ATTACHMENT 1

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY UNITS 1 AND 2

REACTOR COOLANT SYSTEM LOADS

AND

COMPONENT SUPPORT MARGINS

EVALUATION

FOR

ELIMINATION OF REACTOR COOLANT SYSTEM

MAIN LOOP PIPE BREAK PROTECTIVE DEVICES

NOVEMBER 1985

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I. INTRODUCTION

This report is submitted in support of Virginia Electric and Power Company's request for partial exemption to General Design Criterion 4 (GDC-4) as applicable to Surry Units 1 and 2, to the extent that protection against the dynamic effects of postulated pipe rupture on primary system components/supports and piping may be eliminated. The scope of the request would allow elimination of 18 large bore snubbers on the reactor coolant system (RCS) which are required only for pipe rupture loadings. The technical basis for the exemption request is based upon the fracture mechanics analyses referred to as "leak-before-break." The purpose of this report is not to resubmit materials which have been previously reviewed by the NRC staff, but rather to demonstrate that elimination of the 18 large bore snubbers does not compromise the following:

1. The loadings on the primary loop piping are still enveloped by the generic analyses submitted by Westinghouse on behalf of the Unresolved Safety Issue (USI) A-2 Owners Group, and accepted by the NRC staff, as documented in NRC Generic Letter 84-04; and

2. The reactor coolant system equipment, piping, and supports continue to have acceptable margins of safety under licensed loading conditions other than the now eliminated RCS main loop rupture.

II. BACKGROUND

The primary loop piping of Pressurized Water Reactors (PWRs) is highly reliable and for Westinghouse plants (including Surry Units 1 and 2), there is no history of cracking failure. The Westinghouse RCS primary loop has an operating history which demonstrates its inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, or fatigue (for both low and high cycle). This operating history totals over 400 reactor years, including five plants each having 15 years of operation and 15 other plants (including Surry Units 1 and 2) with over 10 years of operation.

The application of the leak-before-break approach to prevent ruptures of the primary coolant loop piping eliminates the requirement to design for the extreme loads associated with these previously postulated pipe rupture events. This provides the opportunity to eliminate selected primary component support snubbers which principally carry pipe rupture loads.

Large bore snubbers, being active components, require periodic removal for functional testing and implementation of a seal service life program. Removal/inspection activities of large bore snubbers have exposed maintenance personnel to high radiation because the snubbers are located in the reactor containment cubicles. The deletion of these snubbers will eliminate this source of occupational exposure and facilitate maintenance and in-service inspections of piping and components by reducing plant congestion.

Support system reliability is also increased with the removal of these active elements. Inadvertent lockup, bleed rate variance and hydraulic fluid leakage are possible problems related to larger bore snubbers that are eliminated.

The functions of the primary loop support snubbers have been reviewed to determine which may be eliminated. The objective of this review was to maximize snubber elimination allowed by the elimination of dynamic effects of main reactor coolant loop breaks while having minimal effect on the design margins for other loads. Except for the application of "leak-before-break" for elimination of dynamic effects of main reactor coolant loop breaks and the combination of remaining pipe rupture loads with seismic loads using SRSS combination, other licensing basis requirements are maintained.

All snubbers on the Steam Generator (SG) lower supports and the Reactor Coolant Pump (RCP) supports may be eliminated by use of "leak-before-break." It is clear from the orientation of these snubbers, as shown in Figures 1 and 2, that these snubbers were introduced into the design to carry pipe rupture loads. Their stiffness is insignificant when compared to that of the unbroken loop piping. The snubbers to be removed are noted in Figures 1 and 2, as described below:

- Both large bore (i.e. 12"-bore) Bergen-Paterson snubbers, acting parallel to the cold leg, in each of the three RCP supports,

- The four large bore (i.e. 12"-bore) Pathon snubbers, acting parallel to the hot leg, and in each of the three SG lower supports, and

- The four small bore (i.e. 4"-bore) Bergen-Paterson snubbers which act as upper diagonal braces of each of the three RCP supports. However, it has been determined that these snubbers could be eliminated even within the current design basis of the RCS primary loop ruptures.

The net effect is the elimination of six of ten large bore snubbers in each of the three reactor coolant loops (i.e. a total of eighteen large bore snubbers per unit) and four small bore snubbers on the diagonals of each RC Pump support (i.e. a total of twelve small bore snubbers per unit).

The steam generator moves out from the reactor vessel several inches due to thermal growth of the reactor coolant loop. The steam generator is allowed to move in the direction radial from the reactor, and is guided by the upper and lower steam generator supports. Seismic support in the direction radial to the reactor is currently provided at the upper support by four large bore (i.e. 12"-bore) Pathon snubbers for each of the three loops. These four snubbers will be maintained as is, while the remaining large bore snubbers are to be eliminated.

This loading evaluation with revised support configuration establishes that the piping components and supports are stressed within UFSAR acceptable limits. Adequate safety margins exist in a seismic event and the maximum moment in the reactor coolant loop piping is within the envelope moment taken as a limit in the safety evaluation provided in Generic Letter 84-04.

III. ANALYSIS

A. Mathematical Models

Two essentially independent analyses of a representative single primary RC loop were performed for this work. Westinghouse Electric Corporation performed analyses using the model of Figure 3 as the run of record to obtain piping stresses. Stone & Webster Engineering Corporation (SWEC) performed analyses using the model of Figure 4 principally to obtain component support loads; therefore, the model incorporates a detailed representation of the support members. This division of analytical responsibility between the two organizations is similar to the original division of design responsibility. Both analytical models were revisions to existing models and incorporated changes due to earlier steam generator replacement efforts and the proposed snubber removal.

B. Loading Conditions

The following loading conditions were analyzed for the revised support configuration:

- Deadweight,
- Thermal expansion,
- Internal pressure,
- Seismic events (OBE and DBE), and

- Dynamic effects of pipe ruptures of other systems as specified in the UFSAR (pressurizer surge, main steam, and feedwater lines).

