Repro 6/30

June 30, 1982

Docket No. 50-237
LS05-82-06-130

Mr. L. DelGeorge
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. DelGeorge:

SUBJECT: SEP SAFETY TOPIC III-6, SEISMIC DESIGN CONSIDERATION AND III-11, COMPONENT INTEGRITY - DRESDEN NUCLEAR POWER STATION UNIT NO. 2

We have completed our seismic review of the Dresden Nuclear Power Station Unit No. 2. Enclosed is a copy of our combined safety evaluation report of the two subject topics.

As discussed in this report, some supports of safety related piping systems were found inadequately designed and required new modifications to resist the postulated seismic loads. In addition, three equipment items, (1) structural integrity of reactor vessel and internal shroud support, (2) structural integrity of recirculation pump and support, and (3) structural integrity and operability of motor operated valves, still remain open due to lack of design information. A supplement of this report will be issued after the review of your responses for these open items are completed.

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. With respect to the potential modifications outlined in the conclusion of this report, a determination of the need to actually implement these changes will be made during the same integrated assessment. 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.

Your response is requested within 30 days of receipt of this letter. If no response is received within that time, we will assume that you have no comments or corrections.

SE04
DSO USD:(6)

Sincerely,

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Paul O'Connor, Project Manager
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Enclosure: As stated

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SEP SAFETY TOPIC EVALUATION

DRESDEN NUCLEAR POWER PLANT UNIT 2

TOPICS: III-6, Seismic Design Consideration
III-11, Component Integrity

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 Builing 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 judgements 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 seismic design; the Dresden 2 plant was placed in Group I for review.

The Dresden 2 plant, a Mark I boiling water reactor (BWR) is located at Goose Lake Township, Grundy County, Illinois, about 60 miles southeast of Chicago, Illinois. General Electric Company supplied the nuclear steam supply systems and Sargent & Lundy Engineers was the architect-engineering and general contractor. The plant received its construction permit on January 10, 1966 and provisional operating license on December 22, 1969. The Commonwealth Edison Company, the owner, filed its application for a full-term operating license on March 16, 1973.

The Dresden 2 plant was originally designed for a design level earthquake (equivalent to the operating basis earthquake, or OBE) with a peak ground acceleration (PGA) of 0.1g and was reviewed to assure that the plant would resist twice the response loads for the 0.1g earthquake without hindering the ability of the plant to be safety shut down. However ground response spectra and N-S component of 1940 El Centro earthquake round scaled to the specific PGAs were used as seismic input motion for the analyses and design. The vertical component of earthquake input was assumed to be two-third of the horizontal components across the entire frequency range. For the dynamic analyses of structures (reactor building, turbine building, and

ventilation stack), the buildings were modeled as two-dimensional lumped mass-spring systems with fixed bases, since the structures were founded on rock and soil-structure interactions was not a consideration. The time-history analysis approach was applied for the analysis of reactor building, turbine building, ventilation stack, and drywall, and generation of in-structure response spectra. The safety related piping systems were analyzed by three different methods: (1) the response spectrum analysis approach with in-structure response spectra as input (recirculation line, main steam line, and feedwater line), (2) lateral-deflection and force-evaluation curve approach (10" in diameter and under), and (3) equivalent static approach. As for the analysis of components, two methods were used: (1) the response spectrum analysis approach with amplified ground response spectra (AGS) as input motion, and (2) the equivalent static method. The dumping ratios recommended by Housner were used for the analysis of structure, systems, and components. Chapter 4 of NRC NUREG/CR-0891 report, "Seismic Review of Dresden Nuclear Power Station - Unit 2 for the Systematic Evaluation Program" (reference 1), summarizes the details of the original analysis and design.

The SEP seismic review of Dresden 2 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 operability 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.

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.

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 Dresden 2 site. The seismic input selected for the confirmatory analysis of

Dresdne 2 facility, namely the Regulatory Guide 1.60 spectrum scaled to 0.2g peak ground acceleration, envelopes the Dresden 2 site specific ground response spectrum as shown in Figure 1; therefore, the results for these four safety topic evaluation will not affect the review of seismic design considerations and component integrity.

EVALUATION

A. General Approach

The seismic reevaluation of Dresden 2 Nuclear Power Plant was initiated by conducting a detailed review of the plant seismic documentation. 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 judgement, 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 analyses 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 judgement. Two levels of qualification were performed, structural integrity and operability. 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 Dresden 2 plant started in early 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 0.2g, 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 generated for the Dresden 2 site was issued to the licensee on June 17, 1981 (reference 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" (reference 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 (R.G. 1.60 spectrum and the site specific spectra).

2. Acceptance Criteria and Scope

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 and design procedures; and

(f) special topics such as under ground piping, tanks and vaults, 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 the Dresden 2 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 three (3) structures, three (3) piping system, seventeen (17) 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, turbine building, and ventilation stack,
- Piping systems - portions of the recirculation loop and LPCI suction line, as well as a small piping seismic example,
- Equipment - 9 mechanical equipment items and 8 electrical equipment items,
- Others - drywell and suppression chamber (ring header, torus, and support system).

3. Analysis of Structures

Analytical procedures and methods conforming with the current state of the art were used. These procedures and methods considered the three-dimensional dynamic response effects (including torsional effects) of buildings, interaction between buildings, equipment masses, structural damping in accordance with calculated stress levels, and so forth.

