ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE ROCHESTER, N.Y. 14649

February 11, 1980

Division of Reactor Regulation
Attention: Mr. Dennis L. Ziemann, Chief
Operating Reactors Branch No. 2
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Anchorage and Support of Safety-Related Electrical Equipment
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Ziemann:

In accordance with Mr. Eisenhut's letter on the above subject dated January 1, 1980 which was received January 11, 1980, RG&E has developed the enclosed Electrical Equipment Anchor Seismic Verification Plan. This plan addresses the concerns outlined in both the letter and the attached draft I.E. Information Notice.

Item (1). Does positive anchorage exist (load carrying mechanism other than friction)?

A field survey of all Class 1E electrical equipment at the Ginna Plant has been made. The findings of this survey are described in Enclosure II of our letter to you, dated July 3, 1979, on the subject of SEP Seismic Review. This survey showed that positive anchorage of all Class 1E equipment exists. Therefore no temporary anchors or supports will be added at this time.

Item (2). If positive anchorage exists, has the anchorage system been engineered with adequate capacity?

Item (3). Was the anchorage fabricated to quality standards?

Items (2) and (3) are addressed in the program described below.

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The R.E. Ginna Nuclear Power Plant was designed and constructed prior to the first issuance of IEEE Std. 344-1971, "Guide for Seismic Qualification of Class I.E. Equipment". Therefore to provide assurances that minimum design requirements have been met, an analysis and verification program has been developed.

The design verification will be accomplished in three phases, inspection, analysis, and test and modification phases.

The output of this plan will be a comparison of the Required Anchor Load Capacity (RALC) as determined by the analysis phase, with the Verified Anchor Load Capacity (VALC) for the anchor bolts associated with that component or assembly as determined by the test and modification phase.

If the VALC is found to be equal to or greater than the RALC, then no modification is required. However, if the VALC is found to be less than the RALC for an electrical assembly, additional anchors will be added.

It is the intent of this program to resolve the overall issue of electrical equipment anchorage seismic capability by September 1, 1980. The schedule for any required modifications will be dependent on their extent and equipment delivery schedules.

L. D. White, Jr.

L. D. White, Jr.

Enclosures

ELECTRICAL EQUIPMENT SEISMIC ACTION PLAN

Description

A three phase program will be initiated to provide assurances that the anchorage system bolts will perform their design function during the SSE. Phase I will consist of inspecting, and preparing "as-built" sketches for all safety related electrical equipment as listed below. Anchor bolts used on this equipment will be field inspected. "As built" sketches will be prepared showing all necessary information to perform Phase II. Phase II will consist of an analysis of each electrical equipment anchoring system, the results of which will be compared to the test information. Phase III will consist of testing the anchor bolts and any resulting modifications required to upgrade the existing anchoring system to the criteria described in Analysis Phase II section.

Equipment Addressed

The action plan will include all Class 1E electrical systems and components. Certain Class 1E equipment installed during recent modifications in accordance with IEEE 344-1975 requirements is known to be seismically anchored and will not be considered in this study. These components are listed below;

Electrical Assemblies Previously Qualified

Foxboro Instrument Racks
7.5 KVA Constant Voltage Transformers
7.5 KVA Inverters
Type W 480 MCC (M&L)

The following electrical assemblies and/or component will be evaluated by the Seismic Action Plan:

Electrical Assemblies Covered by this action plan

Relay Rack Assemblies
480 volt IE buses
480 volt (ac) IE MCC
125 Volt (dc) IE Starters
Power Panels
IE Battery Racks
IE Battery Chargers
Instrument Racks
Control Panels
DG Panels
Non.IE Items (Ancillary Items)

Inspection/Phase I:

An "as built" sketch will be developed for each piece of electrical equipment affected by this program. The diagrams will detail the size and shape of the component bases, the type, size and spacing of the anchor bolts and the physical dimensions and mass of the equipment.

Analysis Phase II

This phase will consist of an analysis of each anchoring system to determine the minimum anchoring requirement to safely secure the equipment during a seismic event using the following criteria and assumptions.

