SECTION 6

LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

6.1 IPEEE PROGRAM ORGANIZATION

Management of the overall IPEEE project at CP&L was provided by Mr. R. E. Oliver, Manager-PWR Safety Analysis, who is responsible for all PRA related work performed for the two PWR plants owned and operated by CP&L.

Responsibility for the technical aspects of the project was divided by technical area. Mr. R. L. Knott was responsible for the seismic analysis, and Mr. A. J. Howe was responsible for the fire analysis and the analysis of Other External Events.

6.1.1 Seismic Analysis

The seismic review was performed through the efforts of a CP&L project engineer and a site project manager. The project engineer facilitated the completion of engineering activities while the project manager provided for effective plant interface. The project engineer and project manager worked closely to coordinate site walkdowns, implement repairs and plan for modifications. CP&L's consultants for engineering activities were as follows:

- EQE International, Inc. (EQE)
- System and Reliability Optimization Services, Inc. (SAROS)
- Vectra Technologies, Inc.

The seismic analysis required various organization structures depending on the task being performed. A summary of the various responsibilities by task is addressed below.

Safe Shutdown Equipment List (SSEL) Development:

The SSEL development was facilitated by a preliminary walkthrough by SAROS, CP&L and EQE personnel to search for potential low seismic capacity components. The success path logic and supporting information for SSEL development was completed by Ricky Summit (formerly SAROS) and reviewed by CP&L.

Seismic Walkdowns and Reviews:

A seismic review team (SRT) was assembled following the guidance provided in EPRI NP-6041 drawing on the experience and expertise of EQE and CP&L personnel.

Each walkdown team included a minimum of two SRT members who had completed the Seismic Qualification Utility Group (SQUG) Walkdown Screening and Seismic Evaluation training course, as well as EPRI's add-on training for IPEEE. Joint walkdown teams generally consisted of at least one EQE Engineer and at least one CP&L Engineer. Component screening and HCLPF analysis candidate selection was performed jointly between CP&L and EQE. HCLPF calculations were performed by EQE and reviewed by CP&L.

Relay Evaluation:

The relay evaluations were primarily performed by SAROS. However, early involvement was provided by operations personnel.

Peer Review:

The seismic peer review was performed by Vectra Technologies, Inc. CP&L and EQE supported the peer reviews by participating in the site walkdown review and providing responses for reviewer questions.

6.1.2 Fires and Other External Events

The analysis was, for the most part, performed by NUS at their Gaithersburg office. In order to ensure that CP&L personnel are fully conversant with the IPEEE methods and are in a position to fully integrate the knowledge gained from performing the work into operating procedures, training programs and appropriate hardware changes, a cognizant CP&L engineer was appointed to be the point of contact throughout the study, and CP&L engineers performed an in-depth review of each of the separate analyses that make up the study, and outlined below.

- Qualitative fire area screening analyses
- Fire frequency analyses
- Deterministic fire modeling assumptions
- Fire induced accident sequence analyses
- Human reliability/recovery action analyses
- Other external events analyses

In addition, CP&L engineers performed the quantification of the conditional core damage probabilities for the various plant damage states that were identified during the course of performing the fire analysis.

6.2 COMPOSITION OF THE INDEPENDENT REVIEW TEAM

An independent review team considered the final results of the IPEEE analysis in order to assess potential vulnerabilities, evaluate alternatives to address them and recommend actions to resolve severe accident issues using the NUMARC closure guidelines. The composition of the team is

shown below. The depth of experience of the plant operations, training, fire protection, licensing and nuclear engineering personnel assigned to this team ensured appropriate disposition of the issues identified in the IPEEE.

Corporate Support

Fred A. Emerson, Director - NEI Regulatory and INPO Affairs
Andrew J. Howe, Project Engineer - Risk Assessment
Rudy E. Oliver, Manager - PWR Safety Analysis

SHNPP Staff

David L. Markle, Senior Specialist - Fire Protection
James F. Nevill, Manager - Design Engineering
A. Wayne Powell, Manager - Operations Training
Robert L. Prunty, Manager - Licensing
Lewis S. Rowell, Project Engineer - Licensing
Vann Stephenson, Manager - Civil Engineering
Anthony Williams, Manager - Operations

The SHNPP and corporate support staff groups were convened to review the technical assumptions, bases, results, and conclusions of the analyses, and either provide confirmation that the results appeared valid or provide recommendations for revising the analyses based on adjusted assumptions. The results were then reviewed by a group of cognizant managers from corporate and SHNPP management. The group then decided (using the NUMARC 91-04 criteria for IPE closure and a qualitative cost-effectiveness criterion) how best to resolve the issues. Possible resolution pathways for each sequence with a core damage frequency above 1E-6/year ranged from modifications to procedure changes or consideration for inclusion in Severe Accident Management Guidance to be developed at a later date. All sequences with core damage frequencies below 1E-6 were not evaluated further, in accordance with the NUMARC 91-04 guidelines. Finally, the results and conclusions were reviewed and accepted by plant senior management.

SECTION 7

PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

As this IPEEE has confirmed, SHNPP has been designed to cope with the occurrence of energetic external events and internal fires. The plant is located in such an area that there are few significant external hazards, and for those that are possible, such as tornadoes, the plant is well designed by protecting equipment required for safe shutdown.

In terms of fire hazard, the risk is minimized by a combination of features including a minimization of potential fire sources, and mitgation of the impact of fires by a combination of measures, which include detection, suppression, and separation of potential targets from the fire sources.

Because of these design features, the IPEEE has been able to demonstrate that there are no significant issues that need to be resolved. The fire analysis identified three fire scenarios that, on application of NUMARC evaluation criteria, require Severe Accident Management Guidelines (SAMG) be in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.

The scenarios are the following:

Switchgear Room B (Fire Area 1-A-SWGRB)

This fire area is located at 286' elevation of the Auxiliary Building. It houses equipment and cabling associated with the safety train B and other non-safety equipment. Significant fire ignition sources for this area include electrical cabinets, transformers, and battery chargers. Cable insulation is the primary source of combustible material for this area. The transient combustible loading for this area is negligible.

The dominant scenario for this area results from fires originating in any bus that is not suppressed within the cabinet, and therefore has the potential for impacting cables from several cabinets. Such fires were, conservatively, assumed to be able to ignite substantial quantity of combustibles before they could be suppressed. The contribution of these fires to CDF was estimated to be 4.0E-6 per year.

Switchgear Room A (Fire Area 1-A-SWGRA)

This fire area is similar to Switchgear Room B, but houses equipment and cabling associated with the safety train A and other non-safety equipment. The contribution of fires in this area to CDF was estimated to be 3.1E-6 per year.

The independent review team identified in section 6.4 reviewed the fire scenarios for the two switchgear rooms. No cost effective modifications were identified which could substantially reduce the likelihood of these fire scenarios. The current plant fire response procedures and operating procedures for plant shutdown after a fire adequately address these scenarios, and no enhancements to the procedures were required.

Control Room Scenarios 1D1 and 6B

The SHNPP Control Room Area is located at elevation 305' of the Auxiliary Building. The area contains control panels, computer consoles, radiation monitoring panels, alarms, incore instrumentation, desk relay panels, exhaust fans, a component cooling water surge tank and associated controls, wiring in conduit, a kitchen, and an office area.

The two scenarios that are not screened out are assumed to require control room evacuation with control of safe shutdown components from the ACP. Operator action at the ACP includes the closing of the appropriate PORV block valve if it is necessary to mitigate a LOCA resulting from a spurious opening of a PORV.

- Scenario 1D1—This scenario results in loss of control for the AFW and ESW systems from the main control room. In addition, control functions for many train A components (including those for feed and bleed) are unavailable in the control room. Shutdown from the ACP is assumed to be necessary. A potential LOCA can be mitigated by the closing of the appropriate PORV Block valve from the ACP. It was conservatively assumed that 10% of all fires in this scenario will result in hot shorts causing a PORV to spuriously open. Thus, the scenario results in a transient with a frequency of 5.01E-5 per year, and a small LOCA with a frequency of 5.57E-6 per year. Even though it is not explicitly identified as an entry condition into AOP-004, it was assumed that the inability to control AFW and ESW from the control room will lead the operators to the remote shutdown procedure. It was further assumed that, using the values for conditional core damage probabilities given in section 4.6.3.4, the human error probability for failing to take control is dominant. The total core damage frequency is estimated as 1.25E-6 per year.

- Scenario 6B—This scenario involves a fire in any cabinet that is not suppressed within 15 minutes. The assumption here is that the effects of smoke produced will necessitate control room evacuation. Plant shutdown is assumed from the ACP. This scenario does not involve damage to any equipment, but requires an orderly shutdown from the ACP. The frequency of core damage is estimated to be 3.0E-6 per year.

The frequency of core damage for each scenario is comprised of a frequency of the fire scenario combined with a human error probability for failure to successfully shut down the plant at the ACP. The independent review team identified in section 6.4 reviewed the fire scenarios for this area. No cost effective modifications were identified which could substantially reduce the likelihood of these fire scenarios. The team identified an enhancement to the procedure for

remote shutdown (outside of the main control room) which would improve the response to these fire scenarios. The procedure will be revised to specifically check the status of the pressurizer PORVs after transfer to the ACP to require closure of a block valve if necessary to isolate a failed open relief valve. The current results reported in this document do not credit this procedural enhancement. This procedure change will be implemented prior to startup after refueling outage 6, which is currently scheduled for fall 1995.

SECTION 8

SUMMARY AND CONCLUSIONS

8.1 SUMMARY

8.1.1 Overview of IPEEE

Carolina Power and Light (CP&L) has completed an examination of the potential for events external to the plant to cause core damage accidents at its Shearon Harris Nuclear Power Plant, Unit No. 1 (SHNPP). This report describes the results of the examination and illustrates the acomplishment of one of the principal goals of the study, namely compliance with the Nuclear Regulatory Commission's Generic Letter 88-20, Supplement 4 (NRC, 1991), requesting that every licensee conduct an Individual Plant Examination of External Events (IPEEE). This analysis complements the analysis presented in the Individual Plant Examination (IPE) (CP&L, 1993), which addressed internal initiating events. By the performance of this project, CP&L has achieved for the most part the four primary objectives of the IPEEE, which were, for initiating events resulting from events external to the plant systems:

- to develop an appreciation of severe accident behavior,
- to understand the most likely severe accident sequences that could occur at the plant under full power conditions,
- to gain a qualitative understanding of the overall likelihood of core damage and fission product releases, and
- if necessary, to reduce the overall likelihood of core damage and fission product release by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

The results of this study are not however directly comparable with those of the IPE. The methodology used to perform the IPE is based on a systems analysis approach that has a achieved an accepted degree of maturity. The analysis of external initiating events, by contrast, has not reached the same degree of maturity. For example, some of the potentially damaging external initiating events have very low frequencies that cannot be estimated using actuarial data without considerable extrapolation, so that the frequency estimates are subject to a large uncertainty. Many of the events can occur with a range of severity with the damage potential being a function of that severity. Analyzing the impact of such events can be very complex and time consuming. Because of this, the methods that have been developed to analyze the impact of external initiating events are essentially screening analyses, designed either to identify the

most significant contributors, while minimizing the need for detailed analyses, or to identify specific weaknesses without explicitly estimating risk.

The method chosen to analyze the impact of seismic events, the seismic margin method, is the latter type of analysis. There is no estimation of core damage frequency. Instead, the analysis is an assessment of whether the plant has sufficient margin over and above the design basis to withstand what is known as the review level earthquake. The analysis of the other external events for SHNPP is essentially a confirmation that the plant meets the Standard Review Plan (NRC, 1975) criteria, and again does not require that core damage frequency be evaluated.

The fire analysis does, on the other hand, result in the evaluation of the core damage frequencies from a set of fire initiated scenarios. However, even in this case, the core damage frequency is not evaluated in the same way as was done for the internal initiating events. The analysis is based on a successive screening approach, at each stage of which, fire scenarios are screened from further consideration on the basis that a conservative analysis shows that the frequency of core damage is less than 1E-6 per year. However, since for scenarios that are screened, the analysis is not further refined, the degree of conservatism is not estimated, and therefore it would be inaccurate to sum the screening core damage frequencies to obtain the overall core damage frequency. Instead, the analysis has been used to identify the scenarios that have the highest likelihood of leading to core damage.

There is an additional difference in that, for the purposes of screening, and for comparison with the Severe Accident Issue Closure Guidelines (NUMARC, 1992), the sequences in the IPE were grouped by functional type. In the fire analysis they were grouped by fire location. The reason for this is that it is the vulnerable locations that are of interest.

8.1.2 Results

The only items resulting fron the seismic margin assessment that are outstanding are minor repairs or modifications which will be completed prior to startup from refueling outage 7, currently scheduled for Spring 1997. No vulnerabilities were identified during the analysis of the other external events.

Only four fire scenarios, one in each switchgear room and two in the control room, were identified as having a contribution to core damage frequency greater than 1E-6 per year. These are described below:

Area 1-A-SWGRB

This fire area is located at 286' elevation of the Auxiliary Building. It houses equipment and cabling associated with the safety train B and other non-safety equipment. Significant fire ignition sources for this area include electrical cabinets, transformers, and battery chargers. Cable insulation is the primary source of combustible material for this area. The transient combustible loading for this area is negligible.

Fire protection consists of early warning ionization detection located throughout the fire area. Hose stations, fire extinguishers and manual alarm stations are located in and adjacent to the area. Safety train A, powered by equipment located in the Switchgear Room A fire area, will be relied upon for plant shutdown in case of a fire in this fire area.

The dominant scenario for this area (scenario 2) results from fires originating in any bus that is not suppressed within the cabinet, and therefore has the potential for impacting cables from several cabinets. Such fires were, conservatively, assumed to be able to ignite a substantial quantity of combustibles before they could be suppressed. The impact of such fires was assumed to be loss of the entire division B safe shutdown path (i.e. loss of 1B-SB and 1E buses and all equipment that are powered from these buses). In addition, bus 1D was assumed to be initially de-energized due to either loss of control power to or spurious opening of its supply breaker. Operator action to locally close the breaker and restore offsite AC power to bus 1D (and therefore to bus 1A-SA), as directed by emergency procedures, was credited.

The frequency of this fire was estimated as:

$$F_{S2} = F_{sf} * P_{fs}$$

where F_{sf} = frequency of significant electrical cabinet induced fires = 3.5E-3
P_{fs} = probability of failure to suppress = 0.1 (EPRI, 1992b)

Therefore, F_{S2} = 3.5E-3 * 0.1 = 3.5E-4 per year.

For cases where operator recovery of offsite power to bus 1D (and hence to 1A-SA) is successful, the consequence of the fire is limited to the loss of the 1B-SB and 1E buses. The CCDP for this case is 7.1E-3 and the resultant CDF was estimated as:

$$CDF_{S2,1} = F_{S2} * CCDP_{S2,1} = 3.5E\text{-}4 * 7.1E\text{-}3 = 2.5E\text{-}6 \text{ per year.}$$

For cases where operator recovery of offsite power is not successful (from the IPE, this HEP is 1E-2), the CCDP is 7.5E-2, and the resultant CDF was estimated as:

$$CDF_{S2,2} = F_{S2} * HEP_{S2} * CCDP_{S2,2} = 3.5E\text{-}4 * 1E\text{-}2 * 7.5E\text{-}2 = 2.6E\text{-}7 \text{ per year.}$$

Thus the total contribution from this scenario to CDF is:

$$CDF_{S2} = 2.5E\text{-}6 + 2.6E\text{-}7 = 2.8E\text{-}6 \text{ per year.}$$

The second most significant scenario is that which results in loss of the 1B-SB bus from fires contained within the cabinets in which they originate. These contributions are estimated to add 1.1E-6 per year to the CDF. The total contribution to CDF for Switchgear Room B is therefore estimated to be 4.0E-6 per year (including a small contribution from welding/cutting fires).

Area 1-A-SWGRA

This area is similar to Switchgear Room B, except that there is no potential for the loss of the turbine-driven AFW pump, and no potential for loss of offsite power to division B. The contribution to CDF from the dominant scenarios is 2.6E-6 per year.

Control Room Scenarios 1D1 and 6B

The SHNPP control room is located at 305' elevation of the Auxiliary Building and contains control panels, computer consoles, radiation monitoring panels, alarms, incore instrumentation, desk relay panels, exhaust fans, a component cooling water surge tank and associated controls, wiring in conduit, a kitchen, and an office area.

The fire postulated for this area assumes ignition and subsequent development into the most severe single fire expected in the area of localized concentrations of combustibles permanently present in the area. Propagation of the postulated fire will be reduced by early detection using ionization-type smoke detectors installed at the ceiling and in the panels of the main control board. The automatic detection system senses products of combustion generated by the incipient fire and alerts employees both at the location of the local fire detection control panel and in the control room via the communications room. As such, manual fire response can be initiated promptly; this fire area is permanently occupied. In addition, ready access is provided to the area from adjacent plant areas facilitating initial use of area fire extinguishers on incipient fires and supplemental use of standpipe hose lines on developing fires by employees responding to the fire. The postulated fire is not considered to have sufficient potential for spread to cause failure of redundant safety-related plant equipment and associated cabling and controls.

The two scenarios that are not screened out are described below. The scenarios are assumed to require control room evacuation with control of safe shutdown components from the auxiliary control panel (ACP). Operator action at the ACP includes the closing of the appropriate power-operated relief valve (PORV) block valve if it is necessary to mitigate a LOCA resulting from a spurious opening of a PORV.

- Scenario 1D1—This scenario results in loss of control for the auxiliary feedwater (AFW) and emergency service water (ESW) systems from the main control room. In addition, control functions for many train A components (including those for feed and bleed) are unavailable in the control room. Shutdown from the ACP is assumed to be necessary. A potential LOCA can be mitigated by the closing of the appropriate PORV block valve from the ACP. It was conservatively assumed that 10% of all fires in this scenario will result in hot shorts causing a PORV to spuriously open. Thus, the scenario results in a transient with a frequency of approximately 5.01E-5 per year, and a small LOCA with a frequency of 5.57E-6 per year. Even though it is not explicitly identified as an entry condition, it was assumed that the inability to control AFW and ESW from the control room will lead the operators to the remote shutdown procedure. It was further assumed that, using the

values for CCDPs given in section 4.6.3.4, the human error probability for failing to take control is dominant. The total core damage frequency is estimated as 1.25E-6 per year.

- Scenario 6B - This scenario involves a fire in any cabinet that is not suppressed within 15 minutes. The assumption here is that the effects of smoke produced will necessitate control room evacuation. Plant shutdown is assumed from the ACP. This scenario does not involve damage to any equipment, but requires an orderly shutdown from the ACP. The frequency of core damage is estimated to be 3.0E-6 per year.

CP&L has reviewed these three fire scenarios and has determined that no plant modifications or administrative changes need to be made. However, the procedure for remote shutdown (outside of the main control room) will be enhanced to check the status of the pressurizer PORVs after transfer to the ACP and require closure of a block valve if necessary to isolate a failed open relief valve. The current results reported in this document do not credit this procedural enhancement. This procedure change will be implemented prior to startup after refueling outage 6, which is currently scheduled for fall 1995.

8.2 CONCLUSIONS

The IPEEE has demonstrated that the SHNPP has no significant vulnerabilities to external events. By performing this IPEEE, CP&L has not only addressed the requirements of the Generic Letter 88-20, Supplement 4 (NRC, 1991), but has also addressed other regulatory requirements.

Three programs, i.e., (1) the external event portion of USI A-45, (2) GI-131, and (3) the Eastern U.S. Seismicity issue, are subsumed in the IPEEE.

Any vulnerabilities associated with decay heat removal (USI-A-45) would have been revealed and resolved during this process. By virtue of the fact that no seismic vulnerabilities were uncovered in the seismic margin study, and that the safe shutdown paths analyzed in that study included equipment for decay heat removal, there are no seismic vulnerabilities specific to decay heat removal. Three of the scenarios identified during the fire analysis involve loss of control or power to the AFW system, and therefore are relevant to the USI-A-45 resolution. In the case of switchgear room fire scenarios, the loss of power to the complete division is a conservative assumption. In the case of the control room fire scenario, the frequency has been assessed to be 1.25E-6 per year, and when compared with the overall CDF, is not a significant contributor.

The Eastern U.S. Seismicity Issue is resolved by the seismic part of the IPEEE. Since CP&L exercised the seismic margins option, the resolution was achieved by an appropriate choice of review level earthquake. GI-131 deals with the seismically induced failure of the flux mapping transfer cart that would lead directly to the rupture of instrumentation tubes at the seal table. Since this is applicable to Westinghouse plants, it is applicable to SHNPP. It has been addressed

in the IPEEE. USI-A-46 has subsumed USI-A-17, "Seismic Interactions in Nuclear Power Plants". Although SHNPP is not an A-46 plant, USI-A-17 was addressed through the seismic walkdown that was performed to meet the requirements of the IPEEE.

The FRSS issues, NUREG/CR-5088, were examined through comparison to standardized checklist questions and through specifically tailored plant walkdowns according to the FIVE Methodology. The FRSS issues are discussed in section 4.8. The issue of seismic-fire interactions has been addressed and is discussed in section 3.1.6.

The revised "Design Probable Maximum Precipitation (PMP)" criteria were assessed with the other external events as requested in Generic Letter 89-22, Supplement 4. The conclusions are presented in section 5.4.

IEN 93-53, Supplement 1 requested that the IPEEE address the lessons learned from the effects of Hurricane Andrew on the Turkey Point Nuclear Generating Station (NRC, 1993). This was addressed during the performance of a walkdown that was conducted to confirm the conclusions of the review of the plant design with respect to Other External Events, as discussed in section 5.

8.3 REFERENCES

(CP&L, 1993), Carolina Power and Light Co., "Shearon Harris Nuclear Power Plant, Unit No. 1, Individual Plant Examination Submittal", August 1993.

(NRC, 1975), USNRC, "Standard Review Plan for Review of Safety Analysis Report for Nuclear Power Plants", NUREG-75/187, December 1975.

(NRC, 1991), USNRC, Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities—10 CFR 50.54(f)", April, 1991.

(NUMARC, 1991), NUMARC, "Severe Accident Issue Closure Guidelines", 91-044, 1991.

SEISMIC IPEEE
SHEARON HARRIS NUCLEAR POWER PLANT

March 1995
Revision 0

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APPROVAL COVER SHEET

TITLE: Seismic IPEEE Harris Nuclear Plant

REPORT NUMBER: 52214-R-001

CLIENT: Carolina Power & Light Company

PROJECT NO.: 52214.15

REVISION: 0

EQE Prepared by: [signature] Date: 3/3/95

EQE Reviewed by: [signature] Date: 3·3·95

EQE Approved by: [signature] Date: 3/3/95

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0	3/95	Original Issue

1. INTRODUCTION AND METHODOLOGY SELECTION

In the Commission policy statement on severe accidents in nuclear power plants issued in 1985, the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on any regulatory requirements for these plants. However, the Commission recognized that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. In 1988 the Commission requested that each licensee conduct an individual plant examination (IPE) for internally initiated events including internal flooding. Many PRAs indicated that, in some instances, the risk from external events could contribute significantly to core damage.

In July 1990, following public comments and a workshop, the Commission issued Supplement 4 to Generic Letter 88-20 (Reference 1) requesting that each licensee conduct an individual plant examination of external events (IPEEE). The general objectives of the IPEEE are similar to that of the IPE - that is, for each licensee (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant under full-power operating conditions, (3) to gain a qualitative understanding of the overall likelihood of core damage and fission product releases, and (4) if necessary, to reduce the overall likelihood of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

The staff has concluded that five external events need to be included specifically in the IPEEE: seismic events, internal fires, high winds, floods, and transportation and nearby facility accidents. This report addresses seismic events.

Acceptable methodologies for performing the seismic IPEEE are summarized in NUREG-1407 (Reference 2). This evaluation may be conducted by performing a

seismic PRA or a Seismic Margins Assessment (SMA). The SMA methodology was designed to demonstrate sufficient margin over the Safe Shutdown Earthquake (SSE) to ensure plant safety and to find any "weak links" that might limit the plant shutdown capacity to safely withstand a seismic event larger than the SSE or lead to seismically induced core damage. The SMA may in turn be performed using the methodology developed by Lawrence Livermore National Laboratories (LLNL), or by Electric Power Research Institute (EPRI). CP&L has opted to perform a SMA using the EPRI methodology (Reference 3).

Harris was placed in the focused-scope category for margin assessment. The basic information used was the 1989 Lawrence Livermore National Laboratory seismic hazard estimates for nuclear power plant locations in the eastern United States (Reference 4) and the EPRI hazard study (Reference 6).

New seismic hazard data were published in October, 1993 that demonstrates that the seismic hazard at existing eastern United States nuclear power plants is much less than what the NRC staff originally believed (Reference 5). The data demonstrate a much lower annual probability of Harris exceeding the 0.15g design basis earthquake based on the 1993 LLNL mean hazard curves than the annual probability of exceeding the 0.3g review level earthquake (RLE) based on the revised hazard curves. Furthermore, the LLNL results confirm that the mean seismic hazard for Harris is lower than the 1989 seismic hazard estimates for the group of plants that were originally designated as reduced-scope.

CP&L elected to complete the Harris SMA following NUREG 1407 and EPRI NP-6041 as a focused-scope plant without schedule delays or major scope changes. The new information and extensive seismic evaluation performed for the recent vintage plant were, however, considered when determining the quantity of components selected for high-confidence-of-low-probability of failure (HCLPF) evaluation and the level of evaluation for issues such as soils, structures and NSSS components.

Detailed plant walkdowns are considered the most cost-effective and beneficial aspect of the SMA program. Harris walkdowns were performed by teams of CP&L and consultant Seismic Review Teams (SRTs) in accordance with EPRI NP-6041. Pre-walkdown activities included prescreening of success path components with available data entered into EHOST, a microcomputer database developed by EQE International. Walkdowns were performed using pen-based computers and the program EWALK. EQE proprietary software EWALK is compatible with EHOST to facilitate efficient data collection and subsequent data management.

These walkdowns identified several housekeeping, maintenance and systems interaction issues that lead to cost effective improvements as a result of the SMA program. Analyses to determine to high confidence of low probability of failure (HCLPF) capacity of selected success path elements confirmed that the plant HCLPF meets or exceeds the 0.3g Review Level Earthquake (RLE).

