ABB

ASEA BROWN BOVERI

February 12, 1992
LD-92-016

Docket No. 52-002

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Response to NRC Requests for Additional Information

Reference: Letter, Structural and Geosciences Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated September 26, 1991

Dear Sirs:

The Reference requested additional information for NRC staff review of the Combustion Engineering Standard Safety Analysis Report - Design Certification (CESSAR-DC). Enclosure I to this letter provides our responses to a number of these questions including corresponding revisions to CESSAR-DC. Responses to the remaining questions of the Reference will be provided by separate correspondence.

Should you have any questions on the enclosed material, please contact me or Mr. Stan Ritterbusch of my staff at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

S. E. Ritterbusch for

C. B. Brinkman
Acting Director
Nuclear Systems Licensing

vs/lw
Enclosures: As Stated

cc: J. Trotter (EPRI)
T. Wambach (NRC)

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Enclosure I to
LD-92-016

RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION
STRUCTURAL AND GEOSCIENCES BRANCH

NRC RAI 210.1

NNS structures, systems, or components whose failure could cause flooding of safety-related structures, systems, or components are designed to Seismic Category I requirements.

The seismic category and safety and quality classification of structures, systems, and components within the System 80+ Standard Design are listed in Table 3.2-1 and on the P&IDs (Chapters 5, 6, and 9). Seismic Category I includes all mechanical components within the safety class boundaries and extends to the first seismic restraint beyond the boundary. All fuel racks are also designated as Seismic Category I. Structures, systems, or components whose failure could reduce the performance of a safety function by a Seismic Category I component to an unacceptable safety level are designed to Seismic Category II requirements for structural integrity only or are separated to the extent required to eliminate that possibility. This ensures that any structures, systems, or components that could potentially have a disabling interaction with Seismic Category I structures, systems, or components are either prevented from doing so or are designed to meet Seismic Category I or II structural integrity requirements, depending on the function of the component. I I D

~~The listing of major electrical components is found in Section 3.11, which also includes safety and quality classifications.~~ Electrical structures, systems, and components not classified as Seismic Category I but whose failure could represent a hazard to the operator or could interfere with the performance of required safety functions of electrical structures, systems and components, are classified as Seismic Category II (Reference 1). Any electrical system or structure or component not in Seismic Category I or II is considered Non-Seismic (see Section 3.10). The use of the Seismic Category II designation for electrical components is limited to non-safety control system components which are designed and documented to maintain structural integrity during an SSE. D

For purposes of this discussion, the motors and solenoids used to provide motive power to mechanical components are treated as part of the mechanical component.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS (SAFETY CLASS)

In general, fluid system components important to safety are classified in accordance with ANSI/ANS 51.1 (Reference 2). For purposes of CESSAR, Safety Class 1, 2, 3 and NNS of ANSI/ANS 51.1 are equivalent to Quality Groups A, B, C and D of Regulatory Guide 1.26. The criteria establish safety classes which are used as a guide to the selection of codes, standards, and quality assurance provisions for the design and construction of the components. The safety class designations are also used as a guide to those fluid system components to be classified as Seismic Category I and II (see Section 3.2.1). The Safety Class definitions in ANSI/ANS 51.1 are summarized as follows: D D

NRC RAI 210.1

Safety classes and safety class changes are shown on the system P&IDs.

A. Safety Class 1 (SC-1) applies to pressure-retaining portions and supports of mechanical equipment that form part of the RCPB whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability and whose requirements are within the scope of the ASME Boiler and Pressure Vessel Code, Section III.

B. Safety Class 2 (SC-2) applies to pressure-retaining portions and supports of primary containment and other mechanical equipment, requirements for which are within the scope of the ASME Boiler and Pressure Vessel Code, Section III, that are not included in SC-1 and are designed and relied upon to accomplish the nuclear safety functions defined in ANSI/ANS 51.1, Section 3.3.1.2.

C. Safety Class 3 (SC-3) applies to equipment, not included in SC-1 or -2, that is designed and relied upon to accomplish the nuclear safety functions defined in ANSI/ANS 51.1, Section 3.3.1.3. D

D. Non-Nuclear Safety (NNS) applies to equipment that is not in Safety Class 1, 2, or 3. This equipment is not relied upon to perform a nuclear safety function.

The safety classifications of major components which are in the System 80+ design scope are listed in Table 3.2-1 ~~and Section 3.11.~~ Seismic category designations and quality assurance requirements are also included. ~~Small components, such as piping, valves and strainers, are not listed; they may be found by reference to the P&IDs (Chapters 5, 6, and 9) where the exact boundaries are indicated.~~ Safety Class 1, 2, and 3 Valves are listed in ~~Tables~~ Table 3.2-2. D I

All pressure containing components in Safety Classes 1, 2, and 3 are designed, manufactured, and tested in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III. Components designated NNS are designed and constructed with appropriate consideration of the intended service using applicable industry codes and standards. The relationship between safety class and code class is shown in Table 3.2-~~2~~3. A higher code class may be used for a component without changing the safety class or affecting the balance of the system in which it is located. D

Fracture toughness requirements are imposed on materials for pressure retaining parts of ASME Class 2 and 3 System 80+ Standard Design components. Test methods, acceptance, and exemption criteria are in conformance with the ASME Code, Section III. I

The safety classification system is also used to identify those components to which the requirements of 10 CFR 50, Appendix B,

~~are applicable. Components in Safety Classes 1, 2, and 3* are designed and manufactured under a rigorous quality assurance program reflecting the requirements of Appendix B, and are designated Quality Class 1. The Quality Class 1 quality assurance program is described in Chapter 17. Components which do not serve a safety related function are designated Quality Class 2. Quality Class 2 components will be designed and manufactured or procured in accordance with the pertinent requirements of the Quality Assurance Program as given in Chapter 17.~~ D

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The use of the above outlined safety and quality classification systems meets the intent of Regulatory Guide 1.26 and the requirements of 10 CFR 50.55a.

are applicable. Components in Safety Classes 1, 2, and 3* are designed and manufactured under a rigorous quality assurance program reflecting the requirements of 10CFR50, Appendix B and are designated with a Q under the Quality Assurance Requirement category in Table 3.2-1. Components which do not serve a safety-related function are not subject to the quality assurance requirements of 10CFR50, Appendix B and are designated with an N in Table 3.2-1. The Quality Assurance Program is described in Chapter 17.

* With the following exception: the CVCS gas stripper is Safety Class 3, ~~Quality Class 2~~, however, pressure retaining portions meet rules applicable to ASME Code Class 3 components. See Table 3.2-1. D

and is not subject to the quality assurance requirements of 10CFR50, Appendix B, RAI 210.1 & 210.4

Piping (CS)	1/2	I	SSCV	Q
Valves (SS)	1/2	I	SSCV	Q

TABLE 3.2-1

(Sheet 1 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

and Internals Structures important to safety

Component Identification	Safety Class	Seismic Category	Location	Quality Assurance Requirement (27) ~~Quality Class~~
Reactor Coolant System				
* Reactor Vessel	1	I	SSCV	~~1~~ Q
* Steam Generators (primary/secondary)	1/2 (1)	I	SSCV	~~1~~ Q
* Pressurizer	1	I	SSCV	~~1~~ Q
* Reactor Coolant Pumps (2) (3) (9)	1	I	SSCV	~~1~~ Q
Piping within Reactor Coolant Pressure Boundary (5)	1/2 (4)	I	SSCV	~~1~~ Q
Control Element Drive Mechanisms	(6)	(6)	SSCV	~~1~~ Q
Core Support Structures (7)	3	I	SSCV	~~1~~ Q
Fuel Assemblies (8)	2	I	SSCV	~~1~~ Q
Control Element Assemblies (8)	3	I	SSCV	~~1~~ Q
Closure Head Lift Rig	NNS	II (10)	SSCV	~~2~~ N
Heated Junction Thermocouple Probe Assembly	1/3 (12)	I	SSCV	~~1~~ Q
HJTC Pressure Housing	1	I	SSCV	~~1~~ Q
ICI Cable Tray Support Frame	3	I	SSCV	~~1~~ Q
ICI Holding Frame	NNS	NS	SSCV	~~2~~ N
ICI Guide Tubes	1	I	SSCV	~~1~~ Q
ICI Guide Tube Supports	1	I	SSCV	~~1~~ Q
ICI Seal Housing	1	I	SSCV	~~1~~ Q
ICI Seal Table	1	I	SSCV	~~1~~ Q
~~Safety Injection System~~				
~~* Safety Injection Pumps~~	~~2~~	~~I~~		~~1~~
~~* Shutdown Cooling Heat Exchangers~~	~~2/3 (1)~~	~~I~~		~~1~~
~~* Safety Injection Tanks~~	~~2~~	~~I~~		~~1~~
~~* Shutdown Cooling Pumps~~	~~2~~	~~I~~		~~1~~
~~* Containment Spray Pumps~~	~~2~~	~~I~~		~~1~~
~~* Containment Spray Heat Exchangers~~	~~2/3 (1)~~	~~I~~		~~1~~
~~* IRWST~~	~~2~~	~~I~~		~~1~~
~~* Shutdown Cooling Mini-flow Heat Exchanger~~	~~2/3 (1)~~	~~I~~		~~1~~
~~* Containment Spray Mini-flow Heat Exchanger~~	~~2/3 (1)~~	~~I~~		~~1~~

Replace with Insert I

Footnotes to this table are given at the end of the table.

~~* Including component supports down to (but not including) embedments.~~

D I D D

RAI 210.1

INSERT 1:

[To be inserted in CESSAR-DC Table 3.2-1 (Page 1 of 17) in response to NRC RAI 210.1]

In-containment Water Storage System				
IRWST		I	SSCV	Q
Holdup Volume		I	SSCV	Q
Steam Relief System				
Piping	1/2	I	SSCV	Q
Valves	1/2	I	SSCV	Q
Spargers	2	I	SSCV	Q
Cavity Flooding System				
Piping	2	I	SSCV	Q
Valves	2	I	SSCV	Q
Safety Depressurization System				
Valves	1/2	I	SSCV	Q
Piping	1/2	I	SSCV	Q
Reactor Coolant Gas Vent System				
Valves	1/2	I	SSCV	Q
Piping	1/2	I	SSCV	Q
Safety Injection System				
Safety Injection Pumps	2	I	RXB	Q
Safety Injection Tanks	2	I	SSCV	Q
Piping (24) (28)	1/2	I	RXB/SSCV	Q
Valves (28)	1/2	I	RXB/SSCV	Q
Shutdown Cooling System				
Shutdown Cooling Heat Exchangers	2/3 (1)	I	RXB	Q
Shutdown Cooling Pumps	2	I	RXB	Q
Shutdown Cooling Mini-Flow Heat Exchanger	2/3 (1)	I	RXB	Q
Piping (28)	1/2	I	RXB/SSCV	Q
Valves (28)	1/2	I	RXB/SSCV	Q
Containment Spray System				
Containment Spray Pumps	2	I	RXB	Q
Containment Spray Heat Exchangers	2/3 (1)	I	RXB	Q
Containment Spray Mini-Flow Heat Exchanger	2/3 (1)	I	RXB	Q
Spray Nozzles	2	I	SSCV	Q
Piping (28)	2	I	RXB/SSCV	Q
Valves (28)	2	I	RXB/SSCV	Q

TABLE 3.2-1 (Cont'd)

(Sheet 2 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	Quality Assurance Requirement ~~Quality Class~~
Chemical and Volume Control System				
* Regenerative Heat Exchanger	2	I	SSCV	~~1~~ G
* Letdown Heat Exchanger	2/[illegible]	I	SSCV	~~1~~ G
* Seal Injection Heat Exchanger	2/[illegible]	I	NA	~~1~~ G
* Purification Ion Exchangers	2	I	NA	~~1~~ G
* Deborating Ion Exchanger	2	I	NA	~~1~~ G
* Volume Control Tank	2	I	NA	~~1~~ G
* Chemical Addition Package	NNS	NS	NA	~~2~~ N
* Boric Acid Batching Tank	NNS	NS	NA	~~2~~ N
* Charging Pumps	2	I	NA	~~1~~ Q
* Boric Acid Makeup Pumps	3	I	NA	~~1~~ G
* Reactor Makeup Water Pumps	NNS	NS	NA	~~2~~ N
* Boric Acid Concentrator	NNS	NS	NA	~~2~~ N
* Pre-holdup Ion Exchanger	3	I	NA	~~1~~ G
Charging Pump * Mini-flow Heat Exchangers	2/[illegible]	I	NA	~~1~~ G
* Boric Acid Condensate Ion Exchanger	NNS	NS	NA	~~2~~ N
* Reactor Drain Pumps	3	I	NA	~~1~~ Q I
* Holdup Pumps	NNS	NS	NA	~~2~~ N
* Reactor Drain Tank	NNS	NS	SSCV	2 N
* Holdup Tank	NNS	NS	YD	~~2~~ N
* Equipment Drain Tank	3	I	NA	~~1~~ G
* Reactor Makeup Water Tank	NNS	NS	[illegible]	~~2~~ N
* Gas Stripper	3	I	NA	~~1~~ G
* Purification Filters	2	I	NA	~~1~~ G
* Reactor Drain Filter	3	I	NA	~~1~~ G
* Seal Injection Filters	2	I	NA	~~1~~ Q
* Reactor Makeup Filter	NNS	NS	NA	~~2~~ N
* Boric Acid Filter	3	I	NA	~~1~~ G
* Letdown Strainer	2	I	NA	~~1~~ G
* Pre-holdup Strainer	3	I	NA	~~1~~ G
* Boric Acid Condensate IX Strainer	NNS	NS	NA	~~2~~ N
* Ion Exchanger Drain Header Strainer	NNS	NS	NA	~~2~~ N
* Boric Acid Batching Strainer	NNS	NS	NA	~~2~~ N
* Chemical Addition Strainer	NNS	NS	NA	~~2~~ N
* Boric Acid Storage Tank	2	I	YD	~~1~~ G
Boric Acid Batching Eductor	NNS	NS	NA	N
Letdown Orifices	2	I	SSCV	Q
Boronometer	NNS	NS	NA	N
Piping (28)	1/2/3/NNS	I/NS	SSCV/NA/YD	Q/N
Valves (28)	1/2/3/NNS	I/NS	SSCV/NA/YD	Q/N

TABLE 3.2-1 (Cont'd)

(Sheet 3 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	Quality Assurance Requirement (27) ~~Quality Class~~	
Emergency Feedwater System					
Cavitating Venturi	2	I	SSCV	Q	
* Motor-Driven Emergency Feedwater Pumps	3	I	RXB	~~1~~ Q	
* Steam-Driven Emergency Feedwater Pumps	3	I	RXB	~~1~~ Q	I
* Emergency Feedwater Pump Turbines	3	I	RXB	~~1~~ Q	
* Emergency Feedwater Storage Tanks	3	I	NA	~~1~~ Q	
Piping (28)	2/3	I	RXB/SSCV/NA	Q	
Valves (28)	2/3	I	RXB/SSCV/NA	Q	
Fuel Handling System					
Spent Fuel Pool	–	I	NA	Q	
Refueling Machine	NNS	II	SSCV	~~2~~ N	
Fuel Transfer System	NNS	II	SSCV/NA	~~2~~ N	
1. Transfer Carriage	NNS	II	SSCV/NA	~~2~~ N	
2. Upending Machine	NNS	II	SSCV/NA	~~2~~ N	D
3. Hydraulic Power Unit	NNS	II	SSCV/NA	~~2~~ N	
Fuel Transfer Tube, Valve, Stand	NNS	II	SSCV/NA	~~2~~ N	
CEA Change Platform	NNS	II	SSCV	~~2~~ N	
Long and Short Fuel Handling Tools	NNS	NS	SSCV	~~2~~ N	
Upper Guide Structure Lifting Rig	NNS	II (11)	SSCV	~~2~~ N	
Core Barrel Lifting Rig	NNS	II (11)	SSCV	~~2~~ N	
Spent Fuel Handling Machine	NNS	II	NA	~~2~~ N	
New Fuel Elevator	NNS	II	NA	~~2~~ N	
Underwater Television	NNS	NS	SSCV	~~2~~ N	
Refueling Pool Seal	NNS	NS	SSCV	~~2~~ N	
In-Core Instrumentation and CEA Cutter	NNS	NS	SSCV	~~2~~ N	
Extension Shaft Uncoupling Tool	NNS	NS	SSCV	~~2~~ N	
Fuel Transfer Tube Blind Flange	2	I	SSCV	~~1~~ Q	
CEA Handling Tools	NNS	NS	SSCV	~~2~~ N	
ICI Insertion and Removal Tools	NNS	NS	SSCV	~~2~~ N	
Spent Fuel Racks	NNS	I	NA	~~2~~ N	
New Fuel Racks	NNS	I	NA	~~2~~ N	E
Condensate and Feedwater System ~~(13)~~					
Condensate Pumps	NNS	NS	TB	~~2~~ N	
Feedwater Pumps	NNS	NS	TB	~~2~~ N	I
Feedwater Pump Controllers	NNS	NS	TB	~~2~~ N	
Feedwater Booster Pumps	NNS	NS	TB	~~2~~ N	
Feedwater Startup Pump~~s~~	NNS	NS	TB	~~2~~ N	
Low Pressure Feedwater Heaters	NNS	NS	TB	~~2~~ N	
High Pressure Feedwater Heaters	NNS	NS	TB	~~2~~ N	
Deaerator	NNS	NS	TB	~~2~~ N	
Piping (13)	2/NNS	I/NS	TB/NA/SSCV/MSVH	Q/N	
Valves (13)	2/NNS	I/NS	TB/NA/SSCV/MSVH	Q/N	

TABLE 3.2-1 (Cont'd)

(Sheet 4 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	Quality Class (Quality Assurance Requirement (27))
Main Condenser System				
Main Condenser	NNS	NS	TB	~~2~~ N
Condensate Storage System				
Condensate Storage Tanks	NNS	NS	YD	~~2~~ N
Condensate Storage Tank Recycle Pumps	NNS	NS	SXB	~~2~~ N
Condensate Drain Tanks	NNS	NS	TB	~~2~~ N
Condensate Drain Tank Transfer Pump	NNS	NS	TB	~~2~~ N
~~Condensate Storage Feedwater Startup Pump~~	~~NNS~~	~~NS~~		~~2~~
Piping	NNS	NS	YD/SXB/TB	N
Valves	NNS	NS	YD/SXB/TB	N
Condensate Cleanup System				
Piping	NNS	NS	TB	N
Polishers/Demineralizers	NNS	NS	TB	~~2~~ N
Resin Traps	NNS	NS	TB	~~2~~ N
Valves	NNS	NS	TB	N
Main ~~Vacuum System~~ Condenser Evacuation System			TB	
Exhaust Silencers	NNS	NS	TB	N
Gas Dryers	NNS	NS	TB	N
Vacuum Pumps	NNS	NS	TB	~~2~~ N
Steam Jet Air Ejectors	NNS	NS	TB	~~2~~ N
Steam Jet Air Ejector Condenser	NNS	NS	TB	~~2~~ N
Piping	NNS	NS	TB	N
Valves	NNS	NS	TB	N
Demineralized Water Makeup System				
Demineralizer Makeup Water Pump	NNS	NS	YD	~~2~~ N
Demineralizers	NNS	NS	YD	~~2~~ N
Vacuum Degasifier	NNS	NS	YD	~~2~~ N
Demineralized Water Storage Tank	NNS	NS	YD	~~2~~ N
Vacuum Pumps	NNS	NS	YD	~~2~~ N
Filters	NNS	NS	YD	~~2~~ N
~~Recirculating Pumps~~ Recycle Pump	NNS	NS	YD	~~2~~ N
Vacuum Degasifier Transfer Pumps	NNS	NS	YD	N
Demineralized Water Transfer Pumps	NNS	NS	YD	N
Demineralizer Waste Tank	NNS	NS	YD	N
Piping (28)	2/NNS	I/NS	ALL	N
Valves (28)	2/NNS	I/NS	ALL	N
Extraction Steam System	~~NNS~~	~~NS~~	~~TB~~	~~2~~
Piping	NNS	NS	TB	N
Valves	NNS	NS	TB	N
Heater Vents	~~NNS~~	~~NS~~	~~TB~~	~~2~~
Piping	NNS	NS	TB	N
Valves	NNS	NS	TB	N

RAI 210.1

Replace with Insert 2

TABLE 3.2-1 (Cont'd)

(Sheet 5 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	Quality Assurance Requirement (27) ~~Quality Class~~
~~Chemical Feed System~~				
~~Chemical Addition Tanks~~	~~NNS~~	~~NS~~		~~2~~
~~Chemical Addition Pumps~~	~~NNS~~	~~NS~~		~~2~~
~~Turbine Generator System~~				
~~Turbine Generator~~				
~~High Pressure Turbine~~	~~NNS~~	~~NS~~		~~2~~
~~Low Pressure Turbines~~	~~NNS~~	~~NS~~		~~2~~
~~Generator~~	~~NNS~~	~~NS~~		~~2~~
~~Moisture Separators~~	~~NNS~~	~~NS~~		~~2~~
~~Steam Reheaters~~	~~NNS~~	~~NS~~		~~2~~
~~Turbine Bypass System~~	~~NNS~~	~~NS~~		~~2~~
~~Turbine Gland Sealing System~~				
~~Gland Seal Condenser~~	~~NNS~~	~~NS~~		~~2~~
~~Gland Seal Regulator~~	~~NNS~~	~~NS~~		~~2~~
~~Turbine Lube Oil System~~				
~~Centrifugal Oil Pumps~~	~~NNS~~	~~NS~~		~~2~~
~~Booster Pump~~	~~NNS~~	~~NS~~		~~2~~
~~Oil Tank~~	~~NNS~~	~~NS~~		~~2~~
~~Oil Turbine~~	~~NNS~~	~~NS~~		~~2~~
~~Oil Coolers~~	~~NNS~~	~~NS~~		~~2~~
~~Turbine Control System~~				
~~EHC Pumps~~	~~NNS~~	~~NS~~		~~2~~
~~EHC Coolers~~	~~NNS~~	~~NS~~		~~2~~
~~EHC Sumps~~	~~NNS~~	~~NS~~		~~2~~
~~Turbine Generator Cooling System~~				
~~Hydrogen Coolers~~	~~NNS~~	~~NS~~		~~2~~
Liquid Waste Management System				
Waste Collection Tanks	NNS	NS	RWF	~~2~~ N
Waste Sample Tanks	NNS	NS	RWF	~~2~~ N
Process Pumps	NNS	NS	RWF	~~2~~ N
Process Demineralizers	NNS	NS	RWF	~~2~~ N
Process Filters	NNS	NS	RWF	~~2~~ N
Piping (28)	2/NNS	I/NS	TB/NA/RWF/SSCV/RXB	G/N
Valves (28)	2/NNS	I/NS	TB/NA/RWF/SSCV/RXB	G/N

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RAI 210.1

INSERT 2:

[To be inserted in CESSAR-DC Table 3.2-1 (Page 5 of 17) in response to NRC RAI 210.1]

Turbine Generator System

Turbine Generator				
High Pressure Turbine	NNS	NS	TB	N
Low Pressure Turbines	NNS	NS	TB	N
Generator	NNS	NS	TB	N
Moisture Separators	NNS	NS	TB	N
Steam Reheaters	NNS	NS	TB	N
Stop Valves	NNS	NS	TB	N
Control Valves	NNS	NS	TB	N
Reheat Stop Valves	NNS	NS	TB	N
Intercept Valves	NNS	NS	TB	N
Valves, other	NNS	NS	TB	N
Piping	NNS	NS	TB	N
Turbine Bypass System				
Turbine Bypass Valves	NNS	NS	TB	N
Valves, other	NNS	NS	TB	N
Piping	NNS	NS	TB	N
Turbine Gland Sealing System				
Gland Seal Condenser	NNS	NS	TB	N
Gland Seal Regulator	NNS	NS	TB	N
Piping	NNS	NS	TB	N
Valves	NNS	NS	TB	N
Turbine Lube Oil System				
Pumps	NNS	NS	TB	N
Oil Tank	NNS	NS	TB	N
Oil Turbine	NNS	NS	TB	N
Oil Coolers	NNS	NS	TB	N
Oil Filters	NNS	NS	TB	N
Piping	NNS	NS	TB	N
Valves	NNS	NS	TB	N
Turbine Control System				
EHC Pumps	NNS	NS	TB	N
EHC Coolers	NNS	NS	TB	N
EHC Sumps	NNS	NS	TB	N
Piping	NNS	NS	TB	N
Valves	NNS	NS	TB	N
Turbine Generator Cooling System				
Hydrogen Coolers	NNS	NS	TB	N
Piping	NNS	NS	TB	N
Valves	NNS	NS	TB	N

RE: RAI 210.1

TABLE 3.2-1 (Cont'd)

(Sheet 6 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	Quality Assurance Requirement (27) Quality ~~Class~~
Gaseous Waste Management System				
Piping (2D)	2/NNS	NS	NA/RWF/SSCI	2/N
Gas Dryers	NNS	NS	RWF	~~2~~ N
Charcoal Beds	NNS	NS	RWF	~~2~~ N
Valves (2D)	2/NNS	NS	NA/RWF/SSCI	2/N
Solid Waste Management System				
Pumps	NNS	NS	NA/RWF	N
Spent Resin Tanks	NNS	NS	NA/RWF	N
HIC Fill/Dewatering Head	NNS	NS	RWF	~~2~~ N
Slurry Pump	NNS	NS	RWF	~~2~~ N
~~Radwaste Building Crane~~	~~NNS~~	~~NS~~	~~RWF~~	~~2~~
Dry Solids Compactor	NNS	NS	RWF	~~2~~ N
Piping	NNS	NS	NA/RWF	N
Valves	NNS	NS	NA/RWF	N
Heater Drain System				
Piping	NNS	NS	TB	N
Reheater Drain Tanks	NNS	NS	TB	~~2~~ N
Moisture Separator Drain Tanks	NNS	NS	TB	~~2~~ N
Heater Drain Tank	NNS	NS	TB	~~2~~ N
Heater Drain Pumps	NNS	NS	TB	~~2~~ N
Valves	NNS	NS	TB	N
Process and Effluent Radiation Monitoring System				
Gaseous Process and Effluent Monitors				
Unit ~~Process~~ Vent	NNS	NS	NA	~~2~~ N
Waste ~~Process~~ Gas	NNS	NS	RWF	~~2~~ N
Unit ~~Ventilation~~ Vent Post-Accident	NNS	NS	NA	~~2~~ N
Containment Purge Exhaust	NNS	NS	NA	~~2~~ N
Condenser Air Ejector	NNS	NS	TB	~~2~~ N
Reactor Building Annulus	NNS	~~II~~ NS	NA	N ~~2~~
Subsphere Ventilation	NNS	NS	NA	N ~~2~~
Liquid Process and Effluent Monitors				
Component Cooling Water	NNS	NS	NA	~~2~~ N
Liquid Waste Discharge	NNS	NS	RWF	~~2~~ N
Plant Discharge Line	NNS	NS	RWF	~~2~~ N
Station Service Water	NNS	NS	CCWX	~~2~~ N
Reactor Coolant Gross Activity	NNS	NS	NA	~~2~~ N
Turbine Building Drains	NNS	NS	TB	~~2~~ N
Steam Generator Blowdown	NNS	NS	TB	~~2~~ N

Move to Airborne Radiation Monitor Section

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TABLE 3.2-1 (Cont'd)

(Sheet 7 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	Quality Class (Quality Assurance Requirement (23))
Airborne Radiation Monitors				
Containment Atmosphere	~~NNS~~ 3	~~NS~~ I	NA	~~2~~ Q
Nuclear Annex	NNS	NS	NA	~~2~~ N
Radwaste Building	NNS	NS	RWF	~~2~~ N
Fuel Building	~~3~~ NNS	~~I~~ NS	NA	~~1~~ N
Ventilation Systems Multisampler	NNS	NS	NA	~~2~~ N
Control Room Intake (A&B)	3	I	NA	~~1~~ Q
Area Radiation Monitors	NNS	NS	SSCV/NA/RWF	~~2~~ N
Special Purpose Area Monitors				
Main Steam Line	NNS	~~II~~ NS	NA	~~2~~ N
Purification Filter	NNS	NS	NA	~~2~~ N
Containment Area High Radiation	3	I	SSCV	~~1~~ Q
Primary Coolant	3	I	SSCV	~~1~~ Q
Containment Isolation System Piping	2	I	SSCV/RXB	~~1~~ Q
Valves	2	I	SSCV/RXB	Q
Component Cooling Water System (14)				
Piping (20)	2/3/NNS	I/NS	CCWX/YD/NA/RXB/SSCV	Q/N
~~CCWS~~ Heat Exchangers	3	I	CCWX	~~1~~ Q
~~CCWS~~ Pumps	3	I	NA	~~1~~ Q
~~CCWS~~ Surge Tanks	3	I	NA	~~1~~ Q
~~CCWS~~ Sump Pumps	NNS	NS	NA	~~2~~ N
~~CCWS~~ Chemical Addition Tank	NNS	NS	NA	~~2~~ N
~~CCWS~~ Heat Exchanger Building Sump Pumps	NNS	NS	CCWX	~~2~~ N
Valves (20)	2/3/NNS	I/NS	CCWX/YD/NA/RXB/SSCV	Q/N
Pool Cooling and Purification System				
Spent Fuel Pool Cooling System				
Piping	3	I	NA	Q
Valves	3	I	NA	Q
~~Spent Fuel Pool Cooling~~ Pumps	3	I	NA	~~1~~ Q
~~Spent Fuel Pool Cooling~~ Heat Exchangers	3	I	NA	~~1~~ Q
Pool Purification System				
~~Pool Purification~~ Pumps	NNS	NS	NA	~~2~~ N
~~Pool~~ Strainers	NNS	NS	NA	~~2~~ N
~~Pool~~ Demineralizers	NNS	NS	NA	~~2~~ N
~~Pool~~ Filters	NNS	NS	NA	~~2~~ N
~~Spent Fuel Pool~~ Skimmer	NNS	NS	NA	~~2~~ N
Piping (20)	2/NNS	I/NS	NA/SSCV	Q/N
Valves (20)	2/NNS	I/NS	NA/SSCV	Q/N
Strainers	NNS	NS	NA	N

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TABLE 3.2-1 (Cont'd)

(Sheet 8 of 17)

Replace with Insert 3

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	~~Quality Class~~ Quality Assurance Requirement (27)
Sampling System				
Process Sample System Heat Exchangers	NNS	NS		2
Station Service Water System				
Piping	3/NNS	I/NS	SSPS/[illegible]	G/N
~~SSWS~~ Pumps	3	I	SSPS	~~1~~ G
~~SSWS~~ Strainers	3	I	SSPS	~~1~~ G
~~SSWS~~ Sump Pumps	~~3~~ NNS	~~I~~ NS	SSPS	~~1~~ N
~~SSWS~~ Traveling Screens	3	I	YD	~~1~~ G
Valves	3/NNS	I/NS	SSPS/[illegible]	G/N
Turbine Building Service Water System				
Piping	NNS	NS	TB	N
Valves	NNS	NS	TB	N
~~TBSWS~~ Pumps	NNS	NS	TB	~~2~~ N
Strainers	NNS	NS	TB	N
Turbine Building Cooling Water System				
Piping	NNS	NS	TB	N
Valves	NNS	NS	TB	N
~~TBCWS~~ Heat Exchangers	NNS	NS	TB	~~2~~ N
~~TBCWS~~ Pumps	NNS	NS	TB	~~2~~ N
~~TBCWS~~ Surge Tank	NNS	NS	TB	~~2~~ N
~~TBCWS~~ Chemical Addition Tank	NNS	NS	TB	~~2~~ N
Chilled Water Systems				
Essential Chilled Water System				
ECW Refrigeration Units	3	I		1
ECW Circulation Pumps	3	I		1
ECW Compression Tanks	3	I		1
Normal Chilled Water System (15)				
NCW Refrigeration Units	NNS	I		2
NCW Circulation Pumps	NNS	I		2
NCW Compression Tanks	NNS	I		2
NCW Air Separators	NNS	I		2
NCW Heat Exchangers	3	I		2
Condenser Circulating Water System				
Condenser Circulating Water Pumps	NNS	NS		2
Condenser Circulating Water Cooling Towers	NNS	NS		2

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Replace with Insert 4

RAI 210.1

INSERT 3:

[To be inserted in CESSAR-DC Table 3.2-1 (Page 8 of 17) in response to NRC RAI 210.1]

Process Sampling System				
Primary Sampling System				
Pump	WNS	NS	NA	N
Heat Exchangers	WNS	NS	NA	N
Sample Vessels	WNS	NS	NA	N
Piping (28)	2/WNS	I/NS	NA/SSCV	QN
Valves (28)	2/WNS	I/NS	NA/SSCV	QN
Sink	WNS	NS	NA	N
Secondary Chemistry Control System				
Heat Exchangers	WNS	NS	NA	N
Strainers	WNS	NS	NA	N
Monitors	WNS	NS	NA	N
Piping (28)	2/WNS	I/NS	NA/SSCV	QN
Valves (28)	2/WNS	I/NS	NA/SSCV	QN

INSERT 4:

[To be inserted in CESSAR-DC Table 3.2-1 (Page 8 of 17) in response to NRC RAI 210.1]

Chilled Water System				
Essential Chilled Water System				
Refrigeration Units	3	I	NA	Q
Pumps	3	I	NA	Q
Compression Tanks	3	I	NA	Q
Chemical Addition Tanks	WNS	NS	NA	N
Essential/Normal Heat Exchangers	3/WNS (1)	I	NA	QN
Piping (28)	2/3/WNS	I/NS	NA/SSCV/RXB	QN
Valves (28)	2/3/WNS	I/NS	NA/SSCV/RXB	QN
Strainers	3/WNS	I/NS	NA	QN
Normal Chilled Water System (15)				
Refrigeration Units	WNS	NS	NA	N
Pumps	WNS	NS	NA	N
Compression Tanks	WNS	NS	[illegible]	N
Air Separators	WNS	NS	NA	N
Chemical Addition Tanks	WNS	NS	NA	N
Piping (28)	2/WNS	I/NS	NA/SSCV	QN
Valves (28)	2/WNS	I/NS	NA/SSCV	QN
Strainers	WNS	NS	NA	N
Condenser Circulating Water System				
Pumps	WNS	NS	YD	N
Cooling Towers (mechanical portion)	WNS	NS	YD	N
Piping	WNS	NS	YD/TB	N
Valves	WNS	NS	YD/TB	N
Strainers	WNS	NS	YD/TB	N
Traveling Screens	WNS	NS	YD	N

TABLE 3.2-1 (Cont'd)

(Sheet 9 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	Quality Assurance Requirements ~~Quality Class~~
Compressed Air Systems				
Instrument Air System				
Air Compressors	NNS	NS	NA	~~2~~ N
~~Aftercoolers~~ Piping (28)	2/NNS	I/NS	ALL	~~2~~ S/N
~~Moisture Separators~~ Valves (28)	2/NNS	I/NS	ALL	~~2~~ S/N
Air Receivers	NNS	NS	NA	~~2~~ N
Desiccant Air Dryers/Filters	NNS	NS	NA	~~2~~ N
~~Air Filters~~	~~NNS~~	~~NS~~		~~2~~
Station Air System				
Air Compressors	NNS	NS	SXB	~~2~~ N
~~Aftercoolers~~ Air Dryers/Filters	NNS	NS	SXB	~~2~~ N
~~Moisture Separators~~ Piping (28)	2/NNS	I/NS	ALL	~~2~~ S/N
Air Receivers	NNS	NS	SXB	~~2~~ N
Valves (28)	2/NNS	I/NS	ALL	S/N
Breathing Air System				
Air Compressors	NNS	NS	SXB	~~2~~ N
~~Aftercoolers~~ Piping (28)	2/NNS	I/NS	ALL	~~2~~ S/N
~~Air Filters~~ Valves (28)	2/NNS	I/NS	ALL	~~2~~ S/N
Air Receivers	NNS	NS	SXB	~~2~~ N
Air Dryers/Filters	NNS	NS	SXB	~~2~~ N
Compressed Gas Systems				
Piping (26)(28)	2/NNS	I/NS	ALL	S/N
High Pressure Gas Cylinders	NNS	NS	YD	~~2~~ N
Pressure Regulators	NNS	NS	YD	~~2~~ N
Leak Detection Systems	NNS	NS	ALL	~~2~~ N
Liquid Nitrogen Evaporators	NNS	NS	YD	~~2~~ N
Valves (28)	2/NNS	I/NS	ALL	S/N
Fire Protection System ~~(16)~~				
Jockey Pump	NNS	NS	FPH	N
Surge Tank	NNS	I	NA	S
Fire Pumps	NNS	~~I~~ NS	FPH	~~2~~ N
~~FPS Tank~~ Storage Tanks	NNS	~~I~~ NS	FPH	~~2~~ N
Water Spray Systems (Deluge and Sprinkler) Piping, Valves (16)(28)	2/NNS	I/NS ~~II~~	TB/NA/SSCV/RXB/DGB/SXB	~~2~~ S/N
~~Detection/Alarm System~~	~~NNS~~	~~NS~~		~~2~~
Hose Systems/Standpipes (16)(28)	2/NNS	I/NS	ALL	~~2~~ S/N
~~Air Packs~~	~~NNS~~	~~NS~~		~~2~~
Portable Fire Extinguishers (16)	NNS	~~I~~ NS	ALL	~~2~~ N
Exterior Distribution System				
Piping	NNS	NS	YD	N
Valves	NNS	NS	YD	N
Strainers	NNS	NS	YD	N

TABLE 3.2-1 (Cont'd)

(Sheet 10 of 17)

Replace with Insert 5

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	Quality ~~Class~~ assurance requirement (27)
Diesel Generator Systems				
DG Fuel Oil System (17)				
Fuel Oil Storage Tank	3	I		1
Recirculation Pump	3	I		1
Booster Pump	3	I		1
Fuel Oil Day Tank	3	I		1
Fuel Oil Transfer Pump	3	I		1
DG Cooling Water System				
DGCW Surge Tank	3	I		1
DGCW Circulation Pump	3	I		1
DGCW Keep Warm Pump	3	I		1
3-way Thermostatic Control Valve	3	I		1
DGCW Jacket Water Cooler	3	I		1
DG Starting Air System (18)				
Compressors	3	I		1
Aftercoolers	3	I		1
Moisture Separator	3	I		1
Filter/Dryer Units	3	I		1
Air Receivers	3	I		1
DG Lube Oil System (19)				
Lube Oil Cooler	3	I		1
Oil Transfer Pumps	3	I		1
Prelube Oil Pump	3	I		1
Lube Oil Sump Tank Heaters	3	I		1
Clean and Used Lube Oil Storage Tanks	3	I		1
DG Intake and Exhaust System				
Turbocharger	3	I		1
Aftercooler	3	I		1
Silencers and Air Filters	3	I		1
Equipment and Floor Drainage System				
Reactor Building Subsphere Sump Pumps	3	I	RXB	~~1~~ Q
~~DG Building Sump Pumps~~	~~3~~	~~I~~		~~1~~
Other ~~Floor Drain~~ Sump Pumps	NNS	NS		~~2~~ N
Piping (28)	2/3/NNS	I/NS	ALL	Q/N
Valves (28)	2/3/NNS	I/NS	ALL	Q/N
Diesel ~~Generator~~ Building Sump Pump System				
Sump Pumps	3	I	DGB	Q
Piping	3/NNS	I/NS	DGB/NA/PWF	Q/N
Valves	3/NNS	I/NS	DGB/NA/PWF	Q/N

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INSERT 5:

[To be inserted in CESSAR-DC Table 3.2-1 (Page 10 of 17) in response to NRC RAI 210.1]

Diesel Generator Systems

DG Engine Fuel Oil System (17)				
Fuel Oil Storage Tanks	3	I	DFS	Q
Recirculation Pumps	NNS	NS	DFS	N
Booster Pumps	3	I	DGB	Q
Fuel Oil Day Tanks	3	I	DGB	Q
Fuel Oil Transfer Pumps	3	I	DGB	Q
Strainers	3/NNS	I/NS	DGB/YD	QN
Filters	3/NNS	I/NS	DGB	QN
Piping	3/NNS	I/NS	DGB/DFS/YD	QN
Valves	3/NNS	I/NS	DGB/DFS	QN
DG Engine Co[illegible]ing [illegible]				
Circulati[illegible] Pump[illegible]	3	I	DGB	Q
Keep Warm [illegible]	3	I	DGB	Q
Jacket Wat[illegible] Cool[illegible]	3	I	DGB	Q
Jacket Wate[illegible]	3	I	DGB	Q
Chemical Pot [illegible]	3	I	DGB	Q
Piping	3	I	DGB	Q
Valves	3	I	DGB	Q
DG Engi[illegible]e Starting Air [illegible]ystem (18)				
Compressors	NNS	NS	DGB	N
Aftercoolers	NNS	NS	DGB	N
Moisture Separators	NNS	NS	DGB	N
Filter/Dryer Units	NNS	NS	DGB	N
Air Receivers	3	I	DGB	Q
Strainers	3/NNS	I/NS	DGB	QN
Traps	NNS	NS	DGB	N
Filters	3/NNS	I/NS	DGB	QN
Piping	3/NNS	I/NS	DGB	QN
Valves	3/NNS	I/NS	DGB	QN
DG Engine Lube Oil System (19)				
Lube Oil Sump Tanks	3	I	DGB	Q
Lube Oil Coolers	3	I	DGB	Q
Oil Transfer Pumps	NNS	NS	DGB/YD	N
Prelube Oil Pumps	3	I	DGB	Q
Clean and Used Lube Oil Storage Tanks	NNS	NS	YD	N
Filters	3	I	DGB	Q
Strainers	3/NNS	I/NS	DGB	QN
Piping	3/NNS	I/NS	DGB/YD	QN
Valves	3/NNS	I/NS	DGB/YD	QN
DG Engine Air Intake and Exhaust System				
Turbochargers	3	I	DGB	Q
Aftercoolers	3	I	DGB	Q
Silencers and Air Filters	3	I	DGB	Q
Piping	3	I	DGB	Q

N2 RAI 210.1

Air Handling Units w/ Filters	3	I	NA	Q
Fans, Ductwork	3	I	NA	Q
Dampers	3	I	NA	Q

TABLE 3.2-1 (Cont'd)

(Sheet 11 of 17)

CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	~~Quality Class~~ Quality Assurance Requirements (12)
Control Building Ventilation System				
Main Control Room Air Handling System				
Air Handling Units w/Filters	3	I	NA	~~1~~ Q
Fans, Ductwork	3	I	NA	~~1~~ Q
Water-cooling Coils	3	I	NA	~~1~~ Q
Heating Coils Dampers	3	I	NA	~~1~~ Q
Technical Support Center Air Handling System	3	I	NA	Q
Air Handling Units w/ Filters	NNS	NS	NA	~~2~~ N
Fans, Ductwork	NNS	NS	NA	~~2~~ N
~~Filters~~ Dampers	NNS	NS	NA	~~2~~ N
Computer Room Air Handling System				
Air Handling Units w/ Filters	NNS	NS	NA	~~2~~ N
Fans, Ductwork Dampers	NNS	NS	NA	~~2~~ N
Vital Instrumentation and Equipment Rooms (inc. Battery Rooms)	NNS ~~3~~	NS ~~I~~	NA	N ~~1~~
Balance of Building Air Handling System				
Filters	NNS	NS	NA	~~2~~ N
Water Cooling Coils	NNS	NS	NA	~~2~~ N
Fans, Ductwork	NNS	NS	NA	~~2~~ N
Dampers	NNS	NS	NA	N
Fuel Building Ventilation System				
Cooling Coil	NNS	NS	NA	~~2~~ N
Heating Coil, Supply	NNS	NS	NA	~~2~~ N
Air Handling Unit w/Filter	NNS	NS	NA	~~2~~ N
Ductwork, Supply	NNS	NS	NA	~~2~~ N
Exhaust System Filter Train	3	I	NA	~~1~~ Q
Exhaust System Fans	3	I	NA	~~1~~ Q
Exhaust System Dampers	3	I	NA	~~1~~ Q
Ductwork, Exhaust	3	I	NA	~~1~~ Q
Dampers, Supply	NNS	NS	NA	N
Nuclear Annex ~~and Radwaste Building~~ Ventilation System (20)				
Recirculation Units	3	I	NA	Q
Supply Air Handling Units	NNS	NS	NA	~~2~~ N
Ductwork, Supply	NNS	NS	NA	~~2~~ N
Cooling Coils	NNS	NS	NA	~~2~~ N
Particulate Exhaust Filter Units	NNS	NS	NA	~~2~~ N
Fans, Ductwork	NNS	NS	NA	~~2~~ N
Dampers	NNS	NS	NA	N
Room Recirculating Unit Cooling Coils	3	I	NA	Q

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Radwaste Building Ventilation System				
Supply Air Handling Units	NNS	NS	RWF	N
Cooling Coils	NNS	NS	RWF	N
Exhaust Filter Units	NNS	NS	RWF	N
Fans	NNS	NS	RWF	N
Ductwork	NNS	NS	RWF	N
Dampers	NNS	NS	RWF	N

TABLE 3.2-1 (Cont'd)

(Sheet 12 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	~~Quality Class~~ Quality Assurance Requirement (27)
Reactor Building Subsphere Ventilation System				
Individual Cooling Units	3/NNS	I/NS	RxB	Q/N
Exhaust Fans	3	I	NA	~~1~~ Q
Cooling Coils and Heating Coils	3	I	NA	~~1~~ Q
Exhaust System Filter Train	3	I	NA	~~1~~ Q
Ductwork, Exhaust	3	I	NA/RxB	~~1~~ Q
Supply Fans	NNS	NS	NA	~~2~~ N
Supply Air Handling Units	NNS	NS	NA	~~2~~ N
Ductwork, Supply	NNS	NS	NA/RxB	~~2~~ N
Dampers, Exhaust	3	I	NA	Q
Dampers, Supply	NNS	NS	NA	N
Diesel Building Ventilation System				
Space Heater	3	I	DGB	Q
Supply/Exhaust Fans	3/NNS	I/NS	DGB	~~1~~ Q/N
Ductwork	3/NNS	I/NS	DGB	~~1~~ Q/N
Dampers	3/NNS	I/NS	DGB	~~1~~ Q/N
Filter, Normal Supply	NNS	NS	DGB	N
Annulus Ventilation System				
Filter Trains	3	I	NA	~~1~~ Q
Fans~~, Ductwork~~	3	I	NA	~~1~~ Q
Dampers	3	I	NA	~~1~~ Q
Ductwork	3	I	NA/RxB	Q
Containment Purge Ventilation System				
Water Cooling Coil	NNS	~~I~~ NS	NA	~~2~~ N
Heating Coil	NNS	~~I~~ NS	NA	~~2~~ N
Supply and Exhaust Fans	NNS	~~I~~ NS	NA	~~2~~ N
~~Dampers~~ Valves (2E)	2/NNS	I/NS	NA/SSCV	~~2~~ Q/N
Filter Trains	NNS	~~I~~ NS	NA	~~2~~ N
Ductwork (2E)	2/NNS	I/NS	NA/SSCV	Q/N
Containment Cooling and Ventilation System				
Containment Cooling Subsystem	NNS	~~I~~ NS	SSCV	~~2~~ N
Control Element Drive Mechanism Cooling Subsystem	NNS	~~I~~ NS	SSCV	~~2~~ N
Containment Air Cleanup System	NNS	~~I~~ NS	SSCV	~~2~~ N
Cavity Cooling Subsystem	NNS	~~I~~ NS	SSCV	~~2~~ N
Ductwork	NNS	~~I~~ NS	SSCV	~~2~~ N
Dampers	NNS	NS	SSCV	N

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TABLE 3.2-1 (Cont'd)

(Sheet 13 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	Quality Class / Quality Assurance Requirement (27)
Turbine Building Ventilation System				
~~Ventilating~~ Fans	NNS	NS	TB	~~2~~ N
~~Intake~~ Dampers	NNS	NS	TB	~~2~~ N
Exhausters	NNS	NS	TB	~~2~~ N
Ductwork	NNS	NS	TB	N
Station Service Water Pump Structure Ventilation System				
~~Vane-Axial Supply~~ Fans	3	I	SSPS	~~1~~ Q
Dampers~~/Check Dampers~~	3	I	SSPS	~~1~~ Q
Ductwork	3	I	SSPS	~~1~~ Q
Main Steam Supply System ~~(21)~~	~~2~~	~~I~~		~~1~~
Piping (21)				
~~Steam Generator to MSIVs~~	2	I	[illegible]	Q
Other	NNS	NS	MSVH, NA, TB	
Valves (21)				
Safety Valves				
MSIVs, MSIV Bypass Valves	2	I	MSVH	Q
Atmospheric Dump Valves	2	I	MSVH	Q
Valves	2	I	MSVH	Q
	2/NNS	I/NS	NA/MSVH,TB	Q/N
Containment Hydrogen Recombiner System ~~Combustible Gas Control System~~				
Piping (28)	2	I	NA/SSCV	Q
Hydrogen Recombiners	2	I	NA	~~1~~ Q
Hydrogen Analyzers	2	I	NA	~~1~~ Q
Hydrogen Recombiner Control Panel	~~2~~ 3	I	NA	~~1~~ Q
Valves (28)	2	I	NA/SSCV	Q
Steam Generator Blowdown System (22)				
Piping (28)	2/NNS	I/NS	SSCV/TB/MSVH	Q/N
Flash Tank	NNS	NS	TB	~~2~~ N
~~Blowdown~~ Heat Exchanger	NNS	NS	TB	~~2~~ N
~~Blowdown~~ Filter	NNS	NS	TB	~~2~~ N
~~Blowdown~~ Demineralizers	NNS	NS	TB	~~2~~ N
Valves (28)	2/NNS	I/NS	SSCV/TB/MSVH	Q/N
Steam Generator Wet Layup Recirculation System (22)				
Piping (28)	2/NNS	I/NS	SSCV/TB/MSVH	Q/N
Valves (28)	2/NNS	I/NS	SSCV/TB/MSVH	Q/N
~~Wet Layup Pumps~~	~~NNS~~	~~NS~~		~~2~~
~~Wet Layup Heat Exchangers~~	~~NNS~~	~~NS~~		~~2~~
Hydrogen Mitigation System				
Hydrogen Igniters	NNS	I	SSCV	~~2~~ N
Potable and Sanitary Water Systems	NNS	NS	YD	N

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TABLE 3.2-1 (Cont'd)

(Sheet 14 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Add Insert 6

Component Identification	Safety Class	Seismic Category	Quality Assurance Requirement (27) ~~Quality Class~~	Location
Structures				
Containment Shield Building		I		
Steel Containment		I		
Internal Structure		I		
Nuclear System Annex		I		
Station Service Water System Pump Structure		I		
Turbine Building		NS		
Radwaste Building		NS		
Component Cooling Water Hx Building		I		
Nuclear Service Water Pump Building		I		
Diesel Generator Building		I		
Main Steam Valve House		I		
Service Building		NS		
Administration Building		NS		
Guard House		NS		
Warehouse		NS		
Fire Pump House		NS		

Replace with Insert 7

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RAI 210.1

INSERT 6:

[To be inserted in CESSAR-DC Table 3.2-1 (Page 14 of 17) in response to NRC RAI 210.1]

Instrumentation and Control Systems				
Plant Protection System (PPS)				
The PPS includes the electrical and mechanical devices and circuitry (from sensors to actuation device input terminals) involved in generating the signals associated with the two protective functions defined below:				
Reactor Protective System (RPS)				
That portion of the PPS which generates signals that actuate reactor trip	-	I	NA/SSCV	Q
Engineered Safety Features Actuation System (ESF)				
That portion of the PPS which generates signals that actuate engineered safety features	-	I	NA/SSCV	Q
Safe Shutdown Systems				
The safe shutdown systems include those systems required to secure and maintain the reactor in a safe shutdown condition	-	I	DGB/NA/CCWX/ ESPS/MSVH/ RXB/SSCV	Q
All other systems required for safety	-	I	NA/DGB/CCWX/ ESPS/MSVH/ RXB/SSCV	Q
Control systems not required for safety	-	NS	All	N
Control Room Panels (safety-related)	-	I	NA	Q
Control Room Panels (other)	-	I	NA	Q
Instrument valves and piping downstream of Safety Class 2 or 3 root valves (For safety-related instruments)				
Piping, tubing, and fittings	2/3	I	All	Q
Instrument valves	NNS	NS	All	N
Electric Systems				
Class 1E AC Equipment (includes associated transformers, protective relays, instrumentation and control devices				
4.16 kV Buses	-	I	NA	Q
480V Load Centers	-	I	NA	Q
480V Motor Control Centers	-	I	NA/CCWX/DGB/ SSPS	Q

RAI 210.1

INSERT 6 (Cont'd):

[To be inserted in CESSAR-DC Table 3.2-1 (Page 14 of 17) in response to NRC RAI 210.1]

Class 1E DC Equipment				
125V Station Batteries and Racks	-	I	NA	Q
Battery Chargers	-	I	NA	Q
125V Switchgear and Distribution Panels	-	I	NA	Q
120V Vital AC System Equipment				
Inverters	-	I	NA	Q
120V Distribution Panels	-	I	NA	Q
Electrical Cables for Class 1E Systems				
125V DC Cables (including cable splices, connectors, and terminal blocks)	-	-	NA	Q
5 kV Power Cables (including cable splices, connectors, and terminal blocks)	-	-	NA/DGB/CCWX/SSPS	Q
600V Power Cables (including cable splices, connectors, and terminal blocks)	-	-	NA/DGB/CCWX/SSPS/MSVH/RXB/SSCV	Q
Control and Instrumentation Cables (including cable splices, connectors, and terminal blocks)	-	-	DGB/CCWX/NA/SSPS/MSVH/RXB	Q
Conduit and cable trays and their supports containing Class 1E cables and those whose failure during a seismic event may damage other safety-related items	-	-	DGB/CCWX/NA/SSPS/MSVH/RXB/SSCV	Q
Miscellaneous Class 1E Electrical Systems				
Containment building electrical penetration assemblies	-	-	SSCV	Q
Non-Class 1E Electrical Systems	-	-	All	N

RAI 210.1

INSERT 7:

[To be inserted in CESSAR-DC Table 3.2-1 (Page 14 of 17) in response to NRC RAI 210.1]

Structures				
Reactor Building	-	I	RXB	Q
Shield Building	2	I	SSCV	Q
Steel Containment	2	I	SSCV	Q
Internal Structure	2	I	SSCV	Q
Equipment Hatch	2	I	SSCV	Q
Personnel Air Lock	2	I	SSCV	Q
Nuclear Annex	-	I	NA	Q
Diesel Generator Building	-	I	DGB	Q
Main Steam Valve House	-	I	MSVH	Q
Turbine Building	-	NS	TB	N
Radwaste Facility	-	I	RWF	Q
Station Service Water Pump Structure	-	I	SSPS	Q
Component Cooling Water Heat Exchanger Structure	-	I	CCWX	Q
Diesel Fuel Storage	-	I	DFS	Q
Service Building	-	NS	SXB	N
Administration Building	-	NS	ADB	N
Warehouse	-	NS	WH	N
Fire Pump House	-	NS	FPH	N
Dike (CVCS Outdoor Tanks)	-	I	YD	Q
Cranes				
Polar Crane	3	II	SSCV	Q
Cask Handling Hoist	3	II	NA	Q
New Fuel Handling Hoist	3	II	NA	Q
Component Supports (23)	1/2/3/NNS	I/NS	All	QN

TABLE 3.2-1 (Cont'd)

(Sheet 16 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

and internals structures important to safety

NOTES: (Cont'd)

(7) Core support structures are designed to the criteria described in Section 3.9.5.4.

(8) CEA and fuel assemblies are designed to the criteria described in Section 4.2.

(9) Reactor coolant pump auxiliary components required for lubrication and cooling of pump seals and thrust bearings are ~~Quality Class 2~~. not subject to the quality assurance requirements of 10CFR 50, Appendix B.

Safety Class NNS

(10) Except Lifting Frame Assembly, which is NS.

(11) During normal plant operation only. D

(12) Safety Class 1 for pressure boundary; Safety Class 3 for electrical portion of system.

(13) The piping, valves, and associated supports/restraints of the Main Feedwater System from (and including) the Main Feedwater Isolation Valves to the steam generator feed nozzles are Safety Class 2, Seismic Category I, ~~Quality Class 1~~; the remainder is ~~ANSI/ASME B31.1~~. and are subject to the quality assurance requirements of 10 CFR 50, Appendix B.

(14) Non-safety Cooling Headers are Safety Class NNS, Seismic Category II, and ~~Quality Class 2~~. are not subject to the quality assurance requirements of 10CFR50, Appendix B.

(15) The Normal Chilled Water System serves no safety function. Portions of the system which are located in non-safety related areas are classed as non-seismic.

(16) Portions of the Fire Protection System piping, ~~and~~ valves, and extinguishers which are not in safety-related areas of the plant are designed as non-seismic. I

(17) Fuel Oil Recirculation System and storage tank fill line strainer are ~~classed as non-nuclear safety.~~ Safety Class NNS.

(18) The Starting Air System is Safety Class NNS from the starting air compressor through the desiccant drying towers, and Safety Class 3 from the starting air receiver tank inlet check valve to the engine connections.

TABLE 3.2-1 (Cont'd)

(Sheet 17 of 17)

CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

NOTES: (Cont'd)

(19) The Clean and Used Oil Transfer System is Safety Class NNS.

(20) Mechanical Equipment Room cooling components are Safety Class 3, Seismic Category I, and ~~Quality Class 1~~, are subject to the quality assurance requirements of 10CFR50, Appendix B.

(21) The piping, valves, and associated supports/restraints of the Main Steam System from each steam generator to (and including) the Main Steam Isolation Valves are Safety Class 2, Seismic Category I, ~~Quality Class 1~~; and are subject to the quality assurance requirements of 10CFR50, Appendix B. the remainder is ~~ANSI/ASME B31.1~~ Safety Class NNS. I

(22) Piping is Safety Class 2 from the Steam Generators through the Containment Isolation Valves.

(23) Component supports are designed to the criteria described in Section 3.9.3.4.

(24) Safety Injection drain and vent piping is Safety Class NNS, Seismic Category NS, and is not subject to the quality assurance requirements of 10CFR50, Appendix B.

(25) Locations:

CCWX = Component Cooling Water Heat Exchanger Structure
DGB = Diesel Generator Building
FPH = Fire Pump House
MSVH = Main Steam Valve House
RWF = Radwaste Facility
RXB = Reactor Building
SSCV = Steel Containment
SSPS = Station Service Water Pump Structure
SXB = Service Building
TB = Turbine Building
NA = Nuclear Annex
YD = Yard
DFS = Diesel Fuel Oil Storage
ALL = Throughout Plant

(26) Hydrogen lines in safety-related areas are either designed to seismic Category I requirements, or sleeved with the outer pipe vented to the outside, or equipped with excess flow check valves so that in case of a line break, the hydrogen concentration in the affected area will not exceed 2%.

Add Insert 8

RAI 210.1

INSERT 8:

[To be inserted in CESSAR-DC Table 3.2-1 (Page 17 of 17) in response to NRC RAI 210.1]

(27) Quality Assurance Requirement

Q = The quality assurance requirements of 10CFR50, Appendix B are applicable

N = The quality assurance requirements of 10CFR50, Appendix B, are not applicable

(28) Containment isolation valves and containment penetration piping are Safety Class 2, Seismic Category I, and are in compliance with the quality assurance requirements of 10CFR50, Appendix B.

Question 210.1

Section 3.2.2, System Quality Group Classifications (Safety Class) states, "The safety classifications of major components which are in the System 80+ design scope are listed in Table 3.2-1 and Section 3.11. Seismic category designations and quality assurance requirements are also included. Small components, such as piping, valves, and strainers, are not listed; they may be found by reference to the P&IDs (Chapters 5, 6, and 9) where the exact boundaries are indicated." The staff considers piping valves, and strainers to be major components and as such should be added to Table 3.2-1. Moreover, it is recommended that Table 3.2.1 be enhanced to include additional information to aid in the staff's overall review of the CE System 80+ design. An acceptable format would be that used for the Palo Verde Nuclear Generating Station (PVNGS) which is a recently licensed plant utilizing the System 80 design. Table 3.2-1 in the PVNGS FSAR contains a comprehensive listing of all principal components (including piping and piping supports) and provides the building location, principal codes and standards, seismic category, safety classification, and quality class that are applicable to each component.

Response 210.1

CESSAR-DC Table 3.2-1, Classification of Structures, Systems, and Components, will be revised to include major components such as piping, valves, and strainers. Table 3.2-1 and CESSAR-DC Sections 3.2.1 and 3.2.2 contain the required information to properly identify and classify structures, systems, and components important to safety. In addition to including the supplemental components referred to above, component locations will be added to enhance Table 3.2-1. These changes will be included in a future submittal of CESSAR-DC.

Question 210.2

Each system listed in the CESSAR-DC SYSTEM 80+ as having a safety related function should have a corresponding Piping and Instrumentation Diagram (P&ID). However, not all systems listed in Table 3.2-1 "Classification of Structures, Systems, and Components" has a corresponding P&ID showing component safety class, intra-system and interfacing system safety class changes, etc. which are necessary to assure the correct safety classification (and subsequently the correct seismic category) of a structure, system, or component. An example is the Diesel Generator System which is designated as safety class 3 in Table 3.2-1 but for which no P&ID for all safety related systems listed in Table 3.2-1. Include the information requested in request for additional information (RAI) 210.3.

Response 210.2

CESSAR-DC will be revised to include P&IDs for safety-related systems listed in CESSAR-DC Table 3.2-1, Classification of Structures, Systems, and Components except systems which are outside the scope of the System 80+ Standard Plant Design (i.e., Station Service Water Pump Structure Ventilation which is site specific). The System 80+ P&IDs will also be revised to include safety classifications and to indicate safety class changes. These revisions will be included in a future amendment to CESSAR-DC. See the response to 210.3 for a list of the P&IDs to be updated.

Question 210.4

Table 3.2-1 in CESSAR-DC provides safety classification information for core support structures and fuel assemblies. Revise this table to include the Safety Class, Seismic Category, and Quality Class for all other safety-related reactor internals. In addition, include this information as part of the discussions in Section 3.9.5, "Reactor Vessel Core Support and Internals Structures."

Response 210.4

Table 3.2-1 Sheet 1 and Table 3.9-16 have been modified to reflect this addition. This revision will be included in a future amendment to CESSAR-DC. (See also responses to RAIs 210.1 and 210.76.

NRC Question 210.5

Section 9.3.4.1.1 states that the Chemical and Volume Control System (CVCS) "...is designed as a non-safety-grade system and is not required to perform any accident mitigation or safe shutdown function." This statement is consistent with the P&IDs for CVCS which show the system to be safety class 4 except where it penetrates the containment and is designed as safety class 2 (Ref. Figure 9.3.4-1, Sht's 1-4). However, Table 3.2-1 lists portions of the CVCS as being safety class 2 or 3. Resolve this discrepancy.

Also, the acceptance of the CVCS being classified as non-safety in Section 3.2.2 is dependent on the resolution of RAI 440.118 concerning GSI-23 which addresses the Reactor Coolant Pump seal issue. It is questioned whether or not the CVCS is required to help prevent an RCP seal failure causing a LOCA. If this were the case then at least a portion of the CVCS would have to be classified as safety related. Final acceptance of the CVCS safety classification will be withheld pending resolution of this issue.

Response to NRC Question 210.5

The EPRI Requirements Document specifies that the Chemical and Volume Control System (CVCS) is not required to perform any safety related functions. The System 80+ CVCS conforms to this requirement by not crediting CVCS operation for accident mitigation. Thus, the CVCS is <u>not safety related</u>. Other dedicated safety related systems are available to perform accident mitigation functions. Section 9.3.4.1.1 of CESSAR-DC Amendment I contains a functional description of the CVCS which is consistent with this position.

However, the EPRI Requirements Document states that because the CVCS is not safety related, it can be classified as Non Nuclear Safety (NNS). The System 80+ CVCS classification exceeds these requirements. The System 80+ CVCS charging and letdown portions outside the reactor coolant pressure boundary, including seal injection and reactor coolant pump bleedoff, are classified as Safety Class 3 to meet Regulatory Guide 1.26 requirements.

Discrepancies in safety classifications for the CVCS, which appear in the current Amendment to CESSAR-DC on Tables 3.2-1 and 3.2-2, and Figure 9.3.4-1, will be resolved. Marked-up versions of Tables 3.2-1 and 3.2-2, and Figure 9.3.4-1 are attached for information.

Please refer to the response to RAI 440.118 for a complete discussion of CVCS operation relating to seal injection.