The static analysis described in Section 5.3 of IEEE 344-1975 will be the basis for establishing shear and tensile stresses expected in electrical equipment anchors being evaluated. Specifically, the seismic response of all floor-mounted equipment is assumed to be the peak of the RRS for the equipment floor location, using damping values in accordance with R.G. 1.61 , multiplied by a static coefficient of 1.5 to account for multifrequency and multimode responses. The inertial forces acting on the equipment center of mass are then evaluated. A multianchor computer model will then be used to determine the shear and tensile stresses for all floor mounted equipment using data from Phase I. The stresses thus determined will establish the Required Anchor Load Capacity (RALC) which will be compared to the Verified Anchor Load Capacity (VALC) determined in Phase III, to establish anchor adequacy.

Wall mounted electrical equipment will be assumed to be rigid and the zero period acceleration (ZPA) values will be used to determine the seismic forces. The tensile and shear stresses will be calculated using the multi anchor model.

Test and Modification Phase III:

After the "as built" sketches are completed, a test of anchor bolts, using existing ISI procedures, will be performed to determine bolt length, embedment, and the bolt verified anchor load capability (VALC).

Where the Required Anchor Load Capacity of an anchoring system is found to be greater than the Verified Anchor Load Capacities, a recommendation for additional anchor bolts or supports will be made. Equipment modifications will be made using existing anchor bolt installation procedures and the "as built" sketches will reflect all modifications.

Results:

The results of the analyses and tests will show the factors of safety for the existing anchor bolts by comparing the RALC to the VALC values.

Cable Tray and Conduit Support Anchors

The cable tray and conduit support anchors were installed using the manufacturers recommended procedures. To verify the adequacy of these anchors, a testing and verification program will be conducted to,

(1) determine the nominal span between anchor centers. This will be documented by sketches of representative cable tray and conduit runs with dimension and anchor locations.

(2) verify the installed capacity of anchors.

(3) compare the verified tray and conduit support anchor configuration with the configurations tested by Bechtel Power Corporation, described in ANCO Report #1053-21.1-4, "Cable Tray and Conduit Raceway Seismic Test Program".

It should be noted that the existing cable tray system at the Ginna Plant is a braced, strut supported system similar to those described in the test report. These tests, which were funded in part by RG&E, showed that in such tray systems, no major structural failures occurred during tests more severe than those required for the Ginna site.

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

LEON D. WHITE, JR.
VICE PRESIDENT

TELEPHONE
AREA CODE 716 546-2700

July 3, 1979

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis L. Ziemann, Chief
Operating Reactors Branch No. 2
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Systematic Evaluation Program - Seismic Review
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Ziemann:

During the April 10-11, 1979 site visit by the NRC Seismic Review Team, members of the team requested that we supply additional information relating to the seismic qualification of mechanical and electrical equipment; and fluid and electric distribution systems.

A subsequent meeting was held in Pittsburgh, Pennsylvania on June 12, 1979 between the NRC, RG&E and their respective consultants. It was agreed at the June 12 meeting that RG&E would submit additional information and expected submittal dates for mechanical and electrical equipment and systems by about June 29, 1979.

Accordingly, Enclosure I lists attached material and submittal dates for the mechanical equipment and systems and Enclosure II lists attached material and submittal dates for electrical equipment and systems.

As requested by your Staff, eight copies of this letter and enclosures are being supplied for your use. If there are any questions regarding this material, please contact us.

Very truly yours,

L D White, Jr.

L. D. White, Jr.

Enclosures

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

LEON D. WHITE, JR.
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July 3, 1979

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Attention: Mr. Dennis L. Ziemann, Chief
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Very truly yours,

L. D. White, Jr.

Enclosures

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Enclosure I

Mechanical Equipment & Systems

The following is a list of mechanical equipment and systems; and the tentative dates when RG&E plans to submit additional information on these items.

1. Essential service water pumps:

   The pump specification, outline drawings and foundation drawings are enclosed. Verification of the installation and seismic integrity will be submitted about October 1.

2. Component cooling water surge tank:

   The tank and foundation drawings are enclosed. Verification of the installation and seismic integrity will be provided about September 1.