A. Analysis of Reactor Building - Turbine Building Complex

The original dynamic models of the reactor and turbine building complex were two-dimensional, planar representations that are considered representative only for symmetric structures. Because of the high degree of asymmetry of this building complex, a new coupled, three dimensional model system was developed to represent the structures. In this model system, the building complex (reactor building, turbine building, drywell, etc.) was modeled as three lumped mass-spring system with fixed bases to simulate rock founded foundations. For most parts of the models, the mass and stiffness properties from the original model were used. The detailed discussion of modeling techniques and

the final model used for the confirmatory analysis are found in reference 1.

In order to generate the building responses (dynamic moments, shears, and axial forces) for the structural evaluation, the model was analyzed through the response spectrum analysis method with R.G. 1.60 spectrum scaled to 0.2g as seismic input. The time-history analysis approach together with an artificial time history record (acceleration) scaled to the same PGA, namely 0.2g, was used for generating in-structure (or floor) response spectra. After the peaks were broadened ±15% of corresponding frequency in accordance with R.G. 1.122, the smoothed response spectra were used as input motions for the evaluation of piping systems and equipment. All building responses and in-structure response spectra were summarized in Chapter 5 of reference 1. The results of structural evaluation showed that reactor-turbine building amplex is capable of withstanding the postulated SSE event (Reference 1 and Attachment 2).

B. Analysis of Ventilation Stack

The ventilation stack is a 310 ft high, tapered reinforced concrete structure with an inside diameter of 11 ft at the top and 22 ft - 7 in at base. The wall thickness is 7 in for the upper 100 ft and 18 in for the lower 30 ft portion.

The stack was originally modeled as a two-dimensional fixed base lumped mass spring system. The same model was used for the confirmatory analysis. Using the response spectrum analyses approach with R.G. 1.60 spectrum anchored to 0.2g PGA as input, the results showed that the stack is capable of withstanding the postulated SSE.

4. Analysis of Piping Systems

As discussed in the subsection B.2 above, three piping lines were selected and analyzed to verify the adequacy of the original design and confirm the original analysis results. The pipe lines selected were portions of the recirculation loop piping, LPCI suction line, and small field run piping. The selections were based on: (1) the expert's judgement 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 analyses 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 (reference 4).

The preliminary results of confirmatory analyses showed that the maximum stresses for the selected piping were within allowable limits and no undesirable deflections were identified. These analyses were completed based on the assumption that all piping supports would retain their integrity under the postulated SSE loading. The review of the adequacy of all safety related piping supports was being reviewed under IE Bulletins 79-02 and 79-14 and was not duplicated as a part of the SEP piping seismic audit analyses. As discussed in Chapter 6 of reference 1, the

methods used for the original seismic piping design, especially for the design of piping supports, may not have been conservative. Recently, in response to NRC's "Notice of Violation," the licensee reported that a significant number of modifications to existing supports and addition of new supports would be required to complete their effort in response to IE Bulletin 79-14. The licensee has proposed to correct piping support deficiencies on a priority basis associated with (1) the reactor coolant system pressure boundary piping up to the first normally closed isolation valve or the first isolation valve capable of being closed and (2) piping necessary to assure at least one path for reactor decay heat removal. The licensee has proposed to complete all work associated with the IE Bulletin 79-14 by December 31, 1983. As shown in Figure 1, the FSAR seismic input (Housner ground response spectrum anchored at 0.2g) is more conservative than the site specific spectrum which was developed by NRC for seismic reevaluation of the Dresden site. Based upon the low probability of an earthquake with ground motion which exceeds the NRC's site specific spectrum, the conservatism of the FSAR seismic input and margins which exist in FSAR design criteria for piping systems it is appropriate to resolve the "adequacy" or conservatism of the original design of piping supports as part of the IE Bulletin 79-14 effort.

5. <u>Analysis of Suppression Chamber (Ring Header, Torus and Support System)</u>

Based on the criteria discussed in NRC NUREG/CR-0098, an analysis with R.G. 1.60 spectrum anchored to 0.2g PGA as input motion was performed for the suppression chamber system. The detailed evaluation was described in Reference 1. The preliminary results of this evaluation showed that

moderate yielding could be expected for the support system. As recommended in reference 1, the swaying rods and their connection details were evaluated by the licensee to ensure they provided an overall ductile/load-resistant system. Considering the additional information provided by the licensee and the fact that the ground response spectrum (0.2g R.G. 1.60 spectrum) used for Dresden 2 seismic reevaluation envelopes the Dresden 2 site specific spectrum (Figure 1), the staff concluded that the suppression chamber system is capable to withstand the postulated seismic loads.

6. Analysis of Sampled 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 engineering 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.

The licensee was asked to provide seismic qualification data for each sampled component including design drawings, specifications, and design calculations. After a detailed evaluation of each sampled component was completed, conclusions were drawn as to the overall seismic capacity of the safety related equipment at the Dresden 2 facility. The description of analytical procedures and evaluations are found in Chapter 6 of Reference 1.

As discussed in References 1 and 6, a total of 12 open items (structures) integrity and/or functionability) out of 18 sampled equipment items were identified as a result of the evaluation. Most of these items remained open due to lack of design information. After the review and incorporation of additional information provided by the licensee (references 6 through 14 and Attachment 1), the results are summarized below:

(a) Six (6) mechanical items and two (2) electrical items were found to be adequately designed.