No other hydraulic transient loading was considered as significant.

For seismic analysis, the 1979 soil structure interaction ARS for 0.5 percent equipment damping (OBE) and 1 percent equipment damping (DBE) were used with appropriate bump factors (UFSAR Section 15A.3.4.3, Reference 1). The vertical and horizontal earthquake responses were combined for piping analysis, as described in UFSAR Section 15A.3.2. For the component support analysis, the responses to the three directions of earthquake loading were combined by SRSS. The combination of closely spaced modes conformed to NRC Regulatory Guide 1.92 (Reference 4).

C. Codes and Standards

The following Codes and Standards were utilized in the analysis:

- Power piping, USAS B31.1 (Reference 5). This is the original Code of record with which the plant was constructed.

- Updated Final Safety Analysis Report (UFSAR), Surry Power Station Units 1 and 2, Virginia Electric and Power Company. Allowable stresses currently documented were used for qualification.

- ASME Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components (Reference 6), was used for the design and construction of the equipment.

D. Computer Programs

The Westinghouse analysis used the WESTDYN computer code (Reference 2) and a simplified representation of the component supports as stiffness matrices. The WESTDYN computer code has been utilized on numerous Westinghouse plants and was reviewed and found acceptable by the NRC for the Surry Units in 1974. The component support stiffness matrices were supplied by SWEC and the computed values were essentially identical to matrices used in earlier analyses, except for stiffnesses representing those eliminated snubbers.

The SWEC analysis used the STARDYNE computer code (Reference 3), and a model incorporating a detailed representation of the supports. STARDYNE is a public domain computer program and is recognized as a Category 1 computer program suitable for nuclear work. The following modules of STARDYNE, Version 3, Level H, were used:

- STAR (Static and Model Extraction)
- DYNRE4 (Seismic Response Spectrum)
- DYNRE6 (Time History Transient Analysis) - only used for evaluating pipe rupture loadings

This program is maintained and monitored with SWEC's Quality Control procedures with respect to any program errors which are encountered through industry usage.

IV. RESULTS AND DISCUSSIONS

A. Stress in Piping

The level of stress (percentage) compared to the Code allowable at the highest stress point in each leg of the RC loop for thermal, deadweight, and seismic conditions are given by:

TABLE 1

Level of Stress as a Percentage of Code Allowable Stress

Loading	Hot Leg	Crossover Leg	Cold Leg	Code Allowable Stress
Thermal	38.6%	15.6%	7.4%	S_A
Pressure + Deadweight	68.0%	48.7%	53.3%	$1.0S_h$*
Pressure + Deadweight + OBE	65.6%	65.0%	60.0%	$1.2S_h$*
Pressure + Deadweight + DBE	49.6%	51.1%	48.9%	$1.8\ S_h$*

*S_h = 15 Ksi

It is interesting to note that even with the elimination of the six large bore snubbers per loop, the maximum level of stress as a percentage of code allowable stress still occurs for the hot leg deadweight + pressure loading condition case.

B. Validation of Generic Fracture Mechanics Evaluation

The maximum resultant bending moment in the primary coolant loop piping is in the steam generator outlet nozzle/piping junction. The magnitude of this maximum moment, which results from the combination of deadweight, pressure, thermal, and design basis earthquake loadings is 28,860 in-kips. The corresponding maximum axial force at the same location is 1,685 kips. Both of these values are less than the enveloped values in the Westinghouse generic report, WCAP-9558, Revision 2 (Reference 8). More importantly, the maximum moment is also less than 42,000 in-kips allowed by the NRC in Generic Letter 84-04.

Component support loads generated from piping analysis were evaluated by SWEC and were found acceptable, as discussed below.

C. Component Support Evaluation

Using the analytical model of Figure 4, SWEC has evaluated the new support system configuration. The frequencies of most vibrational modes are virtually unchanged. The primary loop stress continues to be low as summarized in Table 1. The equipment support loads and stresses also continue to be low as discussed below.

The combination of deadweight, pressure, and seismic loads calculated for the modified support design are low and meet existing UFSAR and code allowables. The factors of safety (allowable load/combined load due to Deadweight + Pressure + Design Basis Earthquake), are shown in Table 2.

The effects of pipe ruptures in the pressurizer surge line, main stream line and feedwater lines have also been investigated. The originally-postulated terminal and intermediate breaks were reviewed by SWEC to determine those breaks which would cause the most severe loadings on the revised support configuration with snubbers removed. Time history forcing functions were applied to the analytical model of Figure 4 representing these potentially limiting breaks, to obtain maximum member loads with the revised support configuration. These loads were combined by SRSS with seismic DBE loads and then summed with deadweight and pressure loads. In all evaluated cases, the supports are within UFSAR and code allowables.

V. ADDITIONAL CONSERVATISM

The analyses performed are in accordance with the existing licensing basis, except for use of SRSS combination of pipe rupture loads with seismic loads. The factors of safety, quoted in Tables 1 and 2, are based upon criteria more conservative than the current industry practice. Additional conservatisms include:

- The use of low equipment damping (0.5 percent for OBE, 1 percent for DBE) compared to higher values recommended in Regulatory Guide 1.61 (Ref. 7). Still higher damping values of 2 percent to 5 percent are now permitted by ASME Code Case N411, but have not been used in this evaluation,

- Comparison of stresses to minimum code-specified material allowables at operating temperature (References 5 and 6), which already include a safety factor, and

- Comparisons based on elastic limits which are not a true indicator of failure.

VI. INDEPENDENT VERIFICATION

As discussed previously, two essentially independent analyses were performed for this work: a pipe stress analysis by Westinghouse and a component support analysis by SWEC. Both completely modeled a single primary loop. The results of both analyses at support-to-component interface points were reviewed and found to be in acceptable agreement.

The interfaces between Westinghouse and SWEC for this work have been carefully monitored. Interface details are provided in Figure 5.