3. Component cooling water heat exchanger:

   The heat exchanger and support drawings are enclosed. Verification of the installation and seismic integrity will be provided about September 1.

4. Diesel generator air tanks:

   The tank and foundation drawings will be provided about July 15. Verification of the installation and seismic integrity will be submitted about September 1.

5. Boric acid storage tank:

   The tank and foundation drawings will be provided about July 15. Verification of the installation and seismic integrity will be submitted about September 1.

6. Refueling water storage tank:

   The tank and foundation drawings will be submitted about July 15. Verification of the installation and seismic integrity will be provided about September 1.

7. Motor operator valves (electric/air) on lines < 4" diameter:

   Details of a typical installation and verification of seismic integrity will be provided about September 1.

8. Primary equipment inside containment (reactor coolant pump, pressurizer, steam generator and control rod drive mechanism):

   Equipment drawings, summary of equipment seismic analysis and equipment support drawings will be submitted about July 15. Verification of support installation and seismic integrity will be provided about September 1.

9. Interaction of seismic and non-seismic equipment (HVAC above panel in diesel generator room and steel platform over oil line to feed pump in diesel generator room):

   Verification of the installation and seismic integrity of these 2 items will be provided about September 1.

10. RHR system dynamic analysis (inside containment)

    Three copies of Gilbert Associates drawing C-381-354 Sheet 1, Revision A were forwarded to Mr. K. Jabbour on June 29, 1979 by express mail. That drawing shows the basic as-built geometry of the "A" RHR

system inside containment.

Additional copies of the piping drawing, pipe support drawings, and piping system design data necessary for analysis of the as-built conditions will be submitted about July 15. Oct. 15

11. Main Steam system dynamic analysis (inside containment):

The as-built piping isometrics of the B Main Steam line inside containment, support drawings, and piping system design data necessary for analysis of the as-built condition will be submitted about August 1.

12. Chemical and Volume Control System - equivalent static analyses (outside containment) Nov. 15

The as-built piping isometrics, support drawings, and piping system design data for a portion of the CVCS system will be submitted about August 1.

Members of the NRC Staff indicated in our June 12 meeting that this information should be submitted instead of the data specified in the meeting minutes of the site visit of April 10-11, 1979.

13. Sample field run of 2" piping: Nov. 15

Copies of an as-built piping isometric, support drawings, and system design data will be submitted about August 1.

Lawrence Livermore National Laboratory

September 11, 1981
SM 81-247
Docket #50-244
FIN A0415

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 information regarding the seismic capacity of a reactor coolant pump at the Ginna plant. While this open item has not been completely resolved, a rather simple action is proposed which should close this item.

Sincerely,

Thomas A. Nelson

Thomas A. Nelson
Structural Mechanics Group
Nuclear Test Engineering Division

TAN/mg
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Enclosure

STRUCTURAL MECHANICS ASSOCIATES
A Calif. Corp.
5160 Birch Street, Newport Beach, Calif. 92660 (714) 833-7552

SMA 12205.20

August 27, 1981

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

Subject: Resolution of Open Items on Ginna Equipment

Dear Tom:

In my June 22 letter, there were still two open items for Ginna equipment which were:

1. Control Rod Drives and Supports
2. Primary Coolant Pump

This letter is to advise you of the current status of these items.

George Wrobel of Rochester Gas & Electric (RGE) Company arranged for me to talk directly to Robert Kelly of Westinghouse, who was responsible for providing much of the Ginna equipment seismic documentation. Regarding loading documentation on the CRD assemblies, R. Kelly will send his analysis of CRD housing and seismic support loading for a 0.8g static coefficient loading. A quick review of the analysis should resolve all outstanding issues on the CRD system.

A legible copy of the primary coolant pump report was transmitted to me from RGE. Conclusions from the review of that report are contained in the attachment.

Tom Cheng suggested that I be present at an upcoming open items meeting for Ginna in September. In view of the current status of open items for which I have been responsible, I don't feel it necessary for me to attend the meeting. After my conversation with R. Kelly of Westinghouse, I feel that the CRD and primary coolant pump issues will be resolved very quickly and can be handled without a meeting. It is therefore recommended that these items not be a topic of the meeting.

