WISCONSIN PUBLIC SERVICE CORPORATION
KEWAUNEE NUCLEAR POWER PLANT

EVALUATION OF REACTOR COOLANT SYSTEM FOR REDUCTION OF STEAM GENERATOR SUPPORT SNUBBERS

REVISION 1

NOVEMBER 1987

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REVISION INDEX

REV. NO.	CHANGE	DATE
0	Original issue	October 1987
1	Revised table 6 (page 25)	November 1987

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I. INTRODUCTION

This report provides the results of the analyses performed to demonstrate the acceptability of a modification to the Kewaunee Nuclear Power Plant's (KNPP) steam generator upper lateral supports under all design basis loading conditions. The modification reduces the number of hydraulic snubbers at each steam generator upper lateral support (figures 1, 2, and 3). Snubber reduction is the direct result of excluding the dynamic effects of postulated reactor coolant loop (RCL) pipe ruptures and the elimination of loads associated with arbitrary intermediate breaks in main steam (MS) lines. These changes to the design basis are in compliance with limited scope General Design Criteria 4 (GDC-4) rule changes (reference 4) and Branch Technical Position MEB 3-1, Rev. 2, June, 1987 (reference 11). The change in support configuration is intended to improve access for snubber surveillance and reduce occupational radiation exposure during maintenance and testing.

The proposed steam generator (SG) upper lateral support design will have one hydraulic snubber instead of four as in the original design. The original design of all other primary equipment supports, including bumper supports on the other sides of the steam generator upper lateral support, will be retained. Adequate seismic support load margins have been demonstrated for the proposed design of the SG upper lateral support and all the other unmodified reactor coolant system (RCS) equipment supports.

The technical basis for this design modification is the use of "leak-before-break" technology for excluding the dynamic effects of postulated pipe ruptures in primary RCL piping from the design basis. Westinghouse topical reports, WCAP-11411, Rev. 1 (Proprietary), WCAP-11410, Rev. 1 (Non-Proprietary), WCAP-11619 (Proprietary), and WCAP-11620 (Non-Proprietary) (references 1 and 2), which documented the fracture mechanics analyses results and responded to NRC's request for additional information, were previously submitted. Reference 2 also reconciled the fracture mechanics analyses results to loads generated from the RCL analyses for the modified steam generator (SG) upper lateral support.

The purpose of this report is to demonstrate that the RCL piping, primary equipment nozzles, and the RCS supports, including the modified SG upper lateral support, are able to withstand all remaining loads with acceptable factors of safety; including those due to the design basis earthquake (DBE), and the limiting high energy line breaks at large auxiliary line branch nozzles. Specifically this report demonstrates that:

1. The maximum stresses in the RCL piping are within Updated Safety Analysis Report (USAR) (reference 5) allowables.

2. The RCS supports continue to have acceptable factors of safety.

3. The RCL piping loads acting on the primary equipment nozzles are within the equipment specifications allowables.

II. BACKGROUND

A. General

The RCL piping of Westinghouse pressurized water reactors (PWRs) has a demonstrated operating history of inherent stability in that no cracking failures have occurred. This indicates a low susceptibility to failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, and fatigue (for both low and high cycle).

An independent review of the design and construction practices, used in Westinghouse PWR plants, by Lawrence Livermore National Laboratory (reference 3) has provided assurance that there are no deficiencies in the Westinghouse RCS design or construction which will significantly affect the probability of double ended guillotine break in the RCL piping. The application of the "leak-before-break" technology (references 1, 2 and 4) to the RCL piping eliminates the requirement to design for the extreme loads associated with these previously postulated pipe rupture events. This provides the opportunity to reduce the number of the steam generator upper lateral support snubbers, which primarily react to pipe rupture loads.

Hydraulic snubbers are active components which require periodic removal for functional testing and seal replacement. Removal and inspection activities of large bore hydraulic snubbers expose maintenance personnel to radiation due to their location in the reactor containment building. Reducing the number of snubbers will reduce occupational exposure by facilitating maintenance and in-service inspections of components. Support system reliability is increased with the implementation of a regular in-service inspection and testing program. Some of the removed snubbers will be refurbished to provide spares for replacement, if required, thereby reducing radiation exposure and plant down time.

The elimination of snubbers, achieved by the exclusion of dynamic effects of postulated RCL pipe ruptures, will have minimal effect on the design margins for other loads. Except for the application of "leak-before-break" for elimination of dynamic effects of postulated RCL pipe ruptures and the elimination of arbitrary intermediate breaks in MS lines, the original licensing basis requirements are maintained.

