August 31, 1982

Docket No. 50-219
LS05-82-08-078

Mr. P. B. Fiedler
Vice President and Director - Oyster Creek
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Fiedler:

SUBJECT: SEP TOPIC III-6, SEISMIC DESIGN CONSIDERATIONS
OYSTER CREEK NUCLEAR GENERATING STATION

The staff has completed the reanalysis of the CRD piping using revised as-built information provided by your consultant MPR Associates, Inc. We have also considered your comments on interpretation of the isolation condenser line results and agree that this line is not representative of safety-related piping at Oyster Creek plant. Both the CRD and isolation condenser lines were sampled and evaluated by the staff to determine the acceptability of chart-methods (lateral-deflection and force evaluation curve methods) by which all safety-related piping in size of 10"ϕ and under were originally analyzed and designed for Oyster Creek plant. We are unable to conclude that the original design methods applied for safety-related piping in this size range provides an adequate seismic design. The bases for our conclusion are described in the revised Safety Evaluation Report (Enclosure 1).

It is the staff's position that:

(1) A sampling approach is to be taken for the evaluation of safety-related piping systems 10"ϕ and under for the Oyster Creek plant; and

(2) If the expanded samples still could not demonstrate adequacy of original design, all safety-related piping 10"ϕ and under should be reanalyzed and upgraded as required.

In addition, information should be provided to demonstrate the design adequacy of piping supports for the two large piping systems. Further, the licensee should provide the requested demonstration of the structural integrity for:

(1) The reactor internals,

SE04 Add: Gary Staley

(2) CRD hydraulic control units,

DSU USE EX (51)

(3) Motor control centers and switch panels; and

(4) Load paths for all electrical equipment.

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P PDR

This evaluation will be a basic input to the Integrated Safety Assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This topic assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the Integrated Assessment is completed.

Sincerely,

Original signed by:

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosures:
As stated

cc w/enclosures:
See next page

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DATE	8/27/82	8/30/82	8/30/82	8/31/82	8/31/82	8/31/82	8/31/82

Mr. P. B. Fiedler

Oyster Creek
Docket No. 50-219
Revised 3/30/82

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SEP SAFETY TOPIC EVALUATION

OYSTER CREEK NUCLEAR POWER PLANT

TOPICS: III-6, SEISMIC DESIGN CONSIDERATION
III-11, COMPONENT INTEGRITY

I. INTRODUCTION

The nuclear power plant facilities under review in the SEP received construction permits between 1956 and 1967. Seismic design procedures evolved significantly during and after this period. The Standard Review Plan (SRP) first issued in 1975, along with the Regulations 10 CFR Part 50, Appendix A and 10 CFR Part 100, Appendix A constitute current licensing criteria for seismic design reviews. As a result, the original seismic design of the SEP facilities vary in degree from the Uniform Building Code up through and approaching current standards. Recognizing this evolution, the staff found that it is necessary to make a reassessment of the seismic safety of these plants.

Under SEP seismic reevaluation, these eleven plants were categorized into two groups based upon the original seismic design and the availability of seismic design documentation. Different approaches were used to review the plant facilities in each group. The approaches were:

Group I: Detailed NRC review of existing seismic design documents with limited reevaluation of the existing facility to confirm judgments on the adequacy of original design with respect to current requirements.

Group II: Licensees were required to reanalyze their facilities and to upgrade, if necessary, the seismic capacity of their facility. The staff will review the licensee's reanalysis methods, scope and results. Limited independent NRC analysis will be performed to confirm the adequacy of the licensee's method and results.

Based upon the staff's assessment of the original design; the Oyster Creek plant was placed in Group I for review.

The Oyster Creek plant, a Mark I boiling water reactor (BWR), is located on the Atlantic coast, about 35 miles north of Atlantic City, New Jersey. General Electric Company, the nuclear steam supply system (NSSS) supplier and prime contractor, engaged Burns and Roe, Incorporated for engineering assistance and construction management. The seismic analysis of plant structures, systems and components was performed by John A. Blume and Associates, Engineers. The plant received its construction permit on December 15, 1964, and provisional operating license on August 1, 1969. The Jersey Central Power and Light Company (JCPLCo), the owner, filed its application for a full-term operating license on March 6, 1972.

