ROCHESTER GAS & ELECTRIC COMPANY
GINNA NUCLEAR POWER PLANT

STEAM GENERATOR HYDRAULIC SNUBBER REPLACEMENT PROGRAM

OCTOBER 1987

Revision 1

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1.0 INTRODUCTION

This report describes a proposed modification to the existing steam generator upper lateral support configuration at Ginna Station, and the analyses which demonstrate the acceptablility of resulting loads from postulated seismic and other design basis events.

1.1 Existing Design

Restraining supports exist for both the upper and lower portion of the steam generator (SG). The lower portion of each SG is restrained laterally and vertically by a set of supports independent of, and not affected by, the proposed modification. The upper portion of each of the two steam generators is restrained against lateral seismic and pipe break loads by eight, large (532,000 lb. capacity) hydraulic snubbers as shown in Figure 1. These snubbers are connected between the building structure and a ring girder which is attached to four lugs welded to the SG shell. The snubbers are installed in four pairs with one pair approximately parallel to the hot leg on the reactor side of the steam generator, and the other pairs placed approximately 90° apart.

1.2 Program Overview

The intent of the proposed upper lateral support modification is to replace six of the eight hydraulic snubbers per SG with rigid

structural members (bumpers), thereby minimizing the number of hydraulic snubbers in service for this application. The redesigned SG upper support configuration will retain two hydraulic snubbers on each steam generator ring girder. These snubbers, along with the rear bumpers, will restrain the steam generator against dynamic motions and loadings along the axis of the hot leg. Restraint of motions and loadings normal to the hot leg will be provided by the replacement bumpers in that direction. The redesigned SG upper support configuration is shown in Figure 2.

The replacement support hardware consists of individual structural assemblies which will be installed wherever an existing hydraulic snubber is removed. A typical assembly is shown in Figure 4. Each assembly is structurally rigid under compression but will allow freedom of movement in the tensile direction. Each assembly is individually adjustable in the field to ensure that clearances at each bumper position are adequate for RCL expansion yet do not exceed those permitted by the RCL analysis. The bumper assembly, and its individual components, will be sized and analyzed to withstand the new design basis loads. Detailed design of the rigid structural members has been performed by RG&E. Fabrication will be performed by a qualified supplier having a Quality Assurance Program meeting the requirements of ANSI N45.2.

1.3. Anticipated Benefits

The required maintenance, in-service inspection and testing of the existing snubbers are performed during annual refueling outages. Surveillance activities are performed periodically throughout the year. By replacing selected snubbers with bumpers, annual maintenance activities and, consequently, annual radiation exposures to maintenance personnel can be minimized. The hydraulic snubbers replaced with bumpers will be refurbished, and stored for use as spares. It is expected that spare parts procurement, as well as utilization of shop facilities and rigging equipment, can be optimized as a result of this snubber replacement program.

1.4 Primary System Qualification

The steam generator hydraulic snubber replacement program has resulted in changes in the response of the primary system. The effect of these changes upon the RCS equipment, piping and piping support system has been analyzed by Westinghouse. An independent review by a consultant with broad experience in RCS support design is also being performed. The use of rigid structural members (bumpers) in the SG upper lateral support system will change the degree of stiffness with which the SGs are restrained against dynamic loads. These new stiffnesses have been calculated and are included in the reanalyses. Loadings from a design basis pipe break (DBPB) postulated to occur in an auxiliary line (RHR, SI accumulator or surge line) branch connection have also been developed using the new upper lateral support stiffnesses, to assess the effect of the new SG upper support configuration on the reactor coolant system. Pipe breaks in the Main Steam and Feedwater piping at the corresponding SG nozzles have also been considered.

The analysis results indicate that RCL stresses and deflections have not changed significantly from previous analyses. The details of the RCL piping system analysis, for the revised SG upper lateral support configuration, are provided in Section 3.1 of this report.

The primary equipment supports were also re-evaluated for new support loads generated from the revised RCS piping system analysis based on the proposed SG upper lateral support configuration. The evaluation was conservatively performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code - 1974 Edition, subsection NF and Appendix F. A detailed discussion of the primary equipment support evaluation is provided in Section 3.2 of this report. Results of the evaluation are summarized in Table 6.

1.5 Intent of Report

This report is intended to present the structural qualifications for the redesigned steam generator upper lateral support configuration. It contains the supporting data to conclude that the maximum stresses in the RCS, and the primary equipment supports, are less than the Code allowable values.

2.0 DESIGN LOADS AND CRITERIA

2.1 Design Basis Loads

2.1.1 Loading Conditions

The SG hydraulic snubber replacement program will assure that adequate support capacity is maintained with respect to the design basis loads.

The RCL, with the modified steam generator upper lateral support configuration, was analyzed for the following loading conditions:

a. Deadweight
b. Internal pressure
c. Thermal expansion
d. Seismic events (OBE and SSE)
e. Postulated pipe ruptures at SG secondary-side nozzles (Main Steam, Feedwater)
f. Postulated pipe ruptures at RCL auxiliary line nozzles (Pressurizer Surge, SI Accumulator, Residual Heat Removal)

The loads are combined in accordance with Tables 1, 2 and 3.

The loading conditions were evaluated with the RCS at full-power conditions. This is consistent with generic analyses of this type, represents the higher probability event, and occurs when the pipe is stressed from design RCL pressures.

2.1.2 Postulated Pipe Ruptures

a. RCS Pipe Ruptures

The probability of rupturing primary system piping is extremely low under design basis conditions. Independent review of the design and construction practices used in Westinghouse PWR Plants by Lawrence Livermore National Laboratory (reference 2) has provided assurance that there are no deficiences in the Westinghouse RCL design or construction which will significantly affect the probability of double-ended guillotine break in the RCL. Westinghouse topical report, WCAP-9558, Rev. 1 (reference 1), provided the technical basis that postulated design basis flaws would not lead to catastrophic failure of the Ginna stainless steel RCL piping. This WCAP documented the plant specific fracture mechanics study in demonstrating the leak-before-break capability. This WCAP was reviewed by the NRC and its conclusions were approved for application to Ginna by letter dated September 9, 1986 (NRC approval of RG&E response to Generic Letter 84-04).

Terminal-end pipe breaks are postulated in the RCL at auxiliary line branch connection nozzles to the Residual Heat Removal (RHR) System, the Safety Injection (SI) Accumulator piping and the Pressurizer Surge piping. The terminal-end break at the SI accumulator line nozzle defines the limiting pipe break design basis loads for the SG upper lateral support system under emergency conditions.

b. Secondary System Pipe Ruptures

Existing postulated pipe break locations in the secondary systems were reviewed. Some intermediate break locations have been eliminated from consideration as described below. Existing postulated terminal-end breaks at Main Steam and Feedwater nozzles continue to be assumed.

i. Main Steam Line Ruptures

The previous controlling design load for the SG upper lateral support system was an arbitrary intermediate pipe break in the horizontal main steam line near the top of the SG (See Figure 3). NRC Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements", provides guidance for elimination of arbitrary intermediate breaks and will be applied to this program. Previous Ginna Seismic Upgrade Program analyses (recently reviewed in NRC Inspection No. 50-244/87-11), using ANSI B31.1 criteria, have been revised as necessary to reflect changes resulting from this snubber replacement program. Consistent with Generic Letter 87-11, these analyses have established that no intermediate pipe breaks need to be postulated in the Main Steam (MS) piping.

ii. Feedwater Line Pipe Ruptures

A terminal-end pipe break is postulated at the steam generator Feedwater inlet nozzle and now defines the limiting pipe break design basis loads for the SG upper lateral support system under faulted conditions.