2. REVIEW OF PLANT INFORMATION

A brief description of the general plant, ground response spectra, structures, equipment, distribution systems is presented below. All the information presented in this section is contained in existing plant licensing documents including the Final Safety Analysis Report (FSAR). The purpose is to provide a review of the plant design.

2.1 GENERAL PLANT DESCRIPTION

The Nuclear Steam Supply System (NSSS) for the Unit is a pressurized water reactor (PWR) consisting of three closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to the "hot" leg of one of the loops. The NSSS, along with the design and fabrication of the initial fuel core, is supplied by Westinghouse Electric Corporation.

The Containment is a steel lined reinforced concrete structure in the form of a vertical right cylinder with a hemispherical dome and a flat base with a recess beneath the reactor vessel. The Containment is designed by Ebasco Services Incorporated, architect/engineer for Harris.

The Unit is designed for an initial licensed power output of 2785 megawatts thermal (Mwt), which includes 10 Mwt from the reactor coolant pumps. This output corresponds to approximately 900 megawatts electric (Mwe). The NSSS is capable of producing approximately 2910 Mwt (approximately 940 Mwe), which includes 10 Mwt from the reactor coolant pumps. Although the license application is for 2785 Mwt, all safety systems, including Containment and engineered safety features, are designed and evaluated for operation at the higher power level of 2910 Mwt.

2.1.1 Site Location and Area

The Harris site is located in the extreme southwest corner of Wake County, North Carolina, and the southeast corner of Chatham County, North Carolina. The city of Raleigh, North Carolina, is approximately 16 miles northeast, and the city of Sanford is about 15 miles southwest.

Maps of the site area are included as Figures 2-1 and 2-2. Indicated on these maps are the site boundary line (which is the same as the station property boundary), the principal plant structures, the exclusion area, and the principal transportation routes. The station requires approximately 10,800 acres. Carolina Power & Light Company owns all land within the site boundary lines. There are no private, residential, industrial, recreational, institutional, or commercial structures (other than those related to plant operation) within this area.

The environment is rural and primarily devoted to farming and dairying. Local industrial activity is centered in an area west-southwest from the plant. Another major center of industrial and research activity is located to the north-northwest.

2.1.2 Hydrology

The plant site is located at the confluence of Buckhorn and Whiteoak Creeks, just north of the Cape Fear River. The power block area is located between Tom Jack and Thomas Creeks. These two creeks are tributaries of Whiteoak Creek; Whiteoak Creek is a tributary of Buckhorn Creek. Figure 2-1 shows a plan of the site development.

The principal water source for the plant is the Main Reservoir which is formed by an impoundment of Buckhorn Creek just below its confluence with Whiteoak Creek. The project design also includes an adjoining and independent Auxiliary Reservoir for emergency cooling purposes. See FSAR Section 2.4 for a more detailed discussion of hydrology.

2.1.3 Geology and Seismology

The region surrounding the site is generally characterized by a gently rolling topography resulting from extensive weathering and erosion of the underlying bedrock. Elevations of the hill tops and ridge crests are mostly between 250 feet and 275 feet (msl) and local relief is generally less than 60 feet. The finished grade elevation is approximately 260 feet.

The site is located in the southeastern part of the Durham Basin, which is in the northern part of the Deep River Triassic Basin. Sediments that underlie much of the southeastern portion of the Durham Basin were placed as alluvial fans and stream channels and flood plain deposits. Below an occasional thin layer of alluvial sand and/or clay, there are from 0 to 15 ft. of residual soil. The depth of weathering below this to sound rock generally varies from about 0 to 15 ft. depending on the type of underlying rock. The foundations have been placed on sound rock.

A small fault was discovered during excavation for the Waste Processing Building. The studies performed showed that this fault is not a capable fault, as documented in the Shearon Harris Fault Investigation Report submitted to the NRC in 1975.

The nearest known fault outside the site is one lying just west of Merry Oaks about three miles to the southwest of the site. Test borings showed nothing that would indicate the development of faults, joints, slickensides or other structural weakness since the late Triassic and early Jurassic time.

Based on historical seismicity, the maximum potential earthquake which might affect the site would be a recurrence of the Charleston, South Carolina earthquake of 1886 which was probably felt as an intensity VI at the site. The largest earthquakes in the site region which are not attributable to any particular geologic structure or seismic zone have been of intensity V. However, it is considered possible that some intensity VII earthquakes in the eastern Piedmont and the Coastal Plain may have been related to exposed or buried Triassic Basins.

Therefore, a shock of intensity VII occurring in the Deep River Basin is considered to be the maximum potential earthquake.

The safe shutdown earthquake is designated as an intensity VII earthquake occurring close to the site. The resulting maximum horizontal ground acceleration at foundation level within the competent bedrock at the site is estimated to be less than 12 percent of gravity. In order to provide an additional margin of conservatism, a value of 15 percent of gravity is assigned as the maximum horizontal ground acceleration. All safety related structures and systems are designed to assure safe plant shutdown for two horizontal excitations and one vertical excitation simultaneously. Seismic Category I systems and components are designed for a minimum of 10 loading cycles under safe shutdown earthquake conditions.

The operating basis earthquake is designated as one with half the accelerations of the safe shutdown earthquake and equivalent to an Intensity VI earthquake near the site. The corresponding horizontal acceleration at foundation level in the bedrock would be less than 7.5 percent of gravity.

The site maximum horizontal ground accelerations for the Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) are 0.150g and 0.075g, respectively. See FSAR Section 2.5 for a more detailed discussion of geology and seismology.

2.2 GROUND RESPONSE SPECTRA

Two earthquake motions were considered in the dynamic analyses of all Seismic Category I structures, systems, subsystems, and equipment: the operating basis earthquake (OBE) and safe shutdown earthquake (SSE). The design value of the maximum horizontal ground acceleration is 0.150g for the safe shutdown earthquake and .075g for the operating basis earthquake.

The design response spectra used for all Seismic Category I structures, systems, and components, except dams and dikes, were developed in accordance with Regulatory Guide 1.60. The horizontal and vertical design response spectra, normalized to 0.150g for the SSE and 0.075g for the OBE, are presented on Figures 2-3 through 2-6 and were applied at the foundation level.

The design response spectra used for the Seismic Category I dams and dikes were based on a modified form of a smoothed response spectra developed from the strong motion record of the 1935 Helena, Montana earthquake, normalized to the maximum horizontal ground accelerations of the safe shutdown earthquake and the operating basis earthquake. This record was obtained from a seismograph that was established on competent bedrock and is, therefore, considered appropriate for the proposed plant site.

The horizontal design response spectra for the dams and dikes, normalized to 0.150g for the SSE and 0.075g for the OBE, are presented on Figures 2-7 and 2-8, respectively. The vertical design response spectra for the dams and dikes, normalized to 0.10g for the SSE and 0.05g for the OBE, are presented on Figures 2-9 and 2-10, respectively. The seismic analysis of the dams and dikes, based on the design response spectra presented on Figures 2-7 through 2-10, is discussed in the FSAR, Section 2.5.6. An evaluation of the behavior of the dams and dikes during an earthquake whose response spectra were developed using the Regulatory Guide 1.60 methodology, is also presented in Section 2.5.6 of the FSAR.

2.3 STRUCTURES

Seismic Category I structures are cast-in-place reinforced concrete structures. The floors are supported on beams and girders which are in turn supported on interior columns and/or exterior walls. Where interior shear walls are installed, the beams and girders are supported on the shear walls. All interior shielding walls and partitions, other than shear walls, are either reinforced concrete or concrete block,

and are not load bearing. The buildings are supported on separate foundation mats 10 feet thick which are founded on suitable rock.

The seismic analyses of the Seismic Category I structures were performed by using the normal mode time-history technique. The structures, considered as seismic systems and analyzed in this manner, are Containment, Reactor Auxiliary Building, Fuel Handling Building, Waste Processing Building, Tank Building, Diesel Generator Building, Diesel Fuel Oil Storage Building, ESWS Intake Structure, ESWS Discharge Structure and ESWS Screening Structure.

These structures are founded on sound rock which has a shear wave velocity of 5600 ft/sec. The lumped mass-spring approach was used to develop the mathematical model for the dynamic analyses of the structures. The mathematical model assumes a single cantilever or multi-cantilever lumped mass system. The lumped masses are connected by weightless elastic bars which represent the stiffness of structural walls and/or columns. Each mathematical model is supported by a mass which represents the foundation mat; the interaction of the foundation mat with the supporting rock medium is represented by linear elastic springs.

The lumped masses are located at floor levels and at any other points where the dynamic responses are important. The dead weights of the structural floor system, steel framing, grating, miscellaneous steel, equipment, piping, and electrical cables and trays (considered as a uniform load distributed over the floor) are included in the lumped mass at the corresponding level. The dead weights of columns and structural walls are evenly distributed between the levels over which they span. The dead weights of block walls are lumped at the levels at which they are supported.

Ebasco's in-house computer program DYNAMIC 2037 was used to generate floor response spectra. The damping values used were 4 percent for structures and 2 percent for rock for OBE and 7 percent and 5 percent, respectively for SSE. The floor response spectra calculations were based on the exact analytical solutions of

the governing differential equations for the successive linear segments of the excitation, specified at equal time intervals. The method is described in detail in Reference 35. The floor response spectra were generated separately for three directions of earthquake motion.

The peaks of the floor response spectra are broadened plus or minus fifteen percent in frequency according to the example shown on Figure 1 of Regulatory Guide 1.122. The broadening on the frequency axis is to account for variations in parameters, such as the material properties of the structures and soil, damping values, soil-structure interaction techniques and approximations in the modeling techniques.

The seismic analysis of all Seismic Category I structures takes into consideration three orthogonal directions of seismic motions; two horizontal and one vertical. The maximum responses to each of the three components of motion are determined separately and combined by the square root of the sum of the squares (SRSS) method to obtain the total seismic responses in accordance with Regulatory Guide 1.92.

The Seismic Category I structures listed above conform with the applicable requirements of the following codes, standards, regulatory guides and specifications listed below:

General Codes and Standards

OSHA - Occupational Safety and Health Administration, Federal Safety Regulations (1975 listing)

NCSBC - North Carolina State Building Code, 1969 Edition

ACI - American Concrete Institute Standards

211.1-1974 Recommended Practices for Selecting Proportions for Normal and Heavy Weight Concrete

301-1975 Specifications for Structural Concrete for Buildings

304-1973 Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete

305-1972 Recommended Practice for Hot Weather Concreting (Use this edition except Paragraph 4.4.3. Comply with ACI 305-1974 Paragraph 4.4.3 only)

306-1966 Recommended Practice for Cold Weather Concreting

309-1974 Recommended Practice for Consolidation of Concrete

315-1974 Manual of Standard Practice for Detailing Reinforced Concrete Structures

318-1971 Building Code Requirements for Reinforced Concrete

347-1968 Recommended Practice for Concrete Formwork

SP-2-1975 Manual of Concrete Inspection

AISC - American Institute of Steel Construction

Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings (AISC Specification) (2/12/69, with Supplements 1- 11/1/70, 2- 12/8/71, and 3- 6/12/74)

ASME - American Society of Mechanical Engineers

"Boiler and Pressure Vessel Code," 1975 Edition

Section II - Material Specifications

Section III, Division 1 - Nuclear Power Plant Components

Subsection ND "Class 3 Components"

Subsection NE "Class MC Components"

Section III, Division 2 - Code for Concrete Reactor Vessels and Containments (as required by ASME Section III, Division 2/ACI 359 Code, exceptions to the ASME Section III, Division 2/ACI 359-74 Code are listed in Appendix 3.8A of the FSAR

AWS - American Welding Society

D 1.1-75 Structural Welding Code, with Revisions 1 (1976) and 2 (1977) for services performed after 4/29/77

NRC Regulatory Guides

1.10 Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures

1.15 Testing of Reinforcing Bars for Category I Concrete Structures

1.18 Structural Acceptance Test for Concrete Primary Containment

1.19 Nondestructive Examination of Primary Containment Liner Welds

1.54 Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants

1.55 Concrete Placement in Category I Structures

1.57 Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

1.60 Design Response Spectra for Seismic Design of Nuclear Power Plants

1.61 Damping Values for Seismic Design of Nuclear Power Plants

1.63 Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants

1.76 Design Basis Tornado for Nuclear Power Plants

1.92 Combination of Modes and Spatial Components in Seismic Response Analysis

1.94 Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel during the Construction Phase of Nuclear Power Plants

1.122 Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components

The following specifications specify the requirements for materials, design criteria, fabrication, erection, inspection, and quality assurance. These specifications, in general, reflect and expand on the requirements set forth in ASME Section III, Division 2/ACI 359 Code.

a) Ebasco Specification CAR-SH-AS-1 "Containment Liner, Air Locks, and Hatch"

b) Ebasco Specification CAR-SH-AS-7 "Structural Steel"

c) Ebasco Specification CAR-SH-M-54 "Mechanical Penetrations"

d) Ebasco Specification CAR-SH-E-30 "Electrical Penetrations"

e) Ebasco Specification CAR-SH-CH-6 "Concrete"

f) Ebasco Specification CAR-SH-CH-7A "Concrete Reinforcing Steel"

g) Ebasco Specification CAR-SH-CH-7 "Weldable Concrete Reinforcing Steel"

h) Ebasco Specification CAR-SH-CH-12 "Waterstops"

i) Ebasco Specification CAR-SH-CH-13 "Waterproofing"

j) Ebasco Specification CAR-SH-CH-15 "Mechanical Splicing of Concrete Reinforcing Steel"

k) Ebasco Specification CAR-SH-CH-16 "Dome Hub Plates and Reinforcing Steel Splice Assembly"

l) Ebasco Specification CAR-SH-CH-22 "Structural Integrity Test of Concrete Containment Building"

2.4 EQUIPMENT SUPPLIED BY THE NSSS VENDOR

Seismic qualification of safety related electrical equipment is demonstrated by either type testing, analysis or a combination of these methods. The choice of qualification method employed by Westinghouse for a particular item of equipment is based upon many factors including; practicability, complexity of equipment, economics, availability of previous seismic qualification to earlier standards, etc. The qualification method employed for a particular item of equipment is identified in the individual equipment qualification document.

The qualification and documentation procedures used for equipment and supports which were purchased prior to March 1, 1977, are in compliance with IEEE 344-1971 and Standard Review Plan 3.10 (Revision 1), Section II.1.a or the Supplemental Qualification Program, Reference NS-CE-692, Letter dated July 10, 1975 from C. Eicheldinger (Westinghouse) to D. B. Vassllo (NRC). The qualification and documentation procedures for equipment and supports purchased on or after March 1, 1977, are in compliance with IEEE 344-1975.

The methods and procedures for equipment qualified in compliance with IEEE 344-1971 and Standard Review Plan 3.10 (Revision 1), Section II.1.a is as follows:

A. Type Test - From 1969 to mid-1974 Westinghouse seismic test procedures employed single axis sine beat inputs in accordance with IEEE 344-1971 to seismically qualify equipment. Much of this early testing was reported in WCAP-7817 and WCAP-7821 as referenced in the FSAR, Table 3.10.1-1. The input form selected by Westinghouse was chosen following an investigation of building responses to seismic events as reported in WCAP-7558, "Seismic Vibration Testing With Sine Beats." Further, this input has been justified with respect to the methods of IEEE 344-1975 and documented in WCAP-8373 "Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974." In addition, Westinghouse has conducted seismic retesting of certain items of equipment as part of the Supplemental Qualification Program. This retesting was performed at the request of the NRC staff on agreed selected items of equipment employing multi-frequency, multi-axis test inputs, Reference WCAP-8624 and WCAP-8373, to demonstrate the conservatism of the original sine-beat test method with respect to the modified methods of testing for complex equipment recommended by IEEE 344-1975.

B. Analysis - The structural integrity of safety related motors is demonstrated by a static seismic analysis in accordance with IEEE 344-1971, with justification. Should analysis fail to show the resonant frequency to be significantly greater than 33 Hz, a test is performed to establish the motor resonant frequency. Motor operability during a seismic event is demonstrated by calculating critical deflections, loads and stresses under various combinations of seismic, gravitational and operational loads. The worst case (maximum) values calculated are tabulated against the allowable values. On combining these stresses, the most unfavorable possibilities are considered in the following areas; 1) maximum rotor deflection, 2) maximum shaft stresses, 3) maximum bearing load and shaft slop at the bearings, 4) maximum stresses in the stator core welds, 5) maximum stresses in the

stator core to frame welds, 6) maximum stresses in the motor mounting bolts and, 7) maximum stresses in the motor feet.

The analytical models employed and the results of the analysis are described in the individual equipment qualification document.

The methods and procedures for equipment qualified in compliance with IEEE 344-1975 is as follows:

A. Type Test - The original single-axis sine beat testing and the additional retesting completed under the Supplemental Test Program have been the subject of generic review by the NRC staff. Both test programs are described later in this section. For equipment which has been previously qualified by the single axis sine beat method and included in the NRC seismic audit and, where required by the NRC staff, the Supplemental Qualification Program (Reference 3.10.2-2), no additional qualification testing is required to demonstrate acceptability to IEEE 344-1975, provided that:

1) The Westinghouse aging evaluation program for aging effects on complex electronic equipment located outside Containment demonstrates there are not deleterious aging phenomena. In the event that the aging evaluation program identifies materials that are marginal, either the materials will be replaced or the projected qualified life will be adjusted.

2) Any changes made to the equipment due to 1) above or due to design modifications do not significantly affect the seismic characteristics of the equipment.

3) The previously employed test inputs can be shown to be conservative with respect to applicable plant specific response spectra. The equipment that requires no additional testing is identified in WCAP-8587, Table 7.1 and the test results in the

applicable Equipment Qualification Data Packages (EQDP's) of Supplement 1 to WCAP-8587.

For equipment tests after July, 1974 (i.e., new designs, equipment not previously qualified, or previously qualified equipment that does not meet 1), 2), and 3), above, seismic qualification by test is performed in accordance with IEEE 344-1975. Where testing is utilized, multi-frequency multi-axis inputs are developed by the general procedures outlined in WCAP-8624 and WCAP-8695. The test results contained in the individual EQDP's of Supplement 1 to WCAP-8587 demonstrate that the measured test response spectrum envelopes the applicable required response spectrum (RRS) defined for generic testing as specified in Section 1 of the EQDP. Qualification for plant specific use is established by verification that the generic RRS specified by Westinghouse envelopes the applicable plant specific response spectrum. Alternative test methods, such as single frequency, single axis inputs, are used in selected cases as permitted by IEEE 344-1975 and Regulatory Guide 1.100.

B. **Analysis** - The structural integrity of safety related motors (Supplement 1 to WCAP-8587, Table 3.10.1-1 EQDP-AE-2 and 3) is demonstrated by a static seismic analysis in accordance with IEEE 344-1975, with justification. Should analysis fail to show the resonant frequency to be significantly greater than 33 Hz, a test is performed to establish the motor resonant frequency. Motor operability during a seismic event is demonstrated by calculating critical deflections, loads and stresses under various combinations of seismic, gravitational and operational loads. The worst case (maximum) values calculated are tabulated against the allowable values. On combining these stresses, the most unfavorable possibilities are considered in the following areas; 1) maximum rotor deflection, 2) maximum shaft stresses, 3) maximum bearing load and shaft slope at the bearings, 4) maximums tresses in the stator core welds, 5) maximum stresses in the

stator core to frame welds, 6) maximum stresses in the motor mounting bolts, and 7) maximum stresses in the motor feet.

The analytical models employed and the results of the analysis are described in Section 4 of the applicable EQDP's, Supplement 1 to WCAP-8587.

2.5 EQUIPMENT SUPPLIED BY OTHER THAN NSSS VENDOR

The safety related electrical (includes instrumentation and control) and mechanical equipment and their supports which are within Ebasco's scope, have been qualified by testing and/or analysis to Seismic Category I requirements to verify their ability to withstand the effects of earthquakes and other applicable accident-related loadings (i.e., dynamic loadings).

In addition, the qualification and documentation procedures for Seismic Category I electrical equipment and their supports have been prepared utilizing the guidance of IEEE 344-1975 and Regulatory Guide 1.100. Such equipment and supports which are Class 1E are qualified in accordance with IEEE 323. IEEE 344 is considered ancillary to IEEE 323 and any exceptions taken to age testing requirements have been evaluated and accepted with applicable justification for the exception.

The purchase specifications for safety related equipment contain seismic input data for the safe shutdown earthquake (SSE) and operating basis earthquake (OBE). This input data consist of floor response spectra (for the appropriate damping values) and/or appropriate "G" values for the various levels of the building (taking into consideration the location of the equipment on the floor) or other mounting locations such as pipes, ducts, etc. Each set of curves consists of two horizontal and one vertical design response spectra curves at the floor elevation of the equipment mounting location.

The equipment supplier's seismic qualification program demonstrates the ability of the equipment to perform its required function during and after the time it is subjected to the forces resulting from the application of five consecutive OBE's

followed by one SSE, with a proper combination of other applicable concurrent loads.

Depending upon the practicability of the method for the type, size, shape, and complexity of the equipment and the reliability of the conclusion, the equipment supplier uses testing, analysis, or a combination of testing and analysis as a method of qualification, as follows:

A. Testing - Testing has been the preferred method of qualification. It is performed by subjecting the equipment to vibratory motions, which conservatively simulate the OBE and SSE responses at the equipment mounting locations. The SSE test is preceded by five events of the OBE. The test input motions are such that the resulting response spectra envelope the design floor response spectra.

   Thermal and radiation aging is performed prior to seismic testing for equipment qualified in accordance with IEEE 323-1974 unless it could be justified that the equipment would not approach the end-of-life condition during the installed life, when subjected to the specified service conditions.

   Instrumentation and electrical equipment are tested in the operational mode and their operability is verified before, during, and after the testing. Test methods described in Section 6.6, "Test Methods," of IEEE 344-1975 are utilized to perform the required qualification testing. The test input motion has generally been of the random type.

B. Analysis - Analysis without testing is accepted when the equipment functional operability can be assured by its structural integrity alone. The procedures described in Sections 5.2 through 5.4 of IEEE 344-1975 are utilized. Component fatigue is checked for the effect of five OBE's and one SSE when the analysis method of qualification is used.

C. **Combination of Testing and Analysis** - When the equipment cannot be qualified by testing or analysis alone because of its size and complexity, a combined testing and analysis method is utilized. Methods described in Section 7 of IEEE 344-1975 are used for qualification.

2.6 PIPING

The stress analysis of Seismic Category I, ASME, Section III, Safety Class 2 and 3 piping is in accordance with ASME B & PV Code, Section III, subarticles NC/ND, and is described below. The design criteria is in accordance with formulations given in subarticle NC/ND-3600.

The seismic analytical procedure using the computer methods, described in Section 3.7.3.8.1.1 of the FSAR, involves an analysis of the piping systems using characteristic spring rates of various rigid constraints and snubbers. Restraint loads based on this analysis were used to design particular controls (i.e., rigid restraints or snubbers). The final design analysis is based on characteristic spring rates.

Equipment having frequencies 33 Hz or higher was assumed rigid for the purpose of analyzing the connected piping. Where the frequency search of equipment indicated a frequency less than 33 Hz, the equipment was considered nonrigid. In such a case, a dynamic model of the equipment having the same response in two orthogonal horizontal directions and the vertical direction was prepared. This dynamic model of the equipment was included in the stress analysis of the piping.

Welded attachments were avoided to the degree practicable. However, where integral attachments could not be avoided, local stress generated in the pipe due to their presence were considered.

Design criteria was based on formulations and allowable stress limits given in ASME Section III, subarticle NC/ND-3600, with load combinations which consider OBE and SSE effects along with other coincident loading conditions as delineated in design specifications.

For the seismic design of piping, the two orthogonal horizontal and vertical loadings were obtained from the floor response spectra that were generated for the appropriate structures and elevations. Relative displacements within buildings and between buildings due to seismic response were also considered in the design of the piping.

An alternative method for broadening of the structure peaks can be based on a probabilistic approach. In the particular case where there is more than one piping frequency located within the frequency range of a widened spectrum peak that is associated with a structural frequency, the floor spectrum curve may be more realistically applied in accordance with the following criterion. Based on the fact that the actual natural frequency of the structure can possibly assume only one single value within the frequency range defined by $f_j \pm \Delta f_j$, but not a range of values, only one of these piping modes can respond with a magnitude indicated by the peak spectral value.

Therefore seismic analysis of piping systems using the broadened floor design response spectra may be accomplished by applying the method of peak shifting as described in the Summer 1984 Addendum of ASME Section III, Appendix N, paragraph N-1226.3(d).

All Seismic Category I, Safety Class 2 and 3 piping systems were seismically analyzed utilizing the methods described in the FSAR, Sections 3.7.3.8.1.1 or 3.7.3.8.1.2. Piping 2-1/2 in. nominal size or larger with design temperature above 275° F, were analyzed by the computer method described in Section 3.7.3.8.1.1. All other piping subsystems were analyzed by either the computer method described in Section 3.7.3.8.1.1 or by the simplified method described in Section 3.7.3.8.1.2.

2.7 BURIED PIPING

Ebasco's design procedure for seismic analysis of Seismic Category I buried piping was based upon Newmark's method, Reference 36 and Hetenyi's theory in beams

on elastic foundations, Reference 37. The analysis procedure included calculation of stresses in the buried portion of the piping due to loads acting on the non-buried portion of the piping inside the building (interaction effect), superimposed on the stresses due to various loads acting on the buried portion of the piping. The resultant stresses were within allowable stress criteria based on the applicable ASME Section III Code.

The buried piping in the yard was analyzed using the above procedure. It was assumed that the piping would be distorted in the same fashion as the earth and, therefore, would assume a sinusoidal wave shape. The wave length and maximum displacement were calculated and the bending moment and stress effects on the piping were obtained. Settlement in the fill along the piping due to differential depth of backfill did not cause any significant stresses in the piping and the resultant stresses were still within allowable stress limits.

At points where piping leaves the ground and is attached to structures, the maximum possible differential movement between the ground and the structure was determined. The differential movement was absorbed either by providing sufficient flexibility in the piping from the ground to the structure or by the use of flexible joints in the piping such as ball joints.

In certain instances, piping which enters structures is supported or anchored within the structure and not at the wall penetration. Wall penetrations were sized to provide sufficient room for differential pipe movement. Flexible membranes provided a moisture seal between the pipe and the structure wall.