The Oyster Creek plant was originally designed for a design level earthquake (equivalent to the OBE) with a peak ground acceleration (PGA) of 0.11g and for a safe shutdown earthquake (SSE) with a PGA of 0.22g. Housner ground response spectra scaled to the specified PGAs were used as seismic input for the analyses and design. The vertical component of ground motion was assumed to be two-third of the horizontal components throughout the frequency range. For the dynamic analyses of structures (reactor building and control room/turbine building), the buildings were modelled as two-dimensional lumped mass-spring systems with only rotational soil springs attached the corresponding lumped masses to account for the soil-structure interaction effects. Response spectrum analysis approach was applied to generate member forces for the structural design. The turbine building was also analyzed by time history approach using 1940 El Centro earthquake record scaled to 0.11g (OBE level) as input. No floor (or instructure) response spectrum was generated for the analyses of systems and components. Two approaches (namely, response spectrum analysis approach and equivalent static analysis approach) were applied for the safety related piping systems using Housner ground response spectra as input. All mechanical and electrical components were analyzed by the static analysis approach. Chapter 4 of NRC NUREG/CR-1981 report, "Seismic Review of the Oyster Creek Nuclear Power Plant as Part of the Systematic Evaluation Program," (Ref. 1), summarizes the details of the original seismic analysis and design.

The SEP seismic review of Oyster Creek facilities addressed only the Safe Shutdown Earthquake, since it represents the most severe event that must be considered in the plant design. The scope of the review included three major areas: the integrity of the reactor coolant pressure boundary; the integrity of fluid and electrical distribution systems related to safe shutdown; and the integrity and functionability of mechanical and electrical equipment and engineered safety features systems (including containment). A detailed review of the facilities was not conducted by the staff; rather our evaluations relied upon sampling representative structures, systems and components. Confirmatory analyses using a conservative seismic input were performed for the sampled structures, systems and components. The results of these analyses served as the principal input for our evaluation of the seismic capacity of the facility.

II. REVIEW CRITERIA

Since the SEP plants were not designed to current codes, standards and NRC requirements, it was necessary to perform "more realistic" or "best estimate" assessments of the seismic capacity of the facility and to consider the conservatisms associated with original analysis methods and design criteria. A set of review criteria and guidelines was developed for the SEP plants. These review criteria and guidelines are described in the following documents:

1. NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," by N. M. Newmark and W. J. Hall, May 1978.

2. "SEP Guidelines for Soil-Structure Interaction Review," by SEP Senior Seismic Review Team, December 8, 1980.

For the cases that are not covered by the criteria stated above, the following SRPs and Regulatory Guides were used for the review:

1. Standard Review Plan, Sections 2.5, 3.7, 3.8, 3.9, and 3.10.

2. Regulatory Guides 1.26, 1.29, 1.60, 1.61, 1.92, 1.100, and 1.122.

III. RELATED TOPICS AND INTERFACES

The related SEP topics to the review of seismic design considerations and component integrity are II-4, II-4.A, II-4.B, II-4.C. These topics relate to specification of seismic hazard at the site, i.e., site specific ground response spectrum for the Oyster Creek site. The seismic input selected for the confirmatory analysis of Oyster Creek facility, namely the Regulatory Guide 1.60 spectrum scaled to 0.22g peak ground acceleration, envelopes the Oyster Creek site specific ground response spectrum as shown in Fig. 1, therefore, the results for these four safety topic evaluations will not affect the review of seismic design considerations and component integrity.

IV. EVALUATION

A. General Approach

The seismic reevaluation of Oyster Creek Nuclear Power Plant was initiated by conducting a detailed review of the plant seismic documentation. The results of this review are summarized in the draft report, "Seismic Review of Oyster Creek Nuclear Power Plant - Phase I Report." Then, the staff and our consultants conducted a site-visit. The purpose of this site-visit were: (1) to observe the as-built plant specific feature relative to the seismic design of the facility, (2) to obtain seismic design information which was not available to the staff in the docket, (3) to discuss, with the licensee, seismic design information that the staff and our consultants had reviewed, and (4) based on the results of this field inspection, experience and judgment, to identify sample structures, systems and components for which the confirmatory analyses (or audit analyses) would be performed. The results of these analyses, then, served as the basis for safety assessment of the plant facility.

When a structure was evaluated, it was judged adequately designed if the results from the structural analysis met one of the following three criteria:

1. The loads generated from confirmatory analysis were less than original loads;

2. The seismic stresses from confirmatory analysis were low compared to the yield stress of steel or the compressive strength of concrete; and

3. The seismic stresses from.confirmatory analysis exceeded the steel yield stress or the concrete compressive strength, but estimated reserved capacity (or ductility) of the structure was such that inelastic deformation without failure would be expected.