B. Steam Generator Upper Lateral Support Configurations

The steam generator upper lateral support consists of a ring girder placed around the steam generator shell. The girder is hung from the steam generator compartment concrete wall by two Grinnel spring hangers. These spring hangers support the dead weight of the ring girder and aid in the vertical positioning of the girder. Once installed, the girder is shimmed to rest directly onto the SG lifting trunnions. Laterally, the girder is connected to four hydraulic snubbers placed parallel to the hot leg on the reactor side of the steam generator. The modified configuration contains one hydraulic snubber in lieu of four. This snubber, and a bumper behind the steam generator, parallel to the hot leg, restrain the steam generator for motions and loadings along the hot leg. Restraint of motions and loadings normal to the hot leg is provided by two additional bumpers that bear against the ring girder. These bumpers are attached to the secondary shield wall with embedded anchor bolt assemblies. Loads are transferred from the steam generator shell to the ring girder by means of twenty-eight bearing plates bolted to the ring girder. The SG upper lateral support system allows unrestrained thermal expansion of the RCS to the final normal operating position. In this position, each bumper to ring girder bearing surface is shimmed to provide proper contact, thus providing restraint to the steam generator in the operating position. The steam generator and its existing support system are shown in figure 1. A sketch of the current steam generator upper lateral support arrangement is shown in figure 2. The modified SG upper lateral support snubber arrangement is shown in figure 3.

Detailed descriptions of the remaining RCS equipment supports (SG lower and RCP supports), which remain unchanged, can be found in the Kewaunee Plant USAR section 5.4.14 (reference 5).

III. CRITERIA

The analyses criteria used were consistent with the licensing basis criteria of the original design basis analyses of record, except for the following:

i) Primary Reactor Coolant Loop piping postulated breaks were eliminated based on the use of "leak before break technology" in compliance with limited scope GDC-4 rule changes (reference 4).

ii) Main steam line arbitrary intermediate breaks were eliminated in accordance with the NRC Branch Technical Position MEB 3-1, Rev. 2 (reference 11), which provides relaxation in arbitrary intermediate pipe rupture requirements. Arbitrary intermediate breaks had been postulated in containment piping systems as design basis conservatisms even though these were not USAR requirements.

The remaining analyses criteria, consistent with the licensing basis, used were:

i) Original design basis floor response spectra,

ii) One half of one percent OBE and DBE damping values,

iii) DBE structural responses were twice OBE values,

iv) 2D seismic analysis with square root of the sum of the squares (SRSS) combination between modes, and the vertical response added absolutely to the horizontal response.

The effects of the postulated pipe rupture from the terminal end breaks at the RCS nozzles of the following large auxiliary piping were included in the faulted condition evaluation of the RCS:

i) Pressurizer surge line,

ii) Residual heat removal (RHR) line,

iii) Accumulator injection line,

iv) Main steam line,

v) Main feedwater line.

The codes used in the evaluation, consistent with the licensing basis were:

i) ANSI B31.1, 1967 Edition (reference 7), for RCL piping qualification,

ii) AISC "Specification for the Design, Fabrication and Erection of Structural Steel Buildings", 7th Edition, 1969 (Reference 6), for the RCS primary equipment supports.

The structural analyses criteria from the USAR for the Kewaunee Nuclear Plant, Wisconsin Public Service Corporation (reference 5), were maintained in the evaluation.

The above criteria is consistent with the criteria presented to the NRC staff at the July 7, 1987 meeting.

In addition, all replacement hardware was designed to meet the requirements of the ASME B&PV Code, Section III, Subsection NF, 1974 Edition (reference 10), and will be fabricated in accordance with the current edition of the ASME B&PV Code, Section III, Subsection NF, with the exception of the NPT stamping requirements. The use of the ASME Code is more conservative, with respect to design and fabrication requirements than the AISC manual (reference 6), which is the licensing design basis for the Kewaunee Nuclear Power plant.

IV. ANALYSIS

A. Mathematical Models

The analyses of the RCL piping were performed using the analytical model shown in figure 4. Analytical models for all loading conditions were revisions to existing design basis models, incorporating the proposed modifications to the steam generator upper lateral support. The results were verified against the original design basis results (reference 9) before updating the model to incorporate the proposed snubber modification.

B. Loading Conditions

The RCS, with the reduced snubber support configuration, was analyzed for the following loading conditions:

- o Seismic events (OBE and DBE), and
- o Postulated pipe ruptures at large auxiliary piping nozzles (pressurizer surge, accumulator injection, residual heat removal, main steam, and main feedwater)

For the seismic analyses, the design basis floor response spectra for one half of one percent critical damping (OBE and DBE) were conservatively used. Responses to the three directions of earthquake loadings were evaluated in accordance with the USAR by combining responses from the higher horizontal directional earthquake absolutely with those from the vertical earthquake.