Very truly yours,

STRUCTURAL MECHANICS ASSOCIATES, INC.

RD Campbell

Robert D. Campbell
Project Manager

RDC:lca
cc: T. Cheng (NRC)

ATTACHMENT

REACTOR COOLANT PUMP SEISMIC DESIGN REVIEW

The original stress report submitted for review, was illegible. A more legible copy was submitted to SMA by RGE on August 17, 1981, and the following conclusions can be reached from review of that submittal.

1. The pump was analyzed for a 0.8g horizontal static coefficient and a 0.54g vertical static coefficient. As reported in NUREG/CR-1821, the 7% damped peak spectral acceleration for both horizontal directions is 0.55 g's resulting in a vector sum of 0.78 g's. Thus, the equivalent static coefficient used in the original analysis is conservative by a small margin.

2. All stresses calculated for the 0.8g H and 0.54g V static coefficients are within allowables designated for the original design basis.

3. Analytical methods used in the design analysis are reasonable except in the case of pump nozzles.

4. The pump nozzles analysis is unrealistic and inadequate for the following reasons:

   a. Only the straight pipe portion of the nozzles were evaluated. Local membrane stresses in the pump casing were not computed.

b. The derived pump nozzle loads have no resemblance to actual achievable loads. The pump was assumed to be supported by the piping for purposes of deriving nozzle loads. This is probably highly conservative but not necessarily so. Actual piping reactions are available and should be used in an evaluation of the pump case.

A conversation between R. Kelly of Westinghouse and R. Campbell of SMA revealed that:

1. The San Onofre pumps are very similar to the Ginna pumps and that a detailed finite element analysis was conducted for the San Onofre Units for a specified set of nozzle loads.

2. A loop analysis of Ginna was conducted by Westinghouse for seismic loading. Actual pump nozzle loads are obtainable from the analysis.

Recommended Actions

RGE should have Westinghouse make a comparison of Ginna vs San Onofre pump casing geometry and pump nozzle loads and scale resulting stresses from the San Onofre pump analysis for Ginna nozzle loading conditions. The load/stress comparison and a comparison of nozzle and casing geometry should be sufficient to demonstrate seismic capability of the Ginna primary coolant pumps.

STRUCTURAL MECHANICS ASSOCIATES
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SMA 12205.20

June 22, 1981

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

Dear Tom:

In the Reference 1 submittal regarding review of open items on Robert E. Ginna Nuclear Power Plant, there were three (3) open items still remaining.

1. Control Rod Drive Mechanism
2. Reactor Coolant Pump
3. Steam Generator Tube Supports

Tom Cheng of the USNRC contacted me on June 16, to inquire if use of a site specific spectra for Ginna, in lieu of the regulatory guide spectrum anchored to 0.2g, would eliminate the oustanding items. I indicated that for two of the three outstanding items, lack of information, not marginal stress conditions, was the principal concern. As a result of the conversation, I have reexamined the three open items and can eliminate the steam generator tube support overstress problem by using Level D Service (faulted condition) allowable stresses and scaling up the original analysis results to applicable floor spectral accelerations. The control rod drive and primary coolant pump items still remain open but should be easily resolved with submittal of the necessary information. The remaining action items and resolution of the steam generator tube support stress problem are summarized in the attachment.

Very truly yours,

STRUCTURAL MECHANICS ASSOCIATES, INC.

R Campbell

Robert D. Campbell
Project Manager

RDC:lca
Attachment

cc: J. Stevenson

ATTACHMENT

SUMMARY OF OPEN ITEMS FOR ROBERT E. GINNA NUCLEAR POWR PLANT AS PART OF THE SEP PROGRAM

CONTROL ROD DRIVE MECHANISM

The submittals provided by the licensee, References 2 and 3, do not contain a correlation between loading used in the analyses and accelerations at the RPV support. A conclusion regarding seismic resistance of the CRD system cannot be reached without such a correlation. Westinghouse should be able to supply the necessary information to Rochester Gas and Electric.