If the above criteria were not satisfied, a more comprehensive reanalysis was required to demonstrate its design adequacy.

For piping reevaluation, the results from the audit analysis of each of the sampled piping systems were compared with ASME Code requirements for Class 2 piping systems at appropriate service conditions. This comparison provided the basis for reevaluating the structural adequacy of piping systems.

Because limited documentation exists regarding the original specifications applicable to procurement of equipment, as well as for the qualification of the equipment, the seismic review of equipment was based on expert experience and judgment. Two levels of qualification were performed, structural integrity and functionability. The results of this reevaluation of equipment served as the basis for modifications or reanalysis to be undertaken by the licensee.

B. Confirmatory Analysis

In order to provide independent analytical results for the reevaluation, a relatively complete seismic confirmatory analysis, which started with a definition of seismic input ground motion and ended with responses of the safety related structures and selected systems and components, during the postulated earthquake event, was performed. The analysis procedures and results are briefly discussed on the following sections.

1. Seismic Input

When seismic review of Oyster Creek plant started in mid 1979, the site specific ground response spectra were not available. In order to perform the review on a sampling basis that could be applied with confidence, a more conservative ground motion, namely Regulatory Guide 1.60 horizontal ground response spectrum (R. G. 1.60 spectra) scaled to 0.22g, the original design peak ground acceleration (PGA), was used as the horizontal component of postulated ground motion for analysis. The input motion in the vertical direction was taken as 2/3 of the value in horizontal direction across the entire frequency range.

Recently, the site specific spectra development program was completed, and the spectrum developed for the Oyster Creek site was issued to the licensee on June 17, 1981 (Ref. 2) for any future work that may be required. The basis for the development of site specific spectra was documented in NRC NUREG/CR-1582 Report, "Seismic Hazard Analysis," (Ref. 3). This site specific spectrum is appropriate for assessing the actual safety margins present for any structures, systems and components that have been identified as open items. In Figure 1, a comparison is made for the ground response spectra that were used for the original plant design and for SEP seismic reevaluation (Reg. Guide 1.60 spectrum and the site specific spectrum).

2. <u>Acceptance Criteria and Scope</u>

The specific SEP reevaluation criteria are documented in NUREG/CR-0098 and SEP Guidelines for Soil-Structures Interaction Review. These documents provide guidance for:

a) selection of the earthquake hazard;

b) design seismic loadings;

c) soil-structure interaction;

d) damping and energy absorption;

e) methods of dynamic analysis;

f) review analysis and design procedures; and

g) special topics such as under ground piping, tanks and valuts, equipment qualification, etc.

These criteria are felt to more accurately represent the actual stress level in structures, systems and components during a postulated earthquake event and consider, to certain extent, nonlinear behavior of the systems.

The SEP seismic reevaluation of Oyster Creek facility was a limited review centering on:

. Assessment of the general integrity of the reactor coolant pressure boundary.

. Evaluation of the capability of essential structures, systems and components required to shutdown the reactor safely and to maintain it in a safe shutdown condition (including the capability for removal of residual heat) during and after a postulated seismic event.

A total of two (2) structures, four (4) piping systems, eighteen (18) equipment components (mechanical and electrical) were fully evaluated and several others samples were evaluated on a limited basis in this work. They are:

- Structures - Reactor Building and Control Room/Turbine Building.
- Piping Systems - Main steam, feedwater, isolation condenser, and CRD return lines.
- Equipment - 8 mechanical equipment items and 10 electrical equipment items.
- Others - Ventilation stack and condensate storage tank.

Additional samples will be selected if any open items cannot be resolved by analysis.

3. Analysis of Structures

Analytical procedures and methods conforming with the current state of the art were used. These procedures considered the three-dimensional dynamic response of buildings, soil-structure interaction effects, a wide range of dynamic properties for the soil foundation, structural damping in accordance with calculated stress levels, equipment masses, and so forth.