The postulated terminal end breaks (at the RCS connection) of the pressurizer surge, residual heat removal (RHR), accumulator injection, main steam, and feedwater lines were analyzed to determine the most severe loadings on the RCL piping and support system, with the revised support configuration. Time-history forcing functions, for the pressurizer surge, RHR, accumulator injection, mainsteam, and main feedwater lines nozzle breaks, were applied to the analytical model of figure 4 to obtain maximum loads.

The absolute sum method was used for combining pipe rupture and DBE loads to demonstrate adequate safety factors for the maximum loadings.

C. Computer Programs

The RCL piping system analyses used the WESTDYN computer code (reference 8). The WESTDYN computer code has been utilized on numerous Westinghouse plants and was reviewed and approved by the NRC (reference 12). The controlled version of the code used for this project is verified for this application and is maintained by Westinghouse.

The modeling techniques used by Westinghouse for the Kewaunee Plant are similar to those used for many other plants. The reliability of these techniques is assured by the Westinghouse design control process and by comparison of results to other computer programs, including STARDYNE, a public domain code, and ME101, a Bechtel

Power Corporation code. The results of the RCL analyses, using the WESTDYN Code, by Westinghouse, were compared to the results obtained by an independent Architect/Engineering firm using the STARDYNE Code. The results were found to be within acceptable tolerances.

D. Structural Qualification of Piping and Supports

The structural analyses of the RCL piping and the primary equipment support system were performed with the modified steam generator upper lateral support configuration, while maintaining the original design for the other supports. The results of the analyses are used to perform the stress evaluations of the RCL piping, the primary equipment supports, and the connecting primary equipment nozzles. Revised primary equipment support concrete embedment loads were compared against previous concrete embedment loads and the differences were found to be insignificant. Therefore, the new loads are considered acceptable.

The stress criteria for the RCL piping and primary equipment supports are presented in the Kewaunee USAR. The piping and support allowable stresses are obtained from the appropriate editions of the ANSI B31.1 Code and the AISC manual, respectively (references 6 and 7). The emergency condition stress limits were conservatively used for the evaluation of the faulted condition pipe stresses.

V. RESULTS AND DISCUSSIONS

A. Stress in Reactor Coolant Loop Piping

Tables 1 through 5 provide the level of stress in the RCL piping and the allowable stresses from the Code (reference 7). A comparison is also shown between the maximum stress in the RCL piping for the existing and modified upper lateral support configuration for the controlling load combinations. The results show that the stresses in the piping are well within allowable limits.

B. Fracture Mechanics Evaluations

Westinghouse topical reports, WCAP-11411 Rev. 1 (Proprietary) and 11410 Rev. 1 (Non-Proprietary), have provided a substantial and adequate technical basis for limiting postulated design basis flaws in the Kewaunee Plant stainless steel RCL piping. The analyses have demonstrated that the probability of rupturing such piping is extremely low under design basis conditions. These WCAP's have documented the plant specific fracture mechanics study in demonstrating the leak-before-break capability.

Revised loads (forces and moments) in the RCL piping were generated for the modified steam generator upper lateral support configuration. These loads were included in analyses for WCAP's 11619 and 11620, (reference 2). WCAP's-11619 and 11620 responded to NRC's request for additional information on the reference 1 reports, and verified that the leak-before-break conclusions of WCAP-11411 Rev. 1 and WCAP-11410 Rev. 1 remain valid for the modified support configuration.

C. Primary Equipment Support Evaluation

The emergency and faulted loading conditions' factors of safety, for the primary equipment supports, including the modified steam generator upper lateral support, are summarized in table 6. Factor of safety is defined as the ratio of the allowable support stress to the actual support stress for combined loads. Loading evaluations, performed for the modified support configuration, demonstrate that the support stresses satisfy USAR limits with adequate factors of safety.

D. Primary Equipment Nozzle Load Conformance

The RCL piping loads on the primary nozzles of the steam generator, reactor coolant pump, and reactor pressure vessel were evaluated. The conformance evaluation consisted of load component comparisons and load combination comparisons, in accordance with each of the respective equipment specifications. All RCL piping loads acting on the primary equipment nozzles were within their allowables and are therefore acceptable.

VI. CONSERVATISMS

The Kewaunee Plant RCS seismic analyses were performed, using the response spectra with a conservative damping value of one-half of one percent for OBE, consistent with the licensing design basis. The DBE results were arrived at by considering loads twice those generated from the OBE analyses.