VII. QUALITY ASSURANCE

Except for elimination of the consideration of dynamic effects due to pipe rupture of the primary RCS piping and SRSS combination of other pipe rupture loads with seismic loads, analyses are in conformance with the existing licensing basis (Reference 1), both with respect to design criteria and the control of the engineering process. The work has been independently reviewed as Category 1 calculations and meets Quality Assurance requirements. The results of the analyses are maintained in Project Document Control.

VIII. ENHANCEMENT OF RELIABILITY

NUREG/CR-3718, "Reliability Analysis of Stiff versus Flexible Piping - Status Report" (Reference 10), established that piping designs using snubbers as support devices may not exhibit the intended reliability because the snubbers may fail to perform the desired function. Inadvertent lock-up, bleed rate variance and hydraulic fluid leakage are a few of the many problems experienced by the nuclear industry with regard to large bore snubbers. It was further demonstrated in the NUREG/CR-3718 that certain piping systems with snubbers removed actually exhibit higher reliability than do those of the original design. The large bore snubbers proposed for elimination here are parallel to both the cold leg and the hot leg of the reactor coolant loop piping. Inadvertent lock-up of these can induce high thermal stresses during normal plant operation.

The revised support configuration will eliminate snubbers in high radiation areas and the more inaccessible areas. The large bore snubbers to be retained in the main reactor coolant system will be in low radiation areas, and more accessible areas; and, therefore can be maintained easier so as to increase their reliability. The snubbers retained can be equipped with individual reservoirs, seals with longer service life, self-flushing control valve, and test-in-place capability. Modifications of this type have already been made for Surry Unit 2 large bore snubbers, and are now under consideration for future long-term implementation for those Surry Unit 1 large bore snubbers not eliminated.

Therefore, the revised support configuration would result in an improved overall reliability in RCS support system.

IX. CONCLUSIONS

Based on the results of loading evaluation, the following concluding remarks can be made:

- Piping, components and supports are stressed within UFSAR allowable limits,
- Adequate safety margins exist and structural integrity will be maintained during seismic events, and
- The maximum moment in the reactor coolant main piping is within the envelope moment in the safety evaluation analysis provided in Generic Letter 84-04.

X. REFERENCES

1. Updated Final Safety Analysis Report (UFSAR), Surry Power Station Units 1 and 2, Virginia Electric and Power Company.

2. WESTDYN, Westinghouse Electric Corporation.

3. STARDYNE, Version 3, Level H, System Development Corporation, February 1, 1984.

4. Regulatory Guide 1.92, Rev. 1, Combining Modal Responses and Spatial Components in Seismic Response Analysis, U.S. Nuclear Regulatory Commission, February 1976.

5. USAS B31.1, Power Piping, American Society of Mechanical Engineers, 1967.

6. ASME Boiler and Pressure Vessel Code, Sections III, Nuclear Power Plant Components American Society of Mechanical Engineers.

7. Regulatory Guide 1.61, Damping Values for Seismic Design of Nuclear Power Plants, U.S. Nuclear Regulatory Commission, October 1973.

8. WCAP-9558, Rev. 2, Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack, Westinghouse Electric Corporation, May 1981.

9. USNRC Generic Letter 84-04, Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops, U.S. Nuclear Regulatory Commission, February 1, 1984.

10. Lu, S. C. and C. K. Chou. 1984. Reliability Analysis of Stiff Versus Flexible Piping-Status Report. NUREG/CR-3718, Lawrence Livermore National Laboratory, Livermore, California.

TABLE 2

FACTORS OF SAFETY FOR COMPONENT SUPPORTS UNDER SEISMIC LOADS

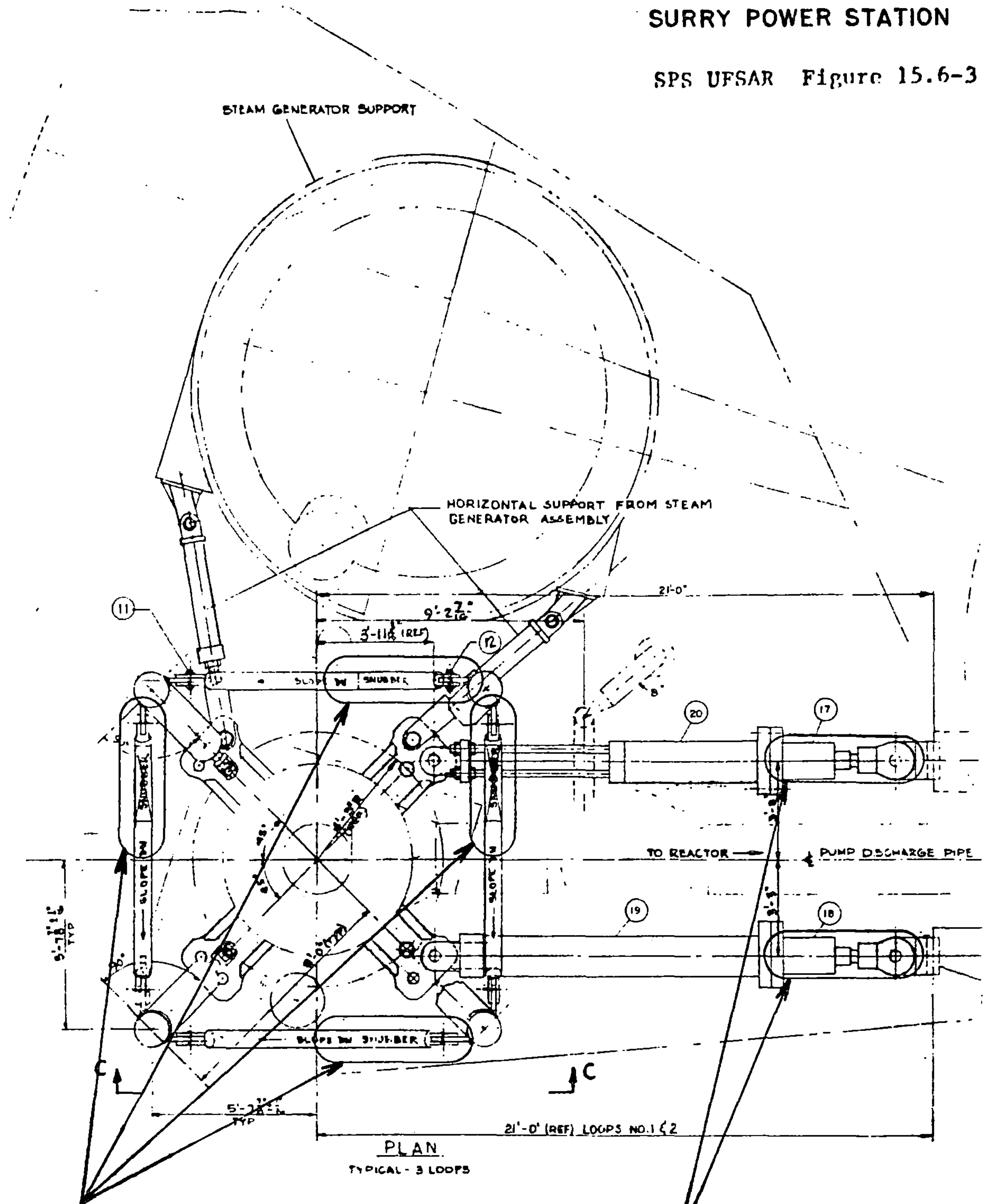
COMPONENT	FACTOR OF SAFETY *
Steam Generator Shell	40.7 (min.)
Steam Generator Upper Support	
Component	17.4
Upper Guides	7.3
Snubbers	14.2
Steam Generator Lower Support	
Hanger Rod	2.3
Swivel End Coupling	16.9
Steam Generator Foot	
Vertical Force	3.1
Tangential Force	15.7
RC Pump Foot	
Vertical Force	5.8
Tangential Force	20.6
Radial Force	22.1
RC Pump Support	
Upper Vertical	7.1
Upper Horizontal	7.1
Lower Vertical	5.3
Lower Diagonal	6.1

