DOE-HTGR-86-128
HFD - 41107
REVISION 1

HTGR

VESSEL SUPPORT SUBSYSTEM DESIGN DESCRIPTION

MASTER

AUTHORS/CONTRACTORS
COMBUSTION ENGINEERING, INC.

ISSUED BY: COMBUSTION ENGINEERING, INC.
FOR GA TECHNOLOGIES
UNDER GA CONTRACT SC-033390
JULY 1987

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CONTENTS

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

LIST OF APPENDICES

LIST OF ILLUSTRATIONS

LIST OF TABLES

LIST OF TABLES (Cont'd)

LIST OF ABBREVIATIONS AND ACRONYMS

AISC	American Institute for Steel Construction
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BSBSG	Buildings, Structures and Building Service Group
BSDD	Building and Structures Design Description
CFR	Code of Federal Regulations
DOE	Department of Energy
ECA	Energy Conversion Area
EFOH	Effective Forced Outage Hours
EG	Electrical Group
FHSS	Fuel Handling and Storage System
HRG	Heat Rejection Group
HTGR	High Temperature Gas-Cooled Reactor
HTS	Heat Transport System
HVAC	Heat/Ventilation/Air-Conditioning
IA	Integrated Approach
ICD	Interface Control Document
ISI	In-Service Inspection
MCIG	Miscellaneous Control and Instrumentation Group
MHTGR	Modular HTGR
NE	Nuclear Energy
NI	Nuclear Island
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OPDS	Overall Plant Design Specification
PAG	Protective Action Guidelines
PCG	Power Conversion Group
PCDIS	Plant Control, Data, and Instrumentation System
PPIS	Plant Protection & Instrumentation System
PRSS	Pressure Relief Subsystem

LIST OF ABBREVIATIONS AND ACRONYMS (continued)

PSG	Plant Service Group
QA	Quality Assurance
RB	Reactor Building
RCCS	Reactor Cavity Cooling System
RS	Reactor System
RSES	Reactor Service Equipment System
RSG	Reactor Service Group
RV	Reactor Vessel
SCHE	Shutdown Cooling Heat Exchanger
SCS	Shutdown Cooling System
SDD	System Design Description
SG	Steam Generator
SGV	Steam Generator Vessel
SSC	Structure, System, and Component
SSDD	Subsystem Design Description
SSE	Safe Shutdown Earthquake
TBD	To Be Determined
VDSS	Vessels and Duct Subsystem
VS	Vessel System
VSSS	Vessel Support Subsystem
10CFR20	Title 10 Code of Federal Regulations, Part 20
10CFR50	Title 10 Code of Federal Regulations, Part 50
10CFR100	Title 10 Code of Federal Regulations, Part 100

DEFINITIONS

Activity: An identifiable set of human actions undertaken to accomplish a particular task within one of the more general industry activities of Design, Construction, Operation, Maintenance, and Decommissioning.

Depressurized Operation: The primary coolant inventory is reduced to slightly subatmospheric (approximately -0.5 psig) pressure, e.g., for depressurized shutdown or refueling.

Design Criterion: A basis for judging the acceptability of a particular action, procedure, or design solution.

Design Value: The nominal value of a plant or system operating parameter plus an appropriate margin.

Energy Conversion Area: That portion of the plant not included within the Nuclear Island.

Equilibrium Plant: A commercial plant that incorporates a design that has matured through construction of replicate facilities and has accrued the benefits of standardization, learning, and optimal use of associated manufacturing facilities.

Equivalent Availability: Equivalent availability is mathematically defined as:

$$Ae = \int_{o}^{Tp} \frac{Pa(t)}{Tp}\, dt$$

where: $Pa(t)$ = available power expressed as a percentage of design power as a function of time (t)

t = time

Tp = time period (operational lifetime)

Function: A statement of what is to be achieved.

Functional Analysis: A systems engineering technique or method used to develop (1) functions required to meet established goals, (2) the relationship between functions and related requirements, and (3) the basis and justification for design selections through which the presecribed goals are met. [Functional analysis is one part of the Integrated Approach.]

Functional Requirement: A bounding quantification that is derived from a functional need versus being imposed by an institutional standard.

Functional Tree: A diagram which displays the interrelationships between functions. The MHTGR functional tree commences with the top level goal, then continues with subordinate goals and plant states.

Goal: An endpoint or accomplishment towards which an effort is directed. The Integrated Approach has identified the following four specific goals supporting the overall objective:

1. Safe, economical power (top level - Goal 0).
2. Maintain plant operation (Goal 1).
3. Maintain plant protection (Goal 2).
4. Maintain control of radionuclide release (Goal 3).
5. Maintain emergency preparedness (Goal 4).

Group: Systems which have different requirements but collectively have a common function.

Institution: A significant practice, relationship, or organization in a society. Institutions include the utility/user, the federal government, industry codes and standards, the Department of Energy, and state and local governments.

Institutional Requirement: A bounding quantification that is imposed by an institution as opposed to a quantification derived from a functional need.

Integrated Approach: A systems engineering technique for establishing and defending a well-developed nuclear plant design.

Interface Requirement: A requirement imposed by a system, subsystem, or component on another system, subsystem, or component, which must be accomplished to facilitate satisfaction of a function(s) of the imposing system, subsystem or component.

Nominal Value: Value of a quantity in name only, and thus not actual. Generally rounded for this usage.

Nuclear Island: That portion of the plant that has within its boundary the following:

1. The standard reactor modules and safety-related buildings, structures, systems, portions of systems, and components dedicated to assuring reactor shutdown, decay heat removal, fission product retention, and security of vital areas including new (unirradiated) fuel.

2. At the designer's discretion, nonsafety-related buildings, structures, systems, portions of systems, and components that support reactor operation or investment protection.

Operating Basis Earthquake (OBE): The earthquake which could reasonably be expectedto affect the plant site during the operating life of the plant.

Operating Life: The calendar time from receipt of the operating license to completion of plant power production.

Plant: All buildings, structures, systems, and components that together accomplishthe process of energy production and conversion and support the humanactivities of administration, operation, and maintenance.

Plant Life: The calendar time from construction permit issuance to completion of plant power production.

Parameter: A specific measurable and quantifiable aspect of a physical item.

Plant State: The condition of the physical plant (or module) at a particular time and place, as described by a set of appropriate variables associated with a plant goal.

Pressurized Operation: The primary coolant inventory (mass of helium) is held constant, equal to its rated-power value.

Primary Coolant: The total helium inventory with its impurities contained within the primary coolant pressure boundary is called primary coolant. Some of the primary coolant continuously flows to the Helium Purification System for purification and to the Moisture Monitor and Analytical Instrumentation Systems. The helium ceases to be primary coolant as soon as it leaves the primary coolant pressure boundary.

Primary Coolant Pressure Boundary: The Vessel System is the primary coolant pressure boundary and includes:

1) all steel vessels and the crossduct
2) all primary closures
3) the Pressure Relief Subsystem, up to and including the pressure relief valves.

Principal Design Criteria: Qualitative statements which specify the design commitments made to ensure that the dose criteria of 10CFR100 will be met, and, therefore, that public health and safety will be protected under accident conditions.

Rated Value: The nominal value of a parameter at 100% plant energy conditions.

Requirement: The bounding quantification (limits) of a function.

Safe Shutdown Earthquake (SSE): The earthquake for which those structures, systems and components required to meet 10CRF100 are designed to remain functional with a high degree of confidence.

Safety-Related: Identifier on equipment necessary to perform the functions required to limit releases under accident conditions to those allowed by 10CFR100.

Standard Reactor Module: That portion of the Nuclear Island which is duplicated with the addition of each reactor. This would include, in part, the reactor, steam generator, helium circulator, reactor building, reactor cavity cooling system, etc.

System: A collection of electrical, mechanical, and/or structural components and associated software and human actions treated as a unit for technical, administrative, or contractual purposes.

PREFACE

The objective of the MHTGR plant is to produce safe, economical power. Supporting this objective, four major goals and their associated plant states are identified as follows:

1. Maintain Safe Plant Operation

   1.1 Maintain Safe Energy Production (State 1)
   1.2 Maintain Safe Plant Shutdown (State 2)
   1.3 Maintain Safe Plant Refueling (State 3)
   1.4 Maintain Safe Plant Startup/Shutdown (State 4)

2. Maintain Plant Protection (in the event that plant operation cannot be maintained in the normal operating envelope)

   2.1 Protect the capability to maintain safe energy production
   2.2 Protect the capability to maintain safe plant shutdown
   2.3 Protect the capability to maintain safe plant refueling
   2.4 Protect the capability to maintain safe plant startup/shutdown

3. Maintain Control of Radionuclide Release (in the low probability event of failure to maintain plant protection)

   3.1 Control radiation
   3.2 Control personnel access

4. Maintain Emergency Preparedness (in the extremely low probability of failure to maintain control of release of radionuclides)

The Overall Plant Design Specification (OPDS) is the top-level technical document for the MHTGR plant. The OPDS (based on utility/user and regulatory requirements) establishes the overall performance, functional, institutional, operational, safety, maintenance, inspection, and decommissioning requirements for design of the plant.

In response to the OPDS, a series of lower tier documents, System Design Descriptions (SDDs), Subsystem Design Descriptions (SSDDs), Buildings and Structures Design Descriptions (BSDDs), Component Design Specifications (CDSs), and Interface Control Documents (ICDs), describe and control the individual designs. Traceability of requirement source from plant-level requirements to equipment-level requirements shall be maintained throughout this hierarchy of design documents.

SUMMARY

The Vessel Support Subsystem is one of three subsystems comprising the Vessel System of the Modular High Temperature Gas-Cooled Reactor 4 X 350 MW(t) Plant. The design of this subsystem has been developed by means of the Integrated Approach.

This document establishes the functions and system design requirements of the Vessel Support Subsystem from the Functional Analysis, and includes institutional requirements from the Overall Plant Design Specification and the Vessel System Design Description. A description of the subsystem design which satisfies these requirements is presented. Lower-tier requirements at the subsystem level are next defined for the component design. This document also includes information on aspects of subsystem construction, operation, maintenance, and decommissioning.

The principal functions of the Vessel Support Subsystem is to support the reactor vessel and the steam generator vessel.

The scope of the Vessel Support Subsystem, as shown in Figures 1-1 and 1-2, includes the following:

- The reactor vessel support columns (3) which support the reactor vessel and its contents while accommodating the radial and axial thermal expansions of the reactor vessel.

- The steam generator vessel sliding pad assemblies (2) which provide the load bearing support to the steam generator vessel and its contents while accommodating the sliding motion along the axis of the crossduct due to the thermal growth of the Vessels and Duct Subsystem.

- The steam generator vessel snubber assemblies (2) which provide the seismic restraint in the direction of sliding (along the axis of the

crossduct) while accommodating the vertical thermal growth, and thermal growth and displacement in the direction of sliding.

The physical envelope of the Vessel Support Subsystem is defined by its connection to the Vessels and Duct Subsystem at the reactor vessel, steam generator and crossduct and by its connection to or interface with the building structure.

The reactor vessel-to-steam generator vessel crossduct is supported solely through its connection to the vessels. The control rod housings are anchored to the top of the RV.

The design life of the Vessel Support Subsystem is 40 years with a goal of no unscheduled downtime. Components of the VSSS can be repaired, or replaced, if necessary.

SECTION 1

SUBSYSTEM FUNCTIONS AND DESIGN REQUIREMENTS

1.1 SUBSYSTEM FUNCTIONS

The Vessel Support Subsystem functions to support the Reactor Vessel, and the Steam Generator Vessel. Support is provided for deadweight, seismic, thermal, mechanical, and vibrational loads.

The VSSS supports the radionuclide control functions of the VS as follows: The VSSS indirectly controls heat generation and assists in removing core heat by controlling the geometry of the core with respect to the reactor vessel and the geometry of the reactor vessel with respect to the RCCS. The VSSS indirectly prevents chemical attack by maintaining the relative geometry of the VDSS components and thereby avoids breaches in the primary pressure boundary.

1.2 VESSEL SUPPORT SUBSYSTEM DESIGN REQUIREMENTS

1.2.1 Subsystem Configuration and Essential Features Requirements

The steel vessel which houses the steam generator shall be located to the side and below the vessel which houses the reactor. (1107.0102.001)*

The VSSS configuration shall accommodate space envelopes of the Vessels and Duct Subsystem and the Reactor Cavity Cooling System. (1107.0102.002)

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* Requirement traceability number, see Appendix A.

Space shall be provided by the Vessel Support Subsystem to allow installation, removal, ISI, and on-line and in-situ maintenance for the main circulator, steam generator, shutdown cooling circulator, and shutdown cooling heat exchanger. The dimensions of the required space are listed in Table 1-1.
(1107.0102.003)

Access shall be provided to the reactor coolant pressure boundary to facilitate in-service inspection as required by Section XI of the ASME Code.
(1107.0102.004)

The reactor vessel supports shall not extend below the shielding in the reactor cavity -28 meters (-92 feet). [The centerline of the crossduct vessel is -22 meters (-72 feet).] (1107.0102.005)

The VSSS sliding pad assemblies of the SG vessel shall not interfere with the steam nozzle and the steam pipe. (1107.0102.006)

The VSSS snubber assemblies shall be oriented as shown in Figure 2-2 so that seismic loads shall be transmitted to the central load bearing wall of the cavity. (1107.0102.007)

The VSSS shall accommodate radiative/convective heat transfer between the RV and RCCS with a minimum obstruction. (1107.0102.008)

1.2.2 Operational Requirements

The VSSS shall be designed for an operating life of 40 calendar years.
(1107.0102.011)

The design temperatures for the VSSS shall be:

Reactor Vessel Support Column - 288°C (550°F)
Steam Generator Vessel Sliding Pad Assembly - 121°C (250°F)
Steam Generator Vessel Snubber - 66°C (150°F) (1107.0102.012)

TABLE 1-1

SPACE REQUIRED FOR INSTALLATION, REMOVAL, ISI, AND ON-LINE AND IN-SITU MAINTENANCE FOR SUBSYSTEMS

Component	Component Dimensions m/ft	Supporting Operations Dimensions m/ft
Main Circulator		
Diameter	2.06/6.75	[TBD]
Steam Generator		
Diameter	4.37/14.3	[TBD]
Shutdown Cooling Heat Exchanger		
Diameter	1.98/6.5	[TBD]
Shutdown Cooling Circulator		
Diameter	1.08/3.54	[TBD]
Reactor Internals Components		
Diameter	6.55/21.5	[TBD]
Hot Duct Components		
Diameter	1.4/4.5	[TBD]
Length	2.8/9.2	[TBD]

The VSSS shall withstand the following radiation environment (total fluences):

Reactor Vessel Support Column (at top)	- 4.0×10^{15} n/cm^2
Steam Generator Vessel Sliding Pad Assembly	- $\ll 1 \times 10^{15}$ n/cm^2
Steam Generator Vessel Snubber	- $\lll 10^{15}$ n/cm^2

(1107.0102.013)

The VSSS shall be designed to operate through the design transients specified in Appendix C for the number of cycles specified in Table 1-2. (1107.0102.014)

1.2.3 Structural Requirements

The VSSS shall be designed to withstand the mechanical and thermal loads resulting from the design transients specified in Appendix C for the number of cycles specified in Table 1-2. (1107.0102.021)

The VSSS shall be designed to provide structural attachments and to support the weights, and seismic, thermal, torque, and vibrational loads imposed by components of interfacing systems given in Table 1-3. (1107.0102.022)

The VSSS in conjunction with the VDSS shall maintain alignment of RS, FHSS, RSES, HTS and SCS components within the tolerance given in Table 1-4. (1107.0102.023)

The VSSS shall withstand the effect of pipe rupture reactions and jet impingement loading as described in Reference 1-2. (1107.0102.024)

The VSSS shall be designed to withstand the seismic response spectra and the corresponding accelerations, deflections, and stresses given in Appendix D. (1107.0102.025)

The SSE load levels shall be twice the OBE values. (1107.0102.026)

The VSSS shall remain functional during and after an SSE. (1107.0102.027)

TABLE 1-2

DESIGN DUTY CYCLE EVENTS

Event	Design No. of Occurrences per Reactor Module)	Level of Service Limits[(a)]
1. Startup from refueling conditions	143	A
2. Startup with full helium inventory	312	A
3. Shutdown to refueling conditions	101	A
4. Shutdown with full helium inventory	105	A
5. Rapid load increase (5% per min) (25%-100%)	1,000	A
6. Normal load increase (0.5% per min) (255-100%)	20,800	A
7. Rapid load decrease (5% per min) (100%-25%)	1,000	A
8. Normal load decrease (0.5% per min) (100%-25%)	17,500	A
9. Step load increase (+15%)	1,000	A
10. Step load decrease (-15%)	1,000	A
11. Depressurized decay heat removal, HTS to SCS transition	80	A
12. Depressurized decay heat removal, SCS to HTS transition	122	A
13. Pressurized decay heat removal, HTS to SCS transition	61	A

TABLE 1-2...(CONTINUED)

DESIGN DUTY CYCLE EVENTS

Event	Design No. of Occurrences (per Reactor Module)	Level of Service Limits[a]
14. Pressurized decay heat removal, SCS to HTS transition	86	A
15. Circulator trip	30	B
16a. Reactor trip from 100%	180[b]	B
16b. Reactor trip from 25%		
17. Turbine trip or load rejection	120	B
18. Sudden reduction of feedwater flow	30	B
19. Steam generator tube leak (small)	9	B
20. Control rod insertion	5	B
21. Main loop overcooling	10	B
22. Earthquake	1	B
23. Slow primary system depressurization	8	B
24a. Rod withdrawal (normal rod speed) (power to flow ratio trip)	1	C
24b. Rod withdrawal (slow). (steam generator helium inlet temperature trip)	1	C
25. Failure of circulator speed control	1	C

Table 1-2 (continued)

DESIGN DUTY CYCLE EVENTS

Event	Design No. of Occurrences (per Reactor Module)	Level of Service Limits[a]
26. Circulator trip with helium shutoff valve failure	1	C
27. Steam generator tube rupture	1	C
28. SCS heat exchanger tube leak	1	C
29. Total loss of feedwater flow	4	C
30a. Total loss of SCS cooling water (HTS operating)	4	C
30b. Total loss of SCS cooling water (SCS operating)	1	C
31. Pressurized conduction cooldown	1	C
32. Main steam pipe rupture	1	D

(a) Level of service limits per ASME Boiler and Pressure Vessel Code 1986 addenda.

(b) For components where reactor trip from 100% load is worse, the breakdown should be 131 trips from 100% and 49 trips from 25%. For components where reactor trip from 25% load is worse, the breakdown should be 63 trips from 100% and 117 trips from 25%.

TABLE 1-3 INTERFACE LOADS BETWEEN THE VESSELS AND DUCT SUBSYSTEM AND VESSEL SUPPORT SUBSYSTEM

CONDITION	PLANE B LOCATION 1 **						PLANE B LOCATIONS 2 AND 3					
	FORCES (KIPS)						FORCES (KIPS)					
	Fx	Fy	Fz				Fx	Fy	Fz			
DEADWEIGHT	—	—	—				—	—	—			
TEMPERATURE NORMAL OP. + LEVEL A	—	—	—				—	—	—			
TEMPERATURE LEVEL B	—	—	—				—	—	—			
TEMPERATURE LEVEL C	—	—	—				—	—	—			
SEISMIC ONLY OBE	—	—	358.4				650.7	—	375.7			
SEISMIC ONLY SSE	—	—	716.8				1301.4	—	751.5			
APPLIED MECHANICAL LOAD	—	—	—				—	—	—			

** See Figure 1-1 for Definitions of Plane and Location

TABLE 1-3 .... CONTINUED

CONDITION	PLANE D LOCATION 1						PLANE D LOCATION 2					
	FORCES (KIPS)			MOMENTS (IN-KIPS)			FORCES (KIPS)			MOMENTS (IN-KIPS)		
	Fx	Fy	Fz	Mx	My	Mz	Fx	Fy	Fz	Mx	My	Mz
DEADWEIGHT	—	960.0	—	—	—	—	—	644.3	—	—	—	—
TEMPERATURE NORMAL OP. + LEVEL A	—	—	—	—	—	—	—	—	—	—	—	—
TEMPERATURE LEVEL B	—	—	—	—	—	—	—	—	—	—	—	—
TEMPERATURE LEVEL C	—	—	—	—	—	—	—	—	—	—	—	—
SEISMIC ONLY OBE	—	231.3	113.5	—	—	—	—	238.5	112.2	—	—	—
SEISMIC ONLY SSE	—	462.6	227.0	—	—	—	—	477.0	224.4	—	—	—
APPLIED MECHANICAL LOAD	—	*	—	—	—	—	—	*	—	—	—	—

* TBD; Conservative Estimate for Steam Pipe Break is 250 Kips

TABLE 1-3 .... CONTINUED

CONDITION	PLANE E LOCATIONS 2 AND 4					
	FORCES (KIPS)			MOMENTS (IN-KIPS)		
	Fx	Fy	Fz	Mx	My	Mz
DEADWEIGHT	—	—	—	—	—	—
TEMPERATURE NORMAL OP. + LEVEL A	—	—	—	—	—	—
TEMPERATURE LEVEL B	—	—	—	—	—	—
TEMPERATURE LEVEL C	—	—	—	—	—	—
SEISMIC ONLY OBE	147.9	—	—	—	—	—
SEISMIC ONLY SSE	295.8	—	—	—	—	—
APPLIED MECHANICAL LOAD	—	—	—	—	—	—

FIGURE 1-1

DEFINITION FOR INTERFACING LOADS

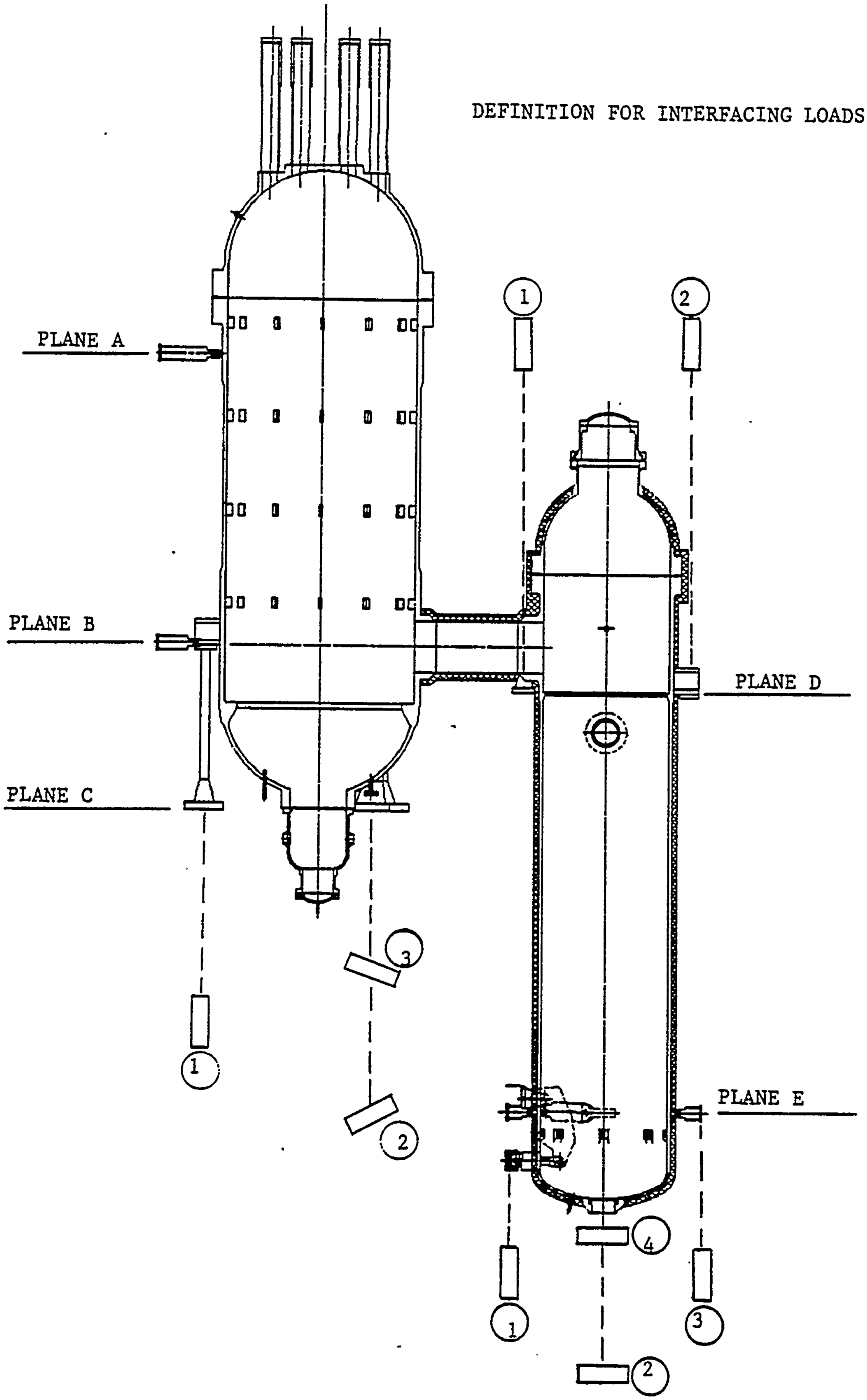

TABLE 1-4

COMPONENT ALIGNMENT TOLERANCES

Component	Tolerances
Neutron Control Assembly Housings	See drawings 029889 and 029894
In-Core Flux Monitors	See drawing 029782
Start-up Detectors	See drawing 029782
Shutdown Cooling Circulator	Concentricity of diffuser and circulator machine assembly at SCC attachment flange shall be within [2.5 mm (0.01 in)]
Core Refueling machine	See drawing 029894
Reactor Services Equipment Alignment for Circulator Handling Equipment Neutron Control Assembly Handling Equipment	[TBD]

The reactor vessel and the steam generator vessel shall be tangentially restrained. (1107.0102.028)

1.2.4 Environmental Requirements

The VSSS shall accommodate the environments in the reactor building as shown in Table 1-5. (1107.0102.031)

1.2.5 Instrumentation and Control Requirements

There are no instrumentation and control requirements.

1.2.6 Surveillance and In-Service Inspection Requirements

The VSSS equipment vendor shall furnish special in-service inspection (ISI) equipment which is not commercially available. (1107.0102.036)

Surveillance and ISI shall be performed on the VSSS in accordance with the Nuclear Island ISI/Surveillance Assessment (Ref. 1-3). (1107.0102.037)

1.2.7 Availability Assurance Requirements

The VSSS shall be designed to meet its overall reliability and equivalent forced outage allocations given in Table 1-6. (1107.0102.041)

Design modifications and improvements which enable the plant to exceed availability requirements shall be considered for incorporation into the design, if a one percentage increase in the total capital investment produces at a minimum, a seven-tenths percentage improvement in the equivalent availability factor. (1107.0102.042)

1.2.8 Maintenance Requirements

The VSSS shall meet an equivalent planned outage allotment of (TBD) h/yr. (1107.0102.046)

TABLE 1-5

ENVIRONMENTAL REQUIREMENTS FOR PORTIONS OF VESSEL SYSTEM EXPOSED TO THE REACTOR BUILDING

Location	Operating Conditions[a]	Air Temperature[b] °C (°F)		Air Relative Humidity %	
		Minimum Allowable Value	Maximum Allowable Value	Minimum Allowable Value	Maximum Allowable Value
Air surrounding the Vessel System side walls	Normal	[TBD]	[TBD]	[TBD]	[TBD]
	Abnormal	[TBD]	[TBD]	[TBD]	[TBD]

(a) The definition of the terms "normal" and "abnormal" can be found in the ASME Section III, Division 2 Code.

(b) Temperature is defined as bulk temperature of air at a given location.

TABLE 1-6 FROM REFERENCE 1-4

RELIABILITY ALLOCATIONS TO VESSEL SYSTEM

System/Subsystem or Feature	Mean Time to Failure (MTTF) of Operation	Probability of Failure to Start or Change State	Mean Time to Repair (MTTR)	Equivalent Forced Outage Hours (h/yr)	Other Description
Vessel System					
Vessels and Duct	103,000 h		69	5.00	
	$>10^8$ yr				Probability per year of leak greater than 84 cm^2 (13 in^2)
Pressure Relief		3×10^{-2}/demand failure to reclose			
		1×10^{-5}/demand failure to open			
Vessel Support					

1.2.9 Safety Requirements

The VSSS shall be designated as safety related and is relied upon to satisfy 10CFR100 dose limits. (1107.0102.051)

The VSSS shall assure that 10CFR100 radionuclide release limits are not exceeded for the Safety-Related Design Conditions in Table 1-7 by:

- o Maintaining the geometry of the reactor core and moveable poisons with respect to the reactor vessel to control heat generation.

- o Maintaining the geometry of the reactor vessel with respect to the RCCS during conduction cooldown. (1107.0102.052)

The VSSS shall be designed to meet the top-level regulatory criteria (Ref. 1-5). (1107.0102.053)

The VSSS shall be designed to meet the lower-level regulatory criteria contained in the regulatory guides identified for this safety-related subsystem Table 1-8. (1107.0102.054)

SSCs designated "safety-related" shall be designed to perform their safety function(s) for the Safety-Related Design Conditions (SRCSs) listed in Table 1-7. The transient design conditions for the SRDCs are included in Ref. 1-2. (1107.0102.055)

Compliance with the safety requirements shall be ensured by designing the VSSS to meet the reliability allocations for the Standard MHTGR (Reference 1-4) and the safety performances requirements for the Standard MHTGR (TBD). (1107.0102.056)

1.2.10 Codes and Standards Requirements

The VSSS design shall meet the industry codes and standards for design and construction of the VSSS and its components as listed in Table 1-9. (1107.0102.061)

TABLE 1-7

SAFETY-RELATED DESIGN CONDITIONS

Pressurized Conduction Cooldown (SRDC No. 1)
Pressurized Conduction Cooldown Without Control Rod Trip (SRDC No. 2)
Pressurized Conduction Cooldown With Control Rod Withdrawal (SRDC No. 3,4)
Pressurized Conduction Cooldown with Earthquake (SSE) (SRDC No. 5)
Depressurized Conduction Cooldown With Moderate Moisture Ingress (SRDC No. 6,7)
Depressurized Conduction Cooldown With Small Moisture Ingress (SRDC No. 8,9)
Depressurized Conduction Cooldown With Moderate Primary Coolant Leak (SRDC No. 10)
Depressurized Conduction Cooldown With Small Primary Coolant Leak (SRDC No. 11)

TABLE 1-8

REGULATORY REQUIREMENTS APPLICABLE TO THE VESSEL SUPPORT SUBSYSTEM

1) Reg. Guide 1.29, Seismic Design Classification

2) Reg. Guide 1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants

3) Reg. Guide 1.61, Damping Values for Seismic Design of Nuclear Power Plants

4) Reg. Guide 1.92, Combining Modal Responses and Spatial Components in Seismic Response Analysis

TABLE 1-9

INDUSTRIAL CODES AND STANDARDS APPLICABLE TO THE VESSEL SUPPORT SUBSYSTEM

A. ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers

1) Section II, Material Specifications
2) Section III, Nuclear Power Plant Components
3) Section V, Nondestructive Examination
4) Section IX, Welding and Brazing Qualifications
5) Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components

B. ANSI Standards, American National Standards Institute

1) ANSI/ASME-NQA-1, Quality Assurance Program Requirements for Nuclear Power Plants.
2) ANSI-Y14.5M, Dimensioning and Tolerancing.
3) ANSI B18.2.1-1981, Square and Hex Bolts and Screws, Including Askew Head Bolts, Hex Cap Screws and Lag Screws.
4) ANSI B1.1-1982, Unified Inch Screw Threads (UN and UNR Thread Form).

C. ANS Standards, American Nuclear Society

1) ANS-18-20, Nuclear Plant Reliability Data Collection and Reporting System.
2) ANS-58-4, Criteria for Technical Specifications for Nuclear Power Stations.

D. AISC Standards, American Institute for Steel Construction

1) AISC-M011, Manual of Steel Construction, Eighth Edition.

E. ASTM Standards, American Society for Testing and Materials

(TBD)

1.2.11 Quality Assurance Requirements

The VSSS shall come under a Quality Assurance Program which fully complies with the requirements of Title 10 Code of Federal Regulations Part 50 (10CFR50), Appendix B. The basic requirements and supplements of ANSI/ASME NQA-1 (as endorsed by Regulatory Guide 1.28, Revision 3) and the four additional supplements from DOE NE F2-10 regarding Management Assessment (NE 02-4.3.0), Engineering Holds (NE 03---1.3.2), Design Reviews (NE 03-1.3.4), and Engineering Drawings Lists (NE 03-1.3.5) shall be implemented on activities that affect the quality of such items. (1107.0102.066)

1.2.12 Construction Requirements

Special installation equipment not commercially available shall be provided by the equipment vendor. (1107.0102.071)

1.2.13 Decommissioning Requirements

Until more specific criteria and/or rules are developed, NUREG-0586, "Draft Generic Environmental Statement on Decommissioning of Nuclear Facilities," January, 1981, shall be used as guidance for anticipating NRC criteria concerning plant decommissioning. (1107.0102.076)

SECTION 2

SUBSYSTEM DESIGN DESCRIPTION

2.1 SUMMARY DESCRIPTION

The VSSS in conjunction with the VDSS provides overall support of the VS. The design concept (Ref. 2.1-2.2) for supporting the reactor vessel (RV) consists of three (3) support columns anchored on the RV at or below the level of the crossduct. Three keys are provided at the top of the reactor vessel and at the support lug elevation to accommodate vertical and radial thermal expansion while providing lateral seismic restraint. Figures 2-1 and 2-2 depict the Vessel Support Subsystem arrangement.

The steam generator vessel (SGV) load bearing support is slightly below the crossduct elevation and consists of two sliding bases, in-line with the crossduct vessel, supported by ledges from the steam generator cavity. These sliding bases also provide lateral seismic restraints. The crossduct side sliding base forms an integral part of the bottom of the SGV nozzle at the crossduct. The other sliding base is formed by a lug welded to the SGV. Pairs of keys and snubbers are provided near the bottom of the steam generator vessel to provide seismic restraint. These components accommodate radial and vertical expansions, translation (sliding) along the axis of the crossduct, and seismic excitations.

The crossduct vessel connecting the RV to the SGV is supported solely through its connections to the vessels.

This vessel support concept maintains the radial center of the reactor core stationary at all times. The SGV, due to a rigid crossduct connection with the RV, slides in-line with the crossduct at the various operating conditions.

The control rod drive housings are anchored at the top of the RV. The shutdown cooling heat exchanger and circulator are anchored at the bottom of the RV. The main circulator is anchored to the top of the SGV. These components are self-supported from their anchor points on the VDSS.

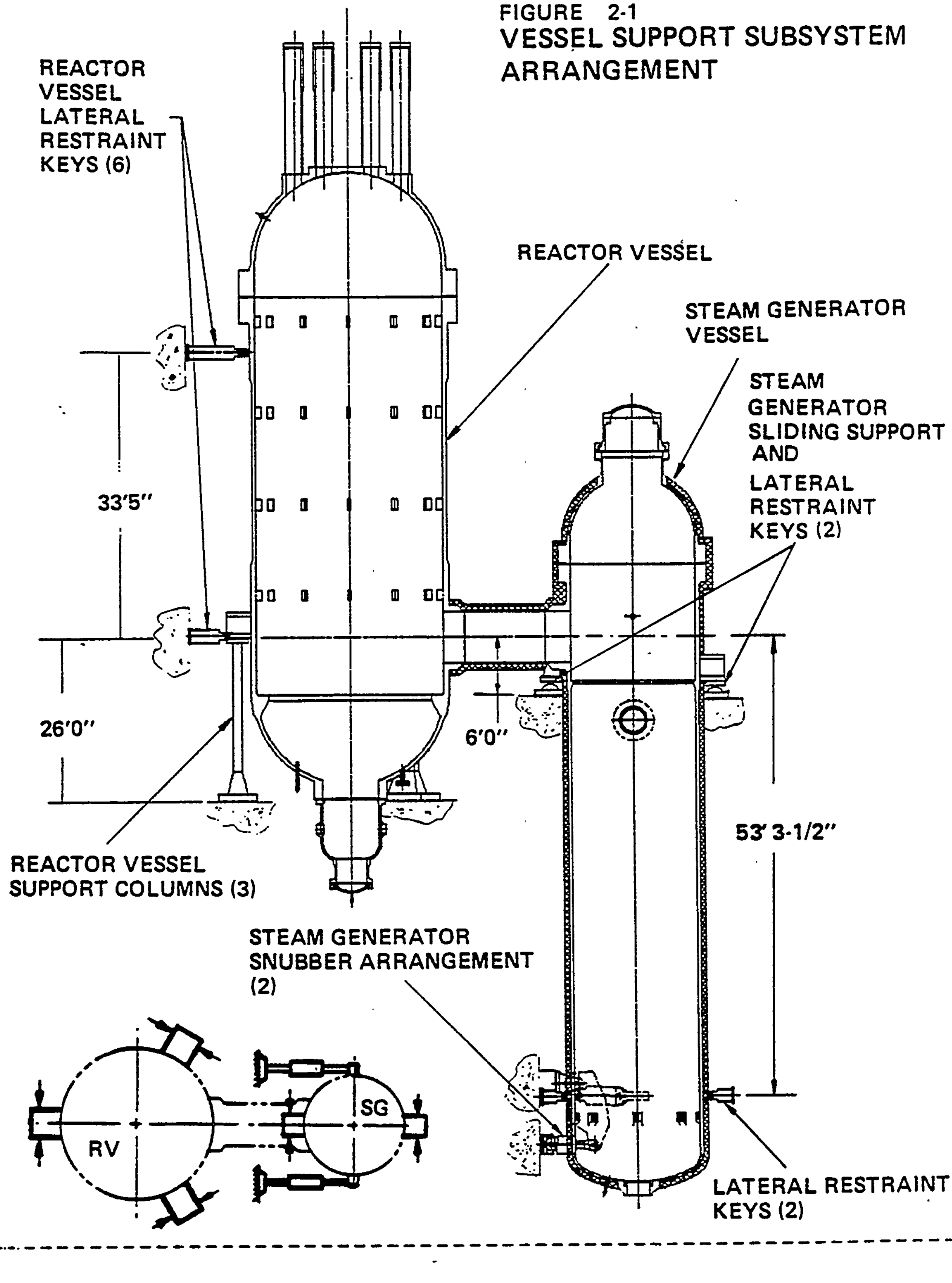

FIGURE 2-1
VESSEL SUPPORT SUBSYSTEM ARRANGEMENT

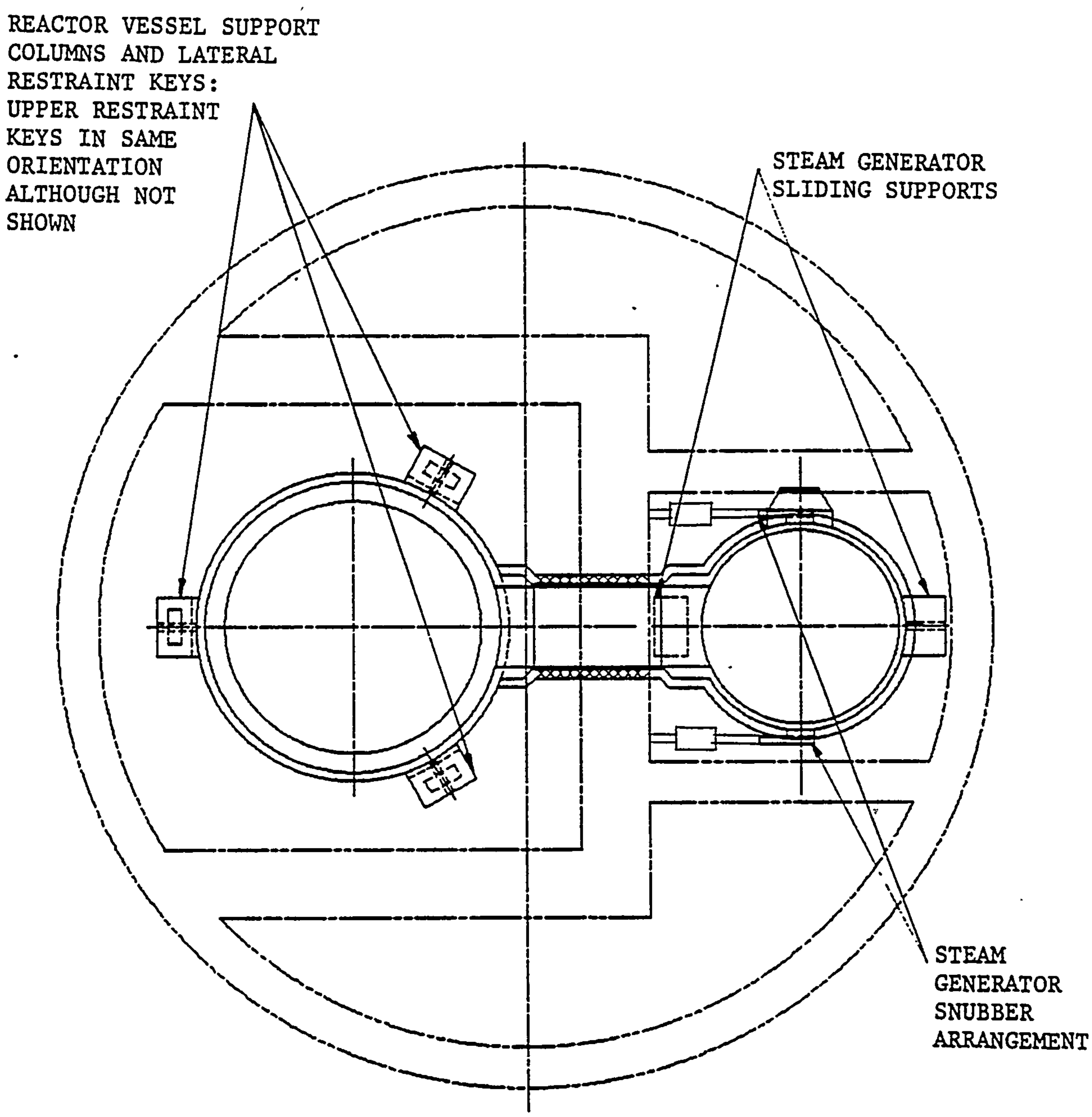

FIGURE 2-2
VESSEL SUPPORT SUBSYSTEM PLAN VIEW

2.2 SUBSYSTEM CONFIGURATION

2.2.1 Interfaces

In its installed state, the VSSS comprises the RV support columns, SGV sliding pad assemblies, and SGV snubber assemblies which are primarily passive in nature. Only the snubbers and sliding pads constitute moving parts. However, their movements are only during transitional phases, e.g., from cold-to-hot operating conditions and vice versa, heatup transients, and seismic excitations.

The primary function of the VSSS is to transmit structural loadings imposed on and by the VS to the RB. Therefore, its major interfaces are primarily structural interfaces with the VDSS (Planes A, B, and C in Figure 2-3) on one side, and with the RB (Planes D, E, and F in Figure 2-3) on the other side. Spatially, the VSSS is located within an annular space between the VDSS and RB. The VSSS also has thermal interfaces with the RCCS (Cylinder G in Figure 2-3) and the Heating, Ventilating, and Air Conditioning Subsystem (Cylinders H and J in Figure 2-3).

Within the interfaces defined in the foregoing paragraphs, the VSSS includes hardware required to transmit the structural and thermal loadings on the VS to the RB. The VSSS also transmits the seismic loadings from the RB to the VS. The VSSS is designed to ensure that, in conjunction with the RB and VDSS, the critical seismic excitation criteria of the Reactor System are met.

2.2.2 Reactor Vessel Support Configuration

The primary components of the VSSS to support the RV include the the RV Support columns (3) and related hardware to anchor the top end to the reactor vessel lug, and to anchor the bottom end to support flanges of the Reactor Building Subsystem (Figures 2-4 and 2-5).

2.2.3 Steam Generator Vessel Support Configuration

The primary components of the VSSS which support the SGV include:

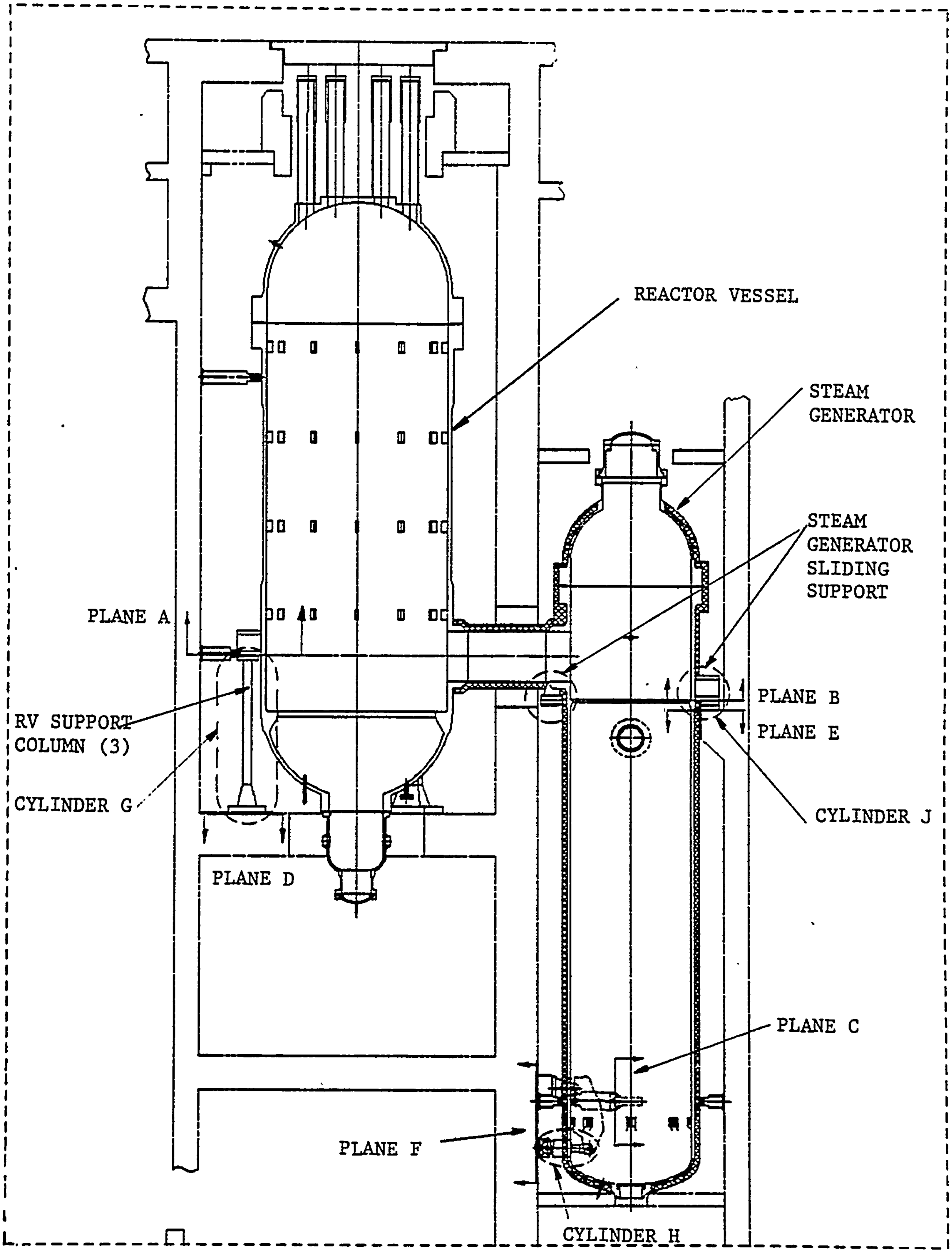

FIGURE 2-3
VESSEL SUPPORT SUBSYSTEM CONFIGURATION AND INTERFACE DESIGNATIONS
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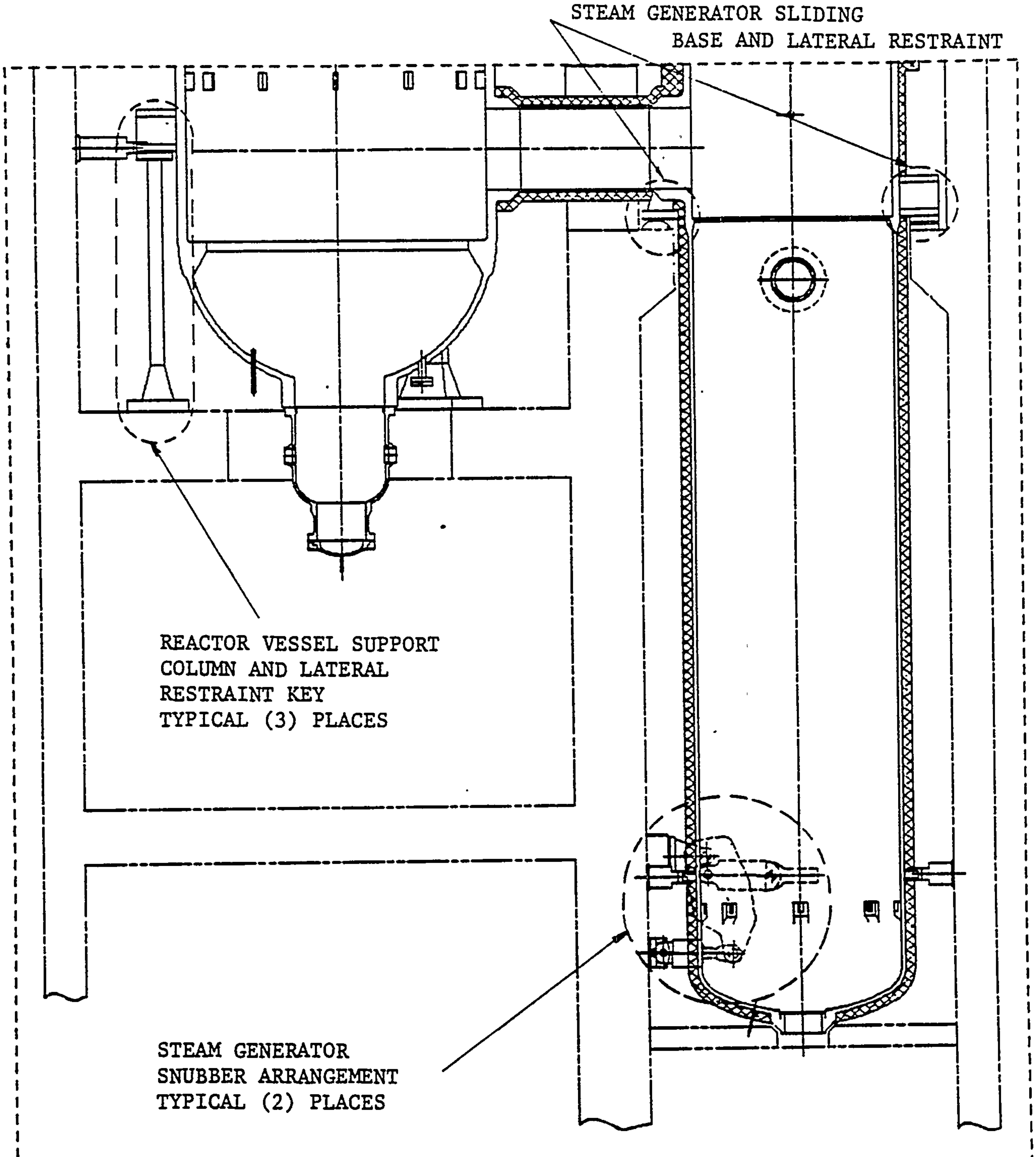


FIGURE 2-4
VESSEL SUPPORT SUBSYSTEM INTERFACES
WITH REACTOR BUILDING

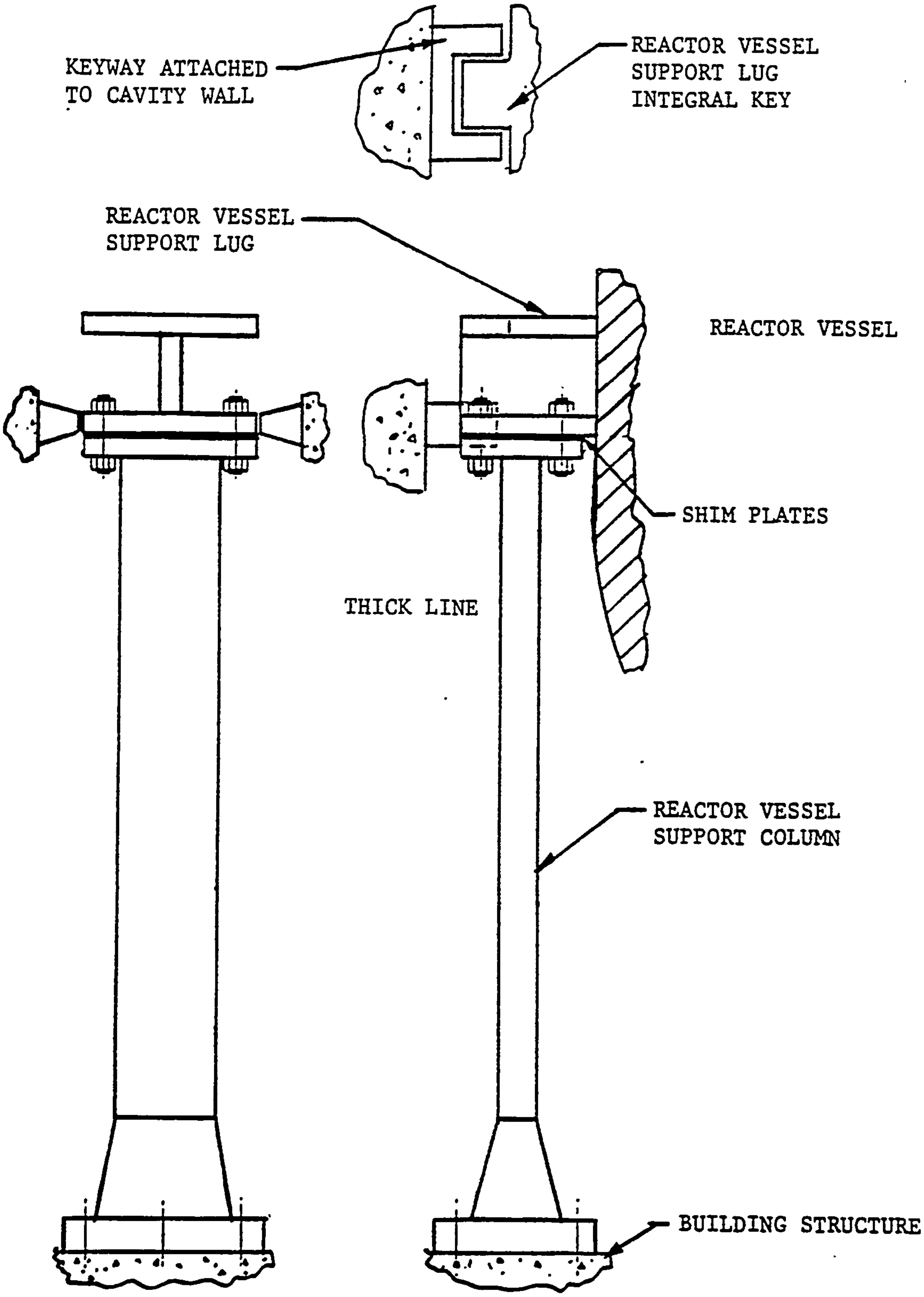


FIGURE 2-5

REACTOR VESSEL SUPPORT
COLUMNS AND LATERAL
RESTRAINT

DOE-HTGR-86-128/Rev. 1

- Spherical bearing assemblies (2) to transmit the structural loads from the SGV to the RB. The spherical bearing assemblies are located just underneath the crossduct level, below the sliding base, such that the steam generator vessel slides along the length of the crossduct (Figures 2-4 and 2-6).

- Snubber assemblies (2) near the bottom of the SGV to accommodate vertical thermal expansion and translational (sliding along the length of the crossduct) motion of the steam generator vessel (Figures 2-4, and 2-7), while providing tangential seismic restraint.

Conceptual design details are shown in the subsystem drawings listed in Appendix D.

2.3 SUBSYSTEM PERFORMANCE CHARACTERISTICS

2.3.1 Subsystem Operating Modes

The VSSS operation is primarily passive in nature. Only the snubbers and sliding bearings constitute moving parts, and their movements are rather limited [5.1 cm (2 in)], only during the transitional phases. During a transitional phase of cold-to-hot (or vice versa) operating conditions, the differential thermal expansion within the Vessel System is accommodated by the sliding pad assemblies (2). The snubbers are required to accommodate the vertical thermal expansion and translational (sliding) motion during the normal operating conditions and provide tangential restraint during the seismic excitations.

2.3.2 Subsystem Steady-State Performance

As discussed in Section 2.3.1, the VSSS operation is primarily passive in nature. The VSSS steady-state performance of the critical components such as RV support columns has been evaluated against the design requirements (Ref. 2-1 and 2-2). Initial sizing results indicate that no technical feasibility issues are anticipated. The VSSS design has sufficient margin to cover

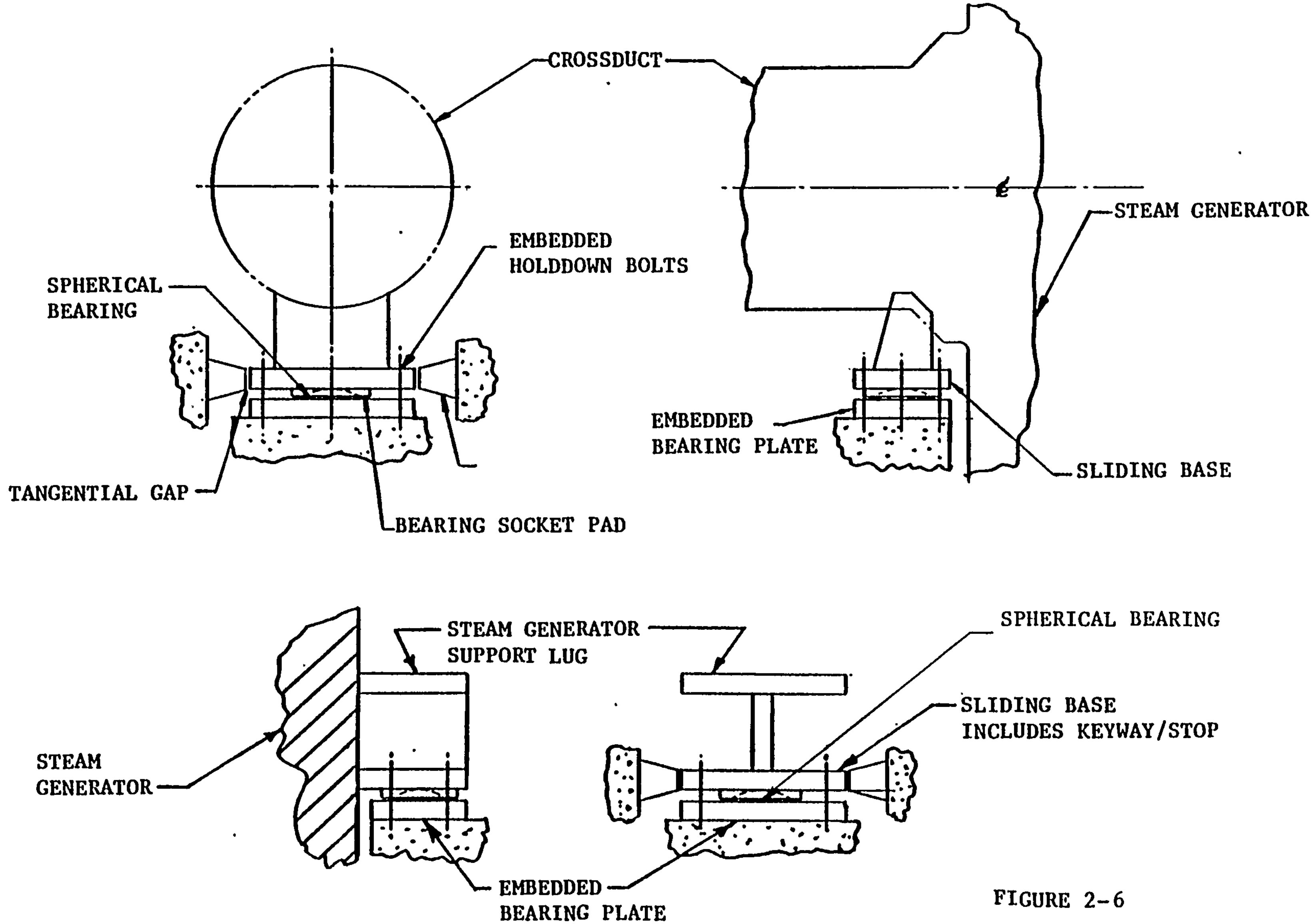


FIGURE 2-6

STEAM GENERATOR VESSEL SLIDING PAD ASSEMBLIES AND LATERAL RESTRAINT

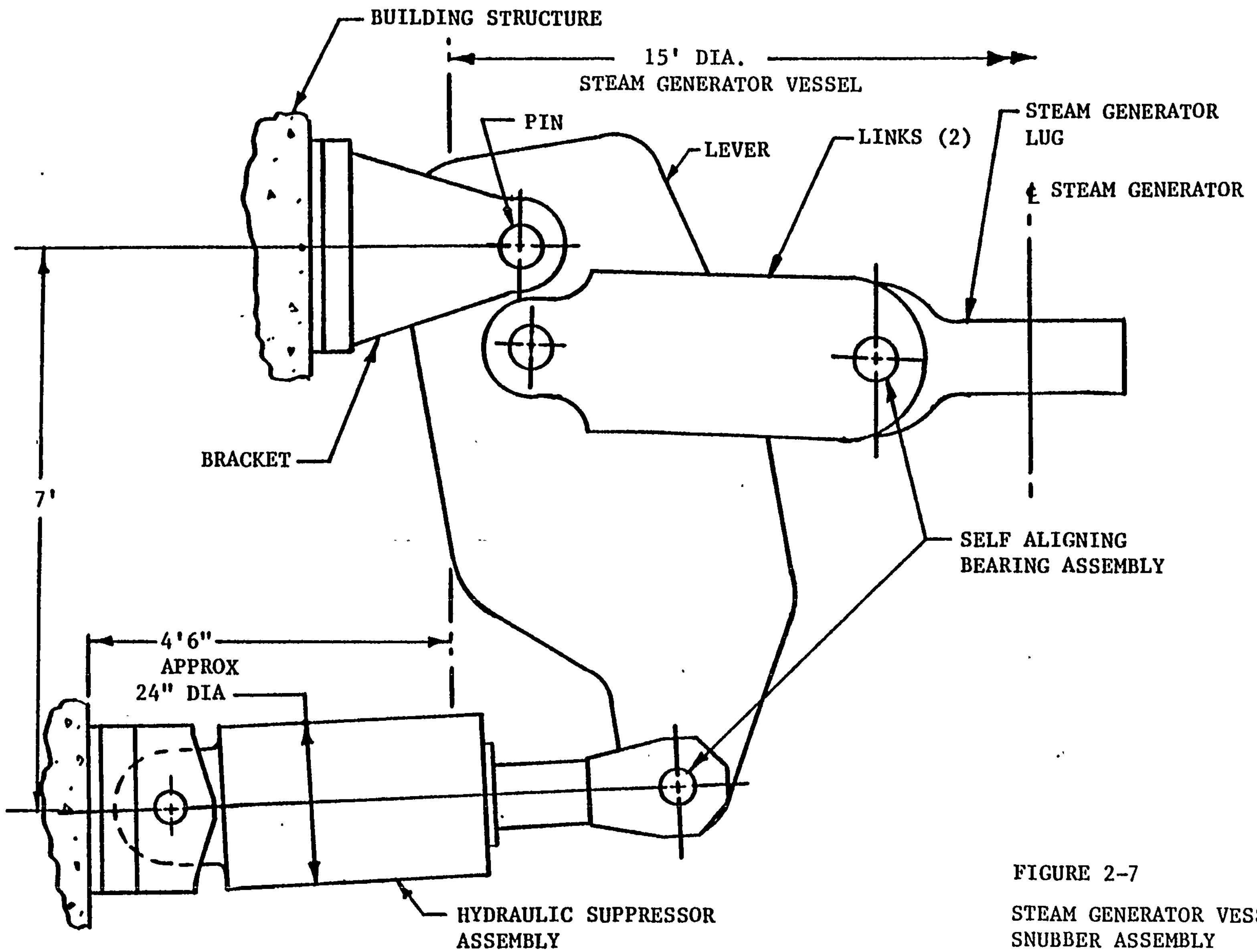


FIGURE 2-7

STEAM GENERATOR VESSEL
SNUBBER ASSEMBLY

contingencies on loadings. The VSSS component designs have been adapted from the well-established PWR component designs which have a very large database on materials and performance.

2.3.3 Subsystem Response to Plant Transients

The plant transients have practically no impact on the performance of the VSSS. It is shown in Appendix C that the peak transient temperatures are enveloped by the design temperature of the RV support columns. Therefore its performance is practically unaffected. The impact of plant transients on other components of the VSSS are expected to be minimal.

2.3.4 Subsystem Failure Modes and Effects

As discussed in Section 2.3.1, the VSSS operation is primarily passive in nature. Only snubbers and sliding bearings constitute moving parts, and their movements [5.1 cm (2.0 in)] are only during transitional phase (cold-to-hot and vice versa). During the transitional phase it is necessary to follow the established startup/shutdown procedures and monitor the VS displacements. If any measurements are beyond the established setpoints, the heatup or cooldown will be terminated until instructions in the manual are followed to avoid overstressing any components. This is standard operating procedure in the operating commercial reactors.

2.4 Subsystem Arrangement

The subsystem arrangement is presented in Figure 2-3 through 2-7. Special provisions for installation, maintenance, and inspection will be established as the design is developed.

2.5 INSTRUMENTATION AND CONTROL

No specific instrumentation and control is planned for the VSSS.

SECTION 3

COMPONENT FUNCTIONS AND DESIGN REQUIREMENTS

3.1 COMPONENT FUNCTIONS

3.1.1 Reactor Vessel Support Column Component Functions

The principal function of the RV support columns is to support the reactor vessel and its contents by transmitting loads from the RV to the RB, while accommodating the thermal growth of the RV.

3.1.2 Steam Generator Vessel Sliding Pad Assembly Component Functions

The principal function of the sliding pad assembly is to support the SGV and its contents by transmitting loads from the SGV to the RB, while accommodating the thermal growth and displacement (sliding) of the SGV in the direction of sliding (i.e., along the axis of the crossduct).

3.1.3 Steam Generator Vessel Snubber Assembly Component Functions

The principal function of the steam generator vessel snubber assembly is to provide seismic restraint to the SGV in the direction of sliding (i.e., along the axis of the crossduct), while accommodating the vertical thermal growth, and thermal growth and displacement in the direction of sliding.

3.2 COMPONENT DESIGN REQUIREMENTS

3.2.1 Reactor Vessel Support Column Component Design Requirements

3.2.1.1 Component Configuration and Essential Features Requirements

The RV and its contents shall be supported by three (3) RV support columns.
(1107.0302.001)

The RV support columns configuration shall accommodate space envelopes of the Vessels and Duct Subsystem and the Reactor Cavity Cooling System.

(1107.0302.002)

Space shall be provided by the RV support columns to allow installation, removal, ISI, and on-line and in-situ maintenance for the shutdown cooling circulator and shutdown cooling heat exchanger. The dimensions of the required space are listed in Table 1-1. (1107.0302.003)

Access shall be provided to the reactor coolant pressure boundary to facilitate in-service inspection as required by Section XI of the ASME Code.

(1107.0302.004)

The RV support columns shall not extend below the shielding in the reactor cavity -28 meters (-92 feet). [The centerline of the crossduct vessel is -22 meters (-72 feet).] (1107.0302.005)

The RV support columns shall accommodate radiative/convective heat transfer between the RV and RCCS with a minimal obstruction. (1107.0302.006)

3.2.1.2 Operational Requirements

The RV support column shall be designed for an operating life of 40 calendar years. (1107.0302.011)

The design temperature for the RV support column shall be 288°C (550°F).

(1107.0302.012)

The RV support column shall withstand the following radiation environment (total fluences):

RV Support Column (at top) - 4.0×10^{15} n/cm^2

(1107.0302.013)

The RV support column shall be designed to operate through the design transients specified in Appendix C for the number of cycles specified in Table 1-2. (1107.0302.014)

3.2.1.3 Structural Requirements

The RV support columns shall be designed to withstand the mechanical and thermal loads resulting from the design transients specified in Appendix C for the number of cycles specified in Table 1-2. (1107.0302.021)

The RV support columns shall be designed to provide structural attachments and to support the weights, and seismic, thermal, torque, and vibrational loads imposed by components of interfacing systems given in Table 3-1. (1107.0302.022)

The RV support columns in conjunction with the VDSS and other components of the VSSS shall maintain alignment of RS, FHSS, RSES, and SCS components within the tolerance given in Table 1-4. (1107.0302.023)

The RV support column shall withstand the effect of pipe rupture reactions and jet impingement loading as described in Reference 1-3. (1107.0302.024)

The RV support column shall be designed to withstand the seismic response spectra and the corresponding accelerations, deflections, and stresses given in Appendix D. (1107.0302.025)

The SSE load levels shall be twice the OBE values. (1107.0302.026)

The RV support column shall remain functional during and after an SSE. (1107.0302.027)

3.2.1.4 Environmental Requirements

The RV support columns shall accommodate the environments in the reactor building as shown in Table 1-5. (1107.0302.031)

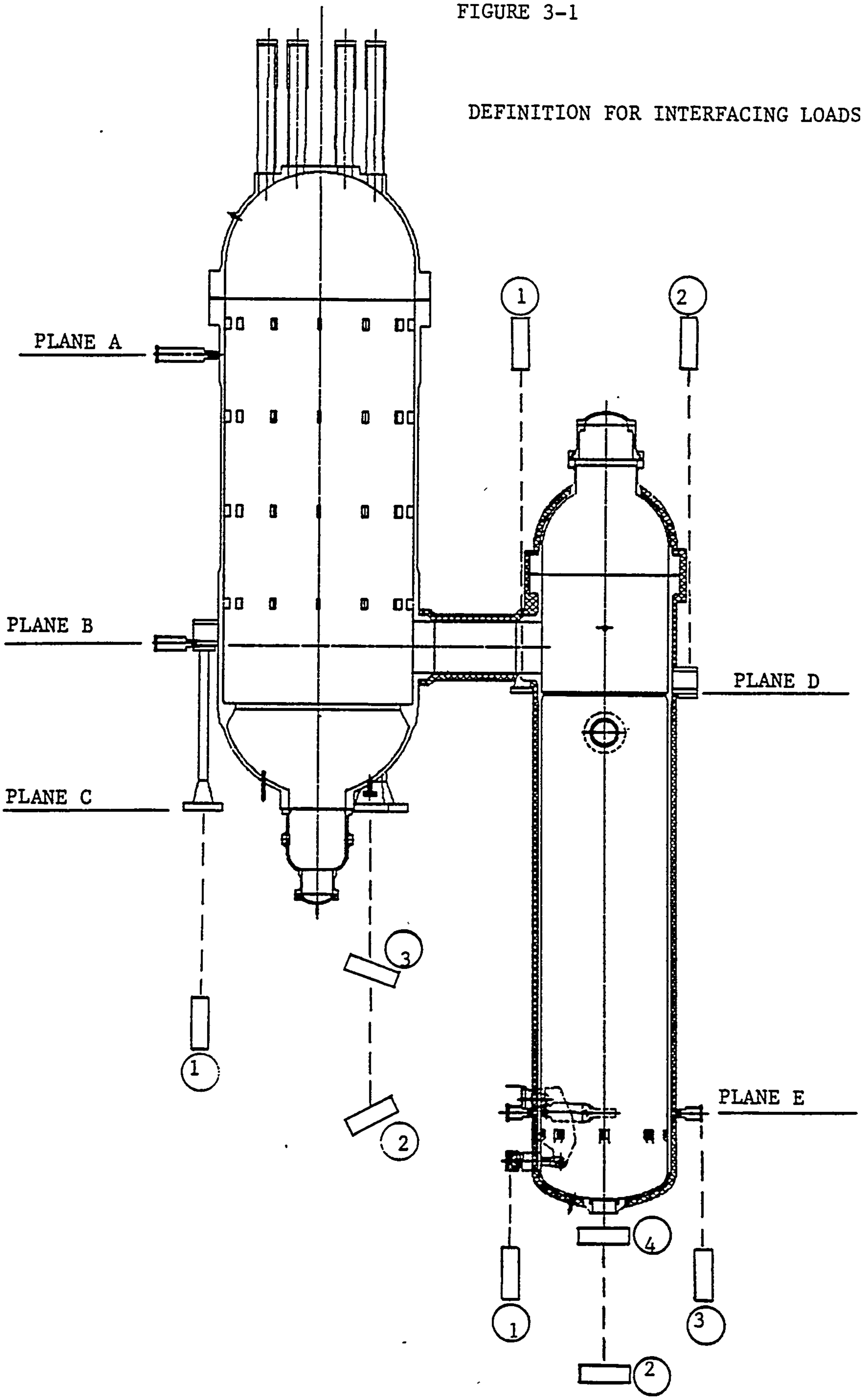
TABLE 3-1 INTERFACE LOADS BETWEEN THE VESSELS AND DUCT SUBSYSTEM AND REACTOR VESSEL SUPPORT COLUMNS

CONDITION	PLANE B LOCATION 1 **						PLANE B LOCATIONS 2 AND 3					
	FORCES (KIPS)						FORCES (KIPS)					
	Fx	Fy	Fz				Fx	Fy	Fz			
DEADWEIGHT	—	—	—				—	—	—			
TEMPERATURE NORMAL OP. + LEVEL A	—	—	—				—	—	—			
TEMPERATURE LEVEL B	—	—	—				—	—	—			
TEMPERATURE LEVEL C	—	—	—				—	—	—			
SEISMIC ONLY OBE	—	—	358.4				650.7	—	375.7			
SEISMIC ONLY SSE	—	—	716.8				1301.4	—	751.5			
APPLIED MECHANICAL LOAD	—	—	—				—	—	—			

** See Figure 3-1 for Definitions of Plane and Location

FIGURE 3-1

DEFINITION FOR INTERFACING LOADS



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3.2.1.5 Instrumentation and Control Requirements

There are no instrumentation and control requirements.

3.2.1.6 Surveillance and In-Service Inspection Requirements

The RV support column equipment vendor shall furnish special in-service inspection (ISI) equipment which is not commercially available.

(1107.0302.036)

Surveillance and ISI shall be performed on the RV support column in accordance with the Nuclear Island ISI/Surveillance Assessment (Ref. 1-3).

(1107.0302.037)

3.2.1.7 Availability Assurance Requirements

The RV support column shall be designed to meet its overall reliability and equivalent forced outage allocations given in Table 1-6. (1107.0302.041)

Design modifications and improvements which enable the plant to exceed availability requirements shall be considered for incorporation into the design, if a one percentage increase in the total capital investment produces at a minimum, a seven-tenths percentage improvement in the equivalent availability factor. (1107.0302.042)

3.2.1.8 Maintenance Requirements

The RV support column shall meet an equivalent planned outage allotment of (TBD) h/yr. (1107.0302.046)

3.2.1.9 Safety Requirements

The RV support column shall be designated as safety related and is relied upon to satisfy 10CFR100 dose limits. (1107.0302.051)

The RV support column shall assure that 10CFR100 radionuclide release limits are not exceeded for the Safety-Related Design Conditions in Table 1-7 by:

- o Maintaining the geometry of the reactor core and moveable poisons with respect to the reactor vessel to control heat generation.

- o Maintaining the geometry of the reactor vessel with respect to the RCCS during conduction cooldown. (1107.0302.052)

The RV support column shall be designed to meet the top-level regulatory criteria (Ref. 1-5). (1107.0302.053)

The RV support column shall be designed to meet the lower-level regulatory criteria contained in the regulatory guides identified for this safety-related subsystem in Table 1-8. (1107.0302.054)

SSCs designated "safety-related" shall be designed to perform their safety function(s) for the Safety-Related Design Conditions (SRCSs) listed in Table 1-7. The transient design conditions for the SRDCs are included in Ref. 1-2. (1107.0302.055)

Compliance with the safety requirements shall be ensured by designing the RV support column to meet the reliability allocations for the Standard MHTGR (Reference 1-4) and the safety performances requirements for the Standard MHTGR (TBD). (1107.0302.056)

3.2.1.10 Codes and Standards Requirements

The RV support column design shall meet the industry codes and standards for design and construction of the RV support column and its component parts as listed in Table 1-9. (1107.0302.061)

3.2.1.11 Quality Assurance Requirements

The RV support column shall come under a Quality Assurance Program which fully complies with the requirements of Title 10 Code of Federal Regulations Part 50

(10CFR50), Appendix B. The basic requirements and supplements of ANSI/ASME NQA-1 (as endorsed by Regulatory Guide 1.28, Revision 3) and the four additional supplements from DOE NE F2-10 regarding Management Assessment (NE 02-4.3.0), Engineering Holds (NE 03-1.3.2), Design Reviews (NE 03-1.3.4), and Engineering Drawings Lists (NE 03-1.3.5) shall be implemented on activities that affect the quality of such items. (1107.0302.066)

3.2.1.12 Construction Requirements

Special installation equipment not commercially available shall be provided by the equipment vendor. (1107.0302.071)

3.2.1.13 Decommissioning Requirements

Until more specific criteria and/or rules are developed, NUREG-0586, "Draft Generic Environmental Statement on Decommissioning of Nuclear Facilities," January, 1981, shall be used as guidance for anticipating NRC criteria concerning plant decommissioning. (1107.0302.076)

3.2.2 Steam Generator Vessel Sliding Pad Assembly Component Design Requirement

3.2.2.1 Component Configuration and Essential Features Requirements

The SGV and its contents shall be supported by two (2) SGV sliding pad assemblies. (1100.0302.101)

The SGV sliding pad assemblies shall accommodate the space envelope of the Vessels and Duct Subsystem. (1100.0302.102)

Space shall be provided by the SGV sliding pad assemblies to allow installation, removal, ISI, and on-line and in-situ maintenance for the main circulator and steam generator. The dimensions of the required space are listed in Table 1-1. (1107.0302.103)

Access shall be provided to the reactor coolant pressure boundary to facilitate in-service inspection as required by Section XI of the ASME Code. (1107.0302.104)

The SGV sliding pad assemblies of the SGV shall not interfere with the steam nozzle and the steam pipe. (1107.0302.105)

3.2.2.2 Operational Requirements

The SGV sliding pad assemblies shall be designed for an operating life of 40 calendar years. (1107.0302.111)

The design temperature for the SGV sliding pad assembly shall be 121°C (250°F). (1107.0302.112)

The SGV sliding pad assembly shall withstand the radiation environment of << 1 x 10^{15} n/cm^2 (total fluence). (1107.0302.113)

The SGV sliding pad assemblies shall be designed to operate through the design transients specified in Appendix C for the number of cycles specified in Table 1-2. (1107.0302.114)

3.2.2.3 Structural Requirements

The SGV sliding pad assemblies shall be designed to withstand the mechanical and thermal loads resulting from the design transients specified in Appendix C for the number of cycles specified in Table 1-2. (1107.0302.121)

The SGV sliding pad assemblies shall be designed to provide structural attachments and to support the weights, and seismic, thermal, torque, and vibrational loads imposed by components of interfacing systems given in Table 3-2. (1107.0302.122)

The SGV sliding pad assemblies in conjunction with the VDSS and other components of the VSSS shall maintain alignment of RS,and HTS components within the tolerance given in Table 1-4. (1107.0302.123)

The SGV sliding pad assemblies shall withstand the effect of pipe rupture reactions and jet impingement loading as described in Reference 1-2. (1107.0302.124)

The SGV sliding pad assemblies shall be designed to withstand the seismic response spectra and the corresponding accelerations, deflections, and stresses given in Appendix D. (1107.0302.125)

The SSE load levels shall be twice the OBE values. (1107.0302.126)

The SGV sliding pad assemblies shall remain functional during and after an SSE. (1107.0302.127)

3.2.2.4 Environmental Requirements

The SGV sliding pad assemblies shall accommodate the environments in the reactor building as shown in Table 1-5. (1107.0302.131)

3.2.2.5 Instrumentation and Control Requirements

There are no instrumentation and control requirements.

TABLE 3-2 INTERFACE LOADS BETWEEN THE VESSELS AND DUCT SUBSYSTEM AND STEAM GENERATOR VESSEL SLIDING PAD ASSEMBLY

CONDITION	PLANE D LOCATION 1 **						PLANE D LOCATION 2					
	FORCES (KIPS)			MOMENTS (IN-KIPS)			FORCES (KIPS)			MOMENTS (IN-KIPS)		
	Fx	Fy	Fz	Mx	My	Mz	Fx	Fy	Fz	Mx	My	Mz
DEADWEIGHT	—	960.0	—	—	—	—	—	644.3	—	—	—	—
TEMPERATURE NORMAL OP. + LEVEL A	—	—	—	—	—	—	—	—	—	—	—	—
TEMPERATURE LEVEL B	—	—	—	—	—	—	—	—	—	—	—	—
TEMPERATURE LEVEL C	—	—	—	—	—	—	—	—	—	—	—	—
SEISMIC ONLY OBE	—	231.3	113.5	—	—	—	—	238.5	112.2	—	—	—
SEISMIC ONLY SSE	—	462.6	227.0	—	—	—	—	477.0	224.4	—	—	—
APPLIED MECHANICAL LOAD	—	*	—	—	—	—	—	*	—	—	—	—

** See Figure 3-1 for Definitions of Plane and Location

* TBD; Conservative Estimate for Steam Pipe Break is 250 Kips.

3.2.2.6 Surveillance and In-Service Inspection Requirements

The SGV sliding pad assemblies shall furnish special in-service inspection (ISI) equipment which is not commercially available. (1107.0302.136)

Surveillance and ISI shall be performed on the SGV sliding pad assemblies in accordance with the Nuclear Island ISI/Surveillance Assessment (Ref. 1-3). (1107.0302.137)

3.2.2.7 Availability Assurance Requirements

The SGV sliding pad assemblies shall be designed to meet its overall reliability and equivalent forced outage allocations given in Table 1-6. (1107.0302.141)

Design modifications and improvements which enable the plant to exceed availability requirements shall be considered for incorporation into the design, if a one percentage increase in the total capital investment produces at a minimum, a seven-tenths percentage improvement in the equivalent availability factor. (1107.0302.142)

3.2.2.8 Maintenance Requirements

The SGV sliding pad assemblies shall meet an equivalent planned outage allotment of (TBD) h/yr. (1107.0302.146)

3.2.2.9 Safety Requirements

The SGV sliding pad assemblies shall be designated as safety related and is relied upon to satisfy 10CFR100 dose limits. (1107.0302.151)

The SGV sliding pad assemblies shall assure that 10CFR100 radionuclide release limits are not exceeded for the Safety-Related Design Conditions in Table 1-7 by:

- o Maintaining the geometry of the reactor core and moveable poisons with respect to the reactor vessel to control heat generation.

- o Maintaining the geometry of the reactor vessel with respect to the RCCS during conduction cooldown. (1107.0302.152)

The SGV sliding pad assemblies shall be designed to meet the top-level regulatory criteria (Ref. 1-5). (1107.0302.153)

The SGV sliding pad assemblies shall be designed to meet the lower-level regulatory criteria contained in the regulatory guides identified for this safety-related subsystem in Table 1-8. (1107.0302.154)

SSCs designated "safety-related" shall be designed to perform their safety function(s) for the Safety-Related Design Conditions (SRCSs) listed in Table 1-7. The transient design conditions for the SRDCs are included in Ref. 1-2. (1107.0302.155)

Compliance with the safety requirements shall be ensured by designing the SGV sliding pad assemblies to meet the reliability allocations for the Standard MHTGR (Reference 1-4) and the safety performances requirements for the Standard MHTGR (TBD). (1107.0302.156)

3.2.2.10 Codes and Standards Requirements

The SGV sliding pad assembly design shall meet the industry codes and standards for design and construction of the SGV sliding pad assembly and its parts as listed in Table 1-9. (1107.0302.161)

3.2.2.11 Quality Assurance Requirements

The SGV sliding pad assemblies shall come under a Quality Assurance Program which fully complies with the requirements of Title 10 Code of Federal Regulations Part 50 (10CFR50), Appendix B. The basic requirements and supplements of ANSI/ASME NQA-1 (as endorsed by Regulatory Guide 1.28, Revision 3) and the four additional supplements from DOE NE F2-10 regarding Management

Assessment (NE 02-4.3.0), Engineering Holds (NE 03-1.3.2), Design Reviews (NE 03-1.3.4), and Engineering Drawings Lists (NE 03-1.3.5) shall be implemented on activities that affect the quality of such items.

(1107.0302.166)

3.2.2.12 Construction Requirements

Special installation equipment not commercially available shall be provided by the equipment vendor. (1107.0302.171)

3.2.2.13 Decommissioning Requirements

Until more specific criteria and/or rules are developed, NUREG-0586, "Draft Generic Environmental Statement on Decommissioning of Nuclear Facilities," January, 1981, shall be used as guidance for anticipating NRC criteria concerning plant decommissioning. (1107.0302.176)

3.2.3 Steam Generator Vessel Snubber Assembly Component Design Requirement

3.2.3.1 Component Configuration and Essential Features Requirements

The SGV and its contents shall be tangentially restrained from seismic excitations by two (2) SGV snubber assemblies (1100.0302.201)

The SGV snubber assemblies configuration shall accommodate space envelope of the Vessels and Duct Subsystem. (1100.0302.202)

Space shall be provided by the SGV snubber assemblies to allow installation, removal, ISI, and on-line and in-situ maintenance for the main circulator and steam generator. The dimensions of the required space are listed in Table 1-1. (1107.0302.203)

Access shall be provided to the reactor coolant pressure boundary to facilitate in-service inspection as required by Section XI of the ASME Code. (1107.0302.204)

The SGV snubber assemblies shall be oriented as shown in Figure 2-2 so that seismic loads shall be transmitted to the central load bearing wall of the cavity. (1107.0302.205)

3.2.3.2 Operational Requirements

The SGV snubber assemblies shall be designed for an operating life of 40 calendar years. (1107.0302.211)

The design temperature for the SGV snubber assemblies shall be 66°C (150°F). (1107.0302.212)

The SGV snubber assemblies shall withstand the radiation environment of $<<< 10^{15}$ n/cm^2 (total fluence). (1107.0302.213)

The SGV snubber assemblies shall be designed to operate through the design transients specified in Appendix C for the number of cycles specified in Table 1-2. (1107.0302.214)

3.2.3.3 Structural Requirements

The SGV snubber assemblies shall be designed to withstand the mechanical and thermal loads resulting from the design transients specified in Appendix C for the number of cycles specified in Table 1-2. (1107.0302.221)

The SGV snubber assemblies shall be designed to provide structural attachments and to support the weights, and seismic, thermal, torque, and vibrational loads imposed by components of interfacing systems given in Table 3-3. (1107.0302.222)

The SGV snubber assemblies in conjunction with the VDSS and other components of the VSSS shall maintain alignment of RS and HTS components within the tolerance given in Table 1-4. (1107.0302.223)

The SGV snubber assemblies shall withstand the effect of pipe rupture reactions and jet impingement loading as described in Reference 1-2. (1107.0302.224)

The SGV snubber assemblies shall be designed to withstand the seismic response spectra and the corresponding accelerations, deflections, and stresses given in Appendix D. (1107.0302.225)

The SSE load levels shall be twice the OBE values. (1107.0302.226)

The SGV snubber shall remain functional during and after an SSE. (1107.0302.227)

3.2.3.4 Environmental Requirements

The SGV snubber assemblies shall accommodate the environments in the reactor building as shown in Table 1-5. (1107.0302.231)

TABLE 3-3 INTERFACE LOADS BETWEEN THE VESSELS AND DUCT SUBSYSTEM AND STEAM GENERATOR VESSEL SNUBBER ASSEMBLY

	PLANE E LOCATIONS 2 AND 4 **					
	FORCES (KIPS)			MOMENTS (IN-KIPS)		
CONDITION	Fx	Fy	Fz	Mx	My	Mz
DEADWEIGHT	—	—	—	—	—	—
TEMPERATURE NORMAL OP. + LEVEL A	—	—	—	—	—	—
TEMPERATURE LEVEL B	—	—	—	—	—	—
TEMPERATURE LEVEL C	—	—	—	—	—	—
SEISMIC ONLY OBE	147.9	—	—	—	—	—
SEISMIC ONLY SSE	295.8	—	—	—	—	—
APPLIED MECHANICAL LOAD	—	—	—	—	—	—

** See Figure 3-1 for Definitions of Plane and Location

3.2.3.5 Instrumentation and Control Requirements

There are no instrumentation and control requirements.

3.2.3.6 Surveillance and In-Service Inspection Requirements

The SGV snubber assembly equipment vendor shall furnish special in-service inspection (ISI) equipment which is not commercially available.
(1107.0302.236)

Surveillance and ISI shall be performed on the SGV snubber assemblies in accordance with the Nuclear Island ISI/Surveillance Assessment (Ref. 1-3).
(1107.0302.237)

3.2.3.7 Availability Assurance Requirements

The SGV snubber assemblies shall be designed to meet its overall reliability and equivalent forced outage allocations given in Table 1-6. (1107.0302.241)

Design modifications and improvements which enable the plant to exceed availability requirements shall be considered for incorporation into the design, if a one percentage increase in the total capital investment produces at a minimum, a seven-tenths percentage improvement in the equivalent availability factor. (1107.0302.242)

3.2.3.8 Maintenance Requirements

The SGV snubber pad assemblies shall meet an equivalent planned outage allotment of (TBD) h/yr. (1107.0302.246)

3.2.3.9 Safety Requirements

The SGV snubber assemblies shall be designated as safety related and is relied upon to satisfy 10CFR100 dose limits. (1107.0302.251)

The SGV snubber assemblies shall assure that 10CFR100 radionuclide release limits are not exceeded for the Safety-Related Design Conditions in Table 1-7 by:

- o Maintaining the geometry of the reactor core and moveable poisons with respect to the reactor vessel to control heat generation.

- o Maintaining the geometry of the reactor vessel with respect to the RCCS during conduction cooldown. (1107.0302.252)

The SGV snubber assemblies shall be designed to meet the top-level regulatory criteria (Ref. 1-5). (1107.0302.253)

The SGV snubber assemblies shall be designed to meet the lower-level regulatory criteria contained in the regulatory guides identified for this safety-related subsystem in Table 1-8. (1107.0302.254)

SSCs designated "safety-related" shall be designed to perform their safety function(s) for the Safety-Related Design Conditions (SRCSs) listed in Table 1-7. The transient design conditions for the SRDCs are included in Ref. 1-2. (1107.0302.255)

Compliance with the safety requirements shall be ensured by designing the SGV snubber assemblies to meet the reliability allocations for the Standard MHTGR (Reference 1-4) and the safety performances requirements for the Standard MHTGR (TBD). (1107.0302.256)

3.2.3.10 Codes and Standards Requirements

The SGV snubber assembly design shall meet the industry codes and standards for design and construction of the SGV snubber assembly and its parts as listed in Table 1-9. (1107.0302.261)

3.2.3.11 Quality Assurance Requirements

The SGV snubber assemblies shall come under a Quality Assurance Program which fully complies with the requirements of Title 10 Code of Federal Regulations Part 50 (10CFR50), Appendix B. The basic requirements and supplements of ANSI/ASME NQA-1 (as endorsed by Regulatory Guide 1.28, Revision 3) and the four additional supplements from DOE NE F2-10 regarding Management Assessment NE 02-4.3.0), Engineering Holds (NE 03-1.3.2), Design Reviews (NE 03-1.3.4), and Engineering Drawings Lists (NE 03-1.3.5) shall be implemented on activities that affect the quality of such items.

(1107.0302.266)

3.2.3.12 Construction Requirements

Special installation equipment not commercially available shall be provided by the equipment vendor. (1107.0302.271)

3.2.3.13 Decommissioning Requirements

Until more specific criteria and/or rules are developed, NUREG-0586, "Draft Generic Environmental Statement on Decommissioning of Nuclear Facilities," January, 1981, shall be used as guidance for anticipating NRC criteria concerning plant decommissioning. (1107.0302.276)

SECTION 4

SUBSYSTEM AND COMPONENT INTERFACES

4.1 SUBSYSTEM INTERFACE REQUIREMENTS

4.1.1 Interface Requirements Imposed on Buildings and Structures and on Other Systems and Subsystems Within Other Systems

Interface requirements imposed on buildings and structures and on other systems and subsystems within other systems by the VSSS are identified in Table 4-1, which also includes a description of the interface, and a quantitative expression for, and location of, the interface. The location of the interfaces is shown in Figure 4.1.

4.1.2 Interface Requirements Imposed on Subsystems Within the System

Interface requirements imposed by VSSS on subsystems within the VS are identified in Table 4-4, which also includes a description of the interface, and a quantitative expression for, and location of, the interface. The location of the interfaces is shown in Figure 4-1.

4.2 SUBSYSTEM BOUNDARY DEFINITION

(LATER)

TABLE 4-1

INTERFACE REQUIREMENTS IMPOSED ON BUILDINGS AND STRUCTURES

AND OTHER SYSTEMS AND SUBSYSTEMS WITHIN OTHER SYSTEMS

Interfacing Systems (With Subsystem/Identification		Nature of Interface	Interfacing Component	Interface Requirements
4.1.1.1	Plant Control Data and Instrumentation System (3700)	None		
4.1.1.2	Reactor System (1000)	None		
4.1.1.3	Reactor Services Group (2000)	None		
4.1.1.4	Heat Transport System (2100)	Transfer of Interfacing Loads	Connection to Steam Generator and Circulator	Connections shall accommodate the thermal and seismic deflections (Table 4-2) of the VS without imposing excessive stresses on the VS. (1107.0401.001)
4.1.1.4	Reactor Cavity Cooling System RCCS (5600)	Limit Support Column Temperatures	RCCS	Assist in maintaining the support column at less than 550°F during operation. (1107.0401.002)

TABLE 4-1 (continued)

INTERFACE REQUIREMENTS IMPOSED ON BUILDINGS AND STRUCTURES

AND OTHER SYSTEMS AND SUBSYSTEMS WITHIN OTHER SYSTEMS

Interfacing Systems (With Subsystem/Identification	Nature of Interface	Interfacing Component	Interface Requirements
4.1.1.5 Shutdown Cooling System (5700)	Transfer of Interfacing Loads	Shutdown Cooling Heat Exchanger	Shutdown Cooling System connections to the Reactor Building shall accommodate the thermal and seismic deflections (Table 4-2) of the VS without imposing excessive stresses on the VS. (1107.0401.003)
4.1.1.6 Plant Protection and Instrumentation System (3200)	None		
4.1.1.7 Fuel Handling and Storage System (3400)	Transfer of Interfacing Loads	Core Refueling Machine	Connections shall accommodate the thermal and seismic deflections (Table 4-2) of the VS without imposing excessive stresses on the VS. (1107.0401.004)

TABLE 4-1 (continued)

INTERFACE REQUIREMENTS IMPOSED ON BUILDINGS AND STRUCTURES

AND OTHER SYSTEMS AND SUBSYSTEMS WITHIN OTHER SYSTEMS

Interfacing Systems (With Subsystem/Identification	Nature of Interface	Interfacing Component	Interface Requirements
4.1.1.8 Miscellaneous Control and Instrumentation System (3000)	None		
4.1.1.9 Heat Rejection Group (5200)	None		
4.1.1.10 Power Conversion Group (5000)	None		
(Feedwater and Condensate) (5002)	Accommodate Thermal and Seismic Deflections	Feedwater Inlet Piping	Connections shall accommodate the thermal and seismic deflections (Table 4-2) of the VS without imposing excessive stresses the VS. (1107.0401.005)

TABLE 4-1 (continued)

INTERFACE REQUIREMENTS IMPOSED ON BUILDINGS AND STRUCTURES

AND OTHER SYSTEMS AND SUBSYSTEMS WITHIN OTHER SYSTEMS

Interfacing Systems (With Subsystem/Identification	Nature of Interface	Interfacing Component	Interface Requirements
	Accommodate Deflections and Thermal Growth	Figure 4-1	Accommodate deflections and thermal growth of the Vessel System (including VSSS). (1107.0401.009)
	Provide Seismic Restraints	Keyways	Provide Keyways for tangential seismic restraints as follows See Figure 4-1): RV top key (3) (Plane A), RV support lug (3) (Plane B), SGV sliding pad assemblies (2) (Plane D), and SGV bottom support (2) (Plane E) (1107.0401.010)
	Anchor points	As in Figure 4-1	Anchor RV support columns (Plane C), anchor SGV sliding assemblies (Plane D), and anchor SGV snubber assemblies (Plane E) (1107.0401.011)

TABLE 4-1 (continued)

INTERFACE REQUIREMENTS IMPOSED ON BUILDINGS AND STRUCTURES

AND OTHER SYSTEMS AND SUBSYSTEMS WITHIN OTHER SYSTEMS

Interfacing Systems (With Subsystem/Identification	Nature of Interface	Interfacing Component	Interface Requirements
Main Steam and Bypass Steam (5004)	Accommodate Thermal and Seismic Deflections	Steam Outlet	Connections shall accommodate the thermal and seismic deflections (Table 4-2) of the VS without imposing excessive stresses on the VS. (1107.0401.006)
4.1.1.11 Buildings Structures and Buildings Service Groups (7000)			
Reactor Building Subsystem (7001)	Withstand Loads	Reactor Building Components as shown in Figure 4-1.	Withstand mechanical, seismic and fluid loads transmitted from the VSSS as depicted in Table 4-3. (1107.0401.007)
	Envelope	Figure 4-1	Envelope the Vessel System including the VSSS. (1107.0401.008)

TABLE 4-1 (continued)

INTERFACE REQUIREMENTS IMPOSED ON BUILDINGS AND STRUCTURES

AND OTHER SYSTEMS AND SUBSYSTEMS WITHIN OTHER SYSTEMS

Interfacing Systems (With Subsystem/Identification	Nature of Interface	Interfacing Component	Interface Requirements
4.1.1.12 Plant Services Group (9000)	Accommodate Access for ISI and maintenance	Steam Generator Cavity	Provide access-path (3 feet diameter minimum) to remove and reinstall SGV snubbers. (1107.0401.012)
HVAC (9011)	Differential Thermal Growth	RV and SG Cavities	Maintain cavity temperatures to limit VSSS component differential growth. (1107.0401.013)
4.1.1.13 Electrical Group (9700)	None		

TABLE 4-2

VESSEL SYSTEM THERMAL AND SEISMIC DEFLECTIONS

AT INTERFACING PIPING CONNECTIONS

[All deflections in mm (in.)]

TYPE	DIRECTION (1) X	Y	Z
A. Reactor Vessel Deflections at Shutdown Cooling System			
Seismic	1.22 (0.048)	0.68 (0.027)	1.22 (0.048)
Thermal	TBD	TBD	TBD
B. Reactor Vessel Deflections at Fuel Handling and Storage System			
Seismic	TBD	TBD	TBD
Thermal	TBD	TBD	TBD
C. Steam Generator Vessel Deflections at Steam Tubesheet Access Extension; Heat Transport System, and Main and Bypass Steam System			
Seismic	1.96 (0.077)	0.19 (0.007)	0.89 (0.035)
Thermal	50.8 (2.0)	TBD	TBD
D. Steam Generator Vessel Deflections at Feedwater Tubesheet Access Extension; Heat Transport System, and Feedwater and Condensate System			
Seismic	1.90 (0.075)	0.23 (0.009)	1.78 (0.070)
Thermal	50.8 (2.0)	TBD	0.0

(1) X is along the axis of the crossduct
Y is along the axis of the reactor vessel
Z is perpendicular to plane XY

TABLE 4-3 INTERFACE LOAD REQUIREMENTS BETWEEN VESSEL SUPPORT SUBSYSTEM AND REACTOR BUILDING

CONDITION	PLANE C LOCATION 1 **						PLANE C LOCATIONS 2 AND 3					
	FORCES (KIPS)			MOMENTS (IN-KIPS)			FORCES (KIPS)			MOMENTS (IN-KIPS)		
	Fx	Fy	Fz	Mx	My	Mz	Fx	Fy	Fz	Mx	My	Mz
DEADWEIGHT	15.2	1265.	—	—	—	1218.	10.2	1403.	49.2	3832.	36.7	811.
TEMPERATURE NORMAL OP. + LEVEL A	84.1	—	—	—	—	737 0.	42.1	—	72.8	3685.	—	6383.
TEMPERATURE LEVEL B	84.1	—	—	—	—	737 0.	42.1	—	72.8	3685.	—	6383.
TEMPERATURE LEVEL C	84.1	—	—	—	—	737 0.	42.1	—	72.8	3685.	—	6383.
SEISMIC ONLY OBE	10.1	644.2	4.7	365.9	1332.	997.4	20.4	729.1	17.7	1674.	2812.	1998.
SEISMIC ONLY SSE	20.2	1288.	9.4	731.8	2664.	1995.	40.8	1458.	35.4	3348.	5624.	3996.
APPLIED MECHANICAL LOAD	—	—	—	—	—	—	—	—	—	—	—	—

** See Figure 4-1 for Definitions of Plane and Location

TABLE 4-3....CONTINUED

	PLANE D LOCATION 1						PLANE D LOCATION 2					
	FORCES (KIPS)			MOMENTS (IN-KIPS)			FORCES (KIPS)			MOMENTS (IN-KIPS)		
CONDITION	Fx	Fy	Fz	Mx	My	Mz	Fx	Fy	Fz	Mx	My	Mz
DEADWEIGHT	—	960.0	—	—	—	—	—	644.3	—	—	—	—
TEMPERATURE NORMAL OP. + LEVEL A	—	—	—	—	—	—	—	—	—	—	—	—
TEMPERATURE LEVEL B	—	—	—	—	—	—	—	—	—	—	—	—
TEMPERATURE LEVEL C	—	—	—	—	—	—	—	—	—	—	—	—
SEISMIC ONLY OBE	—	231.3	113.5	—	—	—	—	238.5	112.2	—	—	—
SEISMIC ONLY SSE	—	462.6	227.0	—	—	—	—	477.0	224.4	—	—	—
APPLIED MECHANICAL LOAD	—	*	—	—	—	—	—	*	—	—	—	—

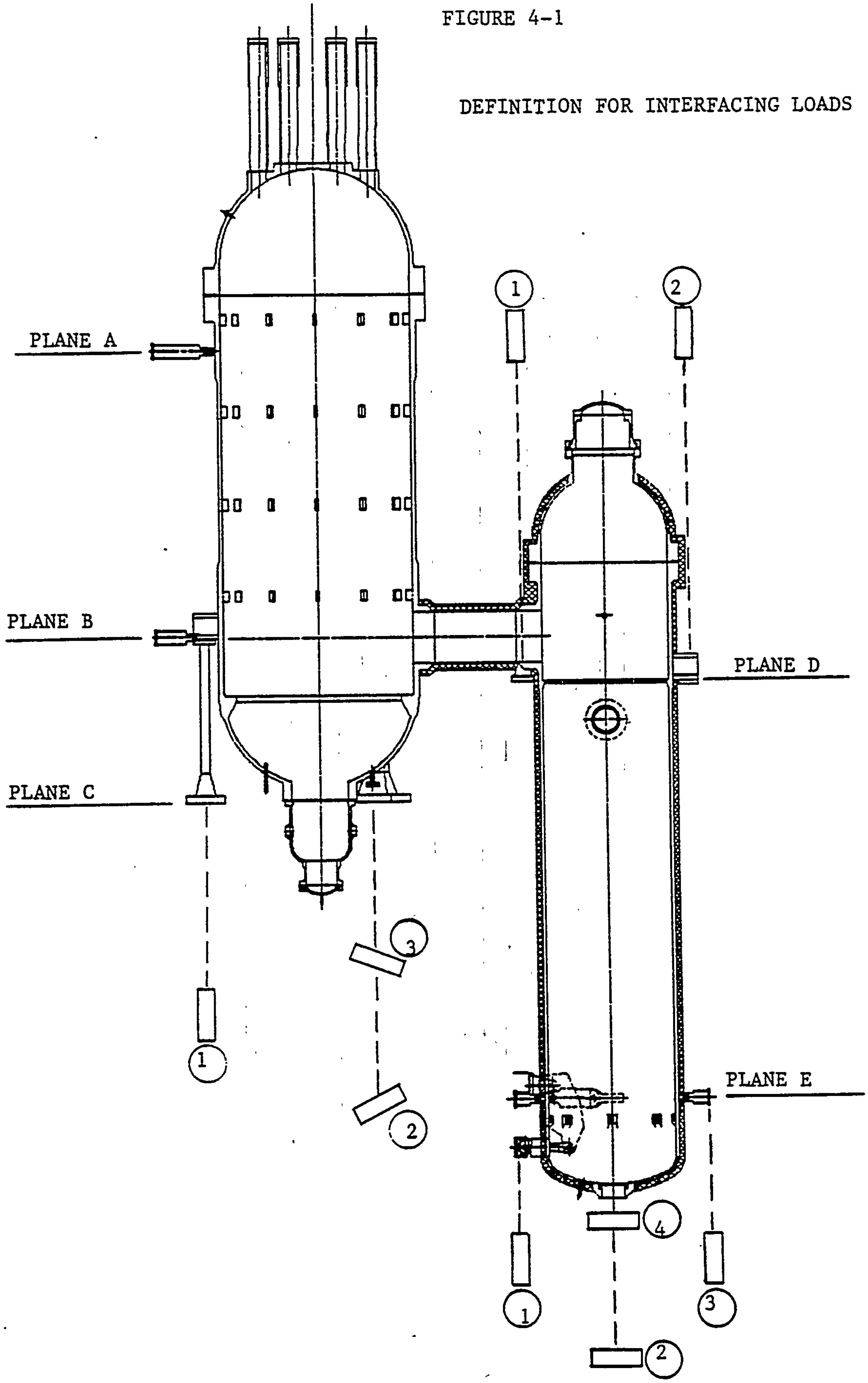
* TBD; Conservative Estimate for Steam Pipe Break is 250 Kips.

TABLE 4-3....CONTINUED

CONDITION	PLANE E LOCATIONS 2 AND 4					
	FORCES (KIPS)			MOMENTS (IN-KIPS)		
	Fx	Fy	Fz	Mx	My	Mz
DEADWEIGHT	—	—	—	—	—	—
TEMPERATURE NORMAL OP. + LEVEL A	—	—	—	—	—	—
TEMPERATURE LEVEL B	—	—	—	—	—	—
TEMPERATURE LEVEL C	—	—	—	—	—	—
SEISMIC ONLY OBE	147.9	—	—	—	—	—
SEISMIC ONLY SSE	295.8	—	—	—	—	—
APPLIED MECHANICAL LOAD	—	—	—	—	—	—

FIGURE 4-1

DEFINITION FOR INTERFACING LOADS



DOE-HTGR-86-128/Rev. 1

TABLE 4-4

INTERFACE REQUIREMENTS IMPOSED ON SUBSYSTEMS WITHIN VESSEL SYSTEM

Interfacing Systems (With Subsystem/Identification	Nature of Interface	Interfacing Component	Interface Requirements
4.1.2.1 Vessel System (1100)			
Vessels and Duct Subsystem (1106)	Provide Support Points	Reactor Vessel	Provide three (3) lugs or support pads to anchor the RV to the support columns. (Figure 4-1, Plane A) (1107.0401.101)
			Provide three (3) keys for tangential seismic restraint (Figure 4-1, Plane A) (1107.0401.102)
		Steam Generator Vessel	Provide two (2) bases, in-line with the crossduct, one at the crossduct and other opposite the crossduct, to anchor the SG vessel to RB. (Figure 4-1, Plane D) (1107.0401.103)

TABLE 4-4 (continued)

INTERFACE REQUIREMENTS IMPOSED ON SUBSYSTEMS WITHIN VESSEL SYSTEM

Interfacing Systems (With Subsystem/Identification	Nature of Interface	Interfacing Component	Interface Requirements
			Provide two (2) keys for tangential seismic restraint, near the the bottom of the SG vessel (Figure 4-1, Plane E) (1107.0401.104)
			Provide two (2) anchor points to anchor snubber assemblies to SG vessel. (Figure 4-1, Plane E) (1107.0401.105)

SECTION 5

SUBSYSTEM CONSTRUCTION

[LATER]

SECTION 6

SUBSYSTEM OPERATION

The Vessel Support Subsystem operation is primarily passive in nature. Only snubbers and sliding pads constitute moving parts, and their primary movements are only during the transitional phases. During a transitional phase of cold-to-hot (or vice versa) operating conditions, the differential thermal expansion within the Vessel System is accommodated by sliding pad assembly (2). The snubbers are required to accommodate the SG vertical thermal expansion and translational (sliding) motion during the normal operating conditions, while providing the seismic restraint in the direction of sliding motion.

6.1 SUBSYSTEM LIMITATIONS, SETPOINTS AND PRECAUTIONS

6.1.1 Subsystem Limitations and Setpoints

The VSSS operation is primarily passive in nature. Primary movements of the VSSS components are only during transitional phase of cold-to-hot (or vice verse) operating conditions. Therefore, limitations and setpoints are limited to this transitional phase. It is necessary that sufficient time (TBD) be allowed for temperature stabilization prior to the VSSS operational measurements. It is also necessary that a detailed operational procedure be developed for this transitional phase. The procedure manual shall establish normal heatup and cooldown rates, frequency and anticipated growth or displacement of the VS, and limitations and setpoints for any corrective actions.

The VSSS installation limitations and setpoints (TBD) will be specified in the detailed design drawings and installation guidelines.

The VSSS requires a very elaborate procedure to reduce interface gaps between the support points and RB. For example, tangential seismic restraints require that various gaps be reduced to 4 mm (1/64 inches) with shims at hot testing. This requires that a procedure manual be developed, as in normal LWR practices.

6.1.2 Precautions

It is necessary to follow the starting up/shutting down procedure manual during transitional phase of cold-to-hot (or vice versa). If any measurement of the VSSS growth is outside the established setpoints, the heatup or cooldown shall be terminated until instructions in the procedure manual are followed. A failure to obey holdpoints could overstress the VS and/or RB components. The VSSS installation continues during hot functional test when shims are installed to reduce interface gaps. A failure to set these gaps to specifications would either overstress the VS and/or RB, or the VSSS will not provide adequate seismic restraints.

6.2 PREOPERATIONAL CHECKOUT

Hot Functional Test objectives will be to demonstrate the unobstructed freedom of the Vessel System to expand and contract during plant heatup and cooldown, and to obtain measurements to allow correct sizing of the vessel stop shims.

6.3 STARTUP/SHUTDOWN

6.3.1 Startup to 25% Steam Flow

There are no operational requirements for hot startup and shutdown. For startup and shutdown from or to cold conditions, the RBS vendor in conjunction with the VS vendor will develop hot testing and startup/shutdown procedures. These procedures will include a checklist of the VS growth measurements with respect to the RB and hold points to eliminate overstressing any components. Details of these procedures will be initiated in the preliminary design.

6.3.2 Shutdown from 25% Steam Flow

Operation procedures are the same as in Section 6.3.1.

6.4 NORMAL OPERATION

The VSSS exhibits a passive nature during normal operation.

6.5 REFUELING

The VSSS continues to operate passively during refueling.

6.6 SHUTDOWN

The VSSS continues to operate passively during shutdown.

6.7 ABNORMAL OPERATION

The VSSS's operation during abnormal operation is TBD when the temperature and loading characteristics of the VS during these operations are established.

6.8 CASUALTY EVENTS

The VSSS's operation during casualty events is TBD when the temperature and loading characteristics of the VS during these events are established.

SECTION 7

SUBSYSTEM MAINTENANCE

7.1 MAINTENANCE APPROACH

The VSSS is designated as a "safety related" system and is subject to in-service inspection (ISI). Detailed maintenance and ISI assessments of the MHTGR Vessel System are documented in References 7-1 and 1-3, respectively. A preliminary maintenance and ISI assessments have also been reviewed in the VSSS Conceptual Design Report (Reference 2-1).

There are no specific maintenance requirements for the Vessel Support Subsystem. The reactor vessel support column, as in the commercial PWR systems, is a single forged piece. The operation of the reactor vessel support column is completely passive, requiring no maintenance.

The SGV sliding pad assembly has sliding bearings. However, based on commercial PWR experience for similar sliding pad assemblies, no maintenance is required.

The SGV snubber assembly has hydraulic snubbers, and bearings at various hinge points of the snubber lever. However, based on commercial PWR experience for a similar snubber assembly, no maintenance is required.

If necessary, any corrective maintenance can be performed with the module off-line.

Based on the ASME Boiler and Pressure Vessel Code ISI requirements, only visual inspection of the VSSS components is required. The steam generator vessel snubbers, as in PWR practices, shall be subjected to a periodic mechanical testing. The state-of-the-art allows in-place testing of only very small size snubbers. However, this approach is unattractive for MHTGR due to the large number of small-sized snubbers required. Therefore, two (2) snubbers and lever assemblies, as in a typical PWR-NSSS, are used for the

MHTGR-VSSS. It is planned to provide an access-path in the RB for removal and reinstallation of snubbers to perform mechanical testing outside the RB during an extended plant outage. The VSSS does not require any material surveillance program.

SECTION 8

SUBSYSTEM DECOMMISSIONING

[LATER]

SECTION 9

REFERENCES

1-1 "Overall Plant Design Specification (OPDS) Modular High Temperature Gas-Cooled Reactor," HTGR-86-004, Rev. 4, may 1987.

1-2 "MHTGR Plant Design Basis Transient Analysis," DOE-HTGR-86-121, Rev. 1, April 1987.

1-3 "Nuclear Island ISI/Surveillance Assessment, MHTGR," DOE-HTGR-86-026, May 1986.

1-4 "Reliability Allocations for the Standard HTGR," DOE-HTGR-85-008, February 1987.

1-5 "Top-Level Regulatory Criteria for the Standard MHTGR," DOE-HTGR-85-002, Rev. 2, October 1986.

2-1 "Vessel Support Subsystem Conceptual Design Report, Modular High Temperature Gas Cooled Reactor," to be Published.

2-2 "PDCO Informal Design Review of the MHTGR Vessel Support Subsystem," PDCO Report PDCO-189-87, April 24, 1987.

7-1 "Nuclear Steam Supply System Maintainability Assessment," DOE-HTGR-86-053, June 1986.

APPENDIX A
TRACEABILITY OF REQUIREMENTS

This appendix provides traceability of requirements to sources in external documents. The requirement traceability summary (Table A-1) identifies the 8 requirement as outlined below and identifies the source. Table A-2 is a list of the references which are identified as sources in Table A-1. Traceability is given for requirements contained in Sections 1, 3 and 4. Requirements in the remaining sections have not been completed to a large extent and is therefore omitted.

Each requirement is given a traceability number which is composed of three groups of digits. The first digit identifies the system and subsystem numbers (e.g., 11 for the Vessel System, 07 for the Vessel Support Subsystem), the second identifies the section and subsection numbers (e.g., 01 for Section 1 and 02 for Subsection 1.2), and the third identifies the requirement.

TABLE A-1

REQUIREMENTS TRACEABILITY SUMMARY

Traceability Number	Source Reference/Section	Traceability Number	Source Reference/Section
1107.0102.001	A-1/3.2.2.1	1107.0102.041	A-1/3.2.2.7
002	A-2/4.1 A-3/4.1	042	A-1/3.2.2.7
003	A-1/3.2.2.1	046	A-1/3.2.2.8
004	A-1/3.2.2.1	051	A-1/3.2.2.9
005	A-1/3.2.2.1	052	A-1/3.2.2.9
006	A-3/4.1	053	A-1/3.2.2.9
007	A-4/4.1	054	A-1/3.2.2.9
008	A-3/4.1	055	A-1/3.2.2.9
011	A-1/3.2.2.2	056	A-1/3.2.2.9
012	A-1/3.2.2.2	061	A-1/3.2.2.10
013	A-1/3.2.2.2	066	A-1/3.2.2.11
014	A-1/3.2.2.2	071	A-1/3.2.2.12
021	A-1/3.2.2.3	076	A-1/3.2.2.13
022	A-1/3.2.2.3		
023	A-1/3.2.2.3		
024	A-1/3.2.2.3		
025	A-1/3.2.2.3		
026	A-1/3.2.2.3		
027	A-1/3.2.2.3		
028	A-1/3.2.2.3		
031	A-1/3.2.2.4		
036	A-1/3.2.2.6		
037	A-1/3.2.2.7		

TABLE A-1 (Continued)

REQUIREMENTS TRACEABILITY SUMMARY

Traceability Number	Source Reference/Section	Traceability Number	Source Reference/Section
1107.0302.001	A-5	1107.0302.041	A-1/3.2.2.7
002	A-2/4.1 A-3/4.1	042	A-1/3.2.2.7
003	A-1/3.2.2.1	046	A-1/3.2.2.8
004	A-1/3.2.2.1	051	A-1/3.2.2.9
005	A-1/3.2.2.1	052	A-1/3.2.2.9
006	A-3/4.1	053	A-1/3.2.2.9
		054	A-1/3.2.2.9
		055	A-1/3.2.2.9
011	A-1/3.2.2.2	056	A-1/3.2.2.9
012	A-1/3.2.2.2	061	A-1/3.2.2.10
013	A-1/3.2.2.2	066	A-1/3.2.2.11
014	A-1/3.2.2.2	071	A-1/3.2.2.12
021	A-1/3.2.2.3	076	A-1/3.2.2.13
022	A-1/3.2.2.3		
023	A-1/3.2.2.3		
024	A-1/3.2.2.3		
025	A-1/3.2.2.3		
026	A-1/3.2.2.3		
027	A-1/3.2.2.3		
031	A-1/3.2.2.4		
036	A-1/3.2.2.6		
037	A-1/3.2.2.7		

TABLE A-1 (Continued)

REQUIREMENTS TRACEABILITY SUMMARY

Traceability Number	Source Reference/Section
1107.0302.101	A-5
102	A-2/4.1
103	A-1/3.2.2.1
104	A-1/3.2.2.1
105	A-4/4.1
111	A-1/3.2.2.2
112	A-1/3.2.2.2
113	A-1/3.2.2.2
114	A-1/3.2.2.2
121	A-1/3.2.2.3
122	A-1/3.2.2.3
123	A-1/3.2.2.3
124	A-1/3.2.2.3
125	A-1/3.2.2.3
126	A-1/3.2.2.3
127	A-1/3.2.2.3
131	A-1/3.2.2.4
136	A-1/3.2.2.6
137	A-1/3.2.2.7
1107.0302.141	A-1/3.2.2.7
142	A-1/3.2.2.7
146	A-1/3.2.2.8
151	A-1/3.2.2.9
152	A-1/3.2.2.9
153	A-1/3.2.2.9
154	A-1/3.2.2.9
155	A-1/3.2.2.9
156	A-1/3.2.2.9
161	A-1/3.2.2.10
166	A-1/3.2.2.11
171	A-1/3.2.2.12
176	A-1/3.2.2.13

TABLE A-1 (Continued)

REQUIREMENTS TRACEABILITY SUMMARY

Traceability Number	Source Reference/Section	Traceability Number	Source Reference/Section
1107.0302.201	A-5	1107.0302.241	A-1/3.2.2.7
202	A-2/4.1	242	A-1/3.2.2.7
203	A-1/3.2.2.1	246	A-1/3.2.2.8
204	A-1/3.2.2.1	251	A-1/3.2.2.9
205	A-4/4.1	252	A-1/3.2.2.9
		253	A-1/3.2.2.9
		254	A-1/3.2.2.9
		255	A-1/3.2.2.9
211	A-1/3.2.2.2	256	A-1/3.2.2.9
212	A-1/3.2.2.2	261	A-1/3.2.2.10
213	A-1/3.2.2.2	266	A-1/3.2.2.11
214	A-1/3.2.2.2	271	A-1/3.2.2.12
221	A-1/3.2.2.3	276	A-1/3.2.2.13
222	A-1/3.2.2.3		
223	A-1/3.2.2.3		
224	A-1/3.2.2.3		
225	A-1/3.2.2.3		
226	A-1/3.2.2.3		
227	A-1/3.2.2.3		
231	A-1/3.2.2.4		
236	A-1/3.2.2.6		
237	A-1/3.2.2.7		

TABLE A-1 (Continued)

REQUIREMENTS TRACEABILITY SUMMARY

Traceability Number	Source Reference/Section	Traceability Number	Source Reference/Section
1107.0401.001	A-5		
002	A-5		
003	A-5		
004	A-5		
005	A-5		
006	A-5		
007	A-5		
008	A-5		
009	A-5		
010	A-5		
011	A-5		
012	A-5		
013	A-5		
101	A-5		
102	A-5		
103	A-5		
104	A-5		
105	A-5		

TABLE A-2

REFERENCES FOR REQUIREMENTS TRACEABILITY

A-1 "Vessel System Design Description," DOE-HTGR-86-125, Rev. 1, HFD-31100 July 1987.

A-2 "Vessels and Duct Subsystem Design Description," DOE-HTGR-86-126, Rev.1, HFD-41106 July 1987.

A-3 "Reactor Cavity Cooling System Design Description," DOE-HTGR-86-064, Rev.0, HFD-35600, July 1987.

A-4 "Reactor Building Subsystem Design Description," DOE-HTGR-87-065, Rev. 0, HFD-47001, July 1987.

A-5 "Vessel Support Subsystem Conceptual Design Report, Modular High Temperature Gas Cooled Reactor," to be Published.

APPENDIX B
DRAWING LIST

The Vessel Support Subsystem configuration and interfaces are shown on the following drawings:

Drawing No.	Revision	Title
SE-7085-800-003	1	Vessel System Arrangement
SE-7085-XXX-XXX	0	Vessel System Interface
SE-7085-800-001	1	Vessel Support Subsystem Arrangement

APPENDIX C

SYSTEM TRANSIENTS

1.0 Introduction

This appendix presents the thermal response of the Vessel Support Subsystem to the transient events in the MHTGR Plant Design Duty Cycle and to the Licensing-Basis Events (LBEs) (Ref. C-1). The duty cycle provides a design-level value for the number of occurrences of each event and the service level designation under Section III of the ASME Boiler and Pressure Vessel Code. These are sufficient to evaluate the ability of subsystems and components to withstand the duty cycle. In order to provide a basis to evaluate the response of the subsystem and components to the LBEs, ASME Section III service level are established (Ref. C-2). Detailed discussion of the transient analysis and the overall results are presented in Ref. C-3. Additional summary discussion on the system transients is presented in the Vessel System Design Description (VSDD) (Ref. C-4).

No specific transient results are available for the thermal response of the VSSS to various transient events. Therefore, enveloping (worst loading) the thermal response of the VSSS to various transients is estimated from the available thermal response of the VS (Ref. C-4).

2.0 Duty Cycle Transient Results

A discussion on events in the design duty cycle is provided in the VSSD, Appendix C (Ref. C-4). The available results indicate that the Safety-Related Design Conditions (SRDCs), Subsection 3.3 of this appendix, enveloped all duty cycle transient results. Therefore, details of this section have been omitted.

3.0 Transient Results For LBEs

The following subsections describe the licensing-basis events (LBEs) which are categorized as anticipated operational occurrences (AOOs), design-basis events (DBEs), and safety-related design conditions (SRDCs).

3.1 Anticipated Operational Occurrences

A discussion on events in the Anticipated Operational Occurrences (AOOs) is provided in the VSDD, Appendix C (Ref. C-4). The available results of the plant-level thermal transient analysis indicate that the Safety-Related Design Conditions (SRDC), Subsection 3.3 of this appendix, enveloped all AOO transient results. Therefore, details of this subsection have been omitted.

3.2 Design-Basis Events

A discussion on the Design-Basis Events (DBEs) is provided in the VSDD, Appendix C (Ref. C-4). The available results of the plant-level thermal transient analysis indicate that the Safety-Related Design Conditions (SRDCs), Subsection 3.3 of this appendix, enveloped all AOO transient results. Therefore, details of this subsection have been omitted.

3.3 Safety-Related Design Conditions

These events all involve the loss of both HTS and SCS cooling. Core residual and decay heat removal is by conduction and radiation to the RCCS. Only events in the reactor cavity are modeled and discussed. These events are all designated as ASME Service Level D for purposes of structural evaluation.

Review of the thermal response of the VS to various SRDC events indicate that the controlling events for the VSSS design, in particular the RV support column design, are enveloped by the three SRDCs discussed in the following subsections. Details of other SRDC events, omitted in this Section, are available in the VSDD, Appendix C (Ref. C-4).

From the results discussed in the following subsections, it is concluded that the design temperature of the VSSS adequately envelopes the thermal response of the RV support column. The impact of the thermal transients on other components of the VSSS, for which results are not available, is anticipated to be considerably lower.

3.3.1 SRDC-1 Pressurized Conduction Cooldown

This event is initiated by a loss of offsite power and turbine trip. The reactor is successfully tripped. The HTS and SCS are not available for cooling, but heat is conducted and radiated to the RCCS. The coolant remains pressurized.

The temperature distribution within the vicinity of the top of the RV support column (at RV surface) is shown in Figure C-1. The peak temperature at the RV outer surface, near the top flange of the RV support column, is below 400°F.

3.3.2 SRDC-5 Pressurized Conduction Cooldown With Earthquake (SSE)

A large earthquake with ground accelerations of 0.3g to 0.5g occurs. The main loop trips causing the reactor to trip on high power-to-flow ratio. The SCS fails to start. The thermal history of this event is the same as shown for SRDC-1 discussed in Section 3.3.1.

3.3.3 SRDC-10 Depressurized Conduction Cooldown with Moderate Primary Coolant Leak

A moderate size leak of 81.9 cm^2 (12.7 in^2) occurs in the primary coolant system which is depressurized in 6 minutes. The reactor is tripped on low pressure at 20 s, and soon after the HTS is tripped at a lower pressure. The HTS circulator coastdown to no forced circulation, and the SCS fails to start. Core heat is removed by conduction and radiation to the RCCS.

The temperature distribution within the vicinity of the top of the RV support column (at RV surface) is shown in Figure C-1. The peak temperature at the RV outer surface, near the top flange of the support column is below 500°F.

(1) SRDC-1, LEVEL D PRESSURIZED CONDUCTION COOLDOWN

(2) SRDC-5, LEVEL D PRESSURIZED CONDUCTION COOLDOWN WITH EARTHQUAKE (SSE)

(3) SRDC-10, LEVEL D DEPRESSURIZED CONDUCTION COOLDOWN WITH MODERATE PRIMARY COOLANT LEAK

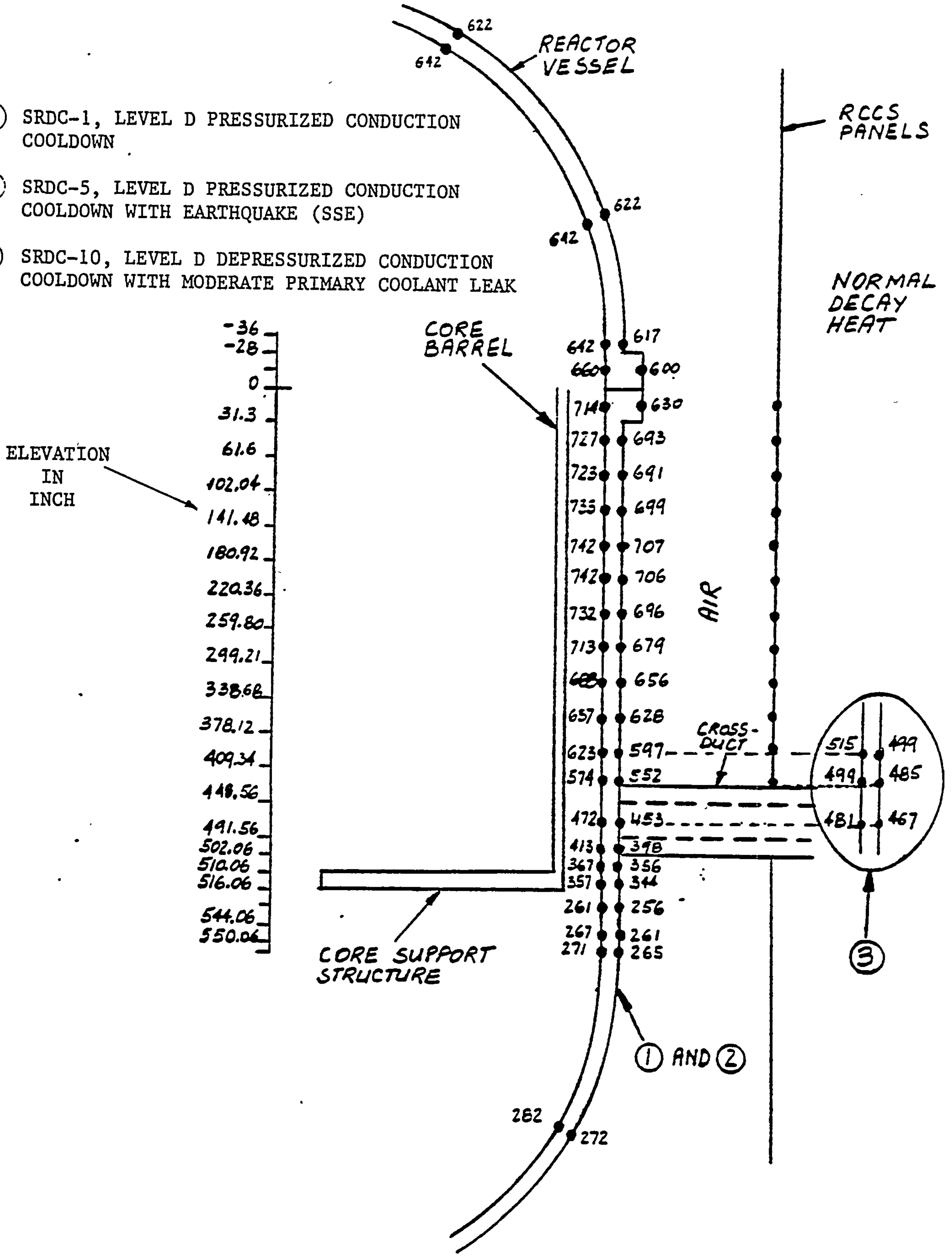


FIGURE C-1
REACTOR VESSEL SUPPORT COLUMN
ENVELOPING PEAK TEMPERATURE

References

C-1 Bellis, E.A. "Licensing-Basis Events for the Modular HTGR." HTGR-86-034. Issued by G.A. Technologies, San Diego, CA, April 18, 1986.

C-2 Breher, W.E. and Lommers, L. Initial Guidance for Evaluation of Component Response During LBEs/SRDCs. GA Internal Correspondence SE:WEB: LL: 021:86, 5/16/86.

C-3 Chan, T.W. and Richards, M.B. MHTGR Plant Design Basis Transient Analysis. HTGR-86-121. Issued by G.A. Technologies, San Diego, CA, September, 1986.

C-4 U. S. Department of Energy. Vessel System Design Description. DOE-HTGR-86-125, Rev. 1, HFD-31100, Issued by Combustion Engineering, Inc., Windsor, Ct., July 1987.

APPENDIX D

DESIGN BASIS SEISMIC INPUT

The subsystems and its components of the Vessel System must withstand seismic loadings as part of the duty cycle. The seismic loads are transmitted from the ground to the Vessel Support Subsystem and then to the reactor and steam generator vessels. The Service Level B seismic event in the duty cycle is an operating basis earthquake (OBE) with a zero-period acceleration (ZPA) of 0.15 g at the ground surface (Refs. D-1 and D-2). Figures D-2, D-3, and D-4 show the design acceleration spectra for the X, Y, and Z directions which are indicated on the nodal model of Fig. D-1. The results of the structural analysis are listed Table D-1 for acceleration, Table D-2 for deflections, Tables D-3 and D-4 for vessel stresses, and Table D-5 for vessel support loads. The details are discussed in Ref. D-1.

The Safety-Related Design Conditions include the safe shutdown earthquake (SSE) with a ZPA of 0.3 g at the ground surface. The design acceleration spectra for the X, Y, and Z directions are shown on Fig. D-5, D-6, and D-7. In absence of an explicit calculation of the SSE, the SSE design spectra is obtained by multiplying the OBE spectra by a constant factor of 2 (Ref. D-2). The resulting accelerations, deflections, vessel stresses, and vessel support loads are obtained from the corresponding OBE quantity multiplied by a factor of 2.

References

D-1 U.S. Department of Energy. FY86 Seismic Assessment, 4 x 300 MW(T) Modular HTGR Plant. DOE-HTGR-86-119. Issued by G.A. Technologies San Diego, CA, September 1986.

D-2 U.S. Department of Energy. Overall Plant Design Specification Modular HTGR. DOE-HTGR-86-004/Rev. 4. Issued by G.A. Technologies San Diego, CA, May 1987.

Table D-1
VESSEL SYSTEM AND STEAM GENERATOR SEISMIC RESPONSE
OBE 0.15 g ZPA
NODAL ACCELERATIONS (g)

	NODE	X	Y	Z
REACTOR VESSEL	1	1.07	.38	.77
	2	1.04	.37	.73
	3	.90	.37	.60
	4	.81	.37	.53
	5	.77	.37	.49
	6	.76	.37	.49
	7	.70	.37	.44
	8	.60	.37	.36
	9	.47	.36	.30
	10	.46	.36	.30
	11	.42	.36	.31
	12	.39	.35	.32
	13	.38	.35	.34
	14	.37	.36	.47
CROSS DUCT	61	.42	.38	.30
	62	.43	.35	.28
	63	.43	.33	.26
	64	.44	.17	.25
	65	.45	.04	.36
	66	.45	.04	.36
	54	.45	.19	.43
ST. GEN. VESSEL	15	.53	.23	.37
	16	.52	.23	.34
	17	.50	.23	.31
	18	.48	.23	.30
	19	.46	.23	.27
	20	.45	.23	.27
	21	.46	.23	.29
	22	.46	.23	.29
	23	.47	.23	.31
	24	.51	.26	.40
	25	.58	.28	.48
	26	.67	.30	.55
	27	.72	.30	.59
ST. GEN. BUNDLE	117	.53	.036	.040
	118	.61	.036	.048
	119	.66	.035	.052
	120	.75	.035	.061
	121	.67	.033	.060
	122	.67	.030	.055

Table D-2
VESSEL SYSTEM AND STEAM GENERATOR SEISMIC RESPONSE
OBE 0.15 g ZPA
NODAL DEFLECTIONS (INCHES)

	NODE	X	Y	Z
REACTOR VESSEL	1	.199	.029	.180
	2	.193	.029	.171
	3	.168	.029	.143
	4	.153	.029	.126
	5	.145	.028	.116
	6	.144	.028	.115
	7	.133	.028	.101
	8	.113	.028	.075
	9	.086	.028	.043
	10	.083	.028	.040
	11	.074	.027	.031
	12	.066	.027	.027
	13	.063	.027	.028
	14	.048	.027	.048
CROSS DUCT	61	.075	.034	.030
	62	.075	.033	.029
	63	.075	.031	.029
	64	.076	.016	.033
	65	.078	.002	.045
	66	.078	.002	.046
	54	.078	.009	.054
ST. GEN. VESSEL	15	.095	.007	.039
	16	.092	.007	.036
	17	.089	.007	.034
	18	.087	.007	.033
	19	.082	.007	.031
	20	.078	.007	.032
	21	.077	.007	.034
	22	.077	.007	.035
	23	.076	.007	.036
	24	.073	.008	.047
	25	.071	.009	.057
	26	.071	.009	.065
	27	.075	.009	.070
ST. GEN. BUNDLE	117	.080	.011	.047
	118	.083	.011	.056
	119	.085	.011	.062
	120	.088	.011	.071
	121	.082	.010	.071
	122	.071	.009	.065

Table D-3
VESSEL SYSTEM AND STEAM GENERATOR SEISMIC RESPONSE
OBE 0.15 g ZPA
STRESSES (psi)

	NODE	σ TENSILE	σ BEND-1	$\sigma_T + \sigma_{B1}$	σ BEND-2	$\sigma_T + \sigma_{B2}$
REACTOR VESSEL	2	30	180	210	240	270
	3	30	360	390	490	520
	4	40	240	280	330	370
	5	40	280	320	390	440
	6	50	280	320	390	430
	7	80	360	440	490	570
	8	90	310	400	420	510
	9	90	350	440	430	520
	10	70	250	330	310	380
	11	130	280	400	330	450
	12	130	300	430	270	390
	13	160	50	200	50	210
CROSS DUCT	62	720	310	1030	1070	1790
	63	720	280	1000	990	1710
	64	710	80	790	470	1190
	65	710	80	800	330	1050
	66	700	90	790	340	1040
ST. GEN. VESSEL	17	10	10	20	20	20
	18	20	30	50	40	60
	19	20	90	110	140	160
	20	70	100	170	200	260
	21	100	110	220	220	330
	22	80	140	220	180	270
	23	110	220	330	230	340
	24	110	220	330	220	320
	25	100	160	260	150	250
	26	80	30	110	30	110

See Figure D-8

Table D-4
VESSEL SYSTEM AND STEAM GENERATOR SEISMIC RESPONSE
OBE 0.15 g ZPA
VESSEL SEISMIC STRESSES AT ATTACHMENTS (psi)

NODE	CIRCUMFERENTIAL STRESS	AXIAL STRESS
41	4060	3470
42	4820	4120
43	4820	4120
44	7660	7520
45	11620	9590
46	11620	9590
61	4500	4000
54	4000	3100
49	5110	4840
50	4230	3760
51	3860	3430
52	2710	2410
53	3860	3430

TABLE D-5

Vessel System Seismic Restraint Reaction Loads

OBE 0.15g ZPA

REACTION FORCES (LB)			
NODES	FX	FY	FX
55	--	231,300	113,500
241	--	--	701,400
242	--	--	836,000 *
243	--	--	836,000 *
Key Loads			
244	--	--	358,400
245	--	--	751,400 *
246	--	--	751,400 *
Column Loads			
244	10,000	644,200	4,700
245	25,400	729,100	8,900
246	25,400	729,100	8,900
249	--	238,500	112,200
250	--	--	162,500
251	147,900	--	--
252	--	--	103,800
253	147,900	--	--
344	10,100	644,200	4,700
345	20,400	729,100	17,700
346	20,400	729,100	17,700

REACTION MOMENTS (IN-LB)			
NODE	MX	MY	MZ
244	729,200	1,332,000	1,453,300
245	1,320,200	2,812,000	3,784,700
246	1,320,200	2,812,000	3,784,700
344	365,900	1,332,000	997,400
345	1,674,000	2,812,000	1,998,000
346	1,674,000	2,812,000	1,998,000

*Forces are locally tangent to vessel

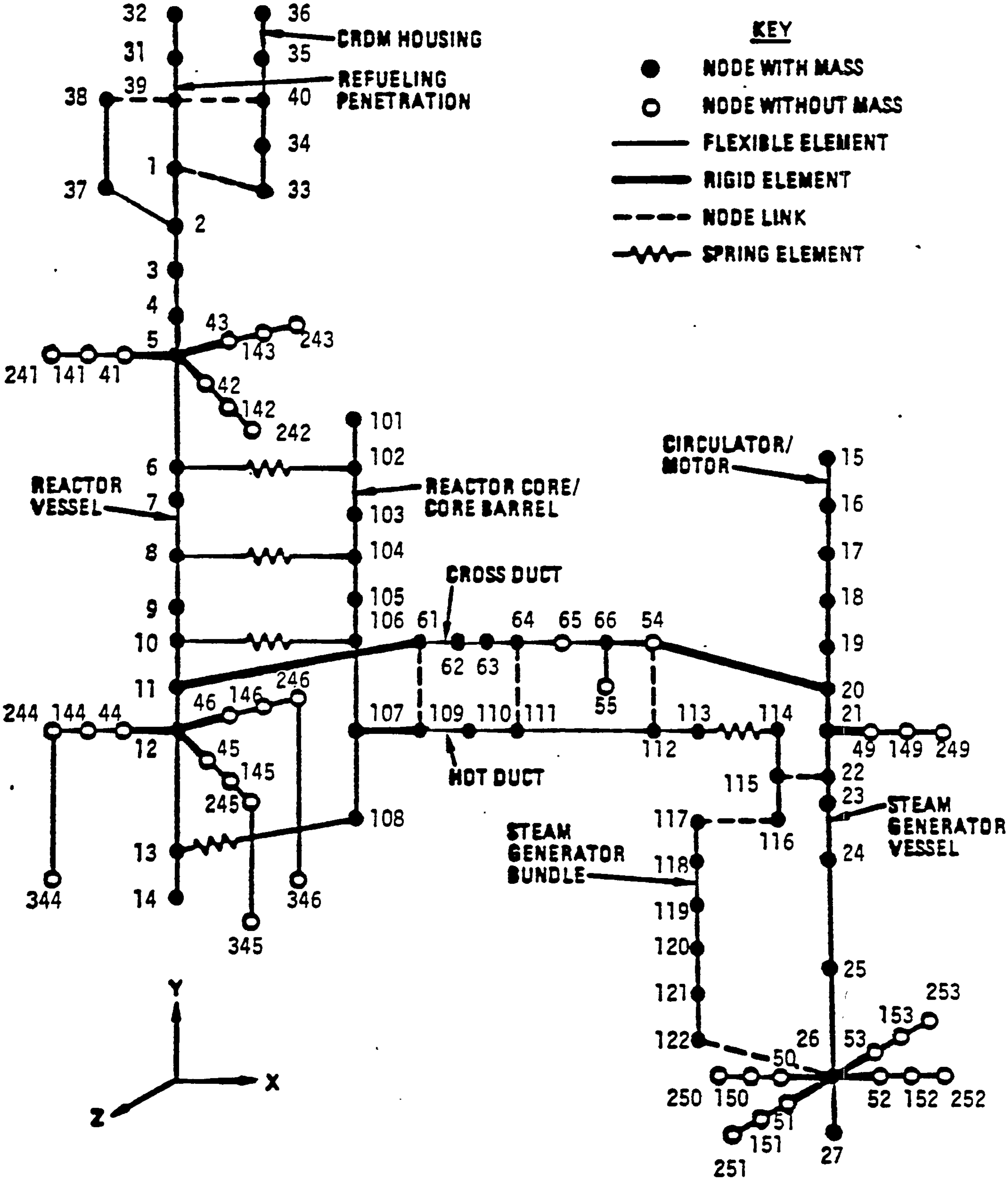


Figure D-1 Reactor Vessels and Internals Seismic Analysis Stick Model

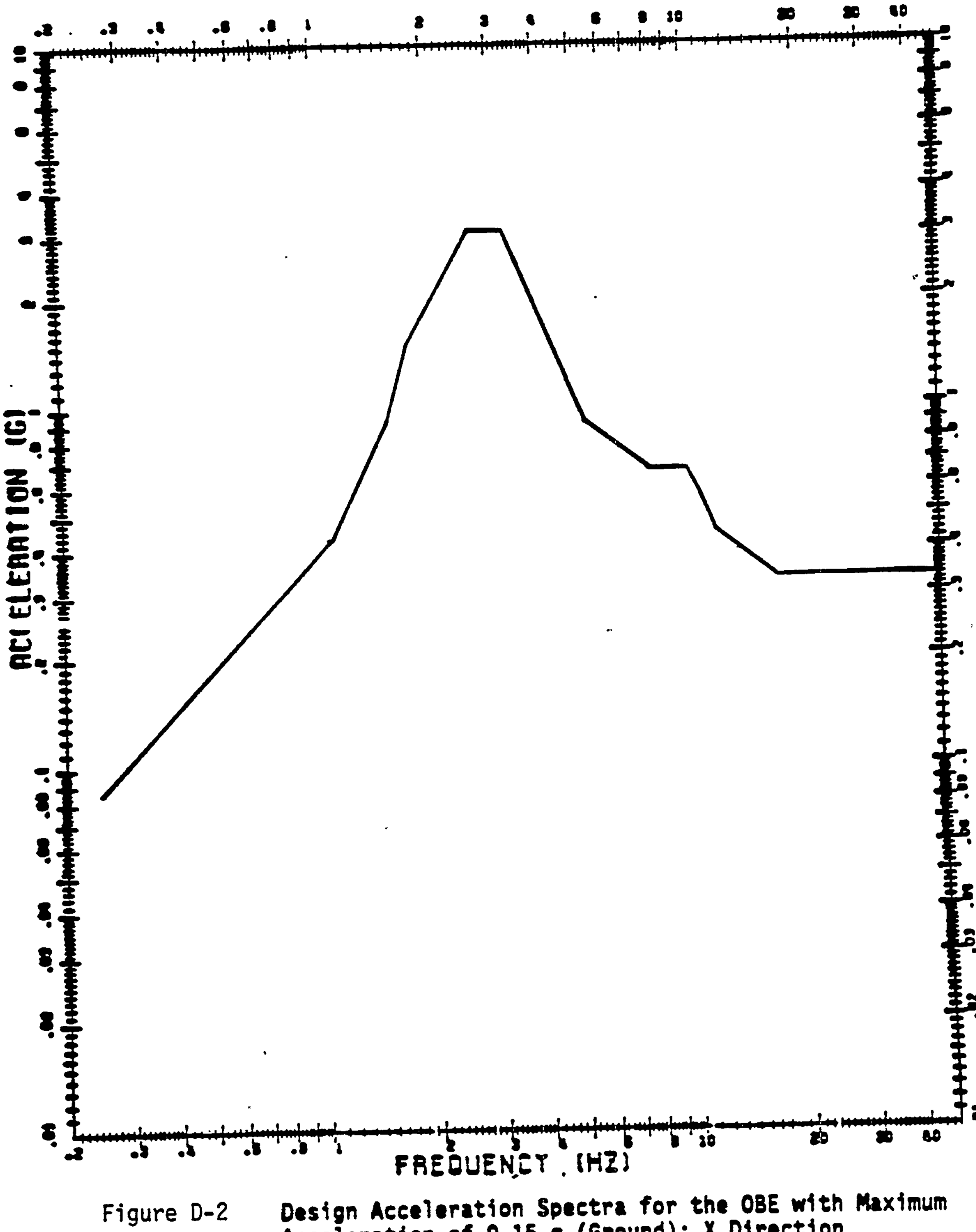


Figure D-2 **Design Acceleration Spectra for the OBE with Maximum Acceleration of 0.15 g (Ground); X Direction**

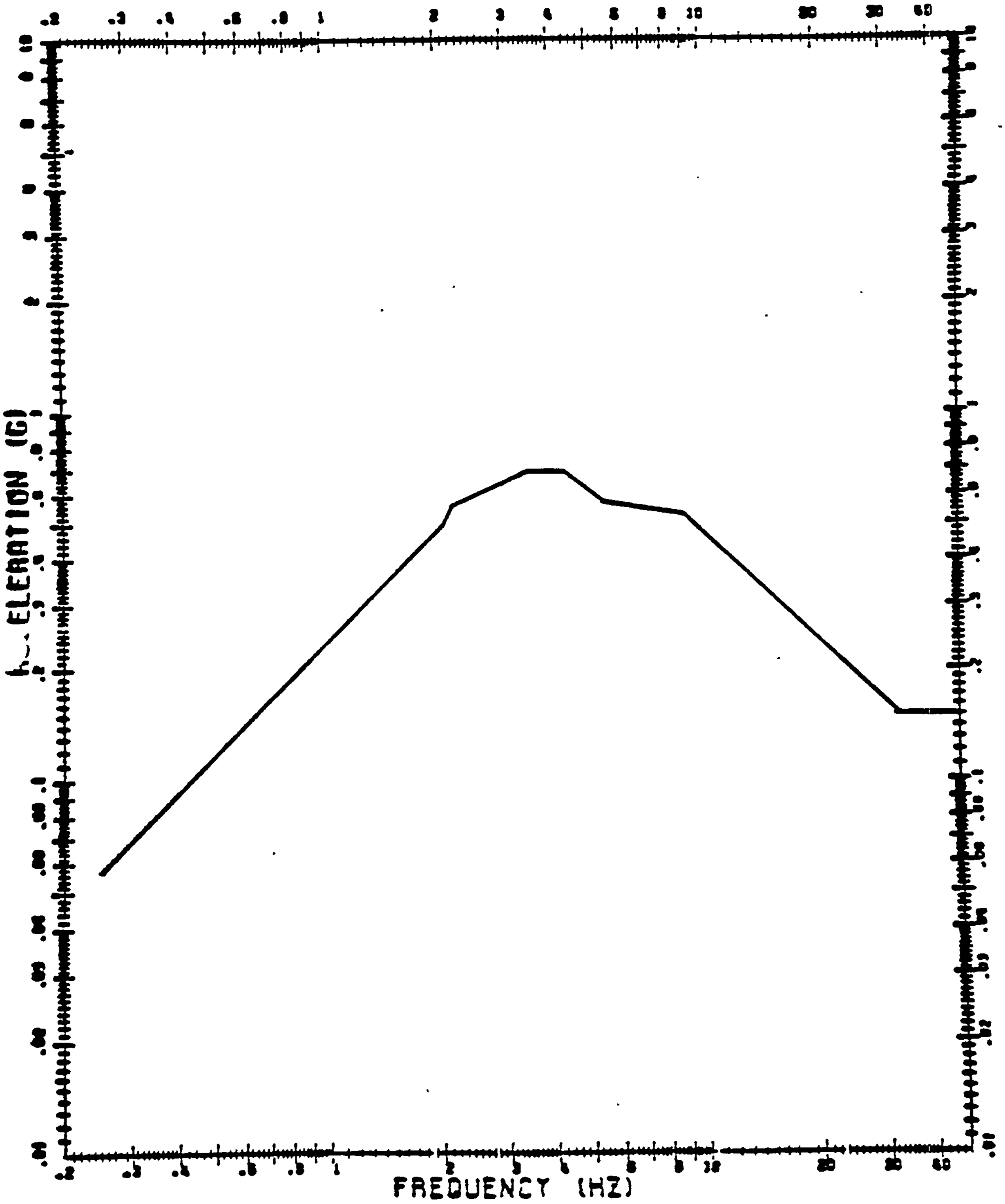


Figure D-3 **Design Acceleration Spectra for OBE with Maximum Acceleration of 0.15 g (Ground); Y Direction**

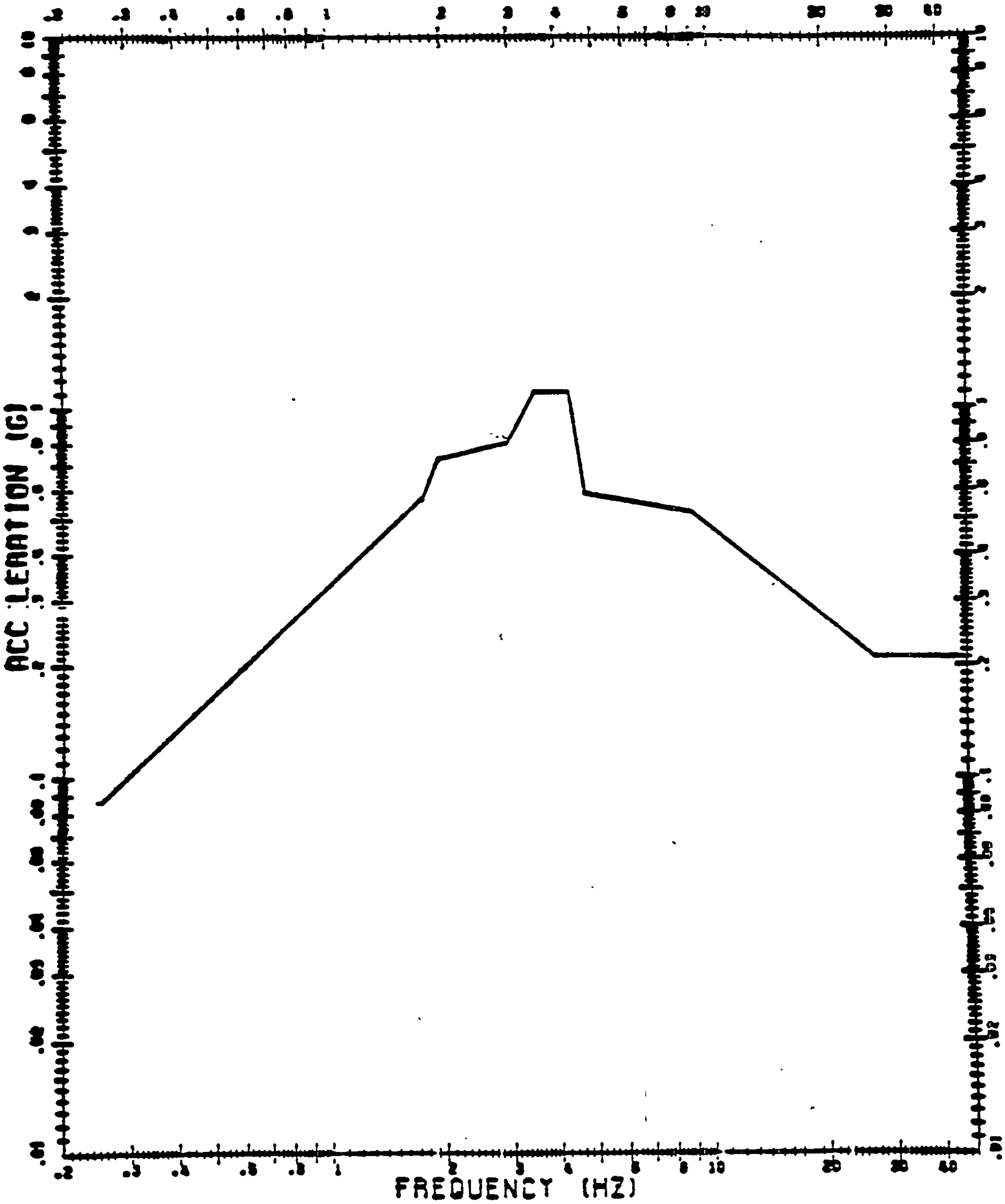


Figure D-4 **Design Acceleration Spectra for OBE with Maximum Acceleration of 0.15 g (Ground); Z Direction**

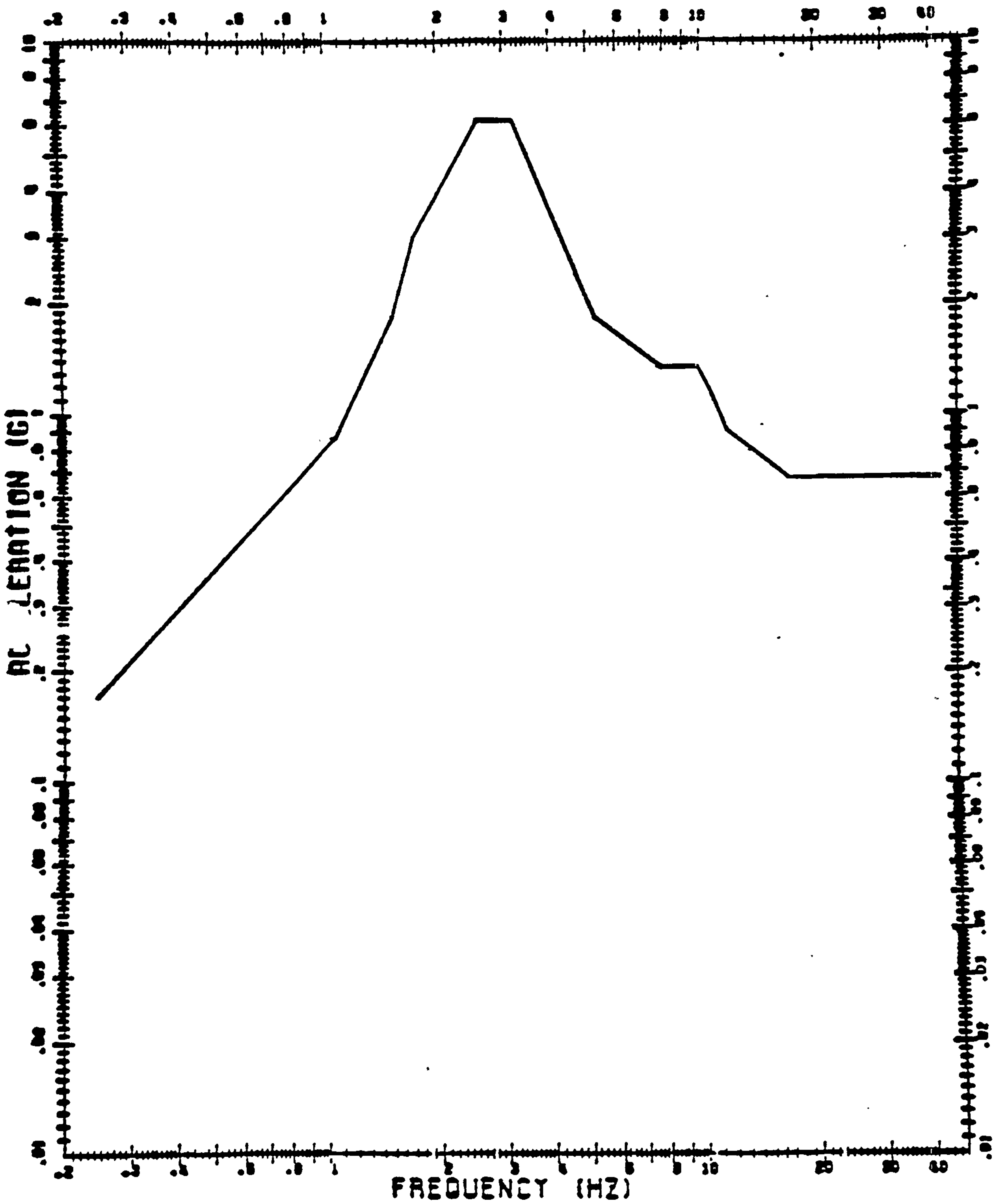
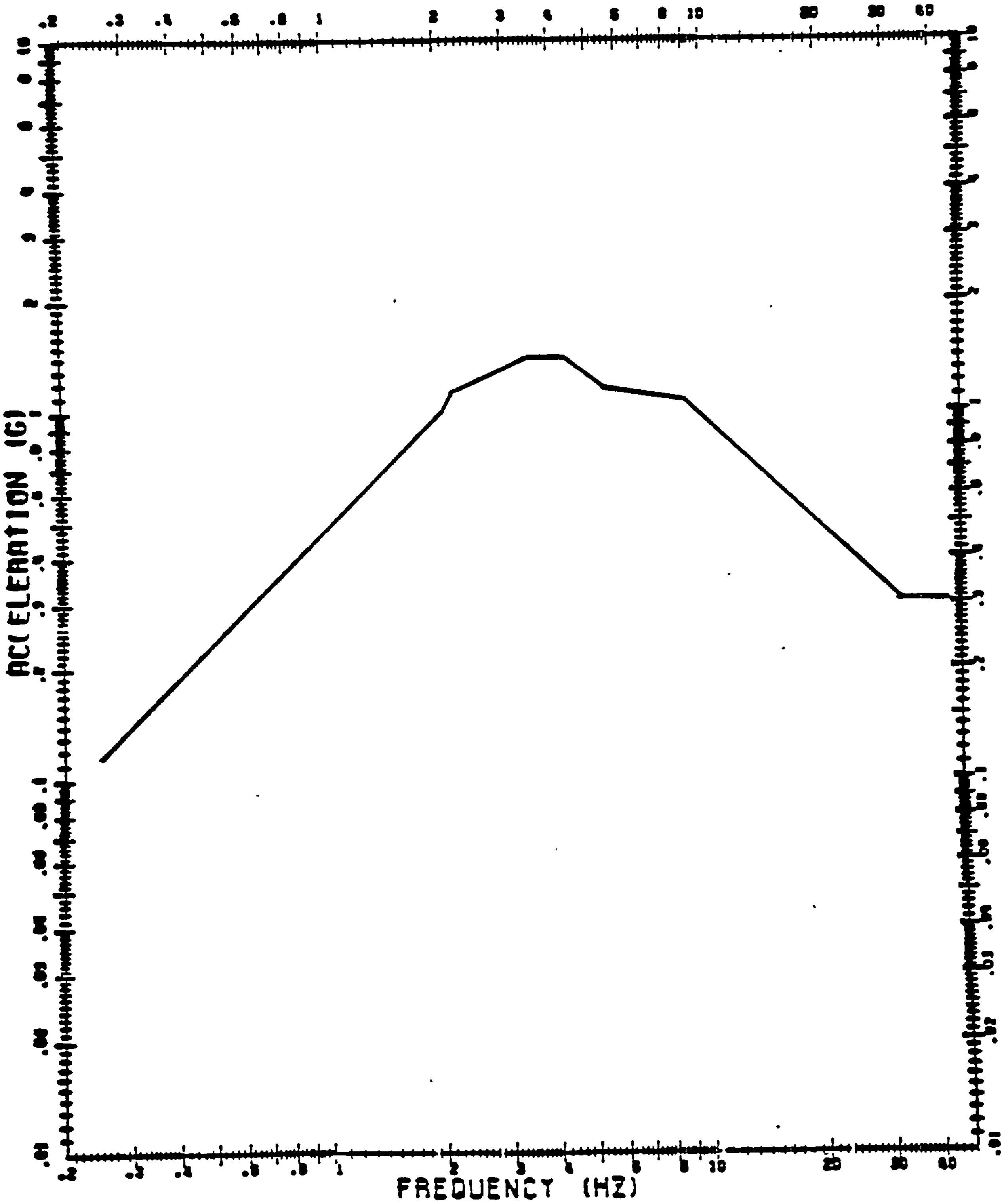
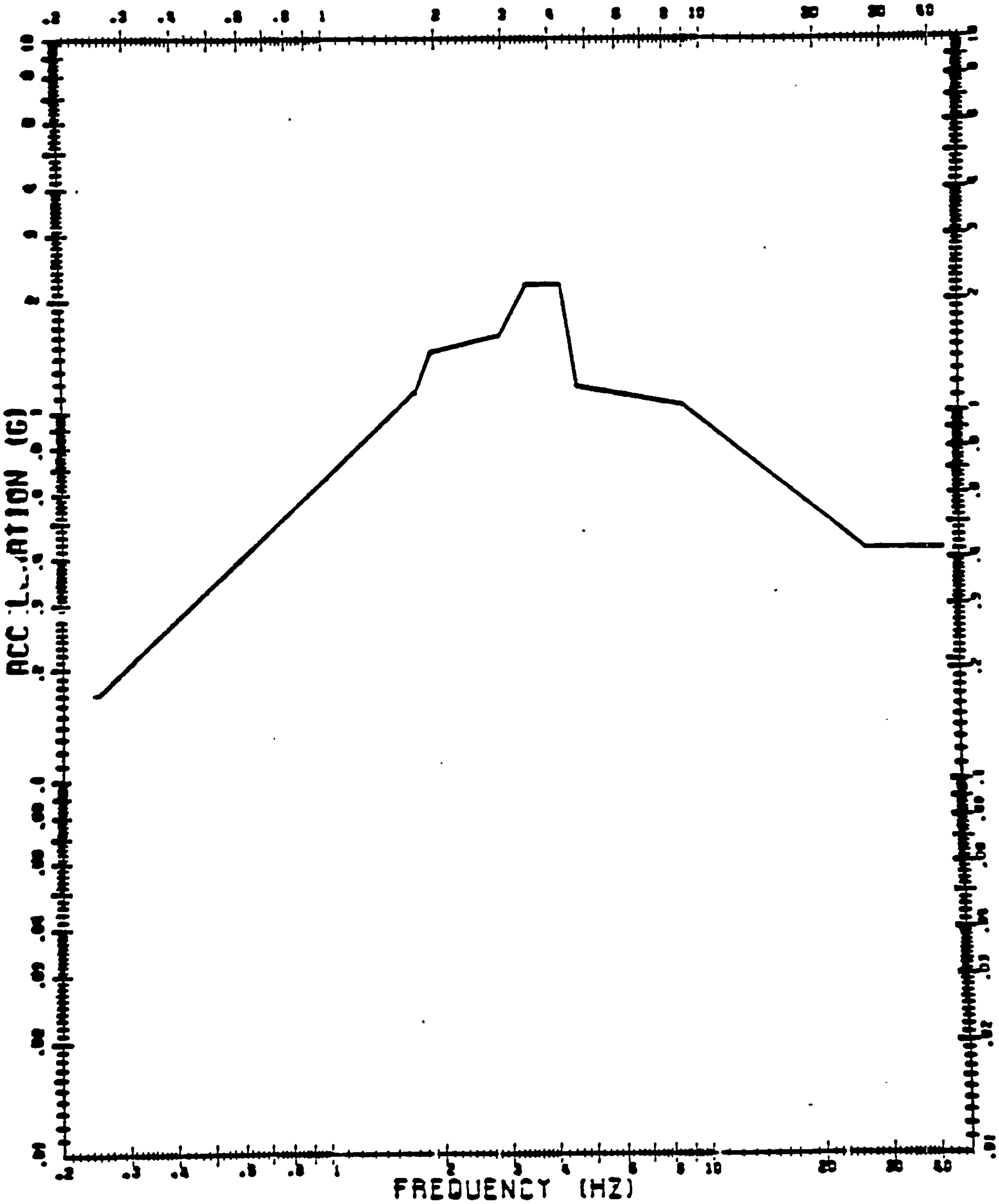


Figure D-5 Design Acceleration Spectra for SSE [redacted] with Maximum Acceleration of 0.30 g (Ground); X Direction



SSE
Figure D-6 Design Acceleration Spectra for [redacted] with Maximum Acceleration of 0.30 g (Ground); Y Direction



SSE
Figure D-7 Design Acceleration Spectra for [redacted] with Maximum Acceleration of 0.30 g (Ground); Z Direction

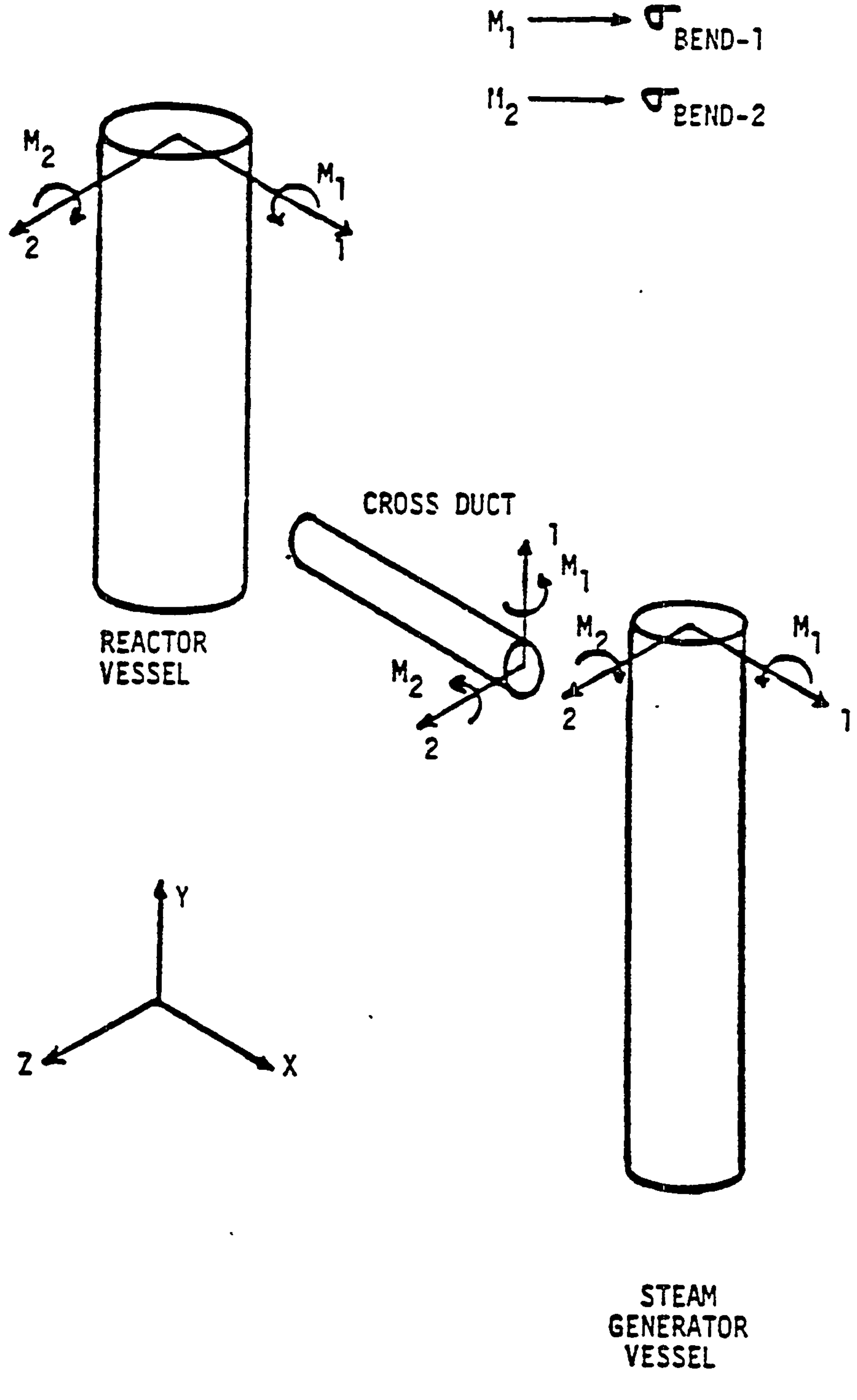


Figure D-8 NSSS VESSEL BENDING STRESS AXIS DEFINITION

APPENDIX E

EQUIPMENT LIST

A list of equipment including component, part, and quantity is depicted in Table E-1.

Table E-1

VESSEL SUPPORT SUBSYSTEM EQUIPMENT LIST

	COMPONENT	PART	QUANTITY
o	Reactor Vessel		
	Column Support		3
o	Steam Generator Vessel		
	Sliding Pad Assembly		2
		Bearing Socket Pad	2
		Spherical Bearing	2
		Bearing Plate	2
o	Steam Generator Vessel		
	Snubber Assembly		2
		Lever Bracket	2
		Lever Assembly	2
		Snubber	2
		Link Assembly	4
		Snubber Wall Bracket	2

APPENDIX F
PARAMETERS LIST

The primary design parameters of the Vessel Support Subsystem are listed in Table E-1. Dimensions in Table E-1 are the best estimates at the conceptual design stage and are provided here for guidelines to interfacing systems and components.

TABLE F-1

VESSEL SUPPORT SUBSYSTEM PARAMETERS LIST

Design Life, Years	40
Reactor Vessel Support Column	
Number	3
Material	ASME SA-508, class 2
Temperature Design, °C (°F)	288 (550)
Temperature, Operating (Top), °C (°F)	TBD
Temperature, Operating (Bottom), °C (°F)	TBD
Neutron Fluence, Maximum (Top), Total, n/cm^2	$4x10^{15}$
Height Including Flanges, m (ft)	6.1 (20.0)
Column Width, cm (in)	76.2 (30.0)
Column Depth, cm (in)	29.2 (11.5)
Top Flange Height, cm (in)	15.2 (6.0)
Top Flange Width, cm (in)	132.1 (52.0)
Top Flange Depth, cm (in)	76.2 (30.0)
Bottom Flange Height, cm (in)	25.4 (10.0)
Bottom Flange Width, cm (in)	185.4 (73.0)
Bottom Flange Depth, cm (in)	134.6 (53.0)
Steam Generator Sliding Pad Assembly	
Number	2
Material, Bearing (slide)	Meehanite GA50
Material, Bearing Plate	Stellite 6-B
Material, Bearing Socket Pad	Carbon Steel
Temperature, Design, °C, (°F)	121 (250)
Temperature, Operating, °C (°F)	TBD
Neutron Fluence, Maximum (Estimated), Total, n/cm^2	$<<10^{15}$
Bearing Height, cm (in)	7.6 (3.0)
Bearing Top Curvature, Spherical Radius, cm (in)	56.0 (22.0)
Bearing Flat Bottom Diameter, cm (in)	48.3 (19.0)
Bearing Socket Pad Height, cm (in)	8.9 (3.5)
Bearing Socket Pad Width, cm (in)	56.0 (22.0)
Bearing Socket Pad Depth, cm (in)	56.0 (22.0)

TABLE F-1 (Cont'd)

VESSEL SUPPORT SUBSYSTEM PARAMETERS LIST

Bearing Plate, Height, cm (in)	0.6 (.25)
Bearing Plate, Width, cm (in)	56.0 (22.0)
Bearing Plate, Depth, cm (in)	56.0 (22.0)
Steam Generator Snubber Assembly	
Number	2
Temperature, Hydraulic Snubber, Design, °C (°F)	66 (150)
Temperature, Hydraulic Snubber, Operating, °C (°F)	TBD
Neutron Fluence, Maximum (Estimated), Total, n/cm^2	$<<<10^{15}$
Snubber Diameter (Estimated), cm (in)	61.0 (24.0)
Snubber Length (Estimated), cm (in)	91.4 (36.0)
Overall Assembly Height (Estimated), m (ft)	3.0 (10.0)
Overall Assembly Width (Estimated), m (ft)	3.0 (10.0)
Overall Assembly Depth (Estimated), m (ft)	0.6 (2.0)

APPENDIX G

PROPRIETARY CLAIMS

The VSSS concept employs certain features which are covered in the following two patents.

1. "Reactor Vessel Supported by Flexure Member," James Darwin Crawford and Bernard Pankow, U.S. Patent 3,916,944 dated November 4, 1975.

2. "Steam Generator Cradle Support," Michael A. Marroni, Jr. and Daniel A. Peck, U. S. Patent 3,771,499 dated November 13, 1973.