UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 29, 1982

Docket No. 50-244
LS05-82-01-070

Mr. John E. Maier
Vice President
Electric and Steam Production
Rochester Gas & Electric Corp.
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: SEP SAFETY TOPICS III-6, SEISMIC DESIGN CONSIDERATION AND III-11, COMPONENT INTEGRITY - GINNA NUCLEAR POWER PLANT

We have completed our seismic review of Ginna Nuclear Power Plant. Enclosed is a copy of our draft combined evaluation report of the two subject topics.

As discussed in this draft report, four items are required to be upgraded to meet SEP requirements for the postulated SSE: (1) steel bracing at northeast corner of auxiliary building, (2) the support system of component cooling surge tank, (3) refueling water storage tank; and (4) essential service water pumps. Six items still remain open due to lack of design information. According to mutual agreement between the staff and your representative, the responses to these items are scheduled by January 31, 1982. A supplement to this report will be issued after the review of your responses for the six 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.

8202090358
PDR ADOCK

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.

Sincerely,

Dennis M. Crutchfield

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

Enclosure:
As stated

cc w/enclosure:
See next page

Mr. John E. Maier

cc
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SEP SAFETY TOPIC EVALUATION

GINNA NUCLEAR POWER PLANT

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 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 seismic design; the Ginna plant was placed in Group I for review.

The Ginna plant, a pressurized light-water moderated and cooled nuclear reactor, is located on the south shore of Lake Ontario, about 16 miles east of Rochester, New York. Westinghouse Electric Corporation was the prime contractor for the plant. The Westinghouse engaged Gilbert Associates, Inc. as the architect-engineer responsible for the plant design and construction specifications. Bechtel Power Corporation was the general contractor for construction. The plant received its Construction Permit on April 25, 1966 and Provisional Operating License on September 19, 1969. Rochester Gas and Electric Corporation (RG&E), the owner, filed its application for a Full-term Operating License on August 9, 1972.

The Ginna plant was originally designed for an operating basis earthquake (OBE) with a peak ground acceleration (PGA) of 0.08g and reviewed for a safe shutdown earthquake (SSE) with a PGA of 0.2g. Housner ground response spectra scaled to the specified PGA's were used as seismic input for the analyses and design. The vertical component of ground motion was assumed to be the same as the horizontal components. For the anlayses of most seismic Class I structures (con-

tainment shell, containment internal structures, auxiliary building, and diesel generator building); the buildings were modelled as lumped mass-spring systems with fixed bases for calculating the fundamental frequency of each building; then, the corresponding spectral accelerations were used for performing the equivalent static analysis and seismic design. For the control building and intermediate building, only the seismic resisting mechanisms (shear walls and steel bracings) were checked to determine if they were capable of resisting the equivalent seismic loads. The same approach used for Class I structures was applied for the analysis and design of the seismic Class I piping systems and equipment with the Housner ground response spectra used as input. The damping ratios recommended by Housner were used for structural and system analyses. Chapter 3 of NRC NUREG/CR-1821 report, "Seismic Review of the Robert E. Ginna Nuclear Power Plant as Part of the Systematic Evaluation Program" (ref. 1) summarizes the details of the original analysis and design.

The SEP seismic review of Ginna 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.

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.T22.

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, and II-4.C. These topics relate to specification of seismic hazard at the site, i.e. site specific ground response spectrum for the Ginna site. The seismic input selected for the confirmatory analysis of Ginna facility, namely the Regulatory Guide 1.60 spectrum scaled to 0.2g peak ground acceleration, envelopes the Ginna site specific gound response

as shown in Fig. 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 Ginna 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 Ginna Nuclear Power Plant - Phase I Report". Then, the staff and our consultants conducted a site-visit. The purposes of this site visit were: (1) to observe the as-built plant specific features relative to the seismic design of the faciltiy, (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 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; or

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 one of 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 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 Ginna plant started in mid 1979, the site specific gound response spectra were not avilable. 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 Ginna 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 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 analysis;

f) review analysis and design procedures; and

g) 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 Ginna facility was a limited review centering on:

- o Assessment of the general integrity of the reactor coolant pressure boundary.
- o 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, two (2) piping systems, seventeen (17) equipment components (mechanical and electrical) were fully evlauated. They were:

- o Structures - Containment building (containment shell and internal structures) and the interconnected auxiliary, turbine, intermediate, control, service, and diesel generator building complex.
- o Piping Systems - Portions of residual heat removal line and safety injection line.
- o Equipment - 12 mechanical items and 5 electrical items.

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 and methods considered the three-dimension dynamic response effects of buildings, interaction between buildings, equipment masses , structural damping in accordance with calculated stress levels, and so forth.

A. ANALYSIS OF CONTAINMENT BUILDING

The containment building is a vertical, cylindrical concrete structure with a flat base mat and a hemispherical dome. The building is 99 ft. high (from base mat to spring line) and has a 105 ft. inside diameter. The concrete wall, which is prestressed vertically and reinforced horizontally, is 3.5 ft. thick. The thickness of the reinforced concrete dome and base mat are 2.5 ft. and 2 ft. respectively. Housed by containment shell, the internal reinforced concrete structures are supported by the same base mat which is founded on bedrock by means of post-tensioned rock anchors.

A hybrid computer model* was used for the containment building (containment shell, internal structures, and base mat). The containment shell was modelled as a fixed-base lumped mass-spring system and the internal structures were modelled as a fixed-base three-dimensional finite element model. These two models are coupled through the crane structure and the NSSS. Because the building is founded on rock, soil-structure interaction effect corrections are not required. The detailed discussion of modelling techniques and the final dynamic model used for the confirmatory analysis are found in NRC NUREG/CR-1821 report.

*The model was originally developed by the licensee and their consultant (Gilbert Assoc., Inc.) and reviewed by the staff.

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 in-structure response spectra were summarized in Chapter 4 of NUREG/CR-1821 report. The results of structural evaluation showed that containment building is capable of withstanding the postulated SSE event.