(b) The structural integrity of the following mechanical equipment items still remain open due to lack of design information:

(i) Reactor vessel and internal shroud support,

(ii) Recirculation pump and support.

(c) The structural integrity and functionability of motor operated valves remain open due to lack of design information.

(d) The structural integrity of all safety related electrical equipment has been demonstrated. All required modifications to the equipment support systems are completed except the support of motor control centers which are scheduled for completion by the next refueling outage. However, the operability as well as the structural integrity of internal mounted devices for all electrical equipment was not evaluated.

(e) Qualification of electrical cable trays is being evaluated by testing through SEP Owners Group program. This program is scheduled for completion by the end of 1982.

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 (reactor-turbine building complex, ventilation stack, and suppression chamber) of the Dresden 2 facility are adequately designed to resist the postulated seismic event (reference 1 and attachment 2).

Piping Systems - According to the results of SEP piping audit analyses, all sampled piping systems were found to be capable for resisting the postulated SSE loads if the existing piping supports were assumed to be adequate. Since deficiencies of the supports have been identified by the licensee, the staff is unable to conclude that the safety related piping systems are currently capable of withstanding the postulated SSE loads until the pipe support deficiencies are resolved.

Mechanical Equipment - A total of 9 mechanical equipment items were sampled. Six out of these 9 items were found to be adequately designed. The remaining items are still open due to lack of design information. This does not necessarily imply that safety deficiencies exist. The licensee has agreed to provide documentation of the design adequacy of these open items. Upon the completion of our review the staff will either issue a supplement to this SER or will address these items in the Integrated Plant Safety Assessment Report.

Electrical Equipment - As a result of SEP seismic review, three (3) activities have been or are being completed by the licensee: (a) upgrading of anchorage and support of all safety related electrical equipment required by NRC letters dated January 1, and July 28, 1980 (references 15 and 16) has been completed, and found to be adequately designed, (b) a program has been initiated for assessing the similarity of electrical equipment to facilitate seismic qualification (operability of the equipment and structural integrity of internal components) of all safety related electrical equipment, namely the SEP Owners Group program, and (c) 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 non-nuclear 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 for which a set of general analytical methodologies is being developed for the seismic qualifications of cable trays and for assessing similarity of other safety related electrical equipment to facilitate qualification for operability.

it is our judgement 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 Dresden 2 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 Dresden 2 facility will be required to address these new requirements along with other operating reactors.

Furthermore, since the ground response spectrum (0.13 R.G. 1.60 spectrum) used for Dresden 2 reevaluation and the 0.2g Housner ground response spectrum used for the original seismic design envelope the Dresden 2 site specific ground response spectrum, additional safety margins in the structures, systems and components do exist for resisting SSE seismic loadings.

The staff concludes that the design of the Dresden 2 plant with the exception of the safety related piping systems meets or is equivalent to current licensing criteria. Questions regarding the design conservatism of piping supports will be resolved as part of IE Bulletin 79-14 which is being reviewed independently of SEP.

TABLE 1. Mechanical and Electrical Components Selected by the SSRT for Seismic Evaluation and the Basis for Selection

Item No.	Description	Reason for Selection
1	Control rod drive control units and associated hydraulic tubing	The supports of this item appear to be of relatively low frequency; hence, significant seismic response is possible. Specific provisions for carrying seismically induced loads are not apparent from visual inspection.
2	Shutdown heat exchanger	This item is supported by two 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	Isolation condenser	This horizontal heat exchanger is supported by three saddles, but the concerns are the same as those expressed for Item 2.
4	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 the attached piping, if the valves are not externally supported.
5	Pump-horizontal, high-pressure coolant injection (HPCI)	Pumps supplied to Dresden 2 had no specified, well-defined code-design requirements that included support and anchor-bolt stress requirements.
6	Pump-vertical, low-pressure coolant injection (LPCI)	Same as Item 1. Also, experience has shown that the suction intake and column support legs of vertical pumps tend to be highly stressed by seismic loads.
7	Liquid control 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.

Item No.	Description	Reason for Selection
8	Reactor vessel and core supports	Items are particularly critical to insure reactor coolant system integrity.
9	Recirculation pump	Item is particularly critical to insure reactor coolant system integrity.
10	Piping systems:	
	(a) Recirculation system	Typical of large, dynamically analyzed pipe systems in Dresden 2, and is particularly critical to insure reactor coolant system integrity.
	(b) LPCI system pump suction seismic analysis	Typical of large, statically analyzed pipe systems in Dresden 2.
	(c) Quad Cities typical piping run test Probe #1	Typical of small, standard layout pipe systems in Dresden 2.
11	Electrical equipment - general	
12	Battery rack	The battery rack for the 250- and 125-V batteries use wooden cell braces. The bracing required to develop lateral load capacity did not appear sufficient to carry the seismic load.
13	Motor control centers - 250 V, dc	Typical seismic-qualified electrical equipment.
14	440 V, ac	Same as Item 13 Also, items were supplied by a different vendor.
15	Switch gear	Same as Item 13.
16	Transformer - 440V	Same as Item 13.
17	Control room electrical panels	The control panels appear adequately anchored at the base. However, there appear to be many components cantilevered off of 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.

Item No.	Description	Reason for Selection
18	Electrical cable raceways	The cable tray support systems did not appear to have positive lateral restraint and load carrying capacity.