2.2 General Criteria - Seismic Upgrade Program

The design codes and criteria utilized in the analysis are consistent with those used for RG&E's Seismic Upgrade Program. That program was initiated in response to IE Bulletins 79-02, 79-14, and the Systematic Evaluation Program (SEP). This program was reviewed during SEP and was approved by the NRC as documented in the SEP SERs for Topic III-6, "Seismic Design Considerations" and the SEP Integrated Assessment. NRC Inspection No. 50-244/83-18 and Inspection No. 50-244/87-11 provided a review of RG&E work performed in response to IEB's 79-02 and 79-14. Since 1979, RG&E has upgraded critical safety-related piping and supports, resulting in the reevaluation and modification of virtually all supports originally covered by the IEB's.

3.0 PRIMARY SYSTEM ANALYSIS

3.1 Piping Analysis

3.1.1 Mathematical Models

The RCL piping model consists of mass and stiffness representations for the two RCLs and the reactor vessel. Each RCL includes the primary loop piping, a steam generator and a reactor coolant pump. The primary equipment supports are represented by stiffness matrices.

The analysis of the RCS was performed using a two-loop model (See Figure 5) to obtain component and support loads and displacements. This model is identical to the one used previously in the Ginna Piping Seismic Upgrade Program except for the following:

a. The new SG upper lateral support design is represented by two stiffness matrices. One matrix provides stiffness along the snubber axis; the second provides stiffness perpendicular to the snubber axis.

b. Each existing pinned-end, tubular support column under the SG's and the RCP's is represented by a stiffness matrix based on revised stiffness values which account for the embedment of the supporting structural frame in the reinforced concrete slab. This is a more realistic representation of

the existing configuration and eliminates the need for translation of loads from global to local coordinates.

3.1.2 Methodology

The seismic analysis is performed by the envelope response spectra method. Peak-broadened floor response spectra for two percent and four percent critical damping (OBE and SSE, respectively) were used in conformance with Regulatory Guides 1.60 and 1.61. The use of four percent critical damping for SSE was developed and justified by low-displacement testing. The testing programs are described in WCAP-7921, which has been accepted by the NRC (reference 9). The modification in the SG upper lateral supports will not affect the conclusion of the damping testing program. Responses to the three directions of earthquake loading were evaluated in accordance with the Ginna Piping Seismic Upgrade Program by combining all three directional earthquakes by the square-root-sum-of-the-squares (SRSS) method. The Westinghouse epsilon-method of closely spaced modes combination was used in the analysis. The combination equations are presented in Appendix A. This method of combination of modal responses and spatial components is consistent with the NRC guidelines in Regulatory Guide 1.92. This method has been used on numerous other Westinghouse PWR's (such as Vogtle and South Texas) as discussed in their respective FSAR's. The NRC has approved the use of this method via the SER's associated with modal response combination on those Westinghouse plants.

Time-history forcing functions for the Pressurizer Surge, RHR and SI accumulator nozzle breaks were applied to the RCL analytical model to obtain the corresponding transient loads. The blowdown fluid thrust forcing functions at the break locations associated with these RCL auxiliary line nozzle breaks are time-history forces in the x, y and z directions. They are applied to the RCL analytical model at the lumped-mass point where each auxiliary line joins the RCL. Jet impingement loads generated by the blowdown of the severed auxiliary lines were also applied at the lumped mass point where the auxiliary line joins the RCL. The time-history internal fluid system loads in the primary loop piping are also applied to the RCL analytical model. These loads represent the traveling decompression blowdown waves and are calculated at each RCL location with a change in direction or change in flow area.

Pipe breaks postulated to occur on the secondary side of the steam generator at the Main Steam outlet nozzle and at the Feedwater inlet nozzle are modeled as step-function forces. The calculation of these forces is based on a simplified thrust coefficient, Ct, multiplied by the initial pressure force, P_oA (oriented along the axial nozzle centerline). Thrust coefficients of 1.26 and 2.0 (1.0 for thrust plus 1.0 for jet impingement) were used for breaks in the Main Steam and Feedwater lines, respectively.

3.1.3 Computer Programs

Piping analyses are performed on the "WESTDYN" Westinghouse computer program (reference 5). WESTDYN performs 3-dimensional, linear, elastic analyses of piping systems subjected to internal pressure and other loadings (static and dynamic). The program is capable of combining loads in accordance with the applicable code class of either ASME Section III or ANSI B31.1. Separate computer runs analyze each loading condition (deadweight, thermal, sustained loads, occasional loads, pipe break and seismic). The primary output from WESTDYN displays information about each analysis performed, including forces, moments, and displacements at each point. The WESTDYN computer code has been utilized on numerous Westinghouse plants and was reviewed and approved by the NRC in 1981 (reference 8). The code is verified for this application and a controlled version is maintained by Westinghouse.

3.1.4 Support Stiffnesses

To accurately represent the equipment supports in the piping analyses, the modified support system stiffness characteristics were developed for input to the piping analysis computer model. Individual spring constants provided in the local directions of restraint were developed for the modified SG upper lateral support configuration and the other RCL primary equipment supports. The stiffness calculations considered the stiffness characteristics of all structural elements in the load path including the supporting concrete, structural members, as well as the tension and compression stiffnesses of the remaining hydraulic snubbers.

During a seismic event loads may shift between the snubbers and the bumper along the axis of the hot leg. This shifting is bounded in the analysis by utilizing three values of the upper support stiffness (Kmin, Kmax and Kavg) in three separate analyses. The bumper is stiffer than the snubber. Thus, the lower bound value is, Case 1, $K_{MIN} = K_{SNUBBER}$ (compression). The upper bound value is, Case 2, $K_{MAX} = K_{BUMPER}$ (compression) + $K_{SNUBBER}$ (tension). K_{MIN} is the actual stiffness when the steam generator moves toward the reactor vessel. K_{MAX} is the actual stiffness when the steam generator moves away from the reactor vessel. Finally, a third value of $K_{AVG} = 1/2\ (K_{MIN} + K_{MAX})$ was used to provide data on an intermediate stiffness.

Several evaluations were performed using Case 1 and Case 2 stiffnesses, and the worst loads on each individual bumper were determined. The results are summarized in Table 8 along with corresponding loads based on the average stiffness value, K_{AVG}. Use of bounding stiffness values produces a decrease in the seismic stress margin at each location as compared with K_{AVG}. Adequate seismic stress margin still exists since the lowest margin, using the bounding stiffness, is 1.73 (SG 1B snubbers).

Based on these changes in seismic margin, and the calculated margins for loop piping (shown in Table 4) and other primary

equipment supports (shown in Table 6), it is concluded that adequate seismic margins exist for the redesigned SG upper lateral supports. The data in Tables 4, 5, 6 and 7 are based on the K_{AVG} value of SG upper support stiffness.

