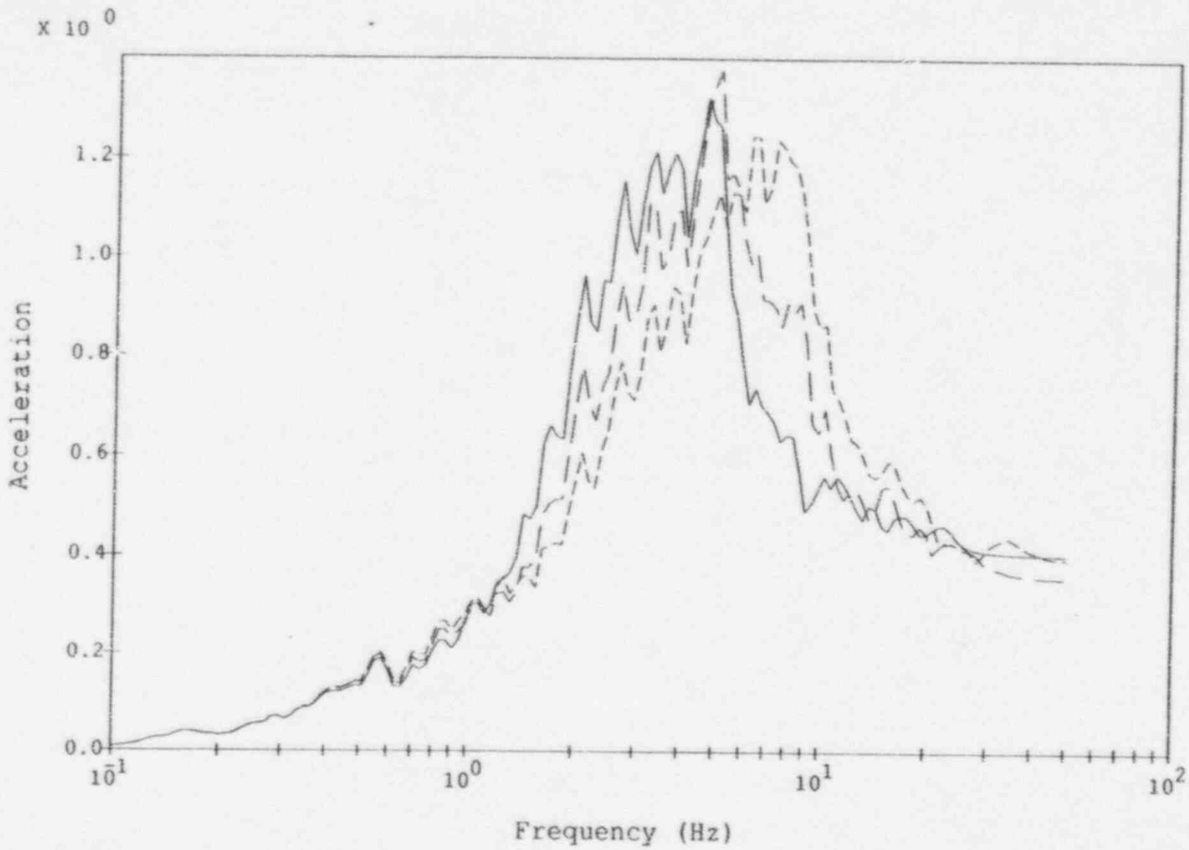
1.6 X Soil Modulus

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 15 (el 109.75 ft), Vertical Direction



Legend:

0.6 X Soil Modulus ————————

1.0 X Soil Modulus — — — — —

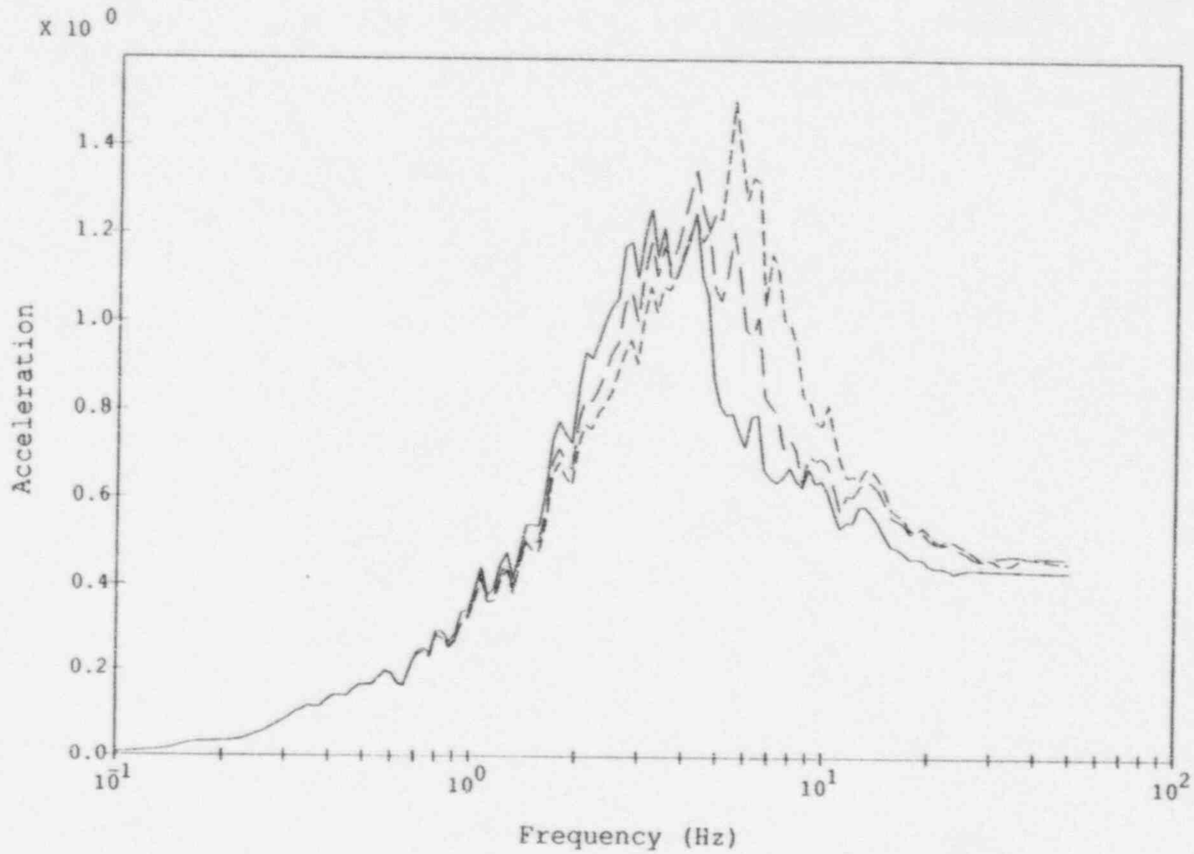
1.6 X Soil Modulus - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 18 (el 127.0 ft), North-South Direction



Legend:

0.6 X Soil Modulus ————————

1.0 X Soil Modulus — — — — —

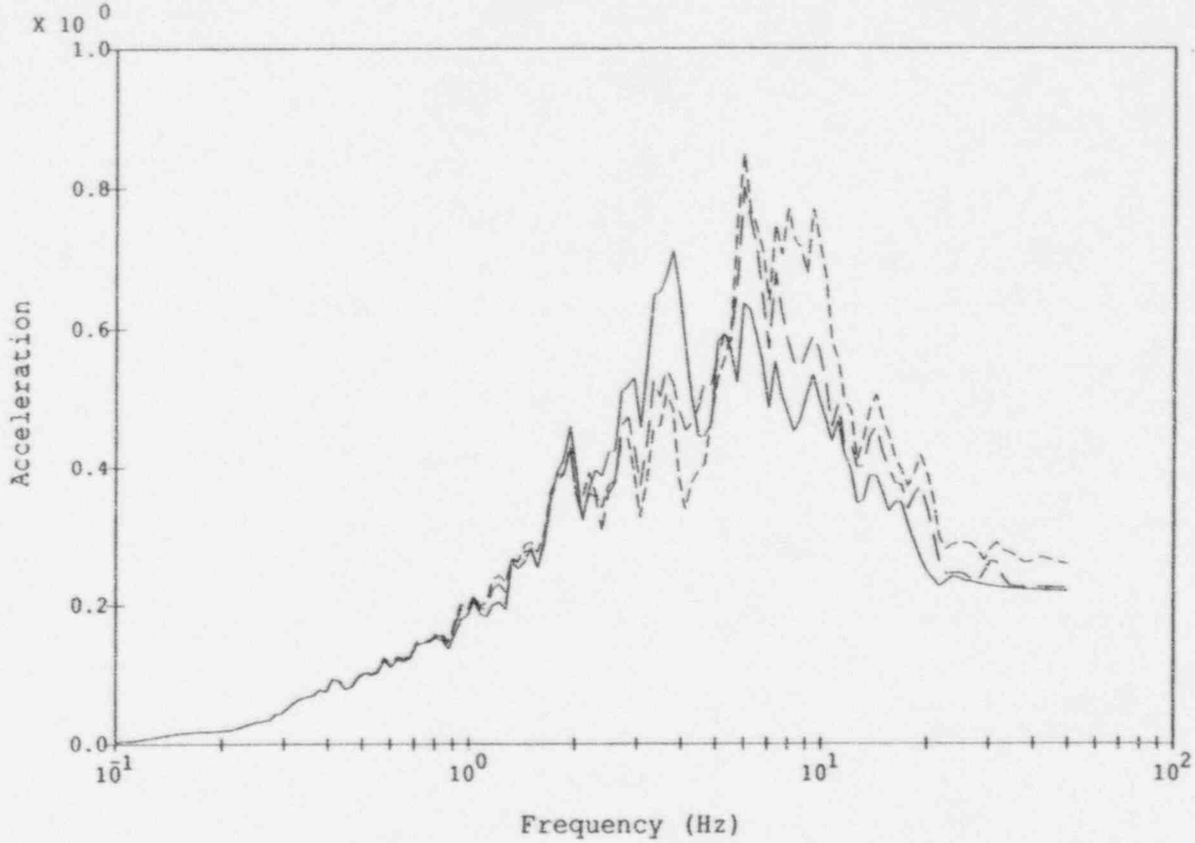
1.6 X Soil Modulus - - - - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 18 (el 127.0 ft), East-West Direction



Legend:

0.6 X Soil Modulus ————

1.0 X Soil Modulus — — — —

1.6 X Soil Modulus - - - - - -

Notes:

Accelerations in g's

5% Spectral Damping

Hatch Intake Structure Seismic Response
Node 18 (el 127.0 ft), Vertical Direction

APPENDIX I
SUMMARY OF EQUIPMENT OUTLIERS
PLANT HATCH UNITS 1 AND 2

Appendix I contains a summary of all Safe Shutdown Equipment List components which require a modification to achive a HCLPF capacity of at least 0.3 g pga.. This appendix contains the following information, sorted by equipment identification number.

Column Heading	Description
Equipment Identification Number	Equipment identification number
Equipment Class	Equipment class (see table 1 of this appendix)
Equipment Description	Brief description of equipment
Equipment Location	Building and elevation where equipment is located
Outlier Description	Brief description of outliers
Outlier Resolution	Brief description of outlier resolution
Modification Method	If an outlier was resolved by a plant modification, this column gives the Design Change Request (DCR) number or states that a Maintenance Work Order (MWO) was used. All modifications listed in this appendix have been completed.

APPENDIX I
TABLE 1
EQUIPMENT CLASS DESIGNATIONS

Equipment Class (Column 2)	Description
01	Motor control centers
02	Low-voltage switchgear
03	Medium-voltage switchgear
04	Transformers
05	Horizontal pumps
06	Vertical pumps
07	Fluid-operated valves
08A	Motor-operated valves
08B	Solenoid-operated valves
09	Fans
10	Air handlers
11	Chillers
12	Air compressors
13	Motor generators
14	Distribution panels
15	Batteries on racks
16	Battery chargers and inverters
17	Engine generators
18	Instruments on racks
19	Temperature sensors
20	Instrumentation and control panels and cabinets
21	Tanks and heat exchangers

TABLE 2

DESCRIPTION OF EQUIPMENT OUTLIERS
PLANT HATCH UNIT 1
(SHEET 1 OF 5)