(a) Analysis of Reactor Building

The reactor building (reactor building structures, portion of office building extension, drywell, shield wall, reactor vessel/ support pedestal, and foundation mat) was modelled as four lumped mass-spring closely coupled systems supported by the foundation mat. Because of the high degree of asymmetry of this building, a fully three dimensional model was developed to present the structure. In order to calculate the soil spring constants to take account of the soil-structure interaction effects, the structural foundation mat was considered as an embeded rigid plate on an elastic half space. The input ground motion, R. G. 1.60 spectrum scaled to 0.22g, was defined at the free field ground surface. As required by SEP soil-structure interaction guidelines, this input ground motion was applied directly at foundation of structures without considering any reduction from the foundation embedment. The response spectrum analysis approach conformed with the SRP requirement in that a combination of modal and directional responses, etc., was used to generate the structural responses. The final analysis results (dynamic moments, shears and axial forces) used for the evaluation of the structure were envelopes of the three sets responses generated by considering three levels of soil shear moduli for the purpose of accounting uncertainty of soil properties. The details of analysis and final results are summarized in Chapter 5 of Oyster Creek NUREG report (Ref. 1).

The time-history analysis approach together with an artificial time history record (acceleration) was used for generating in-structure (or floor) response spectra. Again, smoothed envelopes of the three sets of in-structure response spectra corresponding to three soil conditions were used as input motions for the evaluation of piping systems and equipment. Appendix B to the Oyster Creek NUREG report contains a summary of all the generated in-structure response spectra. The results of evaluation showed that reactor building is capable of withstanding the postulated seismic event.

(b) Analysis of Turbine Building

The same acceptance criteria and analytical approaches used for the reactor building were applied to the turbine building. The details of modelling techniques, analysis procedures and analysis results (dynamic forces used for structural evaluation and in-structure response spectra used for equipment and piping evaluation) are found in Chapter 5 and Appendix B of the Oyster Creek NUREG report. The results of evaluation showed that the turbine building is capable of withstanding the postulated seismic event.

4. Analysis of Piping Systems

As discussed in the Section 2 above, four piping lines were selected and analyzed to verify the adequacy of the original design. The piping selected were portions of the main steam, feedwater, isolation condenser, and CRD return lines. The selections were based on: (1) the expert's judgment and observations during the walkdown of the facility, (2) review of the original analyses and design, and (3) a desire to provide a range of piping sizes. Audit analysis which incorporated current ASME Code and Regulatory Guide Criteria and used the floor response spectra as input motion were performed for each portion of piping system selected. The results from these analyses were compared to ASME Code requirements for Class 2 piping systems at the appropriate service conditions. This comparison provided the bases for assessing the structural adequacy of the piping under the postulated seismic loading condition. Assumptions made for the analysis, methodology employed and detailed preliminary results are found in the INEL report (Ref. 4).

The preliminary results of confirmatory analyses of the sampled piping systems showed that the stresses in main steam (24"Ø) and feedwater (18"Ø) lines were within allowable limits and some portions of control rod drive (CRD) return line and isolation condenser line were found to be overstressed and have relative large deflections under the postulated SSE (Reference 4). During the meeting held on June 16, 1982, at NRC, the licensee's consultant (MPR) pointed out that the design information of CRD return line used by the staff in its confirmatory analysis did not quite represent the "as-built" condition for this system. He also pointed out that the isolation condenser line which is being analyzed and upgraded by the licensee to the SEP acceptance criteria, was not originally classified as safety related piping and, therefore, would not be a representative sample for juding the adequacy of the seismic design. After the incorporation of design information which reflects the as-built condition of CRD return line, a new confirmatory analysis was completed by the staff (Attachment 2). The results of this new analysis again showed that some locations of 1"Ø branch lines were found to be overstressed and have large deflections from the postulated SSE loadings. According to the staff's docket review, all safety related piping systems in size of 10"Ø and under as well as CRD line were designed by chart-methods (lateral-deflection and force evaluation curve approach). The results of new confirmatory analysis of CRD line implies that the design adequacy of piping systems in size of 10"Ø and under has not been demonstrated. 1

5. Analysis of Condensate Storage Tank

The integrity of the condensate storage tank was evaluated for: (1) uplift and overturning, (2) anchor bolt forces, (3) buckling of side wall, (4) hoop stresses due to static and dynamic pressures, and (5) sloshing effect. An equivalent static analysis with 0.22g R. G. 1.60 spectrum as input motion was performed for this tank. The detailed evaluation was described in the Oyster Creek NUREG report. The results of this evaluation showed that the anchor bolts would need to be upgraded. This tank is the only safety related tank at this plant. No additional sample is needed for tank evaluation.