The excavated area under the 30 in. diameter and 8 in. diameter service water pipe lines between the Tank Building and Turbine Building walls and the rock or natural ground was backfilled with concrete which will have insignificant differential settlement.

Seismic Category I electrical conduits in the yard were also analyzed by Newmark's method. The electrical conduits and electrical manholes were both buried in fill or

backfill. Both of them would move with the fill with no local differential settlement to cause shear in the conduit. Moreover, the ends of conduits are not anchored in the wall of manholes and pass through sleeves with elastic boots which permit free movement of the conduits in any direction. Settlement in the fill or backfill along the conduit due to differential depth of fill did not cause any significant stresses in the conduit and the resultant stresses were well within the allowable stress limits.

The sleeves at the electrical manhole will permit rotation of the conduit end due to differential settlement of the manhole and the adjacent soils, if any.

The fill in the yard area supporting Seismic Category I piping and conduits is not subject to liquefaction during a seismic event.

2.8 DISTRIBUTION SYSTEMS

Seismic Category I cable trays, conduits, HVAC duct systems and equipment supports were analyzed by the method of modal response spectra. This accounted for the effects of multiple spans and multiple modes on seismic response.

A three dimensional mathematical model was constructed with a sufficient number of dynamic degrees of freedom to closely simulate the dynamic behavior of the subsystems. All of the significant modes of the subsystems were selected for the determination of the seismic response. When the supports for a subsystem were all mounted at the same floor, the relative displacement among supports was not considered. This relative displacement was considered where the supports of the same subsystem were located at different floors.

For the case where the supports of the same subsystem were located in different buildings, the maximum relative displacements among the different supports were considered in the seismic dynamic analysis of the subsystem.

2.9 SEISMIC SPATIAL SYSTEM INTERACTIONS

The following sections describe the minimum clearance requirements between components and the Seismic II/I criteria for Harris.

2.9.1. Interdiscipline Clearances

Various structure systems and components for Harris have been designed with consideration for minimum clearance requirements to prevent detrimental effects to safety related components due to seismic interactions with non-seismically designed components. Nuclear Engineering Department (NED) Design Guide Number DG-II.17, Reference 30 was written to address minimum clearance requirements. This design guide describes the walkdown that took place at the end of construction to identify any clearance violations that may have occurred during installation due to physical constraints.

The design guide provides Tables with minimum clearances between components, eg. 1" diameter conduit to 1" diameter conduit requires 1.187" clearance, 2" diameter pipe to a 2" diameter conduit requires 1.653" clearance, 2" diameter pipe to tube track requires 0.396" clearance, top of cable tray to duct requires 0.528" clearance and etc. Any violations noted during the walkdown were documented and resolved in accordance with the design guide.

The design guide is a current document and is used for continuing compliance to the clearance requirements.

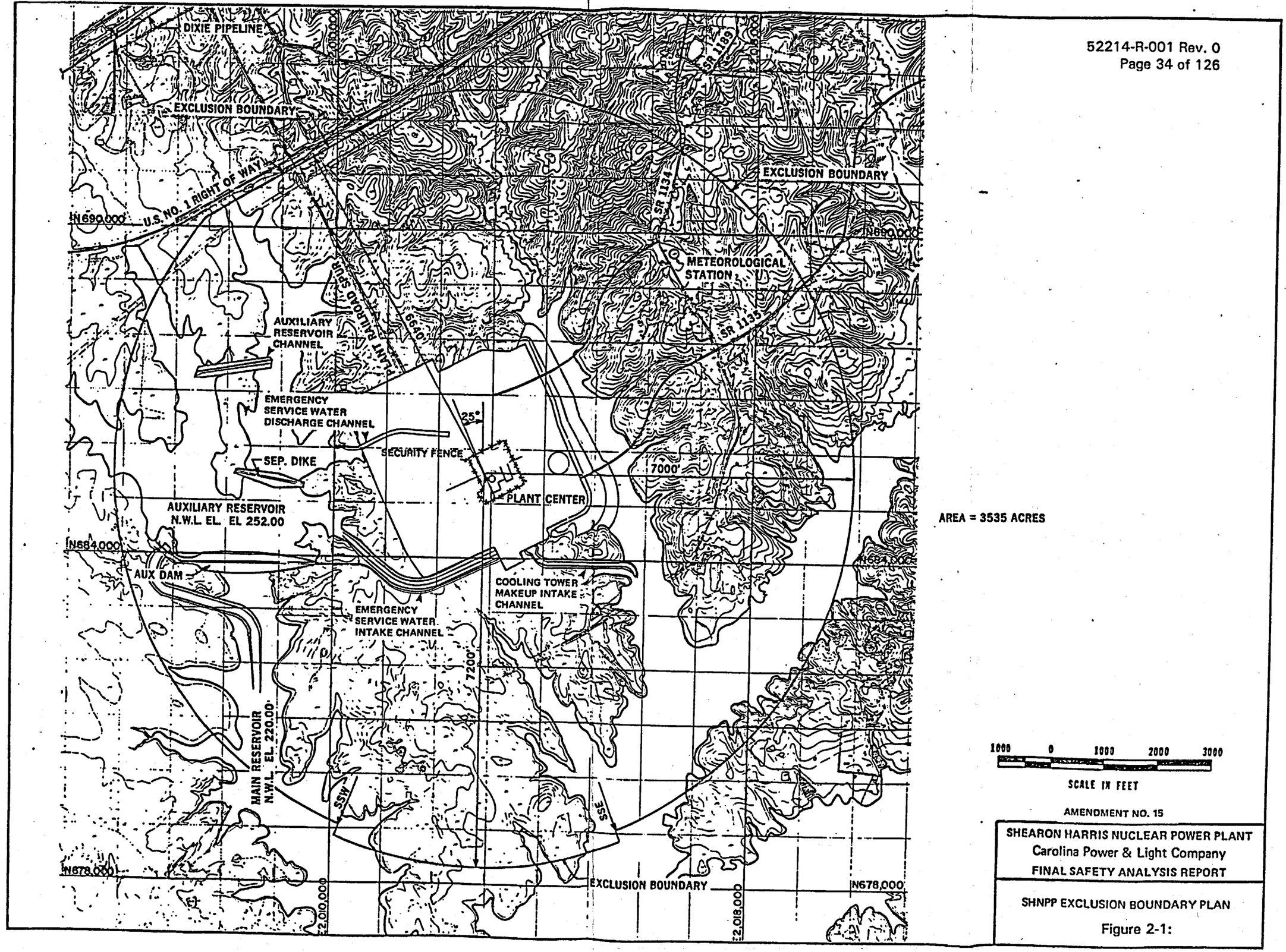
2.9.2 Seismic II/I Criteria

Various structures, systems and components for Harris have been designed in accordance with Regulatory Guide 1.29 to ensure that all safety related components remain functional following the postulated failure of non-seismically designed components. The provisions of this Regulatory Guide have been fully considered in the original design and in design changes. During the design process,

interdiscipline drawings were reviewed to identify interaction between safety related and non-safety related components. These include system piping drawings, HVAC duct layout drawings, conduit and cable tray drawings general arrangements and civil/structural design drawings. Close coordination between design disciplines was maintained to assure that interpretation of the design drawings was accurate and that all spatial considerations were incorporated in the design. During field design and modification work, any potential seismic interactions that were identified were reviewed and dispositioned by design engineering.

Nuclear Engineering Department (NED) Design Guide Number DG-II.19, Reference 31 was written to address seismic II/I criteria. This design guide describes the walkdown that took place at the end of construction. The intent of the walkdown was to confirm that existing design and quality controls required by Regulatory Guide 1.29, Sections C.2 and C.4 were implemented correctly during erection of both design and field-located components. These walkdowns were systematically performed in accordance with established zone areas. All cases identified during the walkdown were documented and resolved in accordance with the design guide.

The design guide is a current document and is used for continuing compliance to Regulatory Guide 1.29.

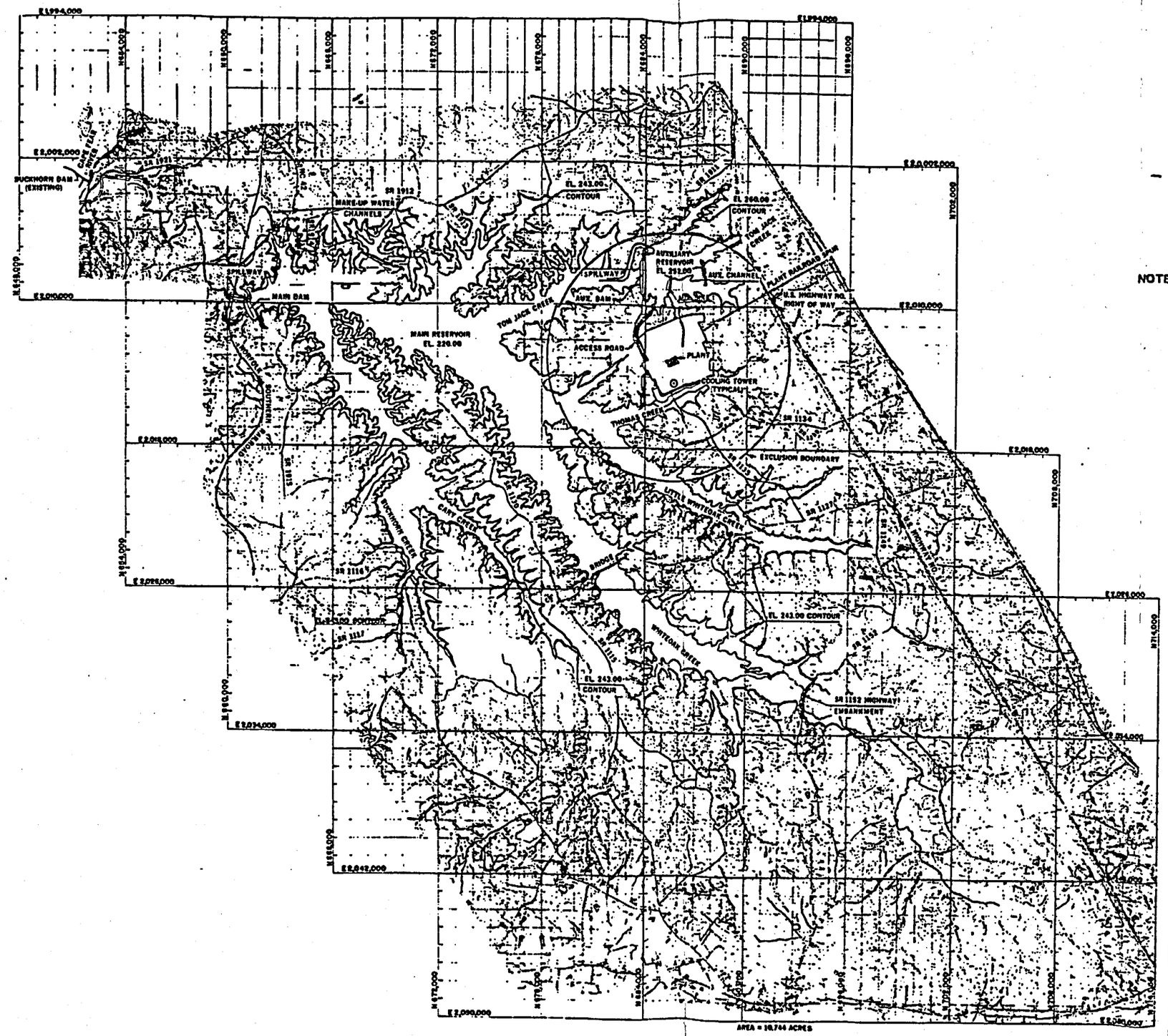


AMENDMENT NO. 15

SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

SHNPP EXCLUSION BOUNDARY PLAN

Figure 2-1:



NOTE: SITE BOUNDARY IS ENCLOSED BY THE EXCLUSION BOUNDARY, THE EL 243 CONTOUR OF THE MAIN RESERVOIR, AND THE EL 260 CONTOUR OF THE AUXILIARY RESERVOIR.

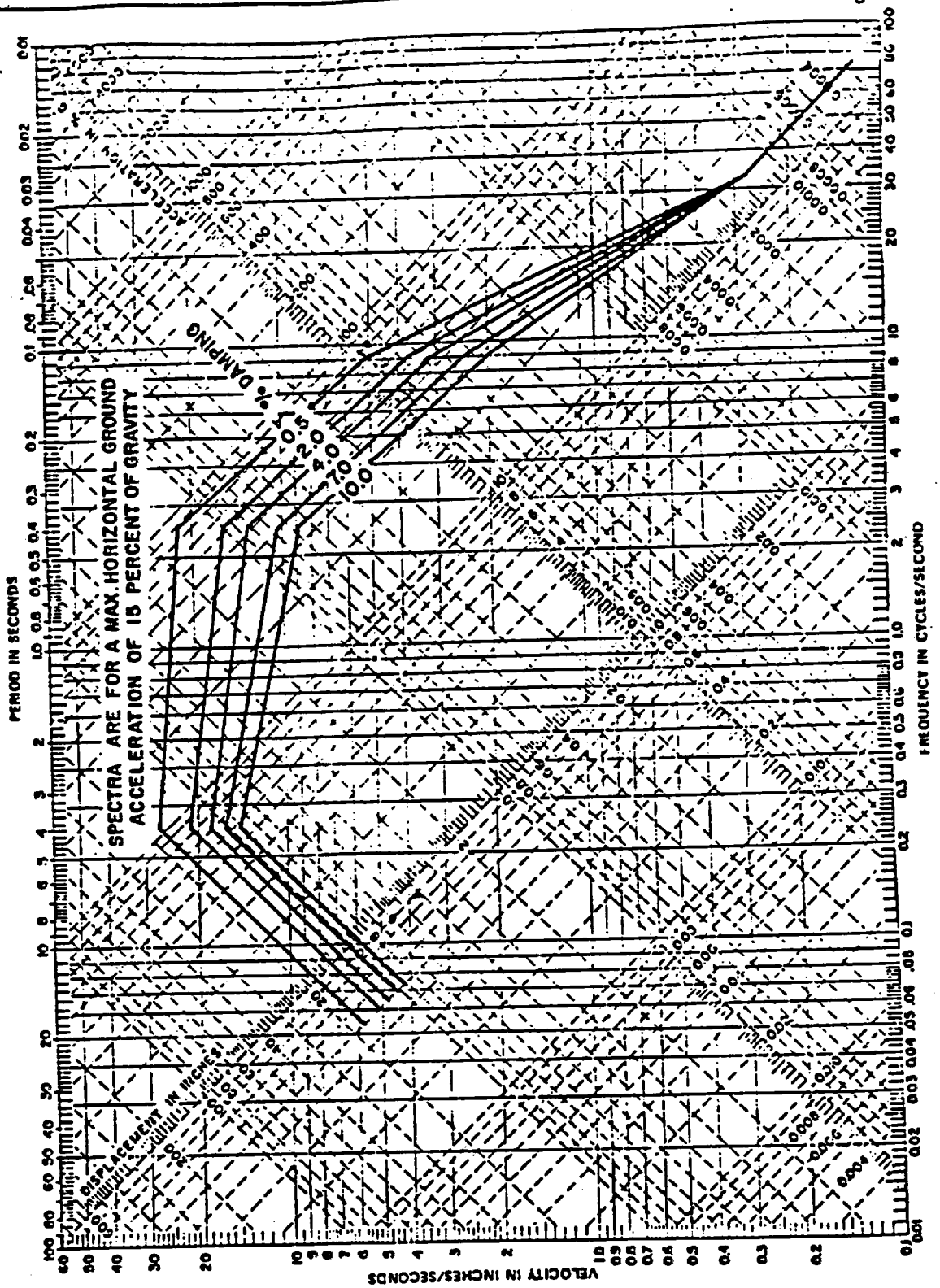
SCALE IN FEET

AMENDMENT NO. 21

SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

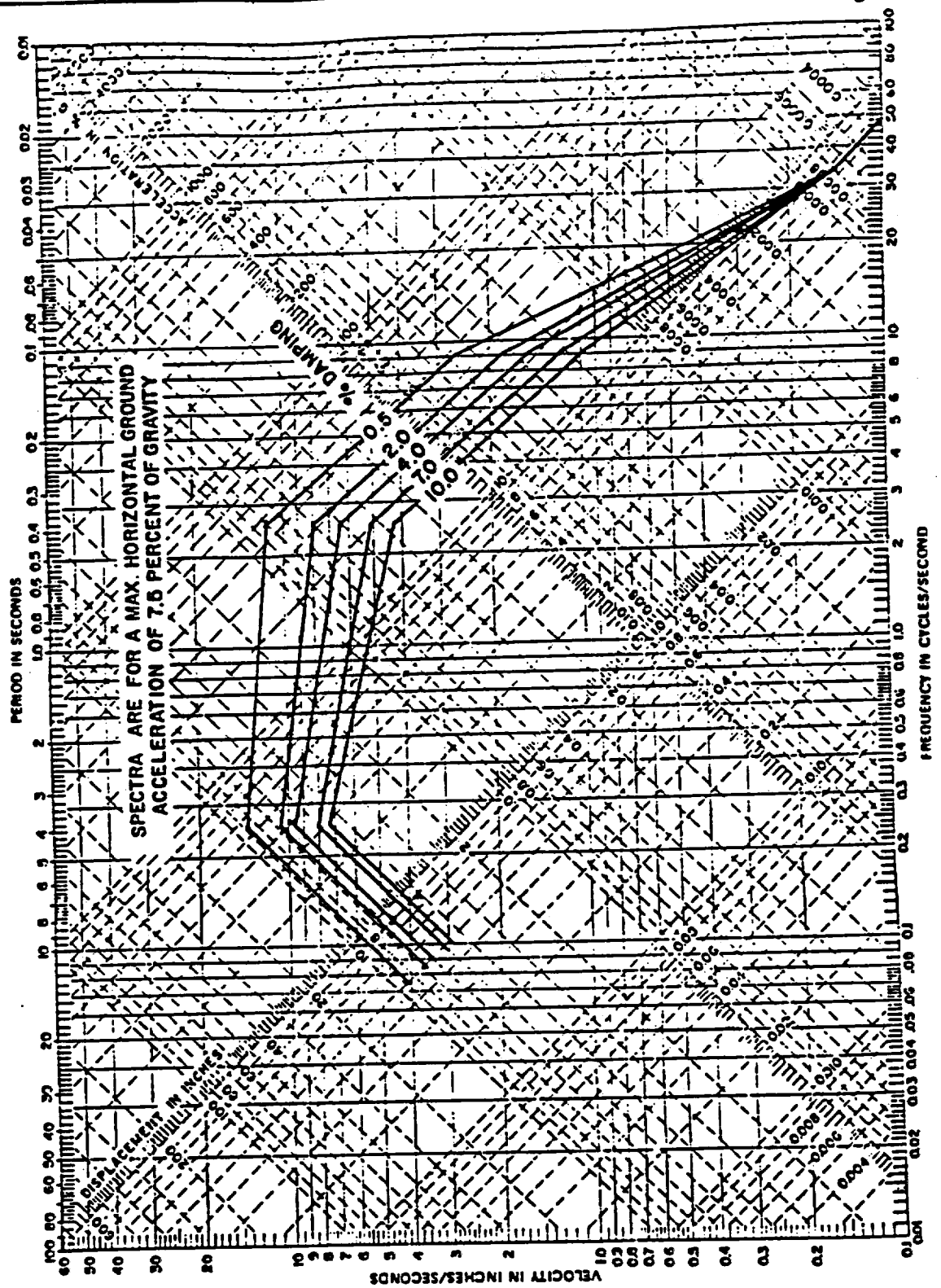
MAP OF SITE EXCLUSION BOUNDARY AND SITE BOUNDARY

Figure 2-2:



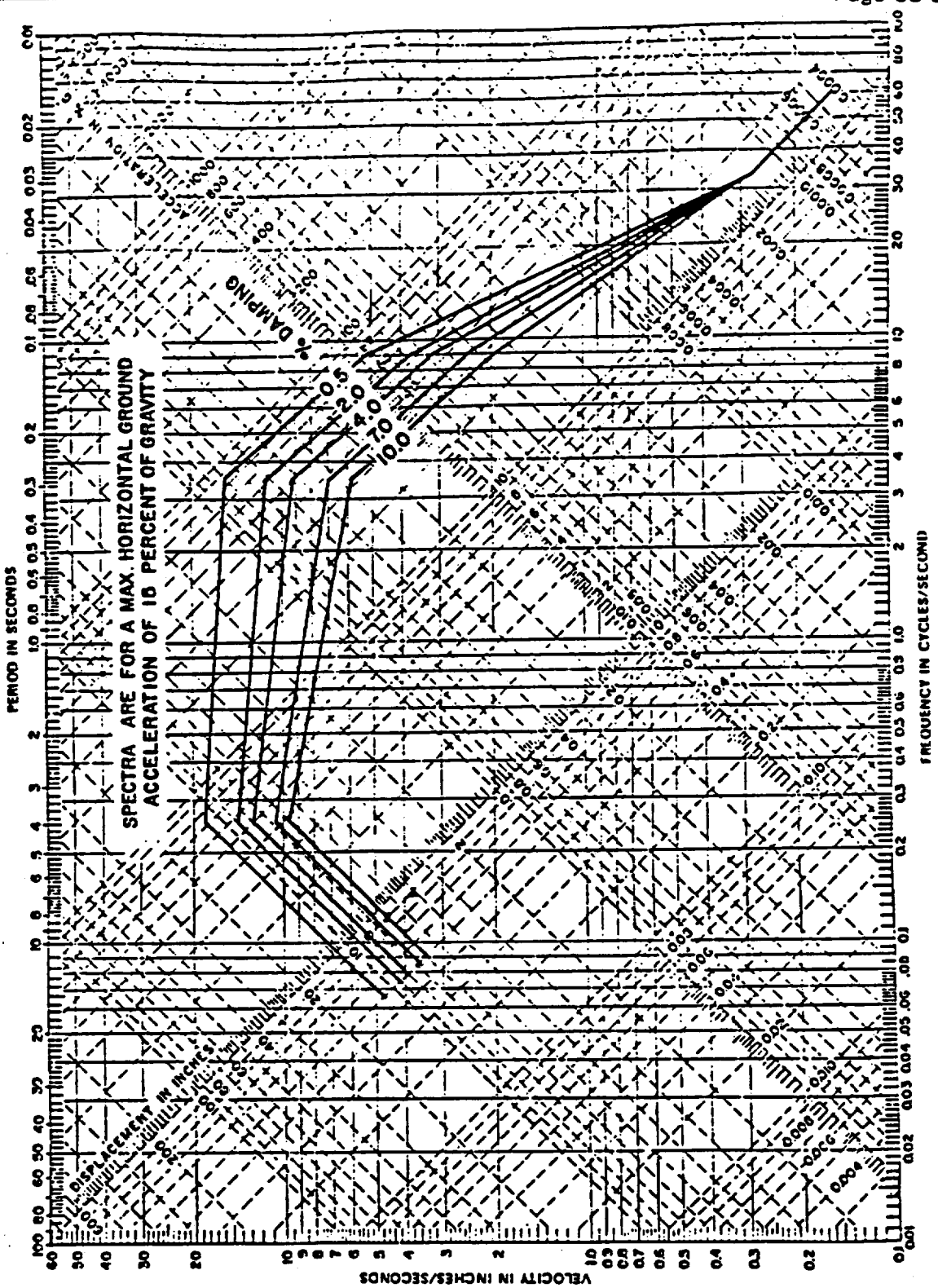
HORIZONTAL RESPONSE SPECTRA
DESIGN BASIS EARTHQUAKE

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	HORIZONTAL DESIGN RESPONSE SPECTRA SAFE SHUTDOWN EARTHQUAKE	FIGURE 2-3:



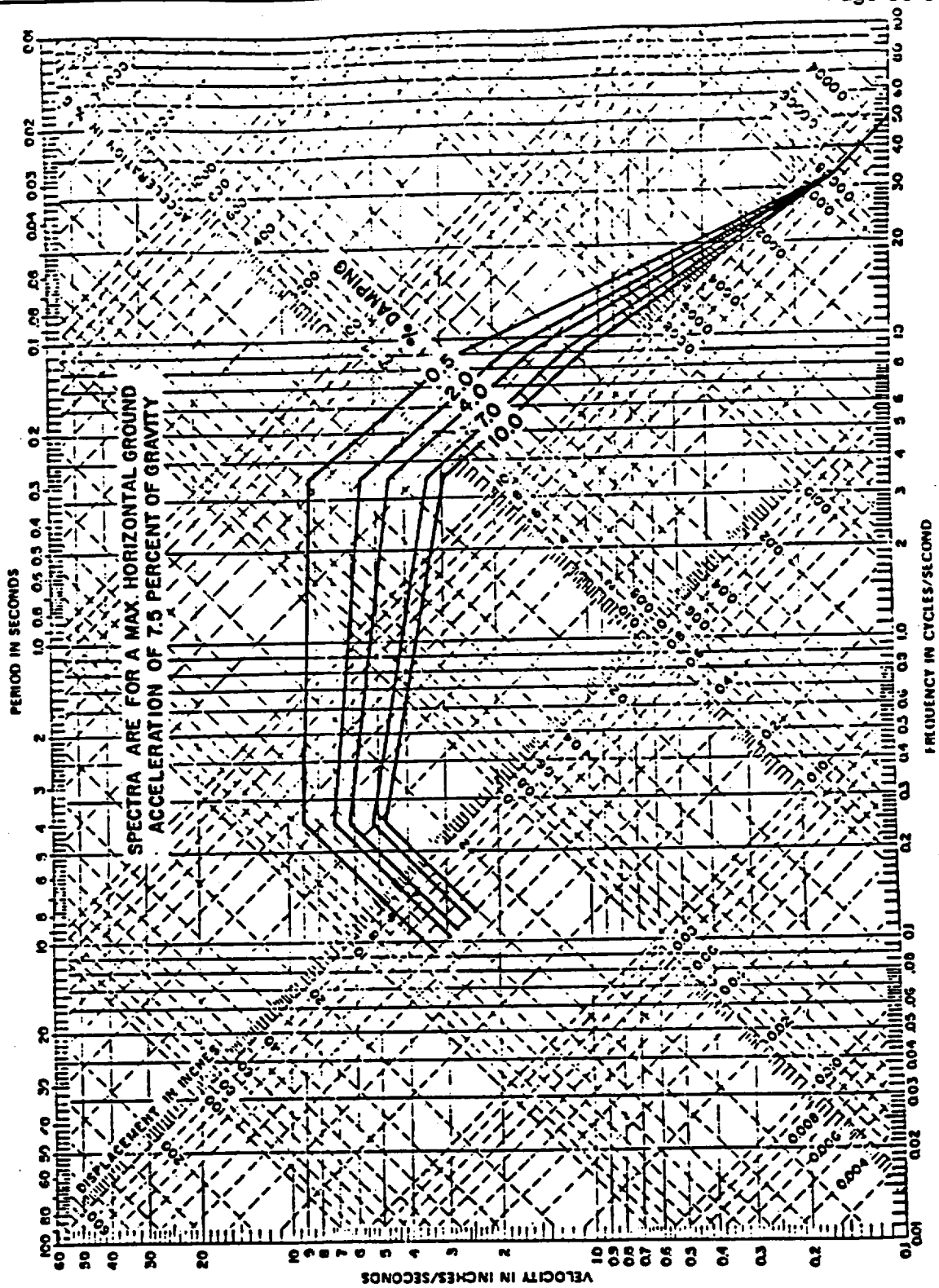
HORIZONTAL RESPONSE SPECTRA
OPERATING BASIS EARTHQUAKE

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	HORIZONTAL DESIGN RESPONSE SPECTRA OPERATING BASIS EARTHQUAKE	FIGURE 2-4:



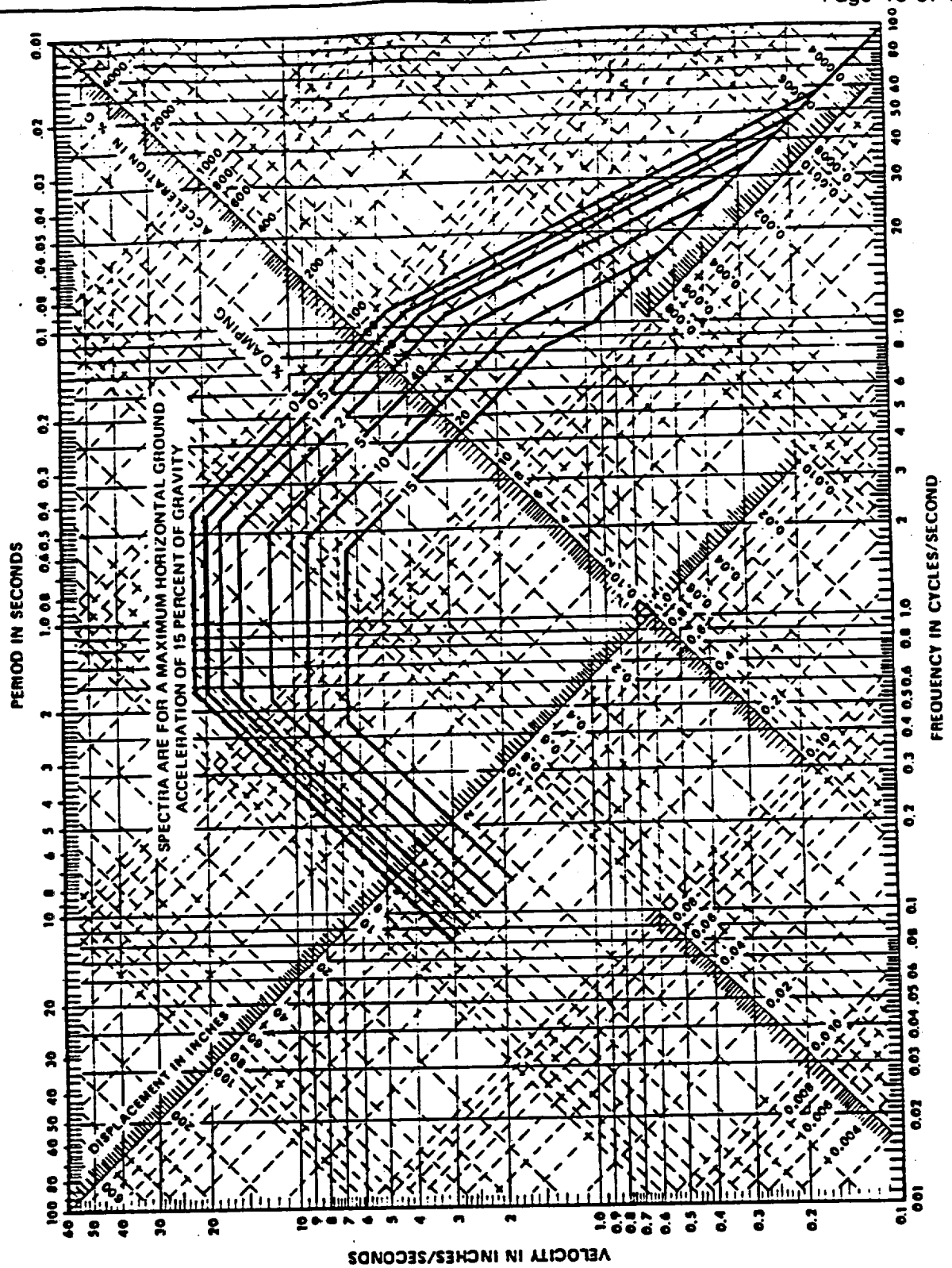
VERTICLE RESPONSE SPECTRA
DESIGN BASIS EARTHQUAKE

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	VERTICAL DESIGN RESPONSE SPECTRA SAFE SHUTDOWN EARTHQUAKE	FIGURE 2-5:



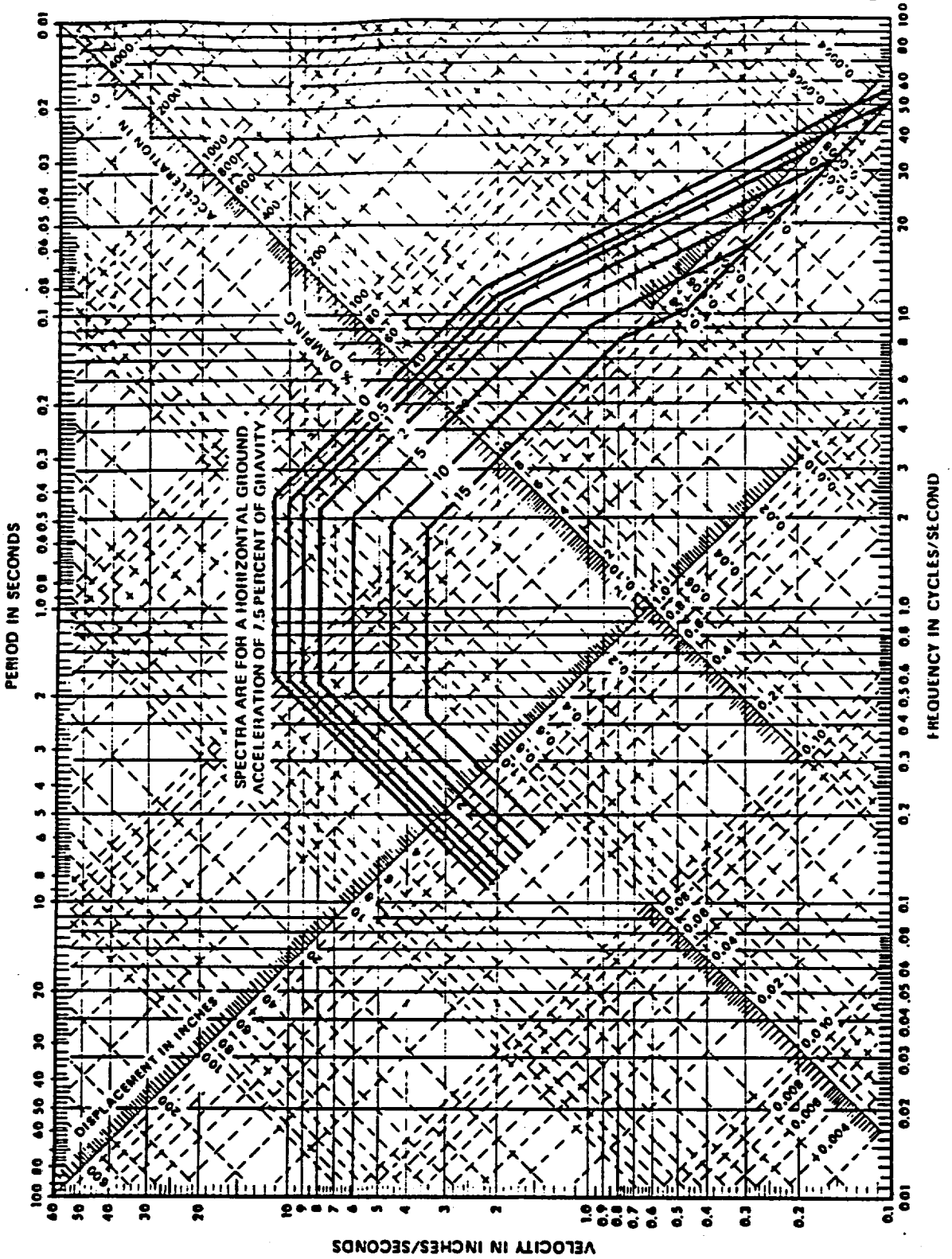
VERTICLE RESPONSE SPECTRA
OPERATING BASIS EARTHQUAKE

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	VERTICAL DESIGN RESPONSE SPECTRA OPERATING BASIS EARTHQUAKE	FIGURE 2-6:



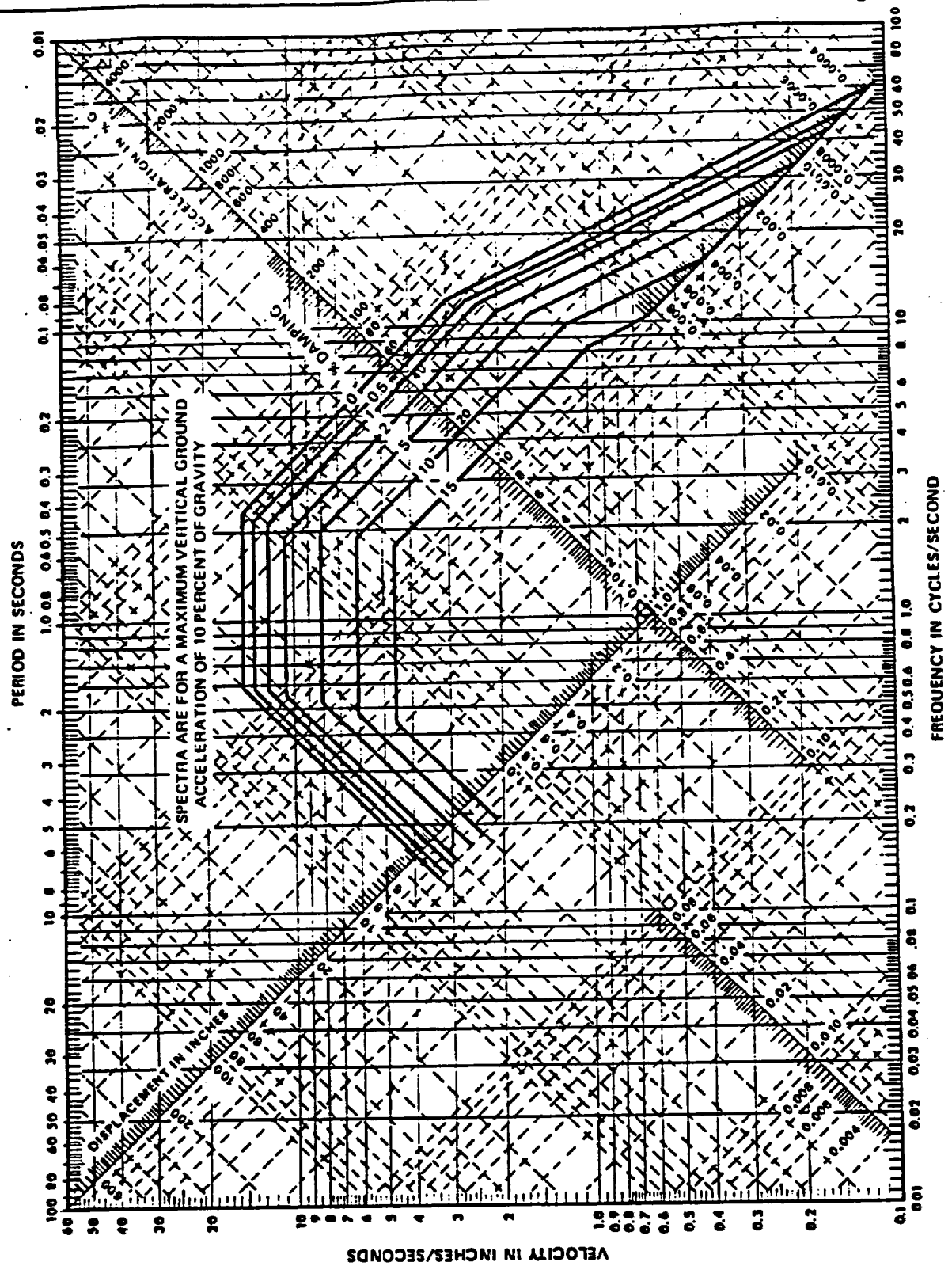
RESPONSE SPECTRA

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	HORIZONTAL DESIGN RESPONSE SPECTRA SAFE SHUTDOWN EARTHQUAKE DAMS AND DIKES	FIGURE 2-7:



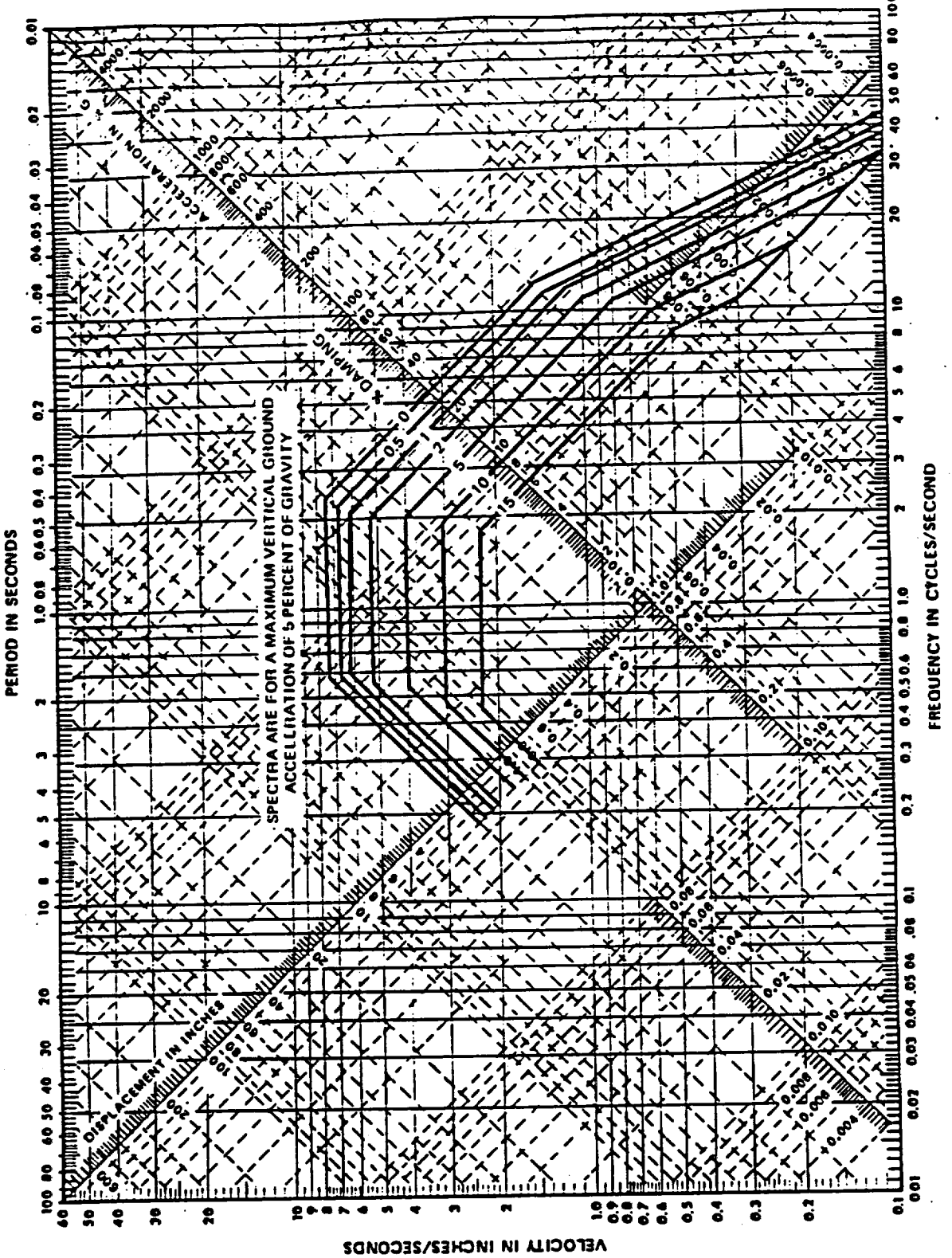
RESPONSE SPECTRA

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	HORIZONTAL DESIGN RESPONSE SPECTRA OPERATING BASIS EARTHQUAKE DAMS AND DIKES	FIGURE 2-8:



RESPONSE SPECTRA

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	VERTICAL DESIGN RESPONSE SPECTRA SAFE SHUTDOWN EARTHQUAKE DAMS AND DIKES	FIGURE 2-9:



RESPONSE SPECTRA

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	VERTICAL DESIGN RESPONSE SPECTRA OPERATING BASIS EARTHQUAKE DAMS AND DIKES	FIGURE 2-10:

3. SYSTEM DESCRIPTION AND SUCCESS PATH SELECTION

This task involves the identification of components and structures for in-plant review. A preliminary walkthrough was performed by Safety and Reliability Optimization Services, Inc. (SAROS), CP&L, and EQE personnel to search for potential low seismic capacity components. Reference 3 was utilized in choosing the items and identifying boundary conditions and assumptions.

The functions involved in the plant response to the RLE are reactivity control, reactor coolant system inventory control, reactor coolant system pressure control, and decay heat removal. In addition, the following were identified as support systems for ensuring critical plant functions:

- Safety and some Non-Safety Related AC Power
- Safety Related DC Power
- HVAC for Charging and Safety Injection Pumps, Diesel Generators, and Control Room
- Emergency Service Water
- Chilled Water System

The structures housing the components included in the above systems are:

- Reactor Auxiliary Building (including main steam tunnel)
- Containment Building
- Emergency Service Water Intake Structure
- Diesel Generator Building
- Tank Building

The resulting success path logic diagram (SPLD) evolved from studying available plant equipment function as well as the plant's normal and emergency operating

procedures. Two or more success paths are required for each of the four major system functions. The SPLD considers two conditions, one being a 1 inch loss of coolant accident (LOCA) and the other being a transient without Reactor Coolant System (RCS) leakage.

The SPLD was reviewed and agreed upon by Harris operations personnel. Equipment selected for inclusion on the (SSEL) was evaluated in a manner similar to that described in the SQUG Generic Implementation Procedure (GIP), Reference 7. Guidance from EPRI NP-6041 (Reference 3) was also used in preparing the format for the list of components. The SSEL (Reference 38) used for the IPEEE walkdown is presented in Appendix B.

4. SEISMIC MARGIN EARTHQUAKE DEMAND

The Harris plant is a relatively modern plant with all Seismic Category I structures, except for Seismic Category I Electrical Manholes, Seismic Category I Underground Electrical Conduits, and some Seismic Category I piping, founded on sound rock (shear velocity of 5600 fps). A shallow layer of soil varying from 15' to 25' feet in depth, with average shear wave velocity of roughly 610 fps exists over the rock base. Using the simple relationship,

$f_{soil} = V_s/4H$ where:

f_{soil} = fundamental soil column frequency

V_s = Shear wave velocity of soil column

H = Height of soil column

the fundamental frequency of the soil column is estimated to be in the 6 - 10 Hz range. The Design Basis Earthquake (DBE) for Harris is the Regulatory Guide 1.60 (Reference 8) spectra anchored to 0.15g. In the design analysis, this motion was introduced at the foundation level of the structure and is therefore directly transmitted into the building basemat. For other structures founded above the rock, a SHAKE analysis was performed to propagate the rock motion through the soil column. The design analysis also used soil springs to represent soil-structure interaction effects. However, these springs actually represent stiff rock and should not result in any inertial interaction effects. A uniform damping of 7% was assumed for the rock/structure system.

For the seismic IPEEE program, the Review Level Earthquake (RLE) is defined as the NUREG/CR-0098 median amplification response spectrum anchored to 0.30g. NUREG/1407 made the general statement that the ground motion should be considered at the surface in the freefield. It also states that if secondary conditions such as shallow soil conditions are considered, appropriate procedures should be used to determine the freefield motion in the vicinity of those affected structures and components. To follow the NUREG/1407 recommendation, a CR-0098 median

soil spectrum geared to 0.30g need to be defined on the soil surface, and a deconvolution study performed to determine the motion on the rock interface at foundation level. Such a deconvolution analysis would show the motion at depth on top of the rock to be deamplified in the 6 - 10 Hz range, when compared to the ground surface motion. Therefore, for structures such as the RAB-Common, RAB-Unit 1, and Containment with structural frequencies within this range, some relief may be obtained. The DGB with fundamental structural frequency of 12.4 Hz falls outside of the deamplified range, and hence is not expected to derive much benefit from deconvolution. The deconvolution effect is difficult to quantify without performing the actual analysis. Hence, it was decided to conservatively ignore deconvolution by specifying the NUREG/CR-0098 median rock spectrum at foundation level. This approach is in line with the original design basis in which the DBE was also applied at foundation level. Given that Reference 3 considers the CR-0098 median spectrum and the Regulatory Guide 1.60 ground response spectrum to be similar in shape, scaling of the design basis floor spectra to the RLE is readily accomplished. Scale factors were conservatively determined in Calculation 5224-C-001 (Reference 9) based on the methodology described in detail in Reference 3. The values indicate both the scaled peak and zero period acceleration (ZPA) in-structure acceleration values for the appropriate elevation in the Reactor Auxiliary Building, Containment Building and Diesel Generator Building. It should be noted that in the scaling process, the available design basis floor response spectra with either N-411 or 4% damping, were treated as 5% damped spectra, thereby introducing some added conservatisms. Scale factors range from 1.60 to 1.67 for the North-South direction, 1.56 to 1.67 for the East-West direction, and 1.11 to 1.33 for the vertical direction. An exception is the diesel generator pedestal, Node 5, which is rigid. The diesel generator scale factors are 2.00 for horizontal, and 1.33 for vertical.

Calculation 52214-C-001 (Reference 9) is provided in Appendix A, "In-Structure Response Spectra".

5. SEISMIC MARGIN ASSESSMENT SCREENING AND WALKDOWN

Section 2 summarizes activities that were performed in preparation of the Seismic Review Team (SRT) walkdowns. The activities include selection of the SRT, systems description and success path selection, development of seismic margin earthquake demand, walkdown preparation and pre-screening, and establishment of screening criteria.

5.1 SEISMIC REVIEW TEAM

The SRT was assembled following guidance provided in Reference 3, drawing on the experience and expertise of EQE International, Inc. (EQE) and Carolina Power & Light Company (CP&L) personnel.

Each walkdown team included a minimum of two SRT members who had completed the Seismic Qualification Utility Group (SQUG) Walkdown Screening and Seismic Evaluation training course, as well as EPRI's add-on training for IPEEE. Joint walkdown teams generally consisted of at least one EQE Engineer and at least one CP&L Engineer. The following persons participated in the SRT walkdowns:

- Jeffrey H. Bond (CP&L)
- Steven R. Bostian (CP&L)
- Martha C. Cook (CP&L)
- Daryl W. Hughes (CP&L)
- Ronald L. Knott (CP&L)
- Robert N. Panella (CP&L)
- Kevin N. Poythress (CP&L)
- Ronald W. Cushing (EQE)
- Gregory S. Hardy (EQE)
- Timothy Mason (EQE)
- Thomas R. Roche (EQE)

Among all the team members there is strong experience in each of the areas listed below:

- Knowledge of the failure modes and performance of structures, tanks, piping, process and control equipment, and active electrical and mechanical components during strong earthquakes.
- Knowledge of nuclear design standards, seismic design practices, and equipment qualification practices for nuclear power plants.
- Ability to perform fragility evaluations including structural/mechanical analysis of essential elements of nuclear power plants.
- Knowledge of the plant system functions and normal and emergency operating procedures.

The qualifications of each of the CP&L and EQE seismic walkdown team members are presented in Appendix A.

5.2 WALKDOWN PREPARATION AND PRE-SCREENING

The purpose of pre-screening was to ensure efficiency in the walkdowns and evaluations with a goal of completing the maximum amount of data entry in advance of the walkdown. This was accomplished by incorporating existing data onto the seismic IPEEE documentation forms prior to the walkdowns. Data that was reviewed consisted of the Final Safety Analysis Report (FSAR), design criteria, stress reports, equipment qualification reports (testing and analysis), structures and equipment support drawings, equipment location drawings, anchorage calculations. and records from other related programs previously performed at Harris. An initial

walkdown was performed by CP&L and EQE personnel as part of the pre-screening task to review the SSEL and to group items according to the "Rule of the Box."

Pre-screening was performed with three purposes in mind:

- To identify critical failure modes to be specifically reviewed on the walkdown.
- Assemble qualification and installation data for use as a basis for screening in the margins review.
- To provide data to be utilized in HCLPF calculations.

A considerable amount of information was extracted from the existing documentation and was subsequently recorded on the Screening and Evaluation Work Sheets (SEWS) prior to commencing the detailed walkdowns. Information entered into SEWS during prescreening was intended to provide available data to the SRT to assist in equipment screening. An example of data is comparison of MCC Test Response Spectrum (TRS) Zero Period Acceleration (ZPA) to the RLE ZPA based on equipment qualification summary data. The information is not intended as the sole basis for screening, but assists the SRT in their review.

Pre-screening was enhanced by the use of the software program EHOST. EHOST is a data base program which has been adapted specifically for use in performing Unresolved Safety Issue (USI) A-46 and IPEEE evaluations. The program is set up so that the data is incorporated onto SEWS forms which are consistent with those recommended in EPRI NP-6041. In this manner the walkdown teams, using portable computers with the companion program EWALK were then able to work more efficiently by having access to SEWS that had already been partially completed.