REFERENCES

1. NRC NUREG/CR-0891 Report, "Seismic Review of Dresden Nuclear Power Station Unit No. 2 for the Systematic Evaluation Program," April 1980.

2. Letter from NRC to CECo dated June 17, 1981.

3. NRC NUREG/CR-01582 Report, "Seismic Hazard Analysis," October 1981.

4. EGG-EA-5065 Report, "Summary of the Dresden Unit No. 2 Piping Calculations Performed for the Systematic Evaluation Program," November 1979.

5. Letter from W. L. Stiede, CECo to J. G. Keppler, IE Region III, NRC, dated April 16, 1982.

6. Letter from D. M. Crutchfield, NRC to D. L. Peoples, CECo, dated January 30, 1981.

7. Letter from R. F. Janecek, CECo to D. M. Crutchfield, NRC, dated April 27, 1981.

8. Letter from R. F. Janecek, CECo to J. G. Keppler, NRC, dated May 27, 1981.

9. Letter from R. F. Janecek, CECo to D. M. Crutchfield, NRC, dated May 28, 1981.

10. Letter from T. J. Rausch, CECo to D. M. Crutchfield, NRC, dated June 4, 1981.

11. Letter from T. J. Rausch, CECo to D. M. Crutchfield, NRC, dated June 8, 1981.

12. Letter from T. J. Rausch, CECo to D. M. Crutchfield, NRC, dated October 14, 1981.

13. Letter from T. J. Rausch, CECo to P. O'Connor, NRC, dated January 8, 1982.

14. Letter from D. G. Eisenhut, NRC to D. L. Peoples, CECo, dated January 1, 1980.

15. Letter from D. M. Crutchfield, NRC to D. L. Peoples, CECo, dated July 28, 1980.

control design of piping systems. For primary stresses, the OBE normally controls design. This phenomenon is applicable to Dresden 2 piping and is discussed in more detail in Sec. 6.3.1.10.

6.3 EVALUATION OF SELECTED COMPONENTS FOR SEISMIC DESIGN ADEQUACY

6.3.1 Mechanical Equipment

6.3.1.1 Control Rod Drive Control Units and Associated Hydraulic Tubing

Seismic qualification tests have been performed on hydraulic control units 761E500 and 729E950.[43] The units tested are for BWR 4 and 5 systems and are not the same as for the BWR 3 system used in Dresden 2. However, it was stated during a visit to General Electric, San Jose, on January 12, 1979, that the construction of the units is similar to the one used on Dresden 2, and that structural response of the units should be similar.

The qualification tests performed on the control units included:

- Uniaxial sine-beat input in the range 0.5 to 1.2 g at predetermined resonant frequencies.
- Uniaxial random input to envelop a reference response spectrum much in excess of the current criteria floor response spectra (Figs. 5-12 through 5-19) that are applicable to Dresden 2.

Based on the evaluation of the seismic qualification tests, the SSRT believes that the hydraulic control units installed in Dresden 2, if properly supported, will withstand the 0.2 g SSE without loss of function. We have reviewed Ref. 70 which indicates that modification to the control rod drive tubing and supports both inside and outside the drywall is required. We have not reviewed the results of the modification in order to evaluate design adequacy.

analysis (0.32 g) times 1.414, to account for the SRSS of two components of horizontal motion, is 2.08. Dividing this value into the original safety factor of 2.81 a new safety margin of 0.35 is determined. The SSRT believes that the isolation condenser will withstand the 0.2 g SSE without loss of function based on the new safety margin determined.

6.3.1.4 Motor-Operated Valves

In response to concerns expressed by the SSRT, the licensee provided a discussion and analysis of stresses induced by attached motor-operated valves in pipe lines 12 in. in diameter and larger.[48]

It has been the experience of the SSRT that, for lines smaller than 4 to 6 in. in diameter, the eccentricity of motor-operated valves may cause additional significant (in excess of 10% of code allowable) piping stresses that should be considered in the computation of total stresses. The applicable 10% stress levels[29] are Class 2, Condition B for active valves and Condition D when only pressure boundary integrity is required (1800 and 3600 psi, respectively, for typical ferritic piping material). This tendency is increased as the lines become smaller.

It was recommended that the licensee evaluate the stresses induced in the supporting pipe from motor-operated valves required to move or change state to perform their safety function. The licensee should show that the stresses are less than 10% of the Service Condition B code allowable stresses in typical pipe lines in the 1-in.- to 4-in.-diameter range. If not, the total stresses at motor-operated valve locations should be calculated to determine that they are within Condition B code allowables.[29] For nonactive valves, Condition D service levels could apply. Alternatively, the SSRT recommends that a requirement to support motor-operated valves external to the pipe on lines 4 in. in diameter and smaller should be developed and implemented on Dresden 2.

Based on the analysis described in Ref. 71 it appears the licensee has evaluated the motor operated values in small lines less than 4 inches in diameter and has concluded they are within Code allowables. The original

FSAR seismic input was used and it was stated this gives more conservative results than would the SEP input. However, no analytical results were transmitted hence we are not able to independently verify the results reported by C.E.