8. ANALYSIS OF INTERCONNECTED AUXILIARY, INTERMEDIATE, TURBINE, CONTROL, SERVICE, AND DIESEL GENERATOR BUILDING COMPLEX

As shown in the plot plan (Fig. 2 of NUREG/CR-1821 report), the auxiliary, intermediate, control, and diesel generator buildings were classified as Class I structures and the turbine service buildings Class III structures. Most of these buildings are steel frame structures with reinforced concrete basements that are structurally connected together. Since the staff and its consultants believed that the coupling between all these buildings would effect the dynamic response of structures, systems and components, the buildings were modeled as a U-shape three dimensional space frame model with a fixed base to simulate the rock foundation. The same approaches, applied for the containment building analysis, were used here for

generating the building responses (dynamic moments, shears, member forces, etc.) and in-structure response spectra that were used as input for the evaluation of the piping systems and equipment. The details of modelling techniques, analysis procedures and analysis results are found in Chapter 4 of Ginna NUREG report. The results of evaluation showed that the buildings have sufficient capacity to withstand the postulated SSE event. However, four sets of steel bracing (bracing at northeast corner of auxiliary building and bracings in the south, north, and west walls of turbine building) were found to exceed the allowable stress level for the postulated SSE. The licensee provided additional information for review on October 28, 1981 and November 13, 1981 (Ref. 4 & 5). This open item is expected to be resolved by January 31, 1982 and will be addressed in a supplement to this Safety Evaluation Report.

4. ANALYSIS OF PIPING SYSTEMS

As a result of SEP preliminary seismic review of Ginna plant, NRC IE Bulletin 79-14, and other NRC Seismic requirements, the licensee initiated a seismic upgrade program after the completion of piping support modifications required by IE Bulletin 79-14. In order to conservatively respond to the SEP seismic review and possible future NRC seismic requirements, a set of analysis procedures and criteria that conform with current NRC review criteria (namely, R.G. 1.60 Spectrum, R.G. 1.61 damping, SRP criteria, etc.) were used for the piping analysis. To date, the analysis of all safety related piping systems

inside containment has been completed. The overall upgrade program is scheduled for the completion by 1984 refueling outage.

As discussed in the section B.2 of this report, two pipe lines from those piping systems completed to date were selected and analyzed independently to verify the adequacy of the as-built design and confirm the upgrade analysis results. The pipe lines selected were portions of residual heat removal (RHR) and safety injection (SI). system piping. 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 requirments 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 analysis results are found in the INEL report (Ref. 9). The results from the confirmatory analysis showed that the sampled piping systems are capable of withstanding the postulated SSE seismic input.

5. ANALYSES OF SELECTED MECHANICAL AND ELECTRICAL EQUIPMENT

The evaluation of equipment was done on a 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 re-evaluations of all individual components.

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 Ginna facility. The description of selected components, analytical procedures and evaluations are found in Chapter 5 of the Ginna NUREG report.

As discussed in the NUREG report, a total of 13 open items (structural and/or functional integrity) out of 18 sampled equipment were addressed as a result of the evaluation. Some of these 13 items remain open due to lack of design information. After the review and incorporation of additional information submitted by the licensee (Ref. 10-15), the results are summarized below:

- 3 Mechanical equipment items and one electrical item were found to be adequately designed.
- The component cooling surge tank support system was found to require upgrading. The staff accepted licensee's design criteria and analysis results.
- Refueling Water Storage Tank (RWST) was found to require upgrading. This item will be resolved as part of the integrated assessment.
- Reactor Coolant Pumps were left open (structural integrity) due to lack of design information. The licensee agreed to provide additional information by January 29, 1982.

- The licensee's structural integrity evaluation of motor operated valves (both valves and piping) larger than 2" under their seismic upgrade program was considered to be adequate. The licensee included the reanalyses of small pipe line (2" in diameter and smaller), to which motor operated valves are attached, in the ongoing seismic upgrade program. A separate evaluation will be performed to determine the effect of valve eccentricity on the pipe stresses when the analysis results become available. The licensee has demonstrated that the functional integrity of motor operated valves will be maintained under the postulated SSE.

- The existing essential service water pumps were determined to be not qualified (structural, and functional integrity) due to the lack of support near the suction of the pumps, resulting in over stress in the pump casing support. These pumps are unique to the service water system.

- The modified anchorage and support systems for safety related electrical equipment as well as the evaluations and modifications of internally mounted elements of safety related electrical equipment are found to be adequate.

- Motor Control Centers and Switchgears - The structural design adequacy of the load path between an internally mounted component or device through the panel frame and bracing to the anchorage system was not evaluated due to lack of design information. The licensee agreed to provide this information by January 27, 1982. This item is expected to be closed out by January 31, 1982.

- Control Room Panels - In order to demonstrate the structural integrity (load path from a internally mounted element to anchorage and support system) of panels, the licensee agreed to conduct a low impedance test for a sample panel to determine the dynamic characteristics of the panels and to perform seismic analysis to demonstrate the design adequacy in the near future.

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

- 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.

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:

Structure - All safety related structures and structural elements of the Ginna facility are adequately designed to resist the postulated seismic event. However, four (4) sets of steel bracing system were found to exceed the allowable stress level for the postulated SSE. The licensee provided additional analysis information for review on October 28, 1981 and November 13, 1981. This open item is expected to be resolved by January 31, 1982.

Piping Systems - According to the results of SEP piping audit analysis performed for the sampled piping systems (Ref. 9), the piping systems have been found to be capable of withstanding the postulated SSE.

Mechanical Equipment - A total of 12 mechanical equipment items were sampled. From the 12 items, 7 have been determined to be adequate and two were determined to be inadequate. Generally, the remaining open items are due to lack of design information. This does not necessarily imply that safety deficiencies exist. Rather, it is the staff's judgement that documentation of the adequacy of these open items can be accomplished by February 28, 1982 and will be addressed in a supplement to this evaluation (Attachment 1). However, our evaluation on three (3) sampled safety related tanks (namely, component cooling surge tank, boric

acid storage tank, and refueling water storage tank) showed that the suprot of component cooling surge tank needs to be upgraded and the refueling water storage tank requires both with regard to support and structural integrity. Since two of the sampled tanks were found to require upgrading, the seismic review of safety related tanks should be performed by the licensee to demonstrate the design adequacy of the remaining safety related tanks (volume control tank and NaOH spray additive tank).