RCL piping stresses due to seismic motions are in hundreds of psi, thus providing considerable margins to allowable stresses. The modified steam generator upper lateral support OBE loads provide a factor of safety of 5.7 for the emergency condition. DBE loads, combined absolutely with the large auxiliary piping branch nozzle rupture loads, provide a factor of safety of 1.6 for the modified SG upper lateral support in the faulted condition.

No other supports were modified, thus maintaining their original design basis capacity. Marginal load increases were experienced at the support locations. These supports were evaluated using the new loads and the acceptance criteria were met.

The original RCS support design predates Subsection NF of the ASME Code. Although fabrication and inspection closely paralleled the requirements of the ASME Code, the allowable stresses were limited to 100 percent of the yield strength at the plant faulted loading conditions. The more liberal ASME Service Level D allowable stresses of up to 120 percent of yield for an elastic analysis were not used. In addition, comparisons of stresses or loads based on elastic limits are very conservative with respect to failure loads. The factors of safety (table 6), based on the 100 percent yield criteria, are substantial for the load combination consisting of seismic DBE, deadweight, thermal, and pressure. Such large margins eliminate seismic risk concerns.

VII. QUALITY ASSURANCE

The work was performed and independently reviewed as a safety-related analysis and meets 10CFR50, Appendix B, Quality Assurance requirements. The results of the analyses are maintained in the Westinghouse Central File system in accordance with WCAP-8370, Rev. 10A (reference 13), which has received NRC review and approval.

VIII. CONCLUSIONS

Based on the results of load evaluations of the RCS with the modified SG upper lateral support configuration, the following conclusions are made:

- RCL piping stresses are within USAR allowable limits.
- Adequate safety factors exist with respect to strength: and RCS structural integrity of primary equipment supports will be maintained for seismic and postulated pipe rupture events.
- The primary equipment nozzle loads are within acceptable limits.

IX. REFERENCES

1. WCAP-11411, Rev. 1 (Proprietary) and WCAP-11410, Rev. 1 (Non-Proprietary), Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee, April 1987.

2. WCAP-11619 (Proprietary) and WCAP-11620 (Non-Proprietary), Additional Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee, September 1987.

3. 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.

4. 10CFR Part 50, Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, Federal Register, Vol. 51, No. 70, April 11, 1986, p. 12502.

5. Kewaunee Updated Safety Analysis Report (USAR).

6. American Institute of Steel Construction, Manual of Steel Construction, Seventh Edition, including the "Specification for the Design, including the Fabrication, and Erection of Structural Steel for Buildings," February, 1969.

7. ANSI B31.1 Power Piping Code, 1967 Edition.

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

9. SD-103 "Structural Analysis of Reactor Coolant Loop/Support System for NSP (Prairie Island) and WPS (Kewaunee) Nuclear Power Plants, February 1972 (Proprietary).

10. ASME B&PV Code Section III, Subsection NF, 1974 Edition.

11. Branch Technical Position MEB 3-1, Rev. 2, June, 1987.

12. WCAP-8252, Rev. 1, "Documentation of Selected Westinghouse Structural Analysis Computer Codes" approved by NRC letter, April, 1981, from R. L. Tedesco (NRC) to T. M. Anderson (W).

13. WCAP-8370 Rev. 10A, "Westinghouse Electric Corporation Water Reactor Divisions Quality Assurance Plan, " August, 1984.

TABLE 1

MAXIMUM OBE STRESSES

REACTOR COOLANT LOOP PIPING

Leg	Existing Support[(1)] Configuration Stress (psi)	Modified Support Configuration Stress (psi)
HL	336	278
XL	336	240
CL	560	537

(1) Per the USAR, the required modal combination is SRSS summation of all modes. The analyses of record for piping conservatively reported the summation of max. mode + SRSS of the remaining modes. This note is also applicable to tables 2 thru 4.

TABLE 2

MAXIMUM DBE STRESSES

REACTOR COOLANT LOOP PIPING

Leg	Existing Support[1] Configuration Stress (psi)	Modified Support Configuration Stress (psi)
HL	672	556
XL	672	480
CL	1,120	1,075

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(1) See note, table 1.

TABLE 3

MAXIMUM UPSET CONDITION STRESSES[2]

REACTOR COOLANT LOOP PIPING

Leg	Existing[1] Support Configuration Stress (psi)	Modified Support Configuration Stress (psi)	B31.1 Code Allowable Stress (psi)	Factor of Safety: Existing[1] Support Configuration	Factor of Safety: Modified Support Configuration
HL	6,705	6,650	17,940	2.7	2.7
XL	6,619	6,487	17,940	2.7	2.8
CL	6,773	6,701	17,940	2.6	2.7

(1) See note, table 1.

(2) Maximum stresses by absolute sum of pressure, weight, and OBE.