6. Analysis of Ventilation Stacks

The 368 ft ventilation stack was modelled as a two-dimensional lumped mass-spring system with soil springs attached to the foundation to account for the soil-structure interaction effects and was analyzed by time-history approach. The details of modelling techniques, analysis procedures and evaluation results were summarized in Chapter 5 of Oyster Creek NUREG report. The results of this evaluation demonstrated that the stack was adequately designed.

7. Analysis of Selected Mechanical and Electrical Equipment

The evaluation of equipment was done on sampling basis. Safety related components required for safe shutdown, the primary pressure boundary, and engineered safeguard features were categorized as active or passive and as rigid or flexible according to the criteria in R. G. 1.45 and SRP 3.9.3. A representative sample (or samples) from each group was selected and evaluated to determine the seismic design margin or adequacy of each group. In this way, groups of similar components were evaluated without the need for detailed reevaluations of all individual components. The sampled mechanical and electrical equipment items and the basis for this sampling are described in Table 1 below:

TABLE 1. Mechanical and electrical components selected by the review team for seismic evaluation, and the basis for selection.

Item No.	Description	Reason for selection
1.	Emergency service water pump	This item has a long, vertical unsupported intake section that was originally statically analyzed for seismic effects.
2.	Emergency isolation condenser	This item is a horizontally mounted component supported by three saddles that do not appear to be seismically restrained. Concern was expressed about the saddles' ability to carry required seismic loads, particularly in the longitudinal direction.
3.	Containment spray heat exchanger	This item is unique in that the heat exchanger is vertically oriented and supported by four brackets. Concern was expressed about the exchanger's ability to withstand overturning effects.
4.	Recirculation pump support	This item is a vertical component supported by hangers and critical to ensuring reactor coolant system integrity.
5.	Emergency diesel oil storage tank	Anchor bolt system for in-structure flat-bottom tanks that are flexible may be overstressed if tank and fluid contents were assumed rigid in the original analysis.
6.	Motor operated valves	A general concern with respect to motor operated valves, particularly for lines 4 in. or less in diameter, is that the relatively large eccentric mass of the motor will cause excessive stresses in piping attached to valves not externally supported.

TABLE 1. (Continued.)

Item No.	Description	Reason for selection
7.	CRD hydraulic control system including tubing and support system	Item is particularly critical to insuring reactor coolant system integrity.
8.	Reactor vessel supports and internals	Same as Item 7.
9.	Battery racks	The bracing required to develop lateral load capacity may not be sufficient to carry the seismic load.
10.	Instrument racks	The racks consist of channel and angle members that may be overstressed due to seismic loads. Anchorage to floor may not be adequate.
11.	Motor control centers	Typical seismic qualified electrical equipment. Functional design adequacy may not have been demonstrated. In addition, anchorage to floor structure may not be adequate.
12.	Transformers	Same as Item 12.
13.	Switchgear panels	Same as Item 12.
14.	Emergency generator	Adequacy of anchorage was questionable. Functionality is important for safe shutdown.
15.	Control room electrical panels	The control panels appear adequately anchored at the base. However, there appear to be many components cantilevered off the front panel, and the lack of front panel stiffness may permit significant seismic response of the panel, resulting in high acceleration of the attached components.
16.	Battery room distribution panels	Same as Item 15.
17.	Isolation phase ductwork supports	The ductwork support system does not appear to have positive lateral restraint and load carrying capacity.
18.	Electrical cabl raceways	The cable tray support system does not appear to have positive lateral restraint and load carrying capacity.

The licensee was asked to provide seismic qualifications data for each sampled component including design drawings, specifications, and design calculations. After a detailed evaluation of each component was completed, conclusions were drawn as to the overall seismic capacity of the safety-related equipment at the Oyster Creek facility. The description of analytical procedures and evaluations are found in Chapter 6 of the Oyster Creek NUREG report.

As discussed in the NUREG report, a total of 16 open items (structural and/or functional integrity) out of 18 sampled equipment were addressed as a result of the evaluation. Some of these items remain open due to lack of design information. After the review and incorporation of additional information submitted by the licensee (Refs. 5 thru 9 and Attachment I), the results are summarized below:

a) The structural integrity of emergency service water pump, emergency isolation condenser, containment spray heat exchanger, recirculation pump support, emergency diesel oil storage tank, motor operated valves, and reactor vessel supports has been demonstrated.