5.3 SCREENING CRITERIA

The Harris seismic IPEEE was completed following the EPRI seismic margins methodology recommended by NUREG-1407 (Reference 2) for a focused scope plant.

Civil structures, equipment and subsystems were screened following the methodology provided in EPRI NP-6041 (Reference 3) for a focused-scope plant. Screening criteria are provided in Tables 2-3 and 2-4 of Reference 3 for civil structures and equipment and subsystems, respectively. The criteria corresponding to 5 percent-damped peak spectral acceleration less than 0.8g were used for Harris based on the RLE. The guidelines are supplemented by Appendix A of the EPRI seismic margins methodology which provides the basis for the seismic capacity screening guidelines. Walkdown data sheets provided by EPRI NP-6041 (Reference 3) were used during the SRT walkdowns.

SEWS were loaded into EQE's computer program EWALK for field screening and data collection using portable pen-based computers. Prescreening information was downloaded from the database program EHOST. The effectiveness of in-plant reviews was improved by access to SEWS forms enhanced with plant specific data. This also allowed the walkdown teams to be alerted to specific concerns that may have been identified during pre-screening.

The SRT had liberal access to plant design drawings, analyses and test reports to use in conjunction with the screening criteria. Much of this information was reviewed and summarized in the SEWS prior to the field walkdowns. This provided the SRT with information such as:

- Seismic coefficients used in motor operated valve weak-link analyses to verify that the intent of valve mass and eccentricity guidelines were satisfied.

- Valve yoke material from vendor drawings to rule out cast iron material.
- Test report for battery chargers used to verify capacity of the base channels.
- Anchorage calculation for CCW heat exchanger indicated that the closely spaced cast-in-place anchors were considered as a single group (shear cone) and not as individual anchors.
- Test report for 6.9 KV switchgear used to verify margin between TRS and SSE level and then compared to RLE level.
- Chiller qualified by using finite element analysis, and separate qualification by test for control panel was reviewed during pre-screening.
- Anchorage data from design drawings were reviewed for anchor bolt type and embedment for various components.

5.4 SEISMIC MARGIN WALKDOWN RESULTS

The Harris seismic margin walkdown was completed during the winter and spring of 1994 in two phases, balance of plant (BOP) and outage walkdowns.

BOP walkdowns were completed in January 1994 prior to refueling outage RFO-5 in order to capture elements that could be reviewed during plant operation. Outage walkdowns were performed in March during RFO-5, capturing items located inside of containment as well as electrical and control equipment that needed to be de-energized for access or was critical to plant operation.

Completed SEWS are included in Reference 32 with copies of photographs included in Reference 33.

The procedure for performing walkdowns is described in the Harris Project Plan, Reference 10. Some equipment was found to be inaccessible during the walkdowns for various reasons, such as high radiation, sealed confined spaces, etc. The following equipment was inaccessible during the walkdown: regenerative heat exchanger 1X-SN; motor operated valves 1SI-300 and 1SI-301; and temperature elements 1TE-410, 1TE-420 and 1TE-430. In these cases, thorough drawing reviews were performed as well as reviews of equipment known to be similar in design and configuration. In addition, photographs of the regenerative heat exchanger were available for review from the radiation control group. These inaccessible equipment were screened out based on these reviews.

The walkdown concentrated on the strength and load path of the equipment to the structure as well as function and integrity. The review of equipment anchorage was a prime objective for the walkdown teams. The anchorage evaluation addressed both physical attributes of the anchorage installation and the capacity relative to other success path items as well as the postulated demand at the RLE. Anchorage capacities were addressed in the pre-screening, as most of the components already had been evaluated for seismic capacities. The anchorage calculation for the Transfer Panels was the only calculation that was not found during the reviews/walkdown. The Transfer Panels were included as a component for HCLPF evaluation based on the anchorage configuration and the fact that the calculation was not reviewed. The walkdown teams also verified that the anchorage was generally in accordance with the design configuration. The anchorage of the components was screened based on the high capacity anchorage and the SRT walkdown.

Interaction reviews were performed to identify falling, impact, spray and flood issues that could affect success path items. Housekeeping issues were also identified, e.g. compressed gas cylinder not sufficiently secured, fuel board not restrained, storage cabinets that were unrestrained, etc. Thirteen items were identified that had minor interaction, housekeeping or maintenance issues. No spray or flood issues were noted during the SRT walkdown.

Tools, test equipment and other maintenance items used during the outage were not noted as housekeeping issues during the outage walkdown. Housekeeping walkdowns were conducted subsequent to the outage to verify that non-plant equipment is in compliance with the Harris housekeeping procedures, Reference 27 and 28. These procedures were revised to enhance the storage and restraint provisions for non-plant equipment.

Suspended systems, such as conduit, cable trays and ductwork were evaluated on a sampling basis in the plant. A general survey was performed to obtain an overview of the suspended system construction throughout the plant. This included a review of the variety of system layouts, support configurations, and construction details. The inspection also included known concerns for suspended systems, such as taut cables, sharp edges, or overloading of cable trays and supports, and potential anchor point displacements. The ceiling above the control room was also reviewed to verify if the light fixtures and ceiling grid were adequately supported, and to evaluate the potential for ceiling panels to fall.

Containment penetrations were reviewed on an area basis to identify anomalies that might affect containment performance. Concerns such as falling and differential building displacement were considered. Also reviewed were displacement concerns between the containment shell and internal structure. Containment isolation valves were also reviewed on a walk-by basis based on the caveats listed on the valve SEWS.

Seismic margin walkdown results are summarized for structures and equipment and subsystems in Tables 5-1 and 5-2, respectively.

Table 5-1 lists civil structures following the format of EPRI NP-6041, Table 2-3, along with screening results for the Harris plant. All Harris civil structures were screened from further review based on EPRI NP-6041, Table 2-3, screening criteria and Section 3.8 of the Harris FSAR.

Table 5-2 lists equipment and subsystems following the format of EPRI NP-6041, Table 2-4, along with screening results for the Harris plant. Equipment and subsystems that were not screened were grouped into 5 categories:

- Thirteen items had minor interaction, housekeeping or maintenance issues that will be resolved through routine maintenance activities via work requests (WR/JO). These items are listed in Table 5-3 along with the corresponding work request number. Work was completed by December 31, 1994 with the exception with WR/JO 95ACLTL which will be completed by the end of RFO-7.

- Six items were identified as requiring repairs or modifications. These items are listed in Table 5-4 along with the description of the repair or modification required for the item. This work is scheduled to be completed by the end of RFO-7.

- One potential interaction issue was identified due to relative building displacement. This issue is addressed in Section 9.

- Two components were identified due to interaction issues with HVAC ducts. These items are addressed in Section 5.8.2.

- Several other items were identified as candidates for HCLPF evaluation.

At the conclusion of plant walkdowns SRT members, including senior level participants from CP&L and EQE, convened to complete the ranking and screening task. SRT members reviewed SEWS and categorized components into the following resolution categories:

- Screened out by the SRT
- Housekeeping or maintenance issue

- Repairs or modification required
- Specific issues require clarification
- Candidate for HCLPF evaluation

Next, SRT members re-walked items not screened out and re-visited existing design and qualification documentation. SRT members that performed the initial walkdowns presented and discussed issues with remaining SRT members. Categorization was refined by group consensus.

The first pass through SSEL items categorized about 45 items representing about 15 specific issues for HCLPF evaluation. Following these group walkdowns and follow-up discussion, 16 items were selected for HCLPF evaluation to address 5 issues. Several of the unscreened components were unique issues such as no top supports on RHR Heat Exchangers while others, such as low voltage switchgear located relatively high in the structure, were selected to represent a group of equipment. These 5 issues are considered to represent the most vulnerable issues observed by the SRT. Other items may have comparable seismic capacity but are considered representative of the selected items.

- Sixteen components were selected for HCLPF evaluation (See Table 5-5). The 16 components were further grouped into five issues:

  Group 1: Six Panels Having Anchorage with 45° Nelson Studs

  Group 2: Four Low Voltage Switchgear

  Group 3: Two Residual Heat Removal (RHR) Heat Exchangers

  Group 4: Two Flat Bottom Storage Tanks

  Group 5: Two Emergency Service Water Pumps

  HCLPF evaluations are summarized in Section 6.

The following sections summarize the review of each major equipment class.

5.4.1 Motor Control Centers

The SSEL contains sixteen motor control centers (MCC's) that are located in the Diesel Generator Building (DGB), Reactor Auxiliary Building (RAB) and the Emergency Service Water Intake Structure (ESWIS) at various elevations. The MCC's are typical Gould Inc. 5600 series, 480 volts, consisting of 20" wide by 20" deep by 90" high sections. Each section weighs approximately 500 lbs. The MCC has a solid load path down to the shipping channels. The anchorage consists of the MCC being bolted to shipping channels and the shipping channels are welded to embedded channels. The MCC's are screened by SRT based on the walkdown, anchorage calculations and the seismic testing performed on the MCC.

Three MCC's were noted to have missing latches on the breaker panel doors. Work requests have been issued to replace the latches, see Table 5-3. Refer to the individual MCC SEWS for specific screening and walkdown results.

5.4.2 Low Voltage Switchgear

The SSEL contains four low voltage switchgear. The switchgear are located in the RAB at elevation 286'. The switchgear are Gould Inc. indoor single ended substation, 480V, type LK, with a 2000kVA transformer. The switchgear were evaluated to be of good seismic design. The switchgear are constructed of structural steel framing and sheet metal panels with an integral structural steel base which provides adequate stiffness and a good load path. The anchorage of the switchgear uses a variation of structural shapes, embedded plates and expansion anchors. No interaction issues or maintenance issues were noted during the walkdown regarding the switchgear. The transformer was noted as not having any top bracing. The low voltage switchgear were selected for HCLPF evaluation based on the anchorage design margin and the transformer anomalies during the seismic testing (without top bracing), see section 6. Refer to the individual SEWS for specific screening and walkdown results.

5.4.3 Medium Voltage Switchgear

The SSEL contains two medium voltage switchgear. The switchgear are located in the RAB (Unit 1) at elevation 286'. The switchgear are Seimens-Allis Inc. 6.9kVA, model FB 500 consisting of thirteen cubicles. Each cubicle is 36" wide by 94" deep by 90" high and weighs approximately 3200 lbs. The switchgear were evaluated to be of good seismic design. The switchgear are mounted on three sill channels. The sill channels are welded to a plate. The plate is anchored via 5/8 inch diameter cast-in-place J-bolt anchors. The anchors are spaced on 36" centers. Each switchgear cubicle has six plug welds, two to each sill channel. No interaction issues or maintenance issues were noted during the walkdown regarding the medium voltage switchgear. The medium voltage switchgear are screened by SRT based on the walkdown, anchorage calculations and the seismic testing performed on the switchgear. Refer to the individual SEWS for specific screening and walkdown results.

5.4.4 Transformers

The SSEL contains eight transformers. Four transformers are mounted in MCC's and the other four transformers are mounted in the low voltage switchgear. These MCC's and switchgear are located in the RAB at elevation 286'. The transformers have been treated as rule-of-box in which the transformers are evaluated as part of the MCC or switchgear.

5.4.5 Horizontal Pumps

The SSEL contains eighteen horizontal pumps. The pumps are located in the RAB (Unit 1) at elevation 236' with the exception of two fuel oil transfer pumps, two chill water pumps and three relief valve oil pumps (rule-of-box). The fuel oil transfer pumps are located in the Fuel Oil Building at elevation 246' and the chill water pumps are located in the RAB (Unit 1) at elevation 261'. The pumps have been evaluated to be of good seismic design. No interaction issues or maintenance

issues were noted during the walkdown regarding the pumps. The horizontal pumps are screened based on the walkdown, anchorage calculations and the review of the seismic qualification documentation. Refer to the individual SEWS for specific screening and walkdown results.

5.4.6 Vertical Pumps

The SSEL contains four vertical pumps. Two of the pumps are RHR motor driven pumps located in the RAB (Unit 1) at elevation 190' and the other two pumps are the ESW suction pumps located in the ESWIS at elevation 261'. The RHR pumps have been evaluated to be of good seismic design. No interaction issues or maintenance issues were noted during the walkdown regarding the RHR pumps. The RHR pumps are screened based on the walkdown, anchorage calculations and the review of the seismic qualification documentation. Refer to the SEWS for specific screening and walkdown results.

The ESW suction pumps were selected for HCLPF evaluation, see section 6. No interaction issues or maintenance issues were noted during the walkdown regarding the ESW suction pumps. Refer to the SEWS for specific screening and walkdown results.

5.4.7 Pneumatic Operated Valves

The SSEL contains twenty-three pneumatic operated valves including one safety relief valve. These valves are located in the RAB and Containment Building (CB) at various elevations. These valves have been evaluated to be of good seismic design. No cast iron components were noted for these valves during pre-screening and the walkdown. The operator height and weight were evaluated/judged to meet the intent of the caveat. No interaction or maintenance issues were noted during the walkdown. The pneumatic operated valves are screened based on the walkdown and the review of the seismic qualification documentation. Refer to the individual SEWS for specific screening and walkdown results.

5.4.8 Motor Operated Valves

The SSEL contains fifty-seven motor operated valves. These valves are located in the RAB and CB and Main Steam Tunnel at various elevations. These valves have been evaluated to be of good seismic design. No cast iron components were noted for these valves during pre-screening and the walkdown. The operator height and weight were evaluated/judged to meet the intent of the caveat. No interaction or maintenance issues were noted during the walkdown except for valve 1AF-55, see Table 5.3. The motor operated valves are screened based on the walkdown and the review of the seismic qualification documentation. Refer to the individual SEWS for specific screening and walkdown results.

5.4.9 Fans

The SSEL contains four fans. The fans are associated with the emergency diesel exhaust system and are located in the DGB. The fans are Joy Manufacturing Company, model S-1000, 54-26½-1170. The weight of the fans is approximately 2161 lbs. The fans are mounted horizontally in-line to structural steel supports with six 1/2" diameter, SAE grade 2 bolts. There are no vibration isolators. No interaction or maintenance issues were noted during the walkdown. The fans are screened by SRT based on the walkdown and the seismic analysis of the fans. Refer to the individual SEWS for specific screening and walkdown results.

5.4.10 Air Handling Units

The SSEL contains four air handling units (AHU's). The AHU's are located in the RAB (Unit 1) with two units at elevation 236' and two units at 305'. The AHU's are associated with cooling the charging/safety injection pump rooms, electrical cabinet rooms and the control room. The AHU's are manufactured by Bahnson. The units at elevation 236' weigh 6,000 lbs and the units at elevation 305' weigh 8200 lbs. The base of the AHU's is constructed of C8 x 13.75 structural steel. The bases are secured to the structure with a minimum of sixteen 5/8 inch

diameter cast-in place anchors. There are no vibration isolators. No interaction or maintenance issues were noted during the walkdown. The AHU's are screened by SRT based on the walkdown. Refer to the individual SEWS for specific screening and walkdown results.

5.4.11 Chillers

The SSEL contains two chillers. The chillers are located in the RAB (Unit 1) at elevation 261'. The chillers provided chilled water to the AHU's. The chillers are manufactured by York (division of Borg Warner Corp.), model OTS4G1-IMBS, 752 tons per unit. The weight of the entire assembly is approximately 60,000 lbs. The fundamental frequency of the unit is greater than 33 Hertz based on the seismic analysis. Each chiller is anchored with (28) 1-3/4 inch diameter A307 cast-in place anchors of length 2'-5". There are no vibration isolators. No interaction or maintenance issues were noted during the walkdown. The chillers are screened by SRT based on the walkdown, anchorage calculations and the seismic analysis. Refer to the individual SEWS for specific screening and walkdown results.

5.4.12 Air Compressors

The SSEL contains four air compressors. The compressors are located in the DGB at elevation 261'. The compressors are associated with the air start system of the emergency diesel generator. The compressors are manufactured by Ingersoll-Rand, model H251BKMX2, Type 40. The weight of the entire assembly is approximately at 2860 lbs. The base of the assembly is constructed of structural steel channels. The base is anchored with six 3/4 inch diameter by 33" long cast-in place anchors . There are no vibration isolators. No interaction or maintenance issues were noted during the walkdown. The air compressors are screened by SRT based on the walkdown. Refer to the individual SEWS for specific screening and walkdown results.

5.4.13 Motor Generators

The SSEL does not contain any motor generators.

5.4.14 Distribution Panels

The SSEL contains fourteen distribution panels. Six of the distribution panels are an integral part of MCC's. The MCC's are located in the RAB at elevation 286'. These distribution panels have been included as rule-of-box components and are evaluated with the associated MCC.

Four of the distribution panels are located in the RAB at elevation 305'. The equipment tag numbers are IDP-1A-SI, IDP-1A-SIII, IDP-1B-SII and IDP-1A-SIV. These distribution panels are manufactured by Gould Inc., and are constructed in a similar manner to the Gould 5600 series MCC. The anchorage is the same as an MCC being bolted to shipping channels and the shipping channels are welded to embedded channels. These distribution panels are screened by SRT based on the walkdown and the seismic testing performed on MCC's with the exception of the maintenance issues listed in Table 5.3. Refer to the individual SEWS for specific screening and walkdown results.

The remaining four distribution panels are located in the DGB at elevation 261' and the RAB at elevation 286'. The distribution panels are manufactured by Gould Inc. The weight of the distribution panels are estimated at 800 lbs (DGB distribution panels) and 1550 lbs (RAB distribution panels). The base of the DGB distribution panels are welded to channels that are welded to embedded plates. No interaction or maintenance issues were noted during the walkdown. The DGB distribution panels are screened by SRT based on the walkdown and seismic testing of the distribution panels. The two distribution panels in the RAB, 1A-SA and 1B-SB were selected for HCLPF evaluation due to the embedded channel with nelson studs at 45 degree angle, see section 6. These two distribution panels were also included in Table 5.4 for repair of the access doors with stripped screws. Refer to the individual SEWS for specific screening and walkdown results.

5.4.15 Battery Racks

The SSEL contains two battery racks. The battery racks are located in the RAB (Unit 1) at elevation 286'. The two step, four bay battery racks were supplied by C&D Batteries. The weight of one rack and 30 cells is approximately 8,550 lbs. The racks are constructed of a welded structural steel frame with cross bracing. Each rack is anchored with (40) 1/2" diameter by 2'-6" long, A193 Grade B8 cast-in-place J-bolt. No interaction or maintenance issues were noted during the walkdown. The battery racks are screened by SRT based on the walkdown, anchorage calculations and the seismic testing performed on the racks. Refer to the individual SEWS for specific screening and walkdown results.

5.4.16 Battery Chargers and Inverters

The SSEL contains two battery chargers and four inverters. The battery chargers are located in the RAB (Unit 1) at elevation 286'. The battery chargers were manufactured by C&D Batteries, model ARR130K150F. The weight of one battery charger is approximately 800 lbs. The battery charger base is constructed of structural steel channels and is anchored with (4) 3/8" diameter by 2'-6" long, A193 Grade B8 cast-in-place J-bolt. No interaction or maintenance issues were noted during the walkdown. The battery chargers are screened by SRT based on the walkdown, anchorage calculations and the seismic testing performed on the battery chargers. Refer to the individual SEWS for specific screening and walkdown results.

The inverters are located in the RAB (Unit 1) at elevation 305'. The inverters were manufactured by Westinghouse. The weight of one inverter is approximately 1150 lbs. The inverters are welded with 1/4" fillet welds by 4" long spaced at 9" centers. The internal frame is bolted to the bottom skid with 5/8 inch diameter bolts in each corner. No interaction or maintenance issues were noted during the walkdown. The inverters are screened by SRT based on the walkdown. Refer to the individual SEWS for specific screening and walkdown results.

5.4.17 Engine Generators

The SSEL contains two engine generators, the emergency diesel generators (EDG). The EDG are located in the DGB at elevation 261'. The EDG were supplied by DeLaval Turbine Inc. model DSRV, 16-4. The engine weighs 235,000 lbs and the generator weighs 63,500 lbs. The engine is anchored with (22) 2" diameter by 65" long (45" embedment) cast-in-place anchors. The generator is anchored with (12) 1-1/4" diameter by 40" long (33-5/8" embedment) cast-in-place anchors. No interaction or maintenance issues were noted during the walkdown. The EDG are screened by SRT based on the walkdown, anchorage calculations and the seismic qualification documentation. Refer to the individual SEWS for specific screening and walkdown results.

5.4.18 Automatic Transfer Switches

This category covers flow switches, flow transmitters, level switches, level transmitters, pressure switches and pressure transmitters. The SSEL contains sixty-two switches and transmitters. These instruments are located through out the plant and are mounted directly to the concrete structures, instrument racks or cabinets. These instruments were generally of rugged construction. No interaction or maintenance issues were noted during the walkdown. The switches and transmitters are screened by SRT based on the walkdown and the seismic qualification documentation. Refer to the individual SEWS for specific screening and walkdown results.

5.4.19 Instrument Racks

The SSEL contains twenty instrument racks. The racks are located in the CB at elevation 236', the Tank Building at elevation 236' and the RAB at elevations 236' and 261'. The instrument racks are generally rigid and construction of structural steel tubing, channel and angle. The racks are either secured to the structure by welds or expansion anchors. No interaction or maintenance issues were noted

during the walkdown except for a fluorescent light interaction with rack A21-R15, see Table 5.3. The instrument racks are screened by SRT based on the walkdown and the seismic qualification documentation. Refer to the individual SEWS for specific screening and walkdown results.

5.4.20 Temperature Sensors

The SSEL contains thirteen temperature indicators, elements, controllers or sensors. These instruments are located through out the plant and are mounted directly to the concrete structures, instrument racks or piping. These instruments were generally of rugged construction. No interaction or maintenance issues were noted during the walkdown except for a misaligned U-bolt attaching 1TIS-658A to a pipe, see Table 5.3. The temperature components are screened by SRT based on the walkdown and the seismic qualification documentation. Refer to the individual SEWS for specific screening and walkdown results.

5.4.21 Control and Instrumentation Cabinets

The SSEL contains thirty control and instrumentation cabinets. The cabinets are located in the DGB at elevation 261' and the RAB at elevations 236', 286' and 305'. The cabinets have all been evaluated to be of good seismic design. In general, the cabinets are constructed of structural steel framing and sheet metal panels with an integral structural steel base which provides adequate stiffness and a good load path. The cabinets are secured to the structure by expansion anchors, cast-in-place anchors, embedded plates or embedded channels. The anchorage of the cabinets were all screened with the exception of four cabinets in the RAB at elevation 286'. The four cabinets are the transfer panels 1A and 1B and the sequencer panels 1A-SA and 1B-SB. These cabinets were selected for HCLPF evaluation due to the embedded channel with nelson studs at 45 degree angle, see section 6. Also, nine cabinets were noted as having interaction or maintenance issues, see Tables 5.3 and 5.4. Refer to the individual SEWS for specific screening and walkdown results.

5.4.22 Buried Tanks

The SSEL contains two buried tanks, main fuel oil storage tanks. The tanks are located in the Fuel Oil Building at elevation 242'. The tanks are associated with the fuel oil system of the emergency diesel generator. The fuel oil storage tanks consists of a concrete structure that has an internal steel liner. The tank is partially buried in the ground. The tank has a concrete cover that is exposed. the cover has a large concrete cap that can be removed by a crane which is used to service the tank. The piping routed to the transfer pumps is well supported. No interaction or maintenance issues were noted during the walkdown. The fuel oil storage tanks are screened by SRT based on the walkdown. Refer to the individual SEWS for specific screening and walkdown results.

5.4.23 Vertical Tanks and Heat Exchangers

The SSEL contains twenty vertical tanks and two vertical heat exchangers. Twelve tanks are associated with the air start system of the emergency diesel generator, four air dryers, four air separators and four air tanks (compressor receiver). These tanks are located in the DGB at elevation 261' and were evaluated to be of good seismic design. No interaction or maintenance issues were noted during the walkdown. The tanks associated with the air start system are screened by SRT based on the walkdown. Refer to the individual SEWS for specific screening and walkdown results.

The eight remaining vertical tanks are the diesel generator fuel oil day tanks (two), boron injection tank, boric acid tank, volume control tank, boric acid filter, condensate storage tank (CST) and refueling water storage tank (RWST). The two largest tanks, CST and RWST were selected for HCLPF evaluation since these tanks are important and high profile elements of the shutdown paths, see section 6. The CST anchorage consists of (80) 3" diameter cast-in-place bolts and the RWST consists of (76) 3" diameter cast-in-place bolts. The HCLPF capacity is estimated to be over 1g based on methodology provided in Reference 3, see Section 6. No

interaction or maintenance issues were noted during the walkdown. These vertical tanks are screened by SRT based on the walkdown, anchorage calculations, HCLPF evaluation, and seismic analysis of the tanks. Refer to the individual SEWS for specific screening and walkdown results.

The two vertical heat exchangers are for the RHR system. The RHR heat exchangers are located in the RAB at elevation 236'. The heat exchangers are manufactured by Joseph Oat & Sons, Inc., and are 40" diameter and 26'-9½" high. The heat exchangers weigh 38,000 lbs. The baseplate is 1½" thick steel plate and is stiffened by a 3/4" thick steel skirt of the heat exchanger. The baseplate is attached to the structure with eight 1½" diameter by 42½" long cast-in-place anchors. Upper lateral restraints were included in the original design but were not installed based on analyses that justifies this configuration for DBE levels. The heat exchangers were selected for HCLPF evaluation based on the upper lateral supports not being installed, see section 6. Refer to the individual SEWS for specific screening and walkdown results.

5.4.24 Horizontal Tanks and Heat Exchangers

The SSEL contains fifteen horizontal tanks and heat exchangers. The horizontal tanks and heat exchangers are located throughout the plant. These tanks and heat exchangers were evaluated to be of good seismic design and to have adequate anchorage. No interaction or maintenance issues were noted during the walkdown. The horizontal tanks and heat exchangers are screened by SRT based on the walkdown, anchorage calculations, and seismic qualification documentation. Refer to the individual SEWS for specific screening and walkdown results.