TABLE 3.2-2 (Cont'd)

(Sheet 5 of 15)

SAFETY CLASS 1, 2 & 3 VALVES

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
CH-374	IX isolation	~~2~~ 3	I	1
CH-375	Letdown to DRDH	~~2~~ 3	I	1
CH-376	Letdown filter	~~2~~ 3	I	1
CH-378	IX isolation	~~2~~ 3	I	1
CH-379	RSSH to IX isolation	~~2~~ 3	I	1
CH-380	IX to SWMS isolation	~~2~~ 3	I	1
CH-381	IX bypass	~~2~~ 3	I	1
CH-382, 383	IX isolation	~~2~~ 3	I	1
CH-384	IX check	~~2~~ 3	I	1
CH-385	IX bypass	~~2~~ 3	I	1
CH-386	IX vent to GWMS	~~2~~ 3	I	1
CH-387	IX resin fill isolation	~~2~~ 3	I	1
CH-389	IX isolation	~~2~~ 3	I	1
CH-390	RSSH to IX isolation	~~2~~ 3	I	1
CH-391	IX to SWMS isolation	~~2~~ 3	I	1
CH-392	IX isolation	~~2~~ 3	I	1
CH-393	Regenerative HX vent	2	I	1
CH-394	IX bypass	~~2~~ 3	I	1
CH-395	IX isolation	~~2~~ 3	I	1
CH-396	SCS check	~~2~~ 3	I	1
CH-397	SCS isolation	2	I	1
CH-398	IX isolation	~~2~~ 3	I	1
CH-399	RSSH to IX isolation	~~2~~ 3	I	1
CH-400	IX to SWMS isolation	~~2~~ 3	I	1
CH-401	IX vent to GWMS	~~2~~ 3	I	1
CH-402	IX resin fill isolation	~~2~~ 3	I	1
CH-403	IX check	~~2~~ 3	I	1
CH-404	IX isolation	~~2~~ 3	I	1
CH-407, 408	IX D/P isolation	~~2~~ 3	I	1
CH-414	Letdown strainer bypass	~~2~~ 3	I	1
CH-415	IX isolation	~~2~~ 3	I	1
CH-418	Letdown to VCT isolation	~~2~~ 3	I	1
CH-419	Letdown strainer to SWMS	~~2~~ 3	I	1
CH-420	IX effluent sample	~~2~~ 3	I	1
CH-425	Charging line pressure indicator isolation	~~2~~ 3	I	1
CH-426	Letdown sample isolation	~~2~~ 3	I	1
CH-427, 428	Charging line flow indicator isolation	~~2~~ 3	I	1
CH-431	Auxiliary spray check	1	I	1

RAI 210.5

TABLE 3.2-2 (Cont'd)

(Sheet 6 of 15)

SAFETY CLASS 1, 2 & 3 VALVES

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
CH-433	Charging line check	1	I	1
CH-434	Charging bypass line	~~1~~ 2	I	1
CH-435	Charging line bypass check	2	I	1
CH-447	Auxiliary spray check	1	I	1
CH-448	Charging line check	1	I	1
CH-450	RDH to EDT check	3	I	1
CH-459	EDT line to GWMS pressure indicator isolation	3	I	1
CH-460, 461	EDT level indicator	3	I	1
CH-462	EDT drain isolation	3	I	1
CH-464	EDT to RDP check	3	I	1
CH-465, 466	RDP suction isolation	3	I	1
CH-468, 469	RDP discharge pressure indicator isolation	3	I	1
CH-470, 471	RDP discharge check	3	I	1
CH-472, 473	RDP discharge isolation	3	I	1
CH-474	Reactor drain filter bypass	3	I	1
CH-475	RDP discharge to RDH	3	I	1
CH-476	Reactor drain filter D/P	3	I	1
CH-477, 478	Reactor drain filter	3	I	1
CH-479	Reactor drain filter D/P	3	I	1
CH-480	IDH to EDT check	3	I	1
CH-485	Pre-holdup IX to RSSH	3	I	1
CH-486	Pre-holdup IX to IDH	3	I	1
CH-488	Pre-holdup IX D/P	3	I	1
CH-489	Pre-holdup strainer to SWMS isolation	3	I	1
CH-490	Pre-holdup IX isolation	3	I	1
CH-491	Pre-holdup IX strainer	3	I	1
CH-492	Pre-holdup IX D/P	3	I	1
CH-493	Pre-holdup IX effluent sample isolation	3	I	1
CH-494	RSSH and RDP to RDH Check	2	I	1
CH-495	Pre-holdup IX to BAST	3	I	1
CH-496	Pre-holdup IX to stripper	3	I	1
CH-500	VCT bypass valve	~~2~~ 3	I	1
CH-501, 504	VCT discharge isolation	~~2~~ 3	I	1
CH-505, 506	RCP CBO contain. isol.	2	I	1
CH-507	RCP bleedoff relief isol.	2	I	1
CH-510	BAST recirculation control	3	I	1

I

RAI 210.5

TABLE 3.2-2 (Cont'd)

(Sheet 7 of 15)

SAFETY CLASS 1, 2 & 3 VALVES

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
CH-512	VCT makeup isolation	3	I	1
CH-513	VCT to GWMS isolation	~~2~~ 3	I	1
CH-514	BAMP to charging pump	3	I	1
CH-515	Letdown isolation	1	I	1
CH-516	Letdown backup isolation	1	I	1
CH-520	Ion exchanger bypass	~~2~~ 3	I	1
CH-523	Letdown isolation	2	I	1
CH-524	Charging line isolation	2	I	1
CH-527	RMW to charging pump	3	I	1
CH-530, 532	BAST discharge isolation	~~2~~ 3	I	1
CH-534	BAST to charging pump	~~2~~ 3	I	1
CH-536	BAST to charging pump	3	I	1
CH-560	RDT suction isolation	2	I	1
CH-561	RDT isolation	2	I	1
CH-562	RDH to EDT isolation	3	I	1
CH-563	EDT to RDP isolation	3	I	1
CH-564	EDT to GWMS isolation	3	I	1
CH-565	Pre-holdup IX bypass	3	I	1
CH-566	Gas stripper bypass to EDT	3	I	1
CH-567	Gas stripper to VCT control	3	I	1
CH-580	RMWS to RDT isolation	2	I	1
CH-590, 591	Charging flow control	~~2~~ 3	I	1
CH-600, 601, 602, 603	Letdown orifice isolation	2	I	1
CH-612, 613	Seal injection line vent	~~2~~ 3	I	1
CH-614	Seal injection vent	3	I	1
CH-639	Chemical addition check	~~2~~ 3	I	1
CH-645	VCT gas supply isolation	~~2~~ 3	I	1
CH-646	RCP bleedoff check	~~2~~ 3	I	1
CH-647	BAST recirc check	~~2~~ 3	I	1
CH-648	BAST recirc sample	3	I	1
CH-649	BABE isolation	3	I	1
CH-653	F-210Y isolation	3	I	1
CH-654	MSH to gas stripper	3	I	1
CH-655	Pre-holdup IX to radiation monitor	3	I	1
CH-656	Gas stripper to HT	3	I	1
CH-657	EDT relief to misc radioactive sump	3	I	1
CH-660	Gas stripper inlet	3	I	1
CH-663	Reactor drain filter vent	3	I	1

I

RAI 210.5

TABLE 3.2-2 (Cont'd)

(Sheet 8 of 15)

SAFETY CLASS 1, 2 & 3 VALVES

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class	
CH-665	RDP discharge sample	3	I	1	
CH-668	BAM line to VCT check	3	I	1	
CH-686	Holdup pump bypass to reactor drain filter	3	I	1	
CH-700	Hydrostatic test connection	2	I	1	
CH-701	Charging pump suction	~~2~~ 3	I	1	
CH-702	Charging suction pressure	~~2~~ 3	I	1	
CH-703, 704	Charging pump bypass	~~2~~ 3	I	1	
CH-705, 706	Charging pump discharge	~~2~~ 3	I	1	
CH-710	Hydrostatic test connection	~~2~~ 3	I	1	
CH-712	Charging suction pressure	~~2~~ 3	I	1	
CH-713	Charging pump suction	~~2~~ 3	I	1	
CH-714, 715	Charging pump bypass	~~2~~ 3	I	1	
CH-716	Charging miniflow HX vent	3	I	1	
CH-717	Charging recirc relief	~~2~~ 3	I	1	
CH-719	Charging pump discharge	~~2~~ 3	I	1	
CH-721, 722	Letdown to pre-holdup IX	3	I	1	
CH-723	Reactor drain line sample	3	I	1	
CH-724	Pre-holdup IX isolation	3	I	1	
CH-725	Pre-holdup IX check	3	I	1	
CH-726	Pre-holdup IX resin fill	3	I	1	
CH-727	Pre-holdup IX D/P	3	I	1	
CH-728	Pre-holdup IX vent	3	I	1	
CH-730	Pre-holdup IX to SWMS	3	I	1	
CH-747	Charging line check	2	I	1	
CH-750	Charging flow control iso.	~~2~~ 3	I	1	
CH-751	Regen. HX charging isolation	2	I	1	
CH-753	BAMP to PCPS	3	I	1	I
CH-754	Charging flow control iso.	~~2~~ 3	I	1	
CH-760, 761	Charging recirculation check	2	I	1	
CH-763	Charging pump discharge	~~2~~ 3	I	1	
CH-764, 766	Charging flow control iso.	~~2~~ 3	I	1	
CH-768	Chemical addition line	~~2~~ 3	I	1	
CH-769, 770	Charging miniflow HX iso.	~~2~~ 3	I	1	
CH-787	Seal injection check	1	I	1	
CH-789, 800	Seal injection flow	2	I	1	
CH-802	Seal injection check	1	I	1	
CH-804, 805	Seal injection flow	2	I	1	
CH-807	Seal injection check	1	I	1	
CH-809, 810	Seal injection flow	2	I	1	

RAI 210.5

TABLE 3.2-2 (Cont'd)

(Sheet 9 of 15)

SAFETY CLASS 1, 2 & 3 VALVES

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class	
CH-812	Seal injection check	1	I	1	
CH-814, 815	Seal injection flow indicator	2	I	1	
CH-816, 818, 819, 821	Seal injection filter	~~2~~ 3	I	1	
CH-822, 823	Seal injection to DRDH	~~2~~ 3	I	1	
CH-825, 826	Seal injection filter D/P	~~2~~ 3	I	1	
CH-830	Nitrogen supply to EDT	3	I	1	
CH-831	Nitrogen supply pressure control	3	I	1	
CH-835	Seal injection check	2	I	1	
CH-839	Seal injection isolation	~~2~~ 3	I	1	
CH-844, 845	Seal injection filter vent	~~2~~ 3	I	1	
CH-858	RSSH line to EDT check	3	I	1	
CH-861	RSSH to EDT isolation	3	I	1	
CH-862	RMWT supply to RDT	3	I	1	
CH-865	Seal injection relief	~~2~~ 3	I	1	
CH-866, 867, 868, 869	Seal injection check	1	I	1	
Pool Cooling and Purification System (PCPS) (1)					
PC-200, 210	Cooling HX inlet pressure	3	I	1	
PC-201, 293	Cooling HX cross-connect	3	I	1	
PC-202, 203	Cooling pump suction isolation	3	I	1	
PC-204, 205	Cooling pump discharge pressure	3	I	1	
PC-206, 207	Cooling pump discharge check	3	I	1	
PC-208, 209	Cooling pump disch. isolation	3	I	1	
PC-211, 212	Cooling HX inlet isolation	3	I	1	
PC-213, 214	Cooling HX outlet isolation	3	I	1	I
PC-249	IRWST return line isolation	3	I	1	
PC-257, 258	Refueling pool disch. isolation	2	I	1	
PC-291, 292	Refueling pool inlet isolation	2	I	1	
PC-300, 301, 302, 303	Cooling flow indication iso.	3	I	1	
Safety Depressurization System (SDS)					
RC-406, 407, 408, 409	Rapid depressurization	1	I	1	
RC-410, 411, 412, 413	Pressurizer vent	1	I	1	
RC-414, 415, 416, 417	Reactor vessel vent	2	I	1	
RC-418	RCGVS vent to RDT	2	I	1	
RC-419	RCGVS vent to IRWST	2	I	1	
RC-263, 264	RD pressure indication	2	I	1	
RC-267	RCGVS pressure indication	2	I	1	

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Question 210.6

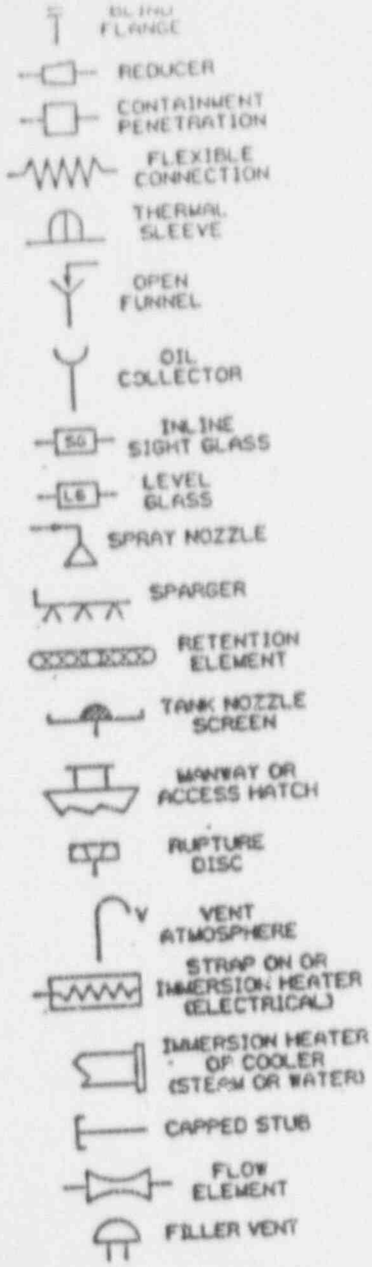
Section 3.2.2, System Quality Group Classifications (Safety Class), states, "For purposes of CESSAR, Safety Class 1, 2, 3, and NNS of ANSI/ANS 51.1 are equivalent to Quality Groups A, B, C, and D of Regulatory Guide (RG) 1.26." While the staff has not formally endorsed ANSI/ANS 51.1, the classification of structures, systems, and components in CESSAR-DC appears to meet the guidelines of RG 1.26 in that the Quality Groups A, B, C, and D in RG 1.26 are found to be equivalent to SC-1, SC-2, SC-3 and NNS as listed in Table 3.2-1 and defined in Section 3.2.2. The safety class designations as given in Section 3.2.2 should be used exclusively throughout CESSAR-DC. Contrary to this, there are other safety class designations used. For example, on the P & ID's the numeral 4 is used in place of NNS to designate non-nuclear-safety related portions of systems. In Section 9.1.3.2.1, reference is made to the system being "Quality Group C" rather than SC-3. Resolve any such discrepancies.

Response 210.6

Due to space limitations, CE limited the field for the safety class designation to one digit on all P&IDs. Consequently, the numeral 4 was used instead of NNS to designate non-nuclear-safety related portions of systems. A statement will be added to Fig 1.7-1, drawing number E-ALWR-310-100 - Piping and Instrumentation Diagram Symbols, and Section 3.2.2 of CESSAR-DC that will define the use of numeral 4 in regard to safety classification.

All discrepancies pertaining to Safety Class designation in CESSAR-DC, Chapter 9 will be corrected.

RAI 210.6



18. The Numeral 4 used in Safety Class designation is an equivalent to the Non Nuclear Safety (NNS) designation of ANSI/ANS 51.1.

17. THE INDICATION (I) AND/OR RECORDING (R) DESIGNATION FOR A SENSOR SHOWS THAT THIS VALUE WILL BE INDICATED OR RECORDED AS APPROPRIATE. THESE FUNCTIONS MAY BE ACCOMPLISHED VIA THE DIAS, THE DPS, OR INDIVIDUAL INDICATORS AND/OR RECORDERS

16. THE SYMBOL IDENTIFIES WHERE THE SENSED VARIABLE IS TO BE DISPLAYED. AN INDIVIDUAL SENSOR OUTPUT MAY BE INDICATED VIA THE DIAS, THE DPS OR AN INDIVIDUAL INDICATOR.

15. ALARMS MAY BE EQUIPMENT, PROCESS OR PLANT MODE DEPENDENT. ALARMS WILL BE DERIVED FROM DIAS VALIDATED PARAMETERS WHEN MULTIPLE SENSORS MEASURING THE SAME PROCESS ARE PRESENT.

14. P&I DIAGRAMS DO NOT IDENTIFY SCOPE OF SUPPLY

13. VALVES WITH DIAPHRAGM, PISTON AND MOTOR OPERATORS ARE EQUIPPED WITH POSITION SWITCHES

12. INSTRUMENT BLOCK VALVES ARE SHOWN ON MEASUREMENT CHANNEL BLOCK DIAGRAMS

11. LOCAL ALARMS ARE GROUPED ON LOCAL PANELS. EACH LOCAL PANEL(S) HAS A CONTROL ROOM ALARM.

10. HAND SWITCHES OR CONTROLS FOR REMOTELY OPERATED VALVES NOT RECEIVING A PROPORTIONAL CONTROL SIGNAL FROM A MEASUREMENT CHANNEL ARE NUMBERED THE SAME AS THE VALVE

9. MEASUREMENT CHANNELS ARE IDENTIFIED AS FOLLOWS: SUFFIXES MAY OR MAY NOT APPEAR AS PART OF CHANNEL NUMBER. THE FOLLOWING RULES APPLY WHEN A SUFFIX APPEARS AS PART OF CHANNEL NUMBER

9A. SAFETY CHANNEL NUMBERS ARE SUFFIXED BY A,B,C & D

9B. SIMILAR MEASUREMENT CHANNELS MAY BE SUFFIXED BY X, Y, Z, ETC.

9C. H & C WHEN USED IN INSTRUMENTATION CHANNELS DENOTE HOT AND COLD

9D. WHEN H & C IS USED WITH SAFETY CHANNELS H OR C IS THE 1ST LETTER AND A, B, C OR D IS THE SECOND

8. SAFETY CLASSES AS INDICATED ON THE PIPING AND INSTRUMENTATION DRAWINGS ARE USED AS A GUIDE TO MINIMUM CODE STANDARDS OF COMPONENTS IN ACCORDANCE WITH THE ASME BOILER AND PRESSURE VESSEL CODE

7. PROCESS DATA POINTS ARE REPRESENTED BY (X) AND DEFINED IN THE SAR REPORT

6. ALL MOTOR OPERATED VALVES ARE EQUIPPED WITH HANDWHEELS

5. FOR INSTRUMENTATION DETAILS, SEE THE APPLICABLE MEASUREMENT CHANNEL BLOCK DIAGRAM

4. E/P CONVERTERS OR POSITIONERS, AND PILOT SOLENOID VALVES CARRY RESPECTIVE VALVE NUMBERS AND ARE NOT NUMBERED ON P&ID'S

3. AIR SOLENOID VALVES ARE SHOWN WITH PORTS IN NORMAL OPERATING POSITION

2. THE NUMBERING OF VALVES CONTROLLED BY A PROPORTIONAL SIGNAL WILL BE THE SAME AS THE CONTROLLING INSTRUMENT OR SWITCH NO. VALVES OPERATED BY THE SAME INSTRUMENT CHANNEL ARE DIFFERENTIATED BY A LETTER SUFFIX (E, F, G, H, ETC.).

1. THE INSTRUMENT IDENTIFICATION LETTERS OF TABLE 1 DENOTE CHANNEL FUNCTION, NOT SPECIFIC EQUIPMENT. (SEE NOTE 5)

REFERENCE DRAWING LIST

DRAWING TITLE	MEASUREMENT CHANNEL BLOCK DIAG. INDEX
REACTOR COOLANT SYSTEM P&I DIAGRAM	100
REACTOR COOLANT PUMP P&I DIAGRAM	
PRESSURIZER & SAFETY DEPRESSURIZATION SYSTEM P&I DIAGRAM	
MAIN STEAM SYSTEM FLOW DIAGRAM	400
CHEMICAL & VOLUME CONTROL SYSTEM P&I DIAGRAM	200
SAFETY INJECTION SYSTEM P&I DIAGRAM	300
INCONTAINMENT WATER STORAGE SYSTEM P&I DIAGRAM	
POOL COOLING & PURIFICATION SYSTEM P&I DIAGRAM	500
[illegible] SAMPLING	[illegible]

E-ALWR-310-100

00

F

E

D

C

A. Safety Class 1 (SC-1) applies to pressure-retaining portions and supports of mechanical equipment that form part of the RCPB whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability and whose requirements are within the scope of the ASME Boiler and Pressure Vessel Code, Section III.

B. Safety Class 2 (SC-2) applies to pressure-retaining portions and supports of primary containment and other mechanical equipment, requirements for which are within the scope of the ASME Boiler and Pressure Vessel Code, Section III, that are not included in SC-1 and are designed and relied upon to accomplish the nuclear safety functions defined in ANSI/ANS 51.1, Section 3.3.1.2.

C. Safety Class 3 (SC-3) applies to equipment, not included in SC-1 or -2, that is designed and relied upon to accomplish the nuclear safety functions defined in ANSI/ANS 51.1, Section 3.3.1.3.

D. Non-Nuclear Safety (NNS) applies to equipment that is not in Safety Class 1, 2, or 3. This equipment is not relied upon to perform a nuclear safety function.

E. INSERT Attachment A HERE

The safety classifications of major components which are in the System 80+ design scope are listed in Table 3.2-1 and Section 3.11. Seismic category designations and quality assurance requirements are also included. Small components, such as piping, valves and strainers, are not listed; they may be found by reference to the P&IDs (Chapters 5, 6, and 9) where the exact boundaries are indicated. Valves are listed in Tables 3.2-2.

All pressure containing components in Safety Classes 1, 2, and 3 are designed, manufactured, and tested in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III. Components designated NNS are designed and constructed with appropriate consideration of the intended service using applicable industry codes and standards. The relationship between safety class and code class is shown in Table 3.2-2. A higher code class may be used for a component without changing the safety class or affecting the balance of the system in which it is located.

Fracture toughness requirements are imposed on materials for pressure retaining parts of ASME Class 2 and 3 System 80+ Standard Design components. Test methods, acceptance, and exemption criteria are in conformance with the ASME Code, Section III.

The safety classification system is also used to identify those components to which the requirements of 10 CFR 50, Appendix B,

RAI 210.6

Attachment A

The Safety Class four (SC-4) designation used on P & IDs is an equivalent to Non Nuclear Safety (NNS)

 RAI 210.6

by use of the system's skimmers. The cleanup system is designed for a flow rate sufficient to ensure adequate circulation of the entire spent fuel pool water volume, and to maintain the specified water chemistry.

The boron concentration in the spent fuel pool water is maintained at approximately the same concentration as in the refueling water. Provisions are made to make up water to the spent fuel pool. The makeup water meets all specified water chemistry requirements.

9.1.3.1.4 System Capacity Bases

For all normal plant operations and normal spent fuel pool heat load conditions, the maximum spent fuel pool bulk water temperature is 120°F. Under heat load conditions of spent fuel in all usable rack spaces (Section 9.1.2.2.2), which includes, as a minimum, a full core offload with 10 years of irradiated fuel in the pool, the maximum bulk water temperature is 140°F. Given a single active failure, the maximum temperatures for normal conditions or a full core offload are 140°F or 180°F respectively. The normal heat load is the decay heat which occurs when an accumulation of spent fuel equal to 10 full power years is in the spent fuel pool, with the newest spent fuel batch having just been placed in the pool during refueling at 120 hours after shutdown. The full core offload heat load is equal to the normal heat load plus the addition of the decay heat from a full core offload 120 hours after shutdown. Design heat loads are evaluated utilizing ANSI/ANS 5.1 (proposed version approved by Subcommittee ANS-5 ANS Standard Committee, October 1971) decay heat correlations. I

A Seismic Category I, ~~Quality Group B~~ Safety Class 2 (SC-2) borated makeup water source is provided to the spent fuel pool. Nonborated water from a non-seismic source is used to make up for the evaporation losses from the spent fuel pool during normal operation.

9.1.3.2 System Description

9.1.3.2.1 General Description

The safety-related spent fuel pool cooling system consists of two independent cooling trains. The system is located in a Seismic Category I building which provides protection from the effects of natural phenomena and missiles. The spent fuel pool cooling system (piping, pumps, valves, and heat exchangers) is safety-related, ~~Quality Group C.~~ Safety Class 3 (SC-3) The spent fuel pool receives

RAI 210.6

Safety Class 2 (SC-2)

normal borated makeup water from a water source which is Seismic Category I, ~~Quality Group B.~~ Non-safety-related sources provide normal nonborated demineralized water to the spent fuel pool.

Each cooling train incorporates one heat exchanger and pump and associated piping, valving, and instrumentation. Each cooling train is designed to service the spent fuel pool, with designed spent fuel assembly loading, and to maintain the bulk fluid temperature of the spent fuel pool below 120°F during normal operation.

The spent fuel pool cooling system removes decay heat from fuel stored in the spent fuel pool. Spent fuel is placed in the pool during the refueling sequence and stored there for decay heat removal. Heat is transferred from the spent fuel pool cooling system, through one of two heat exchangers, to the component cooling water system.

When either cooling train is in operation, water flows from the spent fuel pool to the spent fuel pool cooling pump suction, is pumped through the tube side of the heat exchanger, and is returned to the pool. The spent fuel pool suction connections enter near the normal water level so that the pool cannot be gravity drained. The return line contains an antisiphon device, also to prevent gravity drainage of the pool. To assist in maintaining spent fuel pool water clarity, pool surface is cleaned by a skimmer. I

Each of the two pool purification trains consists of a strainer, a pump, a filter and a demineralizer to maintain spent fuel pool, or refueling pool, water clarity and purity. This purification loop is sufficient for removing fission products and other contaminants which may be introduced if a leaking fuel assembly is transferred to the spent fuel pool. Either cleanup train may be used to clean and purify the refueling water while spent fuel pool heat removal operations proceed.

The spent fuel pool water is separated from the water in the transfer canal by a gate. The gate is installed so that the transfer canal may be drained to allow maintenance of the fuel transfer equipment.

9.1.3.2.2 Component Description

The PCPS cooling pumps and heat exchangers are Safety Class 3 and are designed to ASME B&PV Code, Section III, Subsection ND rules. The pool purification pumps, filters, strainers, and demineralizers are designed as non-nuclear safety. All PCPS containment isolation valves and associated piping are Safety Class 2, and are designed to ASME B&PV Code, Section III, Subsection NC rules.

Question 210.7

Table 3.2-1 in the CESSAR-DC classifies the new and spent fuel storage racks as Non-Nuclear Safety, Seismic Category I, and Quality Class 2. To be consistent with RG 1.29, the staff's position is that, as a minimum, these components should be classified as Quality Class 1 in addition to being Seismic Category I. Revise Table 3.2-1 and any other applicable Section in the CESSAR-DC to reflect the above staff position.

Response 210.7

Combustion Engineering believes that QC-1 for new and spent fuel racks is un-necessarily restrictive. ANSI/ANS-51.1 classifies new and spent fuel racks as NNS, Seismic Catagory I and quality class "consistent with QA requirements considered good practice for power plants" which is a lower quality level than QC 1. C-E has established Quality Class 2 for items that are not safety related, consistent with ANSI 51.1. This position is also consistent with ANS-57.2 design requirements for spent fuel storage facilities and SRP 9.1.2. All of the existing C-E plants have been licensed for operation as a quality class less restrictive than current Quality Class 2.

Question 210.9

In Table 1.8-1, RGs where RG 1.151, Instrument Sensing Lines, is listed, CESSAR-DC Sections 5.1.4, 7.1.2.3(sic 7.1.2.31), and 7.1.3 (E) are the only sections referenced as being applicable. It appears that by referencing only these sections that the intent is to apply the separation and independence guidelines of RG 1.151 and not necessarily positions C.2 and C.3 of the RG relative to the design and construction of instrument sensing lines and their supports. Add a statement to Sections 3.2.1 and 3.2.2 committing to positions C.2 and C.3 of RG 1.151

Response 210.9

Sections 5.1.4 and 7.1.3 (E) were removed by Amendment I to CESSAR-DC. Sections 3.2.1 and 3.2.2 will be revised to specify that instrument sensing lines are designed in accordance with ASME Code Class and Seismic Category criteria of RG 1.151.

Table 1.8-1, under RG 1.151, will be revised to list Sections 3.2.1, and 3.2.2.

The seismic category and safety and quality classification of structures, systems, and components within the System 80+ Standard Design are listed in Table 3.2-1 and on the P&IDs (Chapters 5, 6, and 9). Seismic Category I includes all mechanical components within the safety class boundaries and extends to the first seismic restraint beyond the boundary. All fuel racks are also designated as Seismic Category I. Structures, systems, or components whose failure could reduce the performance of a safety function by a Seismic Category I component to an unacceptable safety level are designed to Seismic Category II requirements for structural integrity only or are separated to the extent required to eliminate that possibility. This ensures that any structures, systems, or components that could potentially have a disabling interaction with Seismic Category I structures, systems, or components are either prevented from doing so or are designed to meet Seismic Category I or II structural integrity requirements, depending on the function of the component. I I D

The listing of major electrical components is found in Section 3.11, which also includes safety and quality classifications. Electrical structures, systems, and components not classified as Seismic Category I but whose failure could represent a hazard to the operator or could interfere with the performance of required safety functions of electrical structures, systems and components, are classified as Seismic Category II (Reference 1). Any electrical system or structure or component not in Seismic Category I or II is considered Non-Seismic (see Section 3.10). The use of the Seismic Category II designation for electrical components is limited to non-safety control system components which are designed and documented to maintain structural ~~integrity during an SSE.~~ D

INSERT Attachment I. HERE

~~For purposes of this discussion~~, the motors and solenoids used to provide motive power to mechanical components are treated as part of the mechanical component.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS (SAFETY CLASS)

In general, fluid system components important to safety are classified in accordance with ANSI/ANS 51.1 (Reference 2). For purposes of CESSAR, Safety Class 1, 2, 3 and NNS of ANSI/ANS 51.1 are equivalent to Quality Groups A, B, C and D of Regulatory Guide 1.26. The criteria establish safety classes which are used as a guide to the selection of codes, standards, and quality assurance provisions for the design and construction of the components. The safety class designations are also used as a guide to those fluid system components to be classified as Seismic Category I and II (see Section 3.2.1). The Safety Class definitions in ANSI/ANS 51.1 are summarized as follows: D D

RAI 210.9

Attachment 1

Instrument sensing lines and their supports are designed in accordance with the seismic category criteria of Regulatory Guide 1.151.

 RAI 210.9

A. Safety Class 1 (SC-1) applies to pressure-retaining portions and supports of mechanical equipment that form part of the RCPB whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability and whose requirements are within the scope of the ASME Boiler and Pressure Vessel Code, Section III.

B. Safety Class 2 (SC-2) applies to pressure-retaining portions and supports of primary containment and other mechanical equipment, requirements for which are within the scope of the ASME Boiler and Pressure Vessel Code, Section III, that are not included in SC-1 and are designed and relied upon to accomplish the nuclear safety functions defined in ANSI/ANS 51.1, Section 3.3.1.2.

C. Safety Class 3 (SC-3) applies to equipment, not included in SC-1 or -2, that is designed and relied upon to accomplish the nuclear safety functions defined in ANSI/ANS 51.1, Section 3.3.1.3.

D. Non-Nuclear Safety (NNS) applies to equipment that is not in Safety Class 1, 2, or 3. This equipment is not relied upon to perform a nuclear safety function.

The safety classifications of major components which are in the System 80+ design scope are listed in Table 3.2-1 and Section 3.11. Seismic category designations and quality assurance requirements are also included. Small components, such as piping, valves and strainers, are not listed; they may be found by reference to the P&IDs (Chapters 5, 6, and 9) where the exact ~~boundaries are indicated.~~ Valves are listed in Tables 3.2-2.

INSERT ATTACHMENT 2 HERE

~~All pressure containing components~~ in Safety Classes 1, 2, and 3 are designed, manufactured, and tested in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III. Components designated NNS are designed and constructed with appropriate consideration of the intended service using applicable industry codes and standards. The relationship between safety class and code class is shown in Table 3.2-2. A higher code class may be used for a component without changing the safety class or affecting the balance of the system in which it is located.

Fracture toughness requirements are imposed on materials for pressure retaining parts of ASME Class 2 and 3 System 80+ Standard Design components. Test methods, acceptance, and exemption criteria are in conformance with the ASME Code, Section III.

The safety classification system is also used to identify those components to which the requirements of 10 CFR 50, Appendix B,

RAI 210.9

Attachment 2

Instrument sensing lines and their supports are designed in accordance with the ASME code class requirements of Regulatory Guide 1.151.

RAI 210.9

TABLE 1.8-1 (Cont'd)

(Sheet 18 of 19)

REGULATORY GUIDES

Document/Title GDC References	Original or Revision Issue Date	Reference CESSAR Section
Reg. Guide 1.144 - Auditing of Quality Assurance Programs for Nuclear Power Plants		Not Applicable
Reg. Guide 1.145 - Atmospheric Dispersion Models for Potential Accident Consequence Assessment at Nuclear Power Plants	Revision 1 11/82	2
Reg. Guide 1.146 - Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants		Not Applicable
Reg. Guide 1.147 - In-service Inspection Code Case Acceptability, ASME Section XI, Division 1	Revision 6 5/88	5.2.1.2
Reg. Guide 1.148 - Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants	3/81	3, 5, 6
Reg. Guide 1.149 - Nuclear Power Plant Simulators for Use in Operator Training	Revision 1 4/87	Not Applicable
Reg. Guide 1.150 - Ultrasonic Testing of Reactor Vessel Welds During Pre-service and In-service Examinations	Revision 1 2/83	5.1.4, 5.3.1.3
Reg. Guide 1.151 - Instrument Sensing Lines	7/83	~~5.1.4,~~ 7.1.2.31, ~~7.1.3(E)~~ 3.2.1 3.2.2 Add
Reg. Guide 1.152 - Criteria for Programmable Digital Computer System Software in Safety Systems of Nuclear Power Generating Stations	11/85	7.1.2.32

E

Question 210.10

The seismic category, safety class, and QA requirements for piping supports and component supports is not clearly defined in CESSAR-DC. Include in either Table 3.2-1 or Sections 3.2.1 and 3.2.2 a clear commitment that piping supports and component supports will have the same seismic category, safety class, and QA requirements as the piping and components to which they apply.

Response 210.10

CESSAR-DC Sections 3.2.1 and 3.2.2 will be revised to include verification that piping supports and component supports will have the same seismic category, safety class, and QA requirements as the piping and components to which they apply. This revision will be included in a future amendment to CESSAR-DC.

NRC RAI 210.10

{Piping supports and component supports are of the same seismic category as the piping and components to which they apply.

The seismic category and safety and quality classification of structures, systems, and components within the System 80+ Standard Design are listed in Table 3.2-1 and on the P&IDs (Chapters 5, 6, and 9). Seismic Category I includes all mechanical components within the safety class boundaries and extends to the first seismic restraint beyond the boundary. All fuel racks are also designated as Seismic Category I. Structures, systems, or components whose failure could reduce the performance of a safety function by a Seismic Category I component to an unacceptable safety level are designed to Seismic Category II requirements for structural integrity only or are separated to the extent required to eliminate that possibility. This ensures that any structures, systems, or components that could potentially have a disabling interaction with Seismic Category I structures, systems, or components are either prevented from doing so or are designed to meet Seismic Category I or II structural integrity requirements, depending on the function of the component. | I | I | D

The listing of major electrical components is found in Section 3.11, which also includes safety and quality classifications. Electrical structures, systems, and components not classified as Seismic Category I but whose failure could represent a hazard to the operator or could interfere with the performance of required safety functions of electrical structures, systems and components, are classified as Seismic Category II (Reference 1). Any electrical system or structure or component not in Seismic Category I or II is considered Non-Seismic (see Section 3.10). The use of the Seismic Category II designation for electrical components is limited to non-safety control system components which are designed and documented to maintain structural integrity during an SSE. | D

For purposes of this discussion, the motors and solenoids used to provide motive power to mechanical components are treated as part of the mechanical component.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS (SAFETY CLASS)

In general, fluid system components important to safety are classified in accordance with ANSI/ANS 51.1 (Reference 2). For purposes of CESSAR, Safety Class 1, 2, 3 and NNS of ANSI/ANS 51.1 are equivalent to Quality Groups A, B, C and D of Regulatory Guide 1.26. The criteria establish safety classes which are used as a guide to the selection of codes, standards, and quality assurance provisions for the design and construction of the components. The safety class designations are also used as a guide to those fluid system components to be classified as Seismic Category I and II (see Section 3.2.1). The Safety Class definitions in ANSI/ANS 51.1 are summarized as follows: | D | D are applicable. Components in Safety Classes 1, 2, and 3* are designed and manufactured under a rigorous quality assurance program reflecting the requirements of Appendix B, and are designated Quality Class 1. The Quality Class 1 quality assurance program is described in Chapter 17. Components which do not serve a safety related function are designated Quality Class 2. Quality Class 2 components will be designed and manufactured or procured in accordance with the pertinent requirements of the Quality Assurance Program as given in Chapter 17. D

I

The use of the above outlined safety and quality classification systems meets the intent of Regulatory Guide 1.26 and the requirements of 10 CFR 50.55a.

Piping supports and component supports are of the same safety class and have the same QA requirements as the piping and components to which they apply.

* With the following exception: the CVCS gas stripper is Safety Class 3, Quality Class 2, however, pressure retaining portions meet rules applicable to ASME Code Class 3 components. See Table 3.2-1. D

Question 210.11

Section 3.2.1 discusses structures, systems, and components that are designated as Seismic Category II, but it does not provide the design criteria to be imposed. To be consistent with applicable portions of RG 1.29, SRP 3.7.3, and SRP 3.9.2, add a statement to CESSAR-DC Section 3.2.1 that all structures, systems, and component classified as Seismic Category II shall be analyzed in accordance with the same seismic criteria that is applicable to Seismic Category I structures, systems, and components.

Response 210.11

In accordance with USNRC Regulatory Guide 1.29, the design and construction criteria for Seismic Category II structures, systems, and components are that the SSE not cause failure of such items that could reduce the performance of a safety function by a Seismic Category I component to an unacceptable safety level and/or result in incapacitating injury to occupants of the control room. As defined in CESSAR-DC Section 3.2.1, these criteria are met by either separation to the extent required to eliminate that possibility or designed to meet Seismic Category I or II structural integrity requirements, depending on the function of the component. Structural integrity may be demonstrated by analyses, testing, or a combination thereof, depending on the nature of the structure, system, or component.

RAI 210.11

Structural integrity requirements may be demonstrated by dynamic or equivalent static analyses, testing, or a combination thereof.

Analyses of Seismic Category II structures, systems, and components are in accordance with the seismic input and methodology criteria described in Sections 3.7.2 and 3.7.3.

The seismic category and safety and quality classification of structures, systems, and components within the System 80+ Standard Design are listed in Table 3.2-1 and on the P&IDs (Chapters 5, 6, and 9). Seismic Category I includes all mechanical components within the safety class boundaries and extends to the first seismic restraint beyond the boundary. All fuel racks are also designated as Seismic Category I. Structures, systems, or components whose failure could reduce the performance of a safety function by a Seismic Category I component to an unacceptable safety level are designed to Seismic Category II requirements for structural integrity only or are separated to the extent required to eliminate that possibility. This ensures that any structures, systems, or components that could potentially have a disabling interaction with Seismic Category I structures, systems, or components are either prevented from doing so or are designed to meet Seismic Category I or II structural integrity requirements, depending on the function of the component.

The listing of major electrical components is found in Section 3.11, which also includes safety and quality classifications. Electrical structures, systems, and components not classified as Seismic Category I but whose failure could represent a hazard to the operator or could interfere with the performance of required safety functions of electrical structures, systems and components, are classified as Seismic Category II (Reference 1). Any electrical system or structure or component not in Seismic Category I or II is considered Non-Seismic (see Section 3.10). The use of the Seismic Category II designation for electrical components is limited to non-safety control system components which are designed and documented to maintain structural integrity during an SSE.

For purposes of this discussion, the motors and solenoids used to provide motive power to mechanical components are treated as part of the mechanical component.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS (SAFETY CLASS)

In general, fluid system components important to safety are classified in accordance with ANSI/ANS 51.1 (Reference 2). For purposes of CESSAR, Safety Class 1, 2, 3 and NNS of ANSI/ANS 51.1 are equivalent to Quality Groups A, B, C and D of Regulatory Guide 1.26. The criteria establish safety classes which are used as a guide to the selection of codes, standards, and quality assurance provisions for the design and construction of the components. The safety class designations are also used as a guide to those fluid system components to be classified as Seismic Category I and II (see Section 3.2.1). The Safety Class definitions in ANSI/ANS 51.1 are summarized as follows:

Question 210.33

CESSAR-DC Section 3.7.2.1.2.2 describes the composite mathematical model used in the analyses of the dynamically coupled components of the reactor coolant system (RCS). The model is described as having 36 mass points and 96 degrees of freedom (DOFs).

Provide data to show that the number of mass points and degrees of freedom in the RCS model are adequate in accordance with the acceptance criterion of SRP 3.7.2, Rev. 2 Subsection II.1.a. (iii).

In addition, in the description of the model, each branch of the hot leg is described as represented by a single mass point with 2 DOF. Provide justification for the use of only 2 DOFs for the hot leg mass point.

Response 210.33

The uncoupled RCS has 50 modes which are active in the coupled analysis. The ratio of degrees of freedom to modes is 1.92 which essentially meets the requirements of SRP 3.7.2 Rev. 2 Subsection II.1.a.

The dynamic properties of each hot leg are represented by three nodes, with a total of 8 DDOF. Two of the nodes, with 3 DDOF each are at either end of the hot leg, one on the RV and one on the SG. The single node with two DDOF is located at about the midpoint of the hot leg and represents the lateral modes of the hot leg, while the axial modes are represented by the nodes on the SG and RV.

Question 210.34

CESSAR-DC Section 3.7.2.1.2.2 states that a multimass model was used for the surge line dynamic analvsis. The model is shown in Figure 3.7-34. However, no details of the number of mass points and degrees of freedom are provided.

Provide the number of mass points and degrees of freedom in the surge line model. Also show that these numbers are adequate in accordance with the acceptance criteria of SRP 3.7.2, Rev. 2, Subsection II.1.a.(iii).

Response 210.34

The representative model of the surge line shown in Figure 3.7-34 has sixteen mass points, each with three degrees of freedom (x, y and z). The number of mass points required for the System 80+ surge line will be determined during the detailed design. The number of mass points and degrees of freedom will meet the requirements of the acceptance criteria of SRP 3.7.2, Rev. 2, Subsection II.1.a.(iii).

CESSAR-DC Section 3.7.2.1.2.2 and Figure 3.7-34 will be revised in a future submittal to note that Figure 3.7-34 is a representive surge line seismic analysis model.

RAI 210.34

A schematic diagram of the composite mathematical model used in the analyses of the dynamically coupled components of the reactor coolant system is presented in Figures 3.7-32 and 3.7-33. This model includes 36 mass points with a total of 96 dynamic degrees of freedom to represent the RCS, including the pressurizer. Additional mass points and dynamic degrees of freedom, not shown in the figures, are used to represent the containment building and interior structures in the coupled seismic model. The surge line is very flexible relative to the rest of the structure, and is not considered in the coupled model analysis. The pressurizer is mathematically coupled to the remainder of the RCS by way of the building structure represented in the coupled seismic model. The mass points and corresponding dynamic degrees of freedom are distributed to provide appropriate representations of the dynamic characteristics of the components, as follows:

A. The reactor vessel, with internals, is represented by 4 mass points with a total of 11 dynamic degrees of freedom.

B. Each of the two steam generators is represented by 4 mass points with a total of 10 dynamic degrees of freedom, each of the four reactor coolant pumps is represented by 2 points with a total of 6 dynamic degrees of freedom.

C. The pressurizer is represented by 6 mass points with a total of 13 dynamic degrees of freedom; each branch of cold leg piping is represented by a mass point with 3 dynamic degrees of freedom. I

D. Each branch of hot leg piping is represented by a single mass point with 2 dynamic degrees of freedom. The representation of the reactor vessel internals is formulated in conjunction with the analysis of the reactor vessel internals discussed in Section 3.7.3.14, and is designed to simulate the dynamic characteristics of the models used in that analysis.

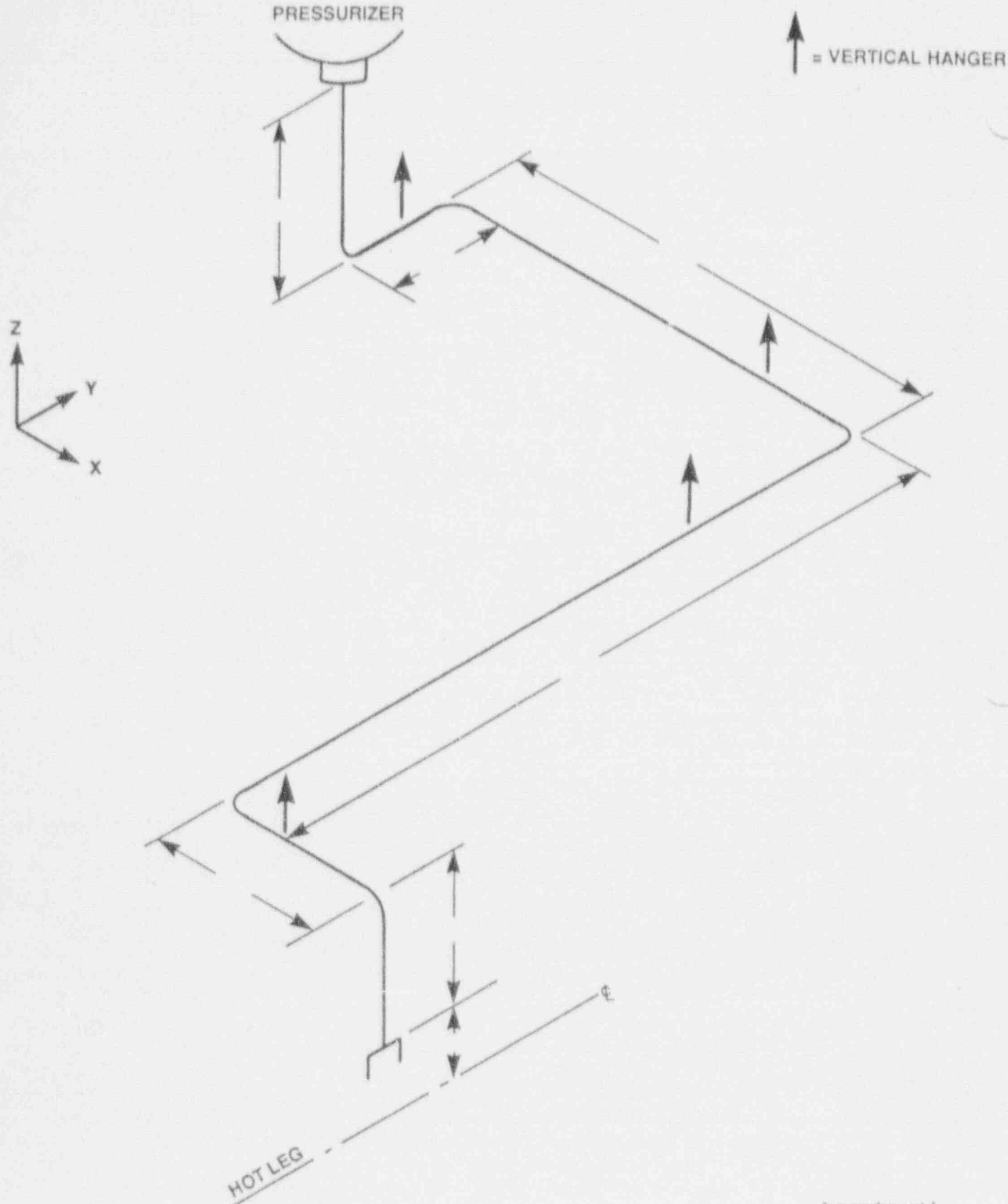
The mathematical model provides a three dimensional representation of the dynamic response of the coupled components to seismic excitations in both the horizontal and vertical directions. The mass is distributed at the selected mass points and corresponding translational degrees of freedom are retained to include rotary inertial effects of the components. The total mass of the entire coupled system is dynamically active in each of the three coordinate directions.

Surge Line

A representative

A lumped parameter, multimass mathematical model is employed in the analysis of the surge line. ~~The~~ model is shown schematically in Figure 3.7-34. The surge line is modeled as a

RAI 210.34



Amendment I
December 21, 1990

SYSTEM 80+™	REPRSENTATIVE 1 SURGE LINE SEISMIC ANALYSIS MODEL	Figure 3.7-34

Question 210.35

CESSAR-DC Section 3.7.2 provides no explanation that a sufficient number of modes was included in the dynamic system analysis to ensure participation of all significant modes.

Provide an explanation to show that a sufficient number of modes was considered in the analysis in accordance with the criteria in SRP, Subsection II.1.a.(iv).

Response 210.35

The number of active modes was chosen to meet the criteria of SRP 3.7.2 II.1.a(IV). This was accomplished by 1) including all modes in the frequency range where amplification could occur, and 2) including additional high frequency modes until at least 90% of the modal mass was active in each of the three excitation directions.

Question 210.36

CESSAR-DC Section 3.7.2.15 describes the analysis procedure for composite modal damping. SRP 3.7.2, Rev. 2, Subsection II.13, specifies that this procedure is limited to 20 percent composite damping. No limitation on composite damping is specified in Subsection 3.7.2.1.5.

Provide justification for not specifying a limitation on composite damping; or modify CESSAR-DC Section 3.7.2.15 in accordance with SRP 3.7.2, Rev. 2, Subsection II.13.

Response 210.36

The structural damping values used to generate the composite damping matrix, as stated in CESSAR-DC-Section 3.7.1.3, are given in Table 3.7-1, the NRC approved damping values for steel, concrete, etc. The largest damping value in the table is 7%. Therefore the largest possible composite damping value is 7%.

Question 210.39

CESSAR-DC Section 3.7.3.4 provides the basis for the selection of frequencies to preclude resonance in equipment and components. This basis is not consistent with the criteria in SRP 3.7.3, Rev. 2, Subsection II.4.

Provide justification for the basis for the selection of frequencies and demonstrate that the criteria in SRP 3.7.3, Rev. 2, Subsection II.4, have been satisfied.

Response 210.39

CESSAR-DC Section 3.7.3.4 provides more stringent requirements than those of SRP 3.7.3 Rev. 2 subsection II.4. The SRP requires the designer to maintain a fixed ratio of less than 1/2 or greater than two between the component frequencies and the support structure dominant frequencies. CESSAR-DC requires that the component frequencies be maintained outside the range of frequencies that would be significantly excited by the forcing frequencies. This requires the designer to evaluate both the frequency of the support structure and the frequency content of the input to the support structure. This may require that the component frequency be more than twice the support structure frequency to meet the requirements of CESSAR-DC for maintaining the component frequency outside the range that is significantly excited.

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TABLE 3.2-1 (Cont'd)

(Sheet 2 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Quality Class
Chemical and Volume Control System			
* Regenerative Heat Exchanger	2	I	1
* Letdown Heat Exchanger	2	I	1
* Seal Injection Heat Exchanger	~~2~~ 3	I	1
* Purification Ion Exchangers	~~2~~ 3	I	1
* Deborating Ion Exchanger	~~2~~ 3	I	1
* Volume Control Tank	~~2~~ 3	I	1
* Chemical Addition Package	NNS	NS	2
* Boric Acid Batching Tank	NNS	NS	2
* Charging Pumps	~~2~~ 3	I	1
* Boric Acid Makeup Pumps	3	I	1
* Reactor Makeup Water Pumps	NNS	NS	2
* Boric Acid Concentrator	NNS	NS	2
* Pre-holdup Ion Exchanger	3	I	1
* Mini-flow Heat Exchanger	~~2~~ 3	I	1
* Boric Acid Condensate Ion Exchanger	NNS	NS	2
* Reactor Drain Pumps	3	I	1
* Holdup Pumps	NNS	NS	2
* Reactor Drain Tank	NNS	NS	2
* Holdup Tank	NNS	NS	2
* Equipment Drain Tank	3	I	1
* Reactor Makeup Water Tank	NNS	NS	2
* Gas Stripper	3	I	1
* Purification Filters	~~2~~ 3	I	1
* Reactor Drain Filter	3	I	1
* Seal Injection Filters	~~2~~ 3	I	1
* Reactor Makeup Filter	NNS	NS	2
* Boric Acid Filter	3	I	1
* Letdown Strainer	~~2~~ 3	I	1
* Pre-holdup Strainer	3	I	1
* Boric Acid Condensate IX Strainer	NNS	NS	2
* Ion Exchanger Drain Header Strainer	NNS	NS	2
* Boric Acid Batching Strainer	NNS	NS	2
* Chemical Addition Strainer	NNS	NS	2
* Boric Acid Storage Tank	~~2~~ 3	I	1

I

TABLE 3.2-2 (Cont'd)

(Sheet 3 of 15)

SAFETY CLASS 1, 2 & 3 VALVES

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
Chemical and Volume Control System (CVCS) (1)				
CH-101	Letdown check to VCT	~~2~~ 3	I	1
CH-103	VCT pressure indicator	~~2~~ 3	I	1
CH-104	VCT vent isolation	~~2~~ 3	I	1
CH-110P, 110Q	Letdown control valve	~~2~~ 3	I	1
CH-112	VCT gas supply check	~~2~~ 3	I	1
CH-113, 114	VCT level indicator	~~2~~ 3	I	1
CH-115	VCT to EDT relief	~~2~~ 3	I	1
CH-116	VCT to RDH isolation	~~2~~ 3	I	1
CH-117	VCT local sample	~~2~~ 3	I	1
CH-118	VCT discharge check	~~2~~ 3	I	1
CH-124	BAST supply isolation	~~2~~ 3	I	1
CH-126	BABT to BAST isolation	3	I	1
CH-127	BAC line to BAST check	3	I	1
CH-128, 129	BAST level indicator	~~2~~ 3	I	1
CH-130	BAMP recirc isolation	3	I	1
CH-131	Boric acid filter D/P	3	I	1
CH-134	BAMP to DRDH isolation	3	I	1
CH-135	BAST level indicator	~~2~~ 3	I	1
CH-139	Gas stripper to VCT	~~2~~ 3	I	1
CH-143	BAMP suction isolation	3	I	1
CH-144	BAST to PCPS isolation	3	I	1
CH-145	BAMP suction isolation	3	I	1
CH-146, 147	BAMP discharge pressure	3	I	1
CH-150	VCT level indication	~~2~~ 3	I	1
CH-152, 153	BAMP discharge isolation	3	I	1
CH-154, 155	BAMP discharge check	3	I	1
CH-156	BAST level indicator	~~2~~ 3	I	1
CH-160	VCT level indication	~~2~~ 3	I	1
CH-161	Boric acid filter isolation	3	I	1
CH-164	Boric acid filter bypass	3	I	1
CH-165	Boric acid filter D/P	3	I	1
CH-166	Boric acid filter	3	I	1
CH-172	F-210Y isolation	3	I	1
CH-174	Boric acid makeup cross-connect	3	I	1
CH-176	BAMP local sample	3	I	1
CH-177	Boric acid to charging pump suction check	~~2~~ 3	I	1

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TABLE 3.2-2 (Cont'd)

(Sheet 4 of 15)

SAFETY CLASS 1, 2 & 3 VALVES

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
CH-179	RMW line to charging pump suction check	~~2~~ 3	I	1
CH-184	RMW line to VCT check	3	I	1
CH-188	RMW to VCT check	~~2~~ 3	I	1
CH-190, 191	BAST gravity feed check	~~2~~ 3	I	1
CH-192	BAMP to BAST recirc	3	I	1
CH-198	RCP controlled bleedoff	~~2~~ 3	I	1
CH-199	RCP controlled bleedoff to RDT relief	2	I	1
CH-205	Auxiliary spray control	~~1~~ 2	I	1
CH-208	Charging backpressure	~~1~~ 2	I	1
CH-210Y	Boric acid flow control	3	I	1
CH-231	Seal injection isolation	~~2~~ 3	I	1
CH-241, 242, 243, 244	Seal injection flow control	2	I	1
CH-255	Seal injection isolation	2	I	1
CH-300	RCP bleedoff pressure	2	I	1
CH-301	Letdown orifice bypass	2	I	1
CH-304	SCS Purification check	2	I	1
CH-307	SCS Purification isolation	2	I	1
CH-308	Letdown heat exchanger vent	3	I	1
CH-330	BAMP line to HT isolation	3	I	1
CH-344	Letdown flow indicator	~~2~~ 3	I	1
CH-346	Letdown pressure control isolation	~~2~~ 3	I	1
CH-347, 348, 349, 350	Letdown control valve iso.	~~2~~ 3	I	1
CH-351	Letdown flow indicator	~~2~~ 3	I	1
CH-353	Sampling system isolation	~~2~~ 3	I	1
CH-354	Letdown to EDT relief	~~2~~ 3	I	1
CH-355	Letdown filter bypass	~~2~~ 3	I	1
CH-356, 357	Letdown filter D/P iso.	~~2~~ 3	I	1
CH 358	Letdown filter isolation	~~2~~ 3	I	1
CH-359	Letdown filter vent	~~2~~ 3	I	1
CH-360	Letdown filter isolation	~~2~~ 3	I	1
CH-361	Letdown to DRDH isolation	~~2~~ 3	I	1
CH-366	Letdown filter vent	~~2~~ 3	I	1
CH-369	Ion exchanger isolation	~~2~~ 3	I	1
CH-370	IX inlet check	~~2~~ 3	I	1
CH-371	IX vent to GWMS	~~2~~ 3	I	1
CH-372	IX resin fill isolation	~~2~~ 3	I	1
CH-373	Letdown filter isolation	~~2~~ 3	I	1

I

Question 210.42

CESSAR-DC Section 3.7.3.12.1 provides seismic design criteria for buried piping. These criteria do not include relative deformation imposed by seismic waves through the surrounding soil as required by SRP 3.7.3, Rev. 2, Subsection II.12.a.(i), and other items defined in SRP 3.9.2, Rev. 2 Subsection II.2.j.

Include the effects of relative deformation imposed by seismic waves through the soil in the seismic design criteria for buried piping and other items defined in SRP 3.9.2, Rev. 2, Subsection II.2.j.

Response 210.42

The following will be included in a future amendment to CESSAR-DC.

"Buried piping is designed for seismic effects according to the criteria established in this section.

In general, buried piping is excluded from areas of direct fault displacement and unstable soil conditions, such as liquefaction.

Buried piping is designed to sustain soil movements during earthquake ground motions. The structural integrity of the piping is evaluated by accounting for two fundamental effects of earthquake ground motion:

a. Strains and associated stresses induced in a long pipe by the free-field vibration resulting from motions of the surrounding soil mass.

b. Seismically induced differential movements of structures which the pipe enters or connects.

The maximum strains associated with the free-field vibration of the soil are computed based on the guidelines of References 10 and 11. Friction between the pipe and the surrounding soil may be considered using conservative estimates of the associated frictional forces.

The effects of seismically induced differential displacements of the ends of buried piping (due to differential movements of buildings) are considered using:

a. Equivalent static analysis.

b. The assumption of out-of-phase differential movements at the entry points.

c. Principles of beams on elastic foundations.

d. Computation of stresses by the SRSS method.

Again, the guidelines of References 10 and 11 are followed."

References:

10 American Society of Civil Engineers, "Seismic Analysis of Safety Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety Related Nuclear Structures", Publication No. ASCE 4-86, September 1986.

11 American Society of Civil Engineers, "Structural Analysis and Design of Nuclear Plant Facilities" ASCE Manual on Engineering Practice No. 58, 1980.

In addition, the following

3.7.3.12 Piping Outside Containment Structure

3.7.3.12.1 Buried Piping

Insert

Seismic design criteria for buried piping are as follows:

A. Intake structure is designed such that the differential movement between this structure and the earth is negligible and the seismic response spectrum utilized is the ground surface response.

B. Conformance to allowable structural and piping stresses after the line penetrates the Auxiliary Building is assured by the use of expansion joints.

An alternate design method is to use flexible seals as the lines pass through pipe sleeves in the structure.

Delete

~~Important factors considered are the flexibility, supports, and restraints of lines which are virtually anchored in earth but which penetrate a structure. A flexibility analysis of these lines is performed to demonstrate that the piping and structures are not overstressed under the additive differential movement of the earth and structure.~~

3.7.3.12.2 Above Ground Piping

Seismic design criteria and methods of accounting for the effects of differential movement of buildings on piping and penetrations are described in Sections 3.7.2.1.2 and 3.7.2.7.

3.7.3.13 Interaction of Other Piping with Category I Piping

The protection of Category I piping from possible adverse effects of other piping during an earthquake is accomplished by several methods. Specifically, these methods are:

A. Category I lines are physically separated from other lines to the extent possible so that failure of a line has no effect on Category I lines.

B. All Category I boundary valves are designed to meet seismic criteria. A valve always serves as a pressure boundary and constitutes the seismic to non-seismic boundary. If failure in the non-seismic portion of the system could cause loss of function of the safety system, then an appropriate automatic or remote manual operator would be used if the valve is open during normal reactor operation.

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Insert

Buried piping ~~are~~ is designed for seismic effects according to the criteria established in this section.

In general, buried piping ~~are~~ is excluded from areas of direct fault displacement and unstable soil conditions, such as liquefaction.

Buried piping ~~are~~ is designed to sustain soil movements during earthquake ground motions. The structural integrity of the piping is evaluated by accounting for two fundamental effects of earthquake ground motion:

a. Strains and associated stresses induced in a long pipe by the free-field vibration resulting from motions of the surrounding soil mass.

b. Seismically induced differential movements of structures which the pipe enters or connects.

The maximum strains associated with the free-field vibration of the soil are computed based on the guidelines of References ~~210.42.1~~ (10) and ~~210.42.2~~ (11). Friction between the pipe and the surrounding soil may be considered using conservative estimates of the associated frictional forces.

The effects of seismically induced differential displacements of the ends of buried piping (due to differential movements of buildings) are considered using:

a. Equivalent static analysis.

b. The assumption of out-of-phase differential movements at the entry points.

c. Principles of beams on elastic foundations.

d. Computation of stresses by the SRSS method.

Again, the guidelines of References ~~210.42.1~~ 10 and ~~210.42.2~~ 11 are followed."

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REFERENCES FOR SECTION 3.7

1. Tsai, N.C., "Spectrum-Compatible Motions for Design Purposes", Journal of Engineering Mechanics Division, ASCE, Vol. 98 EM2, April 1972.

2. Fritz, R.J., "The Effect of Liquids on the Dynamic Motions of Immersed Solids", Journal of Engineering for Industry, Paper No. 71-VIB-100.

3. McDonald, C.K., "Seismic Analysis of Vertical PVMPs Enclosed in Liquid Filled Containers", ASME Paper No. 75-PVP-56.

4. Pahl, P.J., "Modal Response on Containment Structures", Seismic Design for Nuclear Power Plants, MIT Press, Cambridge, Mass.

5. Forsberg, K., "Axisymmetrical and Beam-Type Vibrations of These Cylindrical Shells", AJAA Journal, Volume 7, February 1969. I

6. Lysmer, J., Tabatabaie, M., Tajirian, F., Vahdani, S., Ostadan, F., "SASSI - A System for the Analysis of Soil-Structure Interaction", Report No. UCB/GT/81-02, Univ. of California, Berkeley, April, 1981.

7. Idriss, I.M., "Earthquake Ground Motions - Selection of Control Motion and Development of Generic Soil Sites".

8. ABB Impell Report No. 01-8503-1784, "Seismic Analysis of the Reactor Building of the System 80+ Certified Design".

9. Impell Corporation, Calculation No. ALWR-2, "SSI Analysis of Case B3.5 with Common Basemat", Job No. 8503-003-1355, Revision 6.

Add:

~~210.42.1~~ 10. American Society of Civil Engineers, "Seismic Analysis of Safety Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety Related Nuclear Structures", Publication No. ASCE 4-86, September ~~(9)~~1986.

~~210.42.2~~ 11. American Society of Civil Engineers, "Structural Analysis and Design of Nuclear Plant Facilities", ASCE Manual on Engineering Practice No. 58, 1980.

Question 210.43

CESSAR-DC Section 3.7.3.12.2 references criteria and methods in Sections 3.7.2.1.2 and 3.7.2.7 for differential movement of buildings on above ground piping and penetrations. It is not clear to the staff how these sections address differential movement of buildings. Provide a discussion that describes the seismic design criteria and methods which apply to this issue.

Response 210.43

When the earthquake ground motions are applied, the Nuclear Island structural models undergo deformations in all three directions (two horizontal and the vertical). In general, these deformations differ between adjacent structures. This causes differential movements (and associated stresses) to the above ground piping that are connected to those structures.

To include the effects of differential displacements between adjacent structures in the piping design and analysis, maximum relative displacements (with respect to the foundation basemat) at every major elevation of each Nuclear Island structure are computed in the three principal directions of motion. The maximum relative displacements, defined as Seismic Anchor Movements (SAM), are computed for every generic soil case of analysis.

The appropriate SAM (depending on elevation) are applied to each piping system that runs through adjacent structures. For conservatism, SAM are applied out-of-phase, thus, conservatively neglecting any phasing that might occur in the response of adjacent structures. Piping stresses resulting from SAM are considered in conjunction with those resulting from inertia response.

<u>Question 210.45</u>

CESSAR-DC Section 3.7.3 does not address seismic analysis of above-ground tanks as required by SRP 3.7.3, Rev. 2.

Provide seismic analysis and design criteria for above-ground tanks.

<u>Response 210.45</u>

SRP Section 3.7.3, Rev. 2 requires that seismic Category I above ground tanks which cannot be proven to respond rigidly, allowing use of zero period acceleration information in the seismic analysis, are to be analyzed as separate structures requiring different response spectra.

The major tanks included in the System 80+ design are the IRWST and the EFW tanks. These, however, are integral components of the building structure and are modeled seismically with the interior and annex structures. All other tanks would be vendor supplied via procurement specifications. These specifications will address seismic design considerations assuring that the tanks are qualified using the correct seismic analyses and design criteria.

Question 210.46:

CESSAR-DC Section 3.9.1.1 identifies the transients to be used in the design and fatigue analysis of ASME Code Class 1 components for a 60-year useful plant life. This 60-year life raises questions relative to the margins available in the current ASME fatigue design curves. Based on limited available data, the staff is of the opinion that these margins may not be sufficient to account for variations in the original fatigue test data due to various environmental effects.

Provide a commitment to consider such effects in the designs of applicable ASME Class 1 systems, components and equipment.

Moreover, SRP 3.9.1, Rev. 2, subsection III.1, states that the number of events estimated for each transient and the method used to determine this number is to be compared to the same information on similar and previously licensed applications. Comparing CESSAR(-F) System 80 with CESSAR-DC shows that the useful life of the plants differ. However, both plants use the same list of transients and the same number of events (Table 3.9-1) even though the useful design life differ by 20 years. Justify the use of the same number of events for the System 80+ design.

Response 210.46:

The designs of ASME Class 1 systems, components and equipment for System 80+ will consider the potential influence of environmental effects on the fatigue life of materials over the 60 year design life.

This issue is currently under consideration by a special Steering Committee for Cyclic Life and Environmental Effects in Nuclear Applications of the Pressure Vessel Research Council (PVRC) per the requests of the ASME Boiler & Pressure Vessel (B&PV) Code Committee through the Board on Nuclear Codes & Standards (BNCS). The charter of the PVRC Steering Committee is to provide guidance and direction related to determining the effects of service environment on the cyclic life properties of applicable materials in light-water reactor applications and evaluating application methodologies that include these effects. Recommendations from these activities will be provided for consideration to the ASME Code Committee and to other concerned U.S. and international organizations.

Any revisions to the presently existing fatigue curves or modifications to current fatigue design and evaluation methodologies that are adopted as part of future editions of the ASME B&PV Code will be applied to the design of System 80+ components subject to those editions of the Code.

CESSAR-DC Table 3.9-1 is being revised to include the events and frequency of occurrence expected during a 60 year design life. The latest industry databases are being utilized to assemble this information. Additionally, CESSAR-DC will be modified such that all references to design transients (e.g., sections 3.9.3.1.3.3 (A, B), 5.2.2.4.1 (A-F), 5.4.2.1 (A-H), 5.4.10.1 (B)) will refer to Table 3.9-1 rather than list transients in individual sections. This task will be completed by July 1992.

Question 210.47

SRP 3.9.1, Rev. 2, Subsection II.2, defines the information to be provided to demonstrate the applicability and validity of computer programs to be used in dynamic and static analyses to demonstrate the structural and functional integrity of Seismic Category I, and Code and non-Code items. The information provided in CESSAR-DC, Section 3.9.1.2 is not totally in accordance with the information required by SRP 3.9.1, Rev. 2.

Provide the computer program information required by SRP 3.9.1, Rev. 2. The fact that a program is in the public domain is insufficient for satisfaction of the guidelines in SRP 3.9.1, Rev. 2.

Response 210.47

As stated in CESSAR DC, Section 3.9.1.2, extensive verification of the versions of public domain computer codes used by C-E has been performed to supplement any existing public documentation. Each code (e.g., MDC STRUDL, C-E MARC) has been quality assured in accordance with the QA procedures in force at the time of their documented verification. Documentation for all of the codes described in CESSAR DC, Section 3.9.1.2 is available for NRC audit upon request. The QA calculations for each code contain the author's name, a computer code certificate containing necessary information such as the version, installation date and facility on which the code executes, and the solutions to appropriate test problems.

In addition, a description of the SASSI program will be included in a future amendment of CESSAR-DC.

3.9.1.2.1.22 CEFLASH-4A

A code used to calculate transient conditions resulting from a flow line rupture in a water/steam flow system. The program is used to calculate steam generator internal loadings following a postulated main steam line break.

This program is used in a steam line break accident structural analysis. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.23 CRIBE E

A one-dimensional, two-phase thermal hydraulic code, utilizing a momentum integral model of the secondary flow. This code was used to establish the recirculation ratio and fluid mass inventories as a function of power level. The code is in the public domain and has had sufficient use to justify its applicability and validity. I

This program is used for determining steam generator performance. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

Insert → 3.9.1.2.1.24 SASSI

3.9.1.2.2 Code Class CS Internals, Fuel and CEDMs

The following computer programs are used in the static and dynamic analyses of reactor internals, fuel, and CEDMs. E

3.9.1.2.2.1 MRI/STARDYNE

The MRI/STARDYNE program uses the finite element method for the static and dynamic analysis of two- and three-dimensional solid structures subjected to any arbitrary static or dynamic loading or base acceleration. In addition, initial displacements and velocities may be considered. The physical structure to be analyzed is modeled with finite elements that are interconnected by nodes. Each element is constrained to deform in accordance with an assumed displacement field that is required to satisfy continuity across element interfaces. The displacement shapes are evaluated at nodal points. The equations relating the nodal point displacements and their associated forces are called the element stiffness relations and are a function of the element geometry and its mechanical properties. The stiffness relations for an element are developed on the basis of the theorem of minimum potential energy. Masses and external forces are assigned to the nodes. The general solution procedure of the program is to formulate the total following equations:

Insert

3.9.1.2.1.24 SASSI

The SASSI program is used in soil-structure interaction analyses and it is based on the Flexible Volume Substructuring Method . This method is a general substructuring technique, which uses the finite element method and solves the equations of motion in the frequency domain using the method of complex response.

The SASSI substructuring scheme is an improvement over other substructuring techniques because it simplifies the required solution steps. Nevertheless, it provides rigorous analytical solutions in each step of the SSI problem. In the Flexible Volume Method, the complete soil-structure system is divided into two substructures: the "foundation" and the "structure". The mass and stiffness of the "structure" is reduced by the corresponding properties of the volume of excavated soil. The mass and stiffness of the excavated soil are retained within the "foundation" model. The impedance problem is solved using the "foundation" model, and consists of a series of axisymmetric solutions of a layered site to applied point loads. In general, using SASSI, there is no need to solve the scattering problem since the "foundation" model does not have the intrusion corresponding to the embedded portion of the structure. However, in the System 80+ analysis, the SASSI standard analysis methodology is modified, as discussed in Appendix 3.7B, and the solution of the SSI problem is reduced to three steps:

- Solution of the site response problem to determine the free-field motions within the embedded part of the structure.

- Solution of the impedance and scattering problem.

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- Solution of the structural problem. This involves forming the complex stiffness matrices and load vector and solving the equations of motion for the final displacements.

SASSI Program Capabilities

The SASSI program is the most versatile tool currently available for SSI analysis. It contains basically no limitations of analytical nature, outside of the fact that it is limited to linear-type analysis. The paragraphs below briefly describe the program capabilities.

The analytical model of the soil site consists of semi-infinite elastic or viscoelastic horizontal layers on a rigid base or on a semi-infinite elastic or viscoelastic half-space. To simulate the half-space at the lower boundary, the SASSI program generates soil layers of variable depth which depend upon the analysis solution frequency. Thus, SASSI accurately models energy dissipation through the bottom as well as lateral boundaries.

The seismic environment may consist of an arbitrary superposition of inclined body waves and surface waves. The earthquake excitation is defined by a time history of acceleration, called the control motion. The control motion is assigned to one of the three global directions at a specified control point which lies on the surface or on a soil layer interface within the soil profile. Transient input time histories, such as earthquake records or impact loads, are handled by the Fast Fourier Transform technique. In addition to seismic loads, it is possible to introduce external forces or moments, such as impact loads, wave forces, or loads from rotating machinery acting directly on the structure. This feature is particularly applicable for foundation design for large pieces of equipment such as turbines, diesel generators, etc.

The structures are idealized by standard two or three-dimensional finite elements connected at their nodal points. Material damping is introduced by the use of complex moduli, which leads to effective damping ratios which are frequency independent and which can vary from element to element.

Primary nonlinear effects in the free field and secondary nonlinear effects in a limited region near the structure can be considered by the "equivalent linear method".

The SASSI program can handle embedded structures with flexible foundations, structure-to-structure interaction and the effects of torsional ground motions.

The library of SASSI elements consists of the following:

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1. Three-dimensional solid element (eight-node brick) with three translational degrees of freedom per node.

2. Three-dimensional beam element with three translational and three rotational degrees of freedom per node.

3. Four-node quadrilateral plate/shell element with three translational and three rotational degrees of freedom per node.

4. Two-dimensional, four-node, plane-strain finite element with two translational degrees of freedom per node.

5. Three-dimensional spring element with three translational and three rotational degrees of freedom per node.

6. One-dimensional plane Love wave element with one out-of-plane translational degree of freedom per node.

7. Three-dimensional stiffness element with three translational and three rotational degrees of freedom per node.

8. Three-dimensional mass matrix element.

SASSI Program Structure

The computer program SASSI is structured in a modular form. The code has been segmented in nine independent subprograms which are executed sequentially. Each of these subprograms performs one of the tasks which are required in the sequence of the analysis of soil-structure interaction problems, i.e., the site response analysis, the impedance analysis, the formation of the load vector and, finally, the computation of the transfer functions and the response time histories.