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 of 1980 (Refs. 16 & 17) has been completed, and found to be adequately designed (Attachment 1), (b) a program has been initiated for the documentation of seismic qualification (functionality 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 plant; 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

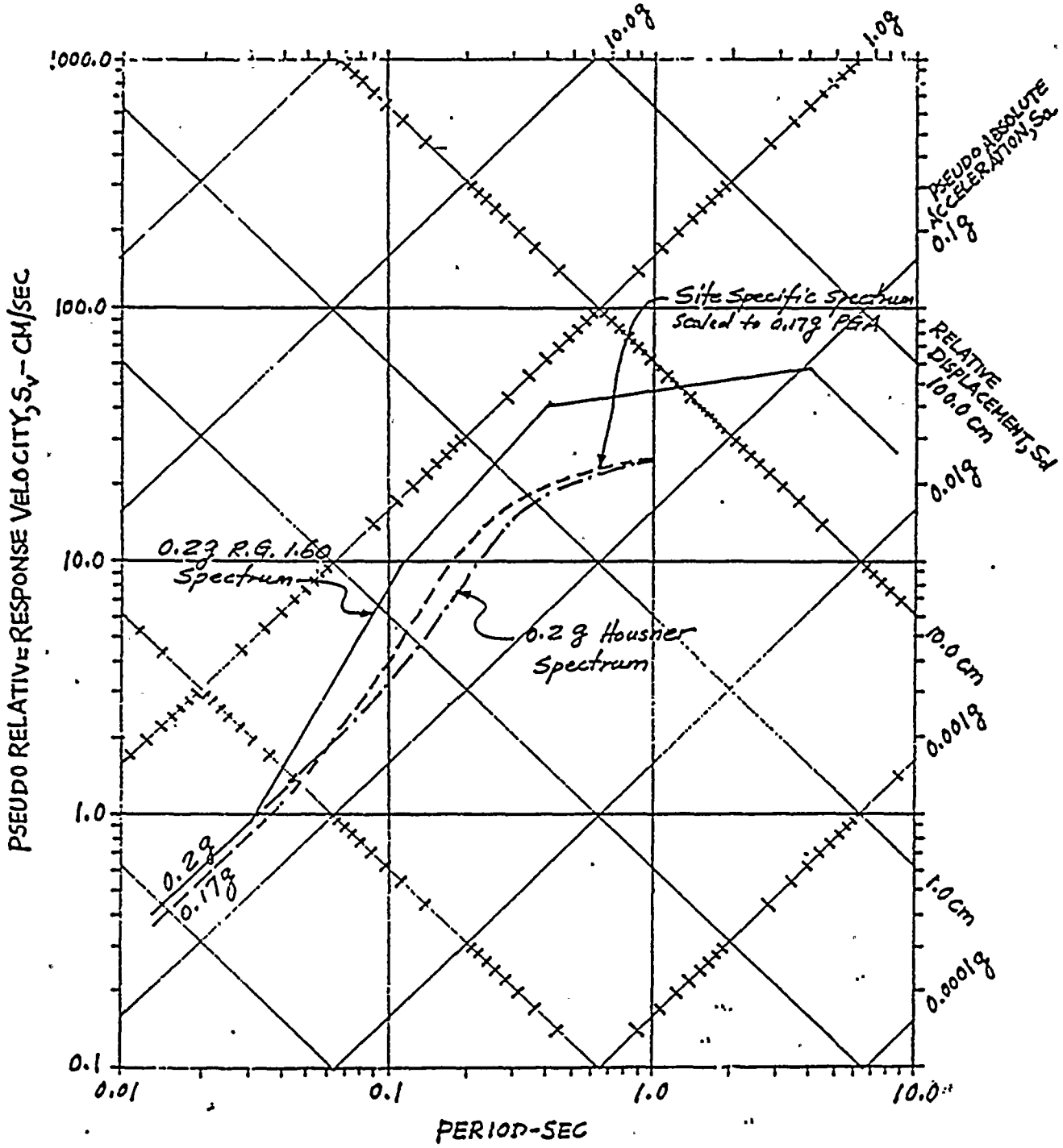
(3) the SEP Owners Group programs from which a set of general analytical methodologies is being developed for the seismic qualifications of cable trays and for documentation of other safety related electrical equipment (functionability);

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

Furthermore, since the ground response spectrum (0.2g R. G. 1.60 spectrum) used for Ginna seismic reevaluation envelopes the Ginna 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 Ginna plant has an adequate seismic capacity to resist a postulated SSE, and therefore, there is reasonable assurance that the operation of the facility will not be inimical to health and safety of the public.

REFERENCES

1. NRC NUREG/CR-1821 Report, "Seismic Review of the Robert E. Ginna Nuclear Power Plant a Part of the Systematic Evaluation Program", December 1980.
2. Letter from NRC to RG&E dated June 17, 1981.
3. NRC NUREG/CR-1582 Report, "Seismic Hazard Analysis", Vol. 4, October 1981.
4. Letter from RG&E to NRC dated October 28, 1981.
5. Letter from RG&E to NRC dated November 13, 1981.
6. Letter from RG&E to NRC dated February 27, 1981.
7. Letter from NRC to RG&E dated February 20, 1981.
8. Letter from RG&E to NRC dated April 1, 1981.
9. EGG-EA-5513 Report, "Summary of the R. E. Ginna Piping Calculations Performed for the Systematic Evaluation Program", July 1981.
10. Letter from NRC to RG&E dated January 7, 1981.
11. Letter from RG&E to NRC dated February 6, 1981.
12. Letter from RG&E to NRC dated May 26, 1981.
13. Letter from RG&E to NRC dated September 24, 1981.
14. Summary of September 9, 1981 meeting held at Rochester, New York dated December 14, 1981.
15. Summary of Integrated Assessment held on December 1, 1981 at Bethesda, Maryland (to be issued in the near future).
16. Letter from NRC to RG&E dated January 1, 1980.
17. Letter from NRC to RG&E dated July 28, 1980.



COMPARISON OF GROUND RESPONSE SPECTRA AT GINNA SITE

FIGURE 1

Attachment 1

Lawrence Livermore National Laboratory

October 30, 1981
SM 81-290

Mr. William T. Russell, Branch Chief
Systematic Evaluation Program Branch
Division of Licensing
Office of Nuclear Reactor Reg.
Washington, D.C. 20555

Dear Bill:

I have enclosed a progress report on open items for Ginna. Of the portion assigned to SMA, Newport Beach, the CRD support has been completed and the one remaining item is the primary coolant pump casing. The report on this item will be forwarded when it becomes available.

Sincerely,

Thomas A. Nelson

Thomas A. Nelson
Project Manager
Structural Mechanics Group
Nuclear Test Engineering Division

TAN/mg
0184m

Enclosure

STRUCTURAL
MECHANICS
ASSOCIATES
A Calif. Corp.

5160 Birch Street, Newport Beach, Calif. 92660 (714) 833-7552

SMA 12205.20

October 15, 1981

Mr. Thomas A. Nelson (L-90)
Nuclear Test Engineering Division
Lawrence Livermore National Laboratory
P.O. Box 808
Livermore, California 94550

Subject: Resolution of Open Items on Ginna Equipment

References: 1) SMA Letter, R. Campbell (SMA) to T. A. Nelson (LLNL) 27 August, 1981.
2) RGE CRDM Seismic Analysis, Westinghouse PWR Systems Division, May 22, 1979.