*Factor of Safety = Allowable Load/Total Load of Deadweight, Pressure and DBE.

REACTOR COOLANT PUMP SUPPORTS GENERAL ARRANGEMENT

SURRY POWER STATION

SPS UFSAR Figure 15.6-3



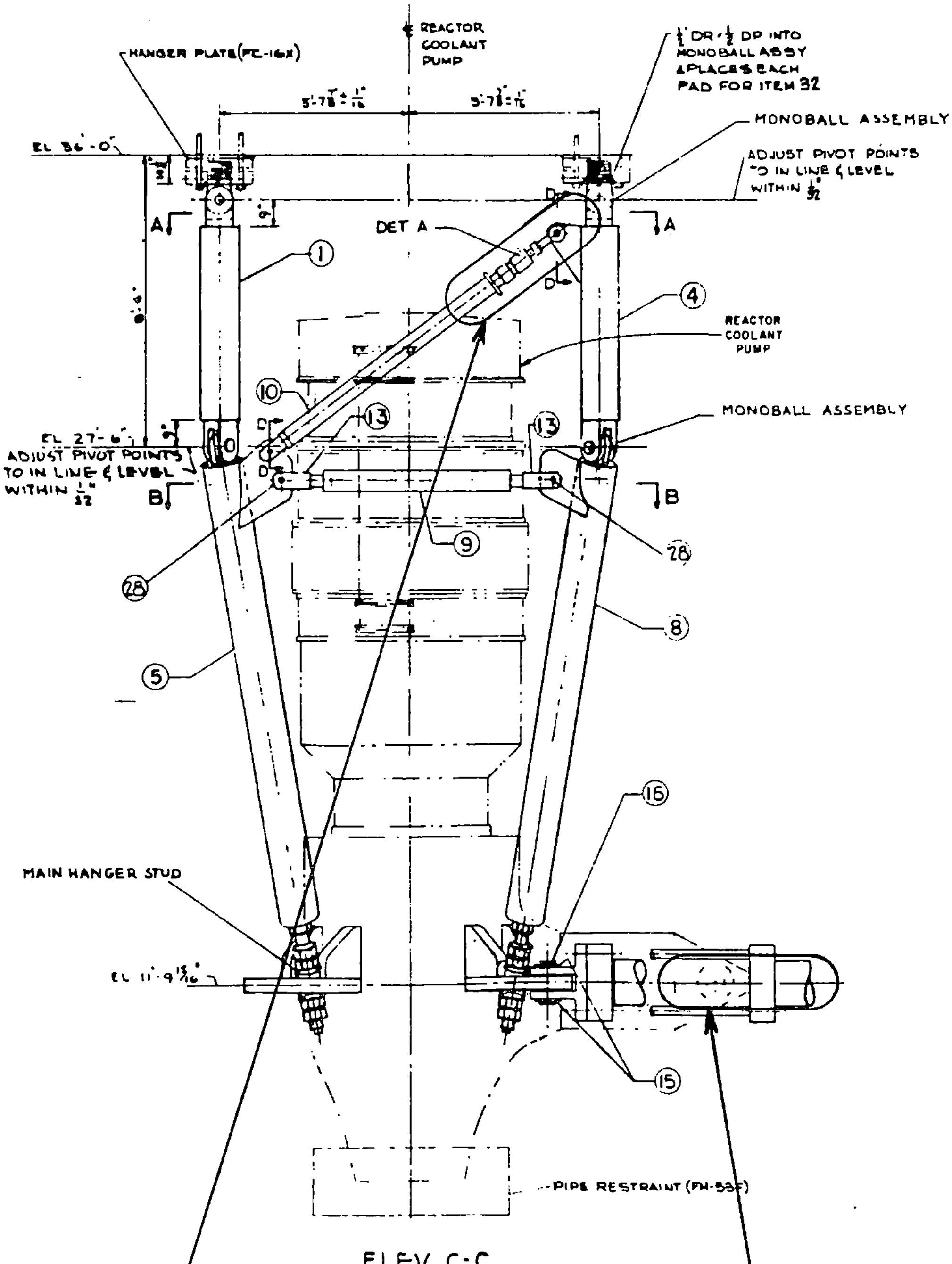
Four 4"-bore Bergen-Paterson snubbers to be eliminated