Equipment ID Number	Equip. Class	Equipment Description	Equipment Location	Outlier Description	Outlier Resolution	Modification Method
B31-F023A	8A	Motor-operated valve	Reactor Building drywell el 117 ft	Potential interaction with platform beam.	Move platform beam to provide adequate clearance.	DCR 89-205.
G11-F003	7	Fluid-operated valve	Reactor Building el 125 ft	Potential interaction between solenoid valve and fire-wrapped conduit.	Modify conduit routing to avoid interaction.	MWO.
H11-P603 H11-P606 H11-P608 H11-P609 H11-P610 H11-P611 H11-P612 H11-P613 H11-P617 H11-P618 H11-P622 H11-P623 H11-P626 H11-P627 H11-P628 H11-P650 H11-P651 H11-P652 H11-P700 2H11-P652	20	Control room panels	Control Building el 164 ft	Potential interaction with control room light fixtures.	Tie up light fixtures to prevent falling.	DCR 90-010.
H11-P620	20	Control room panel	Control Building el 164 ft	1. Missing door latching bars. 2. Potential interaction with control room light fixtures.	1. Repair door latch. 2. Tie up light fixtures to prevent falling.	1. MWO. 2. DCR 90-010.

TABLE 2
(SHEET 2 OF 5)

Equipment ID Number	Equip. Class	Equipment Description	Equipment Location	Outlier Description	Outlier Resolution	Modification Method
H11-P654	20	Control room panel	Control Building el 164 ft	1. Broken door latch. 2. Potential interaction with control room light fixtures. 3. Potential interaction with adjacent panel H11-P663 on vibration isolators.	1. Repair door latch. 2. Tie up light fixtures to prevent falling. 3. Modify anchorage for panel H11-P663.	1. MWO. 2. DCR 90-010. 3. DCR 93-055.
H11-P657	20	Control room panel	Control Building el 164 ft	1. Potential interaction with control room light fixtures. 2. Potential interaction with adjacent panel H11-P663 on vibration isolators.	1. Tie up light fixtures to prevent falling. 2. Modify anchorage for panel H11-P663.	1. DCR 90-010. 2. DCR 93-055.
H11-P691	20	Analog signal conversion/isolation panel	Control Building el 164 ft	1. SRT could not confirm that equipment pad has reinforcing and adequate load path to the floor slab. 2. Potential interaction with control room light fixtures. 3. Potential interaction with adjacent panels H11-P670 through H11-P681 on vibration isolators.	1. Add brace at top of panel to alleviate uplift at the base. 2. Tie up light fixtures to prevent falling. 3. Modify anchorage for panels H11-P670 through H11-P681.	1. DCR 89-206. 2. DCR 90-010. 3. DCR 93-055.
H11-P925 H11-P926	20	Control room panels	Control Building el 164 ft	Potential interaction with adjacent panels H11-P670 through H11-P681 mounted on vibration isolators.	Modify anchorage for panels H11-P670 through H11-P681.	DCR 93-055.
H21-P173	20	Shutdown instrument panel	Reactor Building el 130 ft	Interaction concern from unsecured file cabinet next to panel Panel contains essential relays.	Move and secure file cabinet.	MWO.

TABLE 2
(SHEET 3 OF 5)

Equipment ID Number	Equip. Class	Equipment Description	Equipment Location	Outlier Description	Outlier Resolution	Modification Method
H21-P202 H21-P232	20	Diesel generator relay panels	Diesel Generator Building el 130 ft	1. Bolt bending concern due to gap between base of panel and equipment pad. 2. Potential impact between adjacent panels containing essential relays.	1. Modify anchorage. 2. Connect panels together to prevent impact.	1. DCR 90-015. 2. DCR 90-015.
H21-P230 H21-P231	20	Diesel generator relay panels	Diesel Generator Building el 130 ft	Potential impact between panels containing essential relays.	Connect panels together to prevent impact.	DCR 90-015.
H21-P256	20	MOV and fuel pump control panel 1B	Diesel Generator Building el 130 ft	Potential impact between wall mounted panel containing essential relays and wall due to missing anchor bolts.	Modify anchorage.	DCR 90-015.
P41-N200A P41-N200B	18	Plant service water strainer differential pressure switches	Intake structure el 89 ft	Anchor bolt nuts are corroded.	Modify anchorage.	DCR 90-148.
P41-N520 P41-N521	18	Instrument on rack	Control Building el 180 ft	Corroded bolts.	Repair or modify anchorage.	DCR 93-055.
R24-S002 R24-S003 R24-S029 R24-S031	1	Motor control center	Control Building el 180 ft	Anchorage does not meet GIP screening criteria.	Modify anchorage.	DCR 93-055.
R24-S009 R24-S010	1	600/208V motor control center	Intake structure el 111 ft	Anchorage does not meet GIP screening criteria.	Modify anchorage.	DCR 90-012.
R24-S025 R24-S026 R24-S027	1	600/208V motor control center	Diesel Generator Building el 130 ft	Anchorage does not meet GIP screening criteria.	Modify anchorage.	DCR 88-334.