6.3.1.5 Horizontal Pump--High-Pressure Core-Injection

The high-pressure, core-injection (HPCI) pump-and-motor unit is oriented horizontally in the reactor building and supported at El. 476.5 ft. The unit is located at grade in a rock-supported structure. Therefore, the seismic input criteria should be essentially the R.G. 1.60 ground reponse spectrum

- An explanation of why "guided loads" shown on calculation sheets B-11-1 are considered only for horizontal and not dead load.
- An explanation of how the predicted SSE value of 294,000 lb for Dresden 2 and 3 (shown on page 5/17 of the G.E. calculation on Top Guide Shear Capacity Calculation) was determined.

As a result of the evaluation of the shroud support design calculations, the SSRT believes that they are probably adequate for the new seismic loads defined in Chapter 5, and more detailed analysis is necessary to provide the necessary assurance of seismic design adequacy.

With regard to the seismic reevaluation of the reactor shroud support structure the only new information provided by Ref. 72 and 73 is the G.E. Report No. 257HA718 which appears to reduce the shroud overturning moment used in the original calculation from B&W Report No. 11 "Stress Analyses Shroud Support System" to 5,360 K-ft. from 6,000 K-ft. There is still insufficient information provided in B&W Report No. 11, which was reviewed at the initial preparation of the Dresden-2 Report, to evaluate the seismic design adequacy of the reactor vessel shroud support structure.

In Ref. 72 it is stated that a revised seismic analysis, G.E. Report No. 257HS718 titled "Seismic Analysis of Reactor Internals for the Dresden Ii Plant dated 12/24/68" should be used instead of the original J.A. Blume analysis. The G.E. Report No. 257HA718 revises the basic seismic loads on the reactor vessel supports as follows:

For Original 0.10 g ZPGA Earthquake:

	R. V. Base Shear	Overturning Moment at Base
J.A. Blume Report	820.72 Kips	25.2 K-ft.
G.E. Report No. 257HA817	728.0 Kips	12.8 K-ft.

Thus the G.E. reanalysis reduced the seismic overturning moment by almost a factor of 2. In our opinion a reduction by a factor almost 2 in seismic overturning of the RPV is outside the range of results which would normally be expected from the use of more detailed anaysis. We recommend that C.E.

be asked to verify the results of G.E. Report No. 257HA718 or otherwise explain the major difference between the Blume and G.E. analysis results for moment without significant changes in seismic shear.

Given that the reduced overturning moment determined by the G.E. Report No. 257HA718 is correct the design capability would be increased to approximately 1.2 g. However, design calculations still have not been provided in sufficient detail to permit independent verification of seismic design adequacy of the reactor vessel supports.

6.3.1.9 Recirculation Pump

The SSRT recommends that:

- o Seismic input directly applicable to the recirculation pump supports be developed.
- o The recirculation pump supports be reviewed in detail for design adequacy.

6.3.1.10 Piping

The three methods of seismic qualification of Dresden 2 piping are described in Sec. 4.6. Seismic design margins developed by comparing current design criteria to those used in the original Dresden 2 analysis are found in three areas:

Load ratios
Behavior criteria
Analytical method.

The design margins associated with load ratios are described in Sec. 5.6.1 for piping that was evaluated by equivalent static analysis and the lateral deflection and force evaluation curves.

Three pipe lines--main steam, feedwater, and recirculation--were analyzed by modal superposition using floor response spectra methods. For these lines,

However, it is recommended that supports on piping seismically designed by means of the lateral-deflection and force-evaluation curves be re-evaluated to assure that such piping has a fundamental frequency at least twice the building fundamental frequency. Alternatively, piping supports would have to be evaluated for inertial loads equal to the peak of the applicable floor response spectra.

6.3.2 Electrical Equipment

In Ref. 75 it is indicated that extensive modifications to electrical equipment anchors have been made in response to I&E Bulletin 80-21. In general, the conclusion reached in this section assume components are adequately anchored. It is also understood that many of the components will be restrained from both top and bottom of the component. In such cases there will be an approximate triplicity of the frequency of the component hence the frequency concerns expressed in this section for components which are both top and bottom supported are no longer applicable. Finally, there should be an effort expended for components not qualified by proof testing to evaluate the structural adequacy of the attachment of individual electrical devices within each component.

6.3.2.1 Battery Racks

The battery racks used in the Dresden 2 plant were supplied by Gould-National Batteries, Inc., and are described in Ref. 59. No structural design calculations were available for review. The racks are located in the turbine building at El. 549 ft. Response spectra directly applicable to the battery rack locations are not currently available. However, the responsde spectra given for mass point 7 in the reactor building at El. 545.5 ft should be a reasonable approximation for the battery-rack location.

The existing battery racks were analyzed for dynamic characteristics as well as resultant member stresses as shown in Ref. 67. Results of this analysis indicate the loaded racks are rigid in all cases having fundamental frequencies above 25 Hz.

The rack frame and anchor bolt stress resultants are within acceptable limits of the Standard Review Plan, Sec. 3.8 (Ref. 2).

The wooden battens, which laterally support the batteries, would exceed acceptable limits if the coefficient of friction between the batteries and their support rails is less than 0.2.

The SSRT recommends that the wooden battens, which now laterally restrain the batteries, be strengthened or replaced so that friction between the batteries and their support rails need no longer be relied upon to carry inertial loads.

We understand that the wooden battens have been replaced by steel angles designed to carry the full seismic lateral load. On this basis we believe that the battery racks are qualified to withstand a 0.2 g SSE-ZPGA event without loss of function.

TABLE 6-6. SSRT conclusions regarding equipment reviewed for seismic design adequacy of Dresden 2.