3.1.5 Piping Evaluation Criteria

The piping evaluation criteria are based on ANSI B31.1-1973 edition. The original design basis of the seismic Category I piping at Ginna was in accordance with the 1955 and 1967 editions of USAS B31.1. When USAS B31.1 was updated to the ANSI B31.1, the stress analysis formulae and stress intensification factors were revised. The primary stress equations in the initial B31.1 -1973 edition were similar to those given in the ASME Section III Code of that time. The stress intensification factors given in this version of B31.1 were expanded to include more fittings. In using ANSI B31.1, the Piping Seismic Upgrade Program updated the analysis to reflect ASME Section III concepts while still retaining the philosophy of B31.1. However, the stress intensification factor for butt and socket welds of the original edition of B31.1 have been used because of lack of original weld configuration information.

3.1.6 Piping Load Combinations

The piping was evaluated for the load combination defined in Table 1.

3.2 Primary Equipment Supports Evaluation

3.2.1 Methodology

The steam generator upper lateral support system has been redesigned by replacing six of the eight steam generator snubbers in each loop. The revised configuration is shown in Figure 2.

The RCL analysis model was revised to reflect the new support configurations. Computer analyses were performed, as described in Section 3.1, to generate new RCL loads on the primary equipment support system and the primary equipment supports were evaluated for these new loads. The evaluation was performed for supports associated with the reactor vessel, steam generators and reactor coolant pumps. In appropriate cases, finite element models of supports, via the STRUDL program, were utilized to assist in the evaluation. The supports were requalified for the required combinations of pressure, thermal, deadweight, seismic and applicable pipe rupture loads.

3.2.2 Support Loadings and Load Combinations

The loads used in the analyses and requalification of the equipment support structures are defined in Table 2. These loads were combined for the plant as identified in Table 3. The corresponding load combinations and the allowable service stress limits are also provided in that table.

3.2.3 Evaluation Criteria

The rigid structural members (bumpers) in the SG upper lateral support system are designed to the requirements of the current edition of the original design code (American Institute of Steel Construction, AISC Manual, 8th Edition). However, to evaluate the equipment supports for normal, upset, emergency and faulted conditions, the provisions of ASME Boiler and Pressure Vessel Code Section III, Subsection NF and Appendix F were used - 1974 Edition. The ASME B&PV Code Section III, Subsection NF was used to establish allowable stress criteria for the equipment support evaluation in lieu of the AISC Code because Subsection NF and Appendix F coupled with US NRC Regulation Guide 1.124 establish a more consistent and conservative set of criteria. For example, Subsection NF was developed specifically to address component supports whereas the AISC generally address building structures. Additionally, the use of Subsection NF, Appendix F, and RG. 1.124 require the use of material properties at service temperature, limit buckling to 0.67 critical buckling, and establish upper bound allowables on tension and shear stress. The evaluation was performed by hand calculations, and by computer analysis where appropriate.

3.2.4 Computer Programs

The primary equipment supports were evaluated by hand calculations and, where appropriate, by finite element element computer analysis using "STRUDL." STRUDL, part of the ICES civil engineering computer system, is widely used for the analysis and design of structures. It is applicable to linear elastic two- and three-dimensional frame or truss structures, employs the stiffness formulation, and is valid only for small displacements. Structure geometry, topology, and element orientation and cross-section properties are described in free format. Member and support joint releases, such as pin and rollers, are specified. Otherwise, six restraint components are assumed at each end of each member and at each support joint. Printed output content, specified by input commands, includes member forces and distortions, joint displacements, support joint reactions, and member stresses. The STRUDL computer code has been utilized on numerous Westinghouse plants and was reviewed and approved by the NRC in 1981 (reference 8). The code is verified for this application and a controlled version is maintained by Westinghouse.

4.0 EVALUATION AND RESULTS

4.1 Reactor Coolant Loop Piping

Table 4 provides the level of stress in the RCL piping and the allowable stresses from the Design Code (reference 4). The results show that the stresses in the piping are within allowable limits. A comparison between the maximum stress in the RCL piping for the current and redesigned support configuration shows that there are only very small changes in the calculated stresses.

4.2 Application of Leak-Before-Break

With the redesigned steam generator upper lateral support configuration, revised loads (forces and moments) in the RCL piping have been generated. The revised loads are compared with those loads in Generic Letter 84-04 (reference 7) in Table 5. The calculated axial stress (19.42 ksi) is 60% of the allowable axial stress (32.4 ksi). Based on the comparison, it is verified that the leak-before-break conclusions of WCAP-9558 Rev. 1 remain valid for the redesigned support configuration.

4.3 Main Steam Line Break Locations

The terminal-end break in the main steam line piping at the steam generator nozzle is a design basis pipe break. The maximum calculated stress intensity at intermediate locations for combined pressure, deadweight, thermal and OBE loadings is 27.1 ksi. This is less than the threshold stress intensity of 0.8 (1.2 S_h + S_A) or 29.6 ksi. Therefore, there are no high-stress intermediate break locations in the main steam lines inside containment.

4.4 Primary Equipment Supports

The stress margins for RCL equipment supports resulting from the RCL analysis considering the redesigned steam generator upper lateral support configurations are summarized in Table 6 for all loading combinations. The stress margin is defined as the ratio of the allowable support stress to the actual support stress. Loading evaluations performed with the redesigned support configuration demonstrate that all RCL equipment support stresses satisfy stress limits with an adequate margin of safety. Seismic margin is assessed by the stress margin for the load combination, (DW + TN + SSE). These stress margins are summarized in Table 7 for the existing and redesigned steam generator upper lateral support configuration. The results demonstrate that a significant margin of safety exists for the redesigned steam generator upper lateral support.

4.5 Primary Component Nozzle Load Conformance

The RCL piping loads on the primary nozzles of the reactor vessel, the steam generators, and the reactor coolant pumps were evaluated. The conformance evaluation consisted of load component comparisons, and load combination comparisons, in accordance with each of the respective Equipment Specifications or with applicable nozzle allowable limits. It was concluded that all RCL piping loads acting on the primary component nozzles were acceptable.

4.6 Evaluation of Auxiliary Lines

The RCL piping and primary equipment displacements were compared to the corresponding displacements used in the previous analyses. They are found to be less than the previous analysis results or within ± 1/16 inch. Due to the flexibility of the attached piping systems (designed to be inherently flexible to accommodate thermal growth of the RCS) and the gaps which exist between the pipe and the supporting structure, an increase in anchor motions at the loop connection point of up to 1/16 inch will not cause significant changes in piping stress.

Therefore, auxiliary piping systems attached to the RCL are not affected by the redesigned steam generator upper support configuration.

4.7 Building Structural Evaluation

4.7.1 Evaluation of Local Areas

Corbels and embedments were evaluated for tension loads and their capacity was found to exceed that of the hydraulic snubbers.

Corbels were also evaluated for the rigid strut bearing loads, and were found to be loaded to no more than 60% of allowable.

All evaluations were performed with respect to ACI-349, and Appendix B of ACI-349.

4.7.2 Secondary Shield Walls

Bumper elevations are the same as the Reactor Building Operating Floor. There is no localized bending, since the floor slab acts as a stiffening ring. Resulting tensile stresses are low, with a maximum of about 40% of allowable. All evaluations were done with respect to ACI-349.