TABLE 2
(SHEET 4 OF 5)

Equipment ID Number	Equip. Class	Equipment Description	Equipment Location	Outlier Description	Outlier Resolution	Modification Method
R42-S001A R42-S001B	15 15	125/250V station service batteries	Control Building el 112 ft	Potential interaction with overhead light fixtures.	Tie up light fixtures to prevent falling.	DCR 90-010.
R42-S002A R42-S002B R42-S002C	15	125V diesel system batteries	Diesel Generator Building el 130 ft	1. Bolt bending concern due to gap between base of support members and floor. 2. Potential interaction with overhead light fixtures.	1. Modify anchorage. 2. Tie up light fixtures to prevent falling.	1. DCR 89-261. 2. DCR 90-010.
R42-S032A	16	125V battery charger	Diesel Generator Building el 130 ft	Circuit cards inside battery charger not restrained.	Install circuit card restraining bracket.	MWO.
R43-P001A R43-P001B R43-P001C	20 20 20	Diesel generator control panel	Diesel Generator Building el 130 ft	Potential interaction with overhead light fixtures.	Tie up light fixtures to prevent falling.	DCR 90-010.
R43-S001A R43-S001B R43-S001C	17	Diesel generator	Diesel Generator Building el 130 ft	Springs on the vibration isolation mounts for the engine gauge panel are bent. The engine gauge panel is mounted on the diesel generator skid.	Replace vibration isolation mounts with new mounts of a different design.	DCR 88-172.
R44-S003	16	DC/AC inverter	Control Building el 147 ft	Potential interaction with overhead light fixtures.	Tie up light fixtures to prevent falling.	DCR 90-010.
S11-S012	4	4160/600V station service tansformer	Diesel Generator Building el 130 ft	Missing anchor bolt.	Modify anchorage.	DCR 90-011.
T41-B002A	10	RHR/Core spray pump room cooler	Reactor Building el 107 ft	Bolt bending concern due to gap between base of cooler support member and steel support beam.	Modify anchorage.	DCR 89-202.
T41-B003B	10	RHR/Core spray pump room cooler	Reactor Building el 95 ft	1. Bolt bending concern due to gap between base of cooler support member and steel support beam. 2. Bolts on isolation mounts are not tight.	1. Modify anchorage. 2. Tighten bolts.	1. DCR 89-202. 2. DCR 89-202.

Equipment ID Number	Equip. Class	Equipment Description	Equipment Location	Outlier Description	Outlier Resolution	Modification Method
X41-RH	Relays	Potter & Brumfield PR11DY relays located in panels X43-P006A, P006B, and P006C	Diesel Generator Building el 130 ft	Essential relays not verified for chatter.	Replace relays	DCR 93-055.
X43-P006B	20	Relay panel	Diesel Generator Building el 130 ft	Anchor bolt will not tighten.	Replace bolt.	MWO.
Z41-B003A Z41-B003C	10	Air handling unit	Control Building el 180 ft	Anchorage does not meet GIP screening criteria.	Modify anchorage.	DCR 93-055.
Z41-B003B	10	Air handling unit	Control Building el 180 ft	1. Anchorage does not meet GIP screening criteria. 2. Bolt bending concern due to gap between base of panel and equipment pad.	1.Modify anchorage. 2.Modify anchorage.	1. DCR 93-055. 2. DCR 93-055.
Z41-B008A Z41-B008C	11	Condensing unit	Control Building el 180 ft	Poor installation detail caused poor load path which could induce bolt bending.	Grout in bolt sleeves.	MWO.
Z41-B008B	11	Condensing unit	Control Building el 180 ft	1. Poor installation detail caused poor load path which could induce bolt bending. 2. Potential interaction with pipe insulation.	1. Grout in bolt sleeves. 2. Remove insulation.	1. MWO. 2. MWO.
Z41-F009B	7	Air-actuated damper	Control Building el 180 ft	Air line support inadequate.	Add supports.	DCR 93-055.
2H21-P231	20	Diesel generator relay panel	Diesel Generator Building el 130 ft	Potential impact between panels containing essential relays.	Connect panels together to prevent impact.	DCR 90-149.
2R22-S006	3	4160V switchgear emergency bus 2F	Diesel Generator Building el 140 ft	Anchorage does not meet GIP screening criteria.	Modify anchorage.	DCR 90-013.