TABLE 4

MAXIMUM EMERGENCY CONDITION STRESSES[2]

REACTOR COOLANT LOOP PIPING

Leg	Existing[1] Support Configuration Stress (psi)	Modified Support Configuration Stress (psi)	B31.1 Code Allowable Stress (psi)	Factor of Safety: Existing[1] Support Configuration	Factor of Safety: Modified Support Configuration
HL	7,041	6,928	26,910	3.8	3.9
XL	6,955	6,728	26,910	3.9	4.0
CL	7,333	7,238	26,910	3.7	3.7

(1) See note, table 1.

(2) Maximum stresses by absolute sum of pressure, weight, and DBE.

TABLE 5

MAXIMUM FAULTED CONDITION STRESSES(1)

REACTOR COOLANT LOOP PIPING

Leg	Existing(1) Support Configuration Stress (psi)	Modified Support Configuration Stress (psi)	B31.1 Code Allowable Stress (psi)	Factor of Safety Modified Support Configuration
HL	(2)	15,257	26,910	1.8
XL	(2)	11,274	26,910	2.4
CL	(2)	21,767	26,910	1.2

(1) Maximum stresses by absolute sum of pressure, weight, DBE, and pipe rupture loadings.

(2) USAR design limit curves used for qualification. The analysis of record used principal, axial, shear and hoop stresses and is not readily comparable to evaluation based on B31.1 Code allowables.

(3) The conservative emergency condition allowables are used for the evaluation of the modified support configuration faulted stresses.

TABLE 6

EMERGENCY AND FAULTED CONOITION FACTORS OF SAFETY

REACTOR COOLANT SYSTEM EQUIPMENT SUPPORTS

Support	Emergency Loading[1] Factor of Safety		Faulted Loading[2] Factor of Safety	
	4 Snubber	1 Snubber	4 Snubber	1 Snubber
SG Upper Lateral				
Snubbers (per Snub)	36.44	5.72	1.05[4]	1.58[5]
Ring Girder[3]	212.80	25.0	40.5	7.69
SG Lower Lateral				
Beam	43.48	100	3.05	6.67
Bumpers	>100	100	1.16	10.0
SG Vertical Columns	2.46	3.03	1.73	2.5
RC Pump Eye-Bars	3.89	6.25	1.0	1.03
RC Pump Columns	5.38	2.0	1.89	1.37

Notes: [1] Emergency Loading = Normal + DBE Seismic

[2] Faulted Loading = Normal + DBE Seismic + Pipe Breaks

[3] The four snubber analysis is based on a flexural analysis. The one snubber analysis is based on a bearing type analysis, therefore the results are not directly comparable.

[4] Includes double ended guillotine break in the horizontal run of the mainsteam line.

[5] Includes a double ended guillotine break in the feedwater line nozzle at the steam generator.