The eleven SASSI modules are:

1. SITE
2. POINT
3. HOUSE
4. MOTOR
5. ANALYS
6. COMBIN
7. MOTION
8. STRESS

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9. RANDOM
10. AXSYM
11. RIMP

The version of the SASSI program used in the soil-structure interaction (SSI) analysis of the System 80+ is version 2.0, dated June 1985. The following SASSI modules were part of the version 2.0 program:

- SITE
- COMBIN
- MOTION
- STRESS

To compute foundation impedances and scattering with an axisymmetric approach, SASSI was modified and enhanced. Thus, two of the version 2.0 modules, HOUSE and ANALYS, were modified for the System 80+ project as version 3.0, and a new module, AXSYM, was developed as version 3.0.

The SASSI program is extensively verified and validated and documented using three different methods of verification and correlation:

- Correlation to results of problems with closed form solutions, such as site response and response of simplified structural systems.

- Correlation to solutions of other well known SSI computer codes in the industry such as CLASSI and FLUSH.

- Correlation to experimental results, such as the Lotung Large Scale Experiment sponsored by the Electric Power Research Institute/Nuclear Regulatory Commission/Taiwan Power Company, and others.

Question 210.48

CESSAR-DC Section 3.9.1.4.1 states that inelastic methods of analysis are used to permit significant local inelastic responses in Seismic Category I reactor coolant system (RCS) items.

Identify where inelastic analysis methods have been utilized for Seismic Category I RCS items and provide justification for the methods of analysis in accordance with SRP 3.9.1, Rev. 2 Subsection III.4.

Response 210.48

The reference section of CESSAR-DC allows for inelastic methods to be used where "desirable and appropriate to permit significant local inelastic response." These inelastic analysis methods, however, have not been used for Seismic Category I RCS items in the System 80+ design.

Question 210.50

CESSAR-DC Section 3.9.1.4.2 states that in exceptions to linear elastic models for evaluating faulted conditions, the maximum allowable strain limits from accepted standards will be satisfied.

Identify and justify the use of the accepted standards and the strain limits to be used.

Response 210.50

CESSAR-DC Section 3.9.1.4.2 will be revised to include the following statement:

Pipe rupture restraint energy absorbing members are an exception to the use of linear elastic models. The methods for the dynamic analysis of pipe whip are given in CESSAR-DC Section 3.6.2.2.2.2. For allowable stresses and design criteria, reference CESSAR-DC Sections 3.6.2.3.2.4 and 3.6.2.3.2.5 respectively.

3.9.1.4.1.1 Reactor Internals and CEDMs

See Sections 3.7.3.14 and 3.9.2.5.

3.9.1.4.1.2 Non-Code Items

The components not covered by the ASME Code but which are related to plant safety include:

A. Internal Structures (Class IS).

B. Fuel.

C. Control element drive mechanisms (CEDMs).

D. Control element assemblies (CEAs).

Each of these components is designed in accordance with specific criteria to ensure their operability as it relates to safety. The fuel assembly and control element assembly design is discussed in Section 4.2. The non-code components of the control element drive mechanisms (CEDMs) are proven by testing as described in Section 3.9.4.4. E

3.9.1.4.2 Seismic Category I Non-NSSS Items

The analytical method for evaluating the faulted condition uses a linear elastic model as described in Section 3.7.3. The ASME Section ~~III allowable stress limits will be met for faulted loads,~~ including the safe shutdown earthquake and system transient loads described in Section 3.9.1. ~~For any exceptions to the above, such as the pipe break analysis described in Section 3.6.2, maximum allowable strain limits from accepted standards will be satisfied.~~ INSERT

3.9.2 DYNAMIC SYSTEM ANALYSIS AND TESTING

3.9.2.1 Piping Vibrations, Thermal Expansion, and Dynamic Effects

Safety-related piping systems were designed in accordance with the ASME B&PV Code, Section III. The preoperational test program for the Class 1, 2 and 3 piping systems will simulate actual operating modes to demonstrate that the appurtenances comprising these systems will meet functional design requirements and that piping vibrations are within acceptable levels.

INSERT RAI 210.50

Pipe rupture restraint energy absorbing members are an exception to the use of linear elastic models. The methods for the dynamic analysis of pipe whip are given in CESSAR-DC Section 3.6.2.2.2.2. For allowable stresses and design criteria, reference CESSAR-DC Sections 3.6.2.3.2.4 and 3.6.2.3.2.5 respectively.

Question 210.51

SRP 3.9.2, Rev. 2, Subsection II.I provides criteria for piping vibration, thermal expansion, and dynamic effect testing to be performed during startup testing.

CESSAR-DC Section 3.9.2.1 describes the preoperational test program for the ASME Code, Section III, Class 1, 2, and 3 piping systems. The test program is not in total agreement with the guidelines in SRP 3.9.2, Rev. 2 Subsection II.I.

Modify the descriptions of the test programs in accordance with SRP 3.9.2, Rev. 2 Subsection II.I. In addition, the staff's current position requires a commitment to conduct testing in accordance with ANSI/ASME OM3-1982 and draft OM7 standards. Piping systems to be included in the test programs should also be identified.

Response 210.51

CESSAR-DC, Section 3.9.2.1 contains testing methods, acceptance criteria, and corrective actions for vibration, thermal expansion, and dynamic effects startup testing. The areas that do not agree with SRP 3.9.2, Rev. 2, Subsection II.1 concern level-of-detail. Specific monitoring locations, snubber travel from hot to cold position, etc. cannot be determined until completion of detailed pipe routing and piping support design, which will depend on vendor-supplied information.

CESSAR-DC will be revised to indicate that vibration and thermal expansion startup testing will meet the intent of ASME OM-S/G, Part 3 standard and Part 7 guide. This revision is enclosed for NRC review.

The list of systems requiring vibration and thermal expansion startup testing cannot be finalized until final pipe routing is performed and analysis results are obtained. However, the systems that typically require this testing are listed below:

Piping Systems Included in Vibration and Thermal Expansion Startup Testing

Reactor Coolant System
Safety Depressurization System
Safety Injection System
Shutdown Cooling System
Containment Spray System
Chemical and Volume Control System
Pool Cooling and Purification System
Component Cooling Water System
Station Service Water System

Response 210.51 Cont.

Diesel Generator Engine Fuel Oil System
Diesel Generator Engine Cooling Water System
Diesel Generator Engine Lube Oil System
Essential Chilled Water System
Emergency Feedwater System
Main and Startup Feedwater System
Main Steam System
Steam Generator Blowdown System
Turbine Bypass System
Liquid Waste Management System
Condenser Circulating Water System

3.9.1.4.1.1 Reactor Internals and CEDMs

See Sections 3.7.3.14 and 3.9.2.5.

3.9.1.4.1.2 Non-Code Items

The components not covered by the ASME Code but which are related to plant safety include:

A. Internal Structures (Class IS).

B. Fuel.

C. Control element drive mechanisms (CEDMs).

D. Control element assemblies (CEAs).

Each of these components is designed in accordance with specific criteria to ensure their operability as it relates to safety. The fuel assembly and control element assembly design is discussed in Section 4.2. The non-code components of the control element drive mechanisms (CEDMs) are proven by testing as described in Section 3.9.4.4. E

3.9.1.4.2 Seismic Category I Non-NSSS Items

The analytical method for evaluating the faulted condition uses a linear elastic model as described in Section 3.7.3. The ASME Section III allowable stress limits will be met for faulted loads, including the safe shutdown earthquake and system transient loads described in Section 3.9.1. For any exceptions to the above, such as the pipe break analysis described in Section 3.6.2, maximum allowable strain limits from accepted standards will be satisfied.

3.9.2 DYNAMIC SYSTEM ANALYSIS AND TESTING

3.9.2.1 <u>Piping Vibrations, Thermal Expansion, and Dynamic Effects</u>

Safety-related piping systems were designed in accordance with the ASME B&PV Code, Section III. The preoperational test program for the Class 1, 2 and 3 piping systems will simulate actual operating modes to demonstrate that the appurtenances comprising these systems will meet functional design requirements and that piping vibrations are within acceptable levels. The testing program will meet the intent of ASME OM-S/G, Part 3 standard and Part 7 guide.

Question 210.52

CESSAR-DC Section 3.9.2.3 describes the dynamic system analysis methods for the reactor vessel core support and internal structures.

Analytical methods for determining the responses of the reactor internal components due to both predominantly deterministic and random force inputs are described. For the latter type of inputs, CESSAR-DC Section 3.9.2.3.5.2 indicates that only RMS displacements, loads, and stresses are calculated.

In view of: 1) the 60-year design life stated in CESSAR-DC Section 3.9.1.1; 2) staff concerns relating to possible detrimental environmental effects not currently reflected in current ASME Code design fatigue curves; and 3) the current history of flow induced vibration failures in PWR reactor internal structure support system; CESSAR-DC Section 3.9.2.3 should contain a commitment to include methods for considering peak responses to random type force inputs.

Response 210.52

Three times the RMS stresses, displacements and loads found during random response analysis are used in the reactor internals design stress analysis which combines random results with those of other analyses (i.e.: deterministic, thermal, etc) and compares the total stresses with appropriate ASME code, criteria such as: primary membrane and membrane + bending, primary plus secondary, and fatigue. The use of 3 x RMS values is sufficient to account for peak responses due to random type force inputs.

Question 210.53

CESSAR-DC Section 3.9.2.4 designates Palo Verde Unit 1 as the proto-type plant for the CESSAR-DC System 80+ Standard Design Plant.

Provide a comparison of the Palo Verde, Unit 1 and the CESSAR-DC designs in accordance with SRP 3.9.2, Rev. 2, criteria in support of the designation of Palo Verde Unit 1 as the prototype for the System 80+ design.

Response 210.53

The System 80+ reactor internals as described in CESSAR-DC are essentially the same as Palo Verde Unit 1 as described in the Palo Verde Unit 1 FSAR. The dynamic characteristics will therefore be nearly identical. Since operating conditions are also essentially the same, flow induced vibration responses will be the same as experienced at Palo Verde. Therefore, the requirements of Reg. Guide 1.20 are met and Palo Verde Unit 1 is the valid prototype for the System 80+ design.

Question 210.57

CESSAR-DC Section 3.9.3 states and a number of loading combination tables indicate that dynamic loads are combined by the SRSS method.

Modify Section 3.9.3 and the loading combination tables to indicate that dynamic loads will be combined by the SRSS method in accordance with the guidelines of NUREG-0484, Rev. 1, 1980.

Response 210.57

Section 3.9.3 will be modified to indicate that dynamic loads will be combined by the SRSS method in accordance with the guidelines of NUREG-0484, Rev. 1, 1980. This revision will be included in a future amendment of CESSAR-DC.

in accordance with the guidelines of NUREG-0484, Rev. 1, 1980

(faulted) system condition (postulated pipe rupture for branch line breaks not eliminated by leak before break analysis). The SSE and pipe rupture loadings are combined by the SRSS method or ~~a~~ more conservative method. E

The specific design transients specified for design are discussed in Section 3.9.1.1.

ASME B&PV Code Class 1, 2 and 3 piping and components of fluid systems are designed and constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code. Hydrostatic testing is performed per Section III.

Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems are presented in the sections which describe the systems.

Stress analysis was used to determine structural adequacy of pressure components under the operating conditions of normal, upset, emergency or faulted, as applicable. E

Significant discontinuities were considered such as nozzles, flanges, etc. In addition to the design calculation required by the ASME B&PV Section III code, stress analysis was performed by methods outlined in the code appendices or by other methods by reference to analogous codes or other published literature.

3.9.3.1.1 ASME Code Class 1 Components and Supports

Design transients for ASME Code Class 1 components, supports and piping are discussed in Section 3.9.1.1. Loading combinations for ASME Code Class 1 components are described in Table 3.9-2. Stress limits for ASME Code Class 1 components, supports and piping are described in Table 3.9-3. The operating pressures of Code Class 1 active valves are limited to the pressures taken from the applicable primary pressure class pressure-temperature rating of the ASME Code, Section III, for the maximum temperature for the applicable condition. E

3.9.3.1.2 Core Support Structures (Class CS) and Internal Structures (Class IS) E

Design transients for reactor internals structures are discussed in Section 3.9.1.1. Loading combinations and stress limits are presented in Section 3.9.5.

Question 210.58:

Section 3.9.1.1 implies that the design life of the CESSAR System 80+ plant is 60 years. In Section 3.9.3.1.3, "ASME Code Class 2 and 3 Components and Supports," there is no indication of how this extended life will be considered. For all ASME Class 2 and 3 components, equipment, and supports that are designed for a 60 year life and which are subjected to loadings which could result in thermal or dynamic fatigue, provide a commitment to perform fatigue analyses similar to the requirements for Class 1 components in ASME III, Subsection NB. In addition to the transients discussed in Section 3.9.1.1 of CESSAR-DC, the loadings for these analyses should account for operating vibration loads which may have been observed during piping pre-operational tests and for the effects of mixing hot and cold fluids.

Response 210.58:

ASME Code Class 2 and 3 components and supports will be analyzed for fatigue as required by the ASME Code Section III, Subsections NC and ND. The analysis will include all estimated transient event cycles and vibration over the design life of the plant. This statement will be added to Section 3.9.3.1.3 in a future amendment.

Piping operational vibration loads are considered in the design of ASME Code Class 2 and 3 components. Note that section 3.9.3.1.3 indicates systems specific conditions in addition to the design transients of Table 3.9-1 are reviewed to determine the appropriate parameters to be used in the design of Code Class 2 and 3 components. In addition, preoperational testing confirms that these piping systems, restraints, components and supports have been adequately designed to withstand flow-induced dynamic loadings under the steady state and operational transient conditions anticipated during service.

Transient analyses consider mixing of hot and cold fluids, or the more conservative assumption of step changes of fluid temperature.

RAI 210.58

3.9.3.1.3 ASME Code Class 2 and 3 Components and Supports

Loading combinations applicable to Code Class 2 and 3 components and supports are described in Table 3.9-2. System operating conditions due to the design transients defined in Table 3.9-1, as well as any other auxiliary system specific conditions, are reviewed to determine the appropriate operating parameters to be used in the design of Code Class 2 and 3 components.

Insert A →

The design stress limits for each of the component's loading conditions are presented in Tables 3.9-5 through 3.9-9. Inelastic methods, as permitted by ASME Section III for Class 1 components, were not used for these components. E

3.9.3.1.3.1 Tanks, Heat Exchangers, and Filters

Pressure vessels supplied for the auxiliary systems are:

Shutdown Cooling Heat Exchanger
~~Safety Injection Tanks~~ E
Containment Spray Heat Exchanger
Containment Spray Mini-Flow Heat Exchanger
Shutdown Cooling Mini-Flow Heat Exchanger
Component Cooling Water System Heat Exchangers
Component Cooling Water System Surge Tanks
Essential Chilled Water Compression Tanks
Essential Chilled Water Refrigeration Units
Diesel Generator Fuel Oil Storage Tank
Diesel Generator Fuel Oil Day Tank
Diesel Generator Cooling Water Surge Tank
Diesel Generator Starting Air Aftercoolers
Diesel Generator Starting Air Filter/Dryer Units I
Diesel Generator Starting Air System Air Receivers
Diesel Generator Lube Oil Cooler
Diesel Generator Lube Oil Sump Tank Heaters
Diesel Generator Intake Turbocharger
Diesel Generator Exhaust Aftercooler
Diesel Generator Intake and Exhaust Silencers and Air Filters
Main Control Room Air Handling Units w/Filters
Main Control Room Water-cooling Coils
Main Control Room Heating Coils
Fuel Building Ventilation Exhaust Filter Train
Reactor Building Subsphere Ventilation System Cooling Coils
Reactor Building Subsphere Ventilation System Filters
Annulus Ventilation System Filters
Spent Fuel Pool Cooling System Heat Exchangers
Station Service Water Strainers

RAI 210.58

Insert A

ASME Code Class 2 and 3 components and supports are analyzed for fatigue as required by the ASME Code Section III, Subsections NC & ND. The analysis will include all estimated transient event cycles and vibration over the design life of the plant.

Question 210.59

CESSAR-DC Section 3.9.3.1.3.3 defines the jurisdictional boundary between only ASME Code, Section III, Class 2 and 3 pumps and the building structure.

Revise Section 3.9.3, including Subsection 3.9.3.4, to provide a commitment that the 1987 Addenda to the 1986 Edition of the ASME Code, Section III, Subsection NF will be used to define the jurisdictional boundary between Subsection NF component supports and the building structure.

Response 210.59

The jurisdictional boundary between System 80+ ASME Code Class 1, 2 and 3 component supports and the building structure is in accordance with the ASME Code Section III, Subsection NF. Section 3.9.3 will be revised in a future amendment to CESSAR-DC to include the following:

"Jurisdictional boundaries between ASME Section III Class 1, 2, and 3 component supports and the building structure are established in accordance with ASME Section III, Subsection NF,"

To allow for flexibility in implementing future ASME Code revisions, Combustion Engineering believes that it is not appropriate to identify specific ASME Code addenda in CESSAR-DC. Code addenda requirements for System 80+ plants will comply with the requirements of 10CFR 50.55a.

Jurisdictional boundaries between ASME Section III Class 1, 2 and 3 component supports and the building structure are established in accordance with ASME Section III, Subsection NF.

C. In-shop seat leakage test.

D. Periodic valve exercise and inspection to assure the functional ability of the valve.

Using the methods described, safety-related active valves in the system are qualified for operability during a seismic event.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

Safety valves and relief valves are analyzed in accordance with the ASME Section III Code.

The method of analysis for safety valves and relief valves suitably accounts for the time-history of loads acting immediately following a valve opening (i.e., first few milliseconds). The fluid-induced forcing functions are calculated for each safety valve and relief valve using one-dimensional equations for the conservation of mass, momentum, and energy. The calculated forcing functions are applied at locations along the associated piping where a change in fluid flow direction occurs. Application of these forcing functions to the associated piping model constitutes the dynamic time-history analysis. The dynamic response of the piping system is determined from the input forcing functions. Therefore, a dynamic amplification factor is inherently accounted for in the analysis. Alternatively, an equivalent static analysis may be used following the criteria given in Appendix II of the ANSI/ASME B31.1 Code. This appendix provides a methodology for calculating appropriate dynamic load factors. Where more than one safety relief valve is installed on the same piping run, the sequence of openings that induces the maximum stress will be considered. E

Snubbers or strut-type restraints are used as required. The stresses resulting from the loads produced by the sudden opening of a relief or safety valve are combined with stresses due to other pertinent loads and are shown to be within allowable limits of the ASME Section III Code. Also, the analyses show that the loads applied to the nozzles of the safety and relief valves do not exceed the maximum loads specified by the manufacturer.

3.9.3.4 Component Supports

Supports for ASME Section III Code Class 1, 2 and 3 components are specified for design in accordance with the loads and loading combinations discussed in Section 3.9.3.1 and presented in Table 3.9-2.

Component supports which are loaded during normal operation, seismic and following a pipe break (branch line breaks not

Question 210.60

CESSAR-DC Section 3.9.3.2 describes the pump and valve operability program for both the NSSS and non-NSSS. As described, the program is not in complete agreement with the guidelines in SRP 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment."

Revise the operability program to be in accordance with SRP 3.10.

Response 210.60

CESSAR-DC Section 3.10 is presently being revised to conform with the intent of the guidance provided by SRP Section 3.10, Revision 2, and USNRC Regulatory Guide 1.100, Revision 2. This revision, which will be included in the submittal of a future amendment of CESSAR-DC, will address compliance with the acceptance criteria of applicable subsections of SRP 3.10 for both mechanical and electrical equipment. Consequently, any necessary revisions to CESSAR-DC Section 3.9.3.2 in order to be consistent with Section 3.10 and address the concerns of this RAI will be included in this future submittal.

Question 210.61

CESSAR-DC Sections 3.9.3.2 and 3.9.4.2 contain several references to a number of IEEE standards and a number of associated endorsing RGs. The editions of the IEEE standards referenced are not current and RG revisions are not identified.

Revise Sections 3.9.3.2, 3.9.4.2, and all other applicable sections of the CESSAR-DC to commit to IEEE-323, 1983 and IEEE-344, 1987, as endorsed by R.G. 1.100, Revision 2.

Response 210.61

Applicable sections of CESSAR-DC, to include Sections 3.9.3.2 and 3.9.4.2, will be revised in a future amendment to CESSAR-DC to commit to IEEE Standard 323-1983 and IEEE Standard 344-1987. NRC Regulatory Guides are identified in Section 1.8 of CESSAR-DC, along with their revison and date. To ensure that the complete text of CESSAR-DC remains consistent, only the title and number of the Regulatory Guide will be referenced in other sections.

(See also response to RAI 210.82)

RAI 210.61

Active Components	Active Safety Function
Safety injection pumps	Operate at flow rates to runout
Shutdown cooling pumps	Operate at design flow
Containment spray	Operate at design flow

E

3.9.3.2.1.2 Operability Assurance Program Results for Active Pumps

Operability of the Safety Injection, Shutdown Cooling and Containment Spray pumps under required conditions ~~has been~~ are demonstrated by analyses of the assemblies and by analyses and tests of the motors.

For the safety injection, shutdown cooling and containment spray pumps, allowable stresses are not exceeded, clearances are acceptable and shaft and pedestal bolt deflections do not cause stresses to exceed the normal values.

Where necessary, lumped mass models are used with the computer programs to determine the natural frequencies and displacements. The models are conservative (i.e., simplifications tend to make them more flexible).

1983

To verify "as-built" conditions the pumps are hydrostatically tested in accordance with the ASME B&PV Code, Section III to confirm acceptability of structural integrity of pressure retaining parts, tested for seal leakage, and tested for performance and NPSH characteristics in accordance with the Hydraulic Institute Standard to verify operation within specified parameters. The motors are Class IE and are tested in accordance with IEEE Standard 112A-1978 to verify operation within specified parameters. Additionally, IEEE Standard 323-~~1974~~, as endorsed by Regulatory Guide 1.89, and IEEE Standard 344-~~1975~~, as endorsed by Regulatory Guide 1.100, are applicable for motors to assure operability during and following design basis events. E

1987

3.9.3.2.1.3 Operability Assurance Program for Active Valves

Safety-related active valves must perform their mechanical motion during or after design basis events. The qualification program assures that these valves will operate during a seismic event. Qualification tests and/or analyses are conducted for all active valves. E and test. The vendor considers concurrent loads including seismic, design pressure and pipe loads.

1983 1987

The three-way solenoid valve ~~was~~ is qualified by test and analysis to IEEE Standard 382-1972, as endorsed by Regulatory Guide 1.73, IEEE Standard 323-1974 and IEEE Standard 344-1975. Testing included thermal aging, radiation aging, wear aging, vibration endurance, seismic event simulation, and loss-of-coolant-accident. All test results provide~~d~~ satisfactory evidence of air solenoid valve operability. E

1983

Limit switches, used to determine valve position, ~~were~~ are qualified by testing and analysis to IEEE Standard 323-1974, IEEE Standard 344-1975 and IEEE Standard 382-1972. Switches ~~were successfully~~ performance tested for aging simulation, wear aging, radiation exposure, seismic qualification, and design basis event environmental conditions. For valves outside of containment and utilizing EA-170 limit switches, the switches ~~were~~ are seismically qualified to IEEE Standard 344-1975 and ~~were~~ are tested to sustain radiation dosages up to 2 x 10^8 rads. E

1987 1987

3.9.3.2.1.3.2 Motor Operated Valves

Motor operated valves are qualified by analysis as a minimum as described above. The analysis for each valve assembly considers the effects of seismic loads, design pressure, and piping reaction forces to provide assurance of operability.

To provide full qualification of the motor operated valve actuator, environmental and seismic qualification tests were conducted to simulate the following conditions:

A. Inside Containment (LOCA).

B. Outside Containment.

C. Seismic Qualification.

D. Steam Line Break Accident.

Mid-size valve actuators were subjected to complete environmental qualification consisting of inside containment and outside containment. Each qualification exposed the actuator to thermal and mechanical aging, radiation aging, seismic aging, environmental transient profile test, and steam line break. For the steam line break test an actuator was subjected to a very high superheated temperature to demonstrate that the electrical components of the actuator never exceeded the saturated temperature corresponding to the ambient pressure for the short duration of the test. This short term test provided evidence E that the existing qualification envelopes the steam line break for superheated temperatures as high as approximately 492°F for a few minutes (see Section 3.11). E

The qualification of the mid-size valve actuator ~~was~~ is used to generically qualify all sizes of mid-size valve actuator operators for the environmental test conditions in accordance with IEEE Standard 382-1972. All sizes are constructed of the same materials with components designed to equivalent stress levels, and to the same clearances and tolerances with the only difference being in physical size which varies corresponding to the differences in unit rating.

All the qualifications ~~were~~ are conducted per IEEE Standard 382-1972 and meet the requirements of IEEE Standard 323-~~1974~~ 1983 and IEEE Standard 344-~~1975~~ 1987 as they apply to valve motor actuators. Further, since the actuators perform~~ed~~ satisfactorily without maintenance throughout the various qualifications, the valve actuators are fully qualified for use in CE Nuclear Power Generating Plants.

3.9.3.2.1.3.3 Pressurizer Safety Valves

Pressurizer Safety valves are 6 x 8 valves. Operability has been successfully demonstrated by a combination of dynamic testing and analysis or by static testing. Operability was successfully demonstrated with a 6g seismic load by one vendor or with a 7.1g seismic load by another vendor. Dynamic testing has demonstrated that the natural frequency of both valves was greater than 33 Hz. A summary of the test programs follows:

A. Vendor A Safety Valves

1. Natural Frequency Demonstration

   Vibration input was in a single, horizontal direction. It was established by previous experience that the horizontal direction was more significant than the vertical direction, and that there was no material difference between the various horizontal directions. The frequency of vibration was increased from 5 to 75 Hz at a rate of 1 octave per minute. Accelerometers were mounted on the valve assembly. The actual natural frequency under test conditions was 38 Hz.

2. Operability Demonstration

   A series of tests demonstrated that the valve would fully open and reseat during and after a seismic acceleration. Vibration input ranged from 3 to 6g and

3.9.3.2.2.2 Valves

Safety-related active valves are subjected to the following tests:

A. Shell hydrostatic tests, in accordance with ASME B&PV Code, Section III requirements.

B. Backseat and main seat leakage tests.

C. Disc hydrostatic tests.

D. Functional tests that verify that the valve will open and close with the specified time limits when subjected to the design differential pressure.

E. Operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident, environment simulation) in accordance with IEEE Standards 323-~~1974~~ 1983, 344-~~1975~~ 1987, and 382-1972.

After installation, cold hydrostatic tests, hot functional tests, and periodic inservice operation are performed to verify and assure the functional ability of the valve. These tests enhance reliability of the valve for the design life of the plant.

The valves are designed using either stress analysis or standard design rules for minimum wall thickness requirements. On all active valves with extended topworks, an analysis is also performed for static equivalent OBE loads applied at the center of gravity of the extended structure. E

The maximum stress limits allowed in the analyses are those recommended by the ASME Code for the particular ASME Class of valve analyzed.

In addition to these tests and analyses, valves are tested for verification of operability during a simulated seismic event by demonstrating operational capabilities within the specified limits. The valve is mounted in a manner that represents typical valve installation. The valve unit includes the operator and all appurtenances normally attached to the valve appurtenances in service. The operability of the valve during SSE is demonstrated by satisfying the following criteria:

A. All the active valves with extended topworks are designed to have a first natural frequency greater than 33 Hz. This may be shown by test and/or analysis. Valves with a first natural frequency less than 33 Hz are discussed below.

B. While in the shop and installed in a suitable test rig, the extended topworks of the valve are subjected to a statically applied equivalent seismic load. The load is applied at the center of gravity of the operator in the direction of the weakest axis of the yoke. The design pressure of the valve is simultaneously applied to the valve during the static load tests.

C. The valve is then operated with the equivalent seismic static load applied (i.e., from the normal operating status to the faulted operating status). The valve must perform its safety-related function within the specified operating time limits. Three full-stroke operations are required.

D. Motor operators and other electrical appurtenances necessary for operation are qualified as operable during the SSE by IEEE Standard 344-1975 1987, Seismic Qualification Standards, prior to their installation on the valve.

The piping designer supports the piping in such a way that the equivalent seismic static load accelerations are not exceeded at the valve inlet and outlet support points. If the frequency of the valve with topworks, by test or analysis, is less than 33 Hz, a dynamic analysis of the valve is performed to determine an equivalent acceleration that is to be applied during the static test. The analysis provides the amplification of the input acceleration considering the natural frequency of the valve and frequency content of the plant floor response spectra. The adjusted accelerations are determined using margins similar to that contained in the horizontal and vertical accelerations used for "rigid" valves. The adjusted accelerations are used in the static analysis, and valve operability is assured by the methods outlined in listings B to D above, using the modified acceleration input. E

The above testing program applies only to valves with overhanging structures (e.g., the operator). The testing is conducted on a representative number of valves. Valves from each of the primary safety-related design types (e.g., motor-operated gate valve) are tested. Specific valves are qualified by the tests, and the results are extended to qualify valves within a range of sizes. An analysis is conducted to prove the similarity between the tested valve and the installed ones.

Due to the simple characteristics of check valves and other compact valves, they are qualified by the following tests and analysis:

A. Stress analysis of the attached piping for SSE loads.
B. In-shop hydrostatic test.

5. The upper latch coil is energized engaging the latches with clearance.

6. The lower lift coil is deenergized allowing the lower latch to drop with the drive shaft. The drive shaft will move down 3/8 inch, stopping on the upper latch assembly, which is energized and in its up position. E

7. The lower latch coil is deenergized disengaging the lower latches.

8. The upper lift coil is deenergized lowering the upper latch assembly with the drive shaft 3/8 inch. E

3.9.4.2 Applicable CEDM Design Specifications

The pressure boundary components are designed and fabricated in accordance with the requirements for Class 1 vessels per the applicable Edition and Addenda of Section III of the ASME Boiler and Pressure Vessel Code. The pressure boundary material complies with the requirements of Section III and IX of the ASME Boiler and Pressure Vessel Code and Code Case N4-11. E

The adequacy of the design of the non-pressure boundary components have been verified by prototype accelerated life testing as discussed in Section 3.9.4.4.

1983

The reed switch position transmitter assembly of the CEDM is designed to comply with IEEE 323-~~1974~~, Standard for "Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations," and IEEE 344-~~1975~~, "Recommended Practice Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations." The electrical components are external to the pressure boundary and are non-pressurized.

The test program to verify the CEDM design is discussed in Section 3.9.4.4.

Standard 1987

3.9.4.3 Design loads, Stress Limits and Allowable Deformations

The CEDM stress analyses consider the following loads:

A. Reactor coolant pressure and temperature

B. Reactor operating transient conditions

C. Dynamic stresses produced by seismic loading

D. Dynamic stresses produced by mechanical excitations

Question 210.62

Section 3.9.3.3 in CESSAR-DC states that safety and relief valve mountings are analyzed in accordance with the ASME Section III Code and briefly describes a dynamic analysis which might be used. However, as an alternative, it is stated that an equivalent static analysis may be used following the criteria in the ANSI/ASME B31.1 Code, Appendix II, "Non-Mandatory Rules for the Design of Safety Valve Installations." For this alternative analysis, the staff's position, as stated in SRP 3.9.3, Section II.2, is that such installations should be designed in accordance with ASME Section III, Appendix O, "Rules for the Design of Safety Valve Installations," as supplemented by the additional criteria in SRP 3.9.3, Section II.2. Either delete the reference to ANSI/ASME B31.1, Appendix II and replace it with a commitment to the staff's position in SRP 3.9.3, or provide a justification for using the B31.1 rules in lieu of the staff's position.

Response 210.62

Reference to Appendix II of the ANSI/ASME B31.1 Code will be deleted from Section 3.9.3.3 and will be replaced with the following:

"Alternately, an equivalent static analysis may be used following the criteria of Appendix O of the ASME Code Section III as supplemented by the additional criteria of SRP3.9.3, Section II.2."

C. In-shop seat leakage test.

D. Periodic valve exercise and inspection to assure the functional ability of the valve.

Using the methods described, safety-related active valves in the system are qualified for operability during a seismic event.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

Safety valves and relief valves are analyzed in accordance with the ASME Section III Code.

The method of analysis for safety valves and relief valves suitably accounts for the time-history of loads acting immediately following a valve opening (i.e., first few milliseconds). The fluid-induced forcing functions are calculated for each safety valve and relief valve using one-dimensional equations for the conservation of mass, momentum, and energy. The calculated forcing functions are applied at locations along the associated piping where a change in fluid flow direction occurs. Application of these forcing functions to the associated piping model constitutes the dynamic time-history analysis. The dynamic response of the piping system is determined from the input forcing functions. Therefore, a dynamic amplification factor is inherently accounted for in the analysis. ~~Alternatively, an equivalent static analysis may be used following the criteria given in Appendix II of the ANSI/ASME B31.1 Code. This appendix provides a methodology for calculating appropriate dynamic load factors. Where more than one safety relief valve is installed on the same piping run, the sequence of openings that induces the maximum stress will be considered.~~

Delete

Insert A

Snubbers or strut-type restraints are used as required. The stresses resulting from the loads produced by the sudden opening of a relief or safety valve are combined with stresses due to other pertinent loads and are shown to be within allowable limits of the ASME Section III Code. Also, the analyses show that the loads applied to the nozzles of the safety and relief valves do not exceed the maximum loads specified by the manufacturer.

3.9.3.4 Component Supports

Supports for ASME Section III Code Class 1, 2 and 3 components are specified for design in accordance with the loads and loading combinations discussed in Section 3.9.3.1 and presented in Table 3.9-2.

Component supports which are loaded during normal operation, seismic and following a pipe break (branch line breaks not

RAI 210.62

Insert

" Alternately, an equivalent static analysis may be used following the criteria of Appendix O of the ASME Code Section III as supplemented by the additional criteria of SRP3.9.3, Section II.2. "

Question 210.63

SRP 3.9.3, Rev. 1, subsection II.1, defines design criteria for internal parts of ASME Code, Section III components such as valve discs, seats and pump shafts. CESSAR-DC Section 3.9.3 does not contain such criteria.

Provide the design criteria for internal parts in accordance with SRP 3.9.3, Rev. 1, subsection II.1.

Response 210.63

As noted in CESSAR-DC Section 3.9.3, all components are designed and constructed in accordance with Section III of the ASME Code. The loading combonations, design transients, and stress limits for these components are covered in CESSAR-DC Section 3.9.3.1. Structural integrity of the pumps' and valves' internal parts is covered in CESSAR-DC Section 3.9.3.2, Pump and Valve Operability Assurance.

Question 210.64

SRP 3.9.3, Rev. 1, subsection II.3.a, specify criteria for component supports for active pumps and valves. CESSAR-DC Section 3.9.3 does not contain such criteria.

Provide the design criteria for component supports for active pumps and valves in accordance with SRP 3.9.3, Rev. 1, subsection II.3.a.

Response 210.64

CESSAR-DC contains design criteria throughout Section 3.9.3. Table 3.9-3 provides stress limits for component supports and discusses Regulatory Guide 1.124, Regulatory Guide 1.130, and ASME Section III, Subsection NF. Operability assurance aspects are treated in CESSAR-DC Section 3.9.3.2. Additionally, CESSAR-DC Section 3.9.3.4 discusses Component Supports.

Question 210.65

SRP 3.9.3, Rev. 1, Subsection II.7, states that a listing should be provided of all safety-related components which utilize snubbers. The tabulation should include the following information:

a. Identification of the systems and components in those systems which utilize snubbers.

b. The number of snubbers utilized in each system and on components in that system.

c. The type(s) of snubber (hydraulic or mechanical) and the corresponding supplier identified.

d. Specify whether the snubber was constructed to the rules of ASME Code Section III, Subsection NF.

e. State whether the snubber is used as a shock, vibration, or dual purpose snubber.

f. For snubbers identified as either dual purpose or vibration arrester type, indicate if both snubber and components were evaluated for fatigue strength.

Provide the above information in CESSAR-DC Section 3.9.3.

Response 210.65

A listing of all safety-related components which utilize snubbers including the requested detail information requires detail plant arrangement, piping layout and piping design. As presented to the staff at the meeting of November 26, 1991, detailed piping system design and layout and plant arrangements are (1) not required for certification, (2) depend on plant specific details not finalized at the certification stage and (3) are subject to revision until specific piping, components, and other plant design feature details are finalized.

CESSAR-DC currently provides the general design and operability assurance acceptance criteria.

Response 210.65 (Continued)

Combustion Engineering believes that the detailed information requested including type of snubber and supplier is not required for certification. For information, however, the following snubber applications have been presently identified:

Reactor Coolant System

1. Steam Generator Snubbers
   - Hydraulic
   - ASME III Subsection NF
   - Shock Arrestor

2. Reactor Coolant Pump Snubbers
   - Hydraulic
   - ASME III Subsection NF
   - Shock Arrestor

Question 210.66

In Table 3.2-1 and Section 3.9.3.4 of the CESSAR-DC, provide a commitment that concrete expansion anchor bolts which are used for pipe support base plates are designed to the applicable factors of safety and baseplate flexibility accountability requested in I&E Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Revision 2 dated November 8, 1979.

Response 210.66

The requirements of ACI-349, Code Requirements for Nuclear Safety Related Concrete Structures, are imposed and the issues identified in I&E Bulletin 79-02, are addressed during detailed baseplate design to avoid previous problems experienced with concrete expansion anchor bolts.

Section 3.9.3.4 will be revised in a future amendment to add the following:

Expansion anchors are designed in accordance with ACI-349, Code Requirements for Nuclear Safety Related Concrete Structures. This assures that the design strength of concrete for a given expansion anchor or group of anchors is greater than the strength of the anchor steel, accounts for the effect of shear-tension interaction, and considers minimum edge distance and bolt spacing on expansion anchor capacity. In addition, base plate flexibility is accounted for in the calculation of expansion anchor bolt loads.

Table 3.2-1 will be updated in a future amendment to add Component Supports under Structures and the note below will be referenced. This is in the response to RAI 210.1.

Note (23) will be added to Table 3.2-1.

(23) Component Supports are designed to the criteria described in Section 3.9.3.4.

RAI 210.66

eliminated by leak-before-break) are specified for design for loading combinations (A) through (D) of Section 3.9.3.1. Design stress limits applied in evaluating loading combinations (A), (B), and (C) of Section 3.9.3.1 are consistent with the ASME Code, Section III. The design stress limits applied in evaluating loading combination (D) of Section 3.9.3.1 are in accordance with the ASME B&PV Code, Section III. Loads in compression members are limited to 2/3 of the critical buckling load.

Insert

Where required, snubber supports are used as shock arrestors for safety-related systems and components. Snubbers are used as structural supports during a dynamic event such as an earthquake or a pipe break, but during normal operation act as passive devices which accommodate normal expansions and contractions of the systems without resistance. For System 80+, snubbers are minimized, to the extent practical, through the use of design optimization procedures.

Assurance of snubber operability is provided by incorporating analytical, design, installation, in-service, and verification criteria. The elements of snubber operability assurance for System 80+ include: E

A. Consideration of load cycles and travel that each snubber will experience during normal plant operating conditions.

B. Verification that the thermal growth rates of the system do not exceed the required lock-up velocity of the snubber.

C. Accurate characterization of snubber mechanical properties in the structural analysis of the snubber-supported system.

D. For engineered, large bore snubbers, issuance of a design specification to the snubber supplier, describing the required structural and mechanical performance of the snubber; verification that the specified design and fabrication requirements are met.

E. Verification that snubbers are properly installed and operable prior to plant operation, through visual inspection and through measurement of thermal movements of snubber-supported systems during start-up tests.

F. A snubber in-service inspection and testing program, which includes periodic maintenance and visual inspection, inspection following a transient event, a functional testing program, and repair or replacement of snubbers failing inspection or test acceptance criteria.

RAI 210.66

CESSAR-DC Change

Section 3.9.3.4

Insert as Paragraph #3

Expansion anchors are designed in accordance with ACI-349, Code Requirements for Nuclear Safety Related Concrete Structures. This assures that the design strength of concrete for a given expansion anchor or group of anchors is greater than the strength of the anchor steel, accounts for the effect of shear-tension interaction, and considers minimum edge distance and bolt spacing on expansion anchor capacity. Base plate flexibility is accounted for in the calculation of expansion anchor bolt loads.

Question 210.67

The ASME Code requires that a design specification be prepared for all ASME Class 1, 2, and 3 components. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations and other design data inputs. The code also requires a design report for all such components. As a part of its review of CESSAR-DC, the staff will review documents related to design specifications and design reports for a small number of ASME Class 1, 2, and 3 pumps, valves, and piping systems. The objective of this review will be to provide the staff with the basis for concluding that the System 80+ design documentation meets the applicable requirements of ASME Section III, Subsection NCA. Details of this review will be transmitted later in a separate letter to CE. In the interim, the staff requests CE to agree to submit such documents to the staff on a mutually acceptable date.

Response 210.67

Combustion Engineering agrees to provide design documentation for NRC audit in accordance with this RAI. In addition, programs for design acceptance criteria are being provided. If, as a result of such an audit, NRC staff determines that specific internal documentation is relied upon for SER conclusions, that information will be submitted either as part of CESSAR-DC or as a separate proprietary document. Note, however, that most, if not all Class 1, 2 and 3 pump, valve and piping system Design Reports are provided by vendors and are not available at time of design certification.

Question 210.68

CESSAR-DC Section 9.4, "Air Conditioning, Heating, Cooling and Ventilation Systems" (HVAC) describes the ventilation systems for the different plant buildings and areas. For each system, certain codes, standards, practices, etc. (e.g., SMACNA, ASME/ANSI AG-1, ASME B&PV Code, Section III, Class 3, etc.) are referenced as being applicable to the manufacture, installation, testing, and performance of HVAC equipment. Describe in detail how these codes, standards, and practices address the seismic design criteria, installation criteria, and quality assurance requirements for safety-related HVAC equipment including ductwork, duct supports, cooling and heating coils, piping, piping supports, etc.

Response 210.68

The codes and standards that are referenced in the CESSAR-DC Section 9.4 "Air Conditioning, Heating, Cooling and Ventilation Systems" do not specifically detail the criteria for seismic design, installation or quality assurance for safety-related HVAC equipment. The ductwork is seismically designed in accordance with ASME/ANSI AG-1-1988 Article AA-4000 Structural Design. The cooling and heating coils, piping, pipe supports, etc. are seismically designed in accordance with ASME B&PV Code, Section III, Class 3. Quality assurance requirements will be met according to the program described in CESSAR-DC, Chapter 17 (CENPD-210-A).

Question 210.69:

10 CFR 50.55a contains specific ASME Code requirements for reactor coolant pressure boundary and Quality Groups B and C components. Table 5.2-1 in CESSAR-DC contains some of the required information for the reactor coolant pressure boundary. Supplement the requirements in Table 5.2-1 by adding supports to the Components column and by adding the applicable ASME Code Edition or Addenda for all of the components that are listed in the table. In addition, in either Sections 3.2.2 or 3.9.3 of CESSAR-DC, provide a similar table for all Quality Groups B and C (Safety Class 2 and 3) components, equipment, and their supports.

Response 210.69:

Primary supports will be added to Table 5.2-1 in a future amendment.

Table 3.2-1 in Amendment I is a table similar to Table 5.2-1 and it contains Safety Classes 2 and 3. The table will be revised in a future amendment, and supports will be a part of that amendment.

Combustion Engineering believes that it is not appropriate to specify the applicable ASME Code Edition and Addenda in CESSAR-DC to allow future revisions to the Code to be implemented without reopening design certification proceedings. A sentence will be added to Section 3.2.2 stating that the code edition and addenda requirements for System 80+ plants will comply with requirements of 10CFR50.55a.

RAI 210.69

C. In-shop seat leakage test.

D. Periodic valve exercise and inspection to assure the functional ability of the valve.

Using the methods described, safety-related active valves in the system are qualified for operability during a seismic event.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

Safety valves and relief valves are analyzed in accordance with the ASME Section III Code.

The method of analysis for safety valves and relief valves suitably accounts for the time-history of loads acting immediately following a valve opening (i.e., first few milliseconds). The fluid-induced forcing functions are calculated for each safety valve and relief valve using one-dimensional equations for the conservation of mass, momentum, and energy. The calculated forcing functions are applied at locations along the associated piping where a change in fluid flow direction occurs. Application of these forcing functions to the associated piping model constitutes the dynamic time-history analysis. The dynamic response of the piping system is determined from the input forcing functions. Therefore, a dynamic amplification factor is inherently accounted for in the analysis. Alternatively, an equivalent static analysis may be used following the criteria given in Appendix II of the ANSI/ASME B31.1 Code. This appendix provides a methodology for calculating appropriate dynamic load factors. Where more than one safety relief valve is installed on the same piping run, the sequence of openings that induces the maximum stress will be considered. E

Snubbers or strut-type restraints are used as required. The stresses resulting from the loads produced by the sudden opening of a relief or safety valve are combined with stresses due to other pertinent loads and are shown to be within allowable limits of the ASME Section III Code. Also, the analyses show that the loads applied to the nozzles of the safety and relief valves do not exceed the maximum loads specified by the manufacturer.

ADD INSERT A

3.9.3.4 Component Supports

Supports for ASME Section III Code Class 1, 2 and 3 components are specified for design in accordance with the loads and loading combinations discussed in Section 3.9.3.1 and presented in Table 3.9-2.

Component supports which are loaded during normal operation, seismic and following a pipe break branch line breaks not

RAT 210.69

Insert A

Component supports are designed and constructed to the requirements that are applicable to the class of component they are intended to support. Component classes are identified in Table 3.2-1.

A. Safety Class 1 (SC-1) applies to pressure-retaining portions and supports of mechanical equipment that form part of the RCPB whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability and whose requirements are within the scope of the ASME Boiler and Pressure Vessel Code, Section III.

B. Safety Class 2 (SC-2) applies to pressure-retaining portions and supports of primary containment and other mechanical equipment, requirements for which are within the scope of the ASME Boiler and Pressure Vessel Code, Section III, that are not included in SC-1 and are designed and relied upon to accomplish the nuclear safety functions defined in ANSI/ANS 51.1, Section 3.3.1.2.

C. Safety Class 3 (SC-3) applies to equipment, not included in SC-1 or -2, that is designed and relied upon to accomplish the nuclear safety functions defined in ANSI/ANS 51.1, Section 3.3.1.3. D

D. Non-Nuclear Safety (NNS) applies to equipment that is not in Safety Class 1, 2, or 3. This equipment is not relied upon to perform a nuclear safety function.

The safety classifications of major components which are in the System 80+ design scope are listed in Table 3.2-1 and Section 3.11. Seismic category designations and quality assurance requirements are also included. Small components, such as piping, valves and strainers, are not listed; they may be found by reference to the P&IDs (Chapters 5, 6, and 9) where the exact boundaries are indicated. Valves are listed in Tables 3.2-2. D I

All pressure containing components in Safety Classes 1, 2, and 3 are designed, manufactured, and tested in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III. Insert B Components designated NNS are designed and constructed with appropriate consideration of the intended service using applicable industry codes and standards. The relationship between safety class and code class is shown in Table 3.2-2. A higher code class may be used for a component without changing the safety class or affecting the balance of the system in which it is located. D

Fracture toughness requirements are imposed on materials for pressure retaining parts of ASME Class 2 and 3 System 80+ Standard Design components. Test methods, acceptance, and exemption criteria are in conformance with the ASME Code, Section III. I

The safety classification system is also used to identify those components to which the requirements of 10 CFR 50, Appendix B,

RAT 210.69

Insert B

Code edition and addenda requirements for System 80+ plants will comply with the requirements of 10 CFR 50.55a.

TABLE 3.2-3

RELATIONSHIP OF SAFETY CLASS TO CODE CLASS

Safety Class	Code Class (ASME Section III)
SC-1	1
SC-2 for reactor containment components	2
SC-2 for fluid system components	2
SC-3 for core support structures	CS
SC-3 (otherwise)	3
NNS	Industry Standards

I

D

Code edition and addenda requirements are addressed in Section 3.2.2.

TABLE 5.2-1

REACTOR COOLANT SYSTEM PRESSURE BOUNDARY CODE REQUIREMENTS

Components	Codes and Classes
Reactor Vessel, Steam Generators (primary side), Pressurizer	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1.
Reactor Coolant Pump (structural portions necessary to assure the integrity of the reactor coolant pressure boundary)	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1.
Reactor Coolant Pump Auxiliaries	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 3. Lube oil system designed for Seismic Category I requirements.
Pressurizer Spray and Safety Valves	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1.
Piping and Valves	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1.
Steam Generators (Secondary Side)	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 2.
Control Element Drive Mechanisms	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1.
Primary Component Supports	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Plant Component Supports, Class 1

D

NOTES: Code edition and addenda requirements for System 80+ plants ~~will comply with the requirements of 10 CFR 50.55a~~. are addressed in Section 3.2.2.

Codes listed above are construction codes. In addition, all these components are designed and constructed to meet the test and inspection requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection.

Code requirements for Safety Depressurization System valves, which meet the definition of the Reactor Coolant Pressure Boundary, are given in Section 6.7. D

Question 210.70

CESSAR-DC Section 3.9.4.1 specifies a CEDM design life of 60 years and a total cumulative CEA travel of 100,000 feet of operation without loss of function. In addition, CESSAR-DC Section 3.9.4.4.1.3 states that the System 80+ CEDMs are identical to those in operation at PVNGS. The CEDMs for the 40-year PVNGS plant were also designed for a total cumulative CEA travel of 100,000 feet of operation without loss of function.

Explain why the CEA design travel for both the 40-year design life PVNGS plant and the 60-year design life for a CE System 80+ plant are both 100,000 feet.

In addition, CESSAR-DC Section 3.9.4.4.1.3 states that the PVNGS CEDMs have operated without malfunction and without any measurable wear. Describe the surveillance program that has been implemented on the PVNGS plant to demonstrate satisfactory CEDM operating experience.

Response 210.70

System 80 Control Element Drive Mechanism [CEDM] motor assemblies and drive rods are designed for a cumulative CEA travel of 100,000 feet. This provides sufficient design margin for these plants to insure component reliability under anticipated operating conditions. Should the need arise CEDM motor assemblies may be replaced by removing the upper pressure housing on the CEDM assembly. Replacement can be accomplished without affecting adjacent CEDM's or requiring removal of the reactor vessel closure head. These features are consistent with the 60 year plant design life and the maintenance schedules identified in paragraph 3.9.4.1. In addition, CEDM components that make up the pressure boundary are designed to withstand the number of operating transients expected during a 60 year plant design life.

Though maintenance schedules have been recommended, there is no regular program to specifically monitor component wear in normally functioning plants. Paragraph 3.9.4.4.1.3 will be revised to delete Reference to "measurable wear". Component wear is inferred based on measurements conducted on CEDM motors subjected to operational tests. This revision will be made in a future amendment to CESSAR-DC.

RAI 210.70, 71, 72, 73 & 74

3.9.4 CONTROL ELEMENT DRIVE MECHANISMS

3.9.4.1 Descriptive Information of CEDM

The control element drive mechanism (CEDMs) are magnetic jack type drives used to vertically position and indicate the position of the control element assemblies (CEAs). Each CEDM is capable of withdrawing, inserting, holding, or tripping the CEA from any point within its 153-inch stroke in response to operation signals.

The CEDM is designed to function during and after all normal plant transients. The CEA drop time for 90% insertion is 4.0 seconds maximum. The drop time is defined as the interval between the time power is removed from the CEDM coils to the time the CEA has reached 90% of its fully inserted position. The CEDM pressure boundary components have a design life of 60 years. The CEDM is designed to operate without maintenance for a minimum of 1-1/2 years and without replacing components for a minimum of 3 years. The CEDM is designed to function normally during and after being subjected to the Operating Basis Earthquake loads. The CEDM will allow for tripping of the CEA during and after a Safe Shutdown Earthquake.

The design and construction of the CEDM pressure housing fulfill the requirements of the ASME boiler and Pressure Vessel Code, Section III, for Class 1 vessels. The CEDM pressure housings are part of the reactor coolant pressure boundary, and they are designed to meet stress requirements consistent with those of the vessel. The pressure housings are capable of withstanding, throughout the design life, all normal operating loads, which include the steady-state and transient operating conditions specified for the vessel. Mechanical excitations are also defined and included as a normal operating load. The CEDM pressure housings are service rated at 2500 psi at 650°F. The loading combinations and stress limit categories are presented in Table 3.9-16 and are consistent with those defined in the ASME code.

The design duty requirements for the CEDM is a total cumulative CEA travel of 100,000 feet operation without loss of function.

The test programs performed in support of the CEDM design are described in Section 3.9.4.4.

3.9.4.1.1 Control Element Drive Mechanism Design Description

The CEDMs are mounted on nozzles on top of the reactor vessel closure head. The CEDMs consist of the upper and lower CEDM pressure housings, motor assembly, coil stack assembly, reed switch assemblies, and extension shaft assembly. The CEDM is shown in Figure 3.9-8. The drive power is supplied by the coil stack assembly, which is positioned around the CEDM housing. Two position indicating reed switch assemblies are supported by the upper pressure housing shroud, which encloses the upper pressure housing assembly.

The lifting operation consists of a series of magnetically operated step movements. Two sets of mechanical latches are utilized engaging a notched extension shaft. To prevent excessive latch wear, a means has been provided to unload the latches during the engaging operations. The magnetic force is obtained from large dc magnet coils mounted on the outside of the lower pressure housing.

Power for the electromagnets is obtained from two separate supplies. A control programmer actuates the stepping cycle and moves the CEA by a forward or reverse stepping sequence. Control element drive mechanism hold is obtained by energizing one coil at a reduced current, while all other coils are deenergized. The CEAs are tripped upon interruption of electrical power to all coils. Each CEDM is connected to the CEAs by an extension shaft. The weight of the CEDMs and the CEAs is carried by the pressure vessel head. Installation, removal, and maintenance of the CEDM is possible with the reactor vessel head in place; however, the missile shield placed over the reactor vessel cavity makes the CEDMs inaccessible during operation of the plant. E

The axial position of a CEA in the core is indicated by three independent readout systems. One counts the CEDM steps electronically, and the other two consist of magnetically actuated reed switches located at regular intervals along the CEDM. These systems are designed to indicate CEA position to within ± 2-1/2 inches of the true location. This accuracy requirement is based on ensuring that the axial alignment between CEAs/PLCEAs is maintained within acceptable limits. E

The materials in contact with the reactor coolant used in the CEDM are listed in Section 4.5.1.

RAI 210.70, 71, 72, 73 & 74

3.9.4.1.1.1 CEDM Pressure Housing

The CEDM pressure housing consists of the motor housing assembly and the upper pressure housing assembly. The motor housing assembly is attached to the reactor vessel head nozzle by means of a threaded joint and seal welded. Once the motor housing assembly is seal welded to the head nozzle, it need not be removed since all servicing of the CEDM is performed from the top of the housing. The upper pressure housing is threaded into the top of the motor housing assembly and seal welded. The upper pressure housing encloses the CEDM extension shaft and contains a vent.

3.9.4.1.1.2 Motor Assembly

The motor assembly is an integral unit which fits into the motor housing and provides the linear motion to the CEA. The motor assembly consists of a latch guide tube, upper latches and lower latches.

Both upper latches and lower latches are used to perform the stepping of the CEA and by proper sequencing perform a load transfer ~~function~~ and ~~to~~ minimize latch and extension shaft wear. The upper latch also performs the holding when CEA motion is not required. Engagement of the extension shaft occurs when the appropriate set of magnetic coils is energized. This moves sliding magnets which cam a two-bar linkage moving the latches inward. The upper latches move vertically 7/16 inches while the lower latches move vertically 3/8 inches to perform both the load transfer and stepping action. Total CEA motion per cycle is 3/4 inches.

3.9.4.1.1.3 Coil Stack Assembly

The coil stack assembly for the CEDM consists of four large DC magnet coils mounted on the outside of the motor housing assembly. The coils supply magnetic force to actuate mechanical latches for engaging and driving the CEA extension shaft. Power for the magnetic coils is supplied from two separate supplies. A CEDM control system actuates the stepping cycle and obtains the correct CEA position by a forward or reverse stepping sequence. CEDM hold is obtained by energizing the upper latch coil at a reduced current while all other coils are deenergized. The CEAs are tripped upon interruption of electrical power to all coils. Electrical pulses from the magnetic coil power programmer provide one of the means for transmitting CEA position indication.

A conduit assembly containing the lead wires for the coil stack assembly is located at the side of the upper pressure housing shroud.

RAI 210.70, 71, 72, 73 & 74

3.9.4.1.1.4 Reed Switch Assembly

Two reed switch assemblies provide separate means for transmitting CEA position indication. Reed switches and voltage divider networks are used to provide two independent output voltages proportional to the CEA position. The reed switch assemblies are positioned so as to utilize the permanent magnet in the top of the extension shaft. The permanent magnet actuates the reed switches as it passed by them. The reed switch assemblies are provided with accessible electrical connectors at the top of the upper pressure housing.

3.9.4.1.1.5 Extension Shaft Assembly

The extension shaft assemblies are used to link the CEDMs to the CEAs. The extension shaft assembly is a 304 stainless steel rod with a permanent magnet assembly at the top for actuating reed switches in the reed switch assembly, a center section called the drive shaft and a lower end with a coupling device for connection to the CEA.

The drive shaft is a long tube made of Type 304 stainless steel. It is threaded and pinned to the extension shaft. The drive shaft has circumferential notches in 3/4 inch increments along the shaft to provide the means of engagement to the control element drive mechanism.

The magnet assembly, located in the top of the extension shaft assembly, consists of a housing, magnet and plug. The magnet is made of two cylindrical alnico -5 magnets. This magnet assembly is used to actuate the reed switch position ~~indication~~ indicator and is contained in a housing which is plugged at the bottom of the housing.

3.9.4.1.2 Description of the CEDM Motor Operation

Withdrawal or insertion of the CEA is accomplished by programming current to the various coils. There are three programmed conditions for each coil (i.e., high voltage for initial gap closure, low voltage for maintaining the gap closed and zero voltage to allow opening of the gap).

3.9.4.1.2.1 Operating Sequence for the Double Stepping Mechanism

The initial condition is the hold mode. In this condition, the upper latch coil is energized at low voltage.

RAI 210.70, 71, 72, 73 & 74

A. Withdrawal (Ref. Figure 3.9-8)

1. The upper lift coil is energized causing the 7/16" upper lift gap to close lifting the CEA.

2. Low current is supplied to hold the CEA in the withdrawn position.

3. The lower latch coil is energized causing the lower latches to engage the drive shaft with 1/32-inch clearance.

4. The upper lift coil is deenergized allowing the upper latches to drop 7/16 inches and the drive shaft to lower 1/32 inches placing the load on the lower latches.

5. The upper latch coil is deenergized disengaging the upper latches.

6. The lower lift coil is energized lifting the drive shaft 3/8 inches.

7. The upper latch coil is energized engaging the upper latches in the drive shaft with 1/32-inch clearance.

8. The lower lift coil is deenergized allowing the lower latches to drop 3/8 inches and causing the drive shaft to drop 1/32 inches applying the load on the upper latches.

9. The lower latch coil is deenergized disengaging the lower latches from the drive shaft.

B. Insertion

1. The lower latch coil is energized causing the lower latches to engage the drive shaft.

2. The lower lift coil is energized lifting the lower latches 3/8 inches and lifting the drive shaft 1/32 inches thus applying the load to the lower latches.

3. The upper latch coil is deenergized causing the upper latches to disengage the drive shaft.

4. The upper lift coil is energized moving the deenergized upper latch assembly up 7/16 inches. E

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and the criteria of SRP 3.9.4, Rev. 2 Subsection II.2.

5. The upper latch coil is energized engaging the latches with clearance.

6. The lower lift coil is deenergized allowing the lower latch to drop with the drive shaft. The drive shaft will move down 3/8 inch, stopping on the upper latch assembly, which is energized and in its up position. E

7. The lower latch coil is deenergized disengaging the lower latches.

8. The upper lift coil is deenergized lowering the upper latch assembly with the drive shaft 3/8 inch. E

3.9.4.2 Applicable CEDM Design Specifications

The pressure boundary components are designed and fabricated in accordance with the requirements for Class 1 vessels per the applicable Edition and Addenda of Section III (subsections NCA and NB) of the ASME Boiler and Pressure Vessel Code ~~.~~ The pressure boundary material complies with the requirements of Section III and IX of the ASME Boiler and Pressure Vessel Code and Code Case ~~N4-11~~ N-4-11. E

The adequacy of the design of the non-pressure boundary components have been verified by prototype accelerated life testing as discussed in ~~Section~~ 3.9.4.4.

Standard

The reed switch position transmitter (RSPT) assembly of the CEDM is designed to comply with IEEE 323-~~1974~~ 1983 standard for "Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations," and IEEE 344-~~1975~~ 1987 "Recommended Practice Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.100. The electrical components are external to the pressure boundary and are non-pressurized.

The test program to verify the CEDM design is discussed in Section 3.9.4.4.

3.9.4.3 Design loads, Stress Limits and Allowable Deformations

The CEDM stress analyses consider the following loads:

A. Reactor coolant pressure and temperature

B. Reactor operating transient conditions

C. Dynamic stresses produced by seismic loading and design bases pipe breaks

D. Dynamic stresses produced by mechanical excitations

Full length RSPT assemblies are subjected to biaxial random multi-frequency input motions corresponding to design bases excitations. Testing is performed using four RSPT orientations to account for asymmetries in the design.

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E. Loads produced by the operation and tripping of the mechanism

The methods used to demonstrate that the CEDMs operate properly under seismic conditions are presented in Section 3.7.3.14.

The design and fabrication of the CEDM pressure boundary components fulfills the requirements of the ASME Code, Section III, for Class I vessels. The pressure housings are capable of withstanding throughout the design life all the steady state and transient operating conditions specified in Table 3.9-16.

The adequacy of the design of the CEDM pressure boundary and non-pressure boundary components has been verified by prototype accelerated life testing as discussed in Section 3.9.4.4.

Clearances for thermal growth and for dimensional tolerances were investigated, and tests have proven that adequate clearances are provided for proper operation of the CEDM.

The latch locations are set by a master gauge, and settings are verified by testing at reactor conditions.

A weldable seal closure, per Section III of the ASME Code, is provided for the vent valve in case of leakage.

The motor housing fasteners are mechanically positively captured, and all threaded connections are preloaded before capturing.

The coil stack assembly can be installed or removed simply by lowering or lifting the stack, relative to the CEDM pressure housing, for ease of coil replacement or maintenance.

3.9.4.4 CEDM Performance Assurance Program

3.9.4.4.1 CEDM Testing

3.9.4.4.1.1 Prototype Accelerated Life Tests

The System 80+ CEDM is similar to and based on existing magnetic jack mechanisms presently in use on operating reactors such as Maine Yankee (Docket No. 50-309) and Calvert Cliffs (Docket 50-317), the 150-inch core reactors such as Arkansas Nuclear One Unit 2 (Docket No. 50-368) and San Onofre Units 2 & 3 (Docket No. 50-361/362), and is the same as the System 80 CEDM presently in use at Palo Verde (Docket Nos. 50-528, 529).

The significant differences between the System 80+ drives and pre-System 80 CEDMs are: E

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A. The elimination of the pulldown coil.

B. The use of the lift coils to perform both a load transfer function and stepping action.

The elimination of the pulldown coil required installation of a coil spring to ensure positive resetting of the latch assemblies. In addition, the drive shaft was modified by placing the teeth on 3/4-inch pitch in place of the 3/8-inch spacing of previous drive shafts to allow load transfer and stepping with the same coil. The safety release mechanism uses the same materials and clearances as on all previous magnetic jack mechanisms. The following describes accelerated life tests on both a pre-System 80 mechanism as well as on a prototype System 80 CEDM. Both programs provide design verification for the System 80+ CEDM. E

A pre-System 80 prototype CEDM was subjected to an accelerated life test accumulating a minimum of 157,000 feet of travel on all CEDM components. In addition, the latch guide tube bearings in the motor assembly saw an additional 50,000 feet of operation.

The prototype mechanism was installed on a test facility which was operated at a nominal temperature of 600°F and 2250 psi. After 50,000 feet of operation lifting 230 pounds at 40 inches per minute, the motor was removed from the test motor housing and the bearing surfaces inspected. During this inspection it was found that excessive wear existed on the upper gripper magnet and upper gripper housing bearings.

The gripper housing magnet bearing configuration was revised and replacement parts with this revision were incorporated into the prototype mechanism. This configuration was reinstalled into the test facility and the mechanism operated as before for an additional 157,000 feet of travel. The replacement parts showed a wear of only .001 inches while the latch guide tube bearings had a total wear of 0.012 inches. The mechanism at disassembly was still operational with no abnormalities. This test constituted operation equivalent to 1.5 to 2.0 times the design duty requirements of the mechanism.

A prototype System 80 CEDM was assembled and installed in a test loop, where the accelerated wear test was conducted at 615°F and 2250 psi. The total weight attached to the CEDM was 450 pounds and this was moved at a nominal speed of 30 inches per minute. A total of 34,000 feet of travel was then completed without difficulty. Included in that test footage were 300 full-height gravity scrams.

The mechanism motor was removed from the test facility and disassembled for inspection. The latch guide tube bearings showed a maximum diametral wear of 0.003 inches with negligible wear on the gripper housing to gripper magnetic bearings. Alignment tabs, which maintain orientation of the gripper with the latch guide tube, showed extensive wear but had not caused mechanism malfunctions. These alignment tabs have been replaced in the production units with an improved design.

Upon completion of the accelerated wear test, 300 full height light weight drops were completed utilizing a 75-pound test weight. The maximum CEA drop time to 90% insertion was 2.93 seconds which met the 4.0 second criterion. All release times were less than the 0.3 seconds with normal releases completed in less than 0.200 seconds.

3.9.4.4.1.2 First Production Test

A qualification test program was completed on the first production C-E magnetic jack CEDM. A similar test program was invoked for the System 80 CEDMs. During the course of this program, over 4000 feet of travel was accumulated and 30 full height gravity drops were made without mechanism malfunction or measurable wear on operating parts. The program included the following:

A. Operation at 40 in./min lifting 230 pounds (dry) at ambient temperature and 2300 psig pressure for 800 feet.

B. Six full-height 230 pounds dry weight gravity drops at ambient temperature.

C. Operation at simulated reactor operating condition at 40 in/min lifting 230-pound for 1700 feet.

D. Six full-height drops at simulated reactor operating conditions with 230 pounds of weight.

E. An operational test at ambient temperature and 2300 psig pressure, lifting 335 pounds for 500 feet. | E

F. Six full-height drops of the 335 pound weight.

G. Operation at simulated reactor conditions for 1700 feet at 20 in/min, lifting 335 pounds.

H. Operation at ambient temperature and 2300 psig for 1100 feet and 20 full-height drops with an attached dry weight of 130 pounds.

The mechanism operated without malfunction throughout the test program and, upon final inspection, no measurable wear was found.

3.9.4.4.1.3 Operating Experience at the Palo Verde Nuclear Generating Station

The System 80+ CEDMs are identical to those in operation at PVNGS. That experience has shown that the CEDMS operate without malfunction~~, and without any measurable wear~~. E

3.9.5 REACTOR VESSEL CORE SUPPORT AND INTERNALS STRUCTURES

3.9.5.1 Design Arrangements

The components of the reactor vessel core support structures are divided into two major parts consisting of the core support structure and the upper guide structure assembly. The flow skirt, although functioning as an integral part of the coolant flow path, is separate from the internals and is affixed to the bottom head of the pressure vessel. The arrangement of these components is shown in Figure 3.9-9.

3.9.5.1.1 Core Support Structure

The major structural member of the reactor internals is the core support structure. The core support structure consists of the core support barrel and the lower support structure. The material for the assembly is Type 304 stainless steel.

The core support structure is supported at its upper end by the upper flange of the core support barrel, which rests on a ledge in the reactor vessel. Alignment is accomplished by means of four equally spaced keys in the flange, which fit into the keys in the vessel lodge and closure head. The lower flange of the core support barrel supports, secures, and positions the lower support structure and is attached to the lower support structure by means of a welded flexural connection. The lower support structure provides support for the core by means of support beams that transmit the load to the core support barrel lower flange. The locating pins in the beams provide orientation for the lower ends of the fuel assemblies. The core shroud, which provides a flow path for the coolant and lateral support for the fuel assemblies, is also supported and positioned by the lower support structure. The lower end of the core support barrel is restricted from excessive radial and torsional movement by six snubbers which interface with the pressure vessel wall.

3.9.5.1.1.1 Core Support Barrel

The core support barrel is a right circular cylinder including a heavy external ring flange at the top end and an internal ring flange at the lower end. The core support barrel is supported from a ledge on the pressure vessel. The core support barrel, in

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turn, supports the lower support structure upon which the fuel assemblies rest. Press-fitted into the flange of the core support barrel are four alignment keys located 90 degrees apart. The reactor vessel, closure head, and upper guide structure assembly flange are slotted in locations corresponding to the alignment key locations to provide alignment between these components in the vessel flange region. The core support barrel assembly is shown in Figure 3.9-10.

The upper section of the barrel contains two outlet nozzles that interface with internal projections on the vessel nozzles to minimize leakage of coolant form inlet to outlet. Since the weight of the core support barrel is supported at its upper end, it is possible that coolant flow could induce vibrations in the structure. Therefore, amplitude limiting devices, or snubbers, are installed on the outside of the core support barrel near the bottom end. The snubbers consist of six equally-spaced lugs around the circumference of the barrel and act as a tongue-and-groove assembly with the mating lugs on the pressure vessel. Minimizing the clearance between the two mating pieces limits the amplitude of vibration. During assembly, as the internals are lowered into the pressure vessel, the pressure vessel lugs engage the core support barrel lugs in an axial direction. Radial and axial expansion of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted. The pressure vessel lugs have bolted, captured Inconel X shims. The core support barrel lug mating surfaces are hardfaced with Stellite to minimize wear. The shims are machined during initial installation to provide minimum clearance. The snubber assembly is shown in Figure 3.9-11.