4.7.3 Conclusion

In conclusion, the existing containment building structures are adequate for the new design basis loads associated with the new snubber/bumper SG upper lateral support configuration.

5.0 ADDITIONAL CONSIDERATIONS

5.1 Overtemperature Events

The design basis overtemperature event is the loss-of-load transient. RCL equipment support stress margins for this transient are adequate as shown in Table 6. An evaluation has also been performed for the overtemperature conditions following a feedwater line pipe break. The maximum load on any individual bumper was found to be 23.4 kips. This is significantly less than the 820 kips maximum capacity of each bumper. The corresponding RCL piping stresses were also found to be much less than the code-allowable thermal stress.

6.0 QUALITY ASSURANCE

6.1 Rochester Gas & Electric Corporation

The overall project is being conducted under the RG&E Quality Assurance Program. The replacement rigid structural members (bumpers) will be fabricated by a supplier having a Quality Assurance Program meeting the requirements of ANSI N45.2. RG&E has specified material traceability, welder qualification, non-destructive examination and other requirements in the purchase order.

6.2 Westinghouse Electric Corporation

The structural qualification work performed by Westinghouse has been independently reviewed at Westinghouse as a safety-related calculation and meets 10CFR50, Appendix B, Quality Assurance requirements. The detailed results of the analyses are maintained in Westinghouse Central Files in accordance with Westinghouse Quality Assurance procedures (ref. 10 and 11).

6.3 Altran Corporation

An independent, third party review is being performed by Altran Corporation and Dr. Thomas C. Esselman. Dr. Esselman and his associates will conduct a thorough review of the assumptions, design bases, analyses and other design documents produced by Westinghouse. .

7.0 CONCLUSIONS

Based on the results of loading evaluations of the reactor coolant system with the redesigned SG upper lateral support configuration the following conclusions are made:

a. The combination of hydraulic snubbers and rigid structural members (bumpers) which comprise the revised steam generator upper lateral support system maintain adequate restraint of each steam generator under the design basis loads.

b. The maximum stresses in the RCS piping and primary equipment supports are within Code allowables.

c. The maximum displacements in the RCS piping have been accounted for in analyses of auxiliary piping systems attached to the RCS, and do not significantly affect those analyses.

d. The reactor coolant loop piping and equipment supports continue to have acceptable margins of safety for all design basis events.

e. The Containment Building structures are adequate to carry the loads imposed by the new snubber/bumper SG upper lateral support configuration.

8.0 REFERENCES

1. WCAP-9558, Rev. 1, Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing A Postulated Circumferential Through-Wall Crack June 1980.

2. NUREG/CR-3660, UCID-19988, Volume 3, February, 1985, "Probability of Pipe Failure in Reactor Coolant Loops of Westinghouse PWR Plants," Volume 3, "Guillotine Break Indirectly Induced by Earthquakes," Lawrence Livermore National Laboratory.

3. ASME Boiler and Pressure Vessel Code, Section III, Subsection NF and Appendix F, American Society of Mechanical Engineers, 1974 Edition (for Supports Evalution).

4. ANSI B31.1 Power Piping Code 1967 Edition, including Summer 1973 Addenda.

5. "Piping Analysis Computer Codes Manual II" Westinghouse Proprietary Class 3, Westinghouse Electric Corporation, Pittsburgh, PA.

6. NRC Branch Technical Position MEB 3-1, Rev. 2, 1987 Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment (Generic Letter 87-11)

7. NRC Generic Letter 84-04, 2/1/84.

8. NRC approval letter for WCAP-8252 (WESTDYN), Letter from R.L. Tedesco, NRC, to T.M. Anderson, Westinghouse, dated 4/7/81.

9. WCAP 7921-AR, May 1974, "Damping Values of Nuclear Plant Components."

10. Westinghouse Power System Business Unit Quality Assurance Program for Basic Components Manual, WCAP-9550, Rev. 16, June 30, 1987.

11. Westinghouse NTSD/GTSD Quality Assurance Program Manual for Nuclear Basic Components, WCAP-9565, Rev. 11, Aug. 31, 1987.

Table 1
RCS PIPING
LOAD COMBINATIONS AND STRESS LIMITS

Condition	Loading Combination	ANSI B31.1 Equations
Normal	Design Pressure + Deadweight	11
Upset	Design Pressure + Deadweight + OBE	12
Emergency	Design Pressure + Deadweight + SSE	12
Faulted	Design Pressure + Deadweight + (SSE + DBA)**	12
Max. Thermal	Max. Thermal Stress Range***+ OBE Displacement	13
Normal & Max. Thermal	Design Pressure + Deadweight + Max. Thermal Stress Range + OBE Displacements	14

**SRSS combination of SSE and DBA loads
***Loss-of-load overtemperature transient condition

The piping stress equations are:

$$\frac{PD}{4t} + .75\, i\, \frac{M_A}{Z} \leq 1.0 S_h \qquad \text{Equation (11)}$$

$$\frac{PD}{4t} + .75\, i\, \frac{(M_A + M_B)}{Z} \leq \begin{matrix} 1.2S_h & \text{(Upset)} \\ 1.8S_h & \text{(Emergency)} \\ 2.4S_h & \text{(Faulted)} \end{matrix} \qquad \text{Equation (12)}$$

$$i\, \frac{M_C}{Z} \leq S_a \qquad \text{Equation (13)}$$

$$\frac{PD}{4t} + .75\, i\, \frac{M_A}{Z} + i\, \frac{M_C}{Z} \leq S_h + S_a \qquad \text{Equation (14)}$$

Where:

M_A = Resultant moment due to dead load and other sustained loads.

M_B = Resultant moment due to occasional loads.

M_C = Resultant moment due to range of thermal expansion loadings.

P = Internal Design Pressure.

D = Outside diameter of pipe.

t = Nominal wall thickness of pipe.

Z = Section modulus

S_h = Material allowable stress at maximum temperature.

S_a = Allowable stress range for expansion stress.

i = Stress Intensification Factor.

TABLE 2

DEFINITION OF LOADING CONDITIONS
FOR PRIMARY EQUIPMENT SUPPORTS EVALUATION

	Loading Condition	Abbreviations
1.	Sustained Loads	DW, Deadweight +P, Operating Pressure +TN, Normal Operating Thermal
2.	Transients	SOT, System Operating Transient
	a. Over-temperature Transient	TA
3.	Operating Basis Earthquake	OBE
4.	Safe Shutdown Earthquake	SSE
5.	Design Basis Pipe Break	DBPB
	a. Residual Heat Removal Line	RHR
	b. Accumulator Line	ACC
	c. Pressurizer Surge Line	SURG
6.	Main Steam Line Break	MS
7.	Feed Water Pipe Break	FW

TABLE 3

LOAD COMBINATIONS AND ALLOWABLE STRESS LIMITS FOR PRIMARY EQUIPMENT SUPPORTS EVALUATION

	Plant Event	System Operating Conditions	Service Loading Combinations	Service Level Stress Limits
1.	Normal Operation	Normal	Sustained Loads	A
2.	Plant/System Operating Transients (SOT) + OBE	Upset	Sustained Loads + SOT +OBE	B
3.	DBPB	Emergency	Sustained Loads + DBPB	C
4.	SSE	Faulted	Sustained Loads + SSE	D
5.	DBPB (or MS/FWPB) + SSE	Faulted	Sustained Loads + (DBPB or MS/FWPB) + SSE	D

Note:

1. The pipe break loads and SSE loads are combined by the square-root-sum-of-the-squares method.
2. Stress levels as defined by ASME B&PV Code Section III, Subsection NF, 1974 Edition.