Dear Tom:

The reference letter summarized the status of open items for Ginna NSSS equipment. At that time we were waiting further documentation on the control rod drives, control rod drive support structure and for the primary coolant pump casing at nozzle penetrations.

We received a Westinghouse analysis of the CRD housing, Ref. 2, for a 0.8g static coefficient and verified that the loadings used in the CRD housing seismic support analysis are correct and that stresses are within acceptable levels. Therefore, the only remaining item is the primary coolant pump casing at nozzle penetrations. In our last conversation with Westinghouse, they were comparing the Ginna pump casing and nozzle loading to the San Onofre units for which a detailed finite element stress analysis has been conducted. That comparison will be reviewed when received.

Very truly yours,

STRUCTURAL MECHANICS ASSOCIATES, INC.

R D Campbell

Robert D. Campbell
Project Manager

RDC:mw

Lawrence Livermore National Laboratory

September 30, 1981
SM81-259/0121b
FIN A0415
Docket No. 50-244

RECEIVED
OCT 6 1981
U.S. NUCLEAR REGULATORY COMMISSION

Mr. William T. Russell, Branch Chief
Systematic Evaluation Program Branch
Division of Licensing
Office of Nuclear Reactor Reg.
Washington D.C. 20555

Dear Bill:

I have enclosed a copy of a report addressing resolution of open items for the Ginna plant as a result of the September 9, 1981 meeting at the RG & E offices.

Sincerely,

Thomas A. Nelson

Thomas A. Nelson
Structural Mechanics Group
Nuclear Test Engineering Division

TAN/tlm
enclosure

X003
S1/1

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CF

STRUCTURAL MECHANICS ASSOCIATES

3645 Warrensville Center Road Cleveland, Ohio 44122 (216) 991-8842

16 September 1981

Mr. T. A. Nelson
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Nuclear Test Engineering Division
Lawrence Livermore Laboratory
P. O. Box 808
Livermore, California 94550

Dear Tom:

Attached hereto please find my comments regarding the meeting held at RG&E offices on 9/9/81 to address the unresolved or open items in the mechanical electrical equipment SEP concerning seismic integrity. We have also updated Chapter 5 of NUREG/CR-1821 to reflect the resolution of items discussed at the meeting where appropriate.

Please advise if you require any clarification.

Sincerely,

for

John D. Stevenson
Vice President and General Manager

JDS:clj

Enclosure

c.c. Dr. Tom Cheng

RESULTS OF MEETING BETWEEN RG&E
THEIR CONSULTANTS AND THE NRC-SEP STAFF
AND THEIR CONSULTANTS

The following agenda items were discussed with resolution or comments as follows:

A. Component Cooling Surge Tank

The review comments contained in Section 5.3.1.3 of NUREG/CR-1821 were discussed by GAI personnel acting as consultants to RG&E. Basically, they confirmed that no positive anchorage against sliding currently existed in the longitudinal direction and they provided a stress summary, Attachment A which indicated that the horizontal base plate and anchor bolts are over stressed. They have proposed a solution of adding two additional bolts to each of the two support legs. This has the effect of reducing bending stresses in the horizontal base plate and helping to carry shear stress.

It was pointed out during the discussion by Dr. Stevenson that the addition of the two new bolts in each saddle would induce loads from restraint of free end displacement due to thermal gradients that currently are not identified in the design of supports and tank. GAI representatives stated that they had reviewed the effect of the restraint of thermal expansion in the tank and attached pipe and stated that the resultant stresses were quite low. They have not completed the analysis of their proposed fix but assured that the acceptable stress limits presented in Attachment A would be met. Dr. Stevenson stated that if the stress limit criteria of Attachment A were met the resultant design should demonstrate seismic design adequacy. However, he further stated a personal preference that, if at all possible, the modified support system should not provide thermal restraint.

B. Refueling Water Storage Tank

GAI has not finished the analysis which is scheduled now for a 1 December 1981 completion. Dr. Stevenson reported the concern and conclusions reached in Section 5.3.6.6 of NUREG/CR-1821, that if the potential amplified response of the tank under impulsive load was considered instead of the assumption of tank rigidity used in the original design, linear elastic analysis would indicate that the tank shell would buckle and the anchor bolts fail. Results of the GAI analysis should be available for review by 1 December 1981.

C. Auxiliary Building Bracing

Bracing evaluation of the auxiliary building is scheduled for completion by 1 November 1981.

D. Anchorage of Electrical Equipment and Internally Mounted Components

Dr. Stevenson reviewed typical design fixes supplied by RG&E in response to I&E Bulletin 80-21 concerning anchorage of electrical equipment. The criteria used in modifying the anchorage as expressed in "Final Report Anchorage and Seismic Support of Safety Related Electrical Equipment" RG&E Project No. EWR-2831 dtd. 12/31/80 appeared quite conservative in that a factor of 1.5 times the peak of the applicable floor spectra was used for the design modification. The RG&E analysis also considered the effect of bolt prying in their reevaluation and redesign. In general, they used the expedient of providing new anchorage in the form of stick welded angles to the cabinet plate at the base which was then expansion bolt anchored to the concrete slab rather than evaluating the existing anchorage design and installation integrity. All internally mounted components and devices weighing more than 25 pounds were analyzed as separate assemblies. Attachment of all internal devices and components were surveyed to assure all indicated attachments in the form of bolts, screws, clips, etc. were installed.

Dr. Stevenson concurred that the electrical equipment anchorage design and internal mounted devices and components evaluations and modification appeared quite adequate. However, he expressed a concern that the load path structural design adequacy between an electrical component or device through the panel frame and bracing to the equipment anchorage had not been adequately demonstrated as required (see Attachment 8). This was a notable concern in Ginna as compared to Dresden-2 in that Dresden-2 provided upper lateral supports as well as new base supports to the cabinets thereby effectively halving the reaction forces and reducing bending moments by a factor of four. In addition, cabinet fundamental frequencies are increased by a factor of 3 as a result of the upper lateral restraint which in this case should also reduce the inertia loads.

Dr. Stevenson suggested that RG&E should structurally evaluate, on a sample basis, electrical panelboards, cabinets and racks to demonstrate their structural design adequacy to the requirements of the AISC Code as modified by the SRP Section 3.8.4 for the load combination which included the SSE.