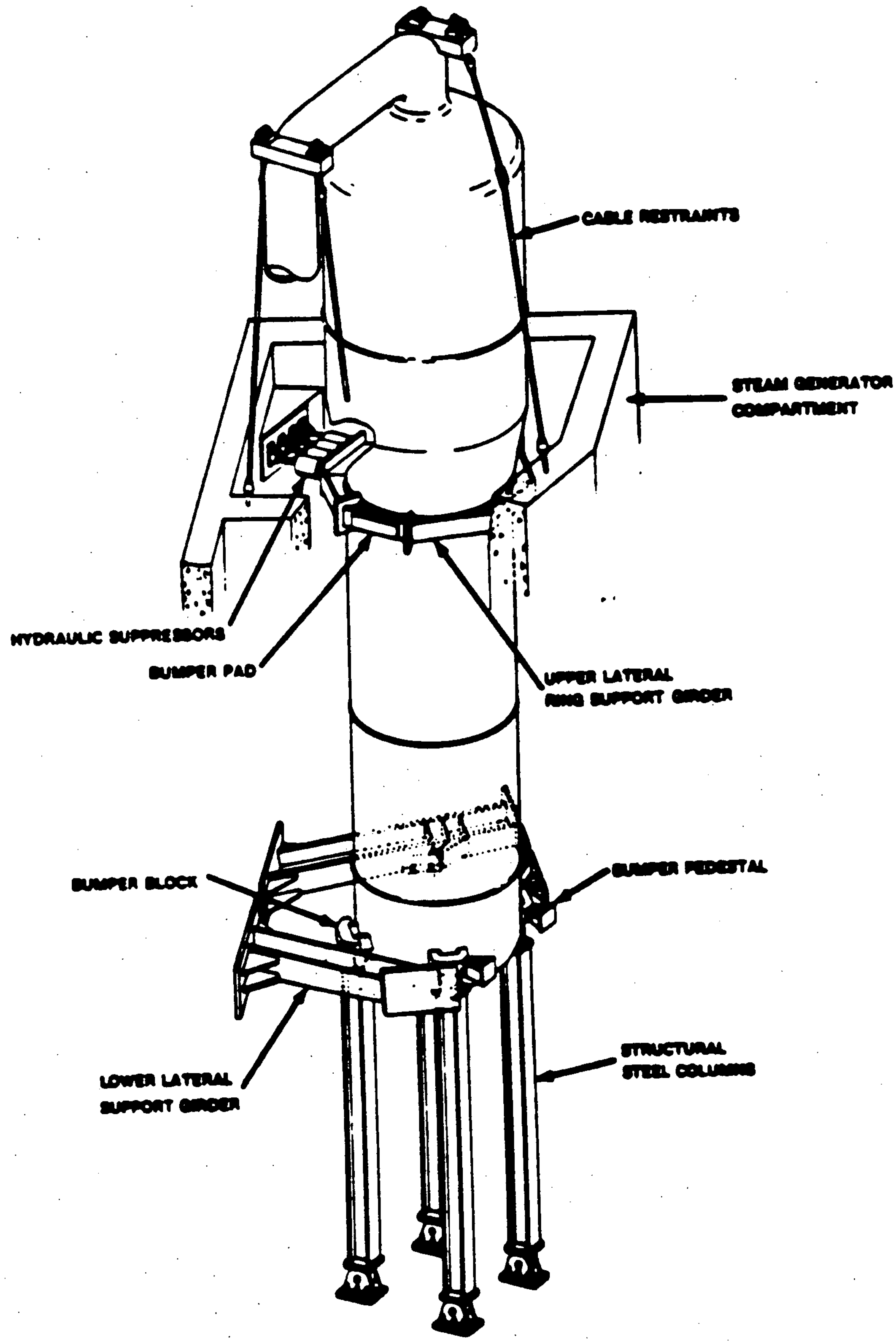

Figure 1

Existing Steam Generator Support System

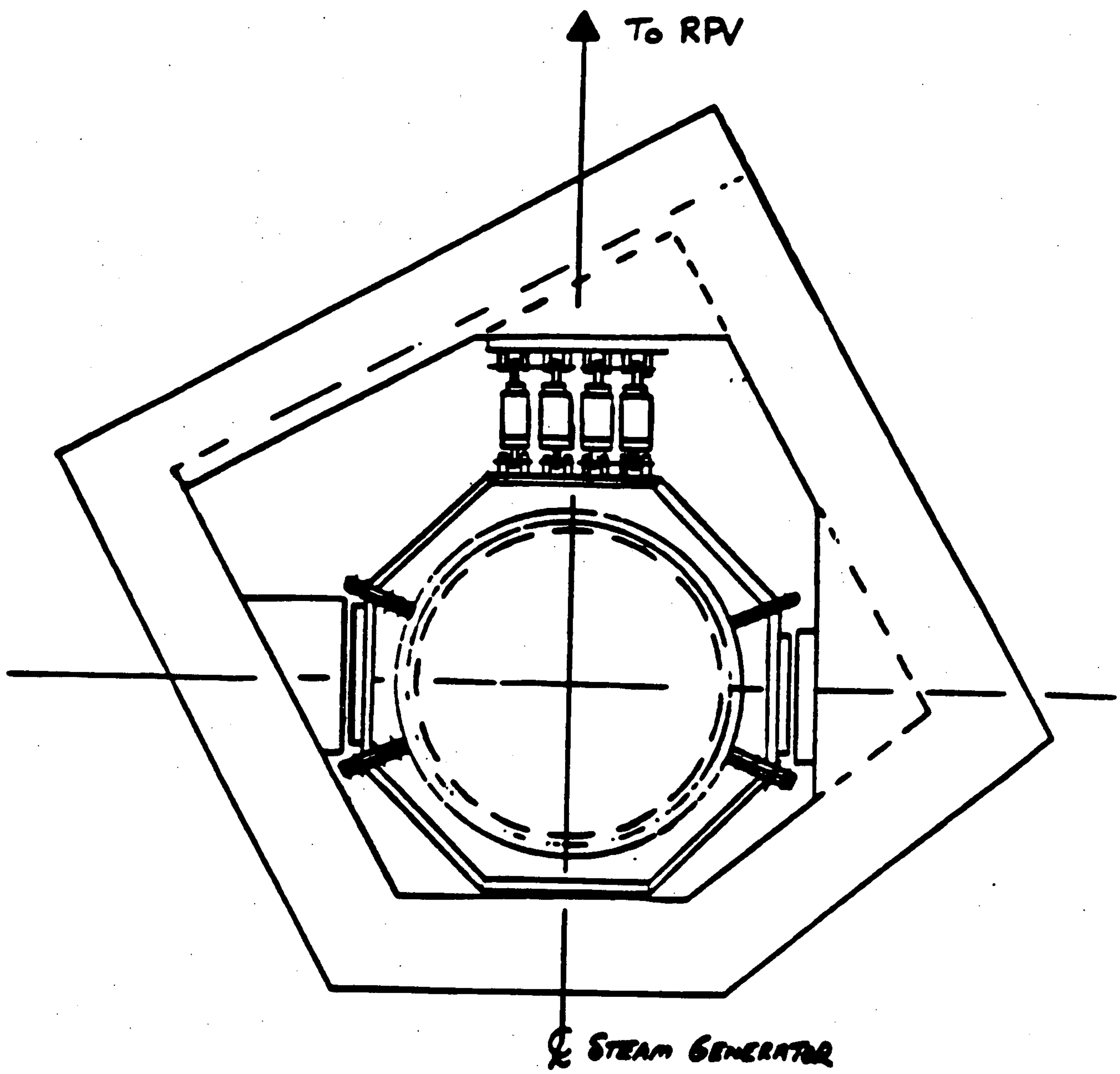