TABLE 4

MAXIMUM REACTOR COOLANT LOOP PIPING STRESSES
(Based on K_{AVG})

ANSI B31.1 Code Equation(2)	(1) RCL Piping	Current Configuration Stress (ksi)	Redesigned Configuration Stress (ksi)	ANSI B31.1 Code Allowable Stress (ksi)	Percentage of Allowable
(11)	HL	7.2	7.2	16.8	43%
	XL	6.9	6.9	16.8	41%
	CL	6.9	6.9	16.8	41%
(12) Design and Upset	HL	9.8	8.0	20.1	40%
	XL	9.8	8.9	20.1	41%
	CL	10.0	9.4	20.1	41%
(12) Emergency	HL	11.7	8.6	30.2	29%
	XL	12.1	10.6	30.2	35%
	CL	12.5	11.5	30.2	38%
(12) (Faulted)	HL	*	19.7	40.3	49%
	XL	*	11.5	40.3	29%
	CL	*	17.8	40.3	45%
(13) See Note 3	HL	9.7	9.7	27.5	36%
	XL	5.3	5.3	27.5	20%
	CL	7.4	7.4	27.5	27%
(14)	HL	16.8	16.8	44.4	38%
	XL	11.1	11.1	44.4	25%
	CL	13.1	13.1	44.4	35%

NOTES:

(1) HL - Hot Leg, XL - Crossover leg, CL - Cold leg
* Pipe rupture loads were not considered. No faulted stresses were calculated for current design.

(2) Load combinations are shown in Table 1.

(3) Loss-of-load overtemperature transient effects are included.

TABLE 5

COMBINED LOADS FOR LOOP PIPING LEAK-BEFORE-BREAK
(Based on K_{AVG})

Load Combination	Axial Force (kips)	Bending Moment (in-kips)	Combined Axial Stress (ksi)
Normal	1939	16760	16.88 (calculated)
SSE	251	2820	2.54 (calculated)
Normal + SSE	2190	19580	19.42 (calculated)
Normal + SSE	1800	45600(2)	32.4 (allowable) (See Note 2)

Notes: (1) Allowable based on WCAP-9558, Rev. 1.
(2) Umbrella bending moment in NRC Generic Letter 84-04 is 42,000 in-kips.

TABLE 6

RCS PRIMARY EQUIPMENT SUPPORTS STRESS MARGIN SUMMARY

(Stress Margin = Allowable/Actual)
(Based on K_{AVG})

Service Level	Normal	Upset	Emergency	SSE	Faulted
Load Combination	DW+TN	DW+TA+ OBE	DW+TN+ DBPB	DW+TN+ SSE	DW+TN+ $[(SSE^2+PIBK^2)]^{1/2}$
SG Upper Supports					
Bumpers	See Note 3	2.53	3.24(ACC)	2.41	1.79(FW)
Snubbers	See Note 3	3.17	6.26(ACC)	2.25	1.11(FW)
SG Lower Supports					
Lateral	See Note 3	1.67	1.57(SURG)	1.77	1.21(SURG)
Columns	3.51	1.65	3.11(ACC)	3.29	2.19(MS)
Reactor Coolant Pumps					
Lateral	See Note 3	4.55	18.12(ACC)	8.10	7.46(ACC)
Columns	5.15	1.87	2.76(ACC)	1.87	1.87(ACC)
Reactor Vessel					
Lateral	See Note 3	4.33	1.31(ACC)	5.94	1.41(ACC)
Vertical	3.05	1.29	2.09(ACC)	4.53	3.45(ACC)

Notes:
1) The load symbols are defined in Table 2.
2) PIBK includes DBPB and MS/FW breaks
3) Under normal conditions no significant loads are imposed on these lateral support elements.

TABLE 7
STEAM GENERATOR UPPER SUPPORTS
SEISMIC LOAD MARGINS
(Based on K_{AVG})

		SEISMIC LOADS (DW+TN+SSE) (kips)			SGUS CAPACITY (Kips)		SEISMIC LOAD MARGIN (Allowable/Actual)	
LOOP NO.	BUMPER ID	EXISTING SGUS(1)	REDESIGNED SGUS	% CHANGE	EXISTING	REDESIGNED	EXISTING	REDESIGNED
1A	SN-1	582.0	410.4	-30	1064	1064	1.83	2.59
	1	582.0	335.4	-42	1064	1640	1.83	4.89
	2	582.6	410.5	-30	1064	1640	1.83	3.99
	3	582.6	410.5	-30	1064	1640	1.83	3.99
1B	SN-2	514.2	472.3	-8	1064	1064	2.07	2.25
	4	470.0	453.3	-4	1064	1640	2.26	3.61
	5	448.0	386.5	-14	1064	1640	2.37	4.24
	6	312.2	309.9	-1	532	820	1.70	2.64
	7	287.2	340.0	+18.4	532	820	1.85	2.41

(1) See Note Attached.

NOTE TO TABLE 7

The original seismic support load calculations included an additional contribution which is not required in the revised support load calculations. In the original case, the total seismic support plane load at the upper support was first calculated by dynamic analysis in global coordinates and then rotated to the local coordinates of the support members. In the revised case, the individual support members were modeled directly in the dynamic model so that a rotation from support plane loads to member loads were not required. The rotation of coordinates must be done conservatively, since there are no signs associated with the total seismic force components in global coordinates. Therefore, the original design loads are more conservatively calculated than the revised design loads.

TABLE 8

STEAM GENERATOR UPPER SUPPORTS
SEISMIC LOAD MARGINS
(Using K_{avg} and K_{max}/K_{min})

		SEISMIC LOADS (DW+TN+SSE) (kips)			SGUS CAPACITY (Kips)	SEISMIC LOAD MARGIN (Allowable/Actual)	
LOOP NO.	BUMPER ID	Kavg	Kmax/Kmin	% CHANGE	REDESIGNED	Kavg	Kmax/Kmin
1A	SN-1	410.4	533.5	+30	1064	2.59	1.99
	1	335.4	436.0	+30	1640	4.89	3.76
	2	410.5	533.7	+30	1640	3.99	3.07
	3	410.5	533.7	+30	1640	3.99	3.07
1B	SN-2	472.3	614.0	+30	1064	2.25	1.73
	4	453.3	589.3	+30	1640	3.61	2.78
	5	386.5	502.5	+30	1640	4.24	3.26
	6	309.9	402.9	+30	820	2.64	2.03
	7	340.0	442.0	+30	820	2.41	1.86