E. Battery Racks

The battery racks are essentially the same as the Gould racks used on Dresden-2. Detailed structural analysis of the Gould racks for Dresden-2 indicate the only area of potential failure is in the wooden battens. In Ginna the existing racks have been stiffened by an external structural steel bracing system which is independently expansion anchored to the floor. In Dr. Stevenson's opinion the design modification to the racks is obviously capable of carrying currently defined seismic loads.

G. Valve Operators

In general, RG&E has made evaluation of Seismic Category I motor operated valves larger than 2" part of their seismic upgrade program where stresses in piping including the effect of eccentricity are determined to be within code allowables. The valve assembly is modeled for analysis as an equivalent tee section.

Dr. Stevenson expressed a concern that it is the smaller diameter piping that is particularly sensitive to eccentric valve laods. RG&E agreed to review its Seismic Category I 2" and under lines to identify any MOV. A separate calculation would be performed to evaluate the effect of valve eccentricity on the piping stresses.

To date no additional evaluation of valve operability has been supplied. See Section 5.3.1.7 of NUREG/CR-1821.

H. Essential Service Water Pumps

No additional information has been supplied. Demonstration of functionality during a seismic disturbance is still an unresolved issue.

ATTACHMENT-A

ANALYSIS REPORT

Component Cooling Water Surge Tank (CCWST) Supports

I. Analysis Basis

A. Models

The CCWST was considered to be an idealized single degree-of freedom rigid body supported by two saddle supports. The saddle supports were considered to be fixed at the top at the weld joint connecting them to the tank body, and pin connected at the base at the anchor bolts connecting them to the supporting structural steel beams.

B. Loads

The three orthogonal components of SSE seismic loads were determined by 1) considering the support system (combined saddle and beam) frequency in each direction, respectively, 2) using damping equal to 3% of critical damping, and 3) interpolating between floor response curves at elevations 271'-0" and 315'-0". Pressure and temperature loads were determined considering the tank design conditions (section 7.0 of the Design Criteria) and the lateral stiffness of the supporting structural steel beams.

C. Stresses

Stresses were calculated by hand using conventional formulas for stress and strain.

II. Analysis Results

Nomenclature is consistent with the definitions given in section 8.0 of the Design Criteria (unless noted). Only maximum Actual Stresses resulting from the load combinations specified in the Design Criteria are presented below. Also, only controling Stress Limits are defined.

Component/Location	Actual Stress (KSI)	Stress Limit (KSI)
A. Saddle		
1. All Vertical Plates	$\sigma_1 = 1.16$	1.5S = 21.75
2. Corner of Outside Vertical Flange Plate	$\sigma_1 + \sigma_2 = 32.23$	2.25S = 32.63
3. Shear Stress in Vertical Flange Plates	$\tau = 2.30$	Not defined by ASME code for Class III plate and shell structures, considered acceptable

Component/Location	Actual Stress (KSI)	Stress Limit (KSI)
4. Shear Stress in Vertical Web Plate	$\tau = 0.39$	"
5. Horizontal Base Plate	$\sigma_1 + \sigma_2 = 76.73$	$2.25S = 32.63$
6. Shear Stress in Welds joining Saddle Plates	$\tau = 33.17$	$F_v = 2.25S = 36.90$
B. Shear Stress in Weld joining Tank and Saddle	$\tau = 16.14$	$F_v = 2.25S = 36.90$
C. Anchor Bolts		
1. Shear Stress	$f_v = 15.90$	$F_v = 1.6S = 16.00$
2. Tension Stress	$f_t = 21.31$	$F_t = 1.6(26) - 1.8f_v = 12.97$

ATTACHMENT B

ATTACHMENT 1
ANCHORAGE AND SUPPORT OF SAFETY
RELATED ELECTRICAL EQUIPMENT
POINTS TO BE ADDRESSED BY SEP
LICENSEES IN DECEMBER 31, 1980
SUBMITTAL

1. Information should be provided not only for the anchorage of electrical equipment but also the entire support that provides a load path (such as bracing and frames), as well as support for internally attached components. The latter is especially important for cabinet or panel type electrical equipment (such as control panels, instrument panels, etc.) which has internally supported components. An example of a potential improperly supported internal component would be a heavy component cantilevered off a front sheet metal panel without additional support to a stronger and stiffer location. These inadequate supports for internal components also should be identified and corrected before December 31, 1980.

2. In order to verify that an anchorage or a support of safety related electrical equipment has adequate capacity, provide justification by test, or analytical means. If expansion anchor bolts exist, justification provided previously for IE Bulletin 79-02 can be utilized if applicable. The acceptance criteria for substantiating these judgements should be provided, this may involve specifying the factor of safety and allowable stress limits used for design and justifying the overturning moment and shear force used.

3. Provide a table listing all (to include both floor and wall mounted) safety related electrical equipment in the plant. For each piece of equipment provide the information described in the attached table (attachment 2).

   These investigations of each piece of equipment should determine:

   a. Whether positive anchorage or support exists

   b. The type of anchorage

   c. Whether internally attached components are properly supported

   d. Identify non-seismic Category I equipment, the dislodgement of which during an earthquake may be detrimental to safety related equipment and render them inoperable. Inspection of the anchorages of such non-seismic Category I equipment should be conducted. If positive anchorages do not exist, they should be identified and modified before December 31, 1980.

4. Wherever modifications of anchorages or supports are required, these modifications should be implemented and thoroughly documented.

5. The seismic design of cable trays may be treated as a separate problem, because of its complexity. Each licensee or the SEP Owner's Group should provide a separate action plan for the resolution of this issue within 30 days of receipt of this letter.

ATTACHMENT 2
SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND NON-SEISMIC CATEGORY I
ITEMS THAT MAY DAMAGE THIS EQUIPMENT

Equip. Name	Equip. ID	System In Which Installed	Location Bldg. & Elev.	Type of Anchorage*	Was Anchorage Modified Since Jan. 1, 1980	Internally Attached Components			Non-Seismic Cat I Items that could potentially interact with this equip.			I.D. of Document Supporting Conclusion
						Equip. Name & ID	Type of Support	Was Support Evaluated	Name & ID	Type of Support	Was Support Evaluated	

*Examples of Type of Anchorage:
1. Bolted to Equipment
2. Bolted to Concrete Wall
3. Bolted to Concrete Slab
4. Bolted to Block Wall
5. Welded to Embedded Channel

SEP SAFETY TOPIC EVALUATION

GINNA NUCLEAR POWER PLANT

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 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 seismic design; the Ginna plant was placed in Group I for review.