3.9.5.1.1.2 Lower Support Structure and Instrument Nozzle Assembly

The lower support structure and ICI nozzle assembly position and support the fuel assemblies, core shroud, and ICI nozzles. The structure is a welded assembly consisting of a short cylinder, support beams, a bottom plate, ICI nozzles, and and ICI nozzle support plate. The lowest support structure is made up of a short cylindrical section enclosing an assemblage of grid beams arranged in egg-crate fashion. The outer ends of these beams are welded to the cylinder. Fuel assembly locating pins are attached to the beams. The bottoms of the parallel beams in one direction are welded to an array of plates which contain flow holes to provide proper flow distribution. These plates also provide support for the ICI nozzles and, through support columns, the ICI nozzle support plate. The cylinder guides the main coolant flow and limits the core shroud bypass flow by means of holes located near the base of the cylinder. The ICI nozzle support plate provides lateral support for the nozzles. This plate is provided

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TABLE 3.9-15

STRESS LIMITS FOR CEDM PRESSURE HOUSINGS

Operating Condition	Stress Categories and Limits of Stress Intensities (a)
1. Level A and Level B: Normal Operating Loading plus Normal Operating & Upset Plant Transients plus Operating Basis Earthquake Forces.	Figures NB-3221-1 and 3222-1, including notes.
2. Level D: Normal Operating Loadings plus Faulted Plant Transients plus Safe Shutdown Earthquake Forces, plus design bases pipe breaks.	Article F-1000, Appendix F, Rules for Evaluation of Service Conditions Loading with Level D Service Limits. (b)
3. Testing: Testing Plant Transients	Paragraph NB-3226

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For the above listed operating conditions, the following limits regarding function apply:

1. Level A and Level B: The CEDMs are designed to function normally during and after exposure to these conditions.

2. Level D: For SSE, the deflections of the CEDM pressure housing are limited to the elastic design limits of Article F-1330, Appendix F (defined above) so that the CEAs can be inserted after exposure to these conditions.

NOTE: a. References listed are taken from Section III of the ASME Boiler and Pressure Vessel Code.

b. Level D dynamic loads due to SSE and design bases pipe breaks are combined by the SRSS method in accordance with the guidelines of NUREG-0484.

Question 210.71

CESSAR-DC Section 3.9.4.2 states that the CEDM pressure boundary material complies, in part, with ASME Code Case N-411. Provide a commitment in this Section that all of the conditions in RG 1.84 relative to the use of this Code Case will be implemented.

Response 210.71

The materials used in the fabrication of the CEDM motor housing are provided in accordance with ASME Code Case N-4-11, special type 403 modified forgings or bars. Combustion Engineering is not aware of any restrictions or conditions imposed on use of Code Case N-4-11 by Regulatory Guide 1.84. Code case N-411-1, addressed in Reg. Guide 1.84 deals with alternative damping values for seismic response spectra. (See CESSAR-DC Section 3.7 for additional information on seismic analysis).

Question 210.72

CESSAR-DC Section 3.9.4.2 states that the CEDM pressure boundary components are designed and fabricated in accordance with the applicable edition of the ASME Code requirements for Class 1 vessels.

Modify the subsection to be in accordance with the criteria of SRP 3.9.4, Rev. 2, Subsection II.2 including "construction" criteria and acceptable codes and standards. In addition, specify the edition of the ASME Code.

Response 210.72

Section 3.9.4.2 will be revised in a future amendment to CESSAR-DC to reflect design and fabrication in accordance with the criteria of SRP 3.9.4, Rev. 2, Subsection II.2.

To allow for flexibility in implementing future ASME Code revisions, Combustion Engineering believes that it is not appropriate to identify specific ASME Code addenda in CESSAR-DC. Code addenda requirements for System 80+ plants will comply with the requirements of 10 CFR 50.55a.

Question 210.73

CESSAR-DC Section 3.9.4.2 states that the CEDM reed switch position transmitter assembly is designed to comply with the IEEE 323-1974 and IEEE 344-1975 standards.

IEEE 323-1974 should be IEEE 323-1983 and IEEE-1975 should be IEEE 344-1987 as endorsed by RG 1.100, Rev. 2 (reference RAI 210.61).

In addition, the input motion(s) for seismic qualification in accordance with the IEEE 344 standard should be clarified. The relationship between the inputs for the range of site types considered for the standard plant design and the IEEE 344 test input(s) should be explained.

Response 210.73

See responses to RAI 210.61 and RAI 210.85.

Question 210.76

CESSAR-DC Table 3.9-16, "Stress Limits for Design and Service Loads," should indicate that the limits are applicable to ASME Code CS components and reactor vessel internals.

In addition, Table 3.9-16 states that core support structures or core support and internal structures shall be "designed" to ASME Code, Section III, Subsection NG design limits.

Revise the table to indicate that: 1) core support structures shall be "constructed" to ASME Code, Section III, Subsection NG requirements where "construction" is as defined in the ASME Code, Section III, NB/NC/ND-1100(a); and 2) reactor internals, other than core support structures, shall meet the guidelines of NG-3000 of the ASME Code and be constructed so as to not adversely affect the integrity of the core support structures (NG-1122).

Response 210.76

Table 3.9-16 will be revised accordingly in a future amendment to CESSAR-DC.

RAI 210.76 & 210.77

Level C Service Limits

The core support and internal structures shall be designed to meet the Level C Service Limits defined in NG-3224 of IBID for Level C Service Loadings.

TABLE 3.9-16

STRESS LIMITS FOR DESIGN AND SERVICE LOADS

Design Limits

The core support and internal structures shall be designed to meet the Design Limits defined in NG-3221 of ASME Boiler and Pressure Vessel Code Section III Subsection NG for Design Loadings. BOTH STRUCTURES ARE SAFETY CLASS 3, SEISMIC CATEGORY I AND QUALITY CLASS I IN ACCORDANCE WITH ANSI/ANS-51.1-1983

Level A Service Limits

The core support and internal structures shall be designed to meet the Level A Service Limits defined in NG-3222 of IB1D for Level A Service Loadings.

Level B Service Limits

The core support and internal structures shall be designed to meet the Level B Service Limits defined in NG-3223 of IB1D for Level B Service Loadings.

Level D Service Limits

The core support structures shall be designed to meet the Level D Service Limits defined in NG-3225 of IB1D for elastic system analysis of Appendix F of Reference 3.1.2 using Level D Service Loadings. MAXIMUM STRESS INTENSITY WILL BE OBTAINED FROM PRINCIPAL STRESSES RESULTING FROM AN SRSS COMBINATION OF LOCA & SSE LOADINGS PLUS NOR. OPERATING LOADS IN ACCORDANCE WITH NUREG-0484 REV. 01

ADD NEW PARAGRAPH:

CORE SUPPORT STRUCTURES SHALL CONFORM TO ALL THE RULES OF CONSTRUCTION IN ACCORDANCE WITH ASME CODE SECTION III SUB-SECTION NG. REACTOR INTERNALS OTHER THAN CORE SUPPORT STRUCTURES SHALL MEET THE GUIDELINES OF NG-3000 AND BE CONSTRUCTED SO AS NOT TO ADVERSLY AFFECT THE INTEGRITY OF THE CORE SUPPORT STRUCTURES

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Question 210.77

In CESSAR-DC Table 3.9-16, CE should indicate that dynamic loads will be combined by the SRSS method in accordance with the guidelines of NUREG-0484, Rev. 1, 1980.

Response 210.77

Loads associated with SSE + DBPB are combined in accordance with NUREG-0484. This will be identified in a future amendment to CESSAR-DC.

Question 210.78

SRP 3.9.5, Rev. 2, Subsection II.d, requires that deformation limits for reactor internals should be established and the basis for the limits provided. In addition, this subsection requires that stresses associated with these displacements should not exceed the specified limits.

Provide the deformation limits and their basis, and associated stress data to demonstrate compliance with the specified limits.

Response 210.78

The deformation limits, and the basis for their selection, that is allowable for reactor internals are provided in Section 3.9.5.4.

The associated stress data to demonstrate compliance is a product of analytical computations which are developed during the detailed component design of the reactor internals. The results are compiled in detailed design calculations and those which are required are presented in the ASME code design report.

Question 210.79:

The information in Section 3.9.6 infers that only ASME Class 1, 2 and 3 pumps and valves will be included in the inservice testing (IST) program for the System 80+. Section 3.9.6.1, "Inservice Testing of Pumps," further limits testing to certain Code Class 2 and 3 pumps. It is the staff's position as stated in Standard Review Plan, Sections 3.9.6.II.1 and 3.9.6.II.2 that all pumps and valves which are considered as safety-related should be included in the IST program even if they are not categorized as ASME Class 1, 2 or 3. Revise Sections 3.9.6, 3.9.6.1 and 3.9.6.2 to agree with this position.

Response 210.79:

Section 3.2.2 of CESSAR-DC gives the safety classification of major components and their equivalence with classification in ANSI/ANS 51.1 and the ASME Code. All safety-related pumps and valves should be categorized as ASME Class 1, 2 or 3. For explicitness, however, sections 3.9.6, 3.9.6.1 and 3.9.6.2 will be revised to read as shown below in a future amendment to CESSAR-DC.

3.9.6 IN-SERVICE TESTING OF PUMPS AND VALVES

The in-service testing program for safety-related active pumps and valves will be developed in accordance with the requirements of Section XI of the ASME B&PV Code. This program will be implemented to assess operational readiness during preservice and in-service inspection.

3.9.6.1 In-service Testing of Pumps

In-service testing will include all safety-related active pumps. The required hydraulic and mechanical parameters will be measured by the methods and with frequency prescribed in subsection IWP of ASME Section XI. The pump test plan and schedule applicability are included in Section 3.0 of the technical specifications.

3.9.6.2 In-service Testing of Valves

All safety-related active valves will be tested to the requirements of ASME B&PV Code Section XI, subsection IWV. The valve test procedure and schedule applicability are included in Section 3.0 of the technical specifications.

In the design of critical reactor vessel internals components which are subject to fatigue, the stress analysis is performed utilizing the design fatigue curve of Figure I-9-2 of Section III of the ASME Boiler and Pressure Vessel Code. A cumulative usage factor of less than one is used as the limiting criterion.

As indicated in the preceding sections, the stress and fatigue limits for reactor internals components are obtained from the ASME Code. Allowable deformation limits are established as 80% of the loss-of-function deflection limits. These limits provide adequate safety factors assuring that so long as calculated stresses, usage factors, or deformations do not exceed these limits, the design is conservative.

3.9.6 IN-SERVICE TESTING OF PUMPS AND VALVES

safety-related active

The in-service testing program for ~~Code Class 1, 2 and 3~~ pumps and valves will be developed in accordance with the requirements of Section XI of the ASME B&PV Code. This program will be implemented to assess operational readiness during preservice and in-service inspection.

3.9.6.1 In-service Testing of Pumps

safety-related active pumps

will ~~[illegible]~~ include all ~~which are considered to~~

In-service testing of pumps ~~is limited to those Code Class 2 and 3 pumps which are required to perform a specific function in shutting down a reactor or in mitigating the consequences of an accident, and that are provided with an emergency power source.~~ The required hydraulic and mechanical parameters will be measured by the methods and with frequency prescribed in Subsection IWP of ASME Section XI. The pump test plan and schedule applicability are included in Section 3.0 of the technical specifications.

E

active

3.9.6.2 In-service Testing of Valves

All safety-related valves, ~~regardless of [illegible]~~,

~~[illegible] be categorized in accordance with Subarticle [illegible] Code, Section XI. Valves~~

will be tested to the requirements of ASME B&PV Code Section XI, Subsection IWV. ~~for each valve category. The testing plan will not include those Code Class 1, 2 and 3 valves which are exempt from testing in accordance with Subarticle IWV-1200 of Section XI~~. The valve test procedure and schedule applicability are included in Section 3.0 of the technical specifications.

<u>Question 210.80</u>

In Section 3.9.6, CE stated that the IST for pumps and valves will be developed in accordance with the requirements of Section XI of the ASME B&PV Code. However, CE has not explicitly committed that provisions to accommodate the IST of applicable pumps and valves will be incorporated in the plant design.

All of the plants which have been licensed by NRC have been permitted to request relief from the ASME Section XI IST rules for pumps and valves. These pumps and valves are generally installed in systems in which it is impractical to meet the Section XI rules because of limitations in the system design which preclude testing without significant design changes. In other cases, the staff granted requests for relief because imposition of the Section XI rules would have resulted in hardships to the license without a compensating increase in the level of safety. The underlying reason for the regulation allowing these reliefs from the code was that the detailed system designs for all of these plants were completed prior to the time that the staff began to require the ASME Code Section XI rules.

A plant such as the CE System 80+, for which the final design is not complete, has sufficient lead time available to include provisions for this type of testing in the detailed design of applicable piping systems. Therefore, requests for relief from the applicable ASME Section XI testing rules for pumps and valves will not be granted for the System 80+. Revise Section 3.9.6 to provide a more explicit commitment that System 80+ systems will be designed to accommodate the applicable Code requirements for IST of pumps and valves. However, with regard to subsequent or future Code revisions to the applicable ASME Code for the System 80+ plant, requests for relief from certain updated Code requirements may still be submitted for staff review in accordance with 10 CFR 50.55a(g).

<u>Response 210.80</u>

System 80+ is an improved design in which provisions for compliance with ASME B&PV Code, Section XI, Subsection IWP and IWV requirements can be implemented without undue hardship. However, there are examples of required testing for the System 80+ design which are clearly impractical and would require relief from the ASME B&PV Code, Section XI, with alternate testing approved by the NRC.

One example is the full flow check valve testing of the safety injection accumulator tank outlet check valves. This is only possible with the RCS depressurized, and such a test

Response 210.80 (cont'd)

would introduce the danger of injecting nitrogen into the RCS, which could lead to shutdown cooling pump cavitation and failure, resulting in RCS heatup. A partial flow stroke test is, however, possible in a non-operating, RCS-pressurized, plant condition. In coordination with nonintrusive check valve testing, full stroke of the check valves could be illustrated at partial flow conditions.

Another example is the containment spray header check valves. It is impossible to full flow stroke such valves in the System 80+ without spraying the containment. An alternate form of testing would be to provide check valves with an exterior means of manually full stroking the check valve without flow present.

A third example to be considered would be isolation valves for the Shutdown Cooling System drop lines. These valves constitute the interface between the RCS and the Shutdown Cooling System. Quarterly stroke tests of these valves while the unit is operating risks inadvertent opening of the valves and a resultant interfacing system LOCA. These valves would be recommended for stroke testing as Cold Shutdown valves, based on their unique situation.

Since these three examples constitute departure from the ASME Code requirements, relief requests would be necessary. Therefore, it is not practical to categorically state that no relief requests will be submitted. Clarification will be added to CESSAR-DC, Section 3.9.6 as follows:

> "The inservice testing program in coordination with the System 80+ design will utilize provisions and features such that minimal departure from ASME B&PV Code, Section XI, Subsections IWP and IWV requirements result."

The ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code 1990), Subsections ISTB and ISTC are currently under development to provide a revised version of Sections IWP and IWV respectively, regarding the need for instituted flexibility in component testing. If adopted and sanctioned by the NRC, these two subsections could eliminate several types of relief requests which have been required in the past for current operating plants.

In the design of critical reactor vessel internals components which are subject to fatigue, the stress analysis is performed utilizing the design fatigue curve of Figure I-9-2 of Section III of the ASME Boiler and Pressure Vessel Code. A cumulative usage factor of less than one is used as the limiting criterion.

As indicated in the preceding sections, the stress and fatigue limits for reactor internals components are obtained from the ASME Code. Allowable deformation limits are established as 80% of the loss-of-function deflection limits. These limits provide adequate safety factors assuring that so long as calculated stresses, usage factors, or deformations do not exceed these limits, the design is conservative.

3.9.6 IN-SERVICE TESTING OF PUMPS AND VALVES

The in-service testing program for Code Class 1, 2 and 3 pumps and valves will be developed in accordance with the requirements of Section XI of the ASME B&PV Code. This program will be implemented to assess operational readiness during preservice and in-service inspection. Insert A

3.9.6.1 In-service Testing of Pumps

In-service testing of pumps is limited to those Code Class 2 and 3 pumps which are required to perform a specific function in shutting down a reactor or in mitigating the consequences of an accident, and that are provided with an emergency power source. The required hydraulic and mechanical parameters will be measured by the methods and with frequency prescribed in Subsection IWP of ASME Section XI. The pump test plan and schedule are included in the technical specifications.

3.9.6.2 In-service Testing of Valves

Code Class 1, 2 and 3 valves will be categorized in accordance with Subarticle IWV-2100 of ASME B&PV Code Section XI. Valves will be tested to the requirements of Subsection IWV for each valve category. The testing plan will not include those Code Class 1, 2 and 3 valves which are exempt from testing in accordance with Subarticle IWV-1200 of Section XI. The valve test procedure and schedule are included in the technical specifications.

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Insert A:

"The inservice testing program in coordination with the System 80+ design will utilize provisions and features such that minimal departure from ASME B&PV Code, Section XI, Subsections IWP and IWV requirements result."

Question 210.81

The staff has also determined that the requirements in Section XI must be supplemented to obtain the level of assurance of operability desired for all of the advanced reactors. The staff requests that in Section 3.9.6 the applicant provide a commitment to design and test the applicable components in the System 80+ as discussed below (reference SECY-90-016).

a. For the reasons discussed above, the System 80+ should be designed to accommodate the applicable Code requirements for quarterly testing of pumps and valves, rather than be designed to accommodate testing that can only be performed during cold shutdowns or refueling outages.

b. For the System 80+ design, the pumps should be provided with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation.

c. For the System 80+ design, system configurations are to be provided to accommodate inservice testing at a flow rate of at least as large as the maximum design flow for the pump. In addition, system designs shall be such that the pumps do not operate below the minimum flow required for pump protection for all operating modes. Minimum recirculation flow lines shall be sized to ensure that degradation will not result from continuous mini-flow operation. The design of the minimum flow lines shall permit periodic testing to verify the flow is in accordance with design. (Reference Bulletin 88-04 and RAI 440.71)

d. For the System 80+ design, pumps and valves are to be periodically disassembled and inspected to determine if there are any indications of unacceptable corrosion or degradation. It is the staff's view that information derived from IST alone is not adequate to assess pump or valve condition and to determine required maintenance. The frequency of inspection and the extent of disassembly may vary depending upon the service condition of the pump and the valve. CE is requested to provide a commitment to periodically disassemble and inspect all pumps and valves important to safety. The staff requires, as a minimum, a commitment to develop a program that will establish the frequency and the extent of disassembly and inspection of all pumps and valves important to safety, including the basis for the frequency and the extent of each disassembly.

Question 210.81 (Cont'd)

e. With respect to check valves IST, the staff's position is that system designs are to incorporate provisions for full-flow testing to demonstrate the operability of the check valves under design conditions. In addition, the valve and system design should permit all check valves to be tested for performance in both forward and reverse flow directions and also allow for movement of the check valves obturator to be verified by observing a position indicator, by evidence obtained from direct instrumentation of the valves or system, or by non-intrusive test methods. The system and component design should assure that proper access and sufficient instrumentation and test connections are provided to allow for monitoring performance and trending degradation.

The staff's position on the use of non-intrusive diagnostic technique is that IST is to incorporate the use of advanced non-intrusive techniques to periodically assess degradation and the performance characteristics of the valves. The system and component design should assure that non-intrusive diagnostic methods can be accommodated.

f. System 80+ design should incorporate provisions to test power operated valves under design basis differential pressure and flow. CE is requested to provide more specific requirements for qualification testing and inservice testing to demonstrate design basis capability before installation, prior to startup, and throughout plant life. This testing is to be done in accordance with the forthcoming ASME/ANSI OM Part 18, "Performance Testing of Hydraulic Operated Valve Assemblies in LWR plants," and OM Part 19, "Performance Testing of Pneumatically Operated Valve Assemblies in LWR plants."

The method of assessing the loads, the method of sizing the actuators, and the setting of control parameters and switches should be specifically addressed. It should also include a study to determine optimal frequency for valve stroking during inservice testing so that unnecessary testing and damage is not done to the valve as a result of the testing.

Furthermore, the IST of power operated valves (POVs) is to rely on diagnostic techniques that are consistent with the state of the art, that are diagnostic of the condition of the valve, and that will permit an assessment of the performance of the valve under actual loading.

Question 210.81 (Cont'd)

g. System 80+ design should incorporate provisions to test motor operated valves (MOVs) under design basis differential pressure and flow. CE is requested to provide more specific requirements for qualification testing and inservice testing to demonstrate design basis capability before installation, prior to startup, and throughout plant life.

The concerns and issues identified in GL 89-10 and its supplements for MOVs should be addressed prior to plant startup. The method of assessing the loads, the method of sizing the actuators, and the setting of the torque and limit switches should be specifically addressed. It should also include a study to determine the optimal frequency for valve stroking during inservice testing so that unnecessary testing and damage is not done to the valve as a result of the testing.

Furthermore, the IST of MOVs is to rely on diagnostic techniques that are consistent with the state of the art, that are diagnostic of the condition of the valve, and that will permit an assessment of the performance of the valve under actual loading.

h. CE should provide a commitment to identify each valve relied upon in the safety analysis to provide a leak tight function and to specify the allowable leakage. The types of valves that are of particular concern to the staff are as follows:

1. Pressure isolation valves - valves that provide isolation of pressure differential from one part of a system to another or between systems. Several safety systems connected to the reactor coolant pressure boundary have design pressures below the rated reactor coolant system (RCS) pressure. Also some systems that are rated at full reactor pressure on the discharge side of pumps have pump suction below RCS pressure. To protect these systems from RCS pressure, two or more isolation valves are placed in service to form the interface between the high pressure RCS and the low pressure system. The staff's position on the leak testing of these pressure isolation valves is described in RAI 210.88 in Appendix A relating to GSI 105. (also reference RAI 440.45)

2. Temperature isolation valves - valves whose leakage may cause unacceptable thermal stress, fatigue, or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps.

Question 210.81 (Cont'd)

3. Containment isolation valves - valves that provide isolation capability for the piping systems penetrating containment. It should be noted that GDC 54 requires that all piping systems penetrating containment be provided with isolation capabilities and designed with a capability to periodically test the operability of the isolation valves and to determine if valve leakage is within acceptable limits. This requirement applies to primary and secondary systems penetrating containment.

These valves as described above must be designated as Category A and the systems and valves designed to accommodate Code leak testing. Applicants will be expected to demonstrate leak tightness of these valves prior to plant startup and throughout plant life.

Response 210.81

a. The System 80+ design incorporates provisions to allow the greatest practical capability to quarterly test required plant pumps and valves. For instance, full flow pump testing of Safety Injection Pumps has in the past been a plant shutdown evolution. The System 80+ Safety Injection System incorporates full flow pump testing capability while the unit is at power by providing a full flow bypass test line. There are instances in which quarterly testing would present hazards to personnel and plant safety, such as quarterly testing of Reactor Coolant System pressure isolation valves while the unit is in a non-cold shutdown condition. A commitment will be added to Section 3.9.6 in a future amendment to reflect this philosophy of the intent of the System 80+ design.

b. The pumps will be provided with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation. Section 3.9.6 will be revised in a future amendment to reflect this commitment.

c. The System 80+ design system configurations accommodate full flow pump testing during plant operation. Miniflow conditions, which in the past have been noted by industry and regulatory sources to cause pump degradation, occur primarily during testing modes of safety-related pumps. The need to operate and test these pumps at miniflow conditions is a result of the inability of prior designs to perform full flow pump testing, and thus plant operators have relied on extended run time at miniflow or low flow conditions in order to obtain pump test data.

Response 210.81 (Cont'd)

The best and most meaningful test point on the pump head curve is the Best Efficiency Point (BEP), and not the miniflow point. Therefore pump testing, except for verification of miniflow, does not require operating the pumps at miniflow for extended periods of time. As a fundamental goal of the System 80+ design, pumps will not be intentionally designed to operate for extended periods at miniflow conditions for normal operation. All miniflow lines are provided with appropriate flow instrumentation to verify the presence of miniflow.

d. The System 80+ design incorporates and commits to the following provisions regarding inservice testing and inspection/disassembly of ASME Class 2 and 3 components. These provisions increase the credibility of using the ASME Code as the primary means for detecting and programmatically rectifying component degradation.

- The System 80+ design provides the capability to perform full flow testing of ASME Code tested pumps, which provides more meaningful data than miniflow point testing.

- An Operational Support Information (OSI) Maintenance Plan which will be prepared to include the trending of all tested component parameters to detect degradation/component wear. The program will also provide a link between trend results and plant engineers responsible for component operability and reliability so that problems may be identified and further analysis may be instituted. The frequency of disassembly/inspection of components important to safety will be outlined in the Maintenance Plan based on the following criteria:

  -Historical performance of the component to identify components which are prone to degradation/wear.

  -Results of trends of test parameters.

  -Analysis of component makeup and whether any parts are subject to aging, such as "O-Rings.", etc.

  Components not considered to be subject to periodic disassembly/inspection may be added to a modified Maintenance Plan/inspection program on a case by case basis based on test data analysis and evaluation.

Response 210.81 (Cont'd)

This position is based upon:

- The ASME Code, Section XI, given the above programmatic refinements committed to in CESSAR-DC, Section 3.9.6, provides a sound basis for operability evaluation and determination of system and component problems and provides guidance in their resolution.

- The burden of an extensive maintenance program as described in the NRC recommendation will result in:

  - Increased radiation exposure to personnel conducting the maintenance and disassembly/inspection of the components without a corresponding increase in reliability.

  - Increased failures due to human error in more frequent disassembly and reassembly.

  - Significant impact on outage time--the plant condition during which the majority of such extensive maintenance may be performed. Action statements in Tecnical Specifications should not be entered for disassembly/inspection of components, unless warranted by test results.

  - Corresponding increase in post-maintenance testing following disassembly/inspection, which will impact outage length and complexity.

  - Inspections being insufficient of themselves, for example, there are cases in which a diagnostic non-intrusive test is vital before the inspection/maintenance effort which follows. Certain important details may be overlooked during a routine inspection, whatever frequency it is based on, which only diagnostic Code testing will alert personnel to look for. Code testing and inspection/maintenance must therefore work together as one effort. This is the intent of the method of the ASME Code: to monitor component vital parameters and to initiate corrective action <u>before</u> a serious problem exists.

Response 210.81 (Cont'd)

e. To the greatest extent practical, the System 80+ design incorporates provisions to allow full flow testing of check valves important to safety to demonstrate valve operability under design conditions. Where plant safety is put at risk in order to perform a full flow check valve stroke test, as in the case of the Safety Injection Tank outlet check valves, where a full flow condition induces the danger of injecting nitrogen into the RCS, a partial flow test shall be substituted in conjunction with appropriate non-intrusive diagnostic testing to verify the position of the check valve obturator. Such departures from accepted Code testing are submitted for NRC approval via relief requests.

   Reverse flow testing to prove whether a check valve seats properly is accomplished by Appendix J and other appropriate leak testing.

   Non-intrusive diagnostic check valve testing technologies are employed where appropriate to effect prudent Code and regulatory required testing.

f. The System 80+ design effort will evaluate on a case by case basis to see if testing of power operated valves under design basis differential pressure is achievable from the standpoint of plant and personnel safety, engineering design practicality, and cost versus derived benefit.

   The method of assessing the loads, the method of sizing the actuators, and the setting of control parameters and switches procurement information require detailed design and not available at time of design certification. Additionally, valve procurement information as well as vendor participation in determining optimal frequency must be addressed on a case by case basis.

   Diagnostic techniques and the criteria for their programmatic usage must be evaluated from a regulatory and functional standpoint. Agreement as to the components requiring such testing, the basis for such requirements, and the cost versus derived benefit are a part of the detailed design process. A benefit must exist to implement design features in the procured valves which will enable the performance of the diagnostic techniques.

g. The System 80+ design effort will evaluate on a case by case basis to see if testing of motor operated valves under design basis differential pressure is achievable from the standpoint of plant and personnel safety,

Response 210.81 (Cont'd)

engineering design practicality, and cost vers s derived benefit.

The method of assessing the loads, the method of sizing the actuators, and the setting of control parameters and switches procurement information require detailed design and not available at time of design certification.

h. The request outlines the programmatic identification of valves relied upon in the safety analysis to provide a leak tight function and to specify the allowable leakage. Combustion Engineering certain detailed valve procurement information is required before leak rates of individual valves may be ascertained. C-E agrees that the valves eventually designated Category A by the commitment to this NRC request will be tested for leak tightness and designed to accommodate appropriate ASME Code/Appendix J testing. Addressing each of the NRC requested items specifically:

   1. Pressure isolation valves - valves that provide isolation of pressure differential from one part of a system to another or between systems. These valves and their required testing will be outlined in the IST program, in accordance with the response of NRC RAI 210.88.

   2. Temperature isolation valves - valves whose leakage may cause unacceptable thermal stress, fatigue, or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps. These valves will be identified by later detailed design as information regarding pipe supports and piping analysis is developed.

   3. Containment isolation valves - valves that provide isolation capability for the piping systems penetrating containment. These valves have been identified with their required testing presented in CESSAR-DC, Table 6.2.4-1.

CESSAR-DC Section 3.9.6.2 will be revised in a future amendment to provide this commitment.

In the design of critical reactor vessel internals components which are subject to fatigue, the stress analysis is performed utilizing the design fatigue curve of Figure I-9-2 of Section III of the ASME Boiler and Pressure Vessel Code. A cumulative usage factor of less than one is used as the limiting criterion.

As indicated in the preceding sections, the stress and fatigue limits for reactor internals components are obtained from the ASME Code. Allowable deformation limits are established as 80% of the loss-of-function deflection limits. These limits provide adequate safety factors assuring that so long as calculated stresses, usage factors, or deformations do not exceed these limits, the design is conservative.

3.9.6 IN-SERVICE TESTING OF PUMPS AND VALVES

The in-service testing program for Code Class 1, 2 and 3 pumps and valves will be developed in accordance with the requirements of Section XI of the ASME B&PV Code. This program will be implemented to assess operational readiness during preservice and in-service inspection.

3.9.6.1 In-service Testing of Pumps

In-service testing of pumps is limited to those Code Class 2 and 3 pumps which are required to perform a specific function in shutting down a reactor or in mitigating the consequences of an accident, and that are provided with an emergency power source. The required hydraulic and mechanical parameters will be measured by the methods and with frequency prescribed in Subsection IWP of ASME Section XI. The pump test plan and schedule are included in the technical specifications. E

Insert A

3.9.6.2 In-service Testing of Valves

Code Class 1, 2 and 3 valves will be categorized in accordance with Subarticle IWV-2100 of ASME B&PV Code Section XI. Valves will be tested to the requirements of Subsection IWV for each valve category. The testing plan will not include those Code Class 1, 2 and 3 valves which are exempt from testing in accordance with Subarticle IWV-1200 of Section XI. The valve test procedure and schedule are included in the technical specifications.

Insert B

RAI 210.81

Insert A:

In addition to Section XI of the ASME B & PV Code, the following provisions will be included as a part of the pump test plan for the pumps specified above:

A. Full flow testing of these Class 2 and 3 pumps on a quarterly basis.

B. Pump suction pressure while pump is operating will be a standard test parameter in addition to static suction pressure (pump shut down).

C. Guidance for ensuring minimal pump miniflow operation while testing.

D. A Pump Maintenance Plan which will ensure the trending of all safety related pump test parameters. This program will also provide a link between trended results and plant engineers responsible for pump operability and reliability so that problems may be identified and further analysis may be instituted. The Plan will also establish a pump disassembly/inspection program based upon:

1. Historical performance of the pump to identify pumps which are prone to degradation/wear.

2. Analysis of trends of pump test parameters.

3. Analysis of pump components, such as "O-Rings," which are subject to aging.

Insert B:

RAI 210.81

In addition to Section XI of the ASME B & PV Code, the following provisions will be included as a part of the valve test plan for the valves specified above:

A. Determination of the optimal test frequency of valves from a regulatory, design, vendor and engineering practicality standpoint.

B. Programmatic use of appropriate non-intrusive diagnostic check valve testing technologies.

C. For those valves which must operate under differential pressure to perform their safety function, tests are to be performed on an appropriate schedule in a manner which best replicates the postulated differential pressure.

D. Categorization and appropriate testing of the following classes of Category A valves:

   1. Pressure isolation valves - valves that provide isolation of a pressure differential from one part of a system to another or between systems.

   2. Temperature isolation valves - valves whose leakage may cause unacceptable thermal stress, fatigue, or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding on pumps.

   3. Containment isolation valves - valves that provide isolation capability for the piping systems penetrating containment.

E. A Valve Maintenance Plan which will ensure the trending of all safety related valve test parameters. This program will also provide a link between trended results and plant engineers responsible for valve operability and reliability so that problems may be identified and further analysis may be instituted. The Plan will also establish a valve disassembly/inspection program based upon:

   1. Historical performance of the valve to identify valves which are prone to degradation/wear.

   2. Analysis of trends of valve test parameters.

   3. Analysis of valve components, such as "O-Rings," which are subject to aging.

Question 210.82

CESSAR-DC Section 3.9.2.2, "Seismic Qualification Testing of Safety-Related Mechanical Equipment", addresses the seismic qualification program for both NSSS and non-NSSS safety-related mechanical equipment. Section 3.9.2.2.1 implies that the qualification program for such NSSS equipment, including supports, is a part of the operability demonstration program described in Section 3.9.3.2, "Pump and Valve Operability Assurance"; and Section 3.9.2.2.2 describes the qualification program for the non-NSSS equipment.

With respect to the scope of the NSSS mechanical equipment requiring seismic qualification, since Section 3.9.3.2 is limited only to pumps and valves, clarify whether or not, except for pumps and valves, there exists any other such equipment. If other such equipment exists, provide details of their seismic qualification. Moreover, the information regarding the seismic qualification provided in Section 3.9.3.2 are not in accordance with the criteria in SRP 3.10, Rev. 2. Provide details of the qualification program in accordance with SRP 3.10, Rev. 2.

Similarly, the description of the seismic qualification program for non-NSSS safety-related mechanical program in Section 3.9.2.2.2 are also not in accordance with SRP 3.10, Rev. 2 criteria and should be modified to be in accordance with these criteria.

Response 210.82

CESSAR-DC Section 3.10 is presently being revised to conform with the intent of the guidance provided by SRP Section 3.10, Revision 2 and USNRC Regulatory Guide 1.100, Revision 2. This revision, which will be included in submittal of a future amendment of CESSAR-DC, will address compliance with the acceptance criteria of applicable subsections of SRP 3.10 for both mechanical and electrical equipment. Necessary revisions, if any, to CESSAR-DC Section 3.9.2.2 in order to be consistent with Section 3.10 and address the concerns of this RAI will be included in this submittal.

Question 210.83

CESSAR-DC Section 3.10 describes the seismic qualification program for Seismic Category I instrumentation and electrical equipment except for valve and pump motors, and their supports. The descriptions of the qualification program in Sections 3.10.2 and 3.10.3 for the equipment and their supports, respectively, do not provide an adequate basis for determining the acceptability of the program.

Modify the descriptions of the qualification program provided in Section 3.10 to specifically address compliance with the acceptance criteria of SRP 3.10, Subsections II.1 through II.5.

Response 210.83

CESSAR-DC Section 3.10 is presently being revised to conform with the intent of the guidance provided by SRP Section 3.10, Revision 2 and USNRC Regulatory Guide 1.100, Revision 2. This revision, which will be included in a future amendment of CESSAR-DC, will address compliance with the acceptance criteria of applicable subsections of SRP 3.10. For instance, SRP 3.10 Subsection II.2 is not applicable since it pertains only to plants for which the CP application was docketed before October 27, 1972.

Question 210.84

CESSAR-DC Section 3.10.2, Item c, states in part that operating experience will be required to substantiate the adequacy of the design of Seismic Category I instrumentation and electrical equipment.

Explain and justify the use of operating experience for the substantiation.

Response 210.84

CESSAR-DC Section 3.10.2, Item C. states that "Analysis, testing or operating experience will be required to substantiate the adequacy of the design ...." and not that, in part, operating experience will be required to substantiate the adequacy of the design of Seismic Category I instrumentation and electrical equipment. Equipment for System 80+ will generally be qualified by test, analysis or a combination thereof.

However, there may be circumstances where use of experience data, which includes operating experience, may be the most practical means of equipment qualification. When qualification of equipment through use of experience data is used, it will be in accordance with IEEE Standard 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations". This will include careful evaluation of the type, size, shape and complexity of equipment, safety and/or structural functions of the equipment and the sources and types of experience data available. Justification of similarity of equipment type with previously qualified equipment or with equipment that has been exposed to other more severe environments will be established.

CESSAR-DC Section 3.10 is currently being revised to conform to the current SRP Section 3.10 and USNRC Regulatory Guide 1.100, Revision 2. This revision will identify use of "experience data" as a qualification alternative rather than "operating experience".

Question 210.85

CESSAR-DC Section 3.10.2, Item F, and elsewhere, and the IEEE 344-1987 standard requires that the input motion for seismic qualification of equipment be specified by a response spectrum, power spectral density function or time history. This item also states that the input be representative of the equipment mounting locations.

Since the standard design is to be qualified for placement at a variety of hypothetical site types, clarify the input motion(s) to be used in the qualification program and their relationship to the motions to be expected at a particular site.

Response 210.85

The input motions to be used in qualification of System 80+ equipment are derived from the structural responses obtained from the system analysis performed for each of the generic site conditions selected for geologic and seismologic evaluation. For each item of equipment to be qualified, input will normally envelope corresponding data for all locations within the plant where that item may be located and for all of the generic soil sites evaluated. However, qualification to this conservative envelope could, in some cases, require unjustified design changes and/or have unjustified cost impacts. An alternative qualification procedure which may be used is to separately qualify equipment for each of the cases which comprise the generic envelope.

The equipment mounting locations referred to in CESSAR-DC 3.10, Item F refers to locations within the building structures and not to plant siting locations. CESSAR-DC Section 3.10 is presently being revised to conform with the guidance provided by the current revision of SRP Section 3.10. This revision, which will be included in a future amendment of CESSAR-DC, will clarify that equipment mounting locations refers to locations within the building structure.

The input motions used for equipment qualification are considered to be conservative compared to the motions expected at any particular plant site meeting the acceptance criteria defined in CESSAR-DC, Section 2.5.

Question 210.86

SRP 3.10, Section II.3 states that GDC 1 of Appendix A and Paragraph XVII of Appendix B to 10 CFR Part 50 establish requirements for records concerning the qualification of equipment. In order to satisfy these requirements, complete records must be documented and maintained. The staff audits the results of tests and analysis to assure that adequate qualification has been demonstrated for all equipment and their supports, and to verify that all applicable loading has been properly defined. Discuss to what extent equipment test results and analyses will be available to the staff for such audits to demonstrate compliance with SRP 3.10 prior to design certification.

Response 210.86

Equipment qualification is vendor and model specific, therefore, specific models and characteristics of equipment will not be available until the time of equipment procurement. Equipment qualification, including required documentation, will be completed subsequent to certification in accordance with established regulations and standards.

Question 210.88

For GSI-105, the staff position is that, for minimizing the likelihood of occurrence of an interfacing system LOCA event, all valves which serve the pressure isolation function to preclude such an event shall verify leak tight integrity. Provide your commitment to perform preservice and periodic inservice leak testing of all safety related pressure isolation valves. Leak tests in accordance with the applicable section of the Technical Specifications for recently licensed CE System 80 plant (i.e. Palo Verde) are acceptable. Your information shall include a description of the surveillance and testing program and a list of all safety related pressure isolation valves which will be leak tested. (also reference RAI's 440.45 & 280.79 (h).

Response 210.88

Preservice and Periodic Inservice Leak Testing

In CESSAR-DC, Appendix A, GSI-105 addresses the "Interfacing Systems LOCA At LWRS." This RAI addresses preservice and inservice valve leak testing of pressure isolation valves.

The System 80+ valve leak test program is developed for preservice and inservice valve testing in accordance with Regulatory Guide 1.68 Initial Test Programs for Water-Cooled Nuclear Power Plants, ASME Boiler and Pressure Vessel Code Section XI, and ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants. Valves are categorized, leakage tested, and exercised in accordance with the test requirements specified for each valve type.

In addition, CESSAR-DC - Chapter 16 provides the proposed plant technical specifications. Technical Specification 3.4.13 addresses RCS Pressure Isolation Valve (PIV) Leakage and defines the frequency, mode change requirements, etc., for testing the PIV's. These requirements are similar to the test requirements specified for the PIV leak tests in the technical specifications for Palo Verde.

In order to clarify this issue for System 80+, GSI-105 in CESSAR-DC, Appendix A will be revised to include a statement requiring preservice and inservice valve leak testing in accordance with Regulatory Guide 1.68, ASME B&PV Code Section XI, ASME OM Code, and the proposed plant technical specifications. A new paragraph will be added which reads:

> In order to identify potential leakage paths, the pressure isolation valves are tested during the preservice and inservice valve test program in accordance with the requirements specified by RG 1.68, ASME B&PV Code Section XI, ASME OM Code, and the proposed plant technical specifications (see CESSAR-DC, Chapter 16 - Section 3.4.13, RCS Pressure Isolation Valve (PIV) Leakage).

See the response to RAI 210.81 (h) which addresses PIV leak testing and RAI 440.45 which addresses the intersystem LOCA.

Description of Surveillance and Test Programs

The PIV's included in the System 80+ test program are tested to meet the valve test program specified by RG 1.68, ASME Code Section XI, and ASME OM Code. In addition, the PIV's listed in the table below will be leak tested in accordance with the proposed plant technical specifications (see CESSAR-DC, Chapter 16 - Section 3.4.13).

Pressure Isolation Valves

The System 80+ pressure isolation valves subject to leak testing include the valves listed below. The listing is consistent with the list of PIV's leak tested at Palo Verde.

	Valve	Description
1.	SI-237	LOOP 2A RC/SI CHECK
2.	SI-247	LOOP 1A RC/SI CHECK
3.	SI-217	LOOP 2B RC/SI CHECK
4.	SI-227	LOOP 1B RC/SI CHECK
5.	SI-235	LOOP 2A SIT CHECK
6.	SI-245	LOOP 1A SIT CHECK
7.	SI-215	LOOP 2B SIT CHECK
8.	SI-225	LOOP 1B SIT CHECK
9.	SI-542	LOOP 2A HEADER CHECK
10.	SI-543	LOOP 1A HEADER CHECK
11.	SI-540	LOOP 2B HEADER CHECK
12.	SI-541	LOOP 1B HEADER CHECK
13.	SI-522	LOOP 1 LONG TERM RECIRCULATION CHECK
14.	SI-523	LOOP 1 LONG TERM RECIRCULATION CHECK
15.	SI-532	LOOP 2 LONG TERM RECIRCULATION CHECK
16.	SI-533	LOOP 2 LONG TERM RECIRCULATION CHECK
17.	SI-651	LOOP 1 SHUTDOWN COOLING ISOLATION
18.	SI-652	LOOP 2 SHUTDOWN COOLING ISOLATION
19.	SI-653	LOOP 1 SHUTDOWN COOLING ISOLATION
20.	SI-654	LOOP 2 SHUTDOWN COOLING ISOLATION

RAI 210.88

(2) Charging pumps can make up lost inventory and allow plant cooldown,

(3) The break can only credibly occur inside containment, or

(4) The flow path contains normally open valves which can be closed to isolate the break.

The System 80+ PRA results show that ISLs provide only a minor contribution to core damage frequency (i.e., a contribution of approximately 3.0E-9 relative to the core damage frequency goal of 1.0E-5).

* Insert A goes here

In summary, the System 80+ Standard Design minimizes the likelihood of an interfacing system LOCA by (1) eliminating low pressure safety injection, one of the most likely leakage paths of previous designs, (2) increasing the over-pressure capability of the SCS, the next most likely leakage path, (3) increasing the design pressure of other components (e.g., the letdown heat exchanger), and (4) improving the location of components to reduce ISL probability. This conclusion is supported by the overall plant PRA, which shows that the contribution of ISL to the overall core damage frequency is insignificant. This issue is therefore resolved for the System 80+ Standard Design.

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues", U.S. Nuclear Regulatory Commission, April 1989. G

2. NUREG/CR-5102, "Interfacing Systems LOCA: Pressurized Water Reactors", U.S. Nuclear Regulatory Commission.

3. 10 CFR, Part 52, "Early Site Permits; Standard Design Certifications", Code Of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

RAI 210.88

INSERT A

In order to identify potential leakage paths, the pressure isolation valves are tested during the preservice and inservice valve test program in accordance with the requirements specified by RG 1.68, ASME B&PV Code Section XI, ASME OM Code, and the proposed plant technical specifications (see CESSAR-DC, Chapter 16 - Section 3.4.13, RCS Pressure Isolation Valve (PIV) Leakage).

Question 210.94

Resolution of GSI C-12, concerning primary system vibration assessment, indicating that experience from the startup of the System 80 plants at Palo Verde is included in System 80+ Standard Design. Provide an explanation of how such experience is utilized and verify that calculations and considerations associated with such experience are included in applicable design documents. In addition, provide a more detailed description of the Integrity Monitoring System and verify that details of the monitoring system, including a list of numbers, types and locations of sensors to be used are documented. If such a list is not available, a commitment to provide such a list as a part of the System 80+ Technical Specifications will be acceptable.

Response 210.94

Design modifications incorporated into the Palo Verde System 80 design as a result of system vibrations experienced during startup have also been implemented into the System 80+ design. The following design areas were affected:

° CEA Shroud
° RCS Temperature Detector Thermowells
° Safety Injection Nozzles
° LPSI Pump

The design features and characteristics for other System 80 systems and components have been proven during startup and successful operation of the Palo Verde units and are also included in the System 80+ design.

The System 80+ Integrity Monitoring system is described in CESSAR-DC Section 7.7.1.6 and the number and location of sensors are given in Tables 7.7-3 and 7.7-4.

QUESTION 210.95

To be acceptable, the discussion of GSI II.D.1, "Performance Testing of PWR Safety and Relief Valves" in Appendix A requires the following additional commitments:

1. To be consistent with the staff's position in NUREG-0737, the qualification of discharge piping by analysis should be included in addition to valves.

2. It is not clear whether the rapid depressurization, gas vent, and associated isolation valves described in CESSAR-DC Section 6.7 are included in these evaluations. The staff's position is that all of these valves should be qualified in a manner consistent with the II.D.1 requirements in NUREG-0737. These valves and associated discharge piping should be qualified by testing for the applicable design fluid inlet conditions including steam, liquid, two-phase, and entrained gas conditions.

3. Provide a commitment that all applicable valves and associated discharge piping in the CESSAR-DC System 80+ Standard Design which are not similar to those which were included in the EPRI Safety Valve Test Program and documented in CEN-227 will be tested in accordance with NURFG-0737 requirements.

Revise the II.D.1 resolution in CESSAR-DC Appendix A to be consistent with the staff positions discussed in 1,2 and 3 above.

RESPONSE 210.95

Generic Safety Issue (GSI) II.D.1 and the corresponding Section in NUREG 0737 address the performance testing of pressurized-water reactor relief and safety valves and are not applicable to the Reactor Coolant Gas Vent System valves and the Rapid Depressurization valves.

Item 1:

The RAI implies that the analysis of the discharge piping may not be a requirement. Nevertheless, analyses will be completed subsequent to design certification when all material, components and detailed piping layouts are specified and become available. These analyses will confirm that the piping layout and consequential backpressure on the primary safety valves are within the range of the backpressures tested in the EPRI valve test program. In addition, these analyses will confirm that the piping loads do not adversely impact valve operability.

RAI 210.95

Items 2 & 3

The Reactor Coolant Gas Vent System (RCGVS) valves are solenoid operated valves and are used for design basis events. During power operation, if these valves were used for plant cooldown, steam would flow through these valves. During reactor coolant system fill, the RCGVS valves will experience liquid flow during the filling and venting process. The system temperature and pressure will be relatively low during these times (i.e., less than a 100 psia), but this venting is not required to support the acceptable performance of design bases events. This is similar to all high point vents in the plant. RCGVS valves used in System 80+ are similar to the valves used at Palo Verde which have been in operation for several years. Since these valves are remote manually operated valves and the reference addresses automatic actuated valves, this reference provides no requirement for testing of the RCGVS valves.

The Rapid Depressurization (RD) valves are used for the beyond design basis event of a Total Loss of Feedwater, and not for any of the design bases events. The requirements referenced in this RAI do not require valve testing for valves used for beyond design basis events. We believe the NRC's concern addresses the sizing of motor operators. We recognize that there are valve test programs currently underway to address these issues. Data from these programs will be considered in the design.

At the time of material and valve procurement, these tests, as well as others that may be conducted, will be considered in specifying the requirements to the valve vendors. It should be noted that the these valves are remote manually operated valves which are actuated by motor operators and are not automatically operated valves such as the primary safety valves discussed in GSI II.D.1.

Question 210.96

For GSI II.E.6.1 concerning all safety related valves, describe your program and provide a commitment to conduct surveillance and testing for verifying valve operability.

A. For motor operated valves (MOVs), the program should commit to implement the following:

   a. conduct surveillance and operability verification testing for each MOV per guidance provided in GL 89-10. The test should be conducted prior to installation, during plant startup, and periodically during plant operation life span.

   b. For thermal overload protection of electric motors on the MOVs, guidelines delineated in RG 1.106 shall be met.

B. For pressure isolation valves, see RAI's 210.88 in CESSAR-DC Appendix A and 210.81h.(1) in CESSAR-DC Section 3.9.6.

C. For check valves, failures and malfunctions have occurred frequently due to various causes, especially problems related to valve design, such as using wrong valve type or over-sized valve, improper valve orientation, and valve located too closely to turbulent flow (see EPRI Report NP-5479 dated January 1988). Your program should address resolution of these concerns for verifying check valve operability.

Response 210.96

The program requested above will be incorporated into the Operational Support Information (OSI) Program. One of the OSI program subcategories will address a maintenance plan which will provide Inservice Testing (IST) guides for safety related valves.

Question 220.0

General: In Table 2.0-1 and a few places in Section 25., there is an indication that there will be a site-specific SSE in plant-specific SARs. Provide a listing of structures, systems, and components (Table 3.2-1) that will be designed using the assumptions and methodology provided in the Design Certification documents, and those that will be designed using the site-specific SSE.

Response 220.0

All structures, systems, and components for System 80+ are designed using the assumptions and methodology provided in CESSAR-DC. There is no intent to provide either site-specific SSE analyses or results in site-specific Safety Analyses Reports (SARs).

Identified in Table 2.0-1 and Section 2.5 of CESSAR-DC is the commitment to provide in the site-specific SAR that information which can only be defined and evaluated once a specific site is selected. This information includes, but is not limited to, site geological features, seismological features, liquefication potential, earthquake activity history, boring data, site instability and ground rupture potential, and man-made conditions. Data and evaluations will be provided in the site-specific SAR to demonstrate that the site acceptance criteria defined in CESSAR-DC are satisfied and documented.

Question 220.1

Section 3.3.1 - The current Standard of Reference should be ASCE 7-88, rather than ANSI A58.1.

Response 220.1

The correct reference is ASCE 7-88, (formerly ANSI A58.1). Section 3.3.1 will be revised to incorporate this editorial change.

REFERENCES FOR SECTION 3.3

1. "Minimum Design Loads for Buildings and Other Structures," ~~ANSI A58.1-1982~~. ASCE 7-88 D

2. "Wind Forces on Structures," ASCE Paper No. 3269, Transactions, ASCE, Vol. 126, Part II, 1961, p. 1124.

3. "Wind Loads on Dome-Cylinder and Dome-Cone Shapes," ASCE Paper No. 4933, Journal of the Structural Division - Proceedings of the American Society of Civil Engineers, Vol. 92, No. ST5, October 1966.

4. Safety Evaluation by the Office of Nuclear Reactor Regulation of Recommended Modification to the R.G. 1.76 Tornado Design Basis for the ALWR, attached to a March 25, 1988 NRC letter to the ALWR Utility Steering Committee. I

Question 220.2

Section 3.4.1 - Provide explanation of the sentence, "The design basis level for the System 80+ Standard design is limited to 1 foot below plant finished yard grade as the minimum flood level value." Does this mean that the probable maximum flood (PMF) level will be 1 foot below the finished grade level?

Response 220.2

The standard plant is designed for a flood level at 1 foot below the finished grade level.

Section 3.4.1 will be revised to clarify this information.

APPENDIX A

CHARACTERISTICS OF GENERIC SOIL SITES

EARTHQUAKE
GROUND MOTIONS

SELECTION OF
CONTROL MOTION
AND
DEVELOPMENT OF
GENERIC SOIL SITES

DONE IN CONJUNCTION WITH
C-E/DOE ALWR CERTIFICATION

September 4, 1990

APPENDIX A

CHARACTERISTICS OF GENERIC SOIL SITES

This appendix presents the 4 generic site categories selected for this project and the maximum shear wave velocity profiles for the cases originally selected and for the cases selected after examining the first set of results.

Figure A-1 shows the general layout of each site category. Figures A-2 through A-13 show the maximum shear wave velocities for the twelve soil cases considered in this study.

3.4 WATER LEVEL (FLOOD) DESIGN

All Seismic Category I structures, components and equipment are designed for applicable loadings caused by postulated floods. Section 2.4 of the site-specific SAR describes, in detail, the relationship of the site-specific flood levels to safety-related buildings and facilities.

3.4.1 FLOOD ELEVATIONS

The elevation level for floods at the reactor site is determined in accordance with Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," and ANSI/ANS 2.8-1983, "Determining Design Basis Flooding at Power Reactor Sites." The design basis level for the System 80+ Standard Design is ~~limited to~~ 1 foot below plant finished yard grade. ~~as the minimum flood level value. The maximum~~ Flood level values ~~is~~ in excess of this 1 foot level are site-specific and protection measures for that flood level are described in Section 2.4 of the site-specific SAR.

3.4.2 PHENOMENA CONSIDERED IN DESIGN LOAD CALCULATION

All safety-related structures of the reactor building complex are designed to withstand the static and dynamic forces of the plant flood level. Other safety-related structures or systems essential for plant operation are designed for the site-related flood level as described in Section 2.4 of the site-specific SAR. D

3.4.3 FLOOD FORCE APPLICATION

The desi flood is used in determining the applicable water level for design of all Seismic Category I structures in accordance with the load combinations discussed in Section 3.8.4. The forces acting on those structures are determined on the basis of full external hydrostatic pressure corresponding to that flood level. All Seismic Category I structures will be in a stable condition due to both moment and uplift forces resulting from the proper load combinations, including design basis flood levels.

3.4.4 FLOOD PROTECTION

3.4.4.1 Flood Protection Measures for Seismic Category I Structures

The flood protection measures for Seismic Category I structures, systems and components are designed in accordance with Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants." The following structures and systems in the reactor complex area are designed for flood level protection:

Question 220.3

Sections 3.7.1.1 and 3.7.1.2 - The time histories discussed in Section 3.7.1.2 are not consistent with the spectra presented in Section 3.7.1.1. Design response spectra should have been presented in Section 3.7.1.1 if RG 1.70 were followed. Are the unsmoothed spectra shown in Figures 3.7-1 through 3.7-24 the design response spectra? How are these unsmoothed spectra used if they are the design response spectra for seismic input?

Response 220.3

The free field design ground motion or control motion is specified on a hypothetical rock outcrop in accordance with USNRC Standard Review Plan 3.7.1, Rev. 2, Section I.1. The design response spectra defined at this control point are shown in CESSAR-DC Figure 2.5-6. The time histories discussed in Section 3.7.1.2 are consistent with these spectra.

The unsmoothed response spectra shown in Figures 3.7-1 to 3.7-24 correspond to the free-field ground surface and foundation level spectra of each generic site. The time histories that produced these unsmoothed spectra were used as seismic control motions in the SSI analyses.

CESSAR-DC Section 3.7.1.1 will be revised to provide this clarification.

RAI 220.3

which correspond to the design response spectra defined at the hypothetical rock outcrop and the representative soil cases evaluated are

The design response spectra which define the free field design ground motion or control motion specified on a hypothetical rock outcrop are shown in Figure 2.5-6.

3.7 SEISMIC DESIGN

3.7.1 SEISMIC INPUT

3.7.1.1 ~~Seismic Input~~ Design Response Spectra

This section discusses the seismic design parameters and methodologies being used for the design of those systems and subsystems important to safety and classified as Category I in Section 3.2.

The System 80+ Standard Design as defined by CESSAR-DC is not based on a specific site. Generic site conditions were selected to cover a range of possible conditions for the System 80+ sites. More specifically, sets of representative cases from each of four generic site categories were evaluated. ~~to create the~~ ground surface and foundation level spectra shown in Figures 3.7-1 through 3.7-24. Out of 12 soil cases analyzed in Section 2.5.2, nine are used in the soil structure interaction (SSI) analysis. The three cases eliminated in the SSI analysis (A1, B3 and D1) were non-governing cases whose soil response levels were enveloped by other cases. See Section 2.5.2 for details of this analysis phase.

The effect of differential seismic displacement on the equipment and supports is included in the analysis as described in Section 3.7.2.1. I

3.7.1.2 Design Time History

For the time history method of analysis, three design time histories are generated that are consistent with the design rock outcrop spectra at the free field. The characteristics of each time history are presented in Section 2.5.2.5.1. The response spectra plots for these time histories are shown in Figures 3.7-25 through 3.7-27.

3.7.1.3 Critical Damping Values

Damping values used for various nuclear safety-related structures systems and components are based upon Regulatory Guide 1.61 or ASME Code Case N-411-1 (See Figure 3.7-41). These values are expressed in percent of critical damping and are given in Table 3.7-1. When the response spectra method of analysis is used for piping, damping values are based on Code Case N-411-1.

Question 220.4

Section 3.7.1.1 - Are the soil cases selected based on Reference 7? If the answer is positive, Reference 7 should be formally submitted to NRC for review. Otherwise, provide additional information regarding the sufficiency of these soil cases to cover the generic site conditions.

Response 220.4

Reference 7 is provided herein in a Draft form. The final version will be issued pending the resolution of the soil cases and ground motion RAIs.

220.4

DRAFT No. 2

EARTHQUAKE GROUND MOTIONS

SELECTION OF CONTROL MOTION AND DEVELOPMENT OF GENERIC SOIL SITES

DONE IN CONJUNCTION WITH C-E/DOE ALWR CERTIFICATION

For
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San Ramon, California 94583

by
I. M. Idriss
P. O. Box 330
Davis, California

September 4, 1990

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APPENDIX C - STRAIN-COMPATIBLE MODULUS AND DAMPING VALUES AND SPECTRAL ORDINATES CALCULATED FOR ALL CASES CONSIDERED

EARTHQUAKE GROUND MOTIONS

SELECTION OF CONTROL MOTION
AND
DEVELOPMENT OF GENERIC SOIL SITES
DONE IN CONJUNCTION WITH
C-E/DOE ALWR CERTIFICATION

1.0 INTRODUCTION

This report presents the results pertaining to the development of earthquake ground motions for use in the seismic design of Combustion Engineering System 80+. The results presented in this report were generated in conjunction with the C-E/DOE ALWR Certification.

To cover a range of possible site conditions where System 80+ may be constructed, a range of generic site conditions was selected. The total depth of each site category and the dynamic soil properties (in terms of maximum shear wave velocities and their variation with depth, and in terms of the variations of modulus and damping with strain) were established to cover a wide range of site conditions and to provide reasonably conservative results. Using these site conditions and the variations of maximum shear wave velocities, 13 cases were developed; 12 soil cases and one rock outcrop case. The cases selected are summarized in Section 2.0 and more details for each case are included in Appendix A of this report.

Each soil case was subjected to an earthquake ground motion which was developed to represent the motion at a free-field rock outcrop (the control motion). The smooth spectrum selected to represent this rock outcrop motion was developed taking into account the characteristics of earthquake ground motions recorded in Western North America and those judged applicable to Eastern North America. The smooth spectra for the two horizontal and the vertical components of this control motion were then developed. The selection of these smooth spectra is summarized in Section 3.0 and details of the synthetic time histories derived to represent the three components of this control motion are provided in Appendix B.

The response of each soil case to each component of the control motion was then obtained and the computed ground surface motion was provided to ABB Impell for their use in the soil-structure interaction studies of the entire system. The results of the response analyses are summarized in Section 4.0 and the spectral ordinates calculated for each case at the ground surface and at the foundation level in the free field are presented in Appendix C.

Section 5.0 summarizes recommended acceptance criteria.

2.0 GENERIC SOIL SITES

generic soil sites were selected by first choosing four generic site categories. These categories were chosen to represent appropriate total thickness of soil overlying bedrock. The four categories are shown schematically in Fig. 2-1. Site Category A consists of 52 ft of soil overlying bedrock; 52 ft is the embedment depth selected for the System 80+. The soils in site Category B extend to a depth of 100 ft and those in Categories C and D extend to depths of 200 and 300 ft, respectively.

One case was selected for Category A and one case for Category D; these were designated Case A-1 and Case D-1. Four cases were initially selected for site Category B; these were designated Cases B-1, B-2, B-3 and B-4. Three cases were initially selected for site Category C; these were designated Cases C-1, C-2 and C-3. Upon examination of the results of the response analyses for these cases, three additional cases were added. The additional cases were designated Cases B-1.5, B-3.5 and C-1.5. These latter cases were selected to provide robust estimate of the response at frequencies that did not seem to be adequately covered by the other cases.

The variations of maximum shear wave velocities with depth assigned for each case are summarized in Figs. 2-2 through 2-4. The shear wave velocity distribution with depth was selected to provide a reasonably wide range and also to provide significant contrast in velocities at ceratin depths for a selected number of cases. The range of maximum shear wave velocities used for all the cases considered in this study is presented in Fig. 2-5. More details about each case are given in Appendix B.

The variation of shear modulus with shear strain was based on using the upper curve from the range published by Seed and Idriss (1970) as shown in Fig. 2-6. The variations of damping with shear strain was based on the lower curve from the range published by the same authors as shown in Fig. 2-7.