b) The structural integrity of CRD hydraulic control units and associated tubing supports remains open due to lack of design information. The licensee should demonstrate the stress in the supports elements are within ASME Service Condition D stress limits for supports when considered dead weight,axial-bending interaction effects and the effects of element curvatures. 1

c) The design of reactor core internals shroud appears adequate. However, no detailed design information nor calculations was available to allow an evaluation of the design adequacy of reactor internals.

d) The review of safety-related electrical equipment showed that the structural integrity of sampled transformers, control room electrical panels, batter racks, emergency generator, instrument racks, isolation phase ductwork supports, and battery room distribution panels would be maintained under the postulated SSE loading condition. However, the integrity of anchorage and support system of motor control centers and switchgear panels have still not been demonstrated by the licensee due to lack of design information. In addition, the licensee has not demonstrated that the structural integrity of the load path from internally mounted components to the equipment anchorage for all electrical panels.

e) The functionability of all safety-related electrical equipment as well as the structural integrity of internal components of all safety-related electrical equipment is being evaluated through SEP Onwer Group program. This program is scheduled for the completion by the end of 1982.

f) Qualification of electrical cable trays is being evaluated by testing through SEP Owners Group program. This program is scheduled for completion by June of 1982.

V. CONCLUSION

Based on the review of the original design analyses, the results of confirmatory analyses performed by the staff and its consultants, and the licensee's responses to the SEP seismic related safety issues, the following conclusions can be drawn:

Structures - All safety related structures and structural elements of the Oyster Creek facility are adequately designed to resist the postulated seismic event (Ref. 1).

Piping Systems - According to the results of SEP piping audit analyses (Reference 4 and Attachment 2), some locations of the sampled piping systems (10"Ø and under) were found to be overstressed under the postulated seismic loading. In addition, the design adequacy of pipe supports has not been reviewed because the implementation of NRC IE Bulletin 79-14 was being continued. Furthermore no design information regarding the design adequacy of pipe supports, although requested, was presented to the staff for review. The staff recommends that on a sampling basis (e.g., 10 randomly selected piping systems), safety-related piping in this size range should be analyzed to demonstrate the adequacy of the original design methods and the design adequacy of piping supports. If the sampled piping does not demonstrate the adequacy of original design methods, all safety-related piping (including supports) in this size range designed by chart-methods should be reanalyzed and upgraded as necessary. In addition, the licensee should provide enough information to demonstrate the design adequacy of piping supports for the two large sample piping, namely main steam and feedwater lines. 1

Mechanical Equipment - A total of 8 mechanical equipment items were sampled. From the 8 items, 5 have been determined to be adequate. The remaining open items (structural integrity and/or functionability) are due to lack of design information. This does not necessarily imply that safety deficiencies exist. The reevaluation of design adequacy of the 3 remaining open items is being conducted by the licensee and the final results will be reviewed during integrated assessment.

Electrical Equipment - As discussed above, the structural integrity of 7 out of 10 sampled safety related electrical equipment items was found to be adequate under postulated SSE and the remaining 3 open items are due to lack of design information. Since the anchorage and support system of all safety related electrical equipment was upgraded based on the same criteria, it is the staff's judgment that the structural integrity of motor control centers and switchgear panels will be demonstrated when the evaluation program conducted by the licensee is complete. As far as the functionability of electrical equipment and the design adequacy of cable trays, two (2) activities are being conducted by the licensee: (a) a program has been initiated for the documentation of seismic qualification (functionability of the equipment and structural integrity of internal components) of all safety related electrical equipment, namely the SEP Owners Group program, and (b) a program for seismic qualification of electrical cable trays based upon testing by the SEP Owners has been implemented. These latter two programs are intended to confirm the adequacy of existing designs and equipment.

Recently, NRC has initiated a generic program to develop criteria for the seismic qualifications of equipment in operating plants; Unresolved Safety Issue (USI) A-46. This program is scheduled for the completion in March 1983. Under this program, an explicit set of guidelines (or criteria) that could be used to judge the adequacy of the seismic qualifications (both functional capability and structural integrity) of safety related mechanical and electrical equipment at all operating plants will be developed. Considering that:

(1) All safety related electrical equipment has been properly anchored;

(2) Past experience and testing results (from both nuclear and nonnuclear facilities) indicate in general that electrical equipment will continue to operate under dynamic loading conditions with only limited transient behavior, if the equipment is adequately anchored; and

(3) The SEP Owners Group programs from which a set of general analytical methodologies is being developed for the seismic qualifications of cable trays and for documentation of other safety related electrical equipment (functionability);

it is our judgment that for the interim period until a technical resolution of USI A-46 is reached regarding methods for assessing seismic qualification of equipment in operating plants, the safety related electrical equipment at Oyster Creek plant will function during and after an earthquake up to and including the postulated SSE. If additional requirements are imposed, as a result of USI A-46, regarding functional capability of safety related electrical equipment, the Oyster Creek facility will be required to address these new requirements along with other operating reactors.