The Ginna plant, a pressurized light-water moderated and cooled nuclear reactor, is located on the south shore of Lake Ontario, about 16 miles east of Rochester, New York. Westinghouse Electric Corporation was the prime contractor for the plant. The Westinghouse engaged Gilbert Associates, Inc. as the architect-engineer responsible for the plant design and construction specifications. Bechtel Power Corporation was the general contractor for construction. The plant received its Construction Permit on April 25, 1966 and Provisional Operating License on September 19, 1969. Rochester Gas and Electric Corporation (RG&E), the owner, filed its application for a Full-term Operating License on August 9, 1972.

The Ginna plant was originally designed for an operating basis earthquake (OBE) with a peak ground acceleration (PGA) of 0.08g and reviewed for a safe shutdown earthquake (SSE) with a PGA of 0.2g. Housner ground response spectra scaled to the specified PGA's were used as seismic input for the analyses and design. The vertical component of ground motion was assumed to be the same as the horizontal components. For the anlayses of most seismic Class I structures (con-

tainment shell, containment internal structures, auxiliary building, and diesel generator building), the buildings were modelled as lumped mass-spring systems with fixed bases for calculating the fundamental frequency of each building; then, the corresponding spectral accelerations were used for performing the equivalent static analysis and seismic design. For the control building and intermediate building, only the seismic resisting mechanisms (shear walls and steel bracings) were checked to determine if they were capable of resisting the equivalent seismic loads. The same approach used for Class I structures was applied for the analysis and design of the seismic Class I piping systems and equipment with the Housner ground response spectra used as input. The damping ratios recommended by Housner were used for structural and system analyses. Chapter 3 of NRC NUREG/CR-1821 report, "Seismic Review of the Robert E. Ginna Nuclear Power Plant as Part of the Systematic Evaluation Program" (ref. 1) summarizes the details of the original analysis and design.

The SEP seismic review of Ginna 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.

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.T22.

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, and II-4.C. These topics relate to specification of seismic hazard at the site, i.e. site specific ground response spectrum for the Ginna site. The seismic input selected for the confirmatory analysis of Ginna facility, namely the Regulatory Guide 1.60 spectrum scaled to 0.2g peak ground acceleration, envelopes the Ginna site specific gound response

as shown in Fig. 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 Ginna 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 Ginna Nuclear Power Plant - Phase I Report". Then, the staff and our consultants conducted a site-visit. The purposes of this site visit were: (1) to observe the as-built plant specific features relative to the seismic design of the faciltiy, (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 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 one of 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 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 Ginna plant started in mid 1979, the site specific gound response spectra were not avilable. In order to perform the review on a sampling basis that could be applied with a 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 Ginna 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 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 analysis;

f) review analysis and design procedures; and

g) 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 Ginna facility was a limited review centering on:

- o Assessment of the general integrity of the reactor coolant pressure boundary.
- o 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, two (2) piping systems, seventeen (17) equipment components (mechanical and electrical) were fully evlauated. They were:

- o Structures - Containment building (containment shell and internal structures) and the interconnected auxiliary, turbine, intermediate, control, service, and diesel generator building complex.
- o Piping Systems - Portions of residual heat removal line and safety injection line.
- o Equipment - 12 mechanical items and 5 electrical items.

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 and methods considered the three-dimension dynamic response effects of buildings, interaction between buildings, equipment masses , structural damping in accordance with calculated stress levels, and so forth.

A. ANALYSIS OF CONTAINMENT BUILDING

The containment building is a vertical, cylindrical concrete structure with a flat base mat and a hemispherical dome. The building is 99 ft. high (from base mat to spring line) and has a 105 ft. inside diameter. The concrete wall, which is prestressed vertically and reinforced horizontally, is 3.5 ft. thick. The thickness of the reinforced concrete dome and base mat are 2.5 ft. and 2 ft. respectively. Housed by containment shell, the internal reinforced concrete structures are supported by the same base mat which is founded on bedrock.by means of post-tensioned rock anchors.

A hybrid computer model* was used for the containment building (containment shell, internal structures, and base mat). The containment shell was modelled as a fixed-base lumped mass-spring system and the internal structures were modelled as a fixed-base three-dimensional finite element model. These two models are coupled through the crane structure and the NSSS. Because the building is founded on rock, soil-structure interaction effect connections are not required. The detailed discussion of modelling techniques and the final dynamic model used for the confirmatory analysis are found in NRC NUREG/CR-1821 report.

*The model was originally developed by the licensee and their consultant (Gilbert Assoc., Inc.) and reviewed by the staff.

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 in-structure response spectra were summarized in Chapter 4 of NUREG/CR-1821 report. The results of structural evaluation showed that containment building is capable of withstanding the postulated SSE event.

B. ANALYSIS OF INTERCONNECTED AUXILIARY, INTERMEDIATE, TURBINE, CONTROL, SERVICE, AND DIESEL GENERATOR BUILDING COMPLEX

As shown in the plot plan (Fig. 2 of NUREG/CR-1821 report), the auxiliary, intermediate, control, and diesel generator buildings were classified as Class I structures and the turbine service buildings Class III structures. Most of these buildings are steel frame structures with reinforced concrete basements that are structurally connected together. Since the staff and its consultants believed that the coupling between all these buildings would effect the dynamic response of structures, systems and components, the buildings were modeled as a U-shape three dimensional space frame model with a fixed base to simulate the rock foundation. The same approaches, applied for the containment building analysis, were used here for

generating the building responses (dynamic moments, shears, member forces, etc.) and in-structure response spectra that were used as input for the evaluation of the piping systems and equipment. The details of modelling techniques, analysis procedures and analysis results are found in Chapter 4 of Ginna NUREG report. The results of evaluation showed that the buildings have sufficient capacity to withstand the postulated SSE event. However, four sets of steel bracing (bracing at northeast corner of auxiliary building and bracings in the south, north, and west walls of turbine building) were found to exceed the allowable stress level for the postulated SSE. The licensee provided additional information for review on October 28, 1981 and November 13, 1981 (Ref. 4 & 5). This open item is expected to be resolved by January 31, 1982 and will be addressed in a supplement to this Safety Evaluation Report.

4. ANALYSIS OF PIPING SYSTEMS

As a result of SEP preliminary seismic review of Ginna plant, NRC IE Bulletin 79-14, and other NRC Seismic requirements, the licensee initiated a seismic upgrade program after the completion of piping support modifications required by IE Bulletin 79-14. In order to conservatively respond to the SEP seismic review and possible future NRC seismic requirements, a set of analysis procedures and criteria that conform with current NRC review criteria (namely, R.G. 1.60 Spectrum, R.G. 1.61 damping, SRP criteria, etc.) were used for the piping analysis. To date, the analysis of all safety related piping systems

inside containment has been completed. The overall upgrade program is scheduled for the completion by 1984 refueling outage.