Two 12"-bore Bergen-Paterson snubbers to be eliminated

REACTOR COOLANT PUMP SUPPORTS GENERAL ARRANGEMENT

SURRY POWER STATION

SPS UFSAR Figure 15.6-3



Four 4"-bore Bergen-Paterson snubbers to be eliminated

Two 12"-bore Bergen-Paterson snubbers to be eliminated

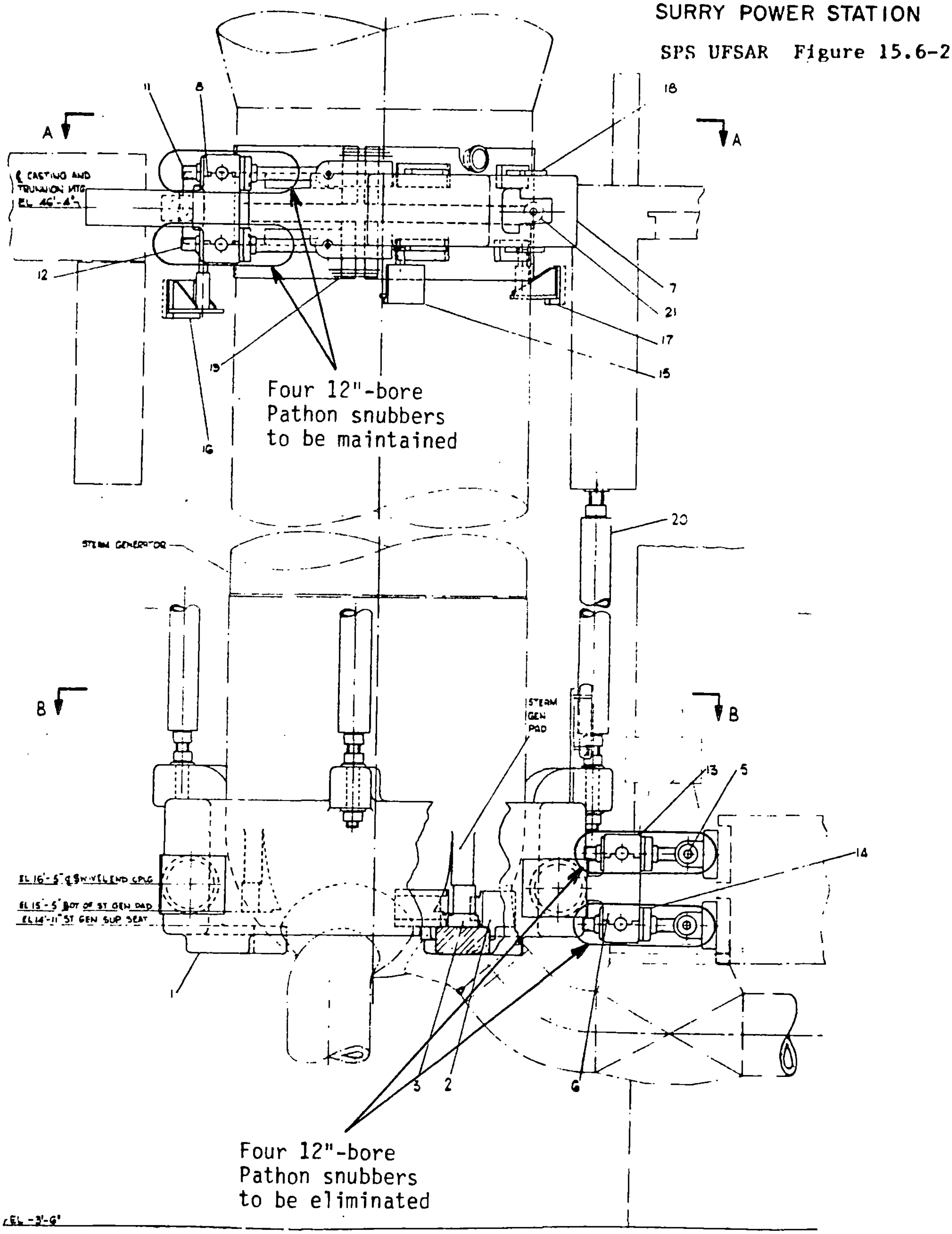