5.4.25 Solenoid Operated Valves

The SSEL contains sixteen solenoid operated valves. Six of the solenoid valves are associated with the power operated relief valves (PORV's). The PORV's are located in the CB at elevation 315'. These solenoid valves have been included as rule-of-

box components and are screened with the PORV's. Eight of the solenoid valves are associated with the air start system of the emergency diesel generator. The emergency diesel generator air start system is located in the DGB at elevation 261'. These solenoid valves have been included as rule-of-box components and are screened with the emergency diesel generator air start system..

The remaining two solenoid operated valves are the fill valves to the diesel fuel oil day tanks. These valves are located in the DGB at an approximate elevation of 263'. The valves are Target Rock, model 79Q-007 and are of good seismic design. No interaction issues or maintenance issues were noted during the walkdown regarding these solenoid valves. These two solenoid operated valves are screened by SRT based on the walkdown and the seismic qualification documentation. Refer to the SEWS for specific screening and walkdown results.

5.5 STRUCTURES

All Harris Seismic Category I buildings are founded upon reinforced concrete mats resting on suitable rock or on concrete fill founded on suitable rock. The structural dynamic analysis of the Seismic Category I buildings is based on use of the response spectra developed for 0.075g for the operating basis earthquake and 0.15g for the safe shutdown earthquake.

Harris Category I structures are all screened from further review based on Reference 3, Table 2.3 and Section 3.8 of the FSAR. A brief description of each of the buildings within seismic IPEEE success paths is provided in the following subsections.

5.5.1 Concrete Containment

The concrete containment is a steel lined reinforced concrete structure in the form of a vertical right cylinder with a hemispherical dome. The 4 ft. 6 in. thick cylindrical wall measures 160 ft. in height from the liner on the base to the springline of the dome and has an inside diameter of 130 ft. The inside radius of

the 2 ft. 6 in. thick dome is equal to that of the cylinder so that the discontinuity at the spring line due to the change in thickness is on the outer surface. The base mat consists of a 12 ft. thick structural concrete slab and a metal liner. The liner is welded to inserts embedded in the concrete slab. The base liner is covered with concrete, the top of which forms the floor of the containment. The base mat is supported by sound rock.

The basic structural elements considered in the design of the containment structure are the basemat, cylinder wall, and dome. These act essentially as one structure under all loading conditions. The liner plate is 3/8 in. thick in the cylinder, ¼ in. thick on the bottom, and ½ in. thick in the dome. The liner is anchored to the concrete shell by means of anchor studs fusion welded to the liner plate so that it forms an integral part of the containment structure. The liner functions primarily as a leaktight membrane.

The containment design loads and loading combinations included earthquake loads using the safe shutdown earthquake horizontal ground acceleration is 0.15g. To account for the simultaneous action of the three spatial components of the earthquake, the representative maximum value of a particular response is obtained by taking the square root of the sum of the squares of the corresponding maximum values of the response to each of the three spatial components calculated independently.

The analysis of the containment shell is based on the classical theory of thin elastic shells of revolution in accordance with Section CC-3300 of the ASME Code, Section III, Division 2. The shell is assumed to be ideally elastic, homogeneous, and isotropic. Reinforcement and the steel liner are neglected in calculating the member stiffness. The design of the Containment demonstrates that, for factored load conditions, the following requirements are met:

a) The summation of external and internal forces and moments satisfies the laws of equilibrium and does not bring any structural section to a general yielding state.

b) Tensile yielding in the reinforcement is acceptable when thermal gradient temperature effects are combined with other applicable loads, provided that the temperature induced forces and moments are reduced as yielding in the reinforcement occurs, and the increased concrete cracking does not cause deterioration of the Containment.

The liner plate is not used as a strength element. Interaction of the liner with the Containment is considered in determining liner behavior.

The containment wall is independent of adjacent interior and exterior structures; sufficient space is provided between the containment wall and adjacent structures to prevent contact under any combination of loading. The interior grating platforms and concrete slabs are supported on steel beams which span between the secondary shield wall and the containment wall. These beams are independently supported, near the containment wall, by steel columns resting on the concrete mat.

5.5.2 Internal Structures

The reinforced Concrete Containment Structure encloses the concrete structures and structural steel components which comprise the Containment Internal Structures. The Containment Internal Structures provide support for the NSSS equipment during all operational phases and, in the unlikely event of an accident, act to mitigate the consequences of the accident by protecting safety-related equipment and other engineered safety features from the effects induced by the accident.

The concrete internal structures, which consist of the primary and secondary shield walls and other concrete enclosures, form compartments within which the entire RCS is located. The main components are the concrete primary shield wall, which encloses the reactor cavity, the semicircular concrete secondary shield walls, which forms the steam generator compartments, the reinforced concrete walls and floors, the fuel storage area, refueling pool and reactor internals laydown areas, the concrete enclosure wall around the pressurizer, the containment steel floors, stairs, and platforms, reactor vessel supports, steam generator supports, and reactor coolant pump supports. The concrete and steel internal structures are supported on a concrete foundation mat 5 ft. thick, resting on the 12-ft. thick concrete containment structure foundation mat. The internal foundation mat is placed on top of the bottom liner plate; no anchorages of the internal structures and internal mat penetrate through the containment liner plate into the external mat. The walls of the internal structures are anchored into the internal foundation mat.

The structural acceptance criteria for the Containment Internal Concrete Structures and the internal and other Seismic Category I structural steel structures consists of compliance with the following requirements:

a) Concrete Structures - To assure that the structural integrity of Category I concrete structures is maintained for the service and factored load conditions, the limits of the stress and strain intensity of concrete generally follow the strength design method requirements of American Concrete Institute (ACI) 318-71.

Using the factored loads, the various components have the required load capacity if the stresses in them do not exceed the yield strengths of the materials used. To provide for the possibility that small, adverse variations in dimensions and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in a net under capacity of the component, the

load capacities of the individual structural members are reduced by a reduction factor "ϕ" for the design cases.

The factors were established for the design on the basis of the function of the component and the effect on its net capacity of the variations enumerated above. These factors are generally in accordance with the ACI 318-71 Code and are tabulated in Table 3.8.3-1 of the FSAR.

a) Steel Internal Structures - Structural steel framing is designed for the loading combinations, given in Section 3.8.3.3.3 of the FSAR, to exhibit either elastic or plastic behavior in all load carrying elements. To assure that the structural integrity of Seismic Category I steel structures is maintained, limits on the resulting stresses and the reduced strength capacities are observed.

5.5.3 Reactor Auxiliary Building

The Reactor Auxiliary Building (RAB) houses engineered safeguards and supporting systems, switchgear, sampling rooms, and the Control Room. The Seismic Category I building consists of two independent structures. One section is designated as RAB-1. The adjacent section is designated as RAB-Common.

The RAB-1 is a reinforced concrete structure, 207 ft. long by 187 ft. wide, varying in height from 69 ft. to 134 ft. from the top of foundation mat to the top of roof. The top of the foundation mat varies from 24 ft. to 70 ft. below finished grade, Elevation 260 ft.

The RAB-Common is a reinforced concrete structure, 120 ft. long by 187 ft. wide by 88 ft. high from the top of mat to the top of roof. The top of the foundation mat is at Elevation 236 ft. except for the pipe tunnel area, which is at Elevation 216 ft.

These buildings are cast-in-place reinforced concrete structures. The floors are supported on beams and girders which are in turn supported on interior columns and/or exterior walls. Where interior shear walls are installed, the beams and girders are supported on the shear walls. All interior shielded walls and partitions, other than shear walls, are either reinforced concrete or concrete block, and are not load bearing. Provisions are made for installation of these walls after the framing, floor systems, and equipment have been installed.

The buildings are supported on separate foundation mats 10 ft. thick which are founded on suitable rock.

5.5.4 Diesel Generator Building

Diesel-Generator Building (DGB) houses the stand-by diesel generators, day tanks, silencers, and associated equipment. The building is a Seismic Category I, missile-proofed, reinforced concrete structure, approximately 153 ft. long and 114 ft. wide.

The building is constructed on concrete fill, which is founded on suitable rock. The top of the foundation mat is 3.5 ft. below the grade floor Elevation 261.0 ft.; there is a space (gap) between the foundation mat and the grade floor (within the building walls) that is filled with sand for the electrical main leads and pipe lines. The foundation mat is 6 ft. thick.

The building is cast-in-place concrete with reinforced concrete exterior and interior shear walls, and reinforced concrete floors. The floors are supported on wall beams. Interior walls, other than shear walls, are reinforced concrete walls or concrete masonry (block) walls and are not load bearing walls. All reinforced concrete floor slabs are designed to act as horizontal diaphragms to transfer horizontal forces to shear walls and to carry vertical loads simultaneously.

5.5.5 Tank Building

The Tank Building houses the refueling water storage tank, reactor make-up water storage tank, condensate storage tank, and other associated equipment. The Tank Building also houses the waste monitoring tanks, secondary waste sampling tank, their associated pumps, and other facilities.

The Tank Building is a reinforced concrete structure, approximately 142 ft. long by 63 ft. wide, and 83 ft. high. The top of the foundation mat is at Elevation 236 ft. and the top of the roof, which provides for missile protection, is at approximately Elevation 319 ft. The foundation mat is 8 ft. thick and is founded on suitable rock.

The top of the foundation mat is 24 ft. below the finished grade elevation of 260 ft. The Tank Building has cast-in-place reinforced concrete exterior walls, interior shear walls, and reinforced concrete floors, supported on shear walls, beams, and columns. All interior shielding or partition walls, other than shear walls, are either reinforced concrete or concrete block walls and are not load bearing walls.

The overall tank building analysis was run using a time history seismic input forcing function. Response spectra were developed from that analysis for Elevation 261 ft., the elevation of the bottom of the tank. These response spectra were broadened by plus or minus 15 percent as required by Regulatory Guide 1.122. These broadened response spectra resulted in higher accelerations for essentially the entire range of frequencies of the spectra curves, thus increasing the seismic input which in turn increased the accelerations and moments at the common points in detailed tank analysis as compared to overall tank building analysis.

The seismic input for the detailed tank analysis for the fundamental frequency of the tank of 15.67 cycles/sec is 0.61g is compared to 0.39g of the actual unbroadened spectra curve indicating a margin of approximately 56 percent.

5.5.6 Diesel Fuel Oil Storage Tank Building

The Diesel Fuel Oil Storage Tank Building consists of a below grade reinforced concrete structure which provides for two reinforced concrete diesel oil tanks and two pumps. The structure is 94 ft. long, 86 ft. wide, and 24 ft. high (including the foundation mat); the top slab is at Elevation 263 ft. The building is supported on a reinforced concrete foundation mat which is founded on sound rock. The top of the foundation mat is at Elevation 242.25 ft; the mat is 3 ft. 3 in. thick.

The tanks have a capacity of 175,000 gallons each. Each compartment is 66 ft. long, 21 ft. wide, and 18 ft. 6 in. high; the free board is at least 12 in. The inside surfaces of the concrete compartments are lined with carbon steel to prevent leakage.

The pumps are housed in below grade cubicles separated by reinforced concrete walls.

5.5.7 Structures for the Emergency Service Water System

The Seismic Category I structures of the ESWS consist of the Emergency Service Water Intake Channel, Emergency Service Water Screening Structure, Emergency Service Water and Cooling Tower Makeup Intake Structure, Emergency Service Water Discharge Structure, and the Emergency Service Water Discharge Channel. Retaining walls are provided at the Emergency Service Water Screening Structure. All Concrete structures are reinforced and founded on sound rock.

Cooling water is drawn from either the Auxiliary Reservoir or the Main Reservoir. Water drawn from the Auxiliary Reservoir is carried by a series of steel pipes from the Emergency Service Water (ESW) Screening Structure to the Emergency Service Water and Cooling Tower Makeup Intake Structure. Cooling water is discharged into the Auxiliary Reservoir through the Emergency Service Water Discharge Channel.

a) Emergency Service Water Intake Channel - The Emergency Service Water Intake Channel extends from the Auxiliary Reservoir to the Emergency Service Water Screening Structure. The bottom of the channel is at Elevation 238 ft., except at the intake screening structure where it slopes down to Elevation 231 ft. Channel side slopes in rock are approximately four vertical to one horizontal; a 15 ft. wide beam is cut at the interface of soil and rock. Side slopes in soil are one vertical to two horizontal. The channel bottom is 50 ft. wide at all sections except at the intake structure. The lowest water level in the channel is Elevation 246.5 ft. and the maximum velocity in the channel is less than one ft. per sec. at this level.

b) Emergency Service Water Screening Structure - The ESW Screening is located at the eastern end of the ESW Intake Channel. It contains eight bays separated by reinforced concrete walls. Only two bays are used for the ESW system. Each ESW bay, 8 ft. 2 in. wide, is sized for seven ft. wide traveling screens. A reinforced concrete enclosure covers the deck to protect the traveling screens and valve pit from tornado missiles. A reinforced concrete skimmer wall, at the front of the intake structure, extends to Elevation 247.5 ft. and prevents ice and floating trash from entering the intake structure.

c) Emergency Service Water and Cooling Tower Makeup Intake Structure - The ESW and Cooling Tower Makeup Intake Structure is located at the northern end of the Cooling Tower Makeup Water Intake Channel. The intake structure has fourteen bays. Two bays are used for cooling tower make up pumps and two bays are used for ESW system. The screen

bays, containing ESW pumps, have a concrete dividing wall with an eight by ten ft. butterfly valve. The dividing wall-butterfly valve arrangement permits operation of the ESW pumps from either the Main or Auxiliary Reservoir. A reinforced concrete enclosure covers the deck to protect all ESWS equipment from tornado missiles.

d) Emergency Service Water Discharge Channel - The ESW Discharge Channel extends from the ESW Discharge Structure to the Auxiliary Reservoir. The bottom of the channel is at Elevation 240 ft. and it is 50 ft. wide. Channel side slopes in rock are approximately four vertical to one horizontal and in soil they are one vertical to two horizontal. A berm 15 ft. wide is cut at the interface of the soil and rock.

e) Emergency Service Water Discharge Structure - The ESW Discharge Structure, is located at the eastern end of the ESW Discharge Channel. It is a reinforced concrete structure which serves as the termination point for the service water discharge piping. The discharge structure has eight bays whereas only two bays are used.

f) Retaining Wall - Reinforced concrete retaining walls, where required, are located at the end of the Cooling Tower Makeup Water Intake Channel and at the ESW Screening structure. The walls are utilized to contain the earth adjacent to the concrete structures.

Harris Category I structures are all screened from further review based on Reference 3, Table 2.3 and Section 3.8 of the FSAR. No further action is required.

5.6 SOILS EVALUATION

Soil failure is deemed not a significant issue based on a review of the FSAR. Brief discussion on soil failure modes, embankments and dams is provided below.

In the spirit of the IPEEE program, CP&L does not believe that a detailed assessment of soils, embankments and dams is warranted. Key objectives of the IPEEE program are to gain useful insights in identifying plant specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. CP&L does not believe that either of these objectives are furthered by a detailed margins assessment of soils, embankments and dams given that no design concerns exist.

5.6.1 Soil Failure Modes

The plant is founded on gently dipping, well-consolidated Triassic siltstones and sandstones. All Seismic Category I structures within the plant area, except for Seismic Category I Electrical Manholes, Seismic Category I Underground Electrical Conduits, and some Seismic Category I pipes, are founded on sound rock. The foundation of the plant has no potential for liquefaction because it consists of hard sound rock.

The Seismic Category I underground piping systems, underground electrical conduits and electrical manholes, founded on soil, compacted fill, of weathered rock, are the only Seismic Category I structures that are not founded on sound rock. The fill in the yard area supporting Seismic Category I piping and conduits is not subject to liquefaction during a seismic event (FSAR 3.7.3.12).

Programs of subsurface exploration based primarily on trenching and borehole sampling and drilling were conducted for both the preliminary site investigation and the site fault investigation. In addition, the floors and walls of excavations for plant foundations were mapped geologically. No areas of actual or potential surface or subsurface subsidence, uplift, or collapse were found.

5.6.2 Embankments and Dams

Two dams were constructed in the Buckhorn Creek watershed to impound cooling water for the Harris Nuclear Plant. The Main Dam impounds a reservoir used primarily for cooling tower makeup water which has a normal water level elevation of 220 ft. and a water surface area of approximately 4,000 acres. The Main Reservoir also serves as a backup source of emergency service water. The Auxiliary Dam impounds a reservoir for emergency service water which has a minimum pond elevation of 250 ft. and a surface area of 317 acres. An Auxiliary Separating Dike and Auxiliary Reservoir Channel control the flow of discharged emergency service water through the east and west arms of the Auxiliary Reservoir.

The dike, constructed across the east arm of the reservoir, prevents discharged emergency service water from flowing directly back to the emergency service water intake area. The Auxiliary Reservoir Channel connects the east and west arms of the Auxiliary Reservoir, allowing emergency service water discharge to enter the west arm of the reservoir for maximum cooling before circulating back to the intake area.

The Main Dam, Auxiliary Dam, Auxiliary Separating Dike, Auxiliary Reservoir Channel, and Emergency Service Water Intake and Discharge Channels are designed and constructed to Seismic Category I criteria and to withstand the effects of natural phenomena. The slope of the dams, dike, and channels are designed to a factor of safety of 1.5 under static conditions, 1.2 for simultaneous OBE and 100-year return period flood level, and 1.1 for simultaneous SSE and 25-year return period flood level.

The Main Dam is a rockfill dam with a maximum height of 108 ft. and a length of approximately 1550 ft. at the beam elevation of 260 ft. Its outside slopes are 2.0 horizontal to one vertical. The main dam's spillway, with a crest elevation of 220

ft., is uncontrolled. The spillway crest has a net length of 50 ft. with a pier at its mid-length.

The Main Dam has a core of compacted silty clay and clayey silt material protected on each side by two 8-ft. thick transitional filter zones and a rockfill shell. The core is founded on suitable rock and the rockfill shell is founded on weathered rock.

The Auxiliary Dam is an earth dam approximately 4060 ft. long with a maximum height of approximately 73 ft. and a beam elevation of 260 ft. Its outside slopes are 2.5 horizontal to one vertical. The basis for its hydraulic design is the probable maximum flood (PMF).

The Auxiliary Dam area is in the Deep River Basin and is underlain by sedimentary rocks which, like those underlying the plant site, belong to the lower part of the Triassic Sanford Formation. The bedrock consists of four major lithologic units: medium-to coarse-grained sandstone, fine-to medium-grained sandstone, siltstone, and shaky siltstone.

The Auxiliary Dam has a compacted core of silty clay and clayey silt material protected by a transition filter zone and a random rockfill shell on each []side. The downstream shell is provided with two horizontal drainage blankets, each three ft. thick, which are connected with the transition filter zone adjacent to the core of the dam. In addition, a 200 ft. wide drainage layer is provided under the shell in each of two areas where pre-existing creeks had been located. The foundation of the dam was excavated into weathered rock and the cutoff trench was excavated to suitable rock.

The Auxiliary Reservoir Separating Dike is located across the east arm of the Auxiliary Reservoir about 1700 ft. north of the Auxiliary Dam. It is approximately 1200 ft. long with a maximum height of approximately 55 ft. Its outside slopes are 2.5 horizontal to one vertical.

The bedrock underlying the Auxiliary Reservoir Separating Dike consists of clastic sedimentary rocks of the Triassic Sanford Formation and is similar in most respects to the strata underlying the Auxiliary Dam.

The Auxiliary Reservoir Separating Dike has a core of compacted silty clay and clayey silt material protected by a random-rockfill shell which is size graded near the core, with the finer material placed adjacent to the core. The core and rockfill shell are founded either on weathered rock or on a thin layer of stiff residual soil overlying weathered rock.. Slope protection is provided by riprap placed on random rockfill.

The Emergency Service Water Intake and Discharge Channels are conservatively designed to carry the service flow required for normal and emergency shutdown of Harris. The intake channel is approximately 3580 ft. long and 50 ft. wide at its invert elevation of 238 ft. The discharge channel is approximately 2170 ft. long. The walls of both channels have a slope of two horizontal to one vertical in soil and one horizontal to four vertical in rock. Portions of the slopes were shaped to grade by backfilling with an impervious material. Diabase dikes were capped with concrete where they crossed channels.

The Auxiliary Reservoir Channel is approximately 1570 ft. long and 140 ft. wide at its invert elevation of 235 ft. Its walls have a slope of two horizontal to one vertical in soil and one horizontal to four vertical in rock. The Auxiliary Reservoir Channel is sized to carry the maximum ultimate discharge of the Service Water System coincident with a PMF flow for the upstream drainage basin.

All three channels are underlain by gently dipping strata of the Triassic Sanford Formation. Bedrock is lithologically and structurally similar to that at the Auxiliary Dam. Foundation conditions were explored by means of borehole drilling, sampling, and testing.

Embankments and dams are designed to a factor of safety of 1.1 for simultaneous SSE and 25 year flood. Further evaluation is not expected to identify any

significant vulnerabilities considering the revised hazard curves which support grouping Harris in the reduced-scope category (References 5 and 6). No further action is required.

5.7 NSSS REVIEW

A review of the Harris NSSS system was performed based on available information in accordance with Reference 3. The system is a modern 3 loop pressurized water reactor supplied by Westinghouse. In the spirit of the IPEEE program, CP&L does not believe that a detailed assessment of soils, embankments and dams is warranted. Key objectives of the IPEEE program are to gain useful insights in identifying plant specific vulnerabilities to severe accident that could be fixed with low-cost improvements. CP&L does not believe that either of these objectives are furthered by a detailed margins assessment of NSSS system and components given that no design concerns exist. The NSSS system and components are screened from further review.

5.7.1 NSSS Primary Coolant Systems and Supports

The primary coolant system and supports are screened from further review based on Appendix A of Reference 3.

5.7.2 Reactor Internals

The fuel assembly component stresses induced by horizontal seismic disturbances were analyzed through the use of finite element computer modeling. The fuel assembly safety analysis was performed for combined seismic and loss of coolant accidents (LOCA). The recent vintage reactor internals, such as Harris are screened from further review since they were designed for combined SSE and LOCA.

5.7.3 Control Rod Drive Mechanisms

The Control Rod Drive Mechanisms (CRDMs) were seismically analyzed to confirm that system stresses under combined loading conditions do not exceed allowable levels as defined by the ASME Code, Section III. The CRDM was modeled as a system of lumped and distributed masses and analyzed under appropriate seismic excitation and the resultant seismic bending moments along the length of the CRDM were calculated. The corresponding stresses were combined with the stresses from the other loadings required and the combination was shown to meet ASME Code, Section III requirements.

Additionally, the SRT verified that the CRDMs have seismic support tie rods to resist seismic loads (References 25 and 26). Therefore, CRDMs are screened from further review.

5.8 DISTRIBUTION SYSTEMS

The following sections address the distribution systems; cable tray and conduit, HVAC duct and piping.

5.8.1 Cable Tray and Conduit

Cable tray and conduit were reviewed on an area basis during SRT equipment and subsystem walkdowns to identify any anomalies that could lead to failure. No such anomalies were observed. Cable and conduit are screened from further review based on Appendix A of Reference 3, and SRT walkdowns.

5.8.2 HVAC Duct

HVAC duct was reviewed on an area basis during SRT equipment and subsystem walkdowns to identify any anomalies that could lead to failure. Two specific cases were noted during these walkdowns which are described below:

1. A large span of duct above the auxiliary feedwater (AFW) motor driven pump, 1A-SA require further review. The duct is 30 inches by 30 inches with a 24 ft. span between supports. This span has a vertical branch duct with a register attached to the bottom. The original design calculations were reviewed and was judged acceptable for RLE. Refer to the SEWS for AFW motor driven pump, 1A-SA for the basis of acceptance.

2. A non-safety non-seismic duct was noted as a potential interaction above motor operated valve, 1SW-40. The duct is 10 inches by 10 inches with duct span of approximately 14 ft. The duct and supports were judged to be acceptable for RLE. Refer to the SEWS for motor operated valve 1SW-40.

HVAC duct are screened from further review based on Appendix A of Reference 3, and SRT walkdowns.

5.8.3 Piping

Harris Seismic Category I, ASME, Section III, Safety Class 2 and 3 piping is analyzed in accordance with ASME B & PV code, Section III, subarticles NC/ND. The design criteria was based on formulations and allowable stress limits given by in ASME, Section III, subarticle NC/ND-3600, with load combinations which consider operating basis earthquake (OBE) and SSE effects along with other coincident loading conditions. The two orthogonal horizontal and vertical loadings were obtained from the appropriate floor response spectra.

Piping system were reviewed on an area basis during SRT equipment and subsystem walkdowns. The SRT looked for any anomalies related to potential displacement induced failure modes. No such anomalies were observed.

Additionally, the SRT looked for potential failure modes of piping system appurtenances such as instrument tubing and associated instruments, vent valves

and drain valves. Seismic interaction and seismic anchor motion were considered potential failure modes for small bore lines attached to larger piping systems. No anomalies noted that could lead to the loss of a pressure boundary of a success path list system were observed.

Containment penetrations were also reviewed on an area basis to identify any anomalies that may affect containment performance. Anomalies such as seismic interaction (falling) and differential building displacement were considered. A walk-by of containment isolation valves for the intent of the caveats identified on the valve SEWS was also performed. No anomalies that could effect containment performance were observed.

Harris piping was screened from further review based on Appendix A of Reference 3, and SRT walkdowns.

5.9 OTHER COMPONENTS

The following sections discuss the in-core flux mapping system and masonry walls.

5.9.1 In-core Flux Mapping System

In 1984, CP&L discovered that the In-Core Flux Mapping System was not seismically designed. Seismic induced failure of the In-Core Flux Mapping System components above the seal table could cause failures of the flux mapping tubing or fittings which would produce a small break LOCA. CP&L notified Westinghouse of the potential failure of the In-Core Flux Mapping System and informed the Nuclear Regulatory Commission (NRC) of a potentially reportable item under both 10CFR50.55(e) and 10CFR21. Information Notice 85-45, (Reference 22) was issued as a result of this information provided by CP&L. CP&L notified the NRC that this item was reportable under 10CFR50.55(e) and 10CFR21 on January 16, 1985.

CP&L designed and installed wheel stops on the Flux Mapping Control System Trolley to prevent seismic interaction between the trolley and the In-Core instrumentation tubing or fittings. The wheel stops are of substantial size and are fabricated from 1/2" and 3/4" thick structural steel plate. The design details for the wheel stops are provided on Drawing CPL-2168-S-9479. The installation of the wheel stops was visually verified by using the Harris Surrogate Tour system. The tour program uses the combined technology of laser videodiscs and the personal computer to effectively simulate travel through the plant and provide detailed visual information about plant areas. The surrogate tour shows the wheel stops installed. The wheel stops have sufficient design margin to be screened out for the RLE.