Item	Description	Conclusion and recommendation
1.	Control rod drive units and associated hydraulic tubing supports	Design adequacy of the tubing and its support system in 0.2 g SSE seismic event should be demonstrated by analysis.
2.	Shutdown heat exchanger	O.K.
3.	Isolation condenser	O.K.
4.	Motor-operated valves	Generic analysis showing motor-operated valves on lines 4 in. should be performed to show resulting stresses are less than 10% of Condition B allowable stresses (1800 psi). Otherwise, stresses induced by valve eccentricity should be introduced into piping analysis to verify design adequacy or provide a procedure whereby all motor valves 4 in. be externally supported. Licensee has evaluated small motor operated valves and determined stresses are within code allowable however, analysis performed has not been reviewed.
5.	Horizontal pump (HPCI)	O.K.
6.	Vertical pump (LPCI)	O.K.
7.	Liquid control tank	O.K.
8.	Reactor vessel and internal should support	Definitive seismic input to reactor supports is not available. Available calculations are incomplete but indicate that the reactor vessel and shroud supports are capable of carrying approximately 0.6 g. A more detailed evaluation is recommended.

TABLE 6-6. Cont'd.

Item	Description	Conclusion and recommendation
9.	Recirculation pump	A detailed evaluation of the pump and supports is recommended to quantify the safety margin.
10.	Piping:[a] a. Recirculation system b. LPCI System, pump suction c. Typical piping run example #1	O.K. in general. However, verification is recommended to assure seismic support spacings of pipe give fundamental frequencies 2 times building fundamental frequency for pipe support design using the lateral-deflection and force-evaluation curves. Otherwise, design stresses in these supports may not meet current behavior limits.
11.	Electrical equipment - general	The support or anchorage of electrical equipment including control instrument racks, switch gear, transformers, motor control centers, etc., do not appear, in general, to have been engineered. Positive anchorage of such components appears to have been decided in the field without any specified material, design, fabrication, or inspection requirements. Supports or anchorage for electrical components should undergo a general engineering review to assure design adequacy. Electrical equipment supports have been modified in a number of instances with installation of angle battens.
12.	Battery Racks	O.K.
13.	Instrumentation and control room panels	O.K. if existing test results can be said to be applicable to panels and instrumentation actually installed in Dresden 2 and modifications suggested by that report have been made.

[a]Seismic design adequacy of steam and feedwater systems has been determined from review of Ref. 18 and 20.

TABLE 6-6. Cont'd.

Item	Description	Conclusion and recommendation
14.	Motor control centers 250 V, dc	O.K. given support modifications have been made to both top and bottom of cabinet otherwise for El. 517.5 ft and below and up to support El. 545 ft if no fundamental cabinet frequencies below 9.0 Hz, and if existing test results are applicable to control centers in Dresden 2. No control centers are located above El. 538 ft.
15.	Motor control centers 480 V, ac	O.K. given support modifications have been made to both top and bottom of cabinets. Centers have not been qualified at frequencies below 5 to 7 Hz. Additional testing or analysis should be performed to determine that there are no resonance frequencies below 5 Hz. Also, test results supplied should be specified as being applicable to motor control centers in Dresden 2.
16.	Switch gear	Test results verifying design adequacy similar to those developed for items 12, 13, and 14 will be required.
18.	Cable trays (pans)	The trays or pans themselves are O.K., but the supports of the trays or pans, consisting primarily of 0.5-in.-diameter threaded rod, do not appear to have adequate lateral load capacity. Additional analysis or test verification is needed to establish lateral load capacity of the supports seismic design adequacy for the applicable response spectra in Chapter 5.

68. Babcock and Wilcox Company, Stress Analysis Shroud Support System, Report No. 11 (April 1970).
69. General Electric Co. Top Guide to Shroud Shear Capability (26 October 1977).
70. C.E. Ltr. to NRC dtd. 27 May 1981 from R.F. Janecek to J.G. Keppler, Subject: Analysis of CRD Piping.
71. C.E. Ltr. to NRC dtd. 8 January 1972 from T.J. Rausch to P. O'Connor, Subject: Qualification of Remot - Operated Valves on Small Pipe
72. C.E. Ltr. to NRC dtd. 8 June 1981 from T.J. Rausch to D.M. Crutchfield, Subject: Seismic Qualification of Reactor Vessel Support.
73. C.E. Ltr. to NRC dtd. 4 June 1981 from T.J. Rausch to D.M. Crutchfield, Subject: Seismic Qualification of Reactor Vessel Support.
74. C.E. Ltr. to NRC dtd. 27 April 1981 from R.F. Janecek to D.M. Crutchfield, Subject: Review of Selected SEP Open Items.
75. C.E. Ltr. to NRC dtd. 28 May 1981 from R.F. Janecek to D.M. Crutchfield, Subject: Positive Anchorage of Safety Related Electrical Equipment.

Attachment 2

DRESDEN 2 REACTOR TURBINE BUILDING INTERFACE

The Dresden Units 2 and 3 Reactor and Turbine Buildings form an integrated structure which is essentially symmetric about the N-S axis. The original design analyses were based on two dimensional models by J. A. Blume. In order to determine the effects of torsional response for the SEP program, LLNL developed a three dimensional model of the structure based on the mass and stiffness properties used in the J.A.B. design models. Loads from the 3-D LLNL model and R.G. 1.60 response spectra indicated significantly higher interface values at the Reactor Building - Turbine Building operating floor diaphragm connection, particularly due to E-W excitation. However, it was observed that the stiffnesses of a number of shear walls in the Turbine Building lower elevations were not included in the original design models. The LLNL model was then modified to include these additional stiffnesses and the response due to E-W excitation was determined. Interface loads from the E-W excitation were substantially reduced as a result of the added stiffnesses.