FIGURE 2

Page 1 of 2

STEAM GENERATOR SUPPORT ASSEMBLY

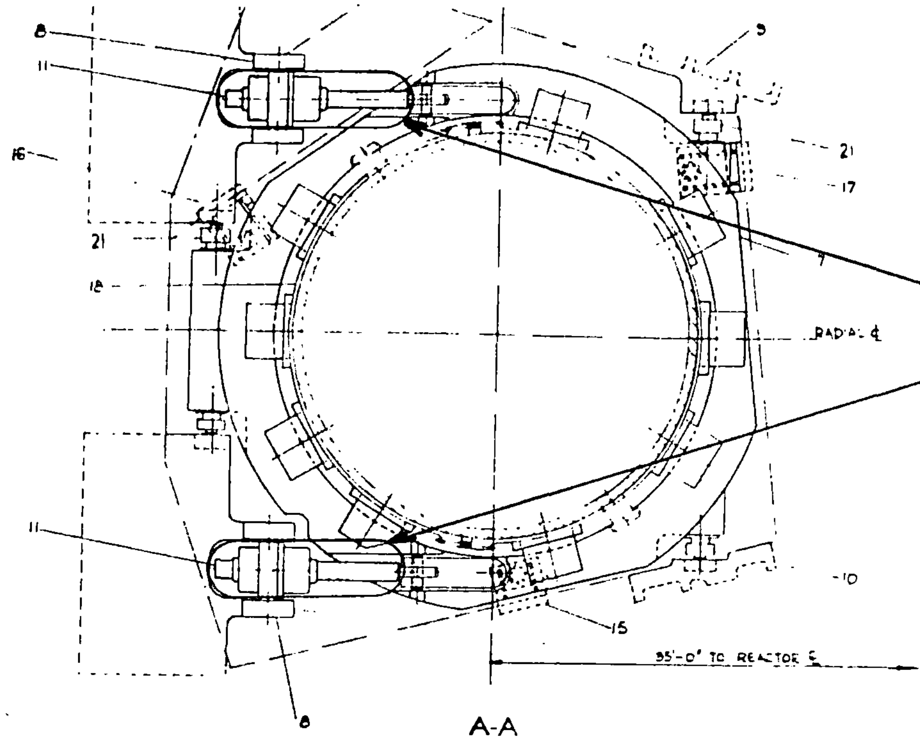
SURRY POWER STATION

SPS UFSAR Figure 15.6-2



ELEVATION

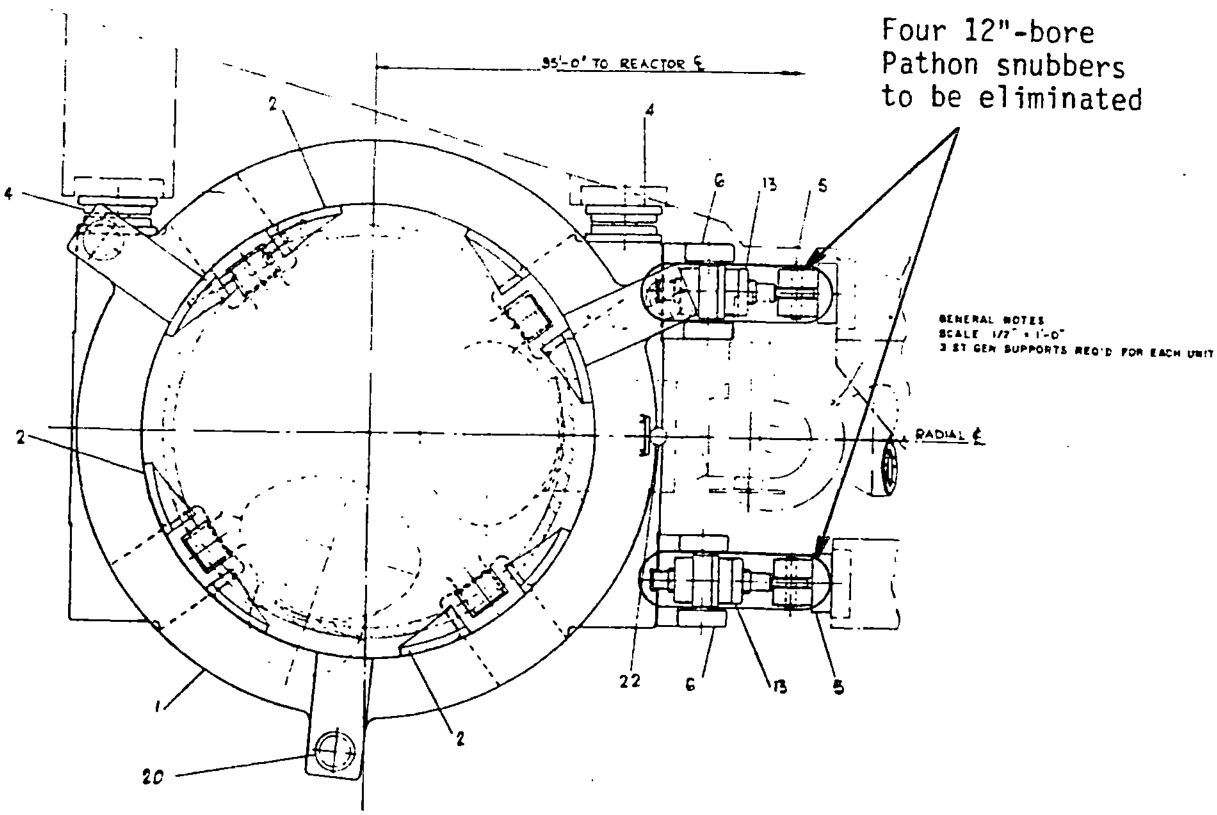
STEAM GENERATOR
SUPPORT ASSEMBLY
SURRY POWER STATION
SPS UFSAR Figure 15.6-2