As discussed in the section B.2 of this report, two pipe lines from those piping systems completed to date were selected and analyzed independently to verify the adequacy of the as-built design and confirm the upgrade analysis results. The pipe lines selected were portions of residual heat removal (RHR) and safety injection (SI). system piping. 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 requirments 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 analysis results are found in the INEL report (Ref. 9). The results from the confirmatory analysis showed that the sampled piping systems are capable of withstanding the postulated SSE seismic input.

5. ANALYSES OF SELECTED MECHANICAL AND ELECTRICAL EQUIPMENT

The evaluation of equipment was done on a 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 re-evaluations of all individual components.

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 Ginna facility. The description of selected components, analytical procedures and evaluations are found in Chapter 5 of the Ginna NUREG report.

As discussed in the NUREG report, a total of 13 open items (stnuctural and/or functional integrity) out of 18 sampled equipment were addressed as a result of the evaluation. Some of these 13 items remain open due to lack of design information. After the review and incorporation of additional information submitted by the licensee (Ref. 10-15), the results are summarized below:

- o 3 Mechanical equipment items and one electrical item were found to be adequately designed.
- o The component cooling surge tank support system was found to require upgrading. The staff accepted licensee's design criteria and analysis results.
- o Refueling Water Storage Tank (RWST) was found to require upgrading. This item will be resolved as part of the integrated assessment.
- o Reactor Coolant Pumps were left open (structural integrity) due to lack of design information. The licensee agreed to provide additional information by January 29, 1982.

- The licensee's structural integrity evaluation of motor operated valves (both valves and piping) larger than 2" under their seismic upgrade program was considered to be adequate. The licensee included the reanalyses of small pipe line (2" in diameter and smaller), to which motor operated valves are attached, in the ongoing seismic upgrade program. A separate evaluation will be performed to determine the effect of valve eccentricity on the pipe stresses when the analysis results become available. The licensee has demonstrated that the functional integrity of motor operated valves will be maintained under the postulated SSE.

- The existing essential service water pumps were determined to be not qualified (structural, and functional integrity) due to the lack of support near the suction of the pumps, resulting in over stress in the pump casing support. These pumps are unique to the service water system.

- The modified anchorage and support systems for safety related electrical equipment as well as the evaluations and modifications of internally mounted elements of safety related electrical equipment are found to be adequate.

- Motor Control Centers and Switchgears - The structural design adequacy of the load path between an internally mounted component or device through the panel frame and bracing to the anchorage system was not evaluated due to lack of design information. The licensee agreed to provide this information by January 27, 1982. This item is expected to be closed out by January 31, 1982.

- Control Room Panels - In order to demonstrate the structural integrity (load path from a internally mounted element to anchorage and support system) of panels, the licensee agreed to conduct a low impedance test for a sample panel to determine the dynamic characteristics of the panels and to perform seismic analysis to demonstrate the design adequacy in the near future.

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

- 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.

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:

Structure - All safety related structures and structural elements of the Ginna facility are adequately designed to resist the postulated seismic event. However, four (4) sets of steel bracing system were found to exceed the allowable stress level for the postulated SSE. The licensee provided additional analysis information for review on October 28, 1981 and November 13, 1981. This open item is expected to be resolved by January 31, 1982.

Piping Systems - According to the results of SEP piping audit analysis performed for the sampled piping systems (Ref. 9), the piping systems have been found to be capable of withstanding the postulated SSE.

Mechanical Equipment - A total of 12 mechanical equipment items were sampled. From the 12 items, 7 have been determined to be adequate and two were determined to be inadequate. Generally, the remaining open items are due to lack of design information. This does not necessarily imply that safety deficiencies exist. Rather, it is the staff's judgement that documentation of the adequacy of these open items can be accomplished by February 28, 1982 and will be addressed in a supplement to this evaluation (Attachment 1). However, our evaluation on three (3) sampled safety related tanks (namely, component cooling surge tank, boric

acid storage tank, and refueling water storage tank) showed that the suprot of component cooling surge tank needs to be upgraded and the refueling water storage tank requires both with regard to support and structural integrity. Since two of the sampled tanks were found to require upgrading, the seismic review of safety related tanks should be performed by the licensee to demonstrate the design adequacy of the remaining safety related tanks (volume control tank and NaOH spray additive tank).

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 of 1980 (Refs. 16 & 17) has been completed, and found to be adequately designed (Attachment 1), (b) a program has been initiated for the documentation of seismic qualification (functionality 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 plant; 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 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 Ginna 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 Ginna facility will be required to address these new requirements along with other operating reactors.

Furthermore, since the ground response spectrum (0.2g R. G. 1.60 spectrum) used for Ginna seismic reevaluation envelopes the Ginna 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 Ginna plant has an adequate seismic capacity to resist a postulated SSE, and therefore, there is reasonable assurance that the operation of the facility will not be inimical to health and safety of the public.

REFERENCES

1. NRC NUREG/CR-1821 Report, "Seismic Review of the Robert E. Ginna Nuclear Power Plant a Part of the Systematic Evaluation Program", December 1980.

2. Letter from NRC to RG&E dated June 17, 1981.

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

4. Letter from RG&E to NRC dated October 28, 1981.

5. Letter from RG&E to NRC dated November 13, 1981.

6. Letter from RG&E to NRC dated February 27, 1981.

7. Letter from NRC to RG&E dated February 20, 1981.

8. Letter from RG&E to NRC dated April 1, 1981.

9. EGG-EA-5513 Report, "Summary of the R. E. Ginna Piping Calculations Performed for the Systematic Evaluation Program", July 1981.