5.9.2 Masonry Walls

The guidelines for assessment of nuclear power plant seismic margin (Reference 3) recommends a review of masonry walls, in particular unreinforced or lightly reinforced walls. There are no unreinforced masonry walls located in safety related areas at Harris.

The masonry (block) walls in the Containment Building and Diesel Generator Building are all seismically designed. All masonry walls in the Reactor Auxiliary Building are seismic except in the hot shop area on Elevation 236 ft., between 'B' and 'H' lines and '43' and '45' line where seismic and non-seismic walls are utilized. Both seismic and non-seismic walls are used in the Fuel Handling Building and Waste Processing Building. These walls are utilized for shielding and equipment removal purposes or support of non-safety equipment.

The following codes are used for the analysis and design of masonry block walls and any associated steel framing:

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

- UBC-79, Uniform Building Code, by International Conference of Building Officials
- AISC, American Institute of Steel Construction, "Specification for Seventh the Design, Fabrication and Erection of Structural Steel for Edition Buildings"
- ACI 318-71, American Concrete Institute, "Building Code Requirements for Reinforced Concrete"

Also, the design criteria utilized in the design of masonry walls located in Seismic Category I structures complies with the NRC Structural Engineering Branch Criteria for Safety Related Masonry Wall Evaluation, dated July 1981.

Four masonry walls were chosen for further evaluation to demonstrate the high design margin that exists. These four specific cases were noted being adjacent to SSEL components during the SRT walkdown. The wall location, description, allowable stress level, calculated stress level and interaction coefficient for the four walls are given in Table 5-6.

The maximum interaction coefficient of the four masonry walls is 0.53. The RLE to DBE scaling factor is approximately 1.6. Therefore, these masonry walls are screened out for RLE. These walls are typical of other safety related masonry walls within the plant which are also screened out for RLE based on the above evaluations. No further action is required.

Table 5-1

SUMMARY OF CIVIL STRUCTURES SEISMIC MARGIN EVALUATION

(Format Follows EPRI NP-6041, Table 2-3)

TYPE OF STRUCTURE	DISPOSITION
Concrete containment	Screened based on EPRI NP-6041, Table 2-3
Containment internal Structure	Screened based on EPRI NP-6041, Table 2-3. The structure was designed for greater than 0.1g.
Shear walls, footing and containment shield walls	Screened based on EPRI NP-6041, Table 2-3. The walls were designed for greater than 0.1g.
Diaphragms	Screened based on EPRI NP-6041, Table 2-3. Diaphragms were designed for greater than 0.1g.
Category I concrete frame structures	Screened based on EPRI NP-6041, Table 2-3. Concrete frame structures were designed for greater than 0.1g. See Section 5.5 for a summary of the evaluation.
Masonry walls	Screened based on existing analyses. See Section 5.9.2 for a summary.
Control room ceilings	Screened. The control room ceiling was judged adequate by the SRT.
Impact between structures	Screened based on EPRI NP-6041, Table 2-3
Category II structures with safety-related equipment or with potential to fail Category I structures	Screened based on EPRI NP-6041, Table 2-3 and SRT walkdowns.
Dams, levees, dikes	Designed to a factor of safety of 1.1 for simultaneous SSE and 25 year flood. Further evaluation is not expected to identify any significant vulnerabilities. See Section 5.6.
Soil failure modes	Screened based on the FSAR. See Section 5.6 for a summary.

Table 5-2

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

(Format Follows EPRI NP-6041, Table 2-4)

EQUIPMENT TYPE	DISPOSITION
NSSS primary coolant system	Screened based on EPRI NP-6041, Table 2-4. See Section 5.7 for a summary of the evaluation.
NSSS supports	Screened. See Section 5.7 for a summary of the evaluation.
Reactor internals	Screened. See Section 5.7 for a summary of the evaluation.
CRDM mechanisms	Screened. See Section 5.7 for a summary of the evaluation.
Category I piping	Screened by the SRT. See Section 5.8.3.
Active valves	Screened based on EPRI NP-6041, Table 2-4 and SRT walkdowns. See Section 5.4.7, 5.4.8, and 5.4.25.
Passive valves	Screened based on EPRI NP-6041, Table 2-4.
Heat Exchangers	The RHR Heat Exchanger HCLPF is estimated to be 0.29g (see Section 6). All other heat exchangers were screened by the SRT, see Sections 5.4.23 and 5.4.24.
Atmospheric storage tanks	RWST and CST HCLPF capacities are much greater than the RLE (see Section 6 for evaluation results). All other tanks were screened by the SRT, see Sections 5.4.23 and 5.4.24.
Pressure vessels	Screened by the SRT. See SEWS for each success path pressure vessel.
Buried tanks	The diesel fuel oil storage tanks were considered to be buried tanks. These tanks consisted of a Seismic Category I concrete structure that has an internal steel liner. See Sections 5.4.22 and 5.5.6.
Batteries and racks	Screened by the SRT. See SEWS for battery racks. See Section 5.4.15.

Table 5-2 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

EQUIPMENT TYPE	DISPOSITION
Diesel generators	Screened by the SRT. See diesel generator SEWS. See Section 5.4.17.
Horizontal pumps	Screened by the SRT. See horizontal pump SEWS. See Section 5.4.5.
Vertical pumps	Service Water Pump HCLPF is greater than the RLE (see Section 6). All other vertical pumps were screened by the SRT. See Section 5.4.6.
Fans	Screened by the SRT. No fans are supported on vibration isolators. See individual SEWS. See Section 5.4.9.
Air handlers	Screened by the SRT. No air handlers are supported on vibration isolators. See individual SEWS. See Section 5.4.10.
Chillers	Screened by the SRT. No chillers are supported on vibration isolators. See individual SEWS. See Section 5.4.11.
Air Compressors	Screened by the SRT. No air compressors are supported on vibration isolators. See diesel air start SEWS. See Section 5.4.12.
HVAC ducting and dampers	Screened by the SRT. See Section 5.8.2.
Cable trays	Screened by the SRT. See Section 5.8.1.
Electrical conduit	Screened by the SRT. See Section 5.8.1.
Electrical power distribution panels, cabinets, switchgear, motor control centers	HCLPF capacities were estimated to be greater than the RLE for low voltage switchgear and miscellaneous electrical panels. See Section 6 for a summary of the evaluations. All other electrical gear were screened by the SRT. See Sections 5.4.1, 5.4.2, 5.4.3 & 5.4.14.

Table 5-2 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

EQUIPMENT TYPE	DISPOSITION
Transformers	Included in the low voltage switchgear and MCC's, see above item.
Battery chargers	Screened by the SRT. See battery charger SEWS. See Section 5.4.16.
Inverters	Screened by the SRT. See inverter SEWS. See Section 5.4.16.
Instrumentation and control panels and racks	Screened by the SRT. See panel and rack SEWS. See Sections 5.4.19 and 5.4.21.
Temperature sensors	Screened by the SRT. See temperature sensor SEWS. See Section 5.4.20.
Pressure and level sensors	Screened by the SRT. See instrument SEWS. See Section 5.4.18.

Table 5-3

COMPONENTS REQUIRING MAINTENANCE REPAIRS

EQUIP ID NO.	EQUIPMENT DESCRIPTION	LOCATION	COMMENT	WR/JO NO.
1A-SA	1DG-E036 Diesel generator control panel	DGB 261'	Close both ends of the "S" hooks on the hanging fluorescent lights in the DG control panel room	94-ANQC1
1B-SB	1DG-E037 Diesel generator control panel	DGB 261'	Close both ends of the "S" hooks on the hanging fluorescent lights in the DG control panel room	94-ANQE1
1A1,1A2 2A1 & 2A2	1EE-E064, 65, 68 and 69 Isolation cabinets	RAB 305'	The row of fluorescent lights needs to be relocated on the existing Unistrut three inches to the west (away from the cabinets)	95-ACLT1
AUX-RLY-PNL-19A	Train A Aux. relay panel 19A	RAB 305'	Install missing mounting screw for relay CR-1-1273	94-AAPQ1
AUX-RLY-PNL-19B	Train B Aux. relay panel 19B	RAB 305'	Install three missing mounting screws relay CR-1191	94-AAPP1
1B31-SB	MCC 1B31-SB (Includes panel and transformer PP-1B311-SB)	RAB 286'	Replace missing latch on breaker panel door missing latch	90-AEMY1
1A32-SA	Motor control center 1A32-SA	ESWS 261'	Replace missing latch on breaker panel door	92-AFHI1
1B32-SB	Motor control center 1B32-SB	ESWS 261'	Replace missing latch on breaker panel door	89-AISZ1
IDP-1A-SI	120 Vac instrument panel (Includes IDP-1A-SIII)	RAB 305'	Replace three missing access panel screws, and replace two missing latches for breaker panel doors	94-ANKZ1 and 94-ANXK1

Table 5-3 (Continued)

COMPONENTS REQUIRING MAINTENANCE REPAIRS

EQUIP ID NO.	EQUIPMENT DESCRIPTION	LOCATION	COMMENT	WR/JO NO.
IDP-1B-SII	120 Vac instrument panel (Includes IDP-1B-SIV)	RAB 305'	Replace three missing access panel screws	94-ANXJ1
1TIS-658A	RHR heat exchanger 1A-SA temp. switch	RAB 236'	U-Bolt has slipped off top of pipe and needs to be reworked	94-ANXL1
A21-R15	Instrument rack, LT-9010B	TANK 236'	Close both ends of the "S" hooks on the hanging fluorescent light above rack	94-APBY1
1AF-55	Electric motor operated valve	MST 261'	Compressed gas cylinder adjacent to valve needs to be removed or restrained in accordance with AP-003	Compressed gas cylinder has been removed

Table 5-4

ITEMS REQUIRING REPAIR OR MODIFICATION

ENGINEERING SERVICE REQUEST (ESR) 94-152

EQUIP ID NO.	EQUIPMENT DESCRIPTION	LOCATION	COMMENT
1B-SB	1DG-E037 Diesel generator control panel	DGB 261'	Four of the hanging fluorescent fixtures in the DG control panel room "B" are not the spring loaded fluorescent tube design. These fixtures require fluorescent tube clamps/clips to prevent the tube from coming loose due to vibration.
AUX-RLY-PNL-4A	Train A Aux. relay panel 4A	RAB 305'	Auxiliary relay panel 4A needs to be physically attached w/ cabinet main termination cabinet 11A-SA . Potential impact/interaction between cabinets is not acceptable since 4A contains essential relays.
AUX-RLY-PNL-1B, 2B, 3B, AND 4B	Train B Aux. relay panels: 1B, 2B, 3B, and 4B	RAB 305'	The fuel status board (which is on wheels) needs to be restrained, possibly with cables to the wall. Potential impact/interaction with cabinets is not acceptable since cabinet contains essential relays.
AUX-RLY-PNL-19B	Train B Aux. relay panel 19B	RAB 305'	
MAIN-CTRL-BOARD	Main control board -1AA, 1A1, 1A2, 1BB, 1B1, 1B2, 1C, 1D1 and 1D2	RAB 305'	Cabinets, carts, and etc. that are to the south of the main control board need to be restrained. Potential impact/interaction with Main Control Board (MCB) is not acceptable.
DP-1A-SA	Distribution panel for vital DC 1A-SA	RAB 286'	Relay access door not properly secured - stripped screws. The door/fastening mechanism needs to be repaired.
DP-1B-SB	Distribution panel for vital DC 1B-SB	RAB 286'	Relay access door not properly secured - stripped screws. The door/fastening mechanism needs to be repaired.

Table 5-5

COMPONENTS REQUIRING HCLPF CALCULATIONS

GROUP	EQUIP ID NO.	EQUIPMENT DESCRIPTION
1	1A	Transfer panel 1A
1	1B	Transfer panel 1B
1	1A-SA	1DG-E038 ESS panel
1	1B-SB	1DG-E039 ESS panel
1	DP-1A-SA	Distribution panel for vital 1A-SA
1	DP-1B-SB	Distribution panel for vital 1B-SB
2	1A2-SA	Low voltage switchgear 1A2-SA
2	1A3-SA	Low voltage switchgear 1A3-SA
2	1B2-SB	Low voltage switchgear 1B2-SB
2	1B3-SB	Low voltage switchgear 1B3-SB
3	1A-SA	RHR heat exchanger 1A-SA
3	1B-SB	RHR heat exchanger 1B-SB
4	1X-SAB	Condensate storage tank
4	1X-SN	Refueling water storage tank
5	1A-SA	ESW suction pump A
5	1B-SB	ESW suction pump B

Table 5-6

REVIEW OF MASONRY WALLS ADJACENT TO SSEL COMPONENTS

WALL	DESCRIPTION	COMPRESSION		INTERACTION fa/Fa + fm/Fm	SHEAR
		AXIAL	FLEX		
DGB EL. 261'	Type IV	fa = 21 psi Fa = 633 psi	fm = 125 psi Fm = 955 psi	0.16	fv = 2 psi Fv = 48 psi
RAB EL. 236' STAIR NO. A4	North Wall Type II	fa = 32 psi Fa = 513 psi	fm = 153 psi Fm = 950 psi	0.22	fv = 10 psi Fv = 48 psi
	South Wall Type II	fa = 37 psi Fa = 547 psi	fm = 382 psi Fm = 950 psi	0.46	fv = 26 psi Fv = 48 psi
	West Wall Type II	fa = 29 psi Fa = 535 psi	fm = 287 psi Fm = 950 psi	0.36	fv = 15 psi Fv = 48 psi
RAB EL. 261' STAIR NO. A3	North Wall Type II	fa = 29 psi Fa = 535 psi	fm = 225 psi Fm = 950 psi	0.25	fv = 12 psi Fv = 48 psi
	East & West Walls Type II	fa = 4 psi Fa = 585 psi	fm = 160 psi Fm = 950 psi	0.18	fv = 23 psi Fv = 48 psi
RAB EL. 305'	West Wall Type II	fa = 39 psi Fa = 553 psi	fm = 433 psi Fm = 955 psi	0.53	fv = 31 psi Fv = 48 psi
	North, South & East Walls Type II	fa = 31 psi Fa = 535 psi	fm = 301 psi Fm = 955 psi	0.38	fv = 26 psi Fv = 48 psi

6. ASSESSMENT OF ELEMENTS NOT SCREENED OUT

Sixteen items were selected for HCLPF evaluation by the SRT. The items were grouped into 5 HCLPF calculations based on similar characteristics. Results of the HCLPF evaluations are summarized below.

1. Panel Anchorage with 45° Nelson Studs

   The SRT observed several panels that met all screening guidelines during the field visual review, however, a suspect load path to structural concrete was noted during a drawing review. Embedded 6 x 8.2 structural channels are anchored to a raised pad via Nelson studs welded to the inside corners of the channel. The detail was considered suspect because the studs are relatively small and short (3/8" diameter, 2-9/16" long, and spaced at 30").

   Initial analyses indicated HCLPF capacities of less than 0.3g. Upon further refinement of the model, assuming shear loads are transferred directly to the concrete by the channel legs rather than through the studs, a HCLPF capacity of 0.59g was calculated. The governing failure mode was determined to be concrete cone failure. The evaluation is documented in Reference 11.

2. Low Voltage Switchgear

   Low voltage switchgear were selected for HCLPF evaluation due to a test response spectrum (TRS) that did not exceed the scaled required response spectrum for RLE by an appreciable margin. Additionally, transformer anchorage anomalies were observed during the test and the anchorage calculation for the switchgear indicated a small margin using SSE loads.

   Initial analyses indicated a HCLPF capacity significantly less than 0.3g due to the low margin between the TRS and RRS. Seismic floor response spectra at the location contain a sharp peak between about 6 and 8 Hz,

within the frequency range of interest. The spectra were peak clipped in accordance with Reference 3 to significantly reduce the RLE seismic demand. The resulting HCLPF is 0.35g.

Transformer anchorage details observed were substantially more robust than the details that experienced anomalies during the test, leading to an anchorage HCLPF of 0.70g. The evaluation is documented in Reference 12.

3. RHR Heat Exchangers

Harris RHR heat exchangers were selected for HCLPF evaluation since upper lateral restraints for the vertical heat exchangers were not installed. Top supports were included in the original design but were not installed based on analyses that justifies the configuration for DBE levels.

The CCW inlet piping attached near the top of the RHR heat exchangers is supported differently for Train A and B. The piping for the Train B heat exchanger has more supports than the Train A piping. The stress analysis of the piping gives higher nozzle loads on the Train B heat exchanger than on the Train A heat exchanger even though Train B has more supports on the inlet piping. This was noted as an observation during the peer review walkdown.

Upon review of the two stress calculations, the response spectra and damping levels used in the two analysis were not the same. The Train B stress analysis conservatively used one percent and two percent damping for OBE and DBE, respectively and enveloped the Reactor Auxiliary Building and Containment Building spectra for elevation 261' and below. The Train A stress analysis used code case N411 (PVRC) damping for Reactor Auxiliary Building, elevation 261' and below. Both methods are acceptable with Train B being more conservative. The piping configuration and supports are correctly reflected in each stress analysis. The higher loads on the Train B

heat exchanger are attributed to the difference in the response spectra used in the stress analysis.

Anchor bolts holding the based of the RHR heat Exchangers to the foundation were identified as the critical item. The HCLPF was estimated by scaling existing analyses and eliminating some of the conservatism. Conservatism in the combination of heat exchanger nozzle loads was eliminated by combining seismic loads by square-root-sum-of-squares and actual concrete strength was used rather than design strength. This led to a HCLPF of 0.29g. It should be noted that additional margin may be extracted from piping dynamic analyses for nozzle loads. The evaluation is documented in Reference 13.

4. CST and RWST Flat Bottom Storage Tanks

The HCLPF capacity of the Condensate Storage Tank (CST) and the Refueling Water Storage Tank (RWST) were estimated since these tanks are important and high profile elements of the shutdown paths.

The HCLPF capacity is calculated to be over 1g based on the methodology provided in Reference 3. The evaluation is documented in Reference 14.

5. Emergency Service Water Pumps

The emergency service water pumps were selected for HCLPF evaluation because the seismic qualification indicated low margin to allowable stress levels in the long vertical column.

The HCLPF capacity was estimated by scaling existing analyses, considering the pump column and sole plate as potential failure modes. The seismic contribution to pump column stresses was low due to the numerous bearing supports. The governing failure mode was determined to be the sole plate with a sufficiently high HCLPF of 0.67g. The evaluation is documented in Reference 15.

Four of the 5 groups of equipment have a HCLPF capacity of greater than the RLE. The RHR Heat Exchanger HCLPF capacity is calculated to be 0.29g. Further refinement of the associated piping analyses to eliminate any unnecessary conservatism in the nozzle loads may raise the HCLPF capacity above the RLE. Conservatism also exists in the RLE spectra as discussed in Section 4. This refinement is not considered necessary since the HCLPF capacity is essentially equal to the RLE.

7. RELAY EVALUATION

As described in NUREG-1407, for the focused-scope plants (including Harris), the evaluation of relay chatter emphasizes consideration of relays with known low seismic ruggedness. These relays are to be evaluated for potential plant impact. The seismic margins analysis (SMA) process developed by EPRI is being used to satisfy the generic letter for Harris. For the SMA methodology, the effect on the plant safety is determined by examining the potential impact of chattering of low-ruggedness relays on the operation of the equipment in the safe shutdown paths. If it can be shown that there are no low-ruggedness relays, or that they would have insignificant effect if they chattered during an earthquake, the evaluation is complete.

The relay study was performed by Safety and Reliability Optimization Services, Inc. (SAROS) and documented in SAROS Report No. 93-9, Reference 20. The relay study focused on identifying low-ruggedness relays. Fifty-one relays were evaluated because they were identified to be potentially low-ruggedness models. These identifications were made based on a comparison of the makes, models, and configurations (energized vs. de-energized, normally open vs. normally closed contact pairs) to the list of low-ruggedness relays provided in Appendix E of EPRI NP-7148-SL, Reference 18.

Further review of these 51 relays were performed with the following results, Reference 20, Section 4:

- Twelve of the 51 relays were determined not to be low-ruggedness relays because they were not in a configuration that had been determined to be subject to chatter for the relevant model.

- Twenty-nine of the 51 relays were determined to be non-essential relays in which chatter is not a concern.

- Four of the 51 relays were determined to be essential. However, chatter of these four relays would not produce any unacceptable consequences.

- The remaining six relays, of the 51, were determined to be essential relays. These are the differential relays (one for each phase) that would actuate the lockout relays on the two 6.9kv emergency buses (switchgear units). These relays are all General Electric model 12PVD21B1A relays. The equipment tag no. for these six relays are 87SA-A-1738, 87SA-B-1738, 87SA-C-738, 87SB-A-1739, 87SB-B-1739 and 87SB-C-1739. The relays are mounted in the doors of cubicle 10 and 6 on the emergency buses 1A-SA and IB-SB, respectively. The relays are mounted within 18 inches of the floor. Further discussion on these relays is given below:

This make and model relay, GE 12PVD21B1A, is on the low-ruggedness relay list as a result of Licensee Event Report (LER) No, 84-020, Docket No. 352, Reference 23. The event involved an inadvertent trip of the D14 bus 'A' phase differential relay which caused the D14 bus to de-energize. The cause of the event was described as follows: "Maintenance personnel were working on a door stop inside breaker cubicle number 6 of bus D14. The D14 bus 'A' phase differential relay is mounted on the door of this cubicle. While work was being performed on the door stop the 'A' phase differential relay was bumped causing the relay to trip and isolate the bus." This is defined as a high frequency vibration issue.

The configuration of these relays provide an inherent seismic margin. The following description of the differential relay function is presented in Reference 24, Section 4: "These differential relays are normally de-energized; when two of three on a particular bus are operated, they cause the lockout relay for that bus to operate. Each of the differential relays has three coils, each with a set of auxiliary contacts. Simultaneous chatter of these contact pairs on two or more of the relays for a

particular bus would cause that bus to be locked out. Operation of the lockout relay would cause the associated emergency diesel generator to trip, and would trip open the supply breakers to the bus and the feeds to the 6.9kv safety equipment and to the 6.9kv/480v transformers. If this were to occur on both buses, a station blackout would result. These breakers could not be re-closed or the diesel generators started until the lockout relays were manually reset. This could only be done locally (i.e., at the respective 6.9kv emergency bus). Although there are seal-in circuits for the differential relays, these relays would be de-energized after the lockout relays had operated. After the initial chatter, the contact pairs should be in their normal (open) positions. They should not impede subsequent attempts to reset the lockout relays."

Sufficient margin exists in the seismic qualification of the relay that inadvertent trip of the relay is not a concern. The 6.9kv emergency buses, including the differential relays and the lockout relays are seismically qualified per the Seimens-Allis Qualification Report No. 90365-1, Reference 24. As described above, chatter on two of three differential relays is needed for the lockout relay to operate. When the lockout relay is operated the associated bus would be de-energized. During the seismic test, the differential relays were not monitored for contact chatter. However, they were energized and configured consistent with design requirements. The lockout relay was monitored for contact chatter greater than 2 milliseconds. This monitoring would detect any unacceptable chatter of the differential relays. No contact chatter greater than 2 milliseconds was recorded for the lockout relay during the seismic testing.

The SSE test response spectra (TRS) envelops the review level earthquake (RLE) for the Reactor Auxiliary Building, Unit 1, at elevation 286'. The SSE TRS horizontal ZPA for the front-to-back and side-to-side directions were recorded as 1.87g (test run 11) and 1.90g (test run 20), respectively, Reference 24, Table I. The RLE ZPA for the Reactor Auxiliary Building, Unit 1, at elevation 286' is 0.773g, N-S and 0.613g, E-W. This results in a margin of 2.4 above the RLE and a margin of 3.9 above the Design Basis Earthquake for Harris.

In addition to the review of the seismic qualification, the emergency busses were reviewed for interaction issues during the SRT walkdown. No interaction issues were noted that would cause high frequency vibration to the emergency buses. The floor in front of cubicle 6 and 10 for emergency buses 1B-SB and 1A-SA, respectively are painted yellow the full width of the door and approximately 18 inches out from the door. This yellow area has black lettering stating "STAY CLEAR." Cautionary labels are also attached to the cubicle doors. These labels state the follow "Vibration Sensitive Relays. Jarring Switches or Cabinet Will Cause a Reactor Trip." These labels are precautionary measures that identify the sensitivity of these components. Therefore, inadvertent relay trip is not a concern for the differential relays located on the emergency buses based on the system configuration, seismic qualification review, no interaction issues, and the cautionary labels on and in front of the cubicles with the relays.

The relay study also noted some relays with unidentified make and models, see Tables 5 and 6 of SAROS Report No. 93-9. Make and models were subsequently identified through further reviews. None of the relays listed in Tables 5 and 6 were identified to be low-ruggedness relays and the Equipment Database System (EDBS) has been updated to reflect the results of the further reviews.

The results of the relay evaluation meets the intent of generic letter 88-20 and no further action is required.

8. SEISMIC INDUCED FIRE AND FLOOD EVALUATION

Seismic/fire interactions, effects of suppressants on safety equipment, and control system interaction should be addressed in the IPEEE per NUREG-1407, Reference 2. The majority of seismic/fire issues are identified in the NUREG/CR-5088 "Fire Risk Scoping Study," Reference 16 and NRC Information Notice 94-12, "Insights Gained from Resolving Generic Issue 57," Reference 17. A description of the fire suppression systems is given below. Specific fire issues are addressed later in this section.

The following four types of automatic fire suppression systems are used at Harris:

1. Water spray system
2. Wet pipe sprinkler system
3. Pre-action sprinkler system
4. Multi-cycle sprinkler system

The water spray system consist of dry pipe with a one-step release. The water flow is controlled by a deluge valve which is activated by a signal from a thermal detector. The water is discharged through open sprinkler heads. This type of sprinkler system is used in the Turbine Building. There is no success path equipment located in this area.

The wet pipe sprinkler system are supplied through hydraulically designed piping systems charged with water up to the sprinkler heads. Water is discharged through the sprinkler heads upon the melting of fusible links. This type of sprinkler system is provided in areas where inadvertent operation will not have a negative effect, such as the Warehouse and Administration Building. There is no success path equipment located in these areas.