APPENDIX A

COMBINATION OF SEISMIC MODAL RESPONSES

For Seismic Category I components within the NSSS scope, the method used to combine modal responses is described below. The total unidirectional seismic response for NSSS equipment is obtained by combining the individual modal responses using the SRSS method. For systems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. The groups of closely spaced modes are chosen such that the difference between the frequencies of the first mode and the last mode in the group does not exceed 10 percent of the lower frequency. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the SRSS of all modes the product of the responses of the modes in each group of closely spaced modes and a coupling factor, ε. This can be represented mathematically as:

$$R_T^{\,2} = \sum_{i=1}^{N} R_i^{\,2} + 2 \sum_{j=1}^{S} \sum_{k=Mj}^{Nj-1} \sum_{\ell=k+1}^{Nj} R_k \; R_\ell \; \varepsilon_{k\ell} \qquad \text{(Equation A-1)}$$

where:

R_T = Total unidirectional response

R_i = Absolute value of response of mode i

N = Total number of modes considered

S = Number of groups of closely spaced modes

Mj = Lowest modal number associated with group j of closely spaced modes

N_j = Highest modal number associated with group j of closely spaced modes

$\varepsilon k\ell$ = Coupling factor defined as follows:

$$\varepsilon k\ell = \left\{ 1 + \left[\frac{\omega'_k - \omega'_\ell}{\beta'_k \omega_k + \beta'_\ell \omega_\ell} \right]^2 \right\}^{-1}$$

and,

$$\omega'_k = \omega_k \, [1 - (\beta'_k)^2]^{\frac{1}{2}}$$

$$\beta'_k = \beta_k + \frac{2}{\omega_k t_d}$$

where:

ω_k = Frequency of closely spaced mode K

β_k = Fraction of critical damping in closely spaced mode K

t_d = Duration of the earthquake

For example, assume that the predominant contributing modes have frequencies as given below:

Mode	1	2	3	4	5	6	7	8
Frequency	5.0	8.0	8.3	8.6	11.0	15.5	16.0	20

There are two groups of closely spaced modes, namely modes 2, 3, 4 and 6, 7. Therefore:

S = 2, Number of groups of closely spaced modes

M_1 = 2, Lowest modal number associated with group 1

N_1 = 4, Highest modal number associated with group 1

M_2 = 6, Lowest modal number associated with group 2

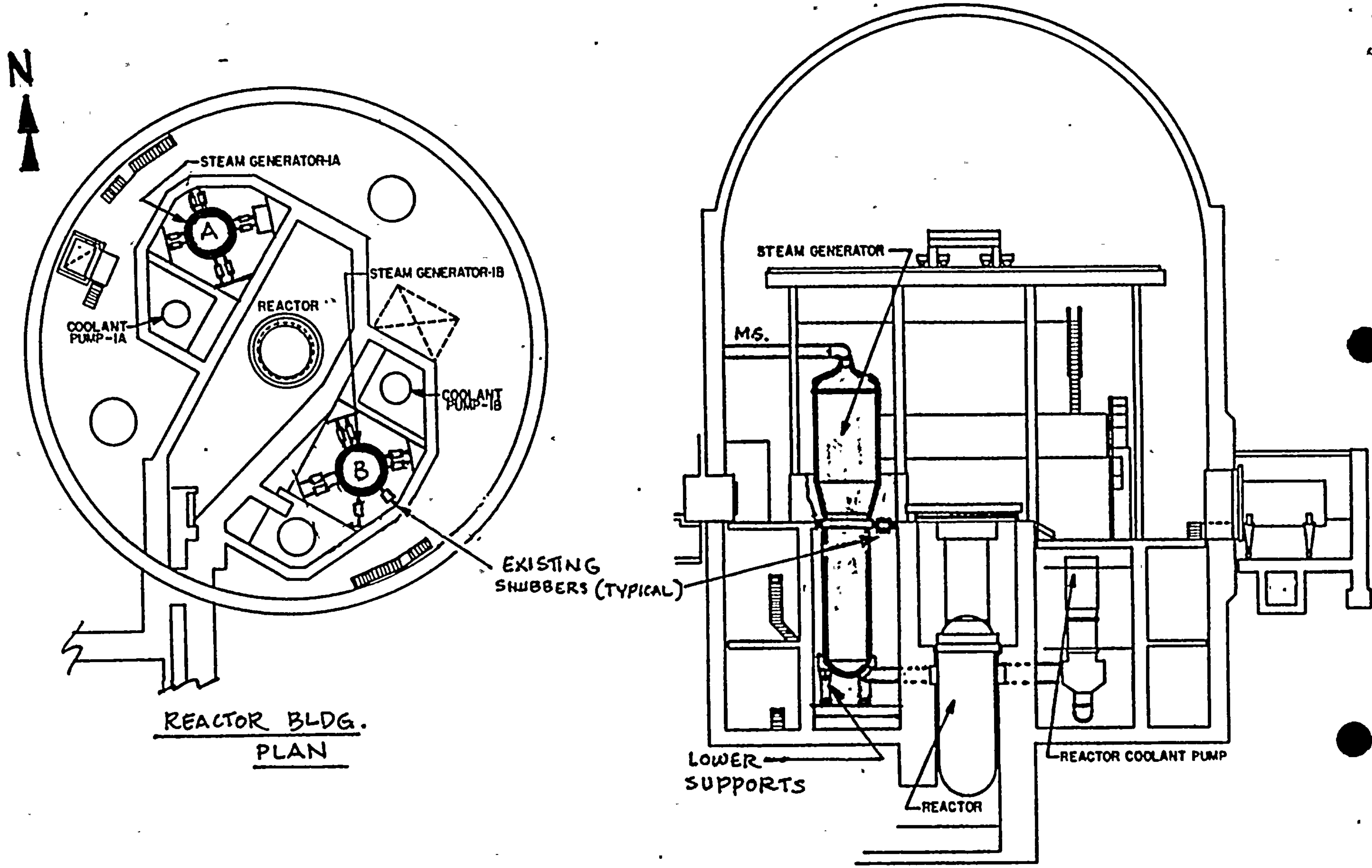
N_2 = 7, Highest modal number associated with group 2

N = 8, Total number of modes considered

The total response for this system is, as derived from the expansion of Equation A-1:

$$R_T^2 = [R_1^2 + R_2^2 + R_3^2 + \ldots + R_8^2] + 2R_2R_3\,\varepsilon_{23} + 2R_2R_4\,\varepsilon_{24} + 2R_3R_4\,\varepsilon_{34} + 2R_6R_7\,\varepsilon_{67}$$

The first term in brackets represents the SRSS summation of each of the eight example modes. The next three terms account for the additional effects due to interaction between example modes 2, 3 and 4. The final term similarly accounts for interaction effects between example modes 6 and 7.