Furthermore, since the ground response spectrum (0.17g R. G. 1.60 spectrum) used for Oyster Creek seismic reevaluation and the 0.22g Housner ground response spectrum used for the original design envelopes the Oyster Creek site specific ground response spectrum, additional safety margins in the structures, systems and components do exist for resisting seismic loadings. Thus, the staff concludes that the Oyster Creek plant can continue to operate with reasonable assurance that the operation of the facility will not be inimical to the health and safety of the public until a resolution is reached for the items identified since the structures meet the SEP reevaluation criteria, the majority of electrical and mechanical equipment meet the SEP reevaluation criteria and unquantified margins exist from the original design.

REFERENCES

1. NUREG/CR-1981 Report, "Seismic Review of the Oyster Creek Nuclear Generating Station as Part of the Systematic Evaluation Program," November 1980.

2. NRC Letter, "Site Specific Ground Response Spectra for SEP Plants Located in the Eastern United States," June 17, 1981.

3. NUREG/CR-1582 Report, "Seismic Hazard Analysis," Volumes 2-4, October 1981.

4. EGG-EA-5211 Report, "Summary of the Oyster Creek Unit 1 Piping Calculations Performed for the Systematic Evaluation Program," July 1980.

5. GPU Service Corporation, "Seismic Analysis of Oyster Creek Emergency Service Water (ESW) Pumps," Revision 1 dated 12/8/81, transmitted by GPU Service Corporation letter dated December 10, 1981.

6. GPU Service Corporation letter from J. T. Carroll to D. M. Crutchfield dated September 2, 1981. Subject: Seismic Evaluation Program, Seismic Considerations - Evaluation of Containment Spray Heat Exchanger Bolts.

7. Oyster Creek Nuclear Generating Station letter dated September 2, 1981 from J. T. Carroll to D. M. Crutchfield.

8. GPU Service Corporation, "Effects of Eccentric Mass of Value Operators on Piping Seismic Stresses," dated November 30, 1981 transmitted by letter from Y. Nagai to T. Cheng, NRC dated December 10, 1981.

9. GPU Service Corporation letter from Y. Nagai to T. Cheng dated November 24, 1981 transmitting October 19, 1981 meeting minutes and enclosures.