It was therefore requested that the licensee verify the additional shear wall stiffness, include any additional stiffnesses in the N-S direction, and verify the adequacy of the structure for both N-S and E-W excitation. Seismic responses of the structure with stiffened Turbine Building for a 0.2g time history artificial earthquake were developed for N-S and E-W excitation by URS/J. A. Blume in References 1 and 2 respectively. Substantially increased diaphragm loads in the Turbine Building are indicated for N-S excitation, however. The adequacy of the structure to withstand these loads was subsequently confirmed by Sargent & Lundy. However, no seismic loads resulting from the E-W excitation were included in S & L's evaluation. In order to ascertain the effect of the simultaneous multi-axial excitation, an evaluation has been conducted using a combination of URS/J. A. Blume loads as the basis.

The Turbine Building operating floor slab is located at El. 561'-6 at the E-W wall along the Reactor/Turbine Building interface. Out-of-plane lateral support for the wall is provided by the Reactor Building floor slabs

at El. 570'-0 and El. 545'-6 together with the N-S Reactor Building walls.

Seismic loads used in this evaluation were those calculated by Blume and reported in References 1 and 2 for the N-S and E-W ground motion components, respectively. These loads were developed from a time history analysis using an artificial earthquake record whose response spectra essentially envelop those in U.S. NRC Regulatory Guide 1.60. The dynamic structural model developed to calculate these loads incorporated revised masses and stiffnesses as discussed in the references noted above. N-S seismic response for the 0.2g earthquake creates an axial load of 31,500k in the Turbine Building diaphragm slab. E-W seismic response for a 0.2g earthquake creates an in-plane shear of 27,300k and an in-plane moment of 801,000k-ft at the Reactor/Turbine Building interface. Combination of earthquake directional components was considered by adding 100% of the loads due to the principal ground motion component to 40% of the loads due to the orthogonal ground motion component.

Two possible modes of failure at the Reactor/Turbine Building interface were considered:

1) Out-of-plane flexural failure of the wall
2) In-plane failure of the Turbine Building diaphragm

The E-W wall at the Reactor/Turbine Building interface must transfer Turbine Building diaphragm axial load and in-plane moment to the Reactor Building floors above and below. In the initial stages of seismic response, these loads are expected to be resisted directly by concentrated reactions in the vicinity of the intersections between the Turbine Building diaphragm and the N-S Reactor Building walls. However, these reactions are expected to lead to localized failures that will tend to degrade this load path. Once this occurs, axial load and moment must be transmitted by the Reactor Building wall in out-of-plane flexure.

The out-of-plane flexural capacity of the wall was calculated to be 328k-ft/ft by Sargent and Lundy in Reference 3 using the provisions of the ACI code. It was assumed that plastic hinges form in the wall along its entire length at the top of the Reactor Building slab at El. 545'-6, above and below the Turbine Building slab at El. 561'-6, and at the bottom of the Reactor Building slab at El. 570'-0. The wall was found able to resist an axial load and moment of 31,600k and 322,000k-ft compared to forces of 31,500k and 321,000k-ft (for 0.2g principal ground motion in the N-S direction and 40% contribution from the E-W direction). The diaphragm shear load is expected to have little effect on the out-of-plane wall capacity. It can be concluded that the wall has sufficient capacity to resist seismic forces for a 0.2g earthquake in the N-S direction with 40% contribution from E-W excitation.

In the evaluation of the diaphragm capacity, structural drawings showing details of the wall-slab connection at El. 561'-6 were not available. Based on calculations from Reference 3, it was inferred that there is a vertical construction joint at the wall-slab interface. The steel reinforcement passing through this construction joint must transfer diaphragm shear through shear-friction as well as axial load and in-plane moment.

The shear-friction strength as given by Equation (11-26) of Reference 4 is a function of reinforcement yield strength. The effect of the concurrent axial load and moment is to cause a tensile stress in most of the reinforcement and thus reduce the diaphragm capacity available to resist shear. The reinforcement stress distribution due to combined axial load and moment was calculated for the entire length of construction joint assuming linear material stress-strain behavior and that the concrete on each side of the joint can resist compressive stresses only. The average reinforcement stress was then included in the shear-friction strength as follows:

V_u = shear-friction strength

$= \Phi A_{vf} (f_y - \bar{f})\mu$

Φ = ACI 318-77 capacity reduction factor = 0.85

A_{vf} = shear-friction reinforcement

f_y = reinforcement yield stress

$\bar{f}$ = average reinforcement stress due to axial load and moment

μ = coefficient of friction

= 1.0 for concrete placed against hardened concrete

The shear-friction strength was calcuated to be 26,500k which is marginally less than the shear load of 27,300k calculated by Blume. If the reduction for capacity due to the Φ factor were neglected, the diaphragm would be considered acceptable.