TABLE 3

DESCRIPTION OF EQUIPMENT OUTLIERS
PLANT HATCH UNIT 2
(SHEET 1 OF 4)

Equipment ID Number	Equipment Class	Equipment Description	Equipment Location	Outlier Description	Outlier Resolution	Modification Method
2E11-C002B	06	RHR pump 2B	Reactor Building el 87 ft	Potential interaction with adjacent beam.	Trim beam flange to provide adequate clearance.	DCR 94-017
2H11-P601 2H11-P606 2H11-P614 2H11-P652	20	Control room panel	Control Building el 164 ft	Potential interaction from HVAC diffuser in ceiling.	Restrain diffuser.	DCR 94-017
2H11-P602 2H11-P650	20	Control room panel	Control Building el 164 ft	1. Potential interaction from HVAC diffuser in ceiling. 2. Potential interaction from nearby furniture.	1. Restrain diffuser. 2. Remove or restrain furniture.	1. DCR 94-017 2. MWO
2H11-P603	20	Control room panel	Control Building el 164 ft	Potential interaction from nearby furniture.	Remove or restrain furniture.	MWO
2H11-P609	20	Control room panel	Control Building el 164 ft	1. Potential interaction from HVAC diffuser in ceiling. 2. Loose cable in panel could potentially impact relays.	1. Restrain diffuser. 2. Restrain cable.	1. DCR 94-017 2. MWO
2H11-P611	20	Control room panel	Control Building el 164 ft	Broken door latch could cause door to rattle.	Repair or replace door latch.	MWO
2H11-P612 2H11-P613	20	Control room panel	Control Building el 164 ft	Instruments on slides not restrained.	Repair or replace retaining clips.	MWO
2H11-P622	20	Control room panel	Control Building el 164 ft	1. Potential interaction from HVAC diffuser in ceiling. 2. Gap under panel could potentially cause relay chatter.	1. Restrain diffuser. 2. Install shims or grout under panel.	1. DCR 94-017 2. MWO

TABLE 3
(SHEET 2 OF 4)

Equipment ID Number	Equipment Class	Equipment Description	Equipment Location	Outlier Description	Outlier Resolution	Modification Method
2H11-P656	20	Control room panel	Control Building el 164 ft	Potential interaction from adjacent panel 2H11-P663 which is supported on isolators.	Modify panel 2H11-P663 anchorage.	DCR 94-017
2H11-P664	20	Control room panel	Control Building el 164 ft	1. Potential interaction from materials stored near panel. 2. Relay has cracked case and missing retainer clips. 3. Anchorage and load path do not meet GIP screening criteria.	1. Remove or restrain materials. 2. Repair or replace relay. 3. Modify anchorage.	1. MWO 2. MWO 3. DCR 94-017
2H11-P670 2H11-P671 2H11-P672 2H11-P673 2H11-P674 2H11-P675 2H11-P676 2H11-P677 2H11-P678 2H11-P679	20	Control room panel	Control Building el 164 ft	Anchorage and load path for these panels does not meet GIP screening criteria. Panels 2H11-P674, P675, and P679 are on the SSEL; panels P670, P671, P672, P673, P676, P677, and P678 are not on the SSEL but could potentially impact other SSEL panels.	Modify anchorage for panels 2H11-P670 through 2H11-P679.	DCR 94-017
2H11-P691 2H11-P700	20	Control room panel	Control Building el 164 ft	SRT could not confirm that equipment pad has reinforcing and adequate load path to the floor slab.	Modify anchorage or add brace at top of panel.	DCR 94-017
2H21-P200	20	Diesel generator relay panel	Diesel Generator Building el 130 ft	1. Potential impact with adjacent panel and MCC not bolted together. 2. Potential interaction from overhead light fixture.	1. Connect panels and MCC together to prevent impact. 2. Tie up light fixture to prevent falling.	1. DCR 91-145 2. DCR 90-10