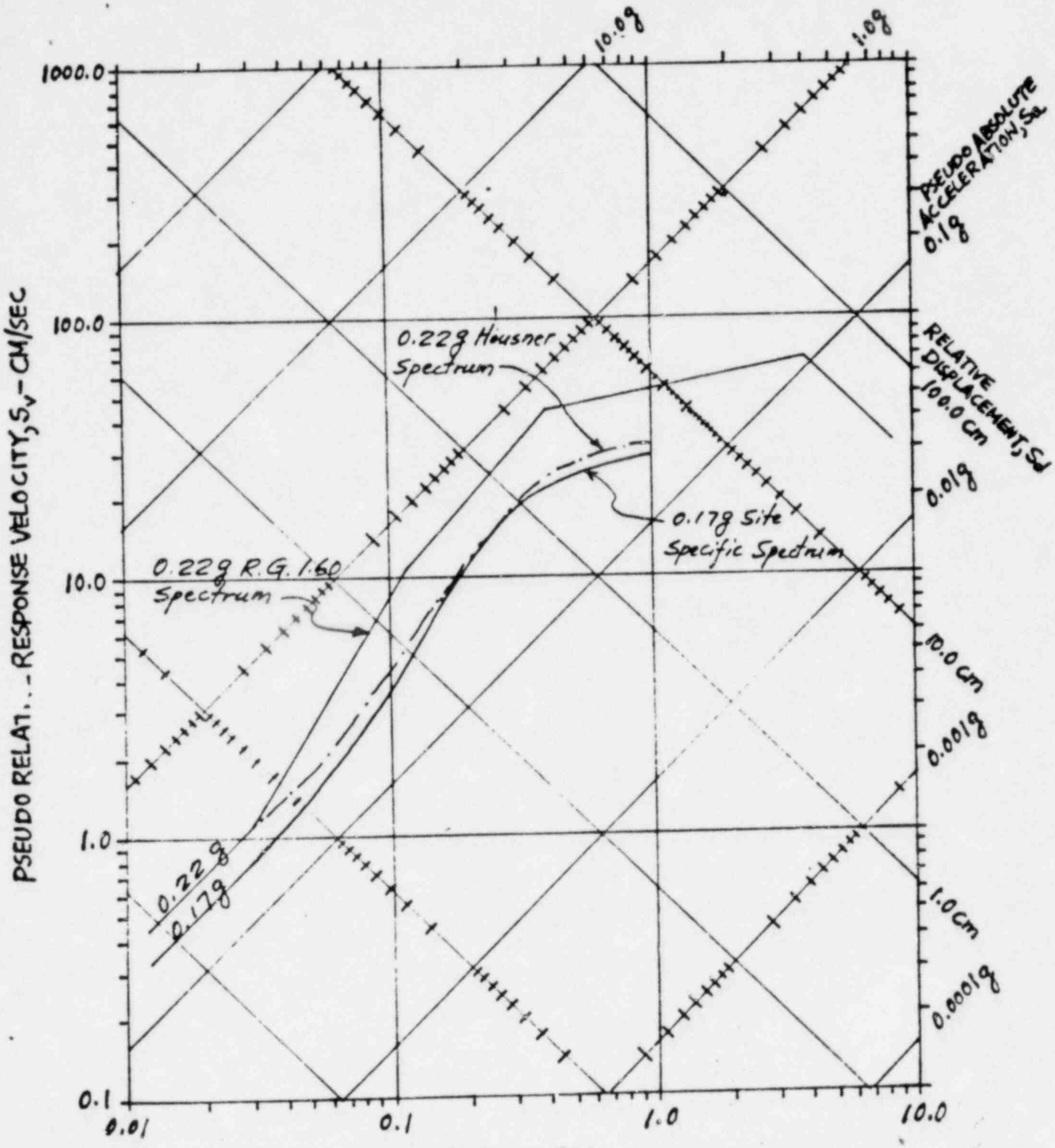
Attachment 1

TABLE 19. Conclusions regarding equipment review for seismic design adequacy of Oyster Creek.

Item	Description	Conclusion and recommendation
1.	Emergency service water pump	O.K.
2.	Emergency isolation condenser	O.K.
3.	Containment spray heat exchanger	O.K.
4.	Recirculation pump support	O.K.
5.	Emergency diesel oil storage tank	O.K.
6.	Motor operated valves	O.K. for structural integrity. Functional adequacy of motor control valves has not been demonstrated.
7.	CRD hydraulic control units	O.K. if detailed evaluation of stress in limiting support elements are within ASME Service Condition D stress limits for supports when considered dead weight, axial-bending interaction effects and the effects of element curvature.
8.	Reactor vessel internals	Design of core internals shroud appears adequate but detailed design calculations are not available to evaluate design adequacy.
9.	Reactor vessel and supports	O.K. for current site specific reduced spectrum if original analysis was adequate.

TABLE 19. continued - Conclusions regarding equipment review for seismic design adequacy of Oyster Creek.

Item	Description	Conclusion and recommendation
10.	Battery racks	O.K.
11.	Instrument racks	O.K. for structural integrity. No information on function.
12.	Motor control centers	Additional analysis demonstrating that MCC will not tip backwards is required. For MCC's identified as "vital" anchorage adequacy has been established.
13.	Transformers	O.K. for structural integrity. Functionality has not been demonstrated.
14.	Switchgear panels	Additional analysis is required. This analysis shall either demonstrate the rigidity of the component or should be based on a maximum acceleration of 1.0g. In addition, shear load calculations must account for the simultaneous occurrence of two mutually orthogonal horizontal components.
15.	Emergency generator	O.K. for structural integrity. Functionality has not been demonstrated.
16.	Control room electrical panels	O.K. for structural integrity. Functionality has not been demonstrated. It should be confirmed that main control room panel has been evaluated.
17.	Battery room distribution panels	O.K. for structural integrity. Functionality has not been demonstrated.
18.	Isolation phase ductwork supports	O.K.
19.	Electrical cable raceways	No evaluation has been made since no drawing or design calcuations are currently ava·lable.



COMPARISON OF GROUND RESPONSE SPECTRA AT OYSTER CREEK SITE

FIGURE 1