REACTOR BLDG. PLAN

ELEVATION

FIGURE 1

GINNA STATION STEAM GENERATOR SNUBBERS - LAYOUT

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F-2

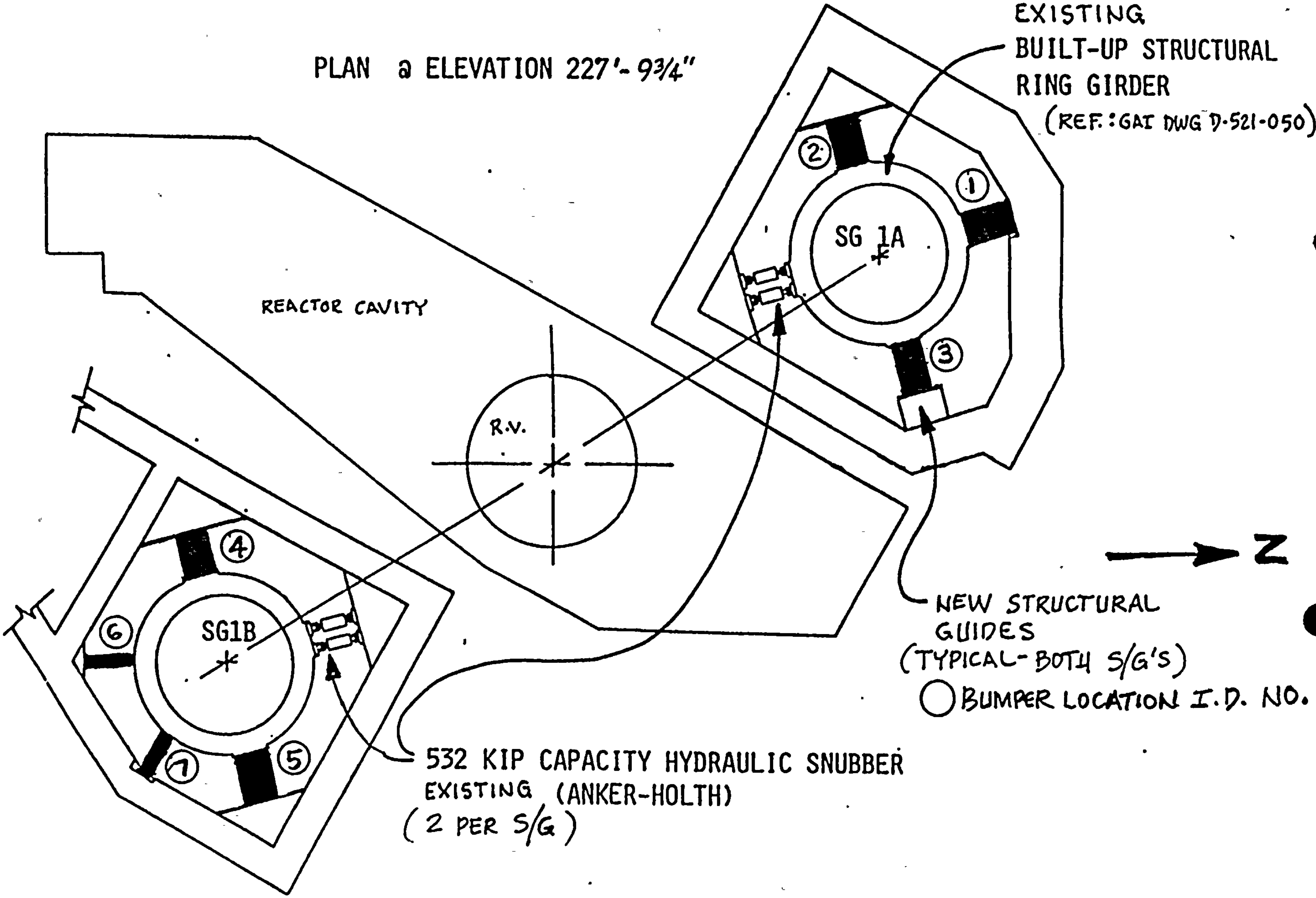

FIGURE 2

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10-15/87

GINNA STEAM GENERATOR SNUBBERS - STRUCTURAL MODIFICATION

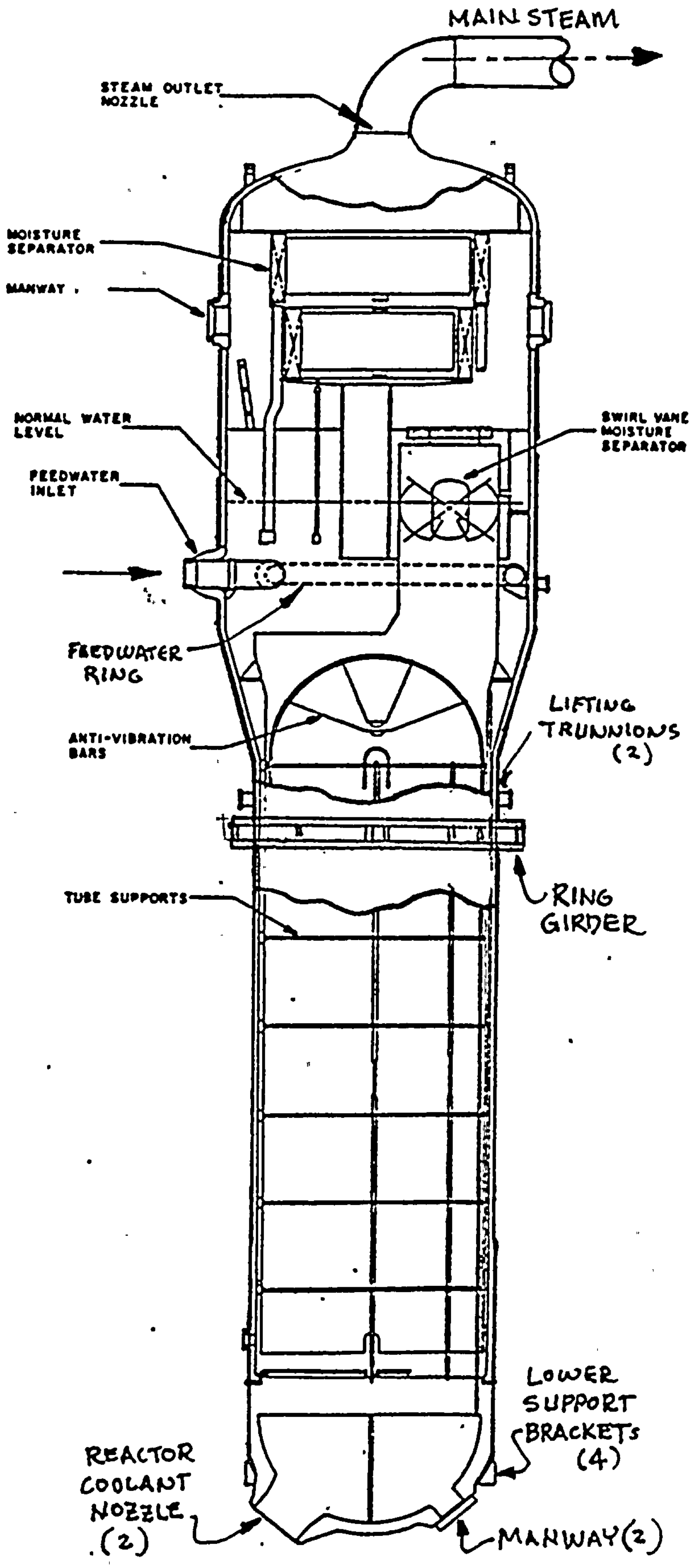

FIGURE 3

STEAM GENERATOR 1A/1B
(TYPICAL)