10. Letter from NRC to RG&E dated January 7, 1981.

11. Letter from RG&E to NRC dated February 6, 1981.

12. Letter from RG&E to NRC dated May 26, 1981.

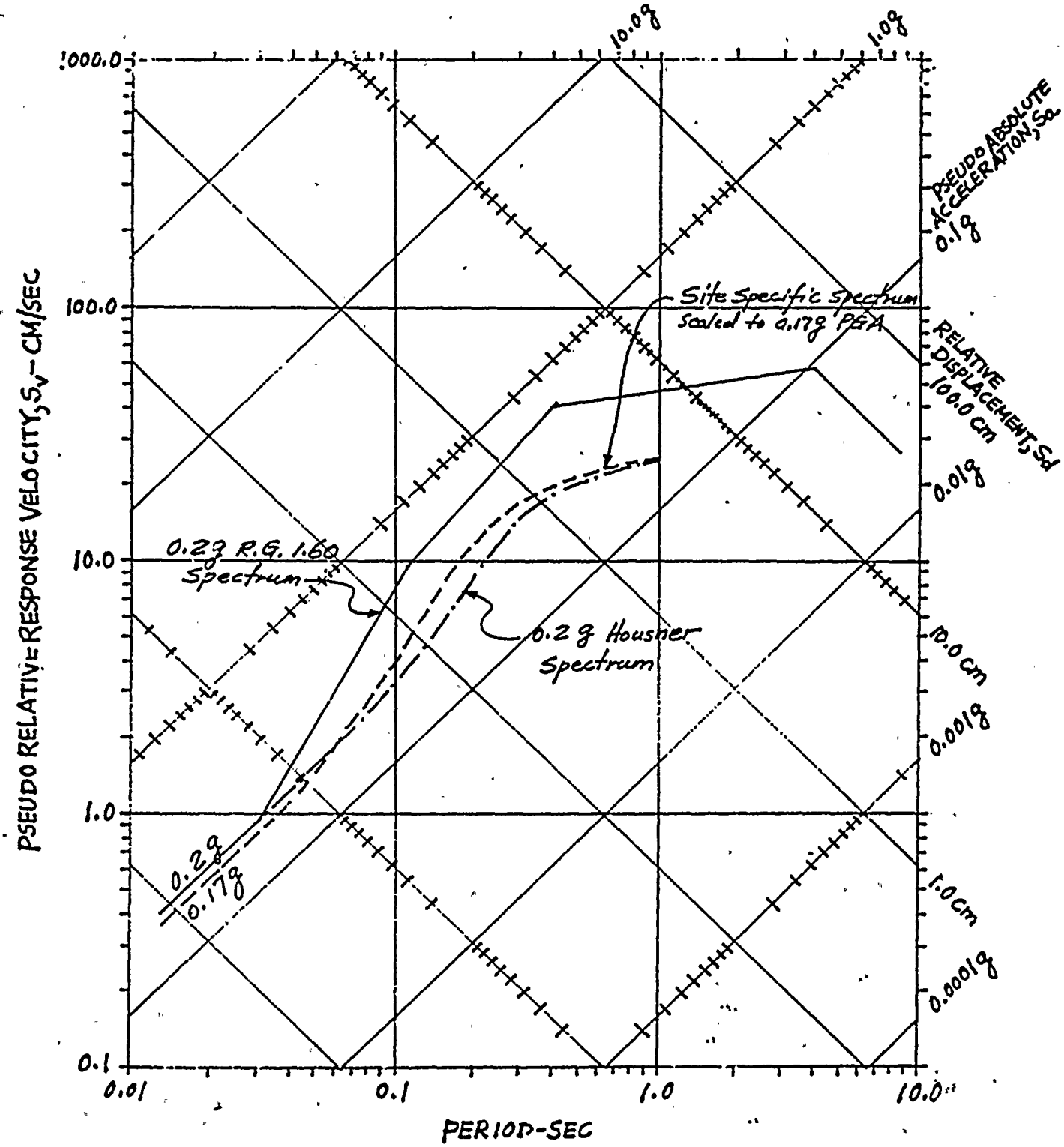
13. Letter from RG&E to NRC dated September 24, 1981.

14. Summary of September 9, 1981 meeting held at Rochester, New York dated December 14, 1981.

15. Summary of Integrated Assessment held on December 1, 1981 at Bethesda, Maryland (to be issued in the near future).

16. Letter from NRC to RG&E dated January 1, 1980.

17. Letter from NRC to RG&E dated July 28, 1980.



COMPARISON OF GROUND RESPONSE SPECTRA AT GINNA SITE

FIGURE 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 29, 1982

Docket No. 50-244
LS05-82-01-070

Mr. John E. Maier
Vice President
Electric and Steam Production
Rochester Gas & Electric Corp.
89 East Avenue
Rochester, New York 14649

SEE Repts.

Dear Mr. Maier:

SUBJECT: SEP SAFETY TOPICS III-6, SEISMIC DESIGN CONSIDERATION AND III-11, COMPONENT INTEGRITY - GINNA NUCLEAR POWER PLANT

We have completed our seismic review of Ginna Nuclear Power Plant. Enclosed is a copy of our draft combined evaluation report of the two subject topics.

As discussed in this draft report, four items are required to be upgraded to meet SEP requirements for the postulated SSE: (1) steel bracing at northeast corner of auxiliary building, (2) the support system of component cooling surge tank, (3) refueling water storage tank; and (4) essential service water pumps. Six items still remain open due to lack of design information. According to mutual agreement between the staff and your representative, the responses to these items are scheduled by January 31, 1982. A supplement to this report will be issued after the review of your responses for the six 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.

SE04 s 1/1 DSU WSE (07)
ADD: G. Staley A. Wang

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.

Sincerely,

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

Enclosures
As stated

cc w/enclosure:
See next page

*See previous concurrence

OFFICE	SEPB *	EPB *	SEPB *	ORB#5 *	ORB#5	AD:SA:DL	
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DATE	12/3/81	1/11/82	1/11/82	1/19/82	1/29/82	1/28/82	

Docket No. 50-244
LS05-

Mr. John E. Maier
Vice President
Electricaand Steam Production
Rochester Gas & Electric Corp.
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: SEP SAFETY TOPICS III-6, SEISMIC DESIGN CONSIDERATION, AND III-11, COMPONENT INTEGRITY - GINNA NUCLEAR POWER PLANT

We have completed our seismic review of Ginna Nuclear Power Plant. Enclosed is a copy of our draft combined evaluation report of the two subject topics.

As discussed in this draft report, some equipment items still remain open due to lack of design information. According to mutual agreement between the staff and your representative, the responses to these open items are scheduled by January 31, 1982. A supplement to this evaluation report will be issued after the review of your responses is completed.

You are requested to examine the facts upon which the staff has based its evaluation and respond either by confirming that the facts are correct, or by identifying errors and supplying the corrected information. We encourage you to supply any other material that might affect the staff's evaluation of these topics or be significant in the integrated assessment of your facility.

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.

Sincerely,

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

Enclosure:
As stated

cc w/enclosure:
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DATE	12/8/81	1/11/82	1/11/82	1/19/82	1/ /82	1/ /82

Mr. John E. Maier

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