Four 12"-bore
Pathon snubbers
to be maintained

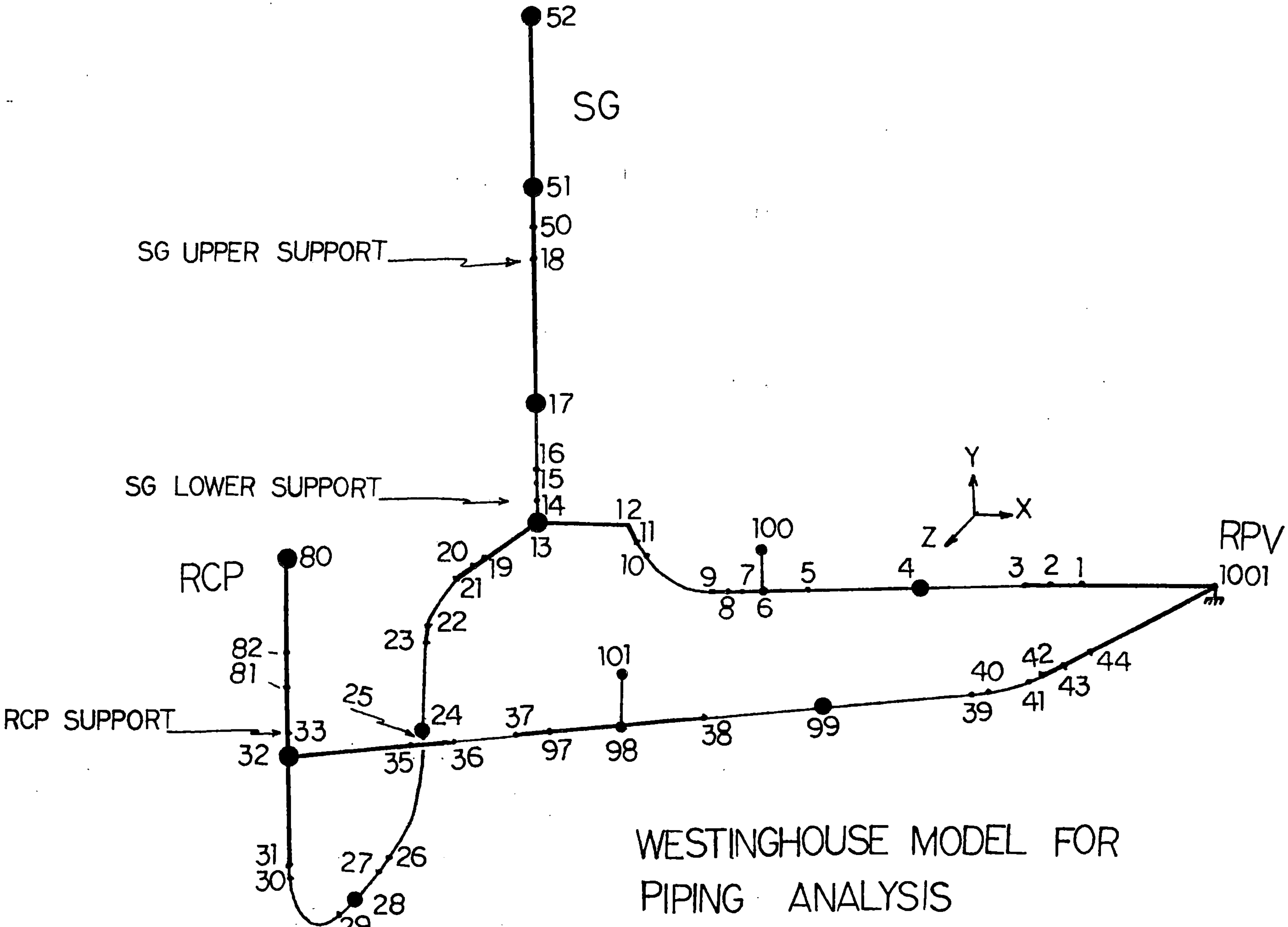
A-A

Four 12"-bore
Pathon snubbers
to be eliminated



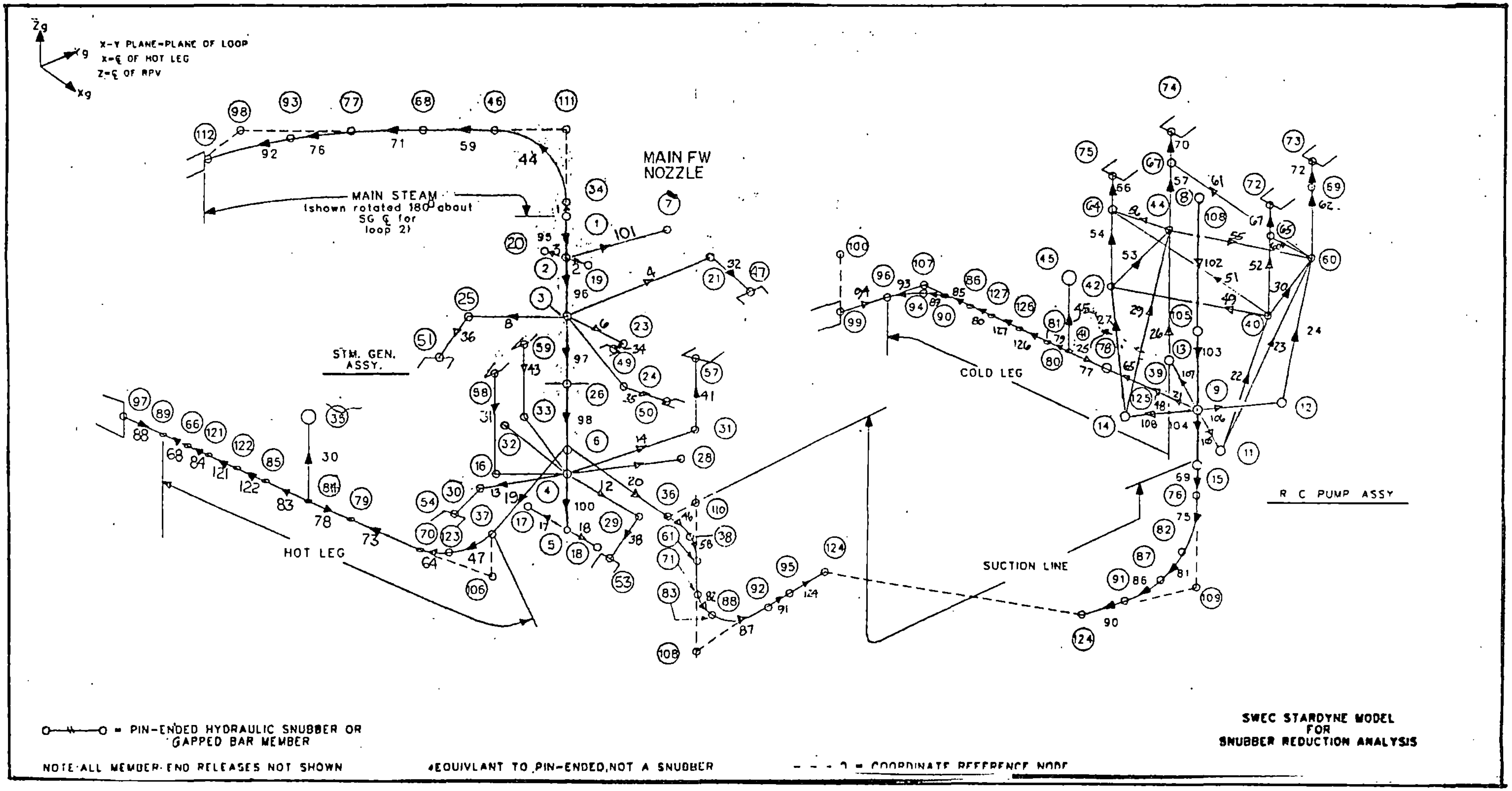
B-B

FIGURE 3



WESTINGHOUSE MODEL FOR PIPING ANALYSIS

FIGURE 4



Stone & Webster Model
for Piping/Structural Analysis

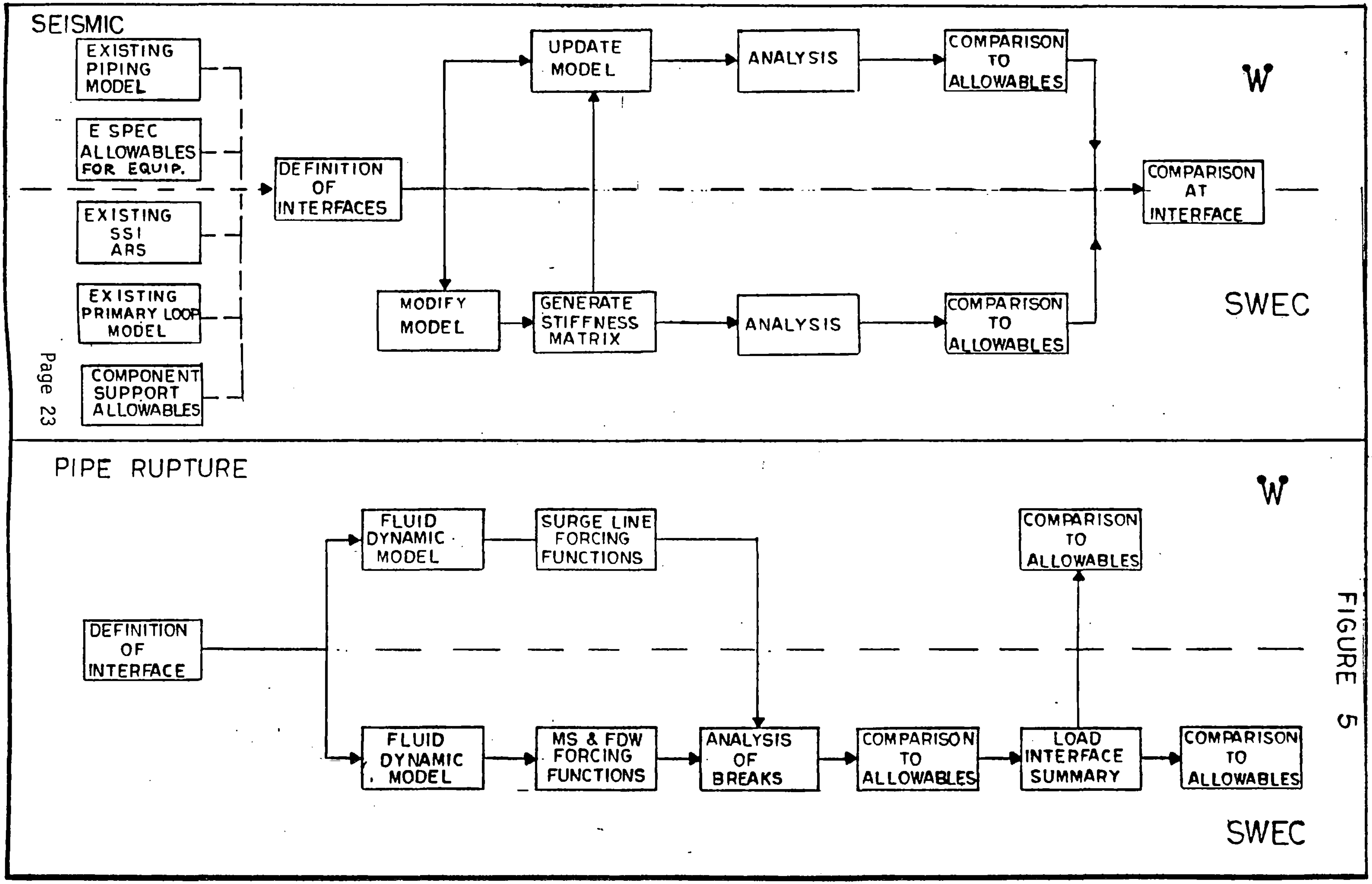

SEISMIC
EXISTING PIPING MODEL
E SPEC ALLOWABLES FOR EQUIP.
EXISTING SSI ARS
EXISTING PRIMARY LOOP MODEL
COMPONENT SUPPORT ALLOWABLES
DEFINITION OF INTERFACES
UPDATE MODEL
ANALYSIS
COMPARISON TO ALLOWABLES
W
COMPARISON AT INTERFACE
MODIFY MODEL
GENERATE STIFFNESS MATRIX
ANALYSIS
COMPARISON TO ALLOWABLES
SWEC
PIPE RUPTURE
W
FLUID DYNAMIC MODEL
SURGE LINE FORCING FUNCTIONS
COMPARISON TO ALLOWABLES
DEFINITION OF INTERFACE
FLUID DYNAMIC MODEL
MS & FDW FORCING FUNCTIONS
ANALYSIS OF BREAKS
COMPARISON TO ALLOWABLES
LOAD INTERFACE SUMMARY
COMPARISON TO ALLOWABLES
SWEC

FIGURE 5