TABLE 3
(SHEET 3 OF 4)

Equipment ID Number	Equipment Class	Equipment Description	Equipment Location	Outlier Description	Outlier Resolution	Modification Method
2H21-P202	20	Diesel generator relay panel	Diesel Generator Building el 130 ft	Potential impact with adjacent panel and MCC not bolted together.	Connect panels and MCC together to prevent impact.	DCR 91-144
2H21-P230	20	Diesel generator relay panel	Diesel Generator Building el 130 ft	1. Relays mounted on flexible plate inside panel could potentially cause relay chatter. 2. Potential interaction from overhead light fixture.	1. Modify or replace mounting plate. 2. Tie up light fixture to prevent falling.	1. DCR 94-017. 2. DCR 90-10
2H21-P231	20	Diesel generator relay panel	Diesel Generator Building el 130 ft	Relays mounted on flexible plate inside panel could potentially cause relay chatter.	Modify or replace mounting plate.	Pending DCR 94-017
2H21-P232	20	Diesel generator relay panel	Diesel Generator Building el 130 ft	1. Potential impact with adjacent panel not bolted together. 2. Relays mounted on flexible plate. inside panel could potentially cause relay chatter.	1. Connect panels together to prevent impact. 2. Modify or replace mounting plate.	1.DCR 91-144 2. DCR 94-017
2P41-N303A 2P41-N303B	18	PSW discharge pressure transmitter	Intake structure valve pit el 88 ft	Excessive corrosion on base plate and anchor bolts.	Modify base plate and anchorage.	DCR 94-017
2R22-S005	03	4160V station service switchgear	Diesel Generator Building el 130 ft	1. Inadequate load path. 2. Potential interaction from overhead light fixture.	1. Install additional anchorage. 2. Tie up light fixture to prevent falling.	1. DCR 94-017 2. DCR 90-10
2R22-S007	03	4160V station service switchgear	Diesel Generator Building el 130 ft	Inadequate load path.	Install additional anchorage.	DCR 94-017
2R24-S009	01	Motor control center	Intake structure el 111 ft	Anchorage does not meet GIP screening criteria.	Modify anchorage.	DCR 94-017
2R24-S012A 2R24-S012B	01	Motor control center	Reactor Building el 164 ft	1. Potential impact between 2R24-S012A and B which are not bolted together. 2. Center sections of 2R24-S012B are not bolted together.	1. Connect 2R24-S012A to 2R24-S012B. 2. Connect center sections of 2R24-S012B to prevent impact.	1. DCR 94-017 2. MWO

TABLE 3
(SHEET 4 OF 4)

Equipment ID Number	Equipment Class	Equipment Description	Equipment Location	Outlier Description	Outlier Resolution	Modification Method
2R24-S025 2R24-S027	01	Motor control center	Diesel Generator Building el 130 ft	Anchorage does not meet GIP screening criteria.	Modify anchorage.	DCR 91-144 & 91-145
2T41-B005B	10	HPCI pump room cooler	Reactor Building el 87 ft	Overhead duct supports could potentially collapse.	Modify duct supports.	DCR 94-017
2T48-A001	21	Nitrogen storage tank	Yard el 130 ft	Wood roof structure could potentially fall on tank and attached piping.	Modify roof structure to prevent collapse.	MDC 94-5028
2X41-RH	Relay	Potter & Brumfield PR11DY relays located in panels 2X43-P003A and 2X43-P003B	Diesel Generator Building el 130 ft	Essential relays not verified for chatter.	Replace relays.	DCR 94-017