REACTOR COOLANT PUMPS

The Reactor Coolant Pump Stress Report submitted for review, Reference 4, is illegible due to poor reproduction quality. A legible report needs to be submitted for review.

STEAM GENERATOR TUBE SUPPORTS

Section 16 of Reference 5 documents the stress analysis of Series 44 steam generator internals for seismic loading. A conservative analysis of the tube supports resulted in a primary membrane stress of 7900 psi for an equivalent static coefficient of 0.19g horizontal. The stress was calculated in the ligaments between the tube holes and circulating holes in a local area adjacent to a wrapper channel. The analysis conservatively ignored a redundant load path from the edge of the tube support to the tube holes. Reference 6, submitted for review, is an update analysis of the Series 44 steam generator but does not address the

tube support for horizontal loading. Neither Reference 5 or 6 describe the dynamic characteristics of the steam generator and its internals nor identify the material of construction for the tube support plates.

Reference 7, obtained during the SSMRP program, documents a generic dynamic analysis of the Series 51 steam generator for varying support stiffnesses and locations. The Series 44 steam generator is similar, but smaller. The fundamental frequency of the Series 51 steam generator ranges from 4.8 to 9.6 Hz depending upon the support stiffness and location. The fundamental mode is predominantly translation and rocking of the steam generator shell. A generic response spectrum that is flat through most of this frequency range was used to compute response accelerations and loadings in the steam generator. A review was conducted of Reference 7 to determine the degree of coupling between the steam generator shell and the internal structures and to establish the validity of considering the steam generator as a SDOF system for estimating an appropriate equivalent static coefficient for evaluation of the tube support plates. It was concluded from review of Reference 7 that at the most critical support location, as determined in Reference 5, that the tubes and tube supports would accelerate as rigid bodies with the shell; thus, using the spectral acceleration from the Ginna spectrum for a fundamental frequency of about 5 Hz is a reasonable approximation of an equivalent static coefficient to use for evaluation of the tube supports.

From the response spectra in Reference 8, for 7% damping, the maximum spectral accelerations at 5 Hz are 0.58 g's in each of two orthogonal directions. Combining the two directional accelerations, the resulting maximum vector is 0.82g. Using this value and scaling the ligament stress computed in Reference 5, the resulting ligament stress is 34,095 psi.

Reference 7 indicates that Series 51 steam generator internals are constructed of SA 285-Grade C carbon steel. A comparison of the allowable stress for this material at the design temperature of 556^{o}F to the allowable stress quoted for the Series 44 steam generator tube

supports in Reference 5, indicates that the Series 44 tube supports are also constructed of SA 285-Grade C or equivalent. This material and a design temperature of 556°F are used as a basis for establishing allowable stresses for the Safe Shutdown Earthquake.

The tube supports are considered to be Class 1 plate and shell type component supports and the allowable primary membrane stress is computed for Level D Service from Appendix F of the ASME Code, Reference 9. The allowable stress is the greater of 1.5 S_m and 1.2 S_y. For SA 285-Grade C material, 1.2 S_y governs and the allowable stress is 27840 psi. Note that the original design criteria limited the tube support stress to S_y.

Comparison of the calculated and allowable stress for Level D Service results in a 22% overstress condition. If site specific spectra anchored to 0.172g are considered in lieu of regulatory guide spectra anchored to 0.2g, the calculated stress decreases. Decreasing the calculated stress by the ratios of the site specific peak ground acceleration divided by the 0.2g peak ground acceleration used to generate floor spectra, the resulting ligament stress is 29,320 psi. This is still about 5.3% over the Level D Service allowable of 27,840 psi. In consideration of the conservatism inherent in obtaining the calculated stress, the computed 5.3% overstress condition is considered acceptable for several reasons.

1. The site specific spectrum envelope has lower spectral accelerations in the frequency range of the containment structure than the regulatory guide spectrum if both are anchored to the same peak ground acceleration. Consequently, the in-structure response spectra will be lower than those in Reference 8.

2. The static analysis from Reference 5 did not account for the redundant load path between the outside diameter of the tube support plate and the outer row of tube holes. The degree of conservatism could not be evaluated since pertinent dimensions are not provided in Reference 5. The degree of conservatism is certainly greater than 5% though.