The pre-action sprinkler system is a two-step water release system. The system is supplied through hydraulically designed piping, which downstream of the deluge valve controlling the water flow contains air under supervisory air pressure. The

deluge valve is automatically actuated by a thermal fire detection system installed in the same area as the sprinklers, and are manually shut off after the fire is extinguished. Pre-action systems are provided in safety related or non-safety related areas where inadvertent sprinkler operation is not acceptable. Pre-action systems limit discharge because water will not flow unless the pre-action valve is opened by a signal from the fire detection system and the sprinkler heads near the fire are fused open by heat from the fire.

The multi-cycle sprinkler system is a pre-action system modified to provide the capability for on and off cycling. Because of their ability to limit water discharge, multi-cycle sprinklers are used in areas where radioactivity may be a problem and the drainage from these areas must be treated or areas where drainage is pumped out of the area, and water discharge should be limited.

Piping for all four types of automatic fire suppression systems is seismically supported to assure system pressure integrity after a safe shutdown earthquake (SSE). Piping and valves are designed to comply with ANSI B31.1.

Standpipe and hose stations are located throughout the plant at approximately 100 ft intervals so that all portions of the plant can be reached by two effective hose streams. This system is basically passive and is not susceptible to inadvertent actuation. The Post SSE Standpipe system piping is analyzed for SSE loading and seismically supported to assure system pressure integrity after a SSE.

The following seismic/fire issues are identified in the NUREG/CR-5088 "Fire Risk Scoping Study," Reference 16:

- **Identify unanchored CO_2, halon, oxygen or hydrogen tanks.**

  Gas bottles are stored in the following locations:

  1. RAB, elev. 236' outside the Reactor Containment Building (RCB) personnel hatch - These bottles are stored in seismically designed storage rack.

2. RAB, elev. 236', hot machine shop located in the Unit 2 side of the RAB common area - The hot machine shop does not contain permanent plant equipment.

3. RAB, elev. 261' between column lines Gz/43 and H/43 in the Unit 2 side of the RAB common area - This area does not contain any permanent plant equipment.

4. RAB, elev. 286' at column line E/13 - A halon bottle/cart is located away from permanent plant equipment and is chained to the wall.

5. RAB, elev. 286' stairwell near column line D/35 - A halon bottle/cart is located under the stairs. The stairwell does not contain permanent plant equipment.

   A first aid storage cabinet and other non-plant equipment is stored in the stairwell. The first aid cabinet potentially could impact the halon bottle during a seismic event. This does not affect any safe shutdown equipment. A potential personnel safety issue is a concern if personnel need to use the stairwell. An adverse condition and feedback report (ACFR) number 95-0128 was initiated to document this condition. As a result of this ACFR, the halon bottle has been relocated to the turbine building.

   The gas bottles are adequately stored/restrained within the power block to preclude any damage to SSEL equipment and permanent plant equipment.

- **Identify actuation systems that are sensitive to vibration, relay chatter, and/or locking circuits:**

Typical relays contained within the fire protection system are listed in Table 8-1.

These are small relays that plug-in to printed circuit board cards within the fire detection panels. These are not low-rugged relays with respect to the list of low rugged relays provide in Appendix E of EPRI NP-7148-SL, Reference 18. Therefore, no further action is required.

- **Identify fire detection systems with only ionization detectors where dust may cause a spurious alarm:**
  Table 8-2 lists areas within buildings that contain success path items that have ionization detectors only. This may result in an alarm but will not cause an inadvertent actuation of the fire suppression system.

- **Identify fire pumps that may have weak mounts or vibration mounts:**
  There are three non-safety related fire pumps located at the Emergency Service Water Screening Structure. The pumps are anchored to the concrete without vibration isolation mounts. In addition, if a seismic event affects the operation/ functionality of these pumps, the backup water supply can be provided by the Emergency Service Water pumps to the Post SSE Standpipe System, per the Design Basis Document (DBD) No. 306, "Fire Protection and Detection System," Reference 19. Therefore, the mounting of the non-safety related fire pumps is not a concern.

- **Identify fire mains that are of cast iron material:**
  The material for the outside underground piping is ductile iron per DBD No. 306, Reference 19. The standpipe, hose system

and automatic suppression system piping material is carbon steel. No cast iron material was identified within the fire piping system.

- **Verify that all electrical cabinets are properly anchored and have sufficient slack in the cables entering the cabinet:**
  The SRT walkdown included an interaction review between SSEL electrical cabinets and adjacent non-SSEL electrical cabinets. The SRT walkdown also verify that there is sufficient slack existing in the cables entering the electrical cabinets. The results from the SRT walkdown did not identify any vulnerabilities in this area.

- **Identify credible interactions between sprinkler systems and adjacent piping:**
  The automatic fire suppression system piping is seismically supported to assure system pressure integrity after a SSE. No interaction issues between the sprinkler system and adjacent piping were noted during the SRT walkdown.

The following seismic/fire issues are identified in NRC Information Notice 94-12, "Insights Gained from Resolving Generic Issue 57," Reference 17:

- **Mercury Relays:**
  No mercury relays have been located in the fire protection circuits.

- **Seismic Dust/Smoke Detectors:**
  Several fire zones within the power block have ionization detectors only which sends an alarm to the control room. Actuation of the fire suppression system (pre-action sprinklers)

is only by thermal detection and, is independent of the detection. Therefore, inadvertent actuation will not occur.

- **Water Deluge Systems:**
  The SRT walkdown included an interaction review for potential sources that could flood or spill onto the electrical cabinets. This is specifically addressed on the SEWS form. No interaction issues, with respect to flooding or fire protection systems flooding electrical cabinets, were noted during the SRT walkdown. No further action is needed.

- **Fire Suppressant Availability During a Seismic Event:**
  There are three non-safety related fire pumps located at the Emergency Service Water Screening Structure. If a seismic event affects the operation/functionality of these pumps, the backup water supply can be provided by the Emergency Service Water pumps to the post SSE standpipe system, per the Design Basis Document (DBD) No. 306, "Fire Protection and Detection System," Reference 19. The post SSE standpipe system piping is analyzed for SSE loading and seismically supported to assure pressure integrity after a SSE. Therefore, the operation/functionality of the non-safety related fire pumps is not a concern.

- **Switchgear Fires:**
  The SRT walkdown included screening/evaluation of switchgear. The SRT walkdown verified adequate switchgear anchorage, sufficient slack in cables entering the switchgear and sufficient separation to other electrical cabinets. The Train "A" and Train "B" switchgear are also physically separated by a reinforced concrete wall. The results from the

SRT walkdown did not identify any seismic/fire interaction concerns with regards to the switchgear.

- **Electro-Mechanical Components in Cable Spreading Room:** The cable spreading rooms only contains two electrical cabinets, Auxiliary Transfer Panel SA and SB. The SSEL includes both of these electrical cabinets and were screened out from further review during the SRT walkdown. These electrical cabinets are seismically anchored which addresses this issue. No interaction issues were noted in the cable spreading rooms.

In addition to the seismic/fire interactions addressed above, fire sources are to be identified and evaluated. The following fire sources were identified:

1. **Hydrogen Hazard** - Hydrogen is supplied to the Turbine Building from the Hydrogen storage tank for the main generator and from the Turbine Building into the Reactor Auxiliary Building to the Volume Control Tank. The hydrogen lines within the Turbine Building are not a concern since this area does not contain any SSEL equipment. The hydrogen line in the Reactor Auxiliary Building is non-safety, non-seismic from the Turbine Building to the air operated valve 1-CS-322 and is Class 2 from valve 1-CS-322 to the Volume Control Tank. Two exceedances of the recommended ANSI B31.1 spans were observed. Both exceedances are approximately 15 foot spans. The 15 foot spans are not considered a significant seismic hazard based on the inherent ruggedness of small bore socket welded pipe as demonstrated by earthquake experience, test and analytical data.

2. **Liquid Fire Hazard** - There are two liquid fire hazards that were identified in areas where safety related equipment are present; the diesel generator fuel oil day tanks located in the Diesel Generator Building, and the reactor coolant pump (RCP) oil collection system in the Containment Building. The seismic walkdown list included the diesel generator fuel oil day tanks and were screened out from further review during the SRT walkdown. The RCP oil collection system was evaluated by reviewing the oil collection system drawings, CP&L drawing no. 1364-53480, Reference 29 and viewing the RCP's using the Harris Surrogate Tour system. The tour program uses the combined technology of laser videodiscs and the personal computer to effectively simulate travel through the plant and provide detailed visual information about plant areas. The oil collection system is well supported and is screened out based on these reviews.

Other sources of liquid fire hazards are the lube oil systems of various pumps (excluding the RCP's). These sources are insignificant in terms of risk and can be ignored in the seismic/fire walkdown. Even though these may be insignificant, the lube oil systems of the pumps included on the SSEL were considered rule-of-box and evaluated/screened with the pump during the SRT walkdown. The results from the SRT walkdown did not identify any vulnerabilities in this area.

No further action is required as a result of the seismic/fire evaluations reviews and walkdown.

Table 8-1

TYPICAL FIRE PROTECTION SYSTEM RELAYS

Description	Make	Model
Relay, Signal, 24VDC	Potter & Brumfield	R10-E6210-1 & R10-E1x4-V700
Relay, Transfer, Power	Potter & Brumfield	R10-E1-M2
Relay, Trouble (Alarm Only)	Telemecanique	TF154-2C-2C

Table 8-2

FIRE ZONES WITH ONLY IONIZATION DETECTORS

BUILDING	ZONE	LOCATION	BUILDING	ZONE	LOCATION
RCB	1-7A	ELEVATOR	RAB	1-43	1-A-ACP
RAB	1-10	1-A-1-FD	RAB	1-44	12-A-5-DIH
RAB	1-11	1-A-1-ED	RAB	1-185	1-A-5-COMA
RAB	1-15	5-W-1-C1	RAB	1-46	12-A-6-RT1
RAB	1-23	1-A-4-COR	RAB	1-47	12-A-6-RCC1
RAB	1-33	1-A-5-HVA	RAB	1-48	12-A-6-CR1
RAB	1-34	1-A-5-HVB	RAB	1-49	12-A-6-ARP1
RAB	1-35	1-A-5-SWGR A	RAB	1-50	12-A-6-CR
RAB	1-36	1-A-5-SWGR B	RAB	1-51	12-A-6-1RR
RAB	1-37	1-A-5-BAT A	RAB	1-52	12-A-6-PICR1
RAB	1-38	1-A-5-BAT N	RAB	1-55	12-A-7-HV
RAB	1-39	1-A-BAT N	RAB	1-56	ELEVATOR
RAB	1-41A	1-A-PICA	RAB	1-122	CONTROL BOARD
RAB	1-42A	1-A-5-PIC-B	RAB	1-123	HVAC ROOM

Duct Detector Zones 1-150 through 1-177 are also only ionization detectors.

9. CONTAINMENT INTEGRITY

The main objective of the containment analysis is to identify vulnerabilities that involve early failure of containment functions. This includes consideration of containment integrity, containment isolation, and other containment functions.

The guidance provided in NUREG-1407, Reference 2 states that "generally containment penetrations are seismically rugged; a rigorous fragility analysis is needed only at review levels greater than 0.3g, but a walkdown to evaluate for unusual conditions (e.g., spatial interactions, unique penetration configurations) is recommended." With regard to containment systems, the guidance provided is that "seismic failures of actuation and control systems are more likely to cause isolation system failures and should be included in the examination." The major concern deals with relay chatter which is addressed in Section 7 of this report.

A review of seismic capacities for containment's of similar design to Harris indicates that the containment structure is expected to have a seismic capacity far above the review level earthquake, Reference 21. In addition to the containment structure, NUREG-1407, Reference 2 suggests that certain considerations could require some additional study. Hatches that employ inflated seals is one potential area for concern. The Harris design does not employ this type of seal. Another concern is the post-operation of penetration cooling that is present in some designs. Harris, however, does not employ this design feature. Finally, air-closed valves used for isolation are also listed as a possible concern. Harris does not utilize air-operated valves for containment isolation that require a supply of air to function. Thus, failures in containment isolation would not be expected due to containment system failures.

Containment heat removal is an important aspect in evaluating containment performance. If heat is not adequately removed from the containment the containment pressure may increase to the containment failure pressure. Two mechanisms can lead to energy being transmitted to the containment. The first is

due to the small LOCA. As RCS inventory is lost through the break it carries stored energy which is then release to the containment and pressurization occurs. Feed-and-bleed cooling also results in energy being transferred to the containment. Containment fan coolers can reduce the pressurization due to these mechanisms.

Based on Modular Accident Analysis Program MAAP analyses performed for the individual plant evaluation (IPE) the containment pressure is not expected to increase to the design limit as long as the RHR heat exchangers are available to remove heat. Thus, the fan coolers represent an additional heat removal mechanism but are not required for successful containment cooling as long as the RHR heat exchangers are present. Failure of this heat removal function will result in containment heatup and pressurization.

The pressurization, however, is predicted to occur over many hours and would not result in an early, rapid containment over pressurization. It is concluded that containment fan coolers are not needed to ensure early containment integrity. As a result, the only containment issues to be addressed are the seismic relay review and walkdown. The relay review is addressed in Section 7.

The containment walkdown consisted of looking/evaluating unusual conditions/configurations (e.g., spatial interactions, unique penetrations, piping hard spots, items/components bridging the seismic gap between the containment liner and interior structure, and etc.). The containment walkdown was performed by the SRT (see section 3).

No unusual conditions/configurations where noted except for the platform in the equipment hatch at elevation 286 ft. The platform is supported/welded to the liner at the equipment hatch barrel and is anchored to the floor of the interior structure at elevation 286 ft. Therefore, the platform bridges between the interior and exterior containment structures. This interaction issue is evaluated in calculation HNP-C/PLAT-1023 and is determined not to be detrimental to the containment integrity.

As stated previously, the main objective of the containment analysis is to identify vulnerabilities that involve early failure of containment functions. The SRT reviews and walkdown performed of the containment did not reveal any significant vulnerabilities. Therefore, the HCLPF for the containment is greater than 0.3g, based on SRT reviews, walkdowns and Appendix A of Reference 3.

10. PEER REVIEW

The Harris IPEEE peer review was performed by Mr. Charbel M. Abou-Jaoude and Mr. Steve Reichle of Vectra Technologies, Inc. during December, 1994. Peer reviewer resumes are included in Appendix A.

Peer review comments on the draft IPEEE report, SSEL report and relay evaluation report were submitted to CP&L (Reference 34). The comments were constructive and incorporated into their respective reports to improve the clarity and presentation of the seismic IPEEE program. Excerpts from the peer review summary are included in Sections 10.1 and 10.2 for the walkdown and systems peer review, respectively.

10.1 WALKDOWN PEER REVIEW

The following documents were reviewed by Charbel M. Abou-Jaoude:

- EQE Document No. 52214-P-001, "Project Plan, CP&L Harris IPEEE", dated January 10, 1994.
- EQE Calculation No. 52214-C-001, Revision 0, "CP&L Shearon Harris: Scaling of In-Structure Spectra for Seismic IPEEE", dated August 6, 1993.
- EQE High Confidence of Low Probability of Failure (HCLPF) calculations:

  52214-C-002, Rev. 0
  52214-C-005, Rev. 0
  52214-C-006, Rev. 0
  52214-C-007, Rev. 0
  52214-C-008, Rev. 0

- EQE Document No. 52214-R-001, "Seismic IPEEE, Shearon Harris Nuclear Power Plant", Draft, dated December 5, 1994.

In addition, a plant visit of all accessible areas, excluding Containment, high radiation areas, and the Emergency Service Water, was conducted. Representative

SEWS and data packages were also sampled. A comparison of the reviewer's field walkdown notes and the corresponding SEWS shows that all plant conditions that required further actions had been captured by the SRT. One minor housekeeping concern near the Diesel Control Panels was noted during the Peer Walkdowns: this condition was apparently introduced after the original Plant-Wide Walkdowns by the SRT.

Based on the above, the Peer Reviewer finds that the IPEEE program is being conducted in a very thorough and competent manner. The seismic examinations follow the guidance of EPRI NP-6041 for seismic margin reviews, and meet the objectives stated in NUREG-1407. The Peer Reviewer concurs with the overall results documented in the draft summary and has validated the judgment that has been exercised by the SRT during the walkdowns; also, the selection of equipment that require further reviews or detailed HCLPF calculations was found to be satisfactory given the vintage of the plant and the conservatisms associated with the original seismic designs. In view of some of the conservatism in the estimation of the in-structure demand for a 0.3g RLE, the Peer Reviewer judges that the plant structures, equipment, and systems are very rugged and are capable of withstanding a Review Level Earthquake greater than 0.3g.

The observations that follow primarily deal with areas that the reviewer believes could use some additional clarifications in order to capture some of the SRT's thought process. Also, one technical issue relative to the In-Structure RLE demand is being identified as requiring further discussions in order to better support some of the screening decisions and the limited scope of the HCLPF calculations.

10.2 SSEL AND RELAY PEER REVIEW

The following CP&L documents were reviewed by Mr. Steve Reichle:

- "Success Path Logic Diagram and Supporting Information Development", RSC 94-01, Revision 1, dated August 1994.

- "Relay Study for the Shearon Harris IPEEE", SAROS 93-9, Volume 1, dated December 1993.

- CP&L Drawings (flow diagrams):

2165-2-0542, Rev. 20	2165-S-1330, Rev. 17	2165-S-1308, Rev. 9
2165-2-0544, Rev. 25	2165-S-1301, Rev. 6	2165-S-1309, Rev. 15
2165-2-0545, Rev. 41	2165-S-1303, Rev. 9	2165-S-1310, Rev. 10
2165-2-0547, Rev. 26	2165-S-1305, Rev. 15	2165-S-1324, Rev. 9
2165-2-0550, Rev. 12	2165-S-1307, Rev. 6	2165-S-1344, Rev. 4

The peer review of the safe shutdown equipment selection and relay review work completed for the HNPP was performed against the guidance provided in EPRI NP-6041 and NUREG-1407. The methodology utilized to select and document the safe shutdown paths and equipment selection, as documented in RSC 94-01, fully meets the intent of EPRI NP-6041.

In addition to reviewing the above reports against the referenced guidance documents, a detailed check of the AFW and CVCS SSELs was performed with their respective flow diagrams. This review was made to determine if all applicable components were identified, and whether the correct review types (i.e. seismic and/or relay) were specified.

As a result of these reviews the following observations and comments were made and provided to Ricky Summitt Consulting. The "Response" to each comment was provided by Ricky Summitt Consulting. Based on a follow-up conversation with Messrs Ricky Summitt and Daryl Hughes of CP&L, action items have been identified under the "Recommended Follow-up" section for Items 3, 18 and 21.

The comments made by the Peer Reviewer are mostly questions presented to the preparer of the documents, and do not necessarily indicate that an error or omission had been made. Following the completion of the recommendations made in this attachment, the Peer Reviewer finds that the SSEL and Relay Review work followed the guidance of both EPRI NP-6041 and NUREG-1407, and are complete and acceptable.

11. SUMMARY AND CONCLUSIONS

The Harris seismic IPEEE was completed in accordance with NUREG 1407 guidelines using the EPRI seismic margins methodology provided in EPRI NP-6041.

The most important aspect of the program was plant walkdowns. Detailed SRT walkdowns were performed using the methodology, criteria and SEWS provided in EPRI NP-6041. Several components were identified for subsequent HCLPF evaluation.

The Harris plant is a modern late vintage pressurized water reactor which underwent thorough engineering review with extensive documentation. Nevertheless, the SRT identified 13 issues related to maintenance, housekeeping and seismic interaction that required work orders to satisfy SRT field issues. Six items were noted as requiring repairs or modifications. The SRT also identified design issues that required aggressive evaluation to establish HCLPF capacities that met or exceeded the RLE. The lowest calculated HCLPF capacity is for the RHR heat exchangers. A HCLPF of 0.29g based on conservative RLE spectra compares favorably to the RLE.

A minor potential plant seismic vulnerability surfaced while preparing for the walkdowns. The systems analyst identified five instruments that are powered from sources that may not be available after a seismic event, off-site power and the non-safety uninterruptible power supply (UPS). Alternate instruments or actions have been identified and have been determined to meet the intent of Generic Letter 88-20. The site Abnormal Operating Procedures (AOP's) are being updated to address the alternate instruments or actions that were identified by the system analyst.

A personnel safety issue was noted during the walkdown with regards to the potential for a first aid storage cabinet to fall on a halon bottle located in the stairwell of the RAB 286 ft. This does not affect safe shutdown equipment and

was documented in ACFR number 95-0128. As a result of this ACFR, the halon bottle was relocated to the Turbine Building.

The relay evaluation identified 51 potentially low-ruggedness relays within the Harris Plant. Six of the relays did not pass systems review screens. These relays are components of the 6.9K switchgear. A detailed review of the seismic qualification documentation for the 6.9K switchgear, including the relay qualification was performed. As a result of this review, it was determined that sufficient margin exists in the seismic qualification in which the intent of Generic Letter 88-20 is satisfied.

All issues identified by the SRT were satisfactory resolved. The evaluation concluded that the Harris plant HCLPF capacity meets the 0.3g RLE.

12. REFERENCES FOR SEISMIC ANALYSIS

1. USNRC, Generic Letter 88-20, Supplement No. 4, "Individual Plant Examination of External Events (IPEEE) for severe accident vulnerabilities," Final, April 1991.

2. USNRC, NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," final report, June 1991.

3. EPRI NP-6041, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin", Revision 1, August 1991.

4. USNRC, NUREG/CR-5250, "Seismic Hazard Characterization of 69 Nuclear Power Plant Sites East of the Rocky Mountains," January 1989.

5. USNRC, NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," October 1993.

6. EPRI NP-6395-D, "Probabilistic Seismic Hazard Evaluation at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charlston Issue," April 1989.

7. Seismic Qualification Utility Group (SQUG), "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 2.

8. US Atomic Energy Commission, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Regulatory Guide 1.60, Rev. 1, 1973.

9. EQE Calculation 52214-C-001, "CP&L Shearon Harris: Scaling of In-Structure Spectra for Seismic IPEEE," Revision 0.

10. EQE International, "Project Plan CP&L Harris IPEEE," EQE Document Number 52214-P-001, Revision 0.

11. EQE Calculation 52214-C-002, "CP&L Shearon Harris IPEEE: HCLPF Capacity Calculation for Panel Anchorage With 45° Nelson Studs," Revision 0.

12. EQE Calculation 52214-C-005, "CP&L Shearon Harris IPEEE: HCLPF Capacity Calculation for Low Voltage Switchgears 1A2- & 1A3-SA and 1B2- & 1B3-SB," Revision 0.

13. EQE Calculation 52214-C-006, "CP&L Shearon Harris IPEEE: HCLPF Capacity Calculation for RHR Heat Exchangers 1A-SA & 1B-SB," Revision 0.

14. EQE Calculation 52214-C-007, "CP&L Shearon Harris IPEEE: HCLPF Capacity Calculation for 1X-SAB CST Tank and 1X-SN RWST Tank," Revision 0.

15. EQE Calculation 52214-C-008, "CP&L Shearon Harris IPEEE: HCLPF Capacity Calculation for Emergency Service Water Suction Pumps A & B," Revision 0.

16. NUREG/CR-5088 "Fire Risk Scoping Study,"

17. NRC Information Notice 94-12, "Insights Gained from Resolving Generic Issue 57,"

18. EPRI NP-7148-SL

19. Design Basis Document (DBD) No. 306, "Fire Protection and Detection System."

20. Safety and Reliability Optimization Services, Inc. Report No. 93-9, "Relay Study for the Shearon Harris Individual Plant Examination of External Events," Volumes 1 and 2, Dated December, 1993, Including Addendum I, Dated February, 1995.

21. Summitt, R. L., Assessment of Containment Capacity for the ALWR Based on Prior PRA Assessments, USDOE Advanced Reactor Severe Accident Program, April 1990

22. IE Information Notice No. 85-45, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System used in Westinghouse Designed Plants," Dated June 6, 1985

23. Licensee Event Report (LER) No, 84-020, Docket No. 352

24. Seimens-Allis Qualification Report No. 90365-1

25. Westinghouse Electric Corporation, "Shearon Harris Units 1,2,3 &4 C.R.D.M. Seismic support Tie Rod Assembly," CP&L Drawing No.: 1364-41579, Revision 1.

26. Westinghouse Electric Corporation, "Reactor Integrated Head Package," CP&L Manual ID: MED, Revision 08.

27. CP&L Administrative Procedure AP-003, "General Plant Personnel Safety and Housekeeping"

28. CP&L Plant Programs Procedure PLP-401, "Ladder, Scaffold, and Equipment Use and Storage"

29. CP&L Drawing 1364-53480, Sheets 1 through 7 of 7, Revision 1, "Oil Spill Protection System for RCP Motors."

30. Nuclear Engineering Department Design Guide Number DG-II.17, "Interdiscipline Clearance Verification Walkdown"

31. Nuclear Engineering Department Design Guide Number DG-II.19, "Verification Walkdown in Accordance with Regulatory Guide 1.29"

32. EQE International, "Screening and Evaluation Walkdown Sheets for Shearon Harris Seismic IPEEE," EQE Document Number 52214-R-002, Revision 0.

33. EQE International, "Photographs From Shearon Harris Seismic IPEEE Walkdown," EQE Document Number 52214-R-003, Rev. 0.

34. Letter from Charbel M. Abou-Jaoude of Vectra Technologies, Inc. to Ronald L. Knott of CP&L dated December 30, 1994, "Carolina Power & Light Company Harris Nuclear Plant - IPEEE Peer Review." Letter No.: 0132-00161.000-001.

35. Nigam, Navin C. And Jennings, Paul C., "Digital Calculation of Response Spectrum for Strong-Motion Earthquake Records," National Science Foundation, June, 1968.

36. Newmark N.M., "Earthquake Response of Reactor Structures," Nuclear Engineering and Design, Vol. 20, Pages 303 to 322, 1972.

37. Hetenyi, M., "Beams on Elastic Foundations," University of Michigan Press, Ann Arbor, Michigan, 1967.

38. Summitt, R.C., Report No. RSC 95-01, Revision 1, "Shearon Harris Nuclear Power Plant Success Path Logic Diagram and Supporting Information Development," August, 1994, Including Addendum 1 Dated February, 1995.