Figure 2

Existing Steam Generator Upper Lateral Support Snubber Arrangement

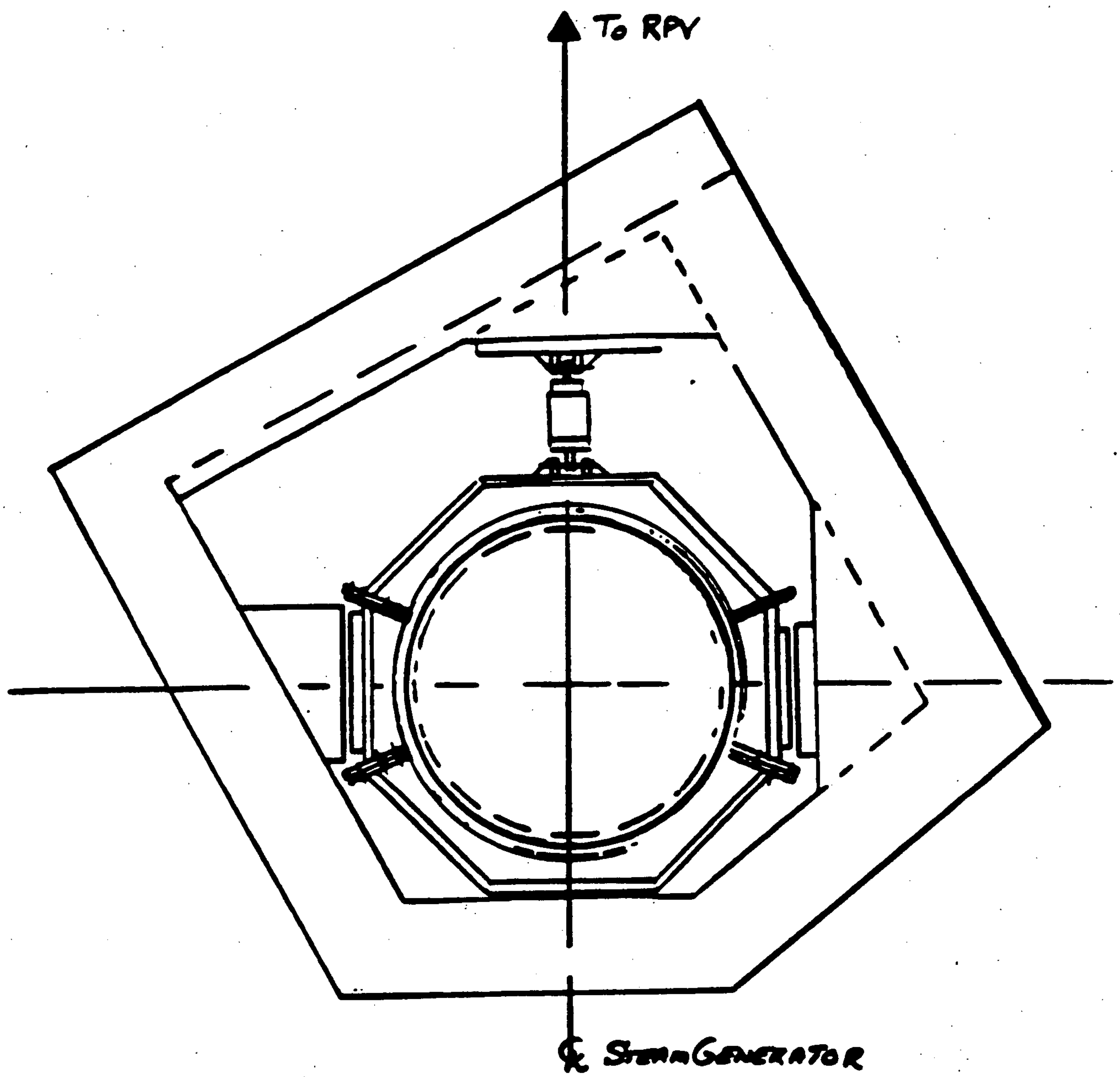

Figure 3