3. The evaluation considered the applicable acceleration to be the vector sum of the two orthogonal directional accelerations. This assumes that both directional responses are in phase and that the resulting vector is aligned in the worst direction.

4. The in-structure response spectra were peak broadened ±15% and smoothed so that the resulting spectra are essentially flat from 2-1/2 to 9 Hz, which covers the range of fundamental frequency for the steam generators.

Items 3 and 4 are consistent with current regulatory criteria and are prudent conservatisms to cover many of the uncertainties in the simplified treatment of the tube support. Recommended acceptance is, therefore, based on the conservatism of Items 1 and 2 being sufficient to overcome the estimated 5.3% overstress condition. Further analyses or submittals from the licensee are not considered necessary.

REFERENCES

1. SMA letter, R. D. Campbell to T. A. Nelson, Review of Open Items Resulting from Seismic Review of the Robert E. Ginna Nuclear Power Plant as Part of the SEP Program, 4 May 1981.

2. High Speed Control Rod Drive Stress Analysis June 26, 1968.

3. Control Rod Drive Mechanism Seismic Frame Calculations, August 13, 1968.

4. Static Seismic Load Stress Analysis, Model RGE Pump (V-11001-81), July 30, 1968.

5. Vertical Steam Generator Stress Report, Westinghouse Electric Corporation, Tampa Division, April, 1969.

6. WTD-SM-75-028, 44 Series Steam Generator Stress Report, External Load Analysis Update, May, 1975.

7. Stress Report, 51 Series Steam Generator, Generic Seismic Analysis, Westinghouse Electric Corporation, Tampa Division, December, 1974.

8. NUREG/CR-1821, Seismic Review of the Robert E. Ginna Nuclear Power Plant as Part of the Systematic Evaluation Program, 15 November 1980.

9. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Appendices, 1980.

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

SMA 12205.20

May 4, 1981

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

Dear Tom:

SMA has reviewed the package of documents transmitted with your April 15 letter addressing open items on Ginna. Our comments and recommended action are contained in the enclosure.

Very truly yours,

STRUCTURAL MECHANICS ASSOCIATES, INC.

R D Campbell

Robert D. Campbell
Project Manager

RDC:mw
Enclosure

cc: J. Stevenson w/encl.

REVIEW OF OPEN ITEMS RESULTING FROM SEISMIC
REVIEW OF THE ROBERT E. GINNA NUCLEAR POWER
PLANT AS PART OF THE SEP PROGRAM

Reference 1 documents a review conducted by the Lawrence Livermore Laboratory and their consultants on the seismic adequacy of the Robert E. Ginna Nuclear Power Plant. Conclusions of the adequacy of the several items in the NSSS system were based upon summary information provided; however, the sources of the summary information were not available for independent review. Those items listed in Section 5.4 of the report as components for which seismic design analyses have not been independently verified are:

Reactor Control Rod Drive
Reactor Vessel Supports
Steam Generator
Reactor Coolant Pumps
Pressurizer and its Supports

References 2 through 7 were provided to SMA in response to the above identified open items. The following summary and conclusions resulted from SMA's review of the submittals.

1) Control Rod Drive Mechanism

In reference 7, the allowable bending moment in the CRDM due to seismic loads is developed. This report does not, however, provide a correlation between bending moments and acceleration levels. Thus, no conclusion can be reached on the basis of the submittal.

Reference 2 contains a stress analysis of the control rod drive support structure. The analysis provides stresses as a function of a static load "P" in pounds. There is no correlation between this static load and acceleration level. Therefore, a conclusion on the seismic capability of the support structure cannot be reached based upon the submittal.

SMA's experience with the SSMRP reference plant, which uses 106A full length control rod drive mechanisms, indicated that a large margin of safety exists for a 1.15g spectral acceleration loading condition and we would not anticipate a seismic problem with the Ginna CRDM.

Recommended action - CRDM loads documentation applicable to Ginna need to be supplied to SMA for final resolution.