If the peak ground acceleration is reduced from 0.2g to 0.137g as indicated by the LLNL site specific spectra for Dresden, both the Reactor Building wall and Turbine Building operating floor are adequate, including the Φ factor capacity reduction. This implies the same shape of response spectra as is developed by the time history used for Blume's analysis. This introduces some additional conservatism since the Blume time history essentially envelops the R.G. 1.60 spectra.

For the case with 100% of the loads resulting from E-W excitation combined with 40% of the loads from N-S excitation, the interface capacity is adequate based on the 0.2g Blume loads.

Based on the results of this evaluation, it is concluded that the connection is marginally overstressed based on code allowables for the 0.2g earthquake. It is expected that some allowance for ductility will show the connection acceptable for the 0.2g earthquake, although the degree of reduction in response is difficult to assess for this mode of failure. The connection has adequate capacity to withstand the 0.137g site specific response spectrum.

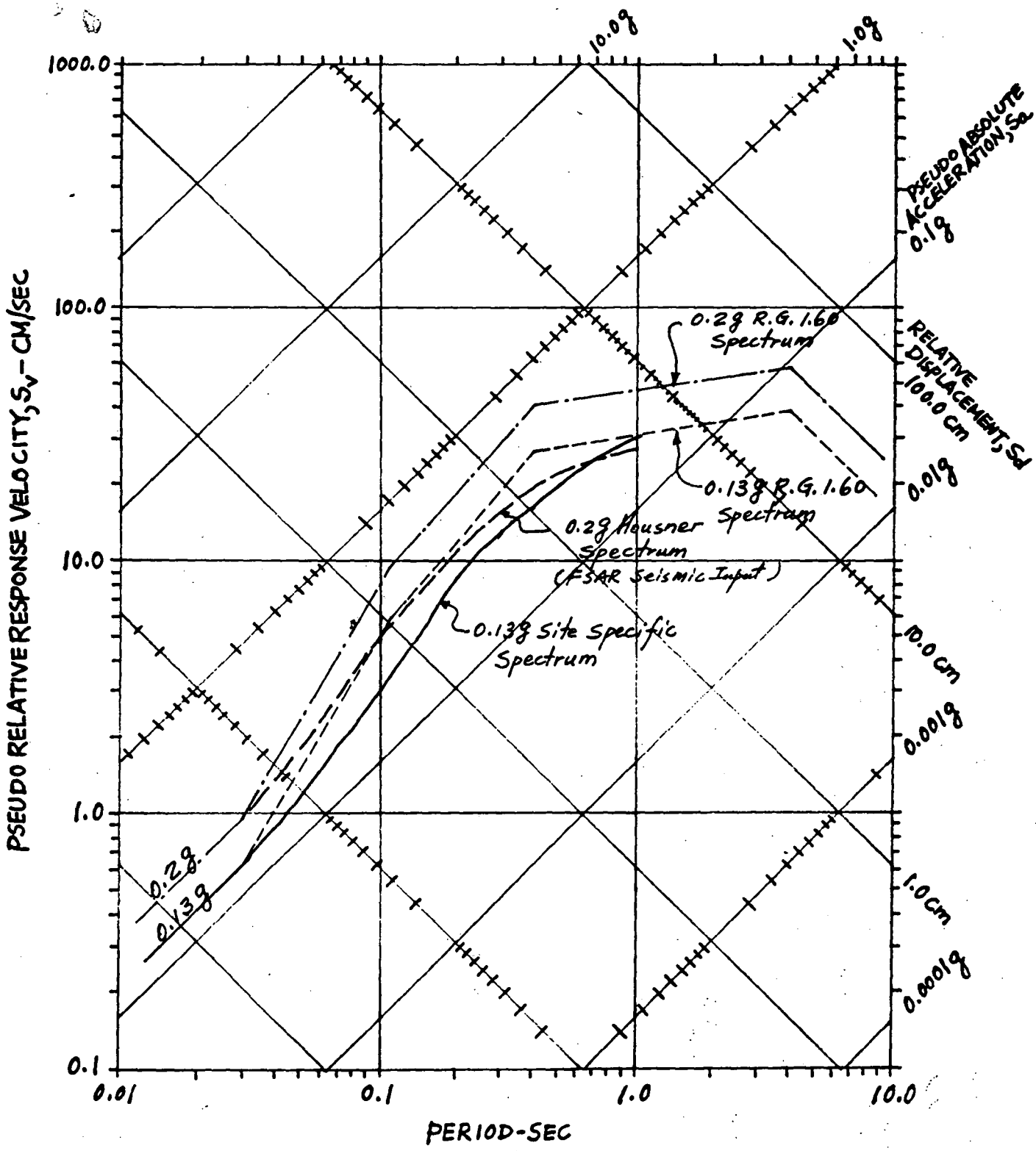
REFERENCES

1) Letter correspondence from T. J. Rausch to P. O'Conner, "Subject: SEP Topic III-6, Seismic Design Considerations Reactor-Turbine Building Complex", September 28, 1981.

2) Letter correspondence from R. F. Janecek to P. O'Conner, "Subject: Dresden 2 SEP Topic III-6, Seismic Design Consideration", March, 30, 1981.

3) Calculations for the Reactor Building-Turbine Building Connection at 561'-6, Sargent and Lundy, July, 1979.

4) "Building Code Requirements for Reinforced Concrete (ACI 318-77)", American Concrete Institute, Detroit, Michigan, 1977.



COMPARISON OF GROUND RESPONSE SPECTRA AT DRESDEN 2 SITE

FIGURE 1