Modified Steam Generator Upper Lateral Support Snubber Arrangement

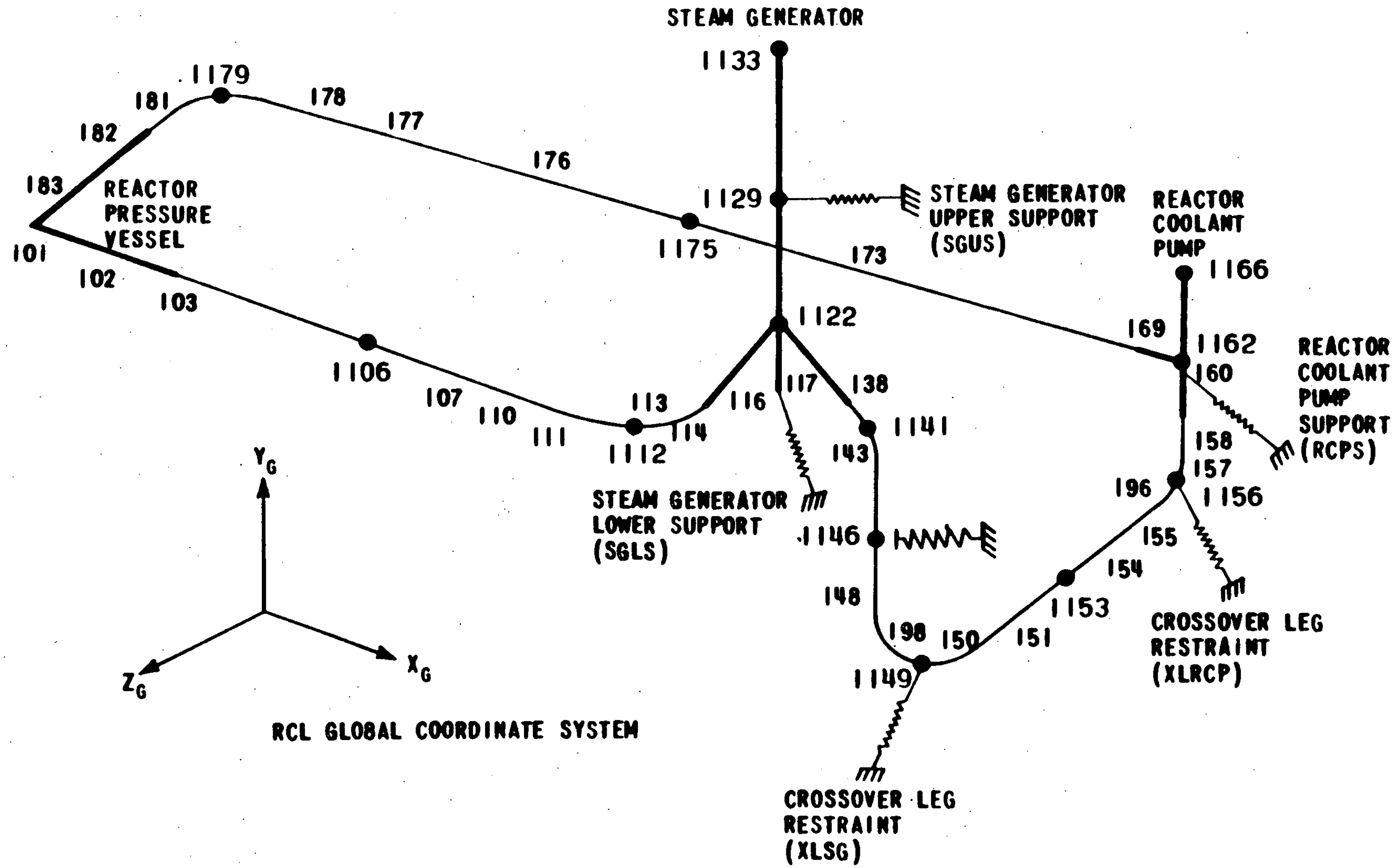

Figure 4
Reactor Coolant Loop Model