2) <u>Reactor Vessel Supports</u>
Documentation verifying the seismic adequacy of the reactor vessel supports was not submitted. Based upon SMA's SSMRP experience for nozzle supported RPV's the seismic induced stresses in the nozzles and adjacent shell are very small and the governing element for RPV support is the concrete shield wall. The shield wall was considered in Ref. 1 to be adequate to withstand the 0.2g SSE.

3) <u>Steam Generator</u>
Reference 6 contains a 1969 static analysis of the Series 44 steam geneator. The most critical area due to seismic loading identified in this report is the tube support baffle ligaments which are stressed past yield for a 0.38g horizontal static load. The SEP revised spectra result in an SRSS response of 0.85g horizontal which will increase the stresses by a factor of 2.24. Thus, based on the submittal, the tube support baffles are overstressed for the 0.2g SSE. The static analysis at these tube support baffles was done quite conservatively, however, and a more rigorous analysis will most likely result in lower state of stress.

In 1975, an update on the series 44 steam generator was conducted (Reference 4) but the tube support area was only evaluated for vertical seismic loading and not for the

horizontal seismic loading. Thus, the results of Reference 4 cannot be used to update the results of Reference 6 in the area of concern.

Recommended Action - Documentation evaluating the tube support baffle ligaments for horizontal seismic loading should be submitted for review.

4) Reactor Coolant Pumps

Reference 3 summarizes the stresses induced in the reactor coolant pump by a 0.8g horizontal and a 0.54g vertical loading. The reported stresses are below the ASME Code allowables, but SMA is unable to evaluate the model or the analysis due to poor reproduction quality of Reference 3. If the static analysis of the pump can be shown to be valid, then the stresses due to the revised Ginna spectra loading will be less than those contained within Reference 3, and thus acceptable.

Recommended Action - A legible copy of Reference 3 should be provided to SMA for review.

5) Pressurizer

Reference 5 contains a 1969 stress report of an 1800 cubic foot pressurizer. All pressurizers from 800 to 1800 cubic feet with cast and fabricated heads utilize the same support skirts, thus conservative generic analysis was conducted for the heavier 1800 cubic foot models. Based on this report, the loads resulting from the new Ginna spectra will cause an overstressed condition within the support flange. This is a very conservative conclusion, however, since the Ginna pressurizer is a smaller 800 cubic foot model plus, the flange analysis itself was very conservatively conducted using a beam bending model in lieu of a more rigorous finite element model.

The 1973 pressurizer report referred to in Reference 1 and the pressurizer summary stress report (reference 45 of Reference 1) were not supplied to SMA for review; thus, the adequate capacity for a 6.7g equivalent static load portrayed in reference 45 of Reference 1 cannot be verified.

Later analysis (References 8 and 9) obtained in the SSMRP program for the Series 51 1800 cubic foot cast and fabricated head pressurizers showed the supports to be acceptable for a 0.96g horizontal and 0.64g vertical loading condition. This later analysis utilized a finite element model of the skirt and flange as opposed to the conservative beam theory used in reference 5.

Recommended Action - No action required. The pressurizer stress analysis within references 8 and 9 and the statements in References 5, 8 and 9 that the support skirts are identical for the 1800 cubic foot and the 800 cubic foot designs substantiate the adequacy of the Ginna pressurizer for a 0.2g SSE.

REFERENCES

1) NUREG/CR - 1821 (UCRL - 53014) Seismic Review of the Robert E Ginna Nuclear Power Plant as Part of the Systematic Evaluation Program, November 1980.

2) Control Rod Drive Mechanism Seismic Frame Calculations, 8/13/68.

3) Static Seismic Load Stress Analysis, Model RGE Pump (V-11001-81), 7/30/68.

4) 44 Series Steam Generator External Load Analysis Update, 5/75.

5) RGE Pressurizer Stress Report, 9/69.

6) Vertical Steam Generator Stress Report, 4/20/69.

7) High Speed Control Rod Drive Mechanism Stress Analysis, 6/26/68.

8) 51 Series Pressurizer Generic Seismic Analysis, Westinghouse Electric Corporation, August 1974.

9) 51 Series Pressurizer Support Skirt and Flange Analysis, Westinghouse Electric Corporation, May 1974.