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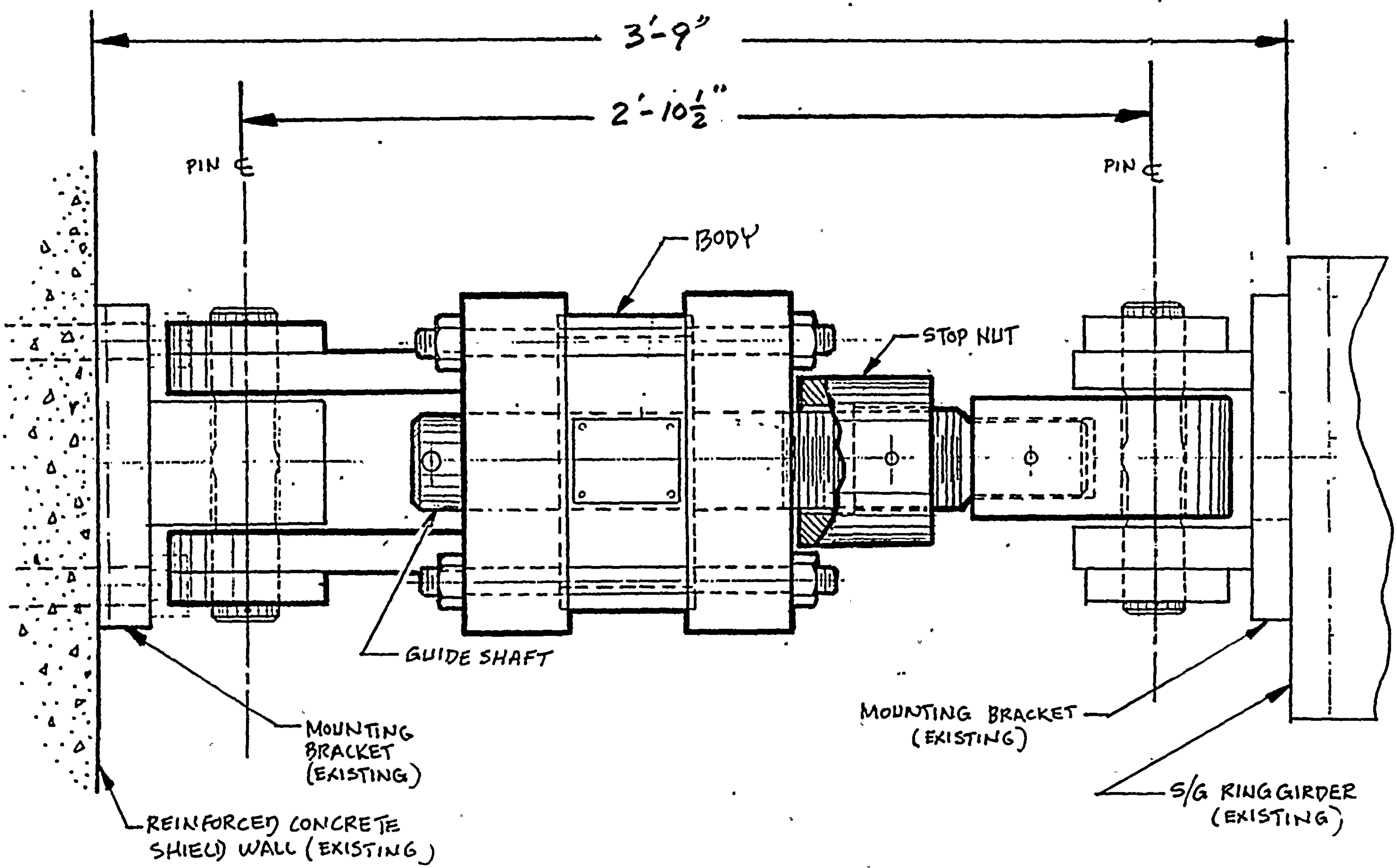

F-4

FIGURE 4

RIGID STRUCTURAL MEMBER (BUMPER)

(TYPICAL PLAN VIEW)

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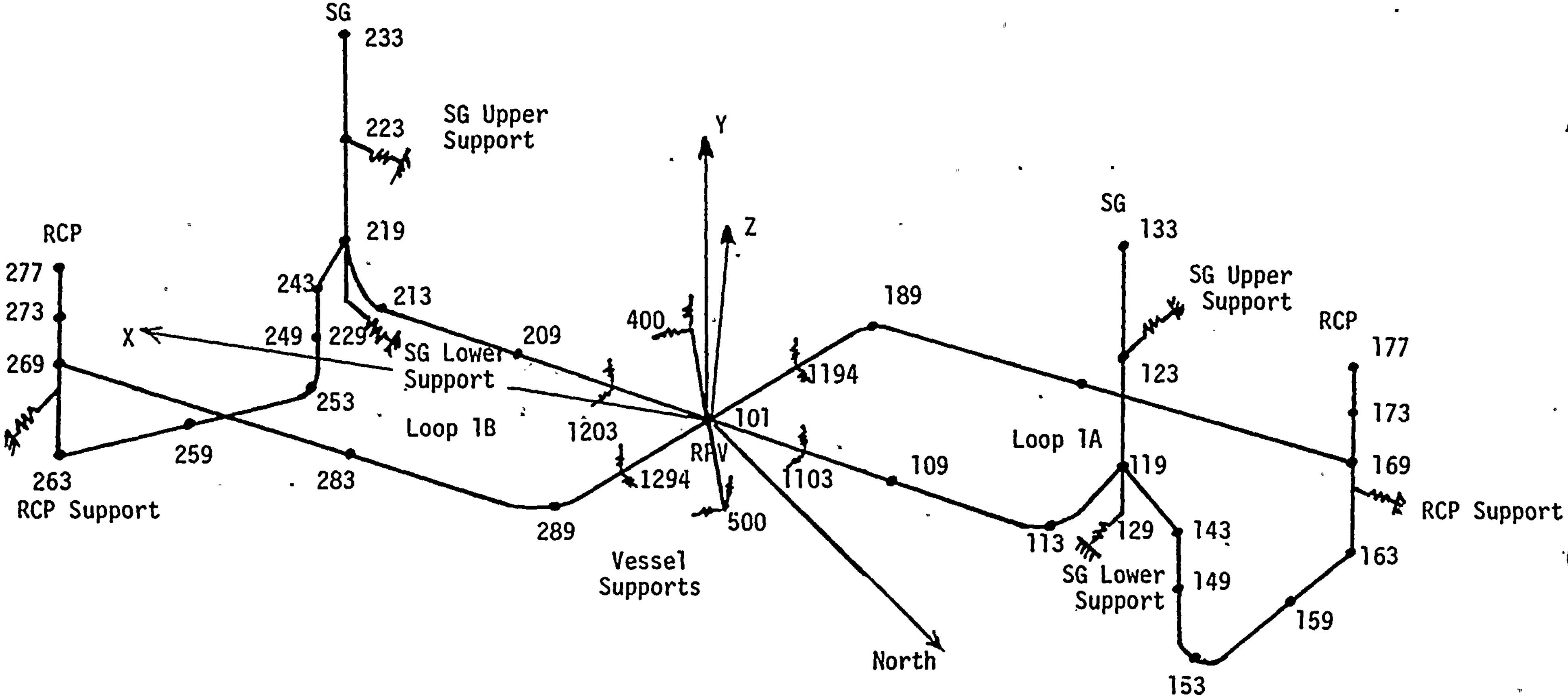

FIGURE 5: RGE GINNA REACTOR COOLANT LOOPS 1A & 1B ANALYTICAL MODEL (SEISMIC AND STATIC)

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