Fig. 2-1

GENERIC SITE CATEGORIES

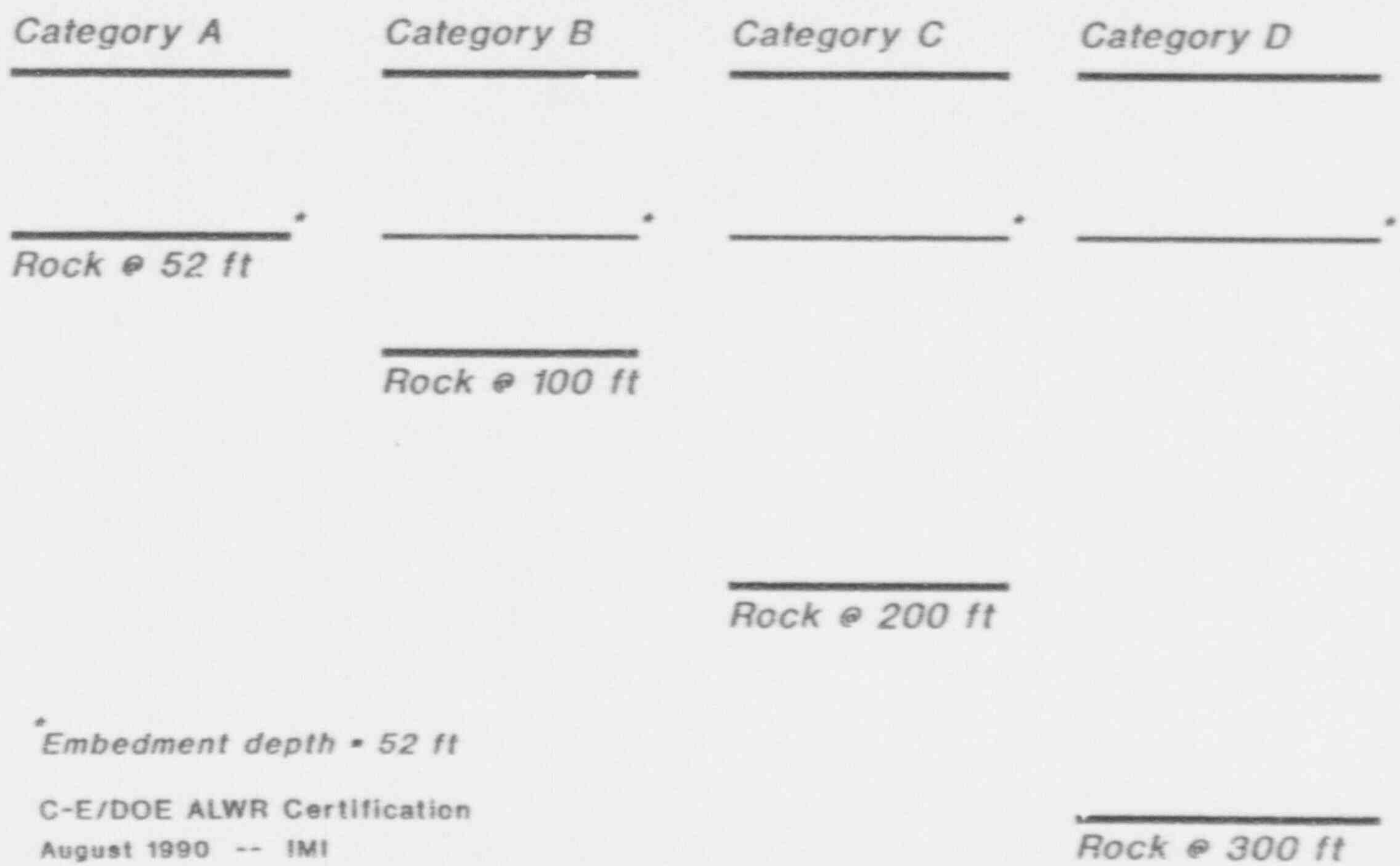

Fig. 2-2
SHEAR WAVE VELOCITIES
CASES B-1, B-2, B-3 AND B-4

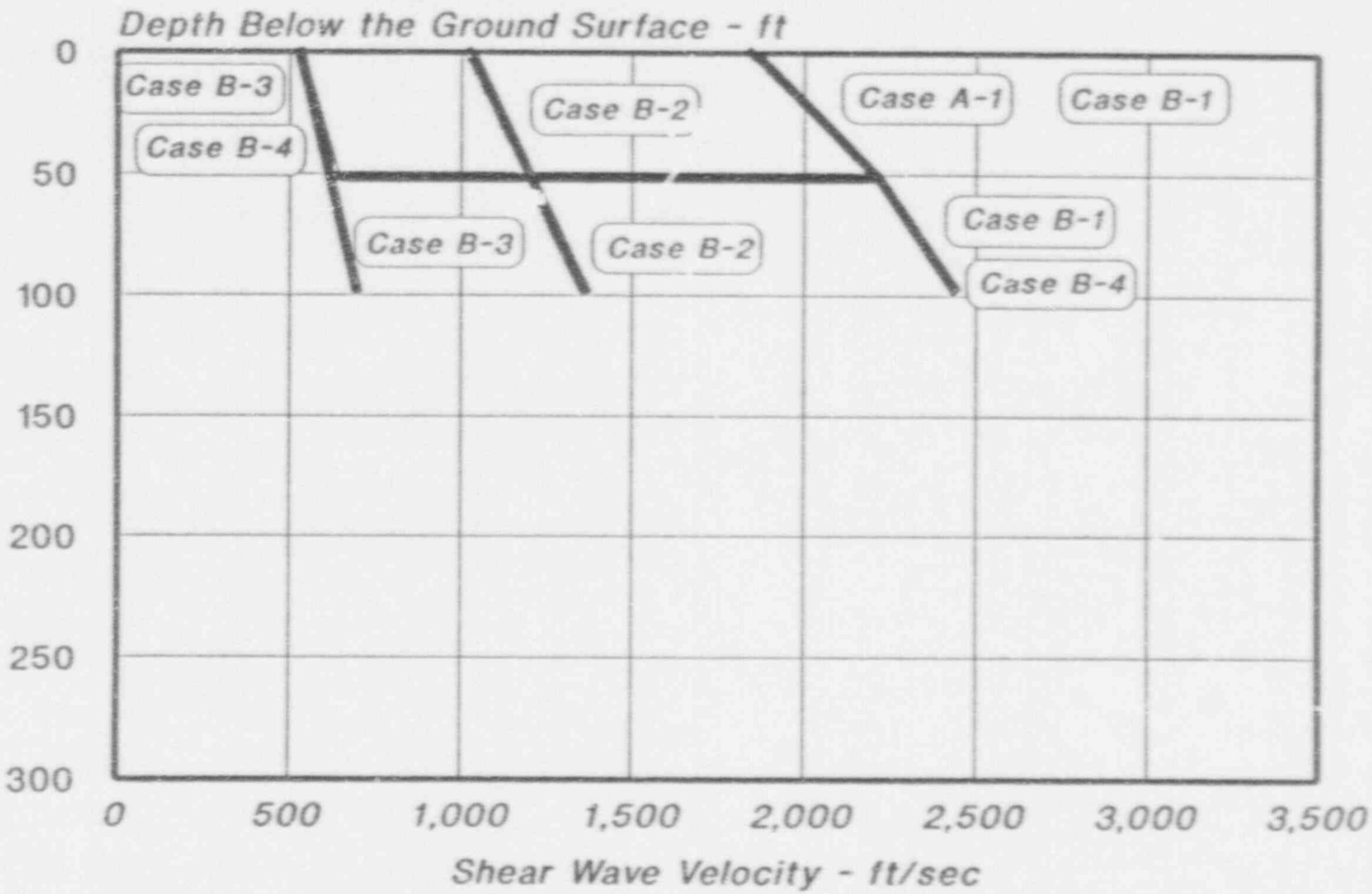

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Fig. 2-3
SHEAR WAVE VELOCITIES
CASES C-1, C-2, C-3 AND D-1

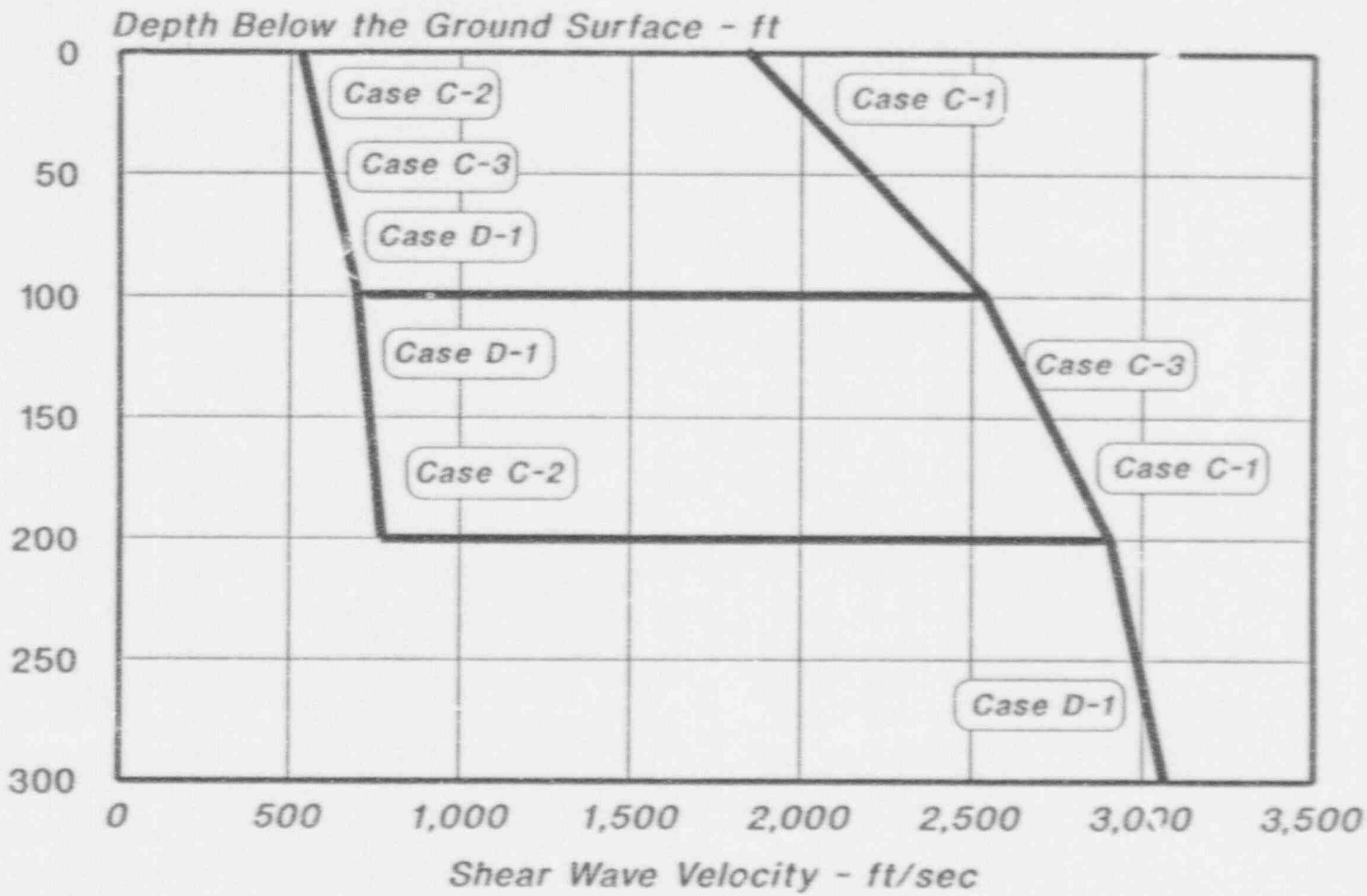

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Fig. 2-4
SHEAR WAVE VELOCITIES
CASES B-1.5, B-3.5 AND C-1.5

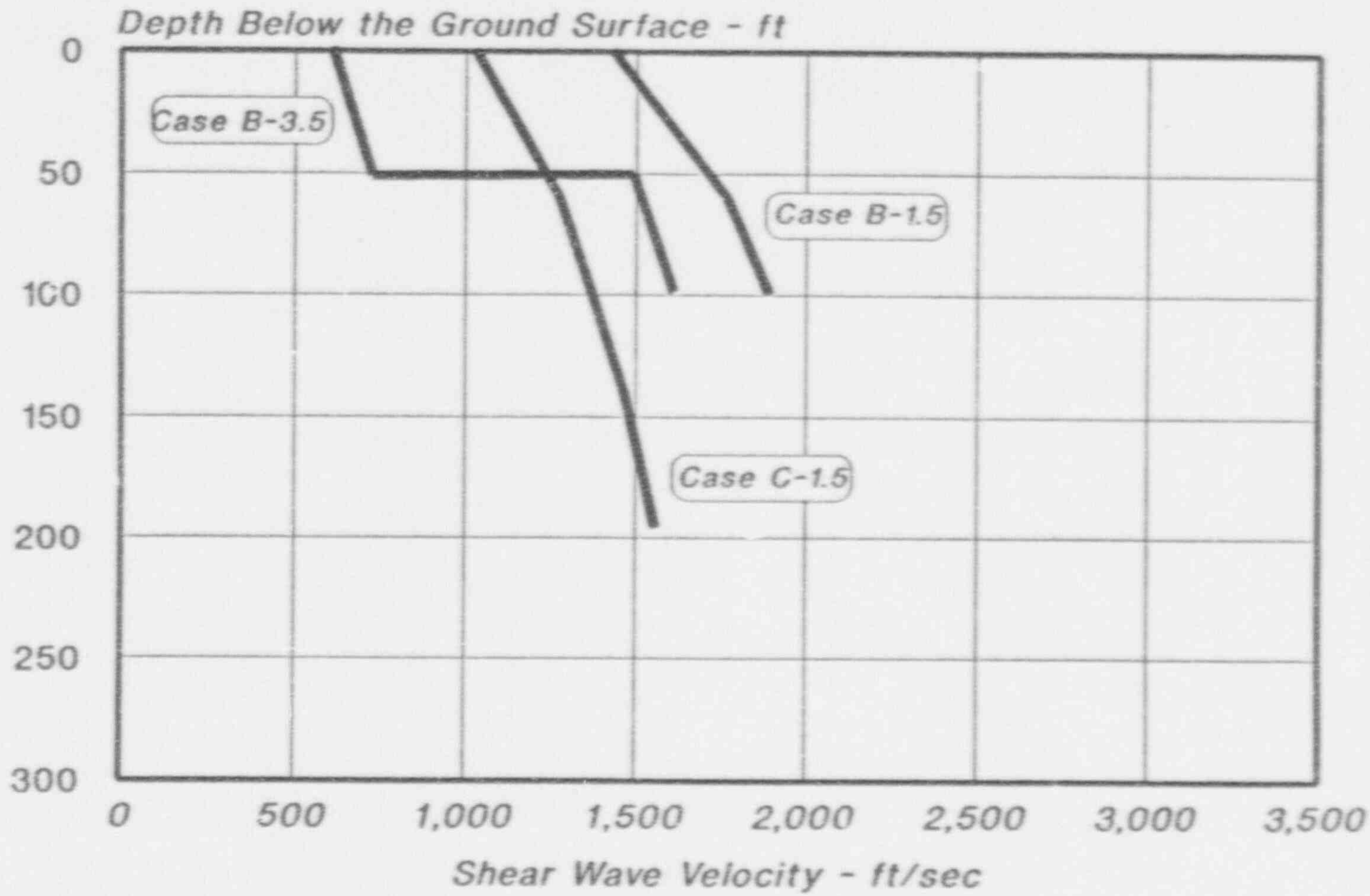

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Fig. 2-5
RANGE OF SHEAR WAVE VELOCITIES
FOR ALL CASES CONSIDERED

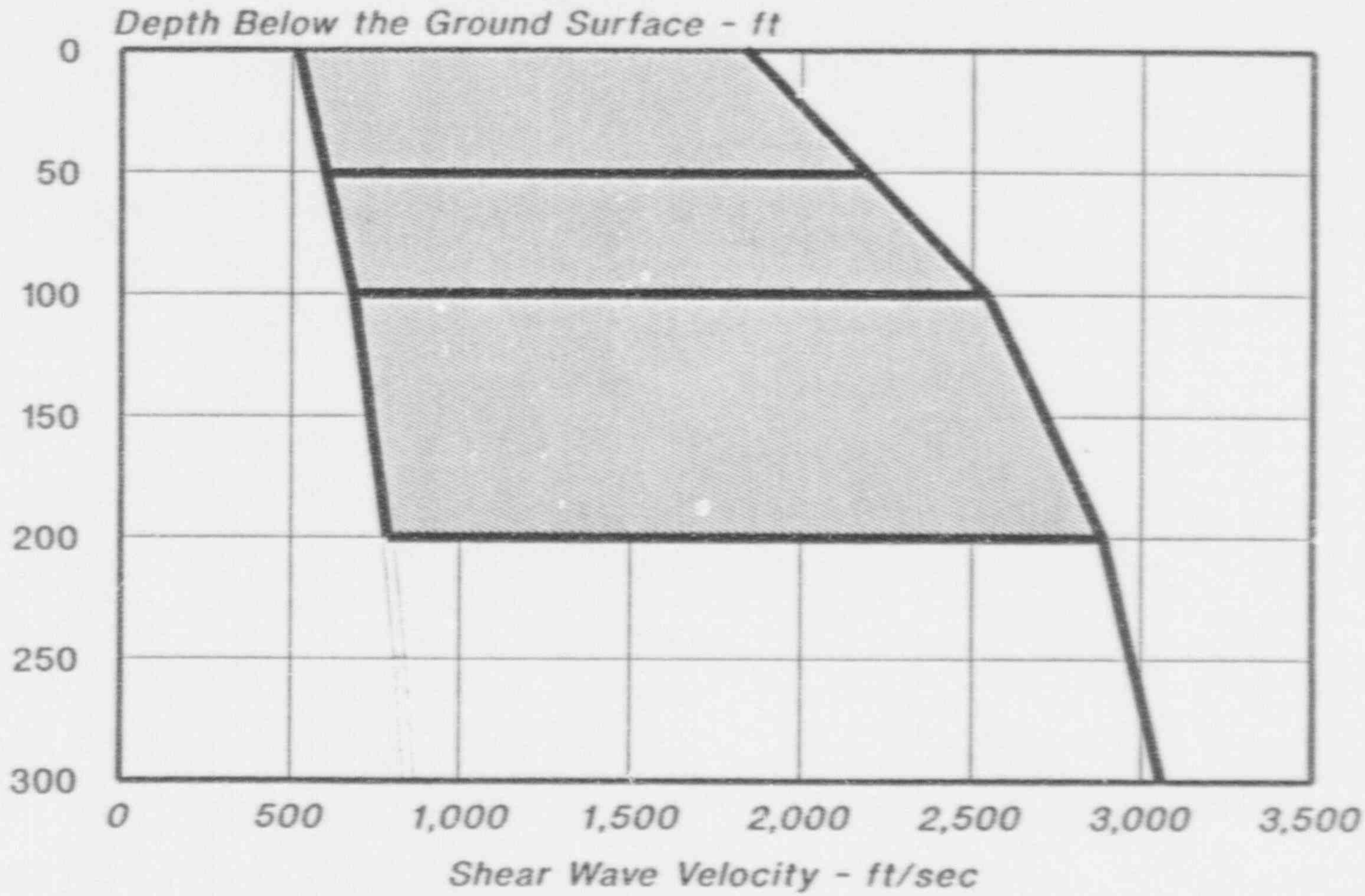

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Fig. 2-6
VARIATION OF SHEAR MODULUS WITH STRAIN
& RELATIONSHIP SELECTED FOR THIS PROJECT

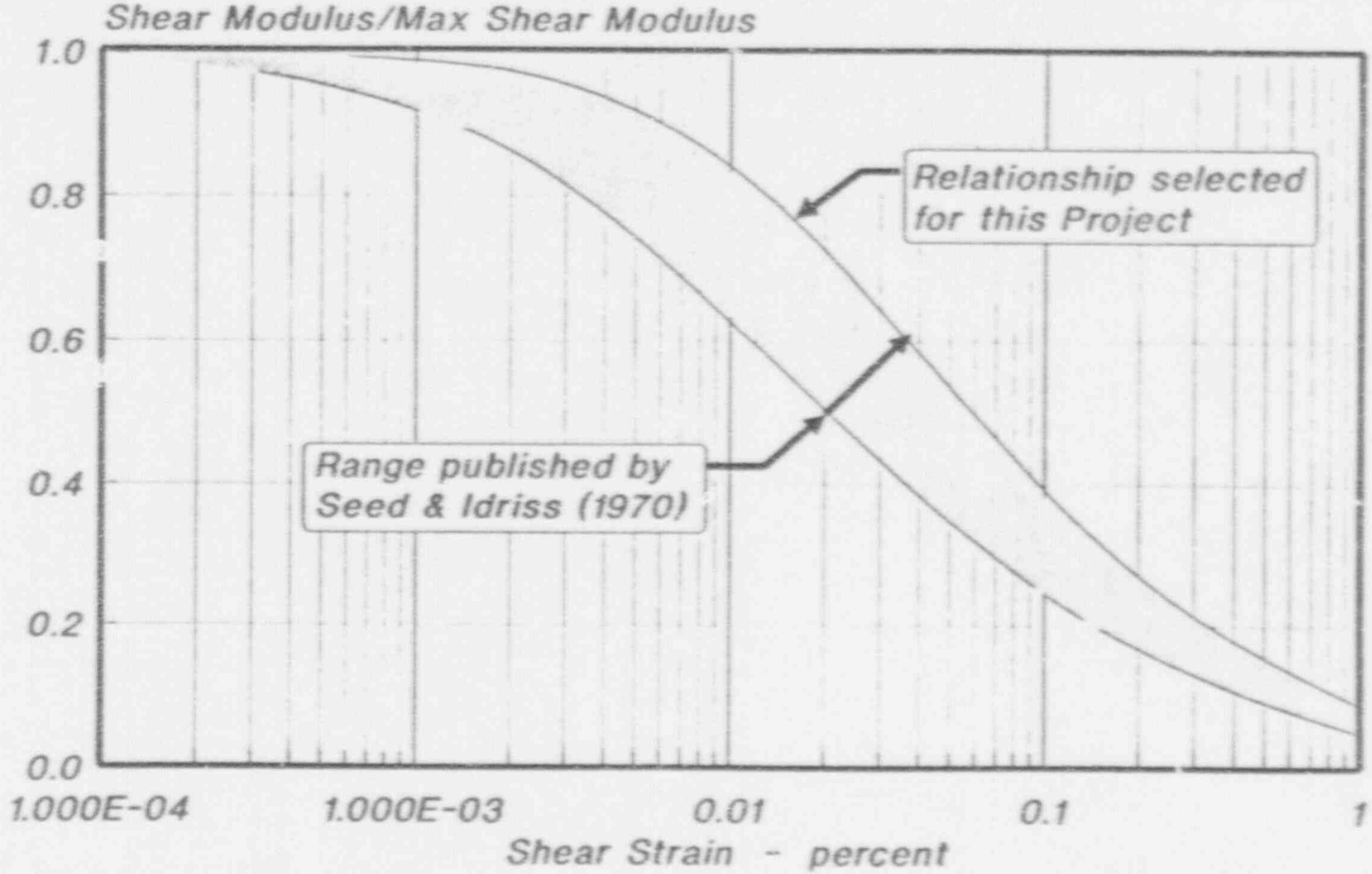

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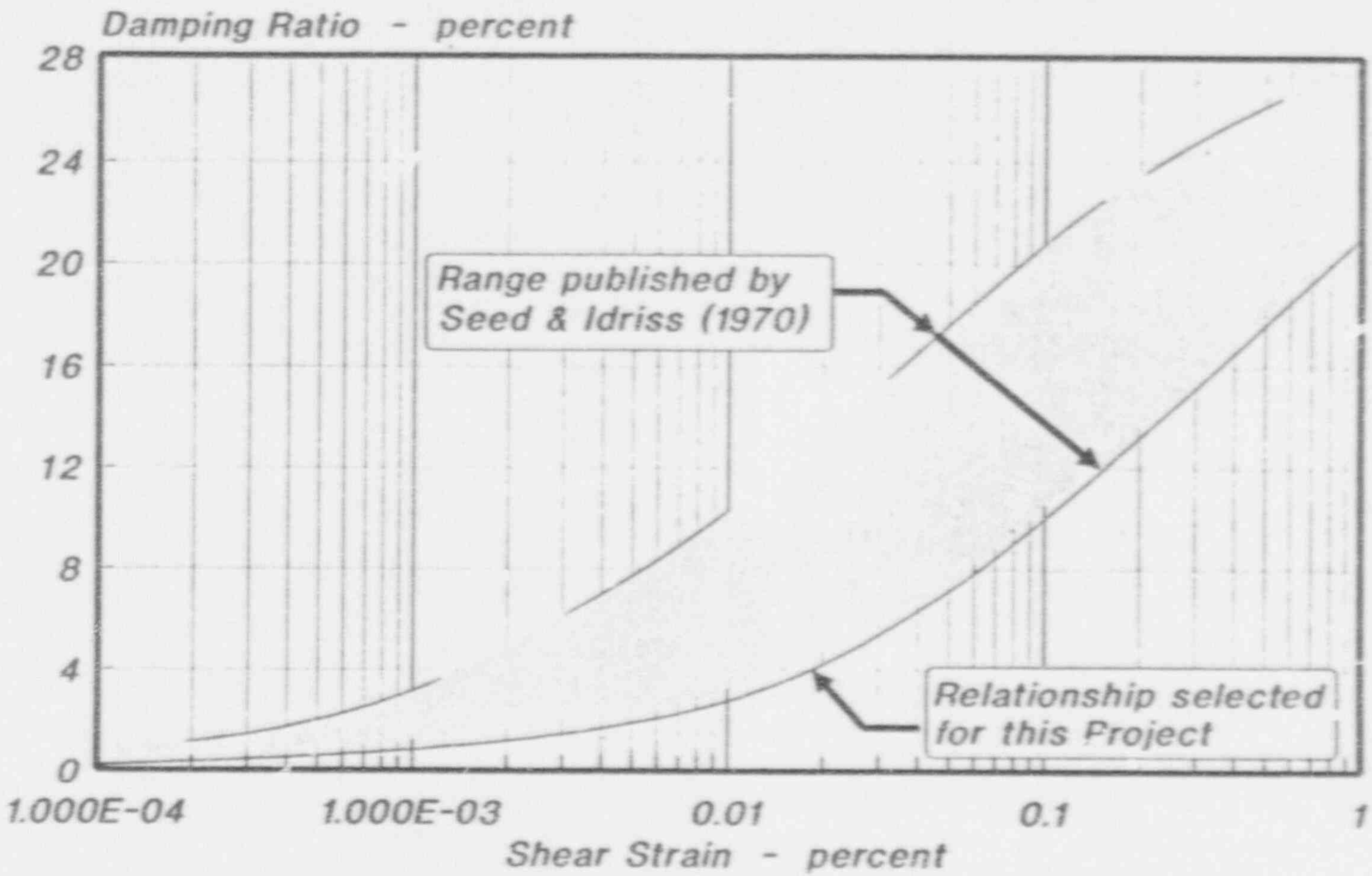
Fig. 2-7
VARIATION OF DAMPING RATIO WITH STRAIN
& RELATIONSHIP SELECTED FOR THIS PROJECT



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3.0 CONTROL MOTION

The smooth spectrum representing the control motion in the free field at a rock outcrop was developed based on spectral shapes for earthquake ground motions considered appropriate for Eastern North America and those based on NUREG-0098. The latter are considered appropriate for Western North America.

A peak horizontal acceleration of 0.3 g was selected for the horizontal components of motion. The selected horizontal smooth spectrum and the other spectra considered in developing this spectrum are shown in Fig. 3-1. As can be noted in this figure, the spectral ordinates were kept equal to those obtained using NUREG-0098 for frequencies lower than about 4 hz. For higher frequencies, particularly above 10 hz, the selected spectral ordinates are significantly greater than those obtained using the NUREG-0098. The shape estimated for Eastern North America influenced this adjustment to these values and the use of the smooth spectrum identified in Fig. 3-1 as "selected horizontal spectrum".

The two horizontal components, H1 and H2, were considered to have identical specua and the vertical component was considered to be equal to 2/3 of the horizontal spectrum at all frequencies. The selected spectra for the horizontal components and for the vertical component are shown in Fig. 3-2.

Synthetic time histories were generated for each component. The spectral ordinates calculated for each synthetic time history and the corresponding selected smooth spectrum are shown in Figs. 3-3, 3-4 and 3-5 for horizontal components H1 and H2 and vertical component, respectively. The spectral ordinates of each synthetic time history conservatively envelops the selected smooth spectra at most frequencies.

The characteristics of each synthetic time history (accelerogram, time histories of velocity and displacement, Fourier amplitudes and Power Spectral Density) are presented in Appendix B. Also presented in Appendix B are the correlation coefficients among these synthetic time histories. The maximum correlation coefficient is about 0.1 indicating that these time histories are statistically independent.

Note that the above values of peak accelerations are for the conditions representing the Safe Shutdown Earthquake (SSE). For the conditions representing the Operating Basis earthquake (OBE) a peak horizontal acceleration of 0.1 g and a peak vertical acceleration of 0.067 g were used. The same spectral shape and time histories (normalized to the appropriate peak acceleration) were used for the OBE analyses.

Fig. 3-1
SPECTRAL ORDINATES FOR ROCK OUTCROP
AND SELECTED HORIZONTAL SPECTRUM

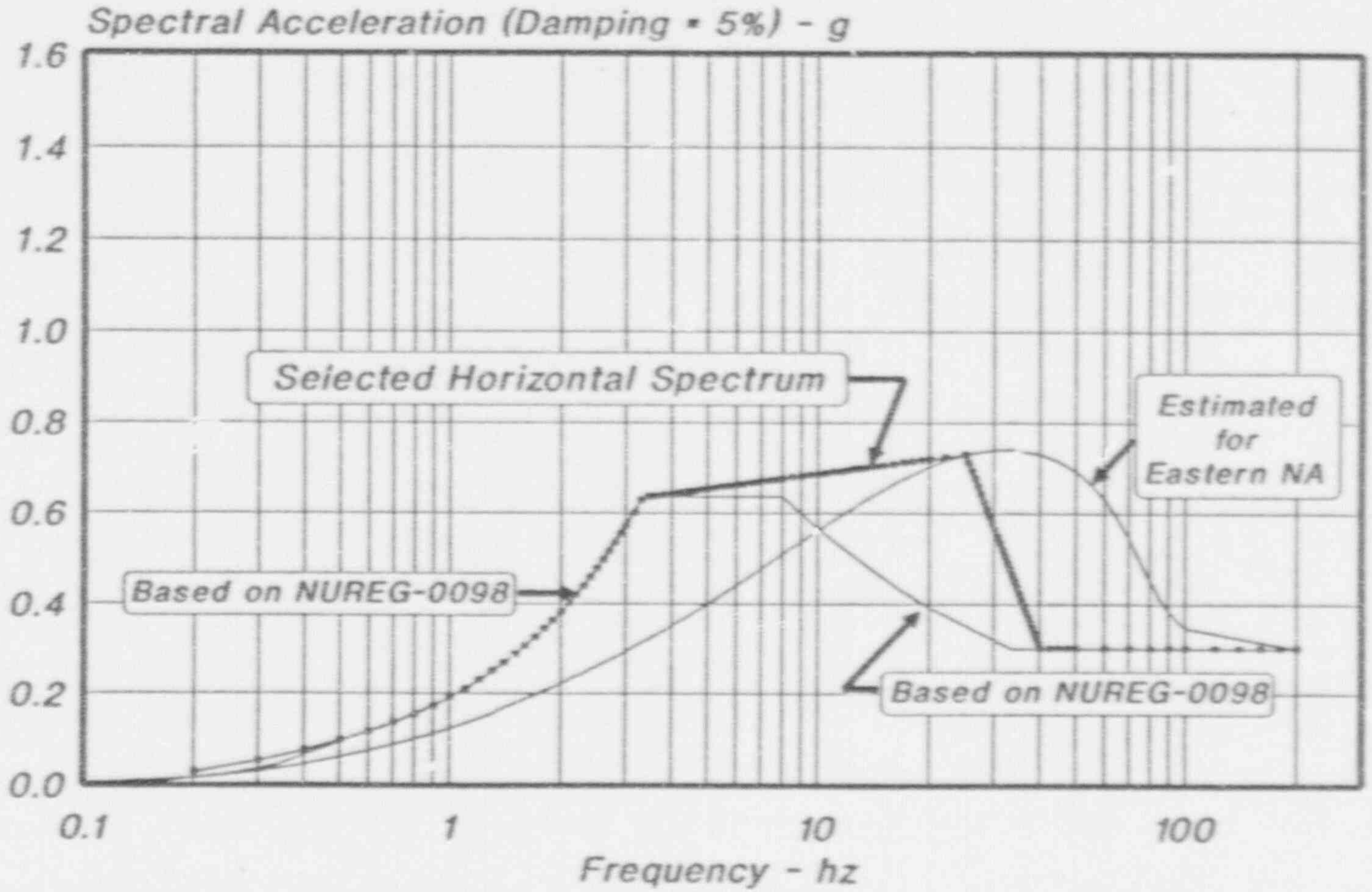

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Fig. 3-2
HORIZONTAL & VERTICAL COMPONENTS OF
SELECTED SMOOTH SPECTRA FOR ROCK OUTCROP

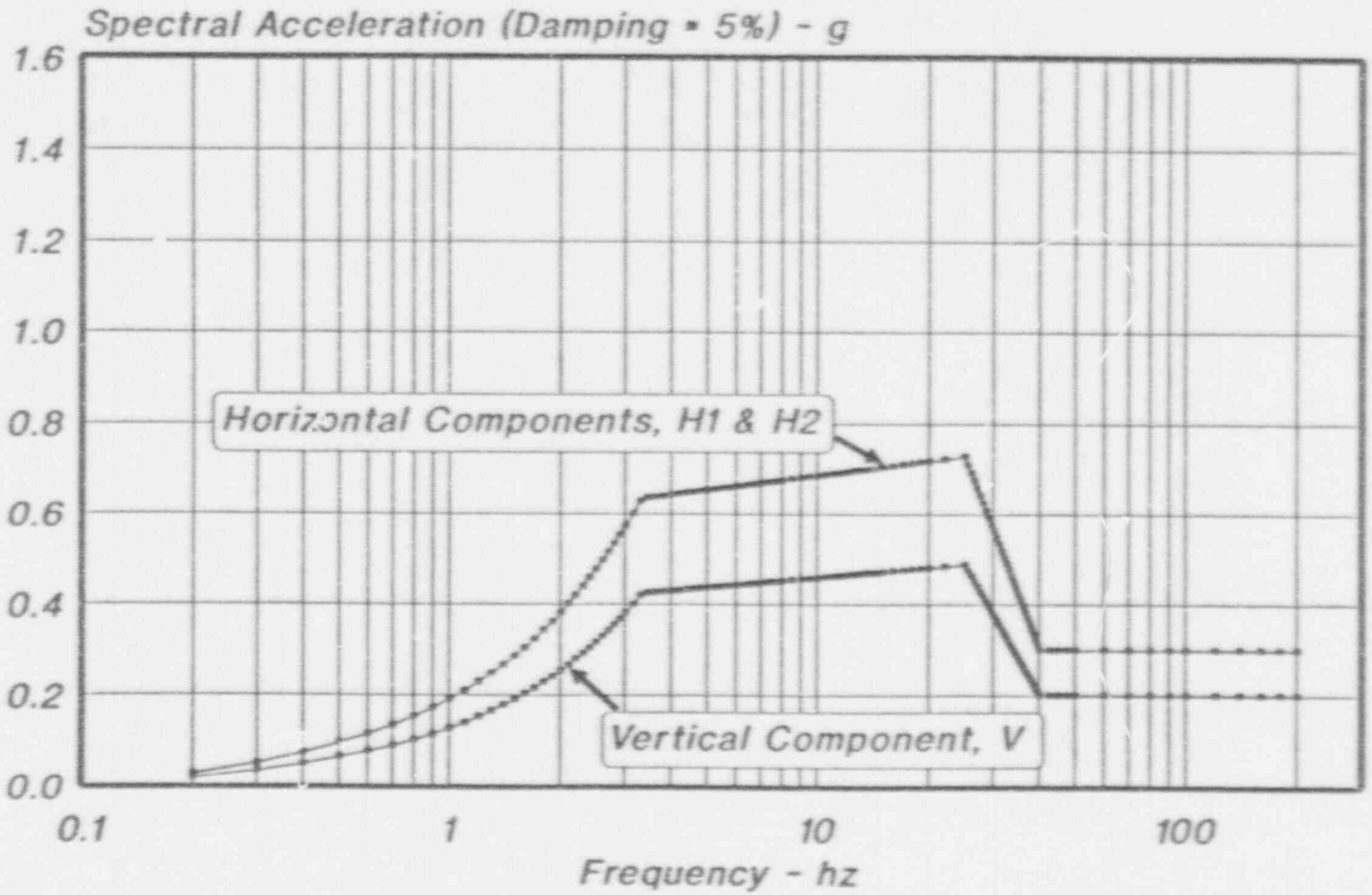

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Fig. 3-3
SELECTED SMOOTH SPECTRUM AND SPECTRUM
FOR SYNTHETIC TIME HISTORY H1

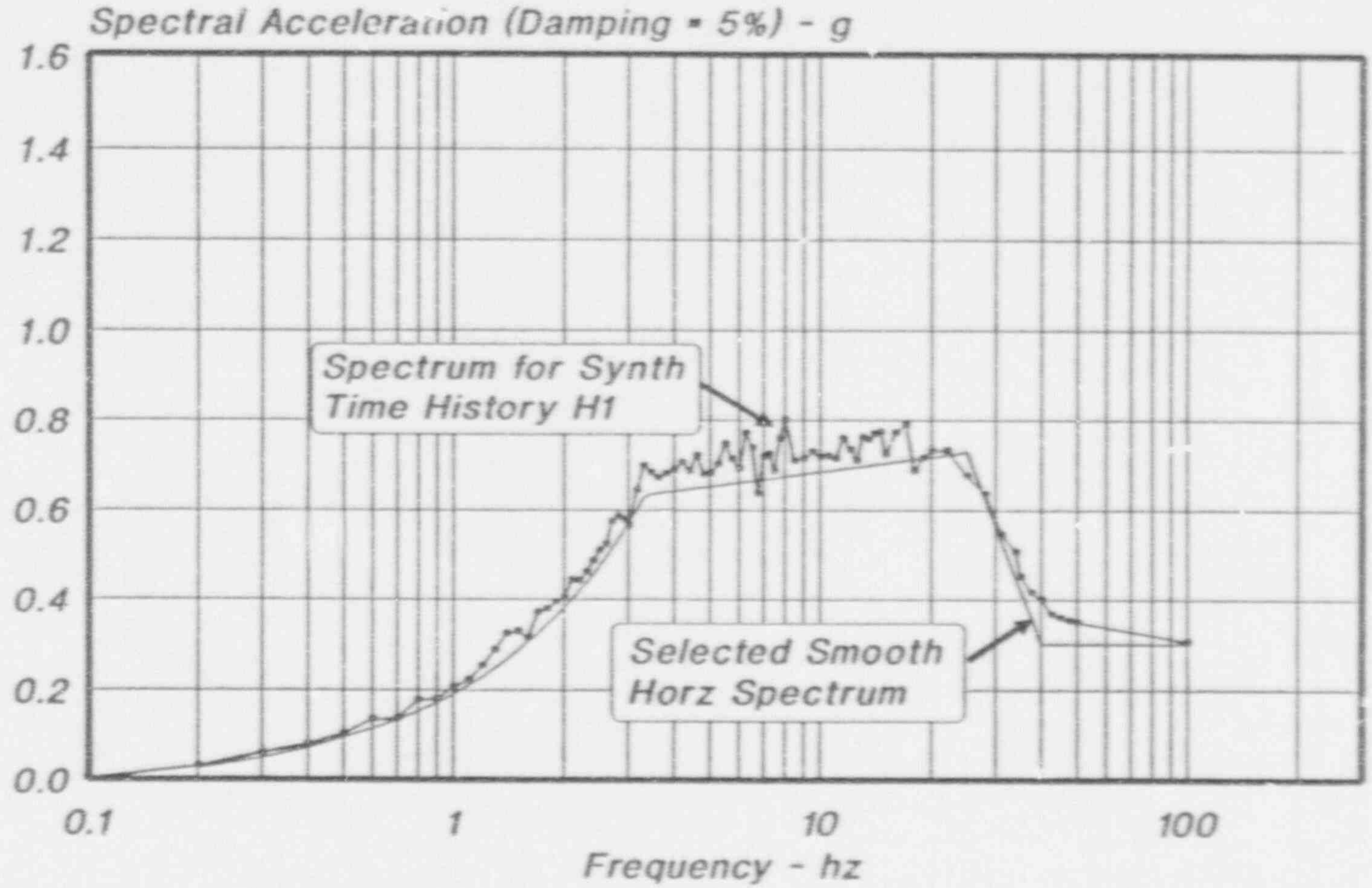

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Fig. 3-4
SELECTED SMOOTH SPECTRUM AND SPECTRUM
FOR SYNTHETIC TIME HISTORY H2

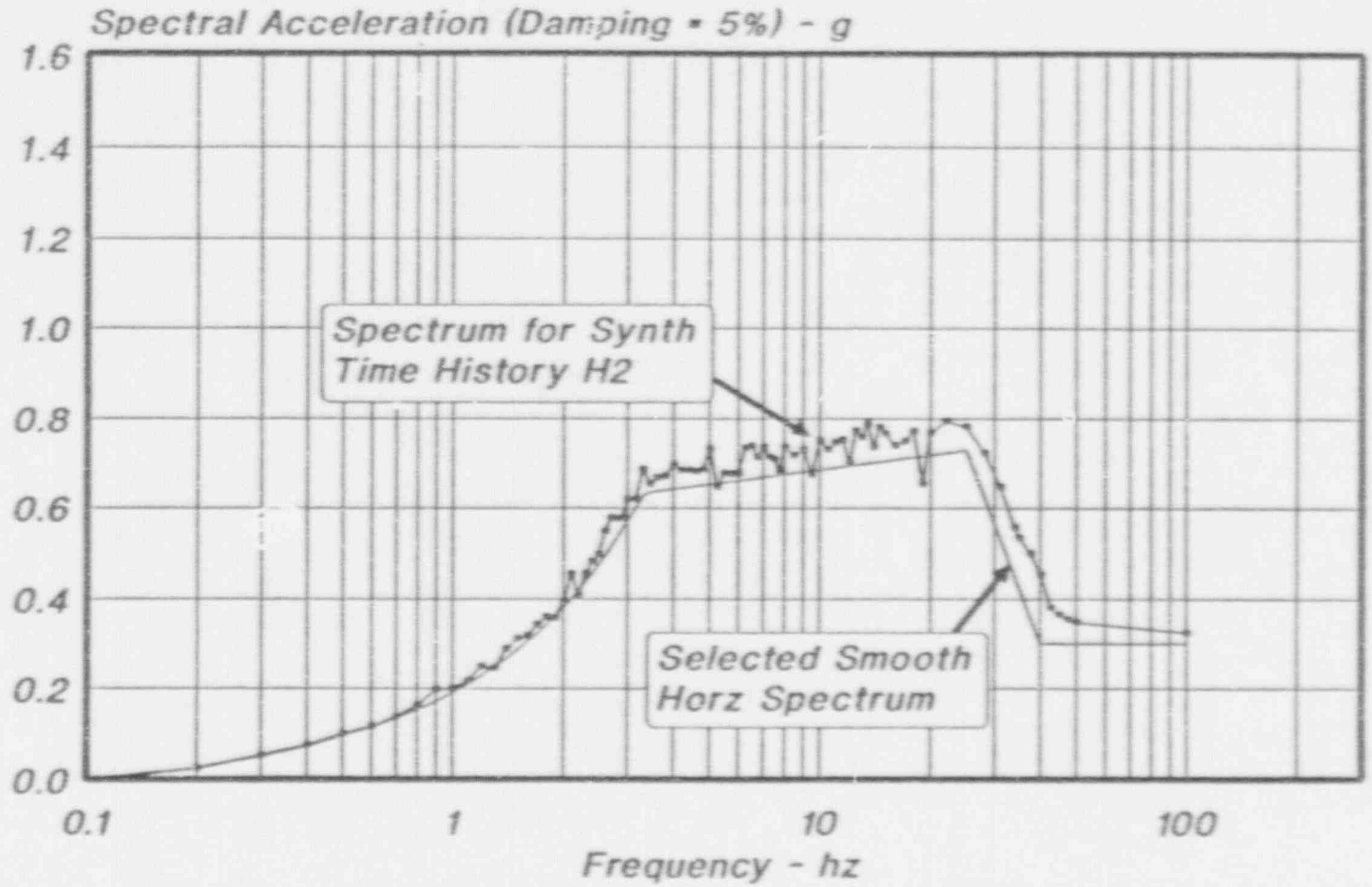

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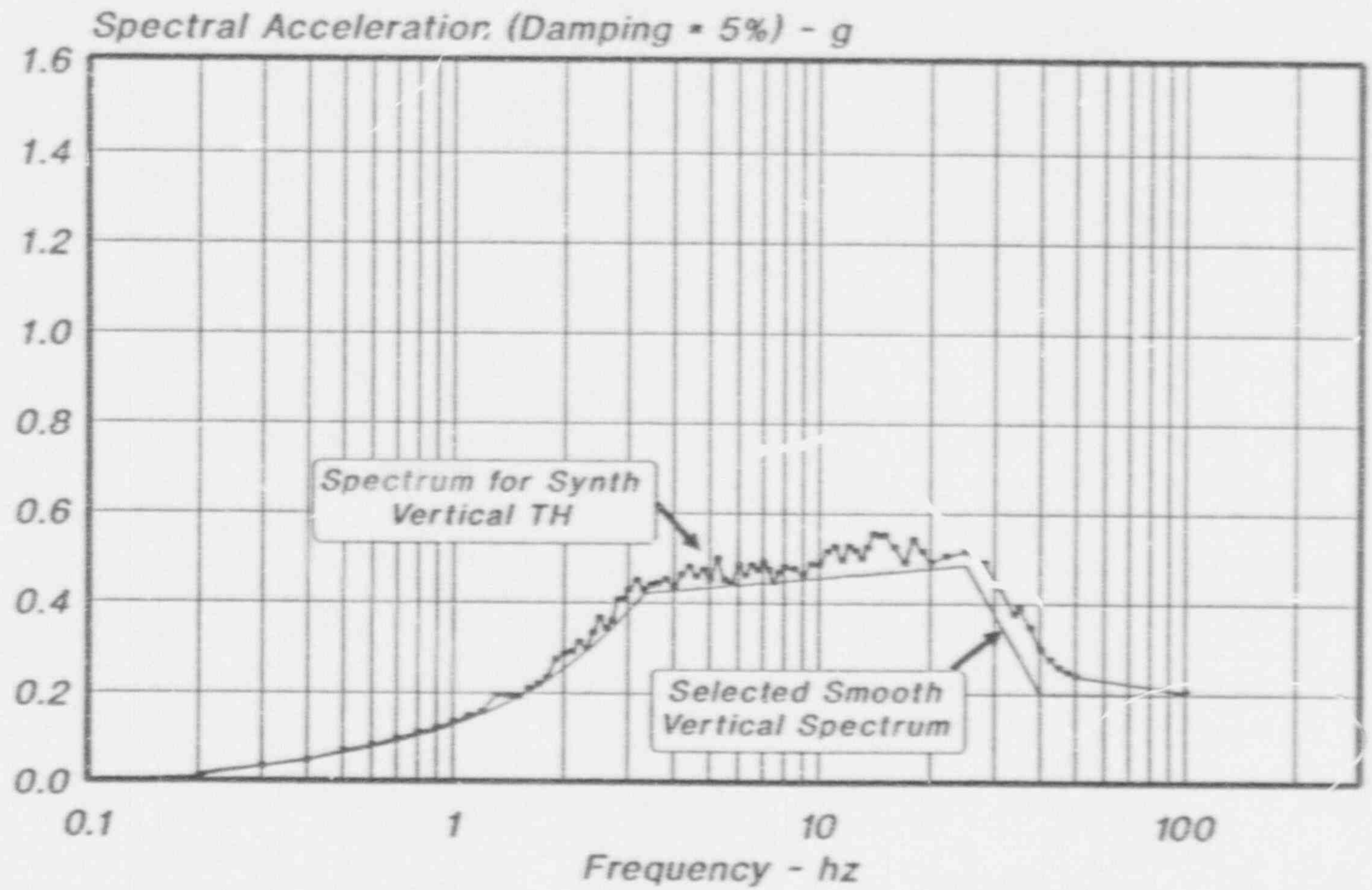
Fig. 3-5
SELECTED SMOOTH SPECTRUM AND SPECTRUM
FOR VERTICAL SYNTHETIC TIME HISTORY V



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4.0 SITE RESPONSE

The response of each soil case was obtained using an equivalent linear response analysis. For each case synthetic time history H1 was applied as the input rock outcrop motion. The strain-compatible modulus and damping values were then obtained for that soil case. These properties were then used without further modifications for the analysis involving synthetic time history H2 as the input rock outcrop motion. (The strain-compatible modulus and damping values thus obtained are listed in the tables included in Appendix C). For analyses involving the vertical component, the strain-compatible shear moduli were converted to constrained moduli and the strain-compatible damping values were multiplied by 1/3 to provide an estimate of the damping associated with the propagation of p-waves (note that this assumption is quite conservative and further evaluation considering higher damping are under way and will be included in the final report).

The response values at the ground surface and at the foundation level (52 ft below the ground surface) for each case are presented in Appendix C.

Selected results are presented in this section in Figs. 4-1 through 4-7?. Figs. 4-1 through 4-4 show the response spectra at the ground surface and at the foundation level for cases A-1, B-1, C-1 and D-1 when the synthetic time history H1 is applied as input rock outcrop motion. Figs. 4-5, 4-6 and 4-7 show the spectral ordinates calculated at the ground surface for all cases considered (also using synthetic time history H1 as input motion). The corresponding spectra calculated at the foundation level are shown in Figs. 4-8, 4-9 and 4-10.

The spectra calculated at the ground surface using synthetic time history H2 are presented in Figs. 4-11, 4-12 and 4-13 and those at the foundation level are presented in Figs. 4-14, 4-15 and 4-16. The corresponding spectra for the vertical component are presented in Figs. 4-17 through 4-22.

Fig. 4-1
SPECTRA AT GROUND SURFACE AND FOUNDATION
LEVEL FOR CASE A-1 USING SYNTH TH H1

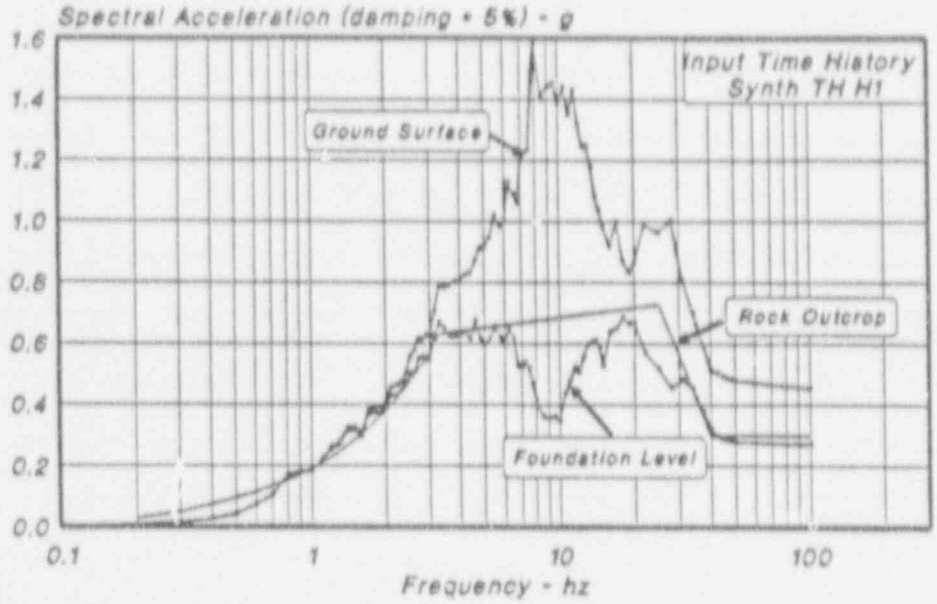

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Fig. 4-2
SPECTRA AT GROUND SURFACE AND FOUNDATION
LEVEL FOR CASE B-1 USING SYNTH TH H1

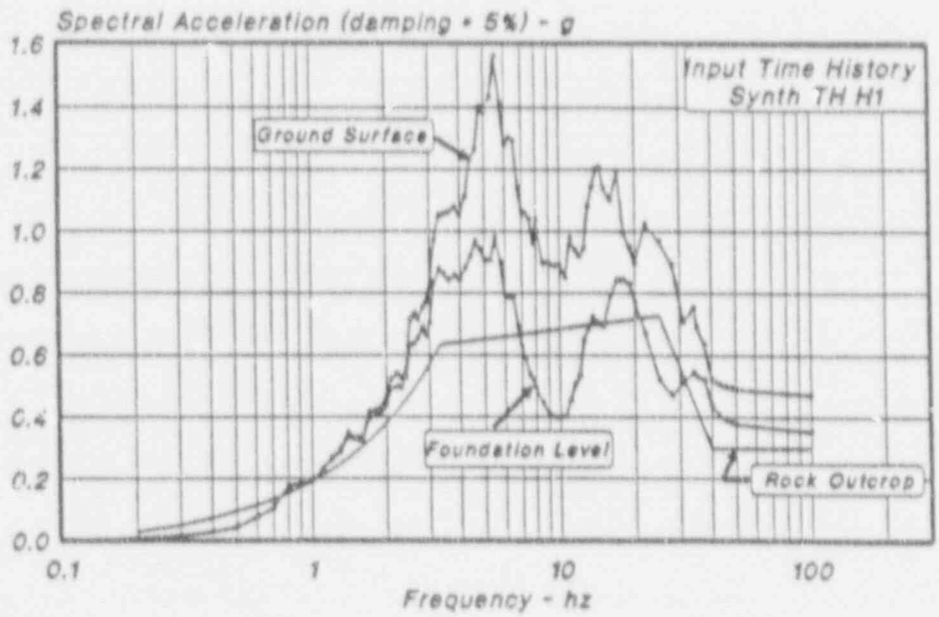

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Fig. 4-3
SPECTRA AT GROUND SURFACE AND FOUNDATION
LEVEL FOR CASE C-1 USING SYNTH TH H1

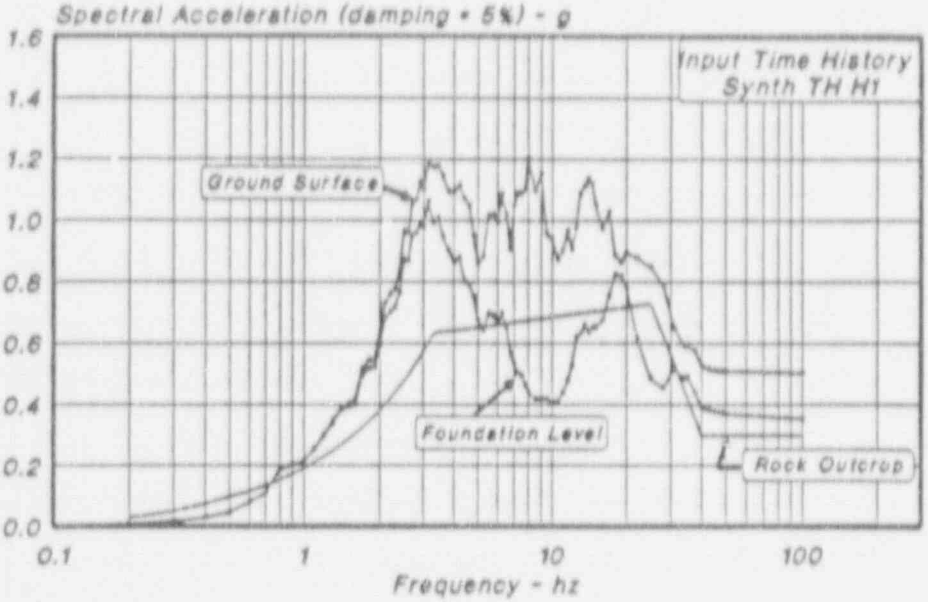

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Fig. 4-4
SPECTRA AT GROUND SURFACE AND FOUNDATION
LEVEL FOR CASE D-1 USING SYNTH TH H1

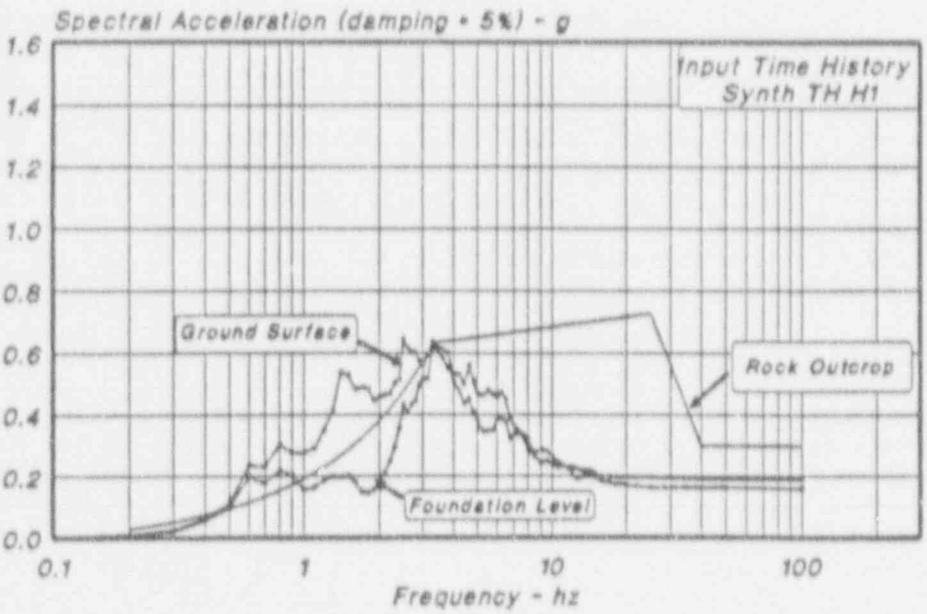

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Fig. 4-5
SPECTRA AT GROUND SURFACE FOR CASES A-1, B-1, B-2, B-3 AND B-4 USING SYNTH TH H1

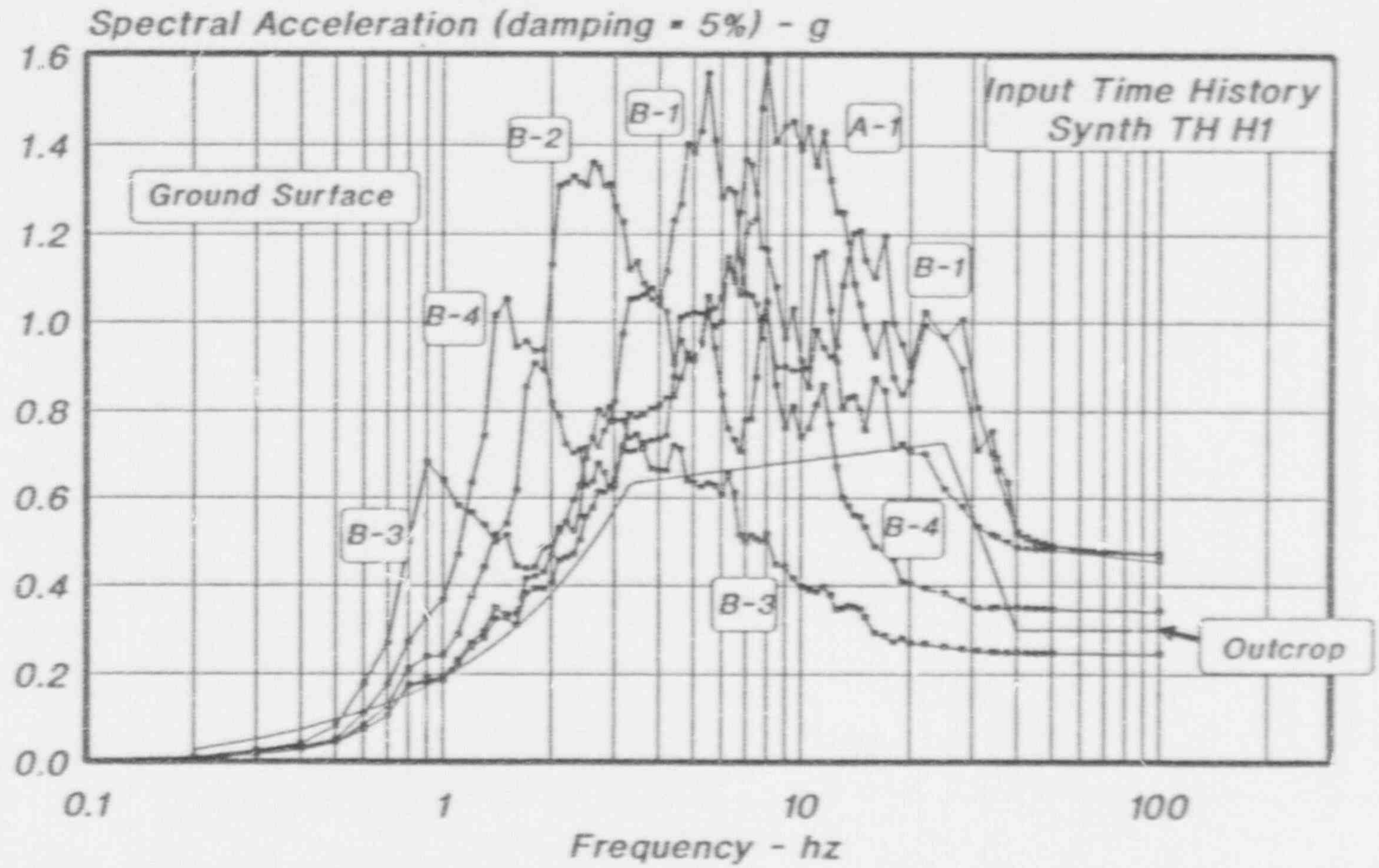

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Fig. 4-6
SPECTRA AT GROUND SURFACE FOR CASES C-1, C-2, C-3 AND D-1 USING SYNTH TH H1

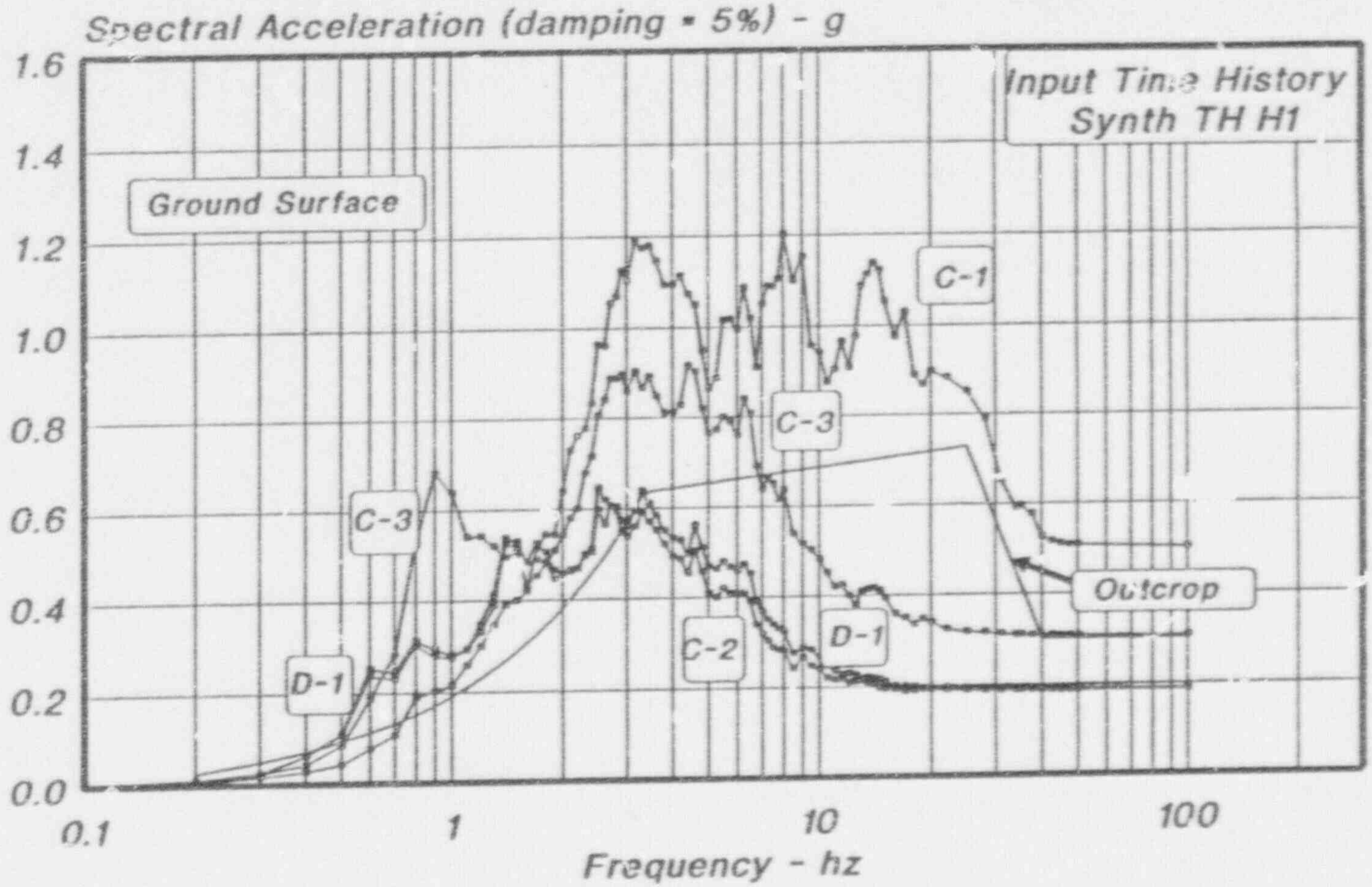

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Fig. 4-7
SPECTRA AT GROUND SURFACE FOR CASES
B-1.5, B-3.5 & C-1.5 USING SYNTH TH H1

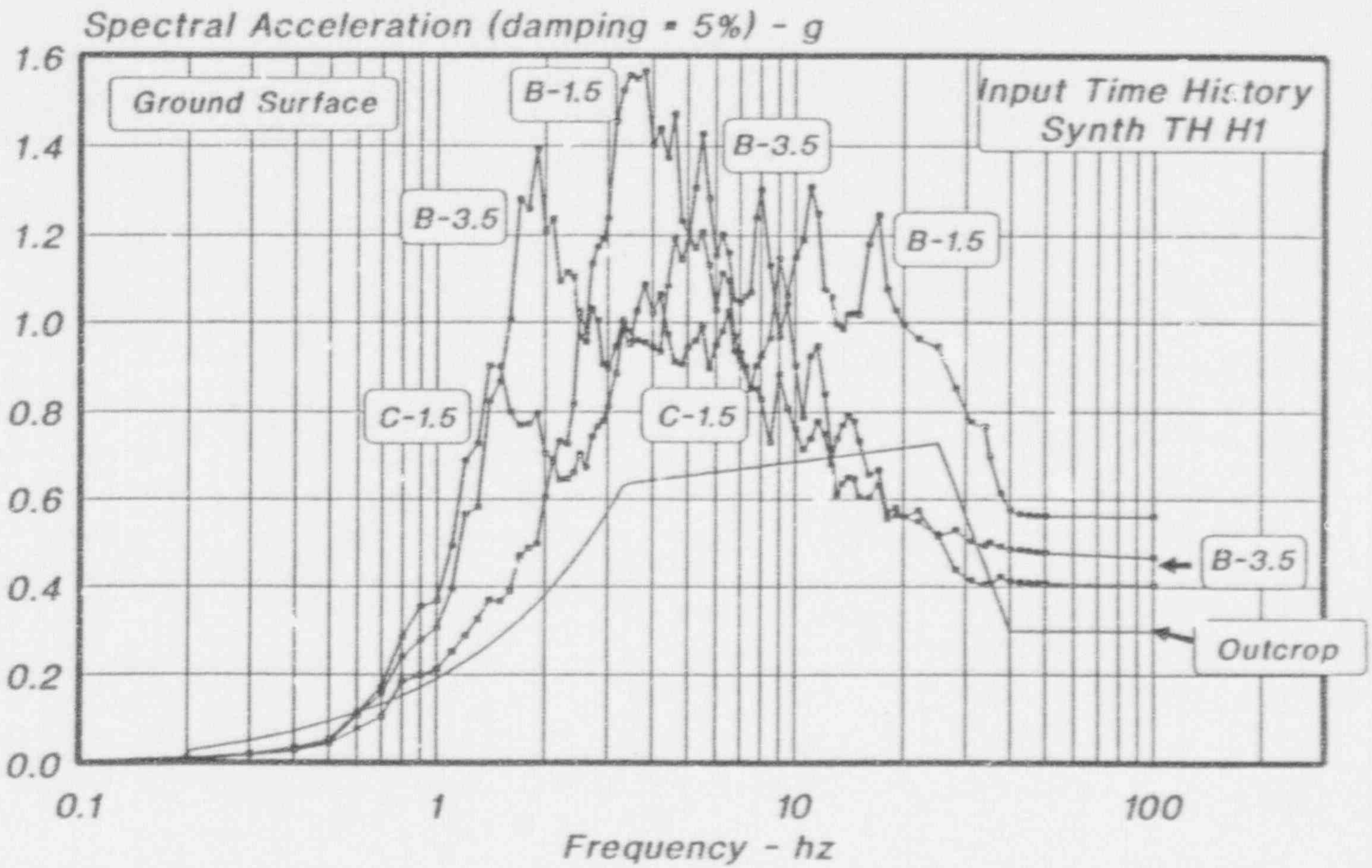

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Fig. 4-8
SPECTRA AT FOUNDATION LEVEL FOR CASES
A-1, B-1 .. B-4 USING SYNTH TH H1

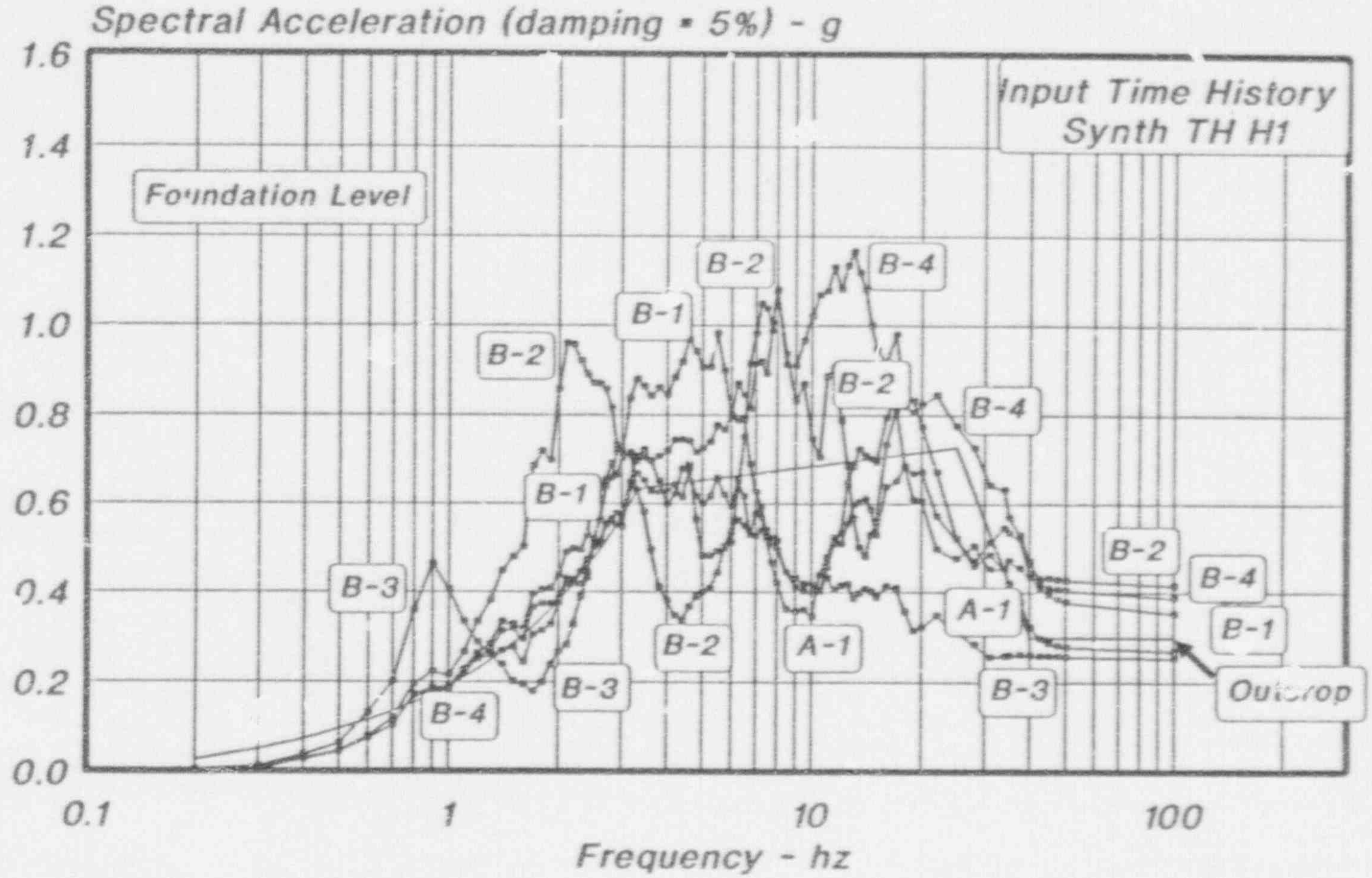

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Fig. 4-9
SPECTRA AT FOUNDATION LEVEL FOR CASES
C-1, C-2, C-3 & D-1 USING SYNTH TH H1

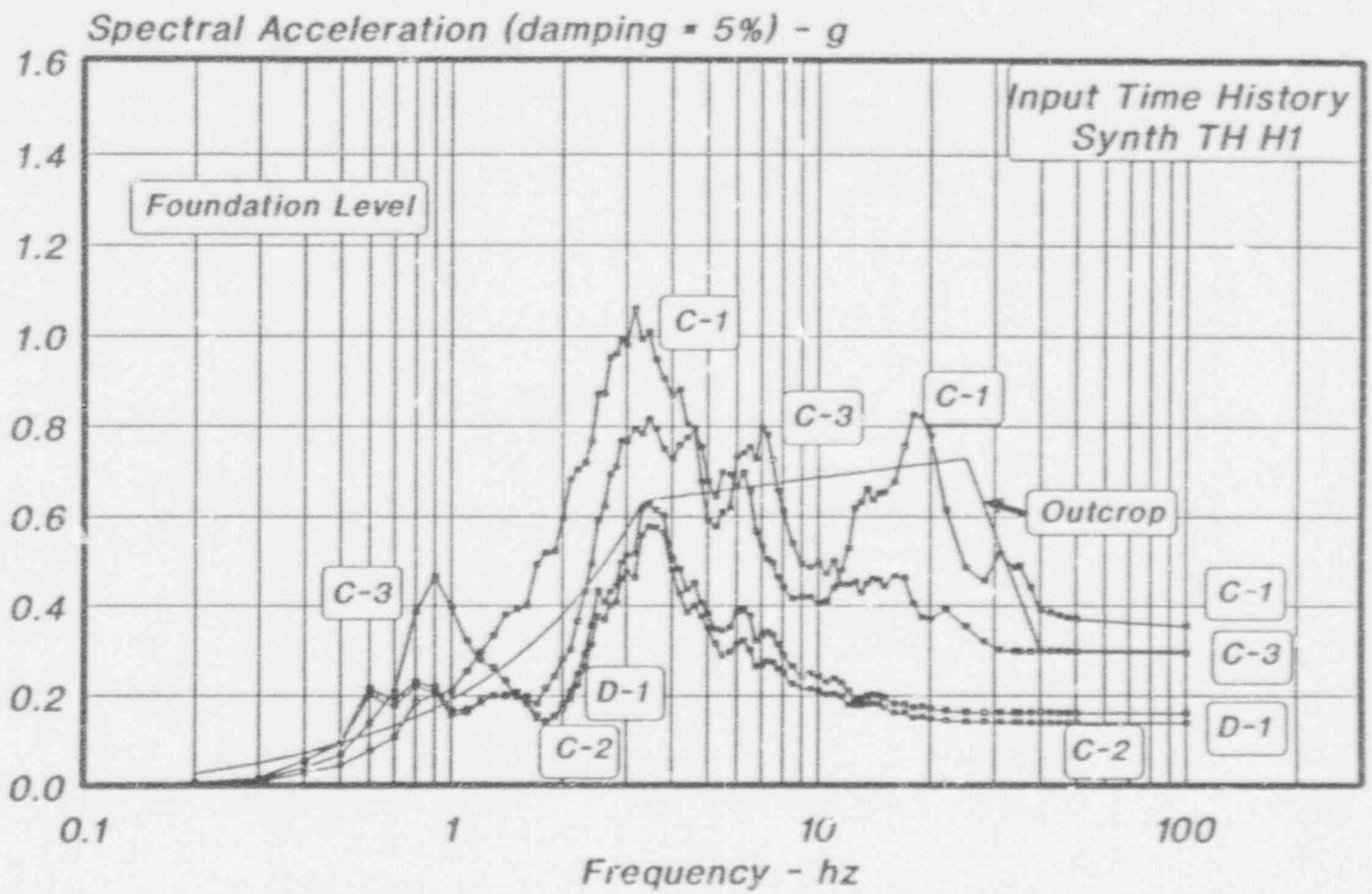

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Fig. 4-10
SPECTRA AT FOUNDATION LEVEL FOR CASES
B-1.5, B-3.5 & C-1.5 USING SYNTH TH H1

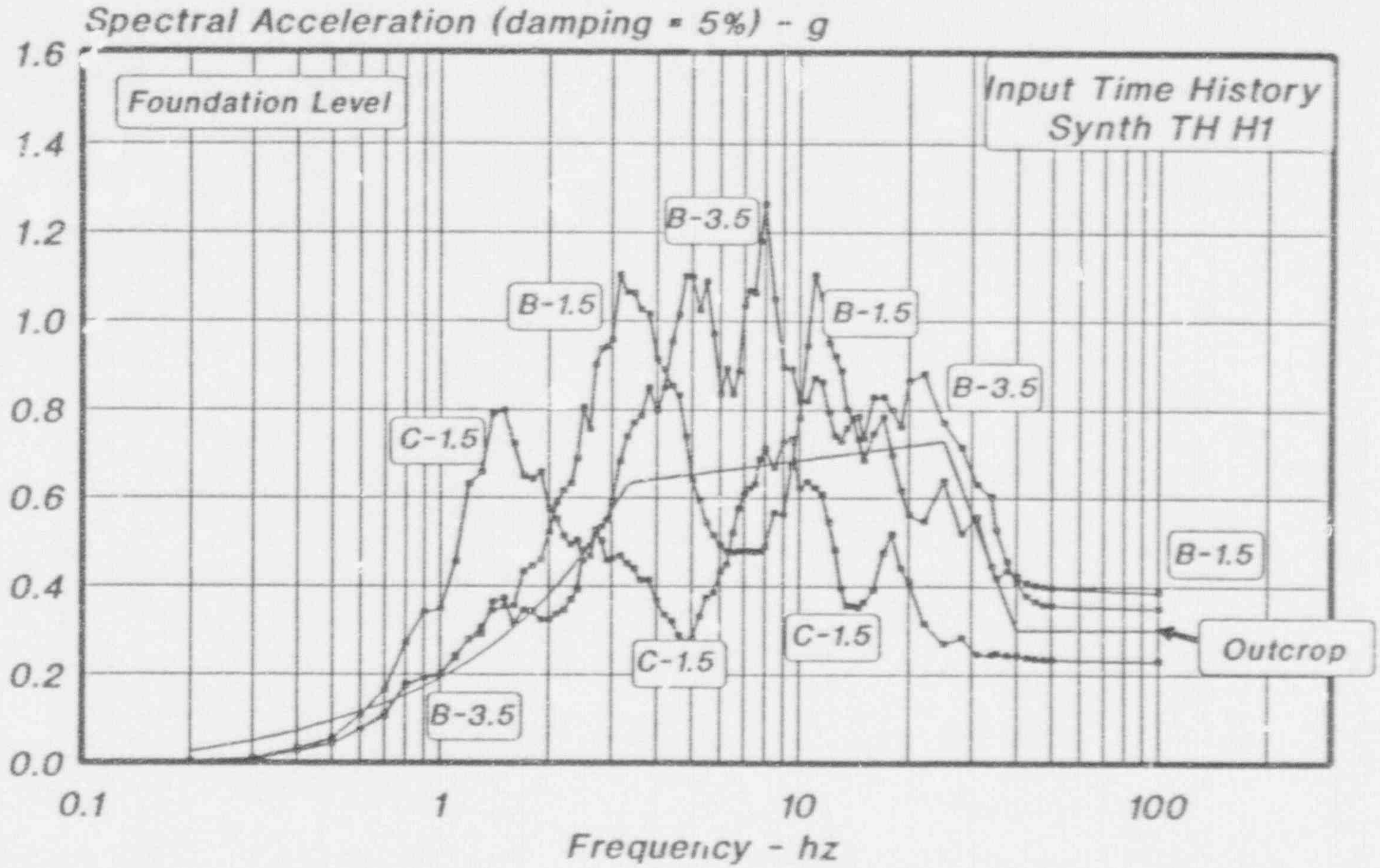

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Fig. 4-11
SPECTRA AT GROUND SURFACE FOR CASES A-1, B-1, B-2, B-3 AND B-4 USING SYNTH TH H2

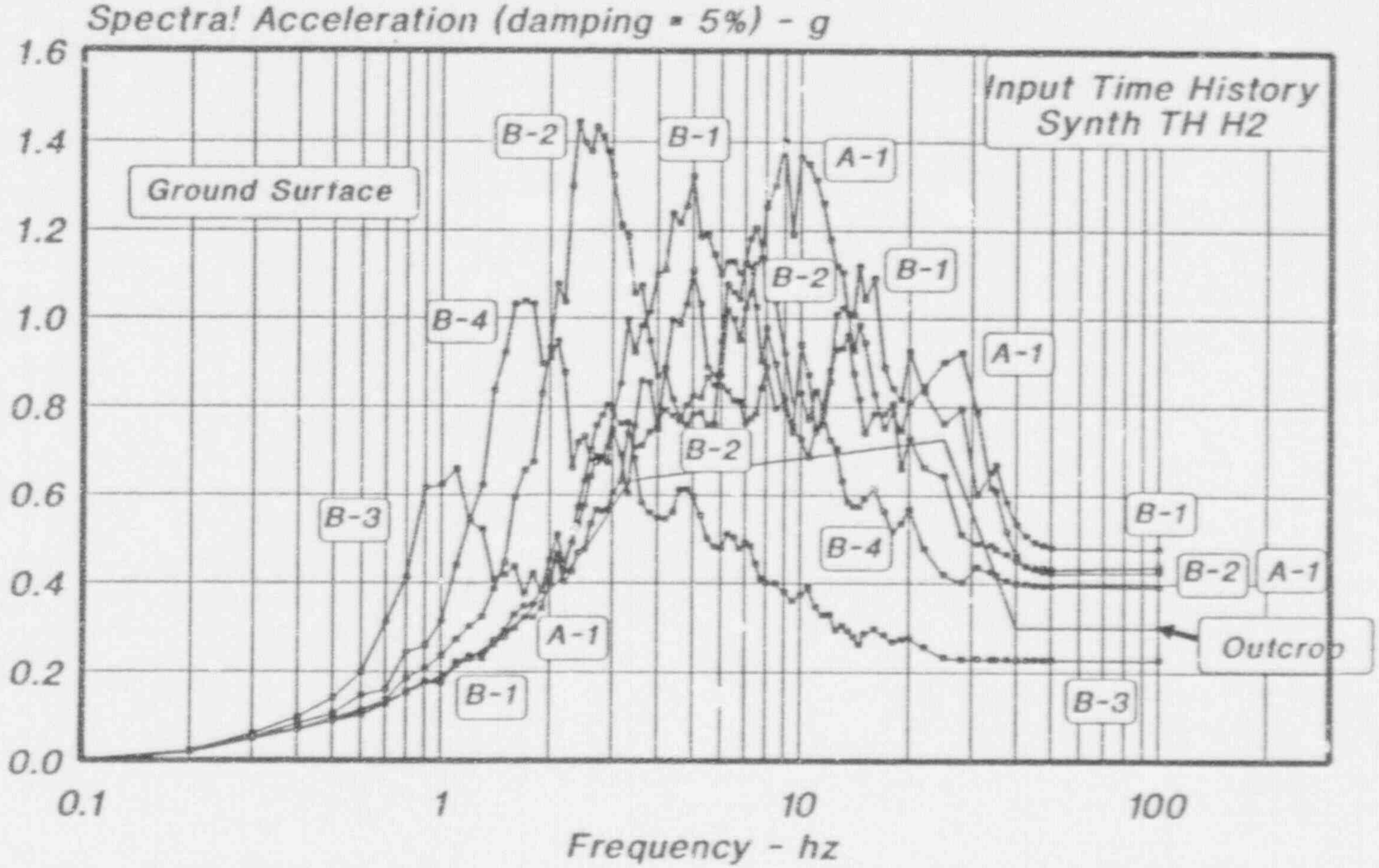

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Fig. 4-12
SPECTRA AT GROUND SURFACE FOR CASES C-1, C-2, C-3 & D-1 USING SYNTH TH H2

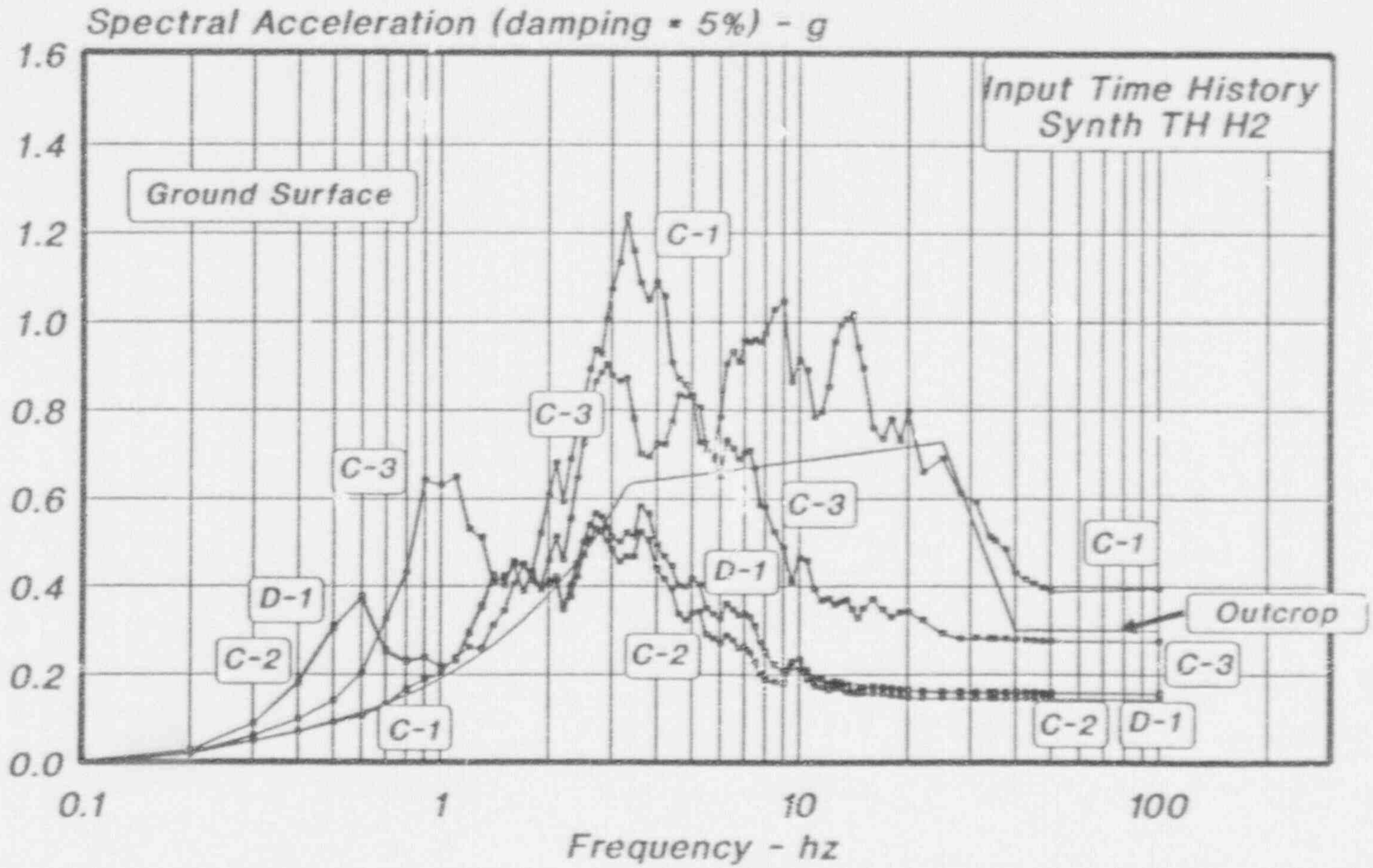

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Fig. 4-13
SPECTRA AT GROUND SURFACE FOR CASES
B-1.5, B-3.5 & C-1.5 USING SYNTH TH H2

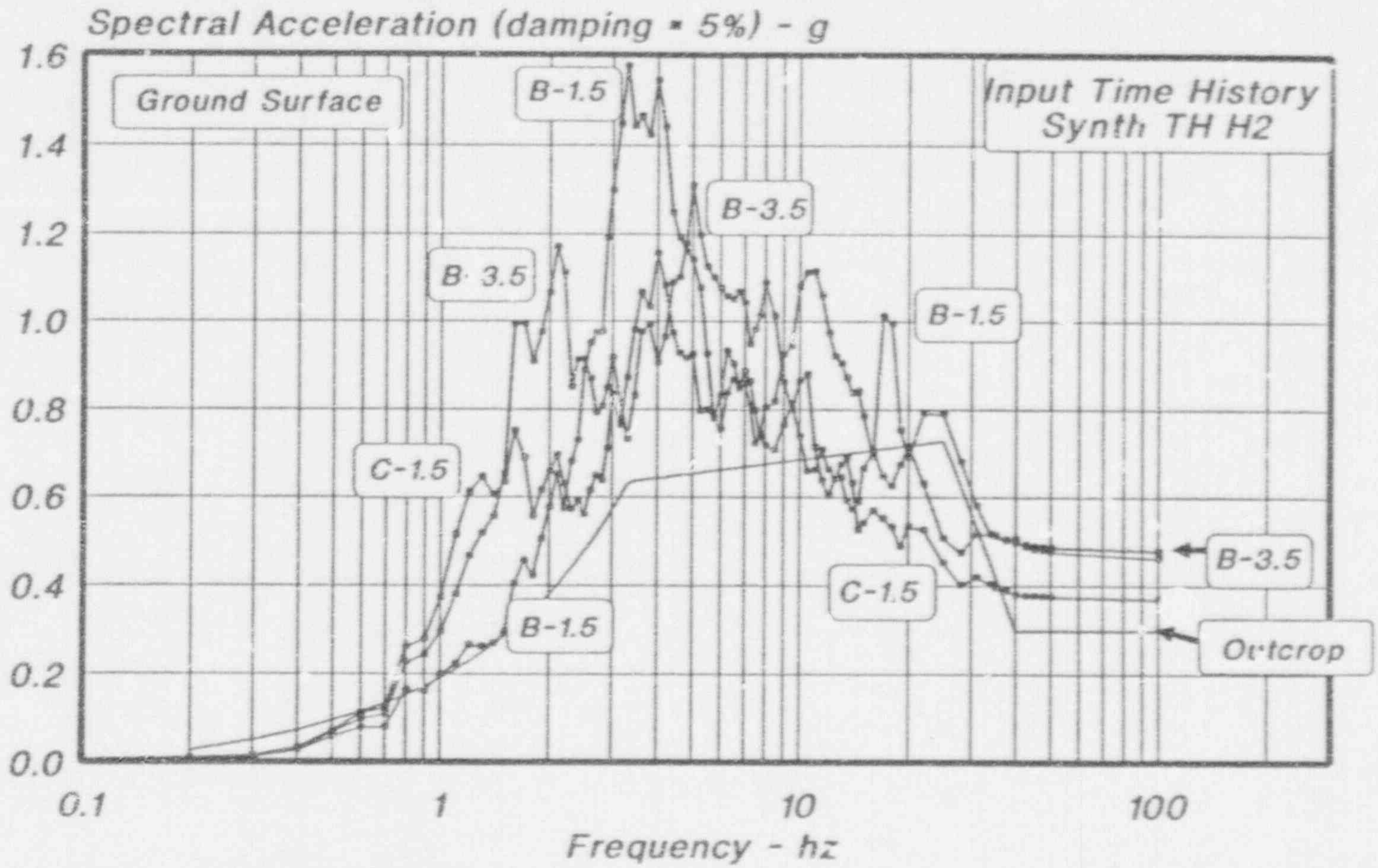

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Fig. 4-14
SPECTRA AT FOUNDATION LEVEL FOR CASES
A-1, B-1 .. B-4 USING SYNTH TH H2

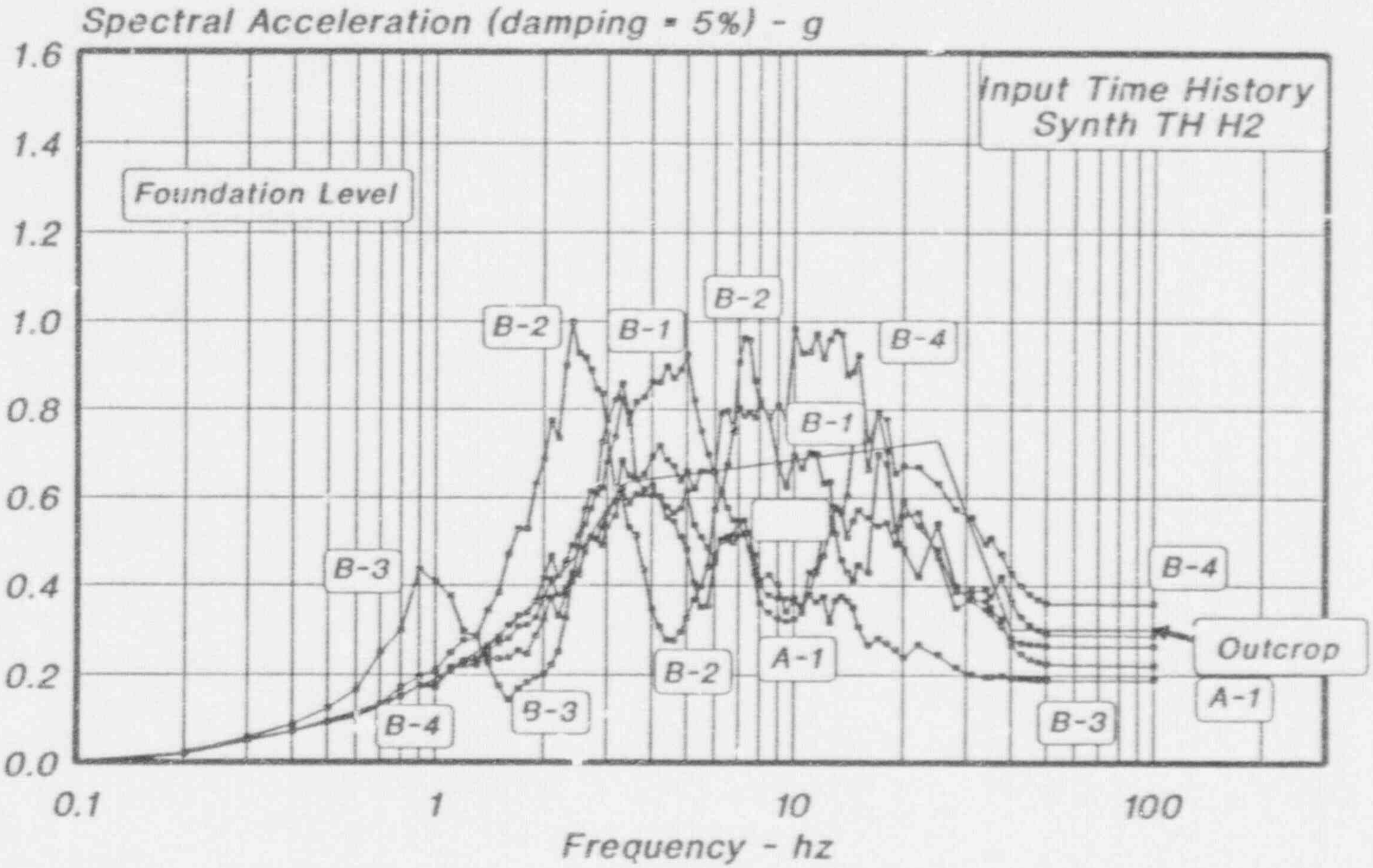

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Fig. 4-15
SPECTRA AT FOUNDATION LEVEL FOR CASES
C-1, C-2, C-3 & D-1 USING SYNTH TH H2

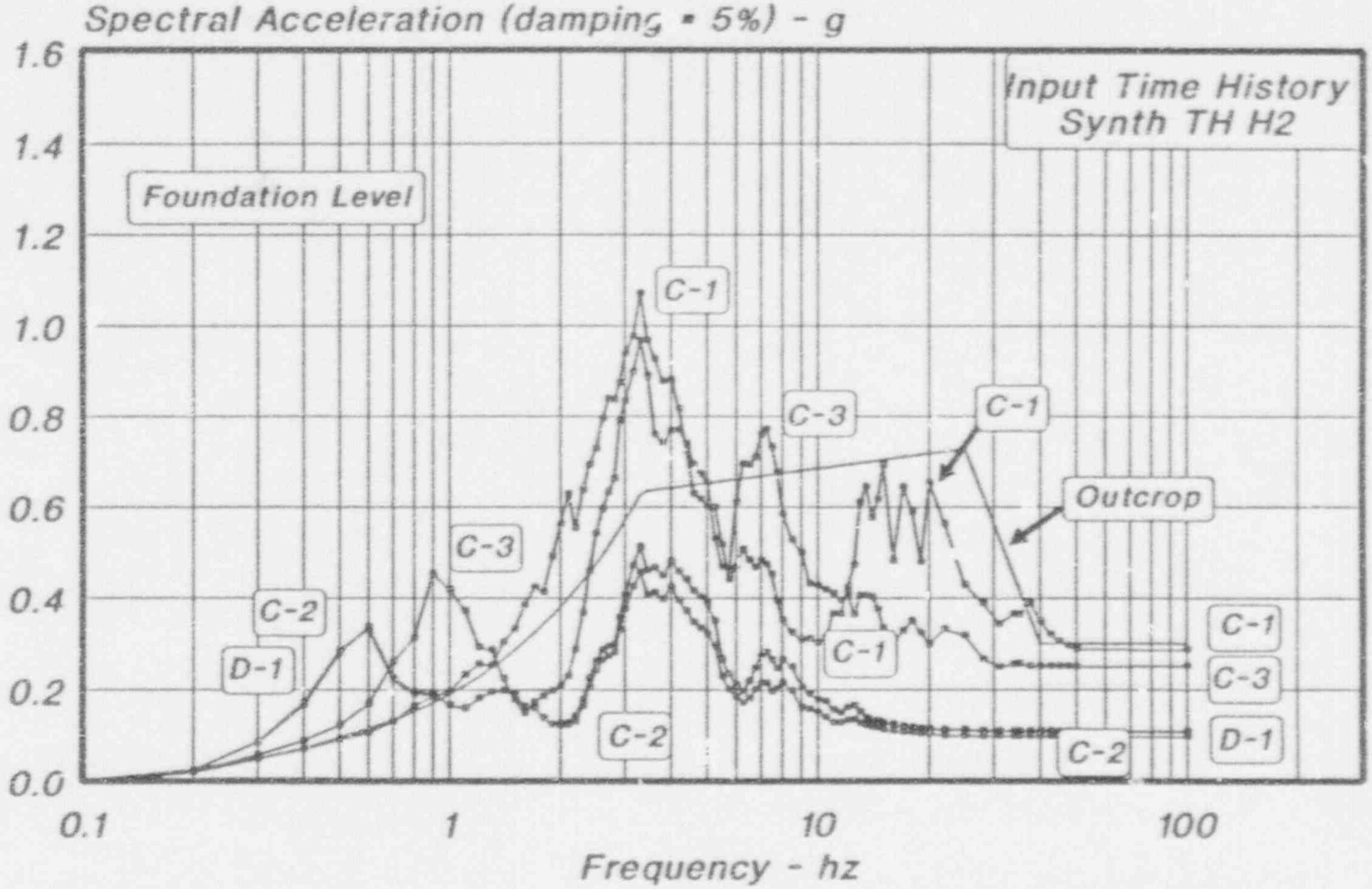

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Fig. 4-16
SPECTRA AT FOUNDATION LEVEL FOR CASES
B-1.5, B-3.5 & C-1.5 USING SYNTH TH H2

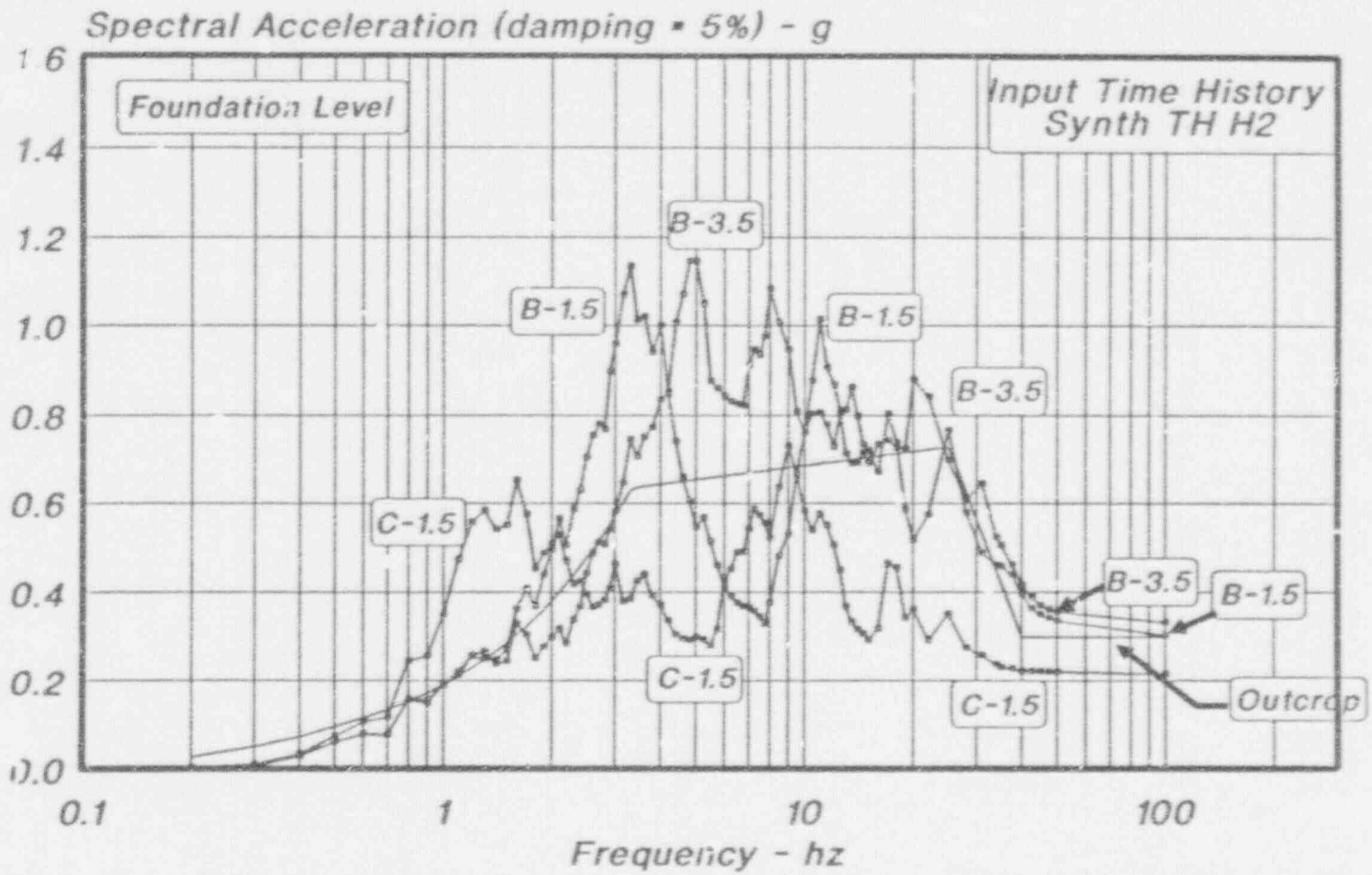

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Fig. 4-17
SPECTRA AT GROUND SURFACE FOR CASES A-1, B-1, B-2, B-3 & B-4 USING VERT SYNTH TH

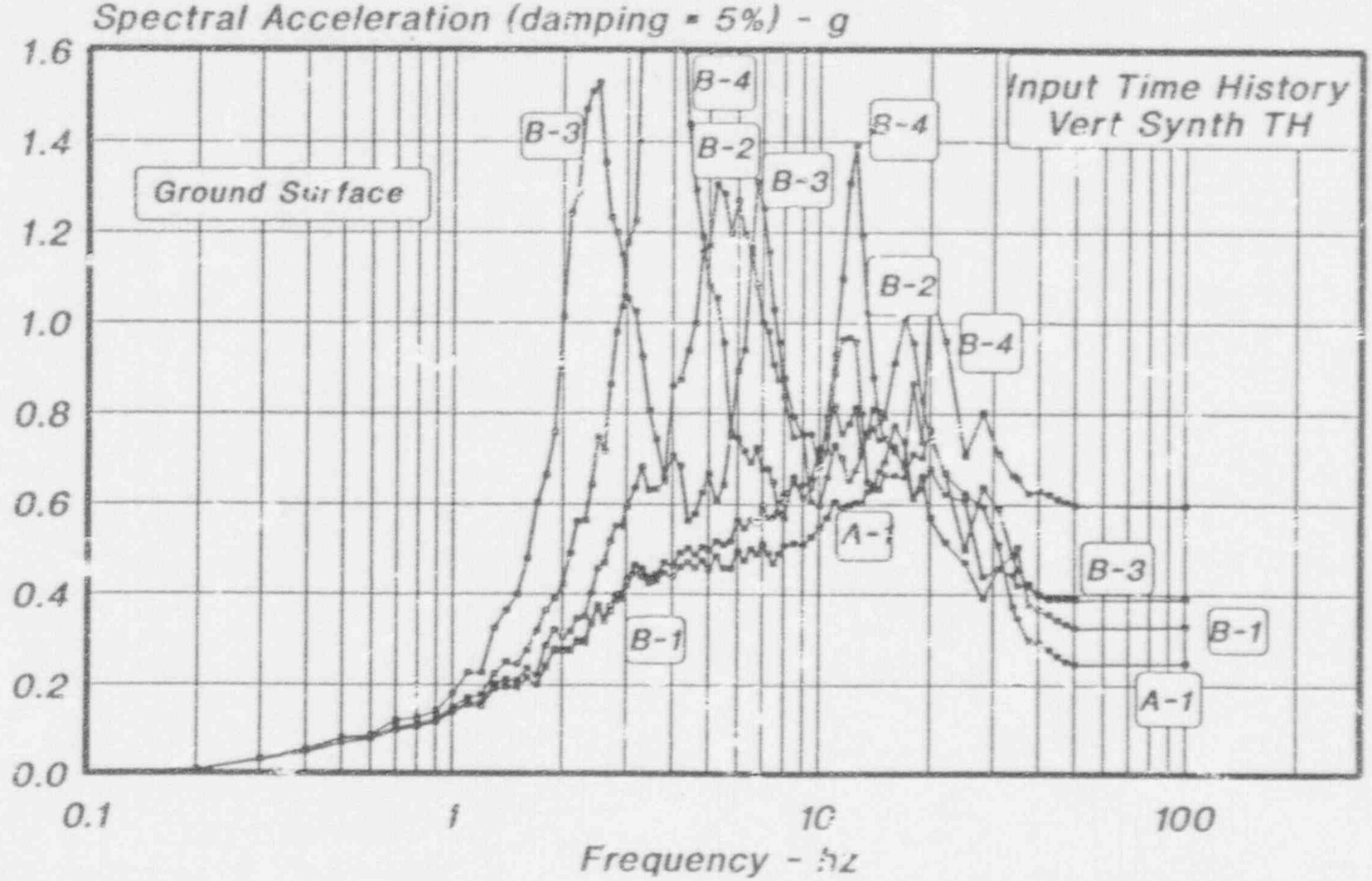

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Fig. 4-18
SPECTRA AT GROUND SURFACE FOR CASES C-1, C-2, C-3 & D-1 USING VERT SYNTH TH

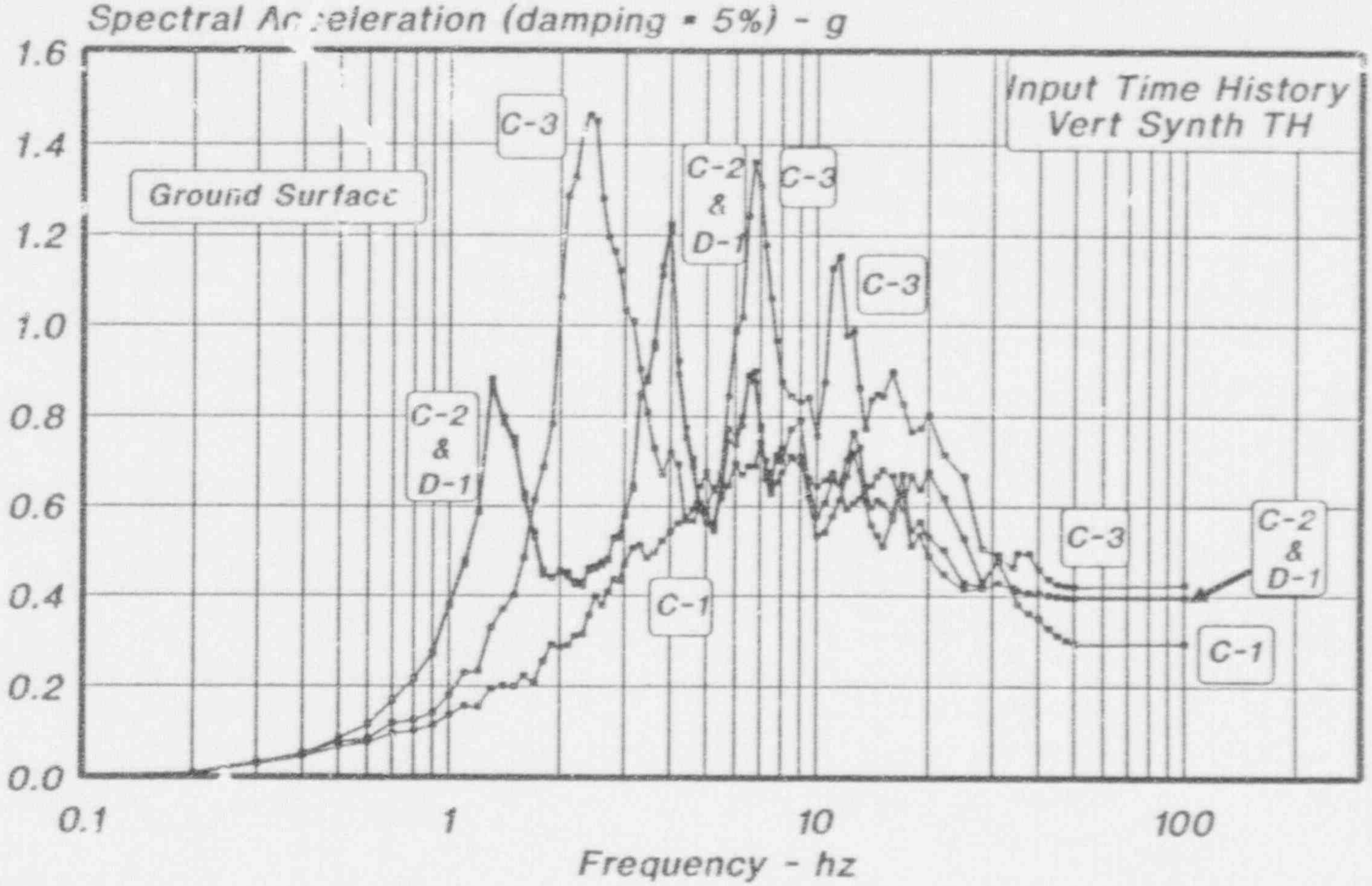

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Fig. 4-19
SPECTRA AT GROUND SURFACE FOR CASES
B-1.5, B-3.5 & C-1.5 USING VERT SYNTH TH

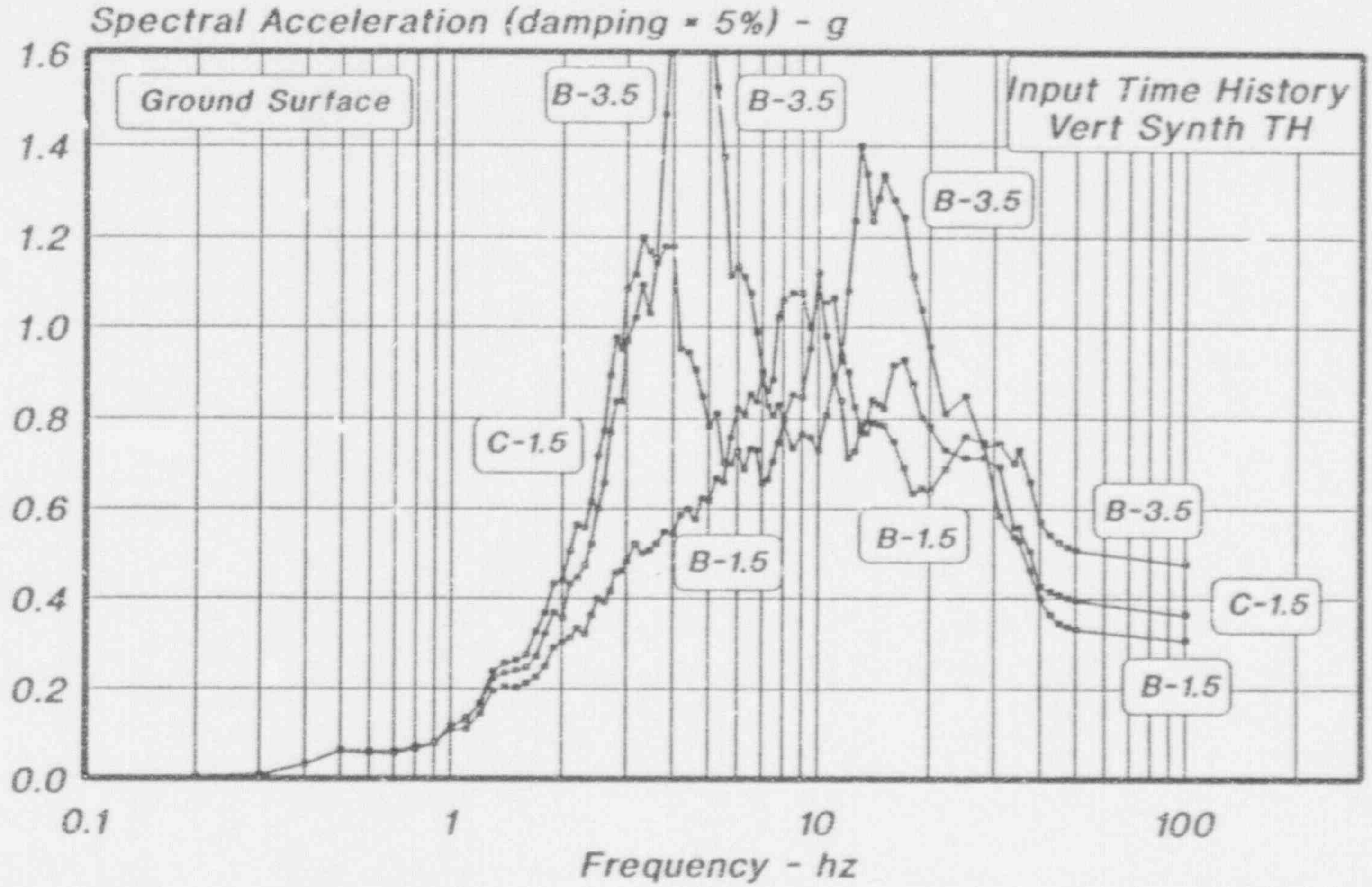

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Fig. 4-20
SPECTRA AT FOUNDATION LEVEL FOR CASES
A-1, B-1 .. B-4 USING VERT SYNTH TH

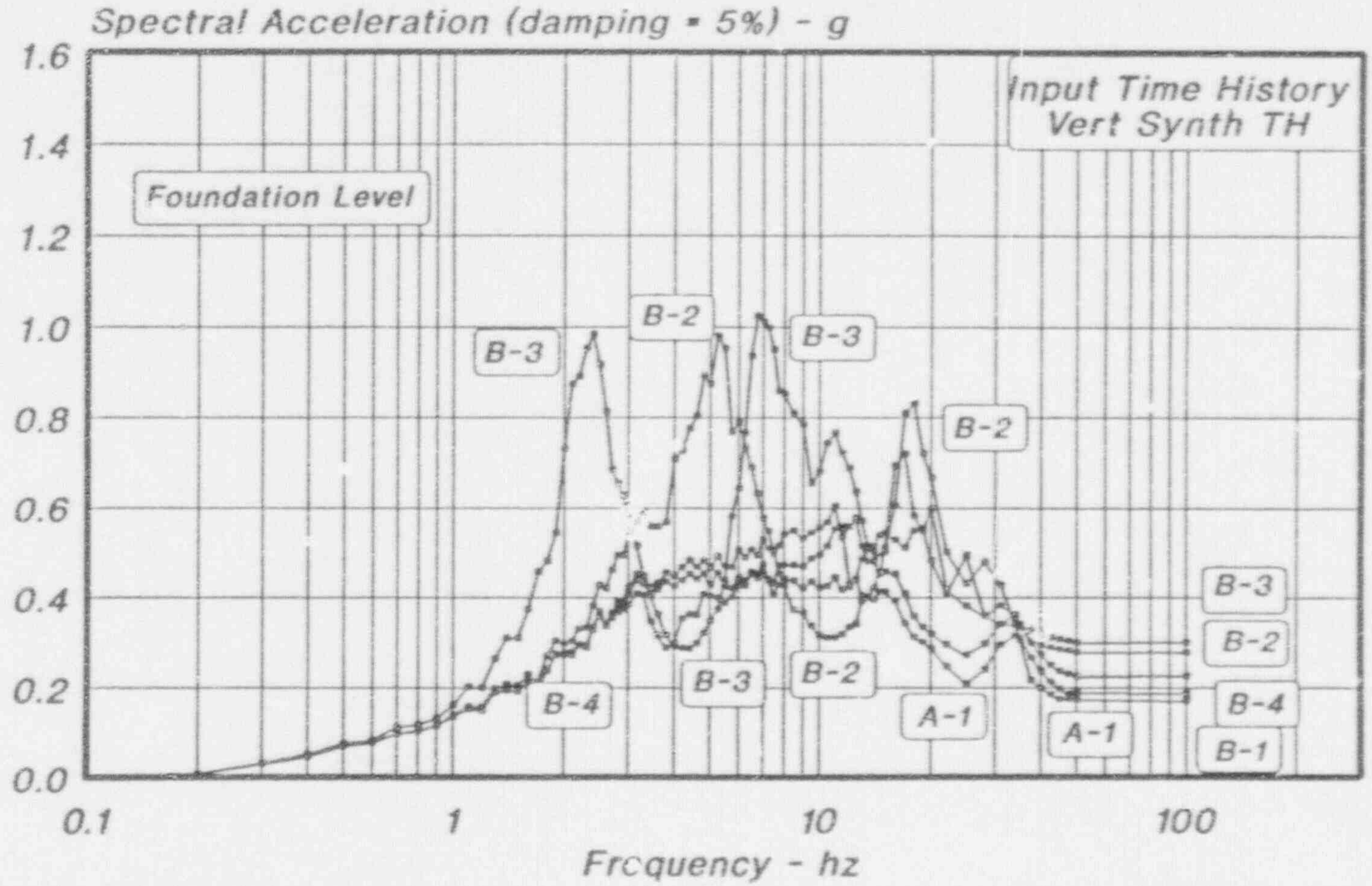

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Fig. 4-21
SPECTRA AT FOUNDATION LEVEL FOR CASES
C-1, C-2, C-3 & D-1 USING VERT SYNTH TH

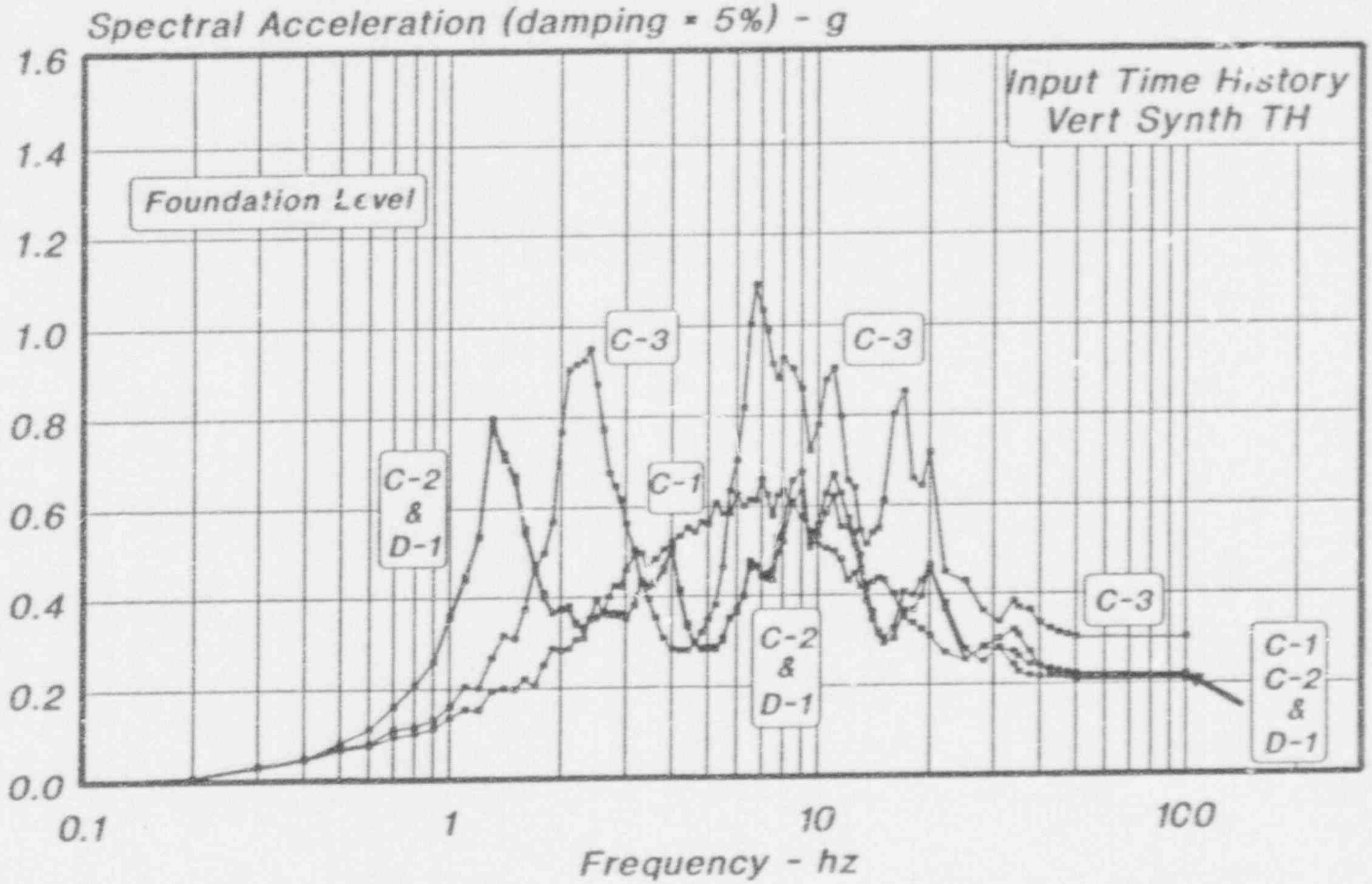

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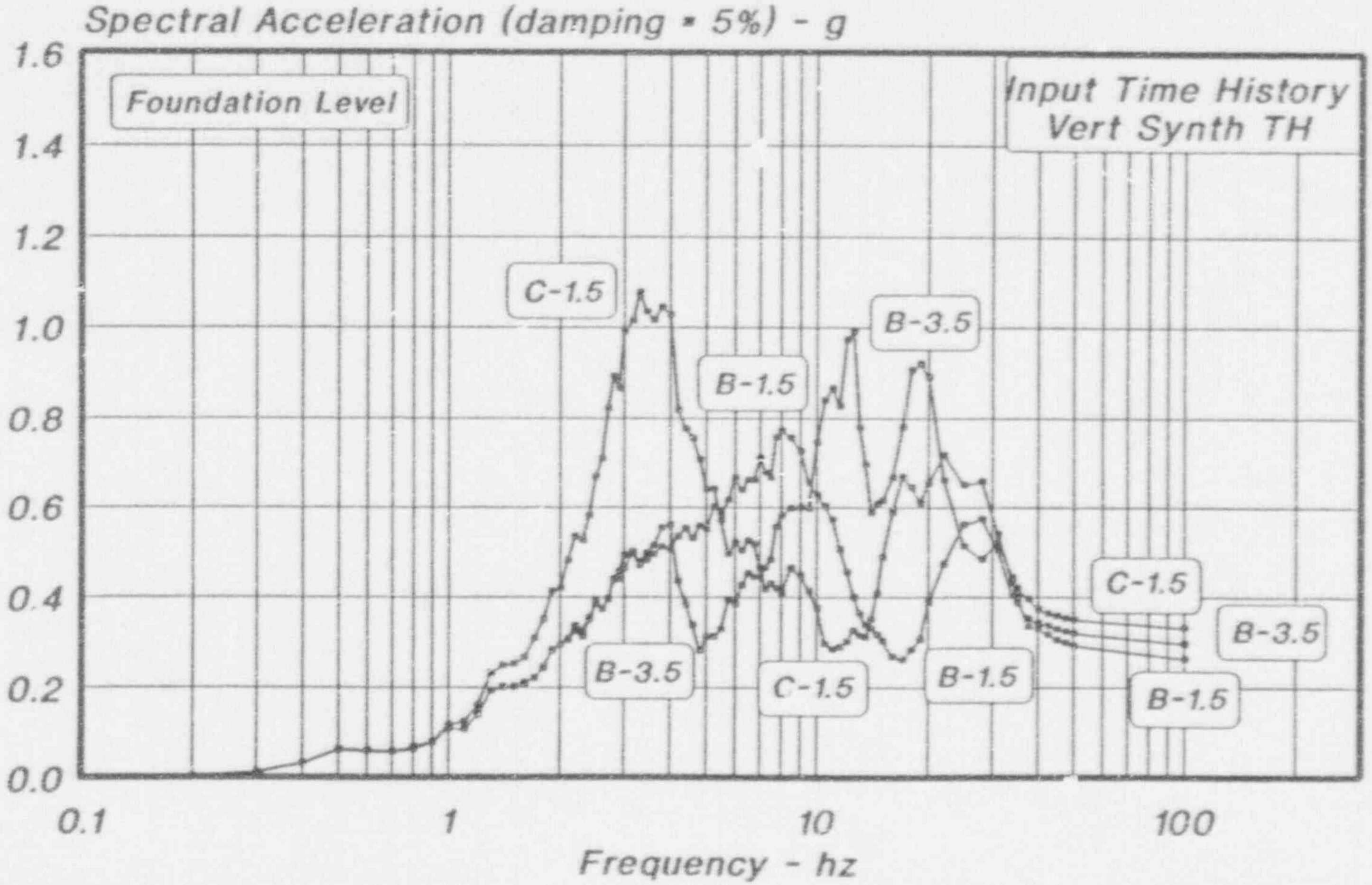
Fig. 4-22
SPECTRA AT FOUNDATION LEVEL FOR CASES
B-1.5, B-3.5 & C-1.5 USING VERT SYNTH TH



5.0 DISCUSSION OF RESULTS

The results presented in this report for the 12 soil cases were obtained using conservative approaches for selecting the free field rock outcrop motion, the range of soil profiles including depths, variation of shear wave velocities with depth and velocity contrasts together with the dynamic material properties.

The results are applicable to a wide range of soil deposits. Thus, a soil profile for which the distribution of maximum shear wave velocities with depth is within the range shown in Fig. 2-5 would have a response well covered by results presented in this report although the results for a specific new case could differ from the results obtained for each of the cases included in this report.

Therefore, a key element for the acceptance criteria would be that the distribution of maximum shear wave velocities with depth be within the range shown in Fig. 2-5. A soil site having a total depth to bedrock greater than that shown in Fig. 2-5 would also be covered. Another key element for the acceptance criteria is that the spectrum for the free field rock outcrop motion be equal to or less than the smooth spectra shown in Fig. 3-2.

Issues related to potential site instability or ground rupture due to steep topography, soft soils, liquefaction or fault rupture should be treated as site-specific issues and dealt with on a case by case basis.

Finally, it may be noted that the analyses presented in this report were based on the distribution of maximum shear wave velocities with depth and thus did not require specification of a depth to water table at the site. Therefore, the water table could be at any depth as long as the variations of maximum shear wave velocities with depth are within the range discussed above and that any local site instability issues are resolved.

6.0 ACKNOWLEDGEMENTS

7.0 REFERENCES

These sections are to be finalized later.

Fig. A-1

GENERIC SITE CATEGORIES

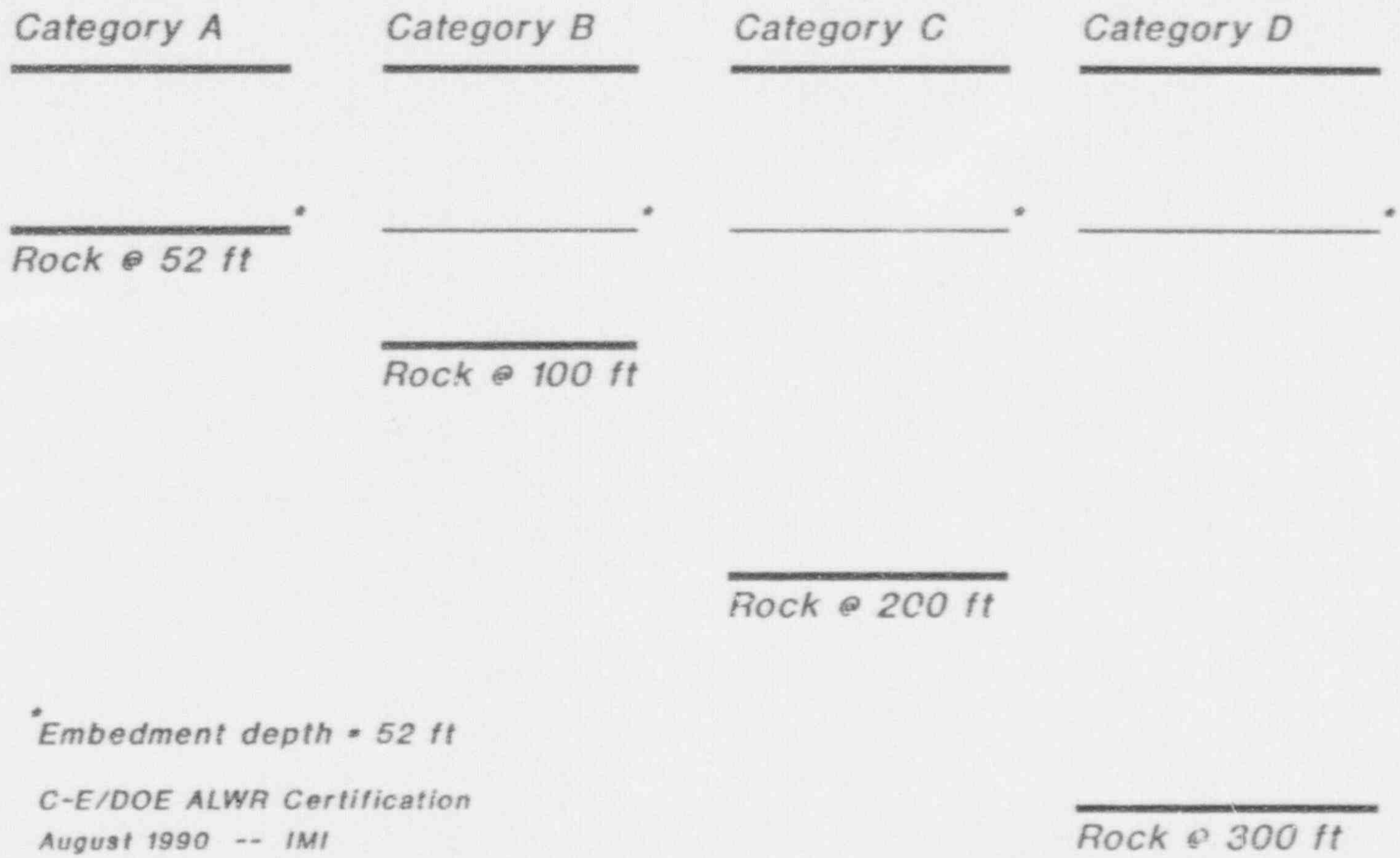


Fig. A-2
ASSIGNED SHEAR WAVE VELOCITIES
CASE A-1

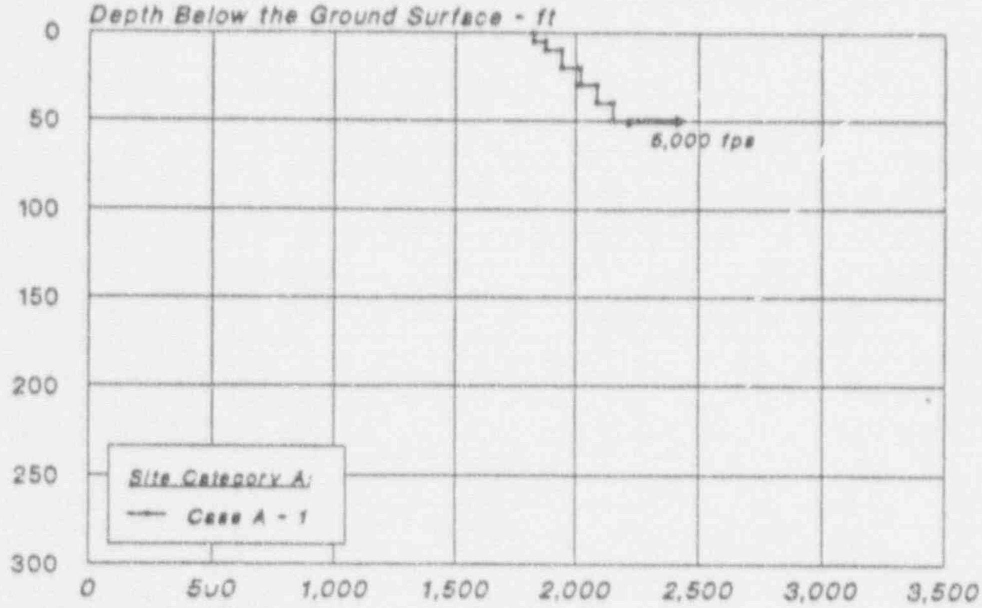


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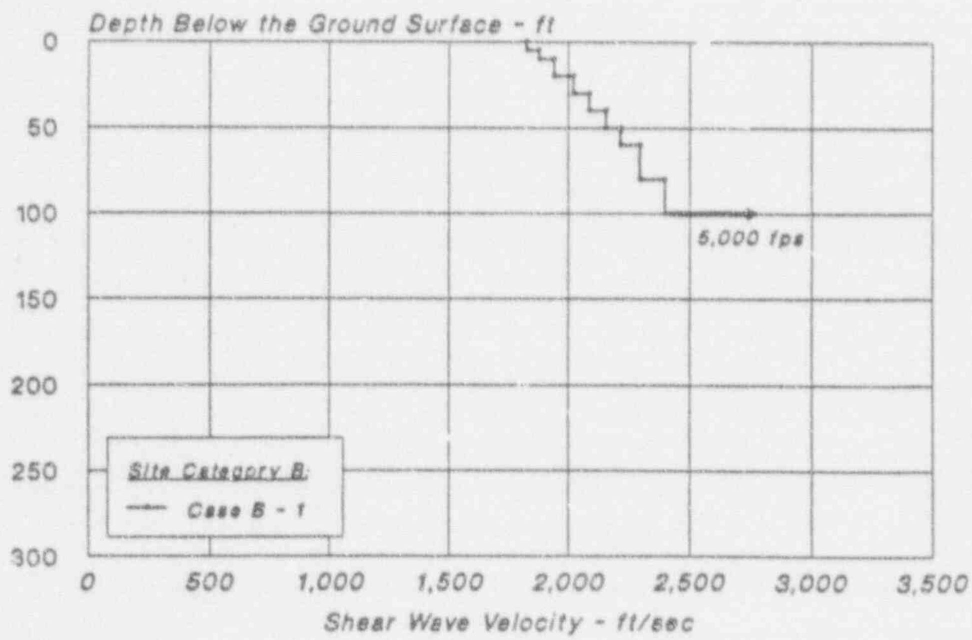
Fig. A-3
ASSIGNED SHEAR WAVE VELOCITIES
CASE B-1



August 1990 -- IMI

Fig. A-4
ASSIGNED SHEAR WAVE VELOCITIES
CASE B-2

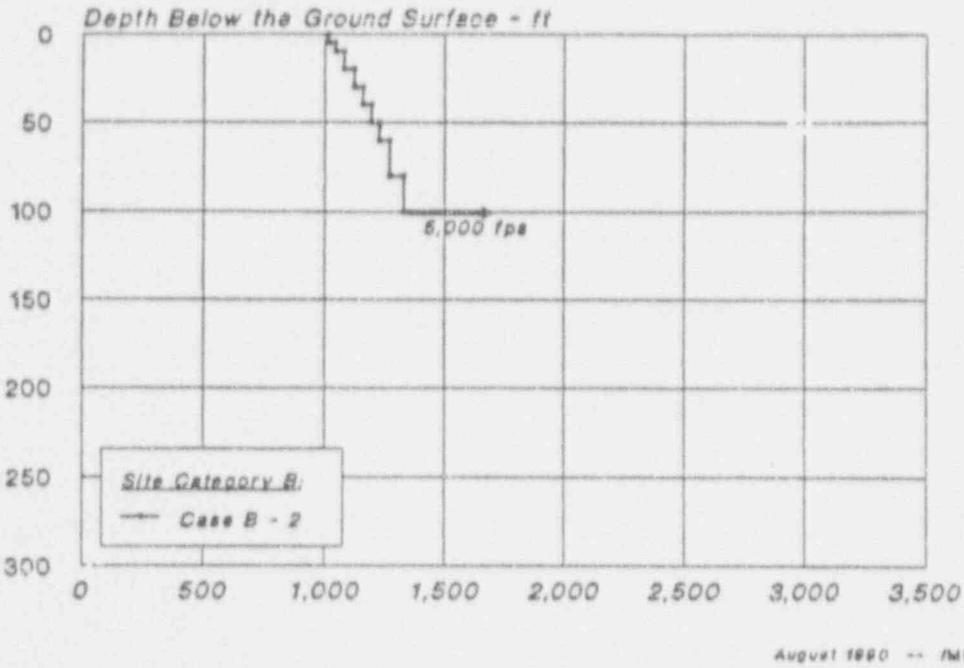


Fig. A-5
ASSIGNED SHEAR WAVE VELOCITIES
CASE B-3

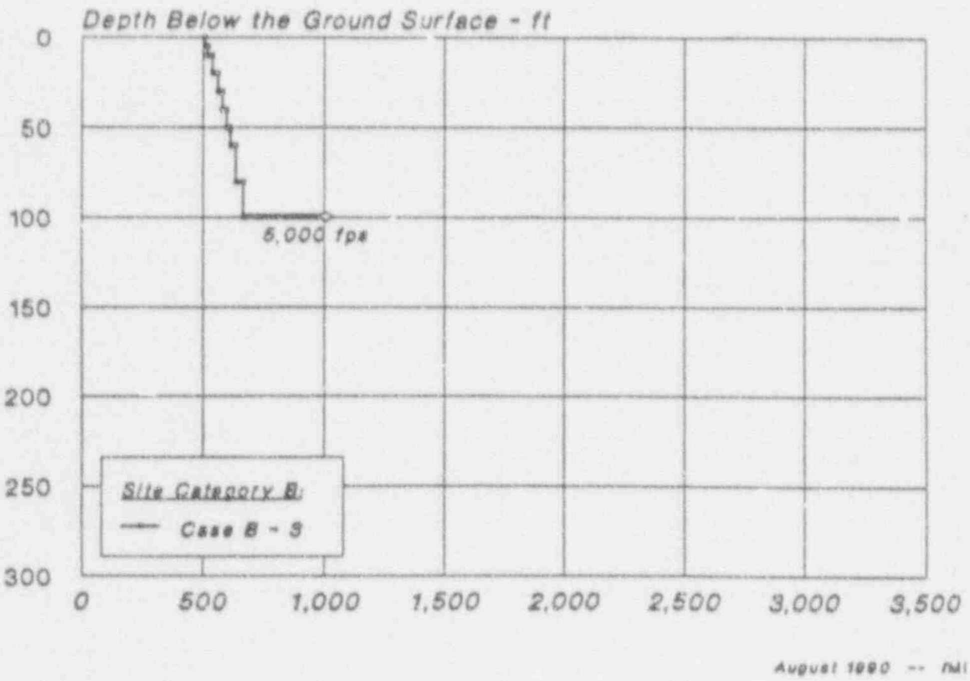
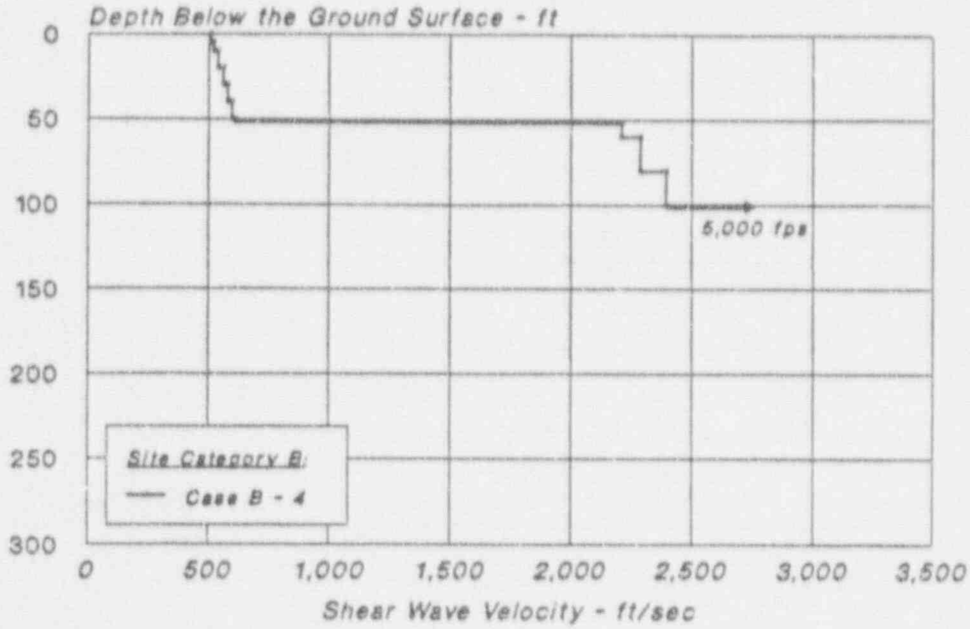
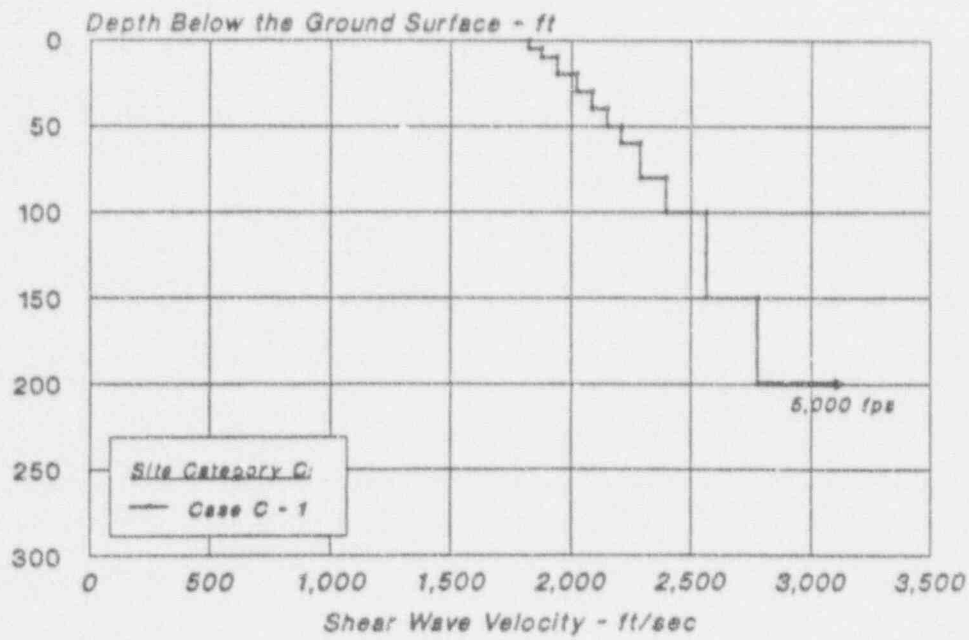


Fig. A-6
ASSIGNED SHEAR WAVE VELOCITIES
CASE B-4



August 1990 -- RMI

Fig. A-7
ASSIGNED SHEAR WAVE VELOCITIES
CASE C-1



August 1990 -- RMI

Fig. A-8
ASSIGNED SHEAR WAVE VELOCITIES
CASE C-2

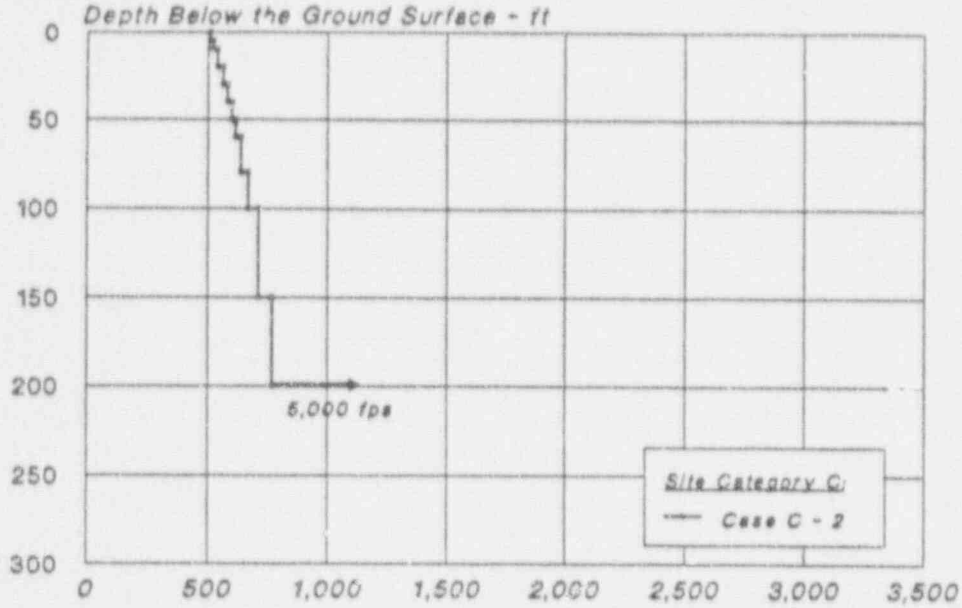


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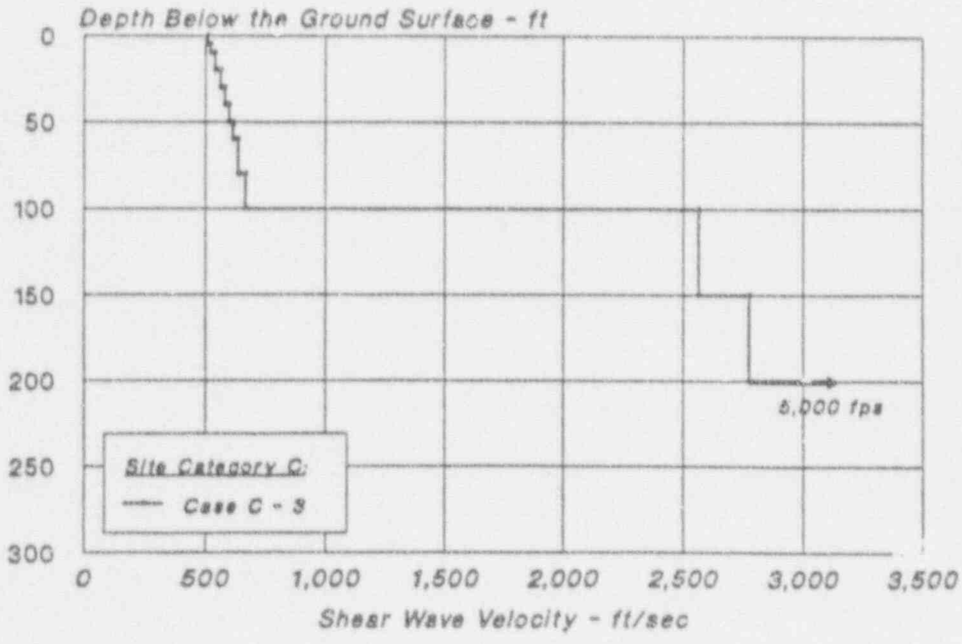
Fig. A-9
ASSIGNED SHEAR WAVE VELOCITIES
CASE C-3



August 1990 -- IMI

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

Fig. A-10
ASSIGNED SHEAR WAVE VELOCITIES
CASE D-1

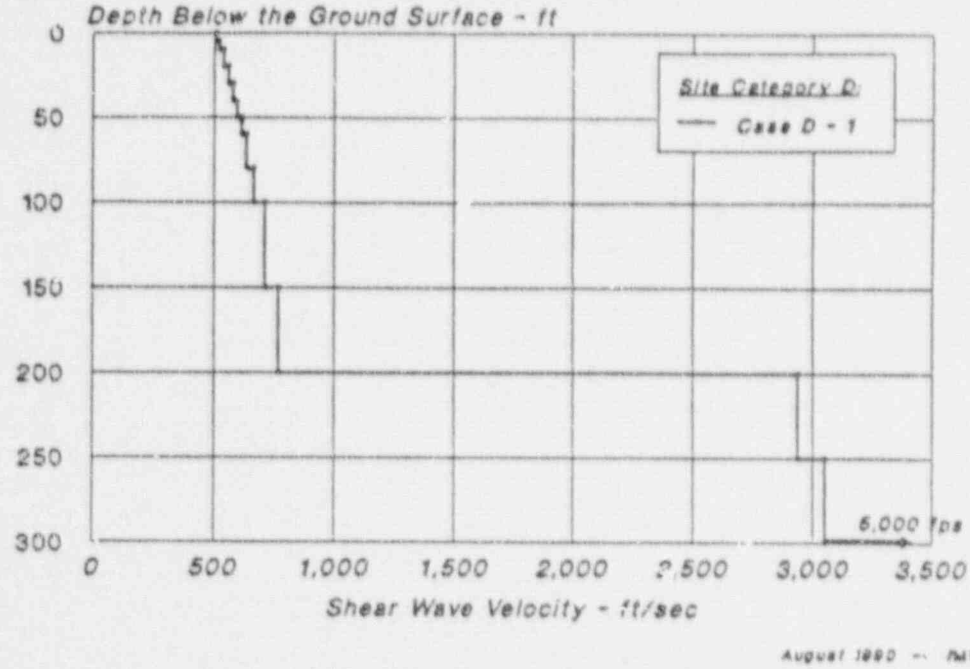


Fig. A-11
ASSIGNED SHEAR WAVE VELOCITIES
CASE B-1.5

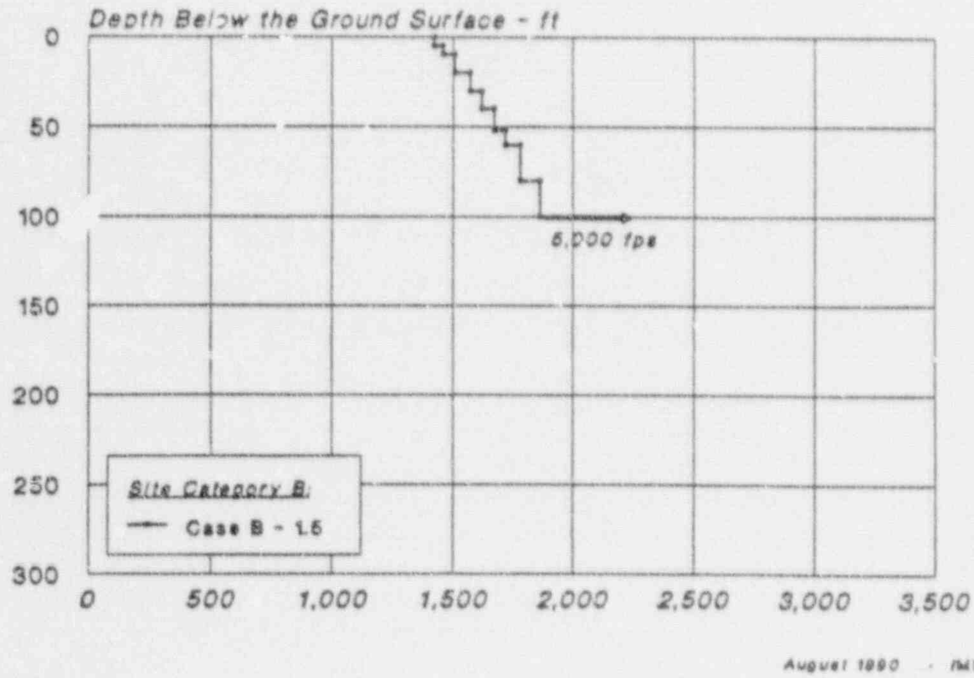


Fig. A-12
ASSIGNED SHEAR WAVE VELOCITIES
CASE B-3.5

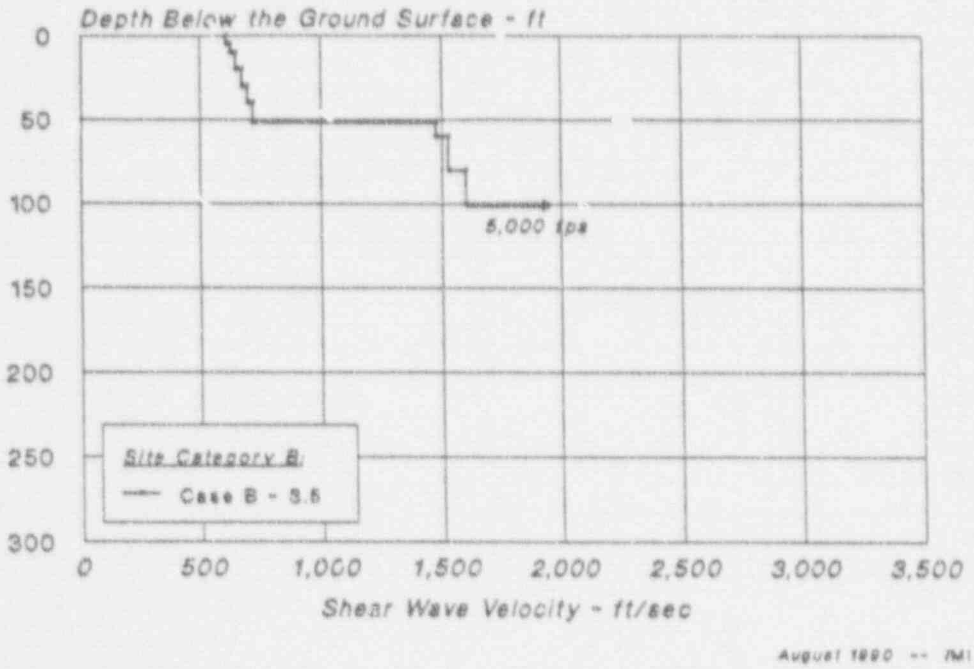
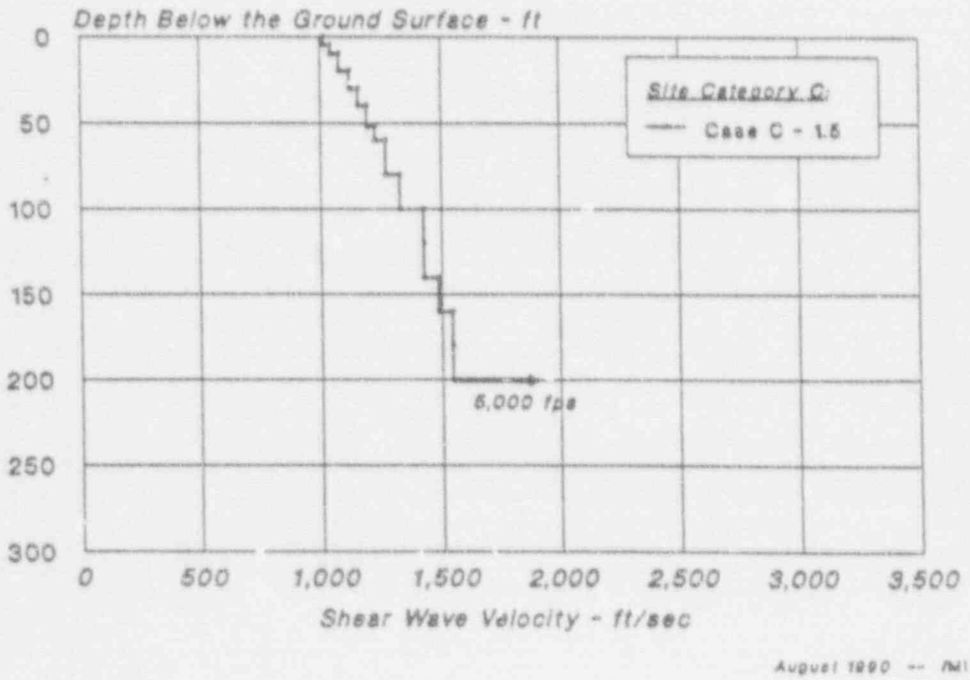


Fig. A-13
ASSIGNED SHEAR WAVE VELOCITIES
CASE C-1.5



APPENDIX B

CHARACTERISTICS OF
SELECTED CONTROL MOTIONS

EARTHQUAKE
GROUND MOTIONS

SELECTION OF
CONTROL MOTION
AND
DEVELOPMENT OF
GENERIC SOIL SITES

DONE IN CONJUNCTION WITH
C-E/DOE ALWR CERTIFICATION

September 4, 1990

APPENDIX B

CHARACTERISTICS OF SELECTED CONTROL MOTIONS

The synthetic time histories generated to represent horizontal components H1 and H2 and to represent the vertical component are presented in this Appendix. The acceleration, velocity and displacement time histories for component H1 are presented in Figs. B-1 through B-3, respectively. Figure B-4 shows the Fourier amplitudes and Figs.B-5 and B-6 show the power spectral density (PSD) and cumulative PSD for this component. Corresponding parameters for component H2 are shown in Figs. B-7 through B-12 and those for the vertical components in Figs. B-13 through B-18.

The correlation coefficients for components H1 and H2 are presented in Fig. B-19, those for H1 and V in Fig. B-20 and those for H2 and V in Fig. B-21.

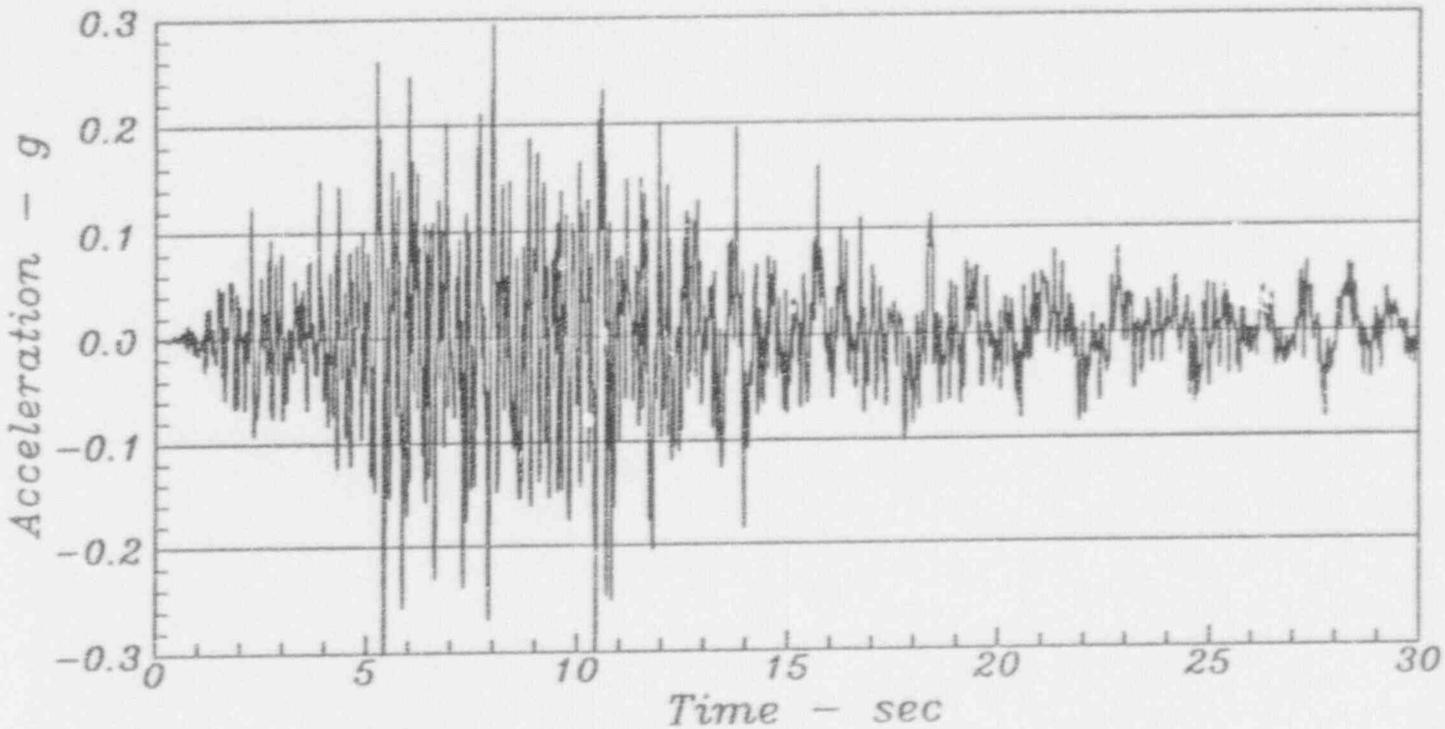


Fig. B-1

SYNTHETIC ACCELERATION TIME HISTORY
COMPONENT H1

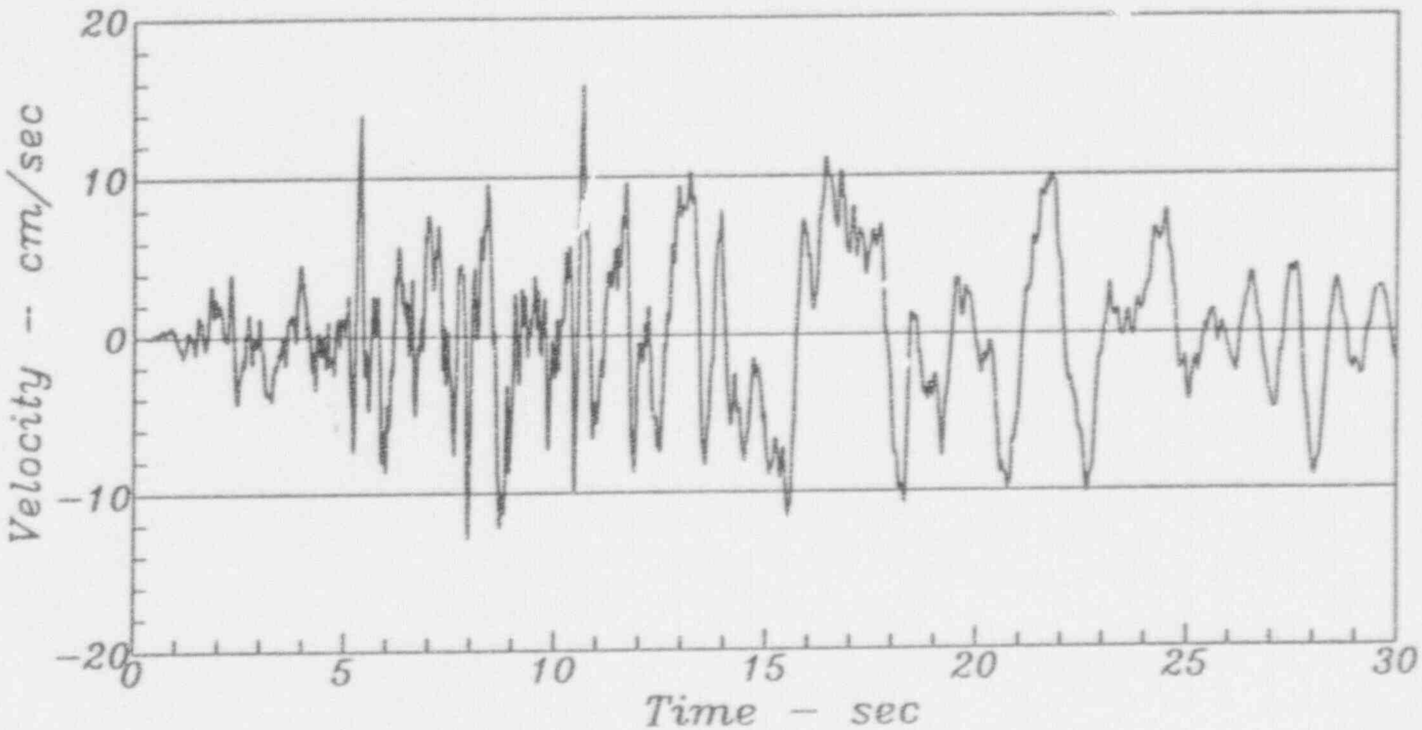


Fig. B-2

VELOCITY -- SYNTHETIC TIME HISTORY
COMPONENT H1

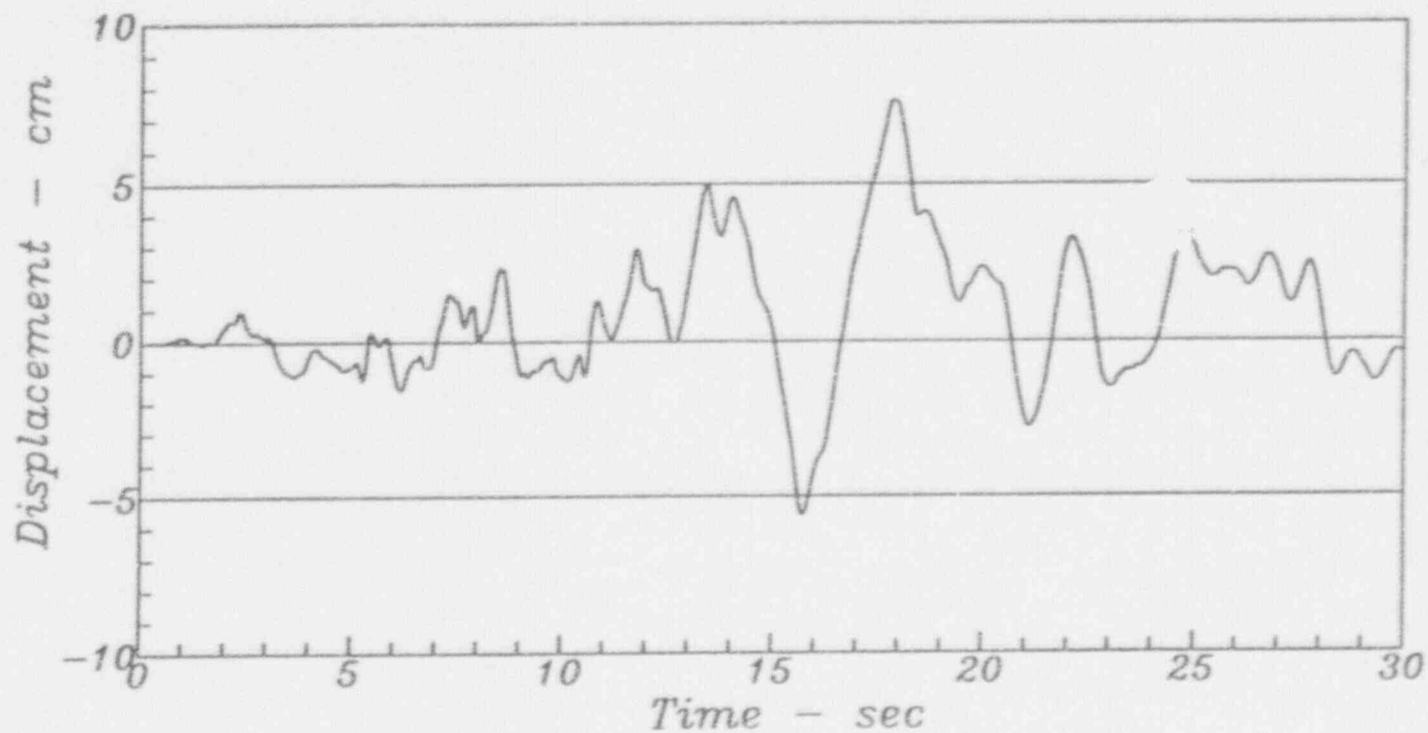


Fig. B-3

DISPLACEMENT -- SYNTHETIC TIME HISTORY
COMPONENT H1

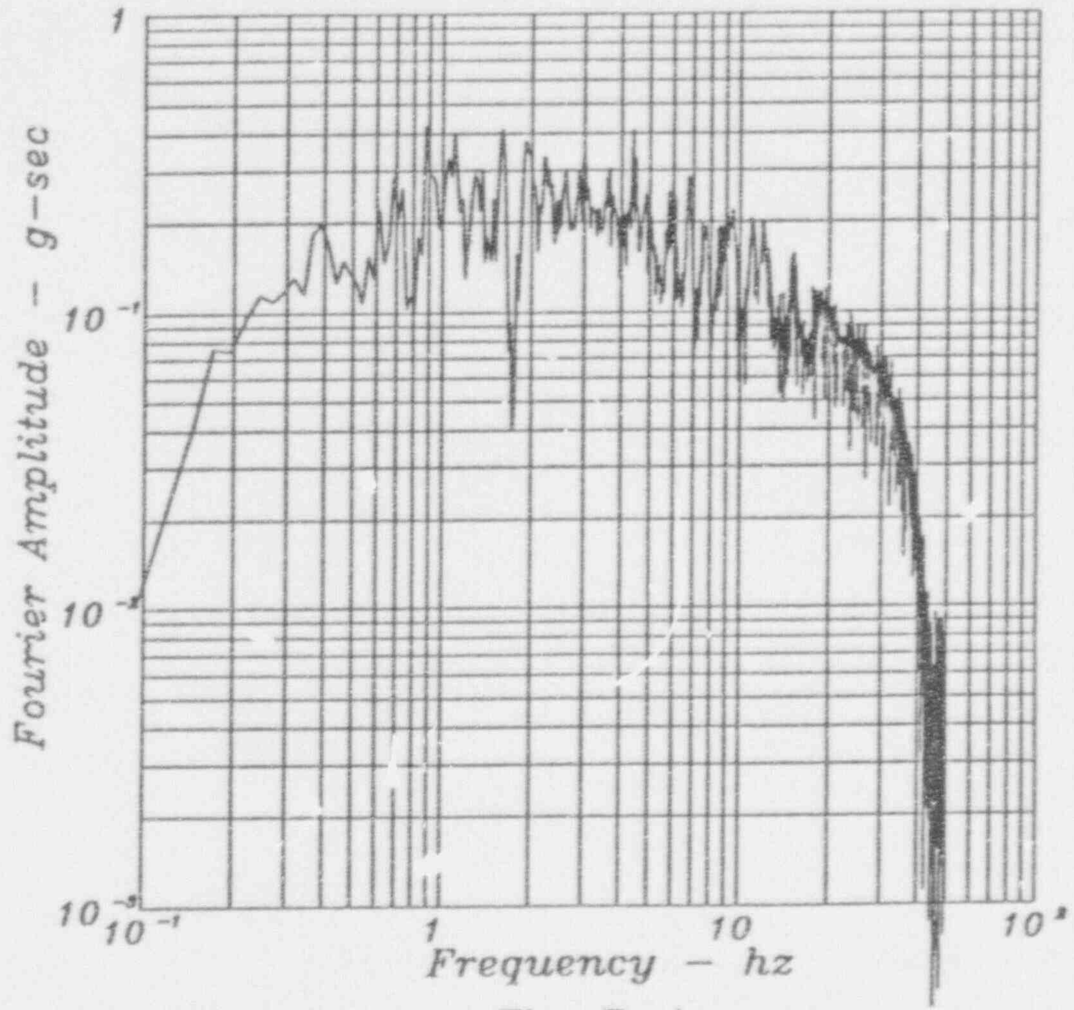


Fig. B-4
FOURIER AMPLITUDES -- SYNTHETIC TIME HISTORY
COMPONENT H1

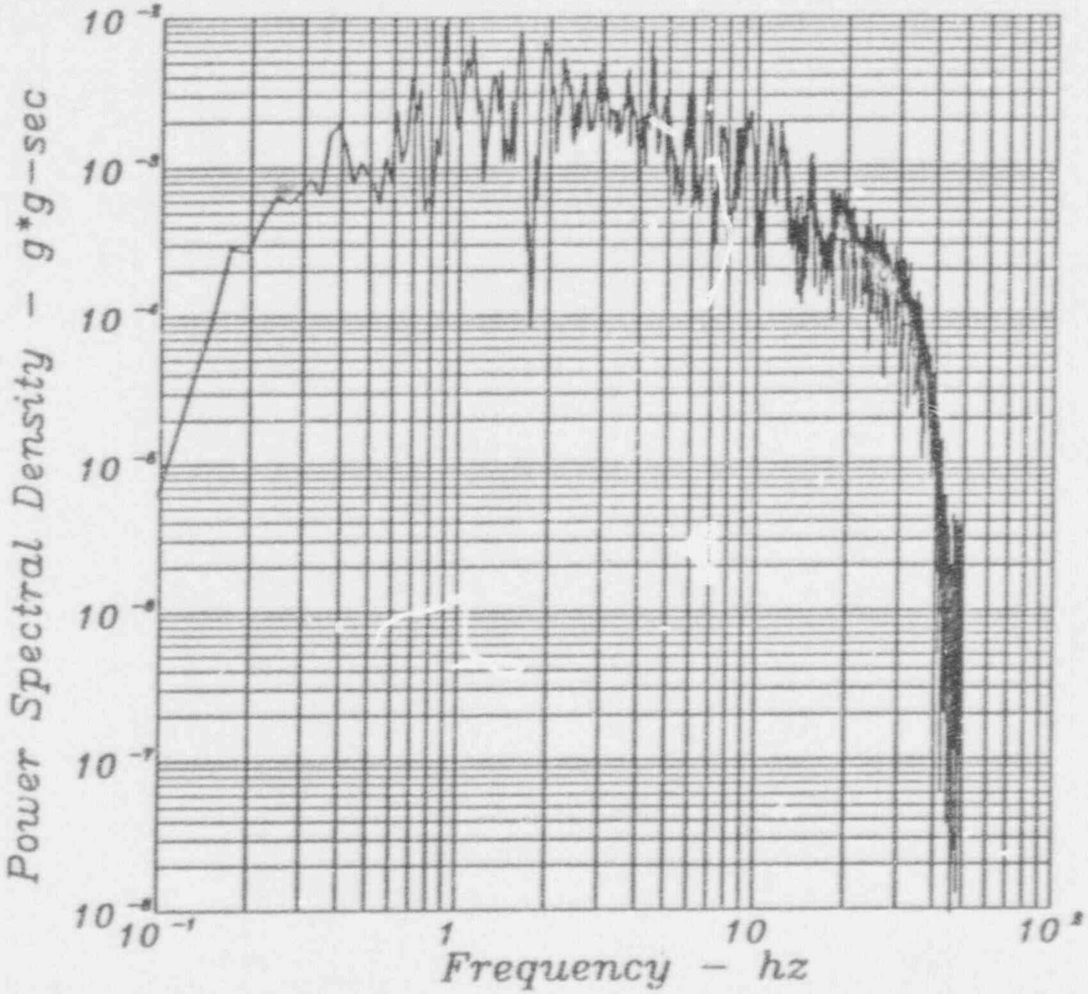


Fig. B-5
POWER SPECTRAL DENSITY -- SYNTHETIC TIME HISTORY
COMPONENT H1

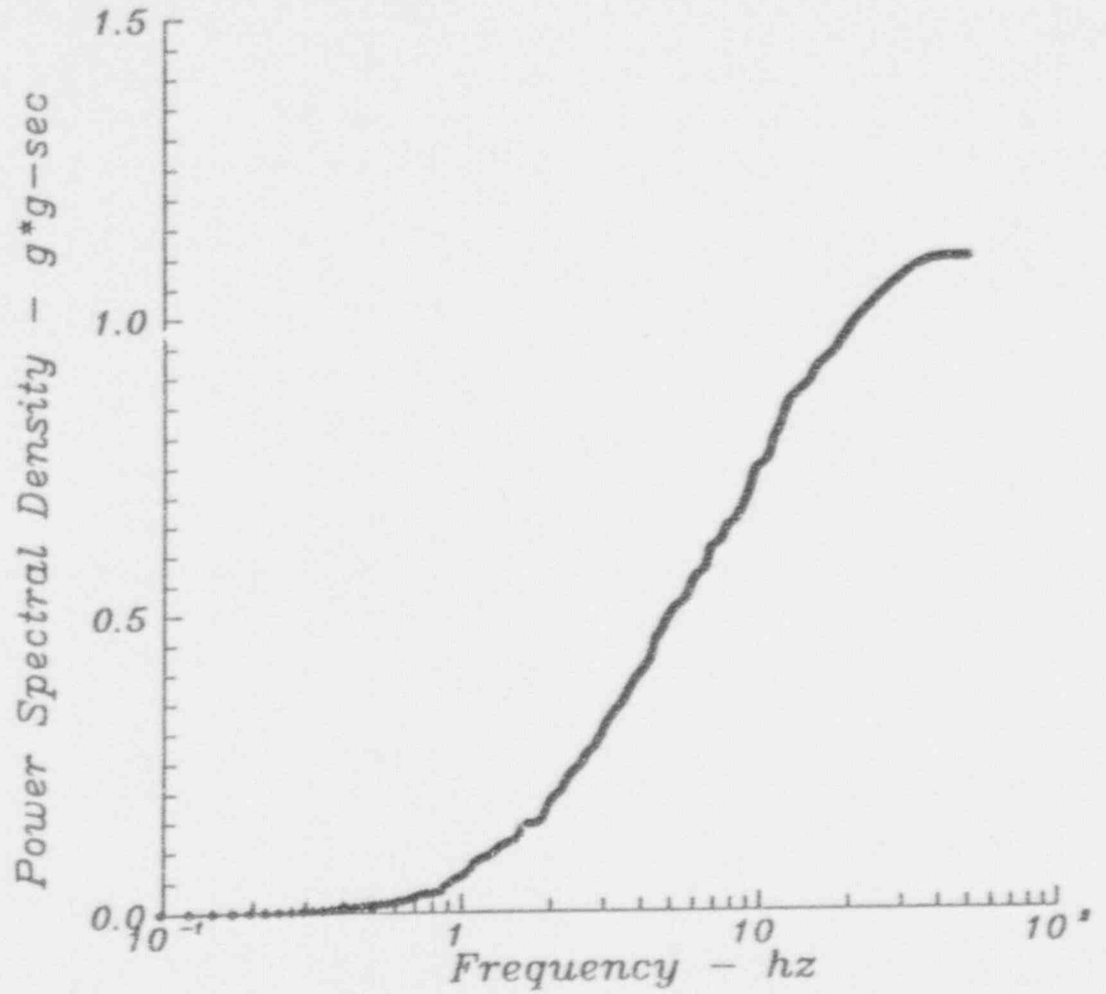


Fig. B-6
CUMULATIVE PSD -- SYNTHETIC TIME HISTORY
COMPONENT H1

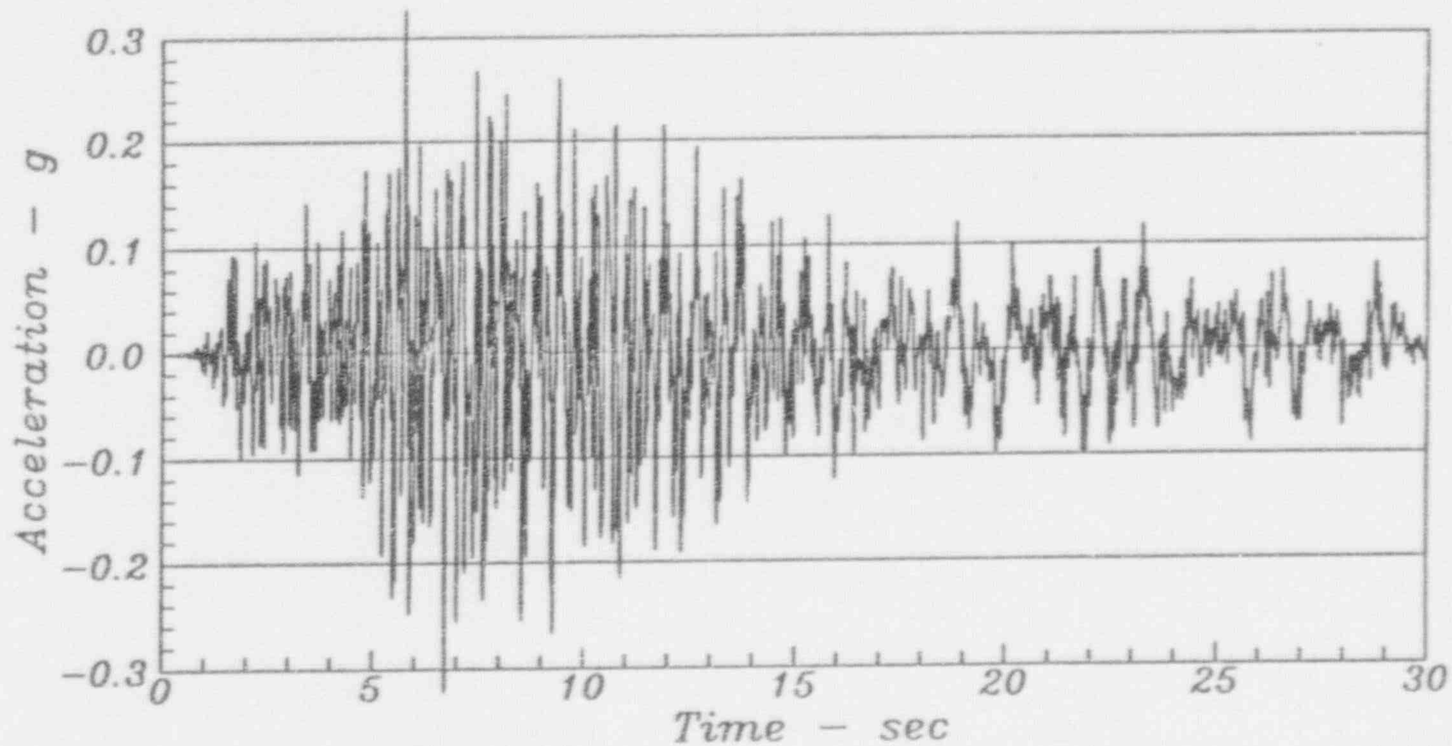


Fig. B-7

SYNTHETIC ACCELERATION TIME HISTORY
COMPONENT H2

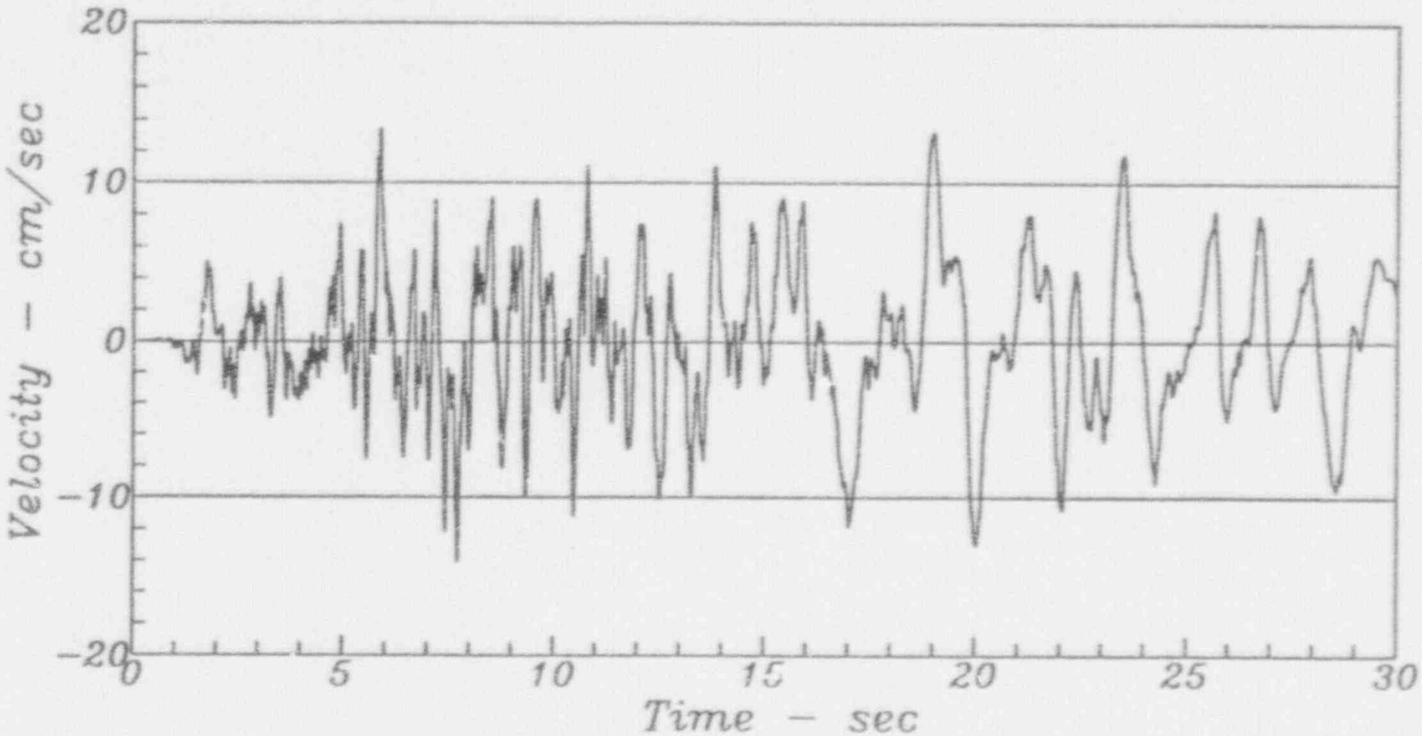


Fig. B-8

VELOCITY -- SYNTHETIC TIME HISTORY
COMPONENT H2

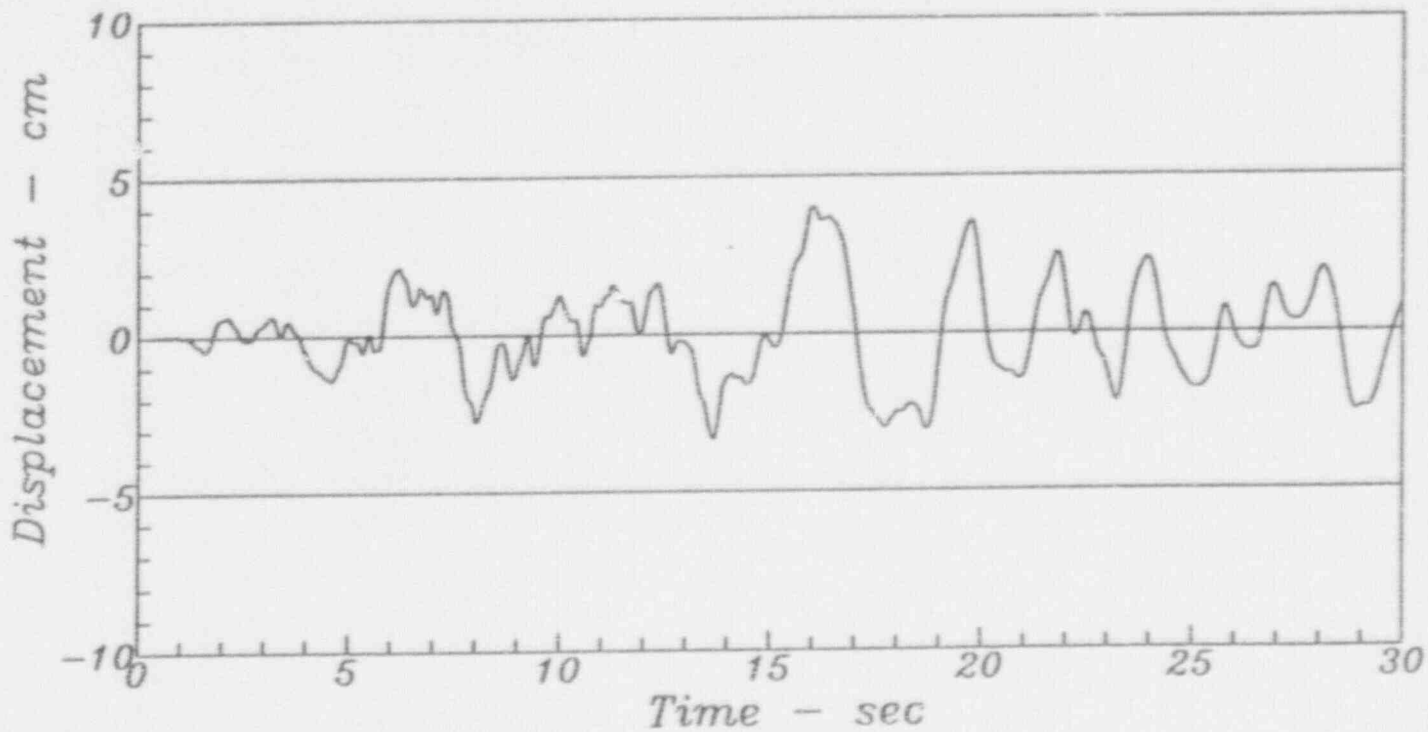


Fig. B-9

DISPLACEMENT -- SYNTHETIC TIME HISTORY
COMPONENT H2

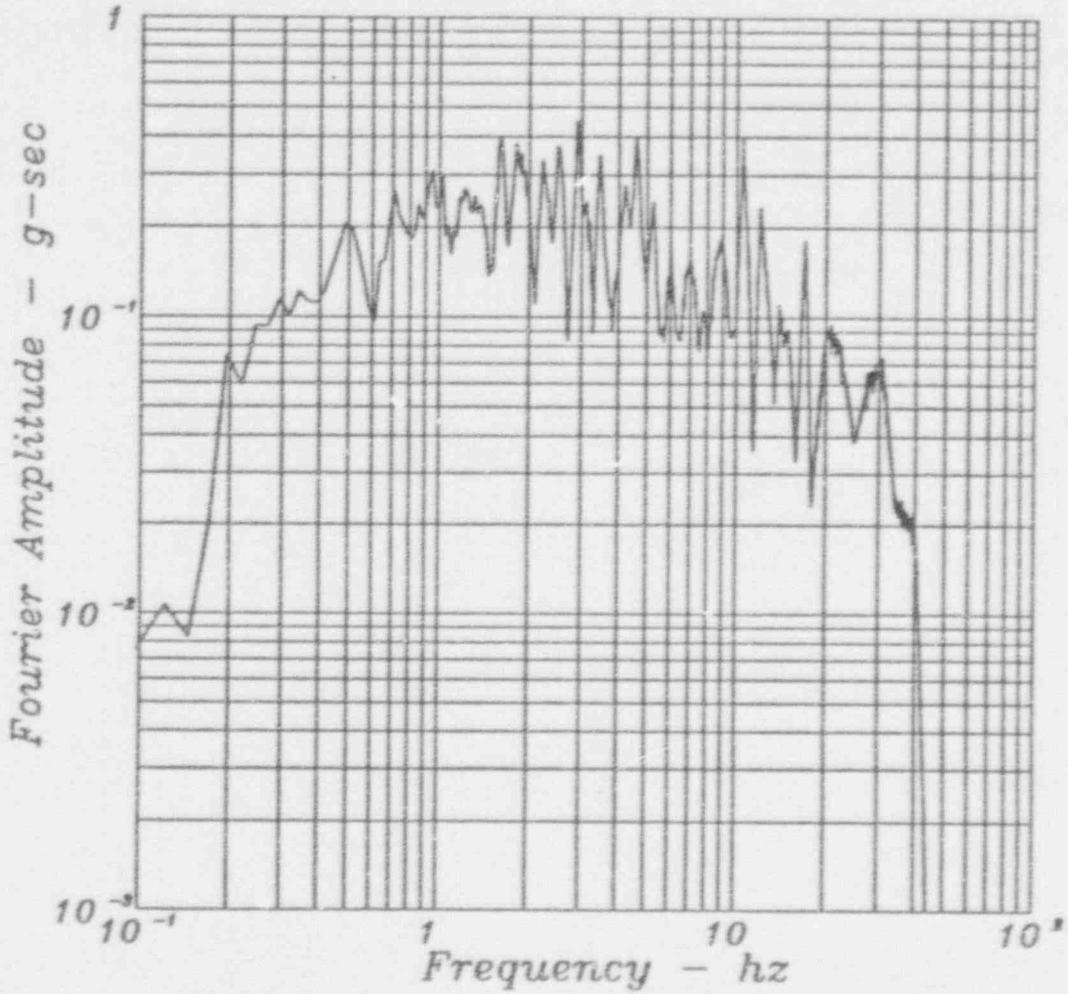


Fig. B-10
FOURIER AMPLITUDES -- SYNTHETIC TIME HISTORY
COMPONENT H2

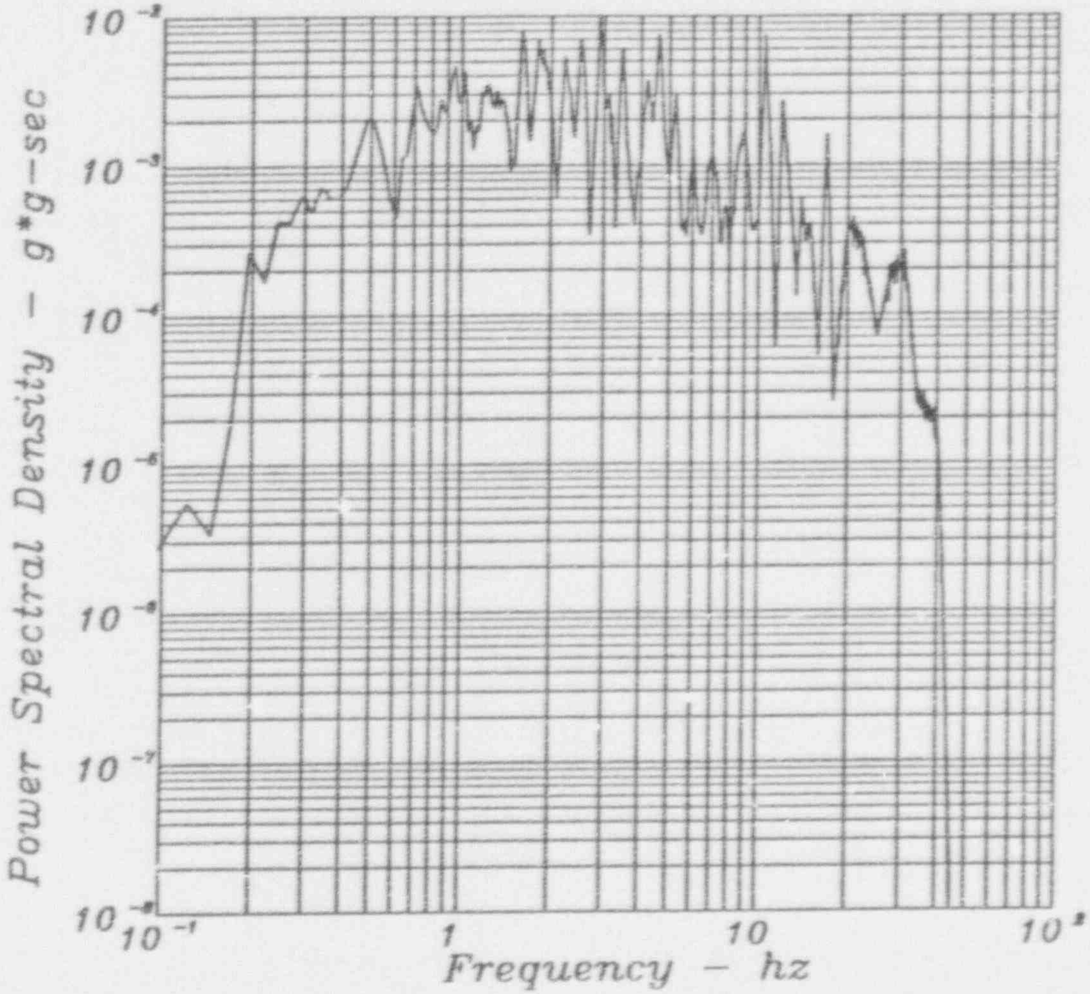


Fig. B-11
POWER SPECTRAL DENSITY -- SYNTHETIC TIME HISTORY
COMPONENT H2

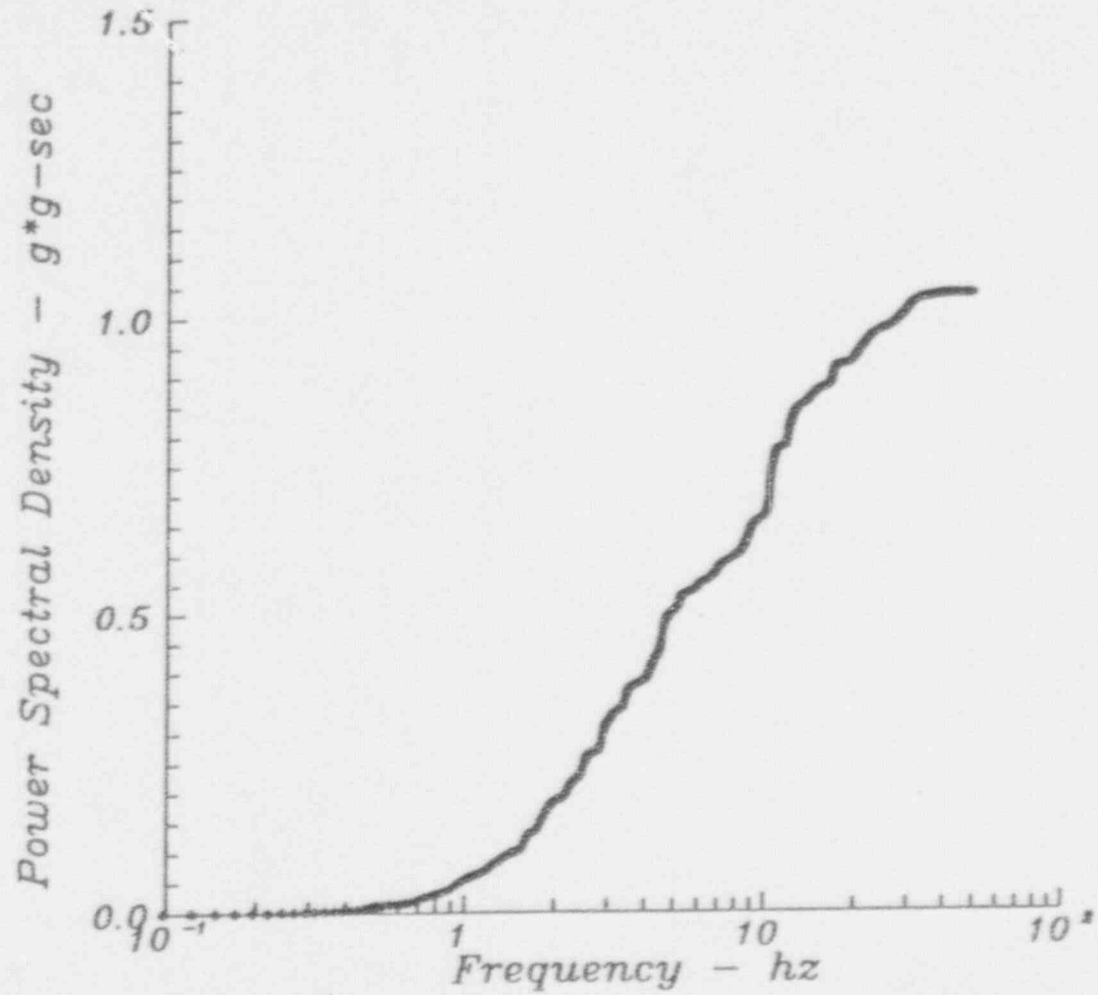


Fig. B-12
CUMULATIVE PSD -- SYNTHETIC TIME HISTORY
COMPONENT H2

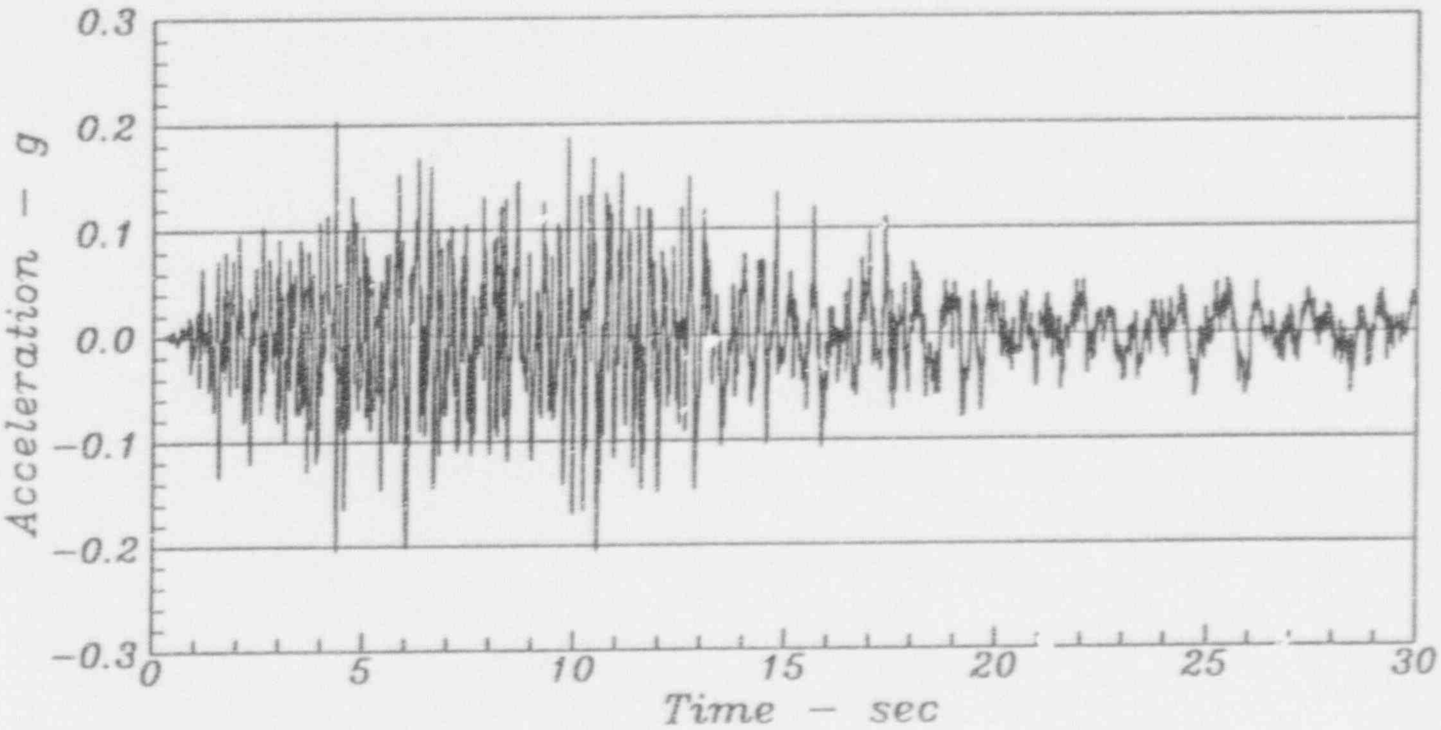


Fig. B-13

SYNTHETIC ACCELERATION TIME HISTORY
VERTICAL COMPONENT

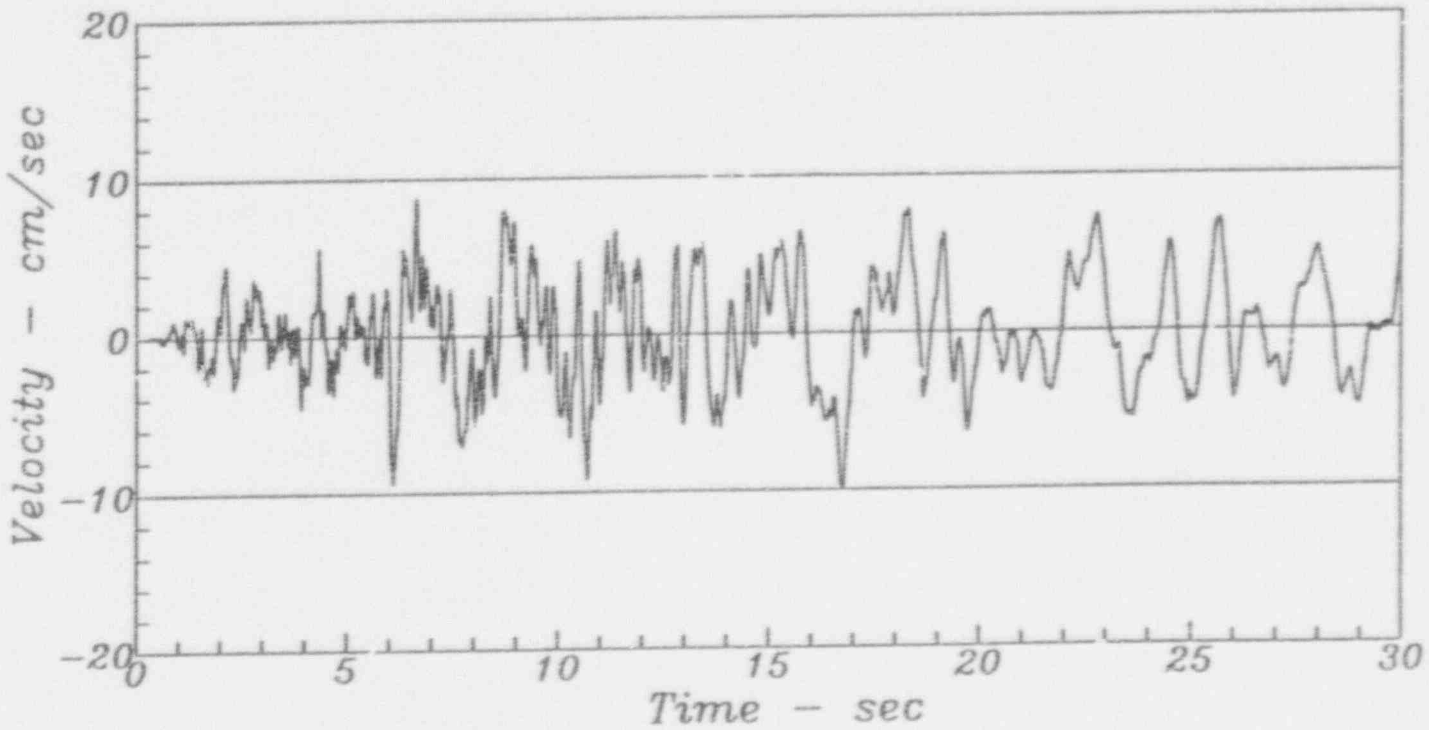


Fig. B-14

VELOCITY -- SYNTHETIC TIME HISTORY
VERTICAL COMPONENT

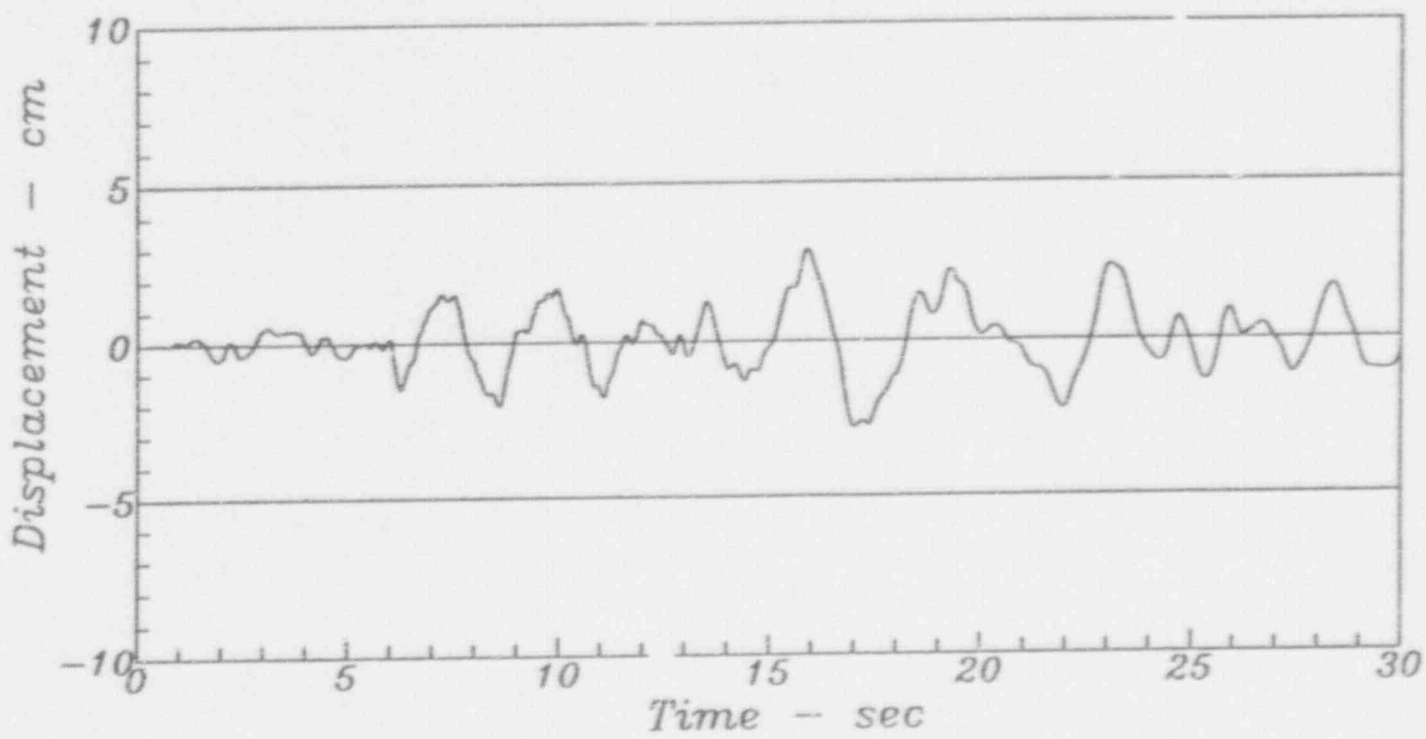


Fig. B-15

DISPLACEMENT -- SYNTHETIC TIME HISTORY
VERTICAL COMPONENT

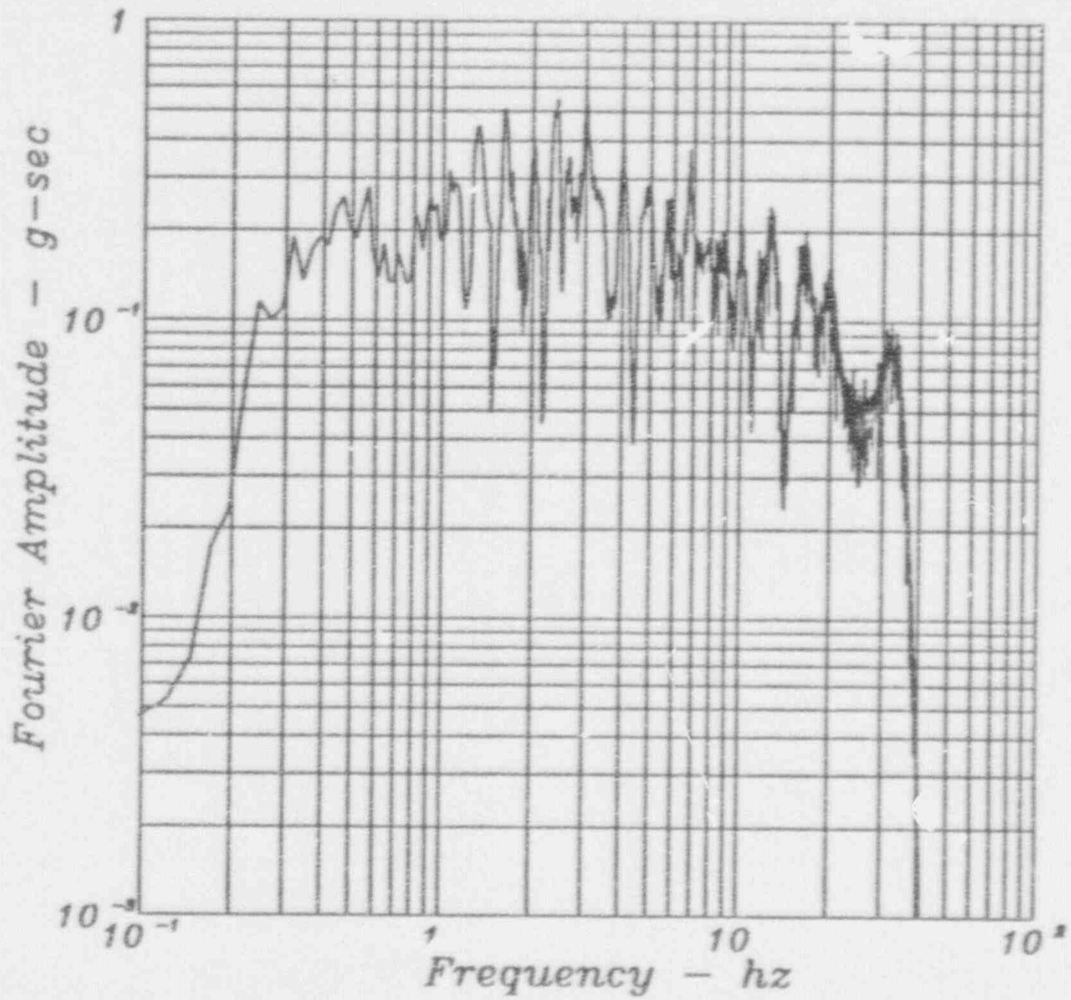


Fig. B-16
FOURIER AMPLITUDES -- SYNTHETIC TIME HISTORY
VERTICAL COMPONENT

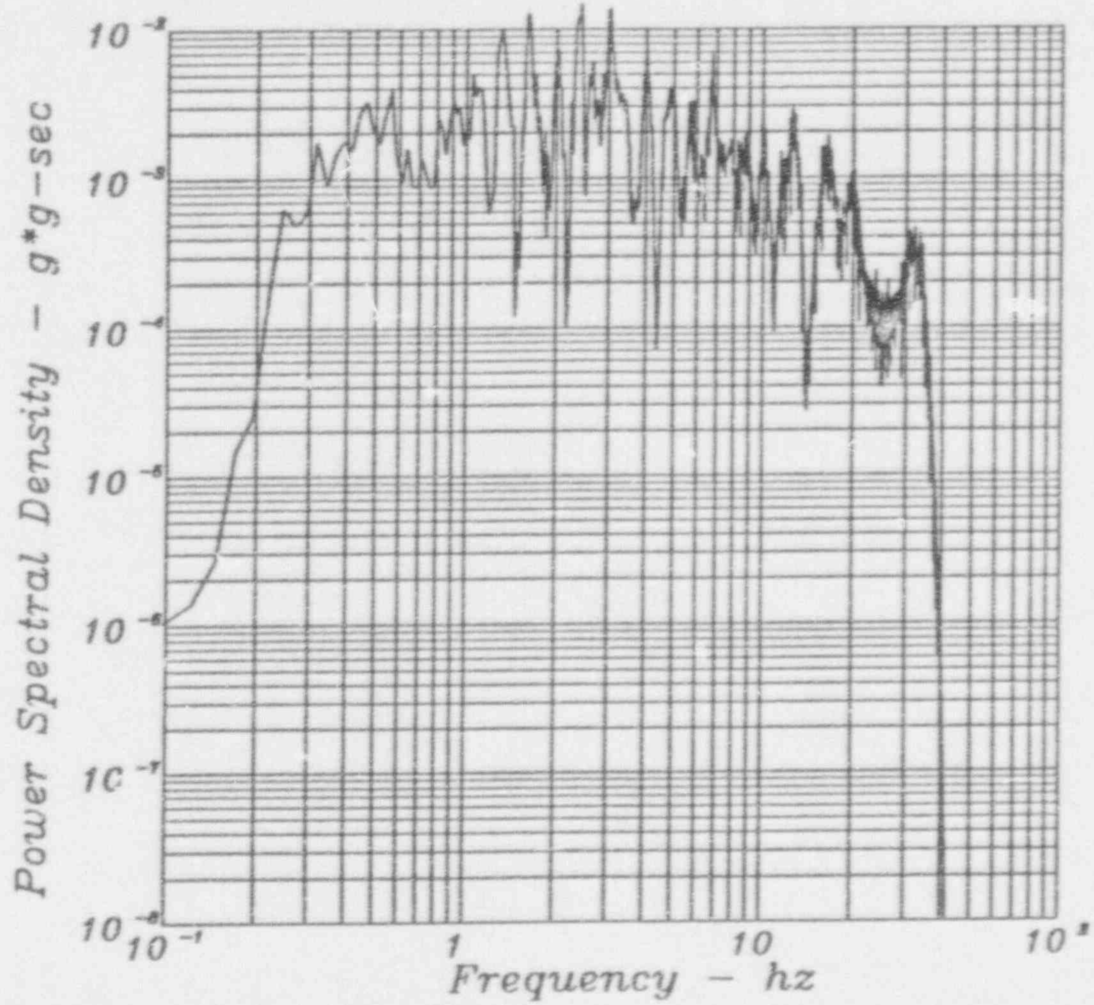


Fig. B-17
POWER SPECTRAL DENSITY -- SYNTHETIC TIME HISTORY
VERTICAL COMPONENT

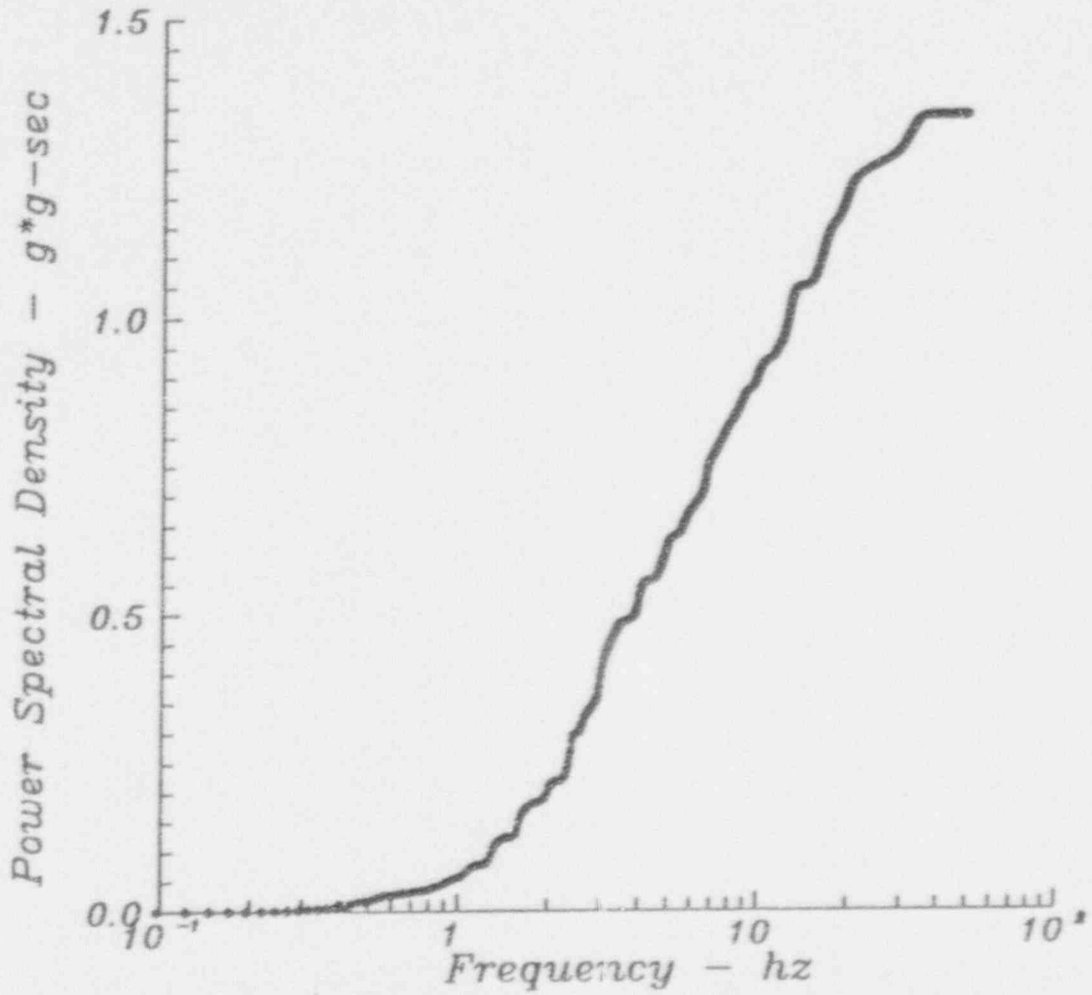


Fig. B-18

CUMULATIVE PSD -- SYNTHETIC TIME HISTORY
VERTICAL COMPONENT

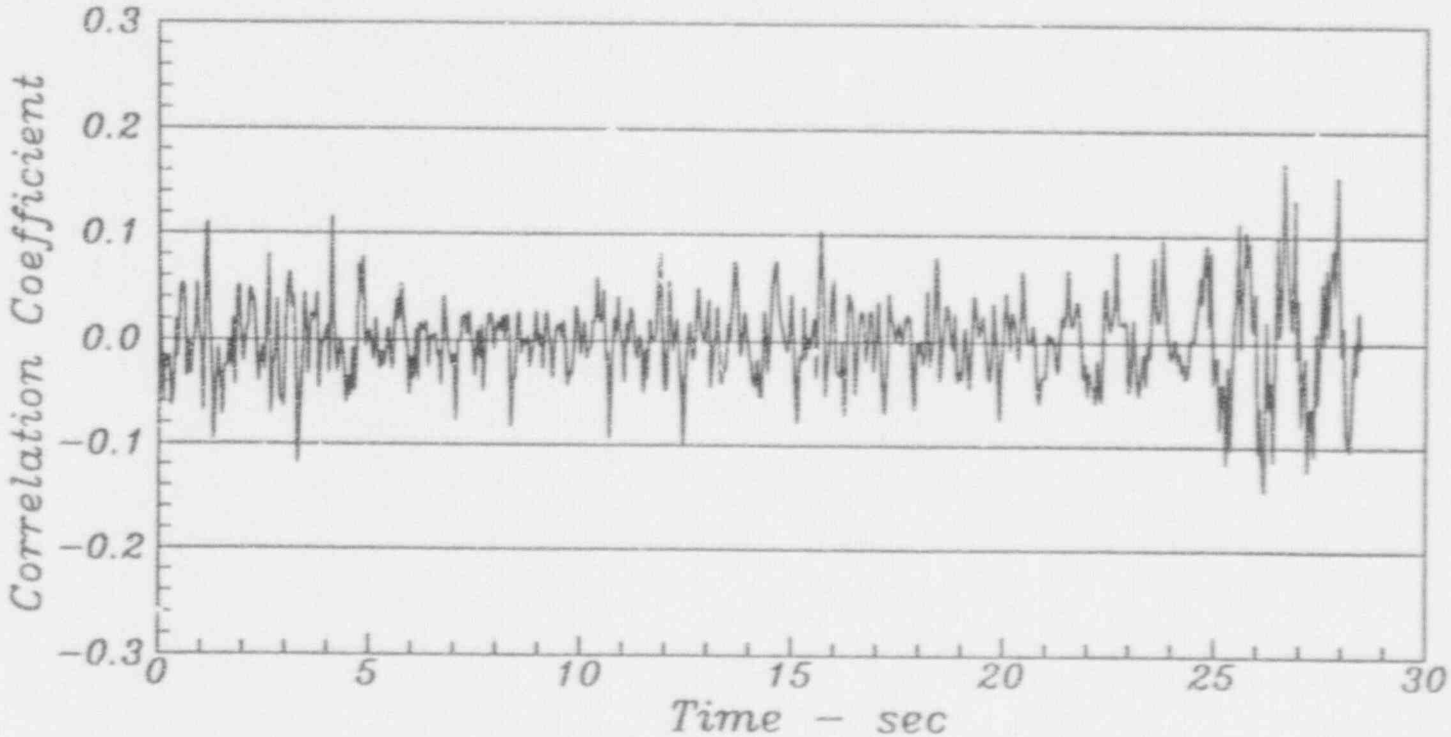


Fig. B-19

CORRELATION COEFFICIENT -- H1 & H2

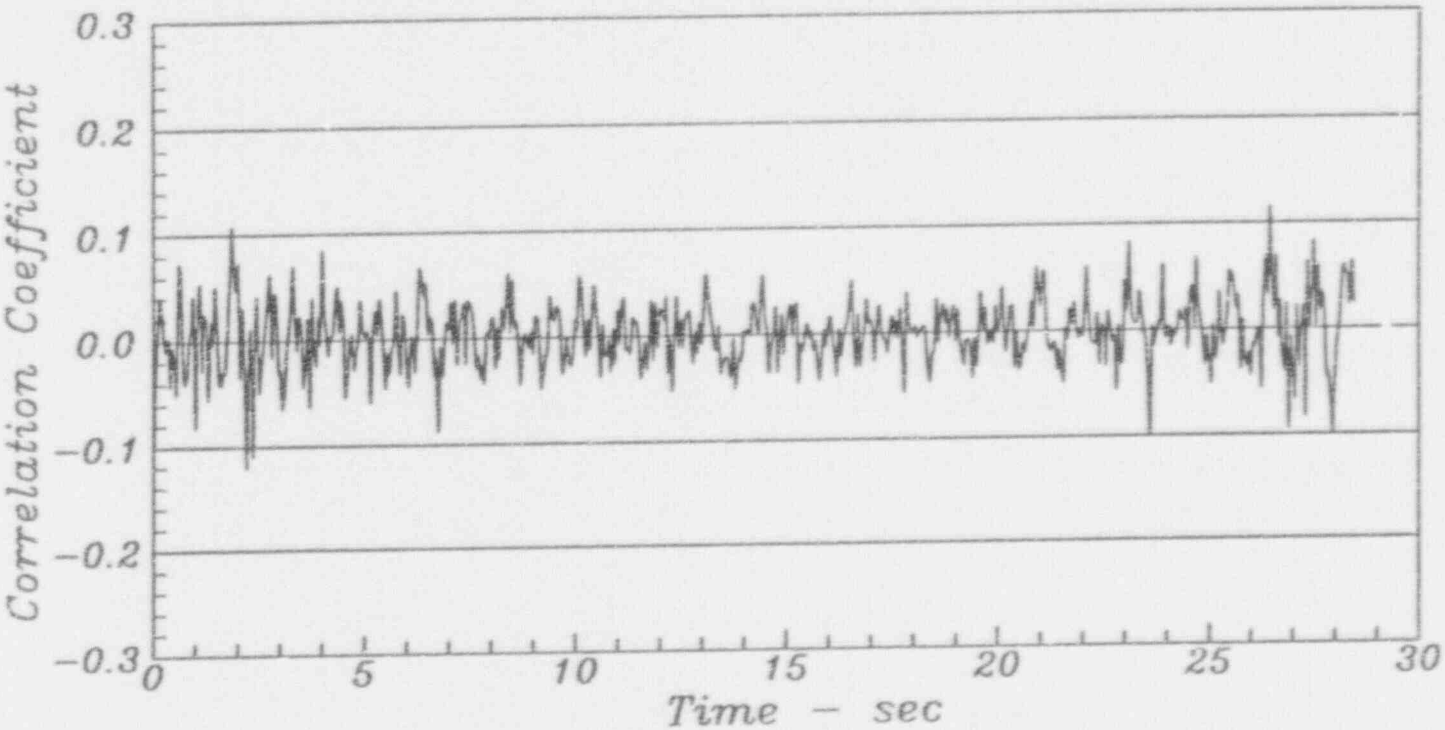


Fig. B-20

CORRELATION COEFFICIENT -- H1 & V

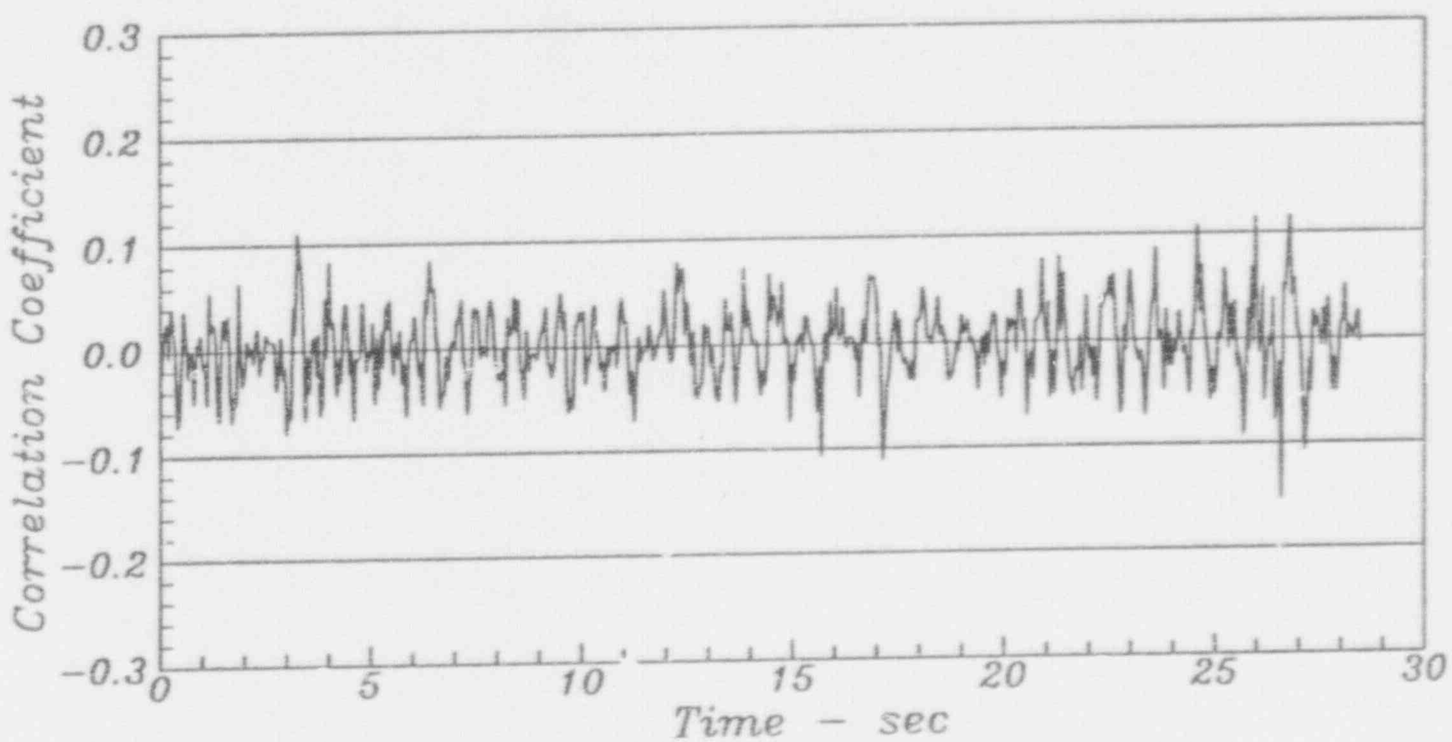


Fig. B-21

CORRELATION COEFFICIENT -- H2 & V

APPENDIX C

STRAIN-COMPATIBLE MODULUS
AND DAMPING VALUES
AND
SPECTRAL ORDINATES CALCULATED FOR
ALL CASES CONSIDERED

EARTHQUAKE
GROUND MOTIONS

SELECTION OF
CONTROL MOTION
AND
DEVELOPMENT OF
GENERIC SOIL SITES

DONE IN CONJUNCTION WITH
C-E/DOE ALWR CERTIFICATION

September 4, 1990

APPENDIX C

STRAIN-COMPATIBLE MODULUS AND DAMPING VALUES AND SPECTRAL ORDINATES CALCULATED FOR ALL CASES CONSIDERED

The strain-compatible modulus and damping values obtained using component H1 are presented in Tables C-1 through C-7 for all the cases considered in this project. Identical values of modulus and damping values were used in conjunction with component H2.

For the vertical motions, the constrained moduli were obtained based on the appropriate strain-compatible shear moduli in Tables C-1 through C-7 and a Poisson's ratio of 0.4. The damping values for the vertical motions were considered to be approximately 1/3 those listed in Tables C-1 through C-7. The actual constrained moduli and damping values used with the vertical component are listed in Tables C-8 through C-14.

Cases B-3.5 and B-4 were also analyzed for the OBE under consideration (peak horizontal acceleration of 0.1 g and peak vertical acceleration of 0.067 g). The moduli and damping values used for these analyses are listed in Tables C-15a, C-15b and C-16.

The calculated spectra at the ground surface and at the foundation level are presented in Figs. C-1 through C-24 for the cases considered in this project.

TABLE C-1

STRAIN COMPATIBLE MODULUS AND DAMPING VALUES
HORIZONTAL MOTIONS
PEAK ACCELERATION = 0.3 g

CASE A - 1

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	12,215	0.008
2	5 to 10	7.5	13,243	0.012
3	10 to 20	15.0	13,719	0.017
4	20 to 30	25.0	14,211	0.022
5	30 to 40	35.0	14,740	0.025
6	40 to 52	46.0	15,437	0.027
7	Below 52	Base	97,000	0.020

CASE B - 1

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	12,218	0.008
2	5 to 10	7.5	13,227	0.012
3	10 to 20	15.0	13,608	0.018
4	20 to 30	25.0	14,091	0.023
5	30 to 40	35.0	14,630	0.026
6	40 to 52	46.0	15,329	0.027
7	52 to 60	56.0	16,026	0.029
8	60 to 80	70.0	16,661	0.031
9	80 to 100	90.0	17,825	0.033
10	Below 100	Base	97,000	0.020

TABLE C-2

STRAIN COMPATIBLE MODULUS AND DAMPING VALUES
HORIZONTAL MOTIONS
PEAK ACCELERATION = 0.3 g

CASE B - 2

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	3,852	0.013
2	5 to 10	7.5	3,677	0.025
3	10 to 20	15.0	3,397	0.039
4	20 to 30	25.0	3,252	0.048
5	30 to 40	35 0	3,203	0.055
6	40 to 52	46.0	3,208	0.062
7	52 to 60	56.0	3,207	0.067
8	60 to 80	70.0	3,254	0.072
9	80 to 100	90.0	3,468	0.075
10	Below 100	Base	97,000	0.020

CASE B - 3

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	906	0.021
2	5 to 10	7.5	775	0.040
3	10 to 20	15.0	635	0.064
4	20 to 30	25.0	523	0.088
5	30 to 40	35.0	468	0.101
6	40 to 52	46.0	468	0.107
7	52 to 60	56.0	500	0.106
8	60 to 80	70.0	548	0.104
9	80 to 100	90.0	633	0.098
10	Below 100	Base	97,000	0.020

TABLE C-3

STRAIN COMPATIBLE MODULUS AND DAMPING VALUES
HORIZONTAL MOTIONS
PEAK ACCELERATION = 0.3 g

CASE B - 4

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	869	0.025
2	5 to 10	7.5	687	0.050
3	10 to 20	15.0	543	0.079
4	20 to 30	25.0	450	0.099
5	30 to 40	35.0	419	0.112
6	40 to 52	46.0	372	0.128
7	52 to 60	56.0	17,508	0.019
8	60 to 80	70.0	18,474	0.021
9	80 to 100	90.0	19,903	0.022
10	Below 100	Base	97,000	0.020

CASE C - 1

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	12,161	0.008
2	5 to 10	7.5	13,204	0.013
3	10 to 20	15.0	13,520	0.018
4	20 to 30	25.0	14,024	0.023
5	30 to 40	35.0	14,558	0.026
6	40 to 52	46.0	15,150	0.029
7	52 to 60	56.0	15,548	0.031
8	60 to 80	70.0	16,249	0.034
9	80 to 100	90.0	17,738	0.034
10	100 to 120	110.0	20,709	0.032
11	120 to 140	130.0	19,927	0.036
12	140 to 160	150.0	21,602	0.036
13	160 to 180	170.0	23,298	0.036
14	180 to 200	190.0	22,812	0.038
15	Below 200	Base	97,000	0.020

TABLE C-4

STRAIN COMPATIBLE MODULUS AND DAMPING VALUES
HORIZONTAL MOTIONS
PEAK ACCELERATION = 0.3 g

CASE C - 2

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	912	0.020
2	5 to 10	7.5	786	0.039
3	10 to 20	15.0	650	0.062
4	20 to 30	25.0	545	0.085
5	30 to 40	35.0	517	0.093
6	40 to 52	46.0	517	0.097
7	52 to 60	56.0	530	0.100
8	60 to 80	70.0	587	0.097
9	80 to 100	90.0	622	0.100
10	100 to 120	110.0	898	0.083
11	120 to 140	130.0	856	0.087
12	140 to 160	150.0	951	0.085
13	160 to 180	170.0	1,029	0.085
14	180 to 200	190.0	919	0.093
15	Below 200	Base	97,000	0.020

CASE C - 3

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	887	0.023
2	5 to 10	7.5	728	0.045
3	10 to 20	15.0	579	0.073
4	20 to 30	25.0	494	0.092
5	30 to 40	35.0	487	0.097
6	40 to 52	46.0	470	0.107
7	52 to 60	56.0	496	0.107
8	60 to 80	70.0	534	0.107
9	80 to 100	90.0	602	0.104
10	100 to 120	110.0	24,260	0.015
11	120 to 140	130.0	23,506	0.019
12	140 to 160	150.0	25,028	0.021
13	160 to 180	170.0	26,775	0.022
14	180 to 200	190.0	26,340	0.024
15	Below 200	Base	97,000	0.020

TABLE C-5

STRAIN COMPATIBLE MODULUS AND DAMPING VALUES
HORIZONTAL MOTIONS
PEAK ACCELERATION = 0.3 g

CASE D - 1

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	905	0.021
2	5 to 10	7.5	767	0.041
3	10 to 20	15.0	618	0.067
4	20 to 30	25.0	504	0.091
5	30 to 40	35.0	481	0.098
6	40 to 52	46.0	509	0.098
7	52 to 60	56.0	529	0.101
8	60 to 80	70.0	585	0.098
9	80 to 100	90.0	602	0.104
10	100 to 120	110.0	888	0.084
11	120 to 140	130.0	847	0.088
12	140 to 160	150.0	916	0.088
13	160 to 180	170.0	976	0.089
14	180 to 200	190.0	855	0.098
15	200 to 220	210.0	31,726	0.016
16	220 to 240	230.0	30,588	0.020
17	240 to 260	250.0	30,793	0.024
18	260 to 280	270.0	31,840	0.024
19	280 to 300	290.0	31,597	0.025
20	Below 300	Base	97,000	0.020

TABLE C-6

STRAIN COMPATIBLE MODULUS AND DAMPING VALUES
HORIZONTAL MOTIONS
PEAK ACCELERATION = 0.3 g

CASE B - 1.5

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	7,658	0.010
2	5 to 10	7.5	7,595	0.019
3	10 to 20	15.0	7,600	0.027
4	20 to 30	25.0	7,454	0.036
5	30 to 40	35.0	7,434	0.041
6	40 to 52	46.0	7,522	0.045
7	52 to 60	56.0	7,745	0.047
8	60 to 80	70.0	8,287	0.047
9	80 to 100	90.0	8,901	0.049
10	Below 100	Base	97,000	0.020

CASE B - 3.5

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	1,246	0.026
2	5 to 10	7.5	982	0.050
3	10 to 20	15.0	770	0.080
4	20 to 30	25.0	588	0.109
5	30 to 40	35.0	604	0.112
6	40 to 52	46.0	655	0.110
7	52 to 60	56.0	6,568	0.036
8	60 to 80	70.0	6,850	0.038
9	80 to 100	90.0	7,211	0.041
10	Below 100	Base	97,000	0.020

TABLE C-7

STRAIN COMPATIBLE MODULUS AND DAMPING VALUES
HORIZONTAL MOTIONS
PEAK ACCELERATION = 0.3 g

CASE C - 1.5

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	3,862	0.012
2	5 to 10	7.5	3,726	0.024
3	10 to 20	15.0	3,583	0.034
4	20 to 30	25.0	3,478	0.043
5	30 to 40	35.0	3,504	0.047
6	40 to 52	46.0	3,653	0.049
7	52 to 60	56.0	3,700	0.052
8	60 to 80	70.0	3,851	0.056
9	80 to 100	90.0	4,191	0.056
10	100 to 120	110.0	4,953	0.053
11	120 to 140	130.0	4,640	0.060
12	140 to 160	150.0	5,157	0.057
13	160 to 180	170.0	5,653	0.056
14	180 to 200	190.0	5,595	0.057
15	Below 200	Base	97,000	0.020

TABLE C-8

MODULUS AND DAMPING VALUES
VERTICAL MOTION
PEAK ACCELERATION = 0.2 g

CASE A - 1

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	73,290	0.0027
2	5 to 10	7.5	79,458	0.0040
3	10 to 20	15.0	82,314	0.0057
4	20 to 30	25.0	85,266	0.0073
5	30 to 40	35.0	88,440	0.0083
6	40 to 52	46.0	92,622	0.0090
7	Below 52	Base	302,000	0.0067

CASE B - 1

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	73,308	0.0027
2	5 to 10	7.5	79,362	0.0040
3	10 to 20	15.0	81,648	0.0060
4	20 to 30	25.0	84,546	0.0077
5	30 to 40	35.0	87,780	0.0087
6	40 to 52	46.0	91,974	0.0090
7	52 to 60	56.0	96,156	0.0097
8	60 to 80	70.0	99,966	0.0103
9	80 to 100	90.0	106,950	0.0110
10	Below 100	Base	302,000	0.0067

Notes:
- Constrained moduli are obtained using the corresponding strain-compatible shear moduli for the cases listed in Tables C-1 through C-7, and a Poisson's ratio of 0.4.
- Damping ratios are approx 1/3 the corresponding strain-compatible damping ratios listed in these tables.

TABLE C-9

MODULUS AND DAMPING VALUES
VERTICAL MOTION
PEAK ACCELERATION = 0.2 g

CASE B - 2

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	23,112	0.0043
2	5 to 10	7.5	22,062	0.0083
3	10 to 20	15.0	20,382	0.0130
4	20 to 30	25.0	19,512	0.0160
5	30 to 40	35.0	19,218	0.0183
6	40 to 52	46.0	19,248	0.0207
7	52 to 60	56.0	19,242	0.0223
8	60 to 80	70.0	19,524	0.0240
9	80 to 100	90.0	20,808	0.0250
10	Below 100	Base	302,000	0.0067

CASE B - 3

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	5,436	0.0070
2	5 to 10	7.5	4,650	0.0133
3	10 to 20	15.0	3,810	0.0213
4	20 to 30	25.0	3,138	0.0293
5	30 to 40	35.0	2,808	0.0337
6	40 to 52	46.0	2,808	0.0357
7	52 to 60	56.0	3,000	0.0353
8	60 to 80	70.0	3,288	0.0347
9	80 to 100	90.0	3,798	0.0327
10	Below 100	Base	302,000	0.0067

Notes:
- *Constrained moduli are obtained using the corresponding strain-compatible shear moduli for the cases listed in Tables C-1 through C-7, and a Poisson's ratio of 0.4.*
- *Damping ratios are approx 1/3 the corresponding strain-compatible damping ratios listed in these tables.*

TABLE C-10

MODULUS AND DAMPING VALUES
VERTICAL MOTION
PEAK ACCELERATION = 0.2 g

CASE B - 4

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	5,214	0.0083
2	5 to 10	7.5	4,122	0.0167
3	10 to 20	15.0	3,258	0.0263
4	20 to 30	25.0	2,700	0.0330
5	30 to 40	35.0	2,514	0.0373
6	40 to 52	46.0	2,232	0.0427
7	52 to 60	5[illegible].0	105,048	0.0063
8	60 to 80	[illegible]0.0	110,844	0.0070
9	80 to 100	90.0	119,418	0.0073
10	Below 100	Base	302,000	0.0067

CASE C - 1

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	72,966	0.0027
2	5 to 10	7.5	79,224	0.0043
3	10 to 20	15.0	81,120	0.0060
4	20 to 30	25.0	84,144	0.0077
5	30 to 40	35.0	87,348	0.00[illegible]7
6	40 to 52	46.0	90,900	0.0097
7	52 to 60	56.0	93,288	0.0103
8	60 to 80	70.0	97,494	0.0113
9	80 to 100	90.0	106,428	0.0113
10	100 to 120	110.0	124,254	0.0107
11	120 to 140	130.0	119,562	0.0120
12	140 to 160	150.0	129,612	0.0120
13	160 to 180	170.0	139,788	0.0120
14	180 to 200	190.0	136,872	0.0127
15	Below 200	Base	302,000	0.0067

Notes:
- *Constrained moduli are obtained using the corresponding strain-compatible shear moduli for the cases listed in Tables C-1 through C-7, and a Poisson's ratio of 0.4.*
- *Damping ratios are approx 1/3 the corresponding strain-compatible damping ratios listed in these tables.*

TABLE C-11

MODULUS AND DAMPING VALUES
VERTICAL MOTION
PEAK ACCELERATION = 0.2 g

CASE C - 2

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	5,472	0.0067
2	5 to 10	7.5	4,716	0.0130
3	10 to 20	15.0	3,900	0.0207
4	20 to 30	25.0	3,270	0.0283
5	30 to 40	35.0	3,102	0.0310
6	40 to 52	46.0	3,102	0.0323
7	52 to 60	56.0	3,180	0.0333
8	60 to 80	70.0	3,522	0.0323
9	80 to 100	90.0	3,732	0.0333
10	100 to 120	110.0	5,388	0.0277
11	120 to 140	130.0	5,136	0.0290
12	140 to 160	150.0	5,706	0.0283
13	160 to 180	170.0	6,174	0.0283
14	180 to 200	190.0	5,514	0.0310
15	Below 200	Base	302,000	0.0067

CASE C - 3

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	5,322	0.0077
2	5 to 10	7.5	4,368	0.0150
3	10 to 20	15.0	3,474	0.0243
4	20 to 30	25.0	2,964	0.0307
5	30 to 40	35.0	2,922	0.0323
6	40 to 52	46.0	2,820	0.0357
7	52 to 60	56.0	2,976	0.0357
8	60 to 80	70.0	3,204	0.0357
9	80 to 100	90.0	3,612	0.0347
10	100 to 120	110.0	145,560	0.0050
11	120 to 140	130.0	141,036	0.0063
12	140 to 160	150.0	150,168	0.0070
13	160 to 180	170.0	160,650	0.0073
14	180 to 200	190.0	158,040	0.0080
15	Below 200	Base	302,000	0.0067

Notes:
- Constrained moduli are obtained using the corresponding strain-compatible shear moduli for the cases listed in Tables C-1 through C-7, and a Poisson's ratio of 0.4.
- Damping ratios are approx 1/3 the corresponding strain-compatible damping ratios listed in these tables.

TABLE C-12

MODULUS AND DAMPING VALUES
VERTICAL MOTION
PEAK ACCELERATION = 0.2 g

CASE D - 1

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	5,430	0.0070
2	5 to 10	7.5	4,602	0.0137
3	10 to 20	15.0	3,708	0.0223
4	20 to 30	25.0	3,024	0.0303
5	30 to 40	35.0	2,886	0.0327
6	40 to 52	46.0	3,054	0.0327
7	52 to 60	56.0	3,174	0.0337
8	60 to 80	70.0	3,510	0.0327
9	80 to 100	90.0	3,612	0.0347
10	100 to 120	110.0	5,328	0.0280
11	120 to 140	130.0	5,082	0.0293
12	140 to 160	150.0	5,496	0.0293
13	160 to 180	1[illegible].0	5,856	0.0297
14	180 to 200	190.0	5,130	0.0327
15	200 to 220	210.0	190,356	0.0053
16	220 to 240	230.0	183,528	0.0067
17	240 to 260	250.0	184,758	0.0080
18	260 to 280	270.0	191,040	0.0080
19	280 to 300	290.0	189,582	0.0083
20	Below 300	Base	302,000	0.0067

Notes:
- *Constrained moduli are obtained using the corresponding strain-compatible shear moduli for the cases listed in Tables C-1 through C-7, and a Poisson's ratio of 0.4.*
- *Damping ratios are approx 1/3 the corresponding strain-compatible damping ratios listed in these tables.*

TABLE C-13

MODULUS AND DAMPING VALUES
VERTICAL MOTION
PEAK ACCELERATION = 0.2 g

CASE B - 1.5

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	73,308	0.0033
2	5 to 10	7.5	79,362	0.0063
3	10 to 20	15.0	81,648	0.0090
4	20 to 30	25.0	84,546	0.0120
5	30 to 40	35.0	87,780	0.0137
6	40 to 52	46.0	91,974	0.0150
7	52 to 60	56.0	96,156	0.0157
8	60 to 80	70.0	99,966	0.0157
9	80 to 100	90.0	106,950	0.0163
10	Below 100	Base	302,000	0.0067

CASE B - 3.5

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	7,476	0.0087
2	5 to 10	7.5	5,892	0.0167
3	10 to 20	15.0	4,620	0.0267
4	20 to 30	25.0	3,528	0.0363
5	30 to 40	35.0	3,624	0.0373
6	40 to 52	46.0	3,930	0.0367
7	52 to 60	56.0	39,408	0.0120
8	60 to 80	70.0	41,100	0.0127
9	80 to 100	90.0	43,266	0.0137
10	Below 100	Base	302,000	0.0067

Notes:
- ♦ Constrained moduli are obtained using the corresponding strain-compatible shear moduli for the cases listed in Tables C-1 through C-7, and a Poisson's ratio of 0.4.
- ♦ Damping ratios are approx 1/3 the corresponding strain-compatible damping ratios listed in these tables.

TABLE C-14

MODULUS AND DAMPING VALUES
VERTICAL MOTION
PEAK ACCELERATION = 0.2 g

CASE C - 1.5

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	23,172	0.0040
2	5 to 10	7.5	22,356	0.0080
3	10 to 20	15.0	21,498	0.0113
4	20 to 30	25.0	20,868	0.0143
5	30 to 40	35.0	21,024	0.0157
6	40 to 52	46.0	21,918	0.0163
7	52 to 60	56.0	22,200	0.0173
8	60 to 80	70.0	23,106	0.0187
9	80 to 100	90.0	25,146	0.0187
10	100 to 120	110.0	29,718	0.0177
11	120 to 140	130.0	27,840	0.0200
12	140 to 160	150.0	30,942	0.0190
13	160 to 180	170.0	33,918	0.0187
14	180 to 200	190.0	33,570	0.0190
15	Below 200	Base	302,000	0.0067

Notes:
- Constrained moduli are obtained using the corresponding strain-compatible shear moduli for the cases listed in Tables C-1 through C-7, and a Poisson's ratio of 0.4.
- Damping ratios are approx 1/3 the corresponding strain-compatible damping ratios listed in these tables.

TABLE C-15a

MODULUS AND DAMPING VALUES
HORIZONTAL MOTIONS
PEAK ACCELERATION = 0.1 g

CASE B - 3.5

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	1,246	0.016
2	5 to 10	7.5	983	0.033
3	10 to 20	15.0	771	0.049
4	20 to 30	25.0	589	0.074
5	30 to 40	35.0	605	0.075
6	40 to 52	46.0	656	0.075
7	52 to 60	56.0	6,572	0.021
8	60 to 80	70.0	6,851	0.023
9	80 to 100	90.0	7,214	0.024
10	Below 100	Base	97,000	0.010

CASE B - 4

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	869	0.015
2	5 to 10	7.5	685	0.031
3	10 to 20	15.0	544	0.046
4	20 to 30	25.0	488	0.055
5	30 to 40	35.0	421	0.071
6	40 to 52	46.0	371	0.084
7	52 to 60	56.0	17,502	0.011
8	60 to 80	70.0	19,835	0.011
9	80 to 100	90.0	19,901	0.013
10	Below 100	Base	97,000	0.010

Notes:
- Moduli are the strain-compatible moduli obtained for cases with peak acceleration = 0.3 g at rock outcrop.
- Damping values are compatible with strains developed for peak acceleration = 0.1 g at rock outcrop.

TABLE C-15b

MODULUS AND DAMPING VALUES
HORIZONTAL MOTIONS
PEAK ACCELERATION = 0.1 g

CASE B - 3.5

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	1,354	0.016
2	5 to 10	7.5	1,259	0.030
3	10 to 20	15.0	1,141	0.044
4	20 to 30	25.0	1,123	0.051
5	30 to 40	35.0	1,160	0.055
6	40 to 52	46.0	1,161	0.061
7	52 to 60	56.0	7,435	0.024
8	60 to 80	70.0	7,957	0.024
9	80 to 100	90.0	8,768	0.023
10	Below 100	Base	97,000	0.010

CASE B - 4

No.	Depth Range	Avg Depth	Shear Modulus	Damping Ratio
1	0 to 5	2.5	928	0.018
2	5 to 10	7.5	838	0.034
3	10 to 20	15.0	770	0.046
4	20 to 30	25.0	714	0.061
5	30 to 40	35.0	705	0.068
6	40 to 52	46.0	734	0.070
7	52 to 60	56.0	18,424	0.012
8	60 to 80	70.0	19,782	0.012
9	80 to 100	90.0	21,573	0.012
10	Below 100	Base	97,000	0.010

Notes: • Modulus and damping values are compatible with strains developed for peak acceleration = 0.1 g at rock outcrop.

TABLE C-16

MODULUS AND DAMPING VALUES
VERTICAL MOTION
PEAK ACCELERATION = 0.067 g

CASE B - 3.5

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	7,476	0.0053
2	5 to 10	7.5	5,898	0.0110
3	10 to 20	15.0	4,626	0.0163
4	20 to 30	25.0	3,534	0.0247
5	30 to 40	35.0	3,630	0.0250
6	40 to 52	46.0	3,936	0.0250
7	52 to 60	56.0	39,432	0.0070
8	60 to 80	70.0	41,106	0.0077
9	80 to 100	90.0	43,284	0.0080
10	Below 100	Base	302,000	0.0067

CASE B - 4

No.	Depth Range	Avg Depth	Cnstrnd. Mod.	Damping Ratio
1	0 to 5	2.5	5,214	0.0050
2	5 to 10	7.5	4,122	0.0103
3	10 to 20	15.0	3,258	0.0153
4	20 to 30	25.0	2,700	0.0183
5	30 to 40	35.0	2,514	0.0237
6	40 to 52	46.0	2,232	0.0280
7	52 to 60	56.0	105,048	0.0037
8	60 to 80	70.0	110,844	0.0037
9	80 to 100	90.0	119,418	0.0043
10	Below 100	Base	302,000	0.0067

Notes:
- *Constrained moduli are obtained using the corresponding strain-compatible shear moduli for the cases listed in Table C-15a, and a Poisson's ratio of 0.4.*
- *Damping ratios are approx 1/3 the corresponding strain-compatible damping ratios listed in Table C-15a.*

Fig. C-1
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE A-1

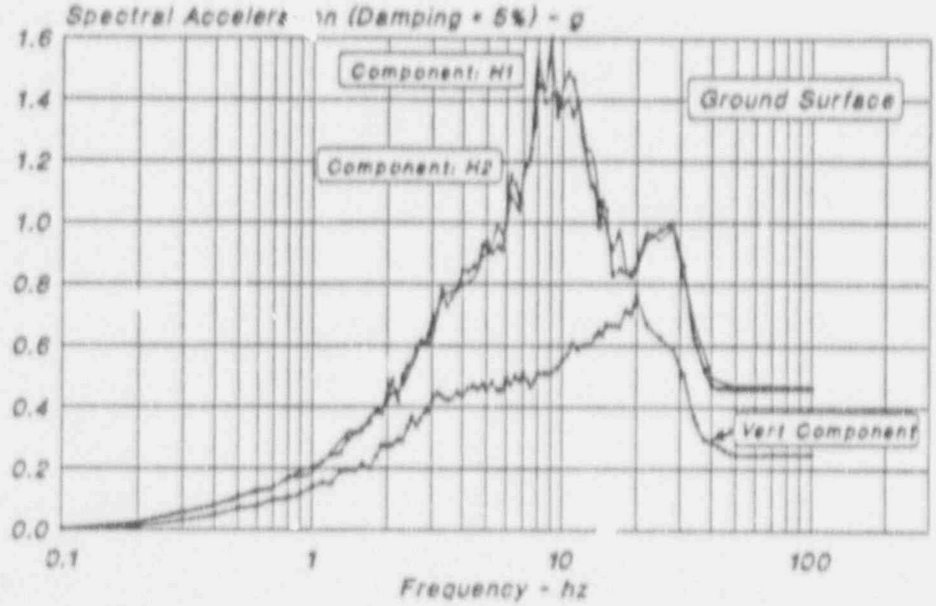


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Fig. C-2
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE A-1

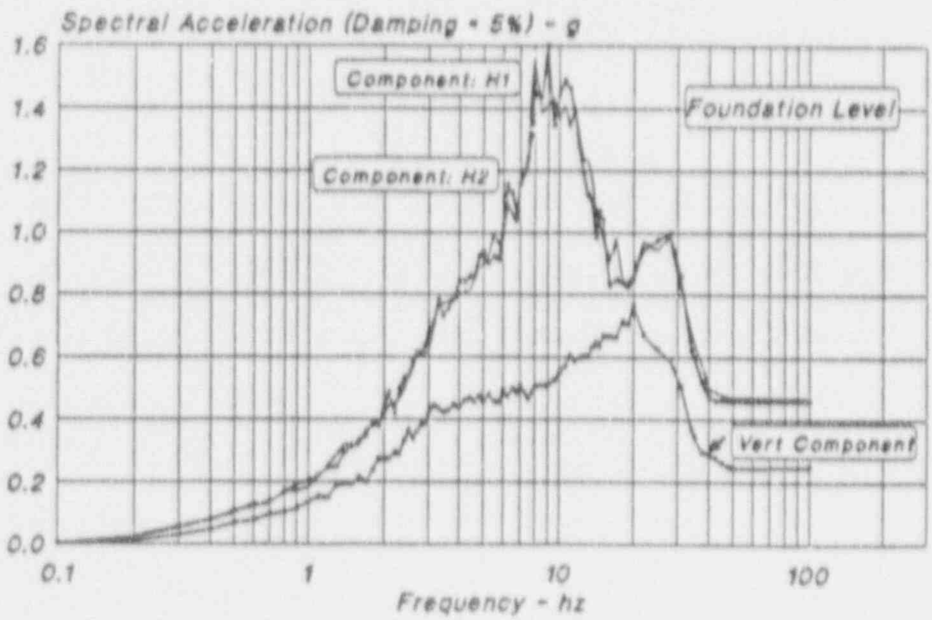

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Fig. C-2
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-1

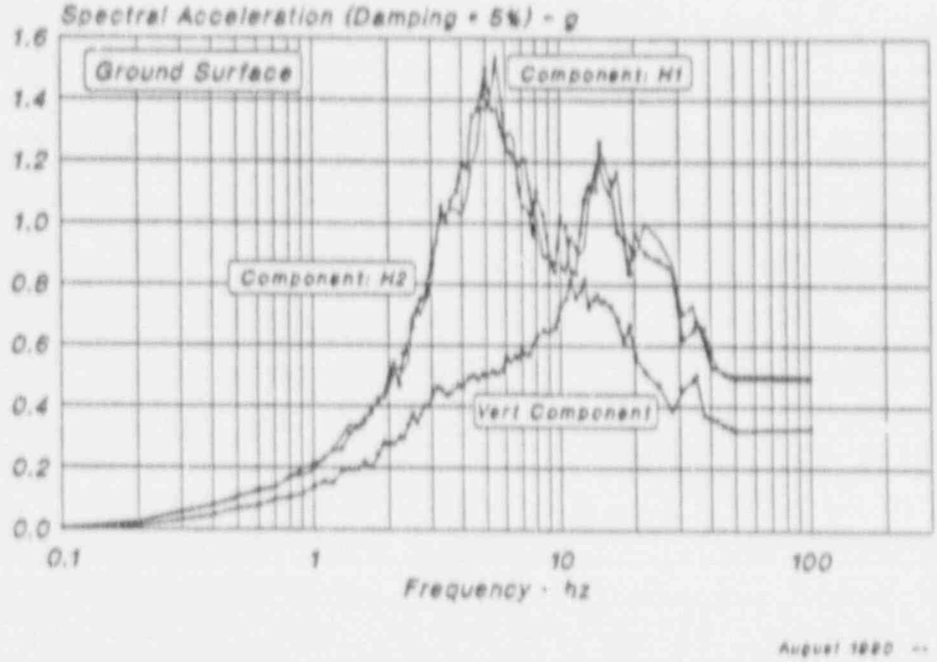

Fig. C-4
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-1

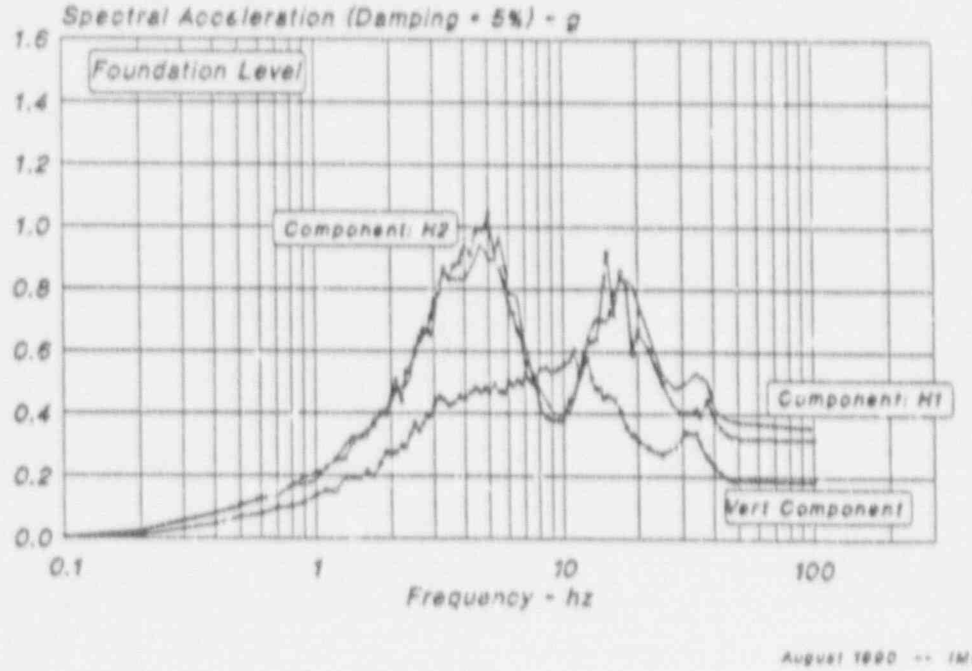

Fig. C-3
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-2

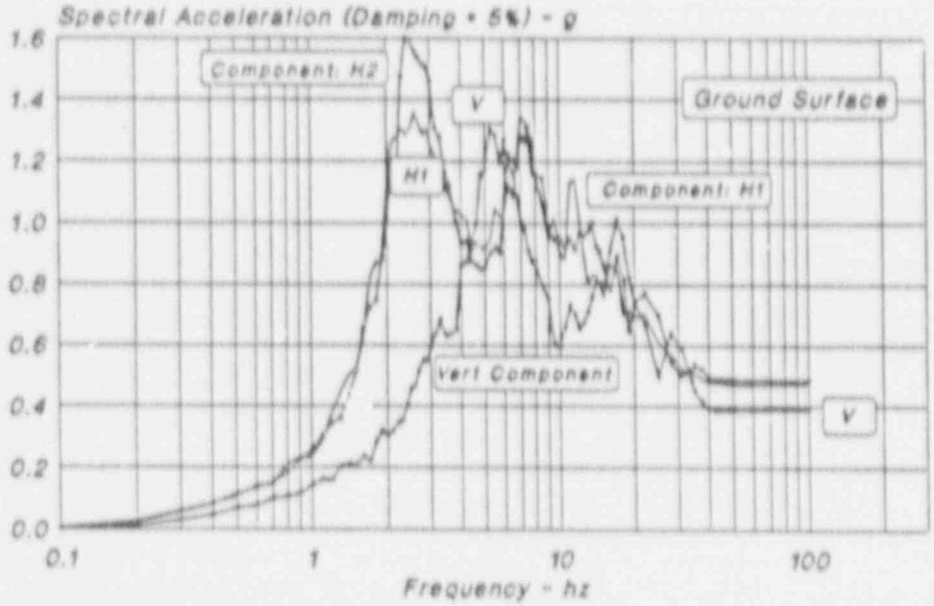

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Fig. C-6
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-2

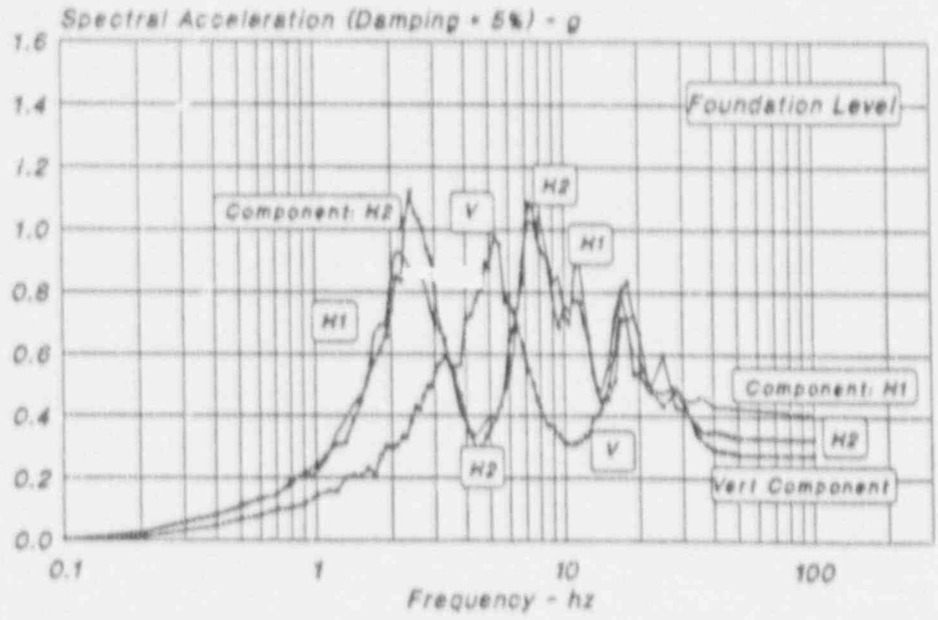

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Fig. C-7
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-3

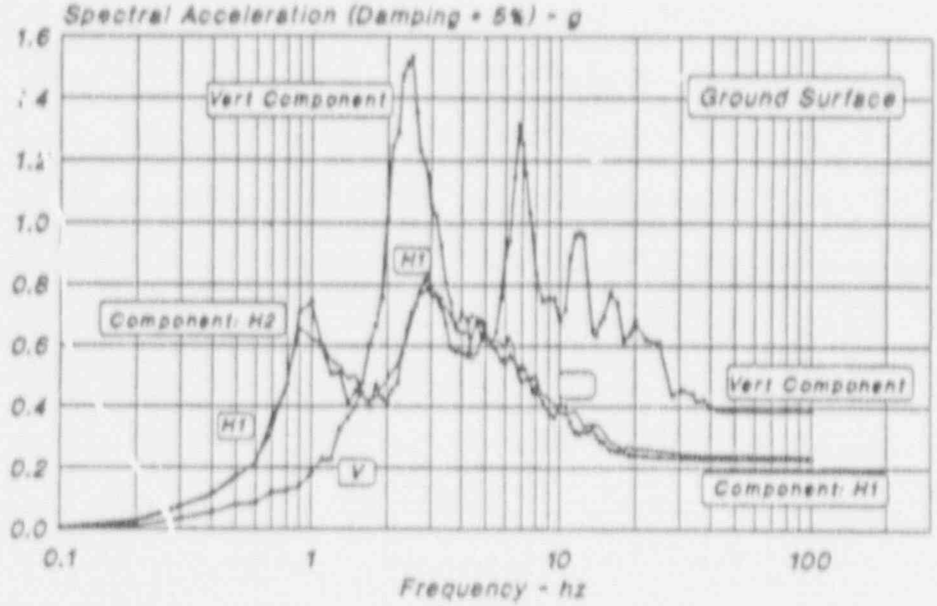

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Fig. C-8
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-3

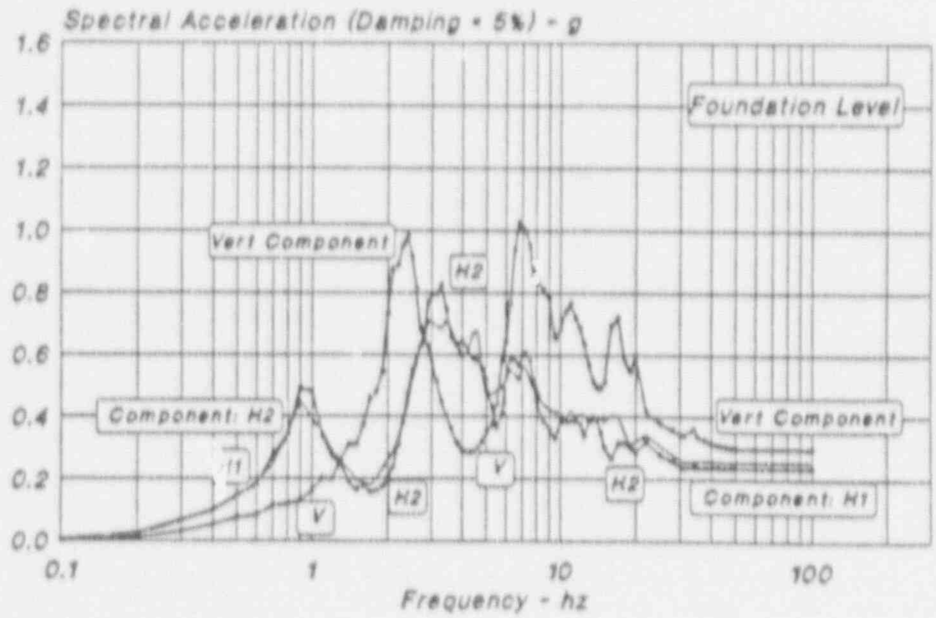

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Fig. C-9
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-4

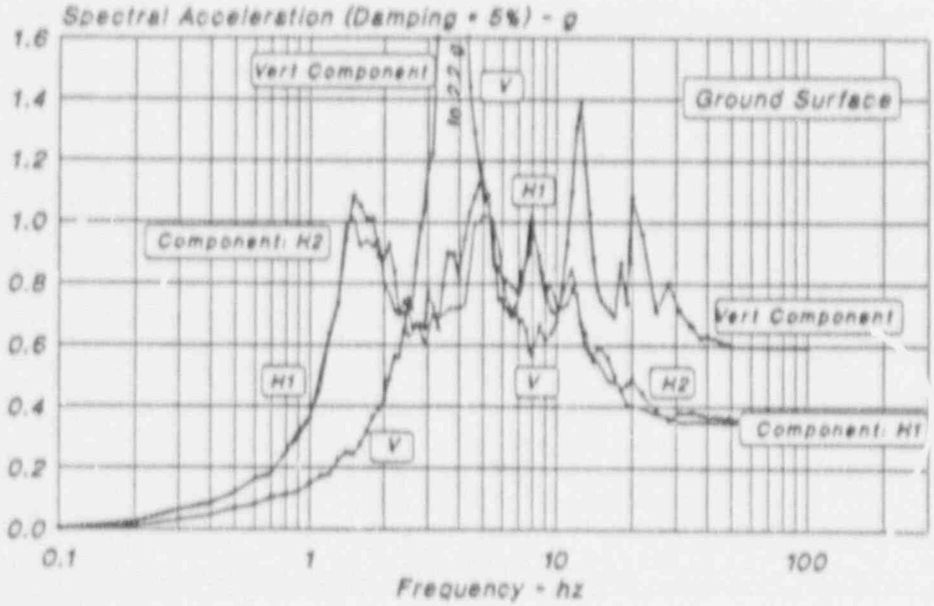

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Fig. C-10
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-4

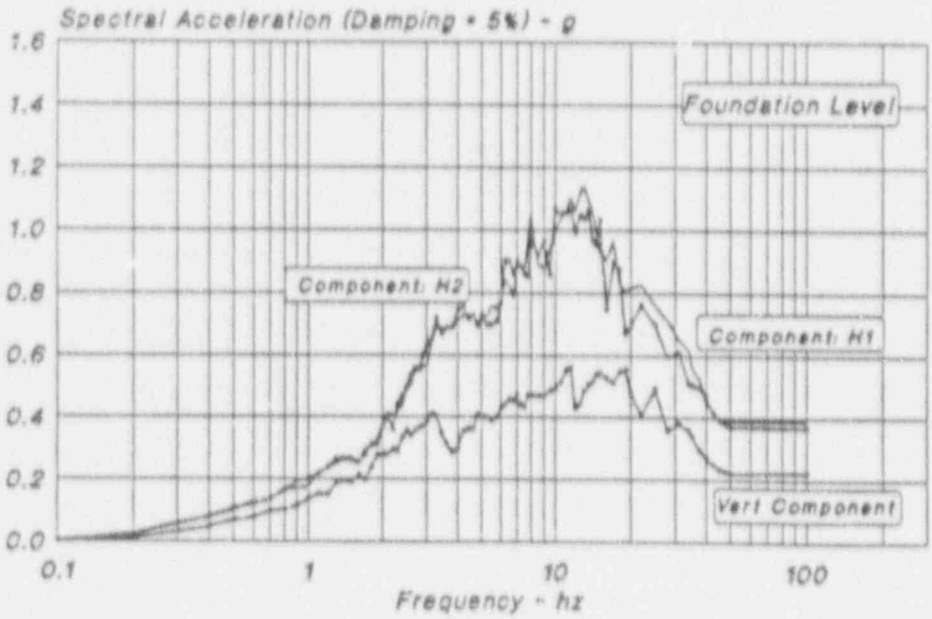

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Fig. C-11
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE C-1

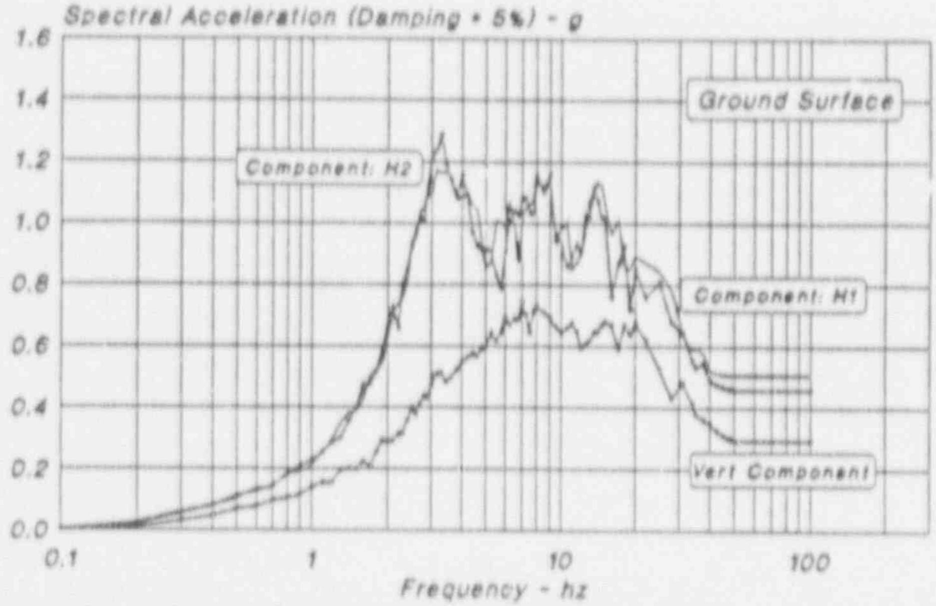

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Fig. C-12
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE C-1

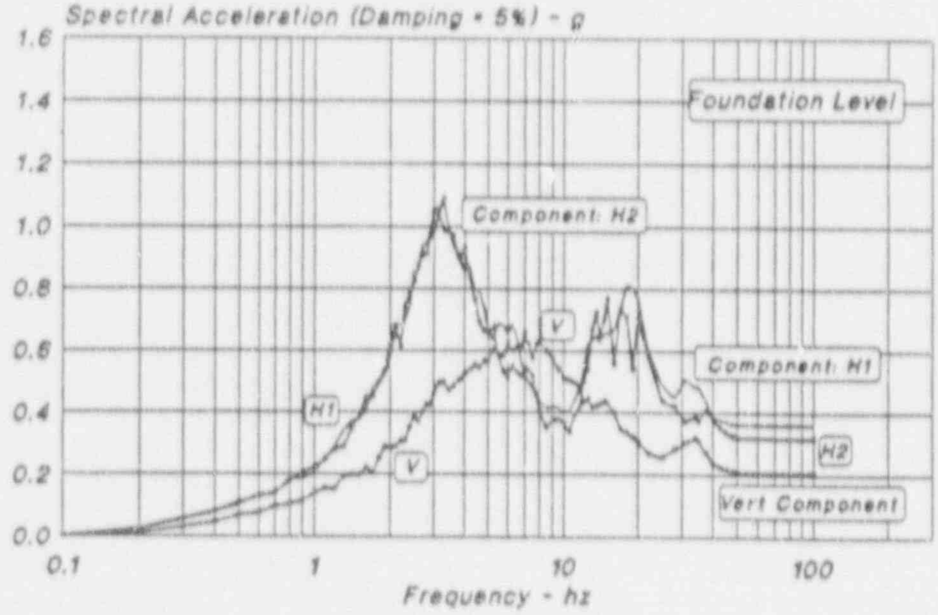

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Fig. C-13
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE C-2

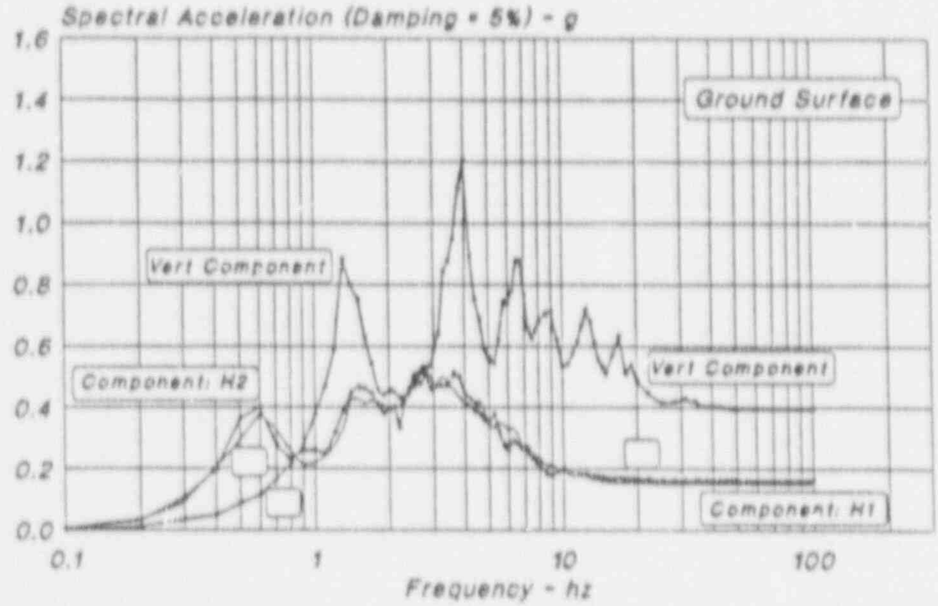

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Fig. C-14
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE C-2

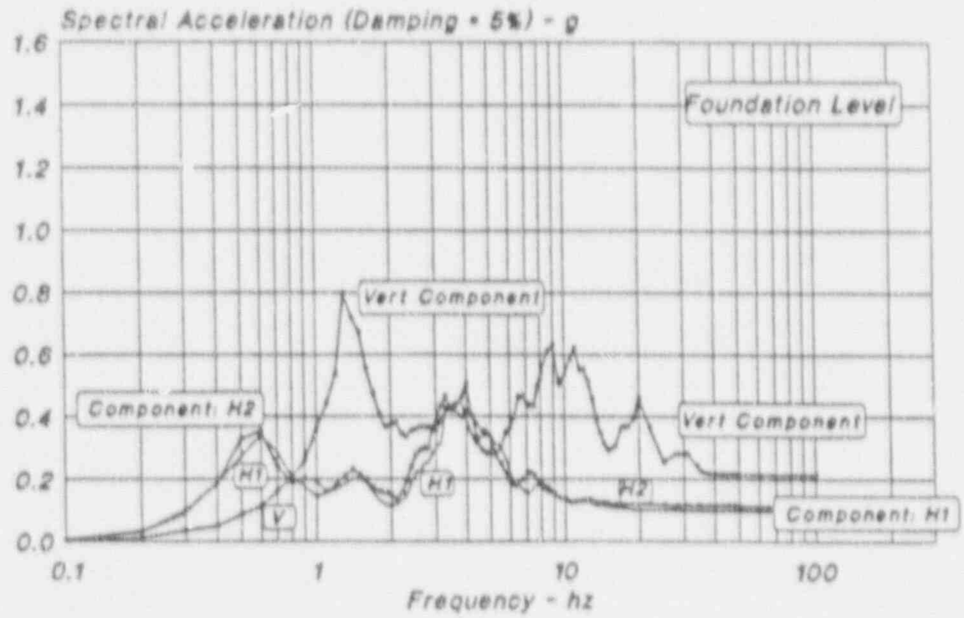

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Fig. C-15
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE C-3

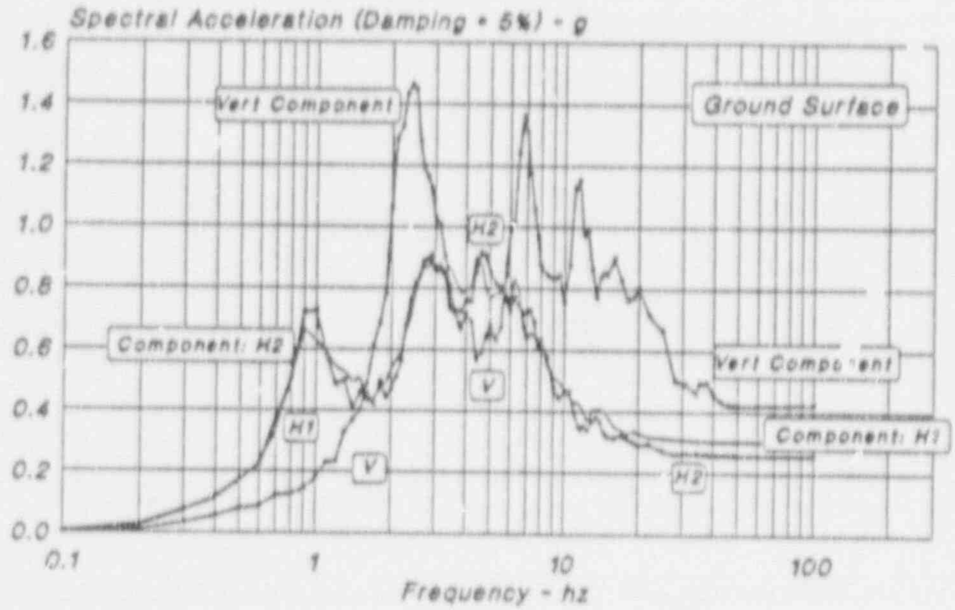

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Fig. C-16
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE C-3

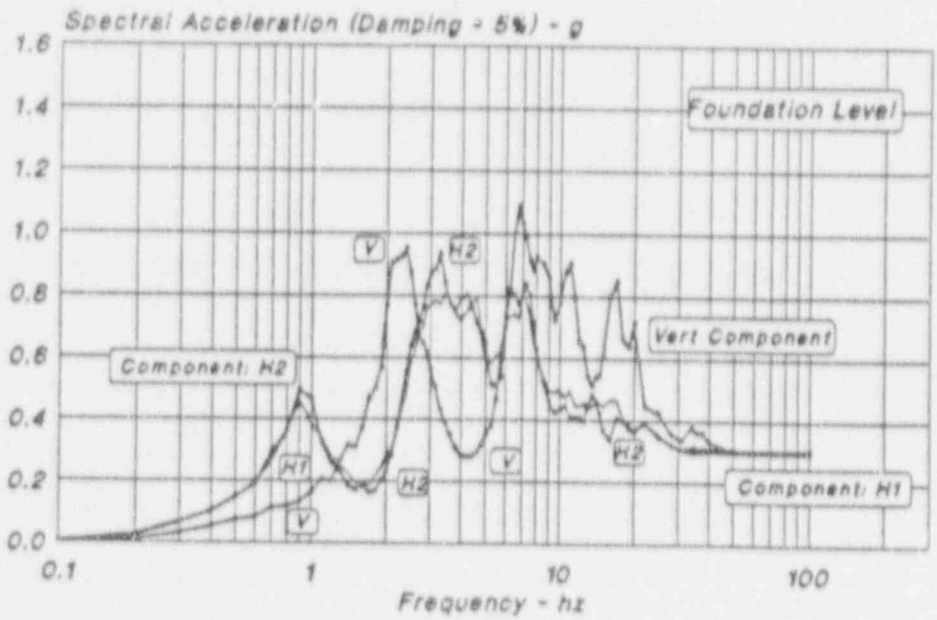

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Fig. C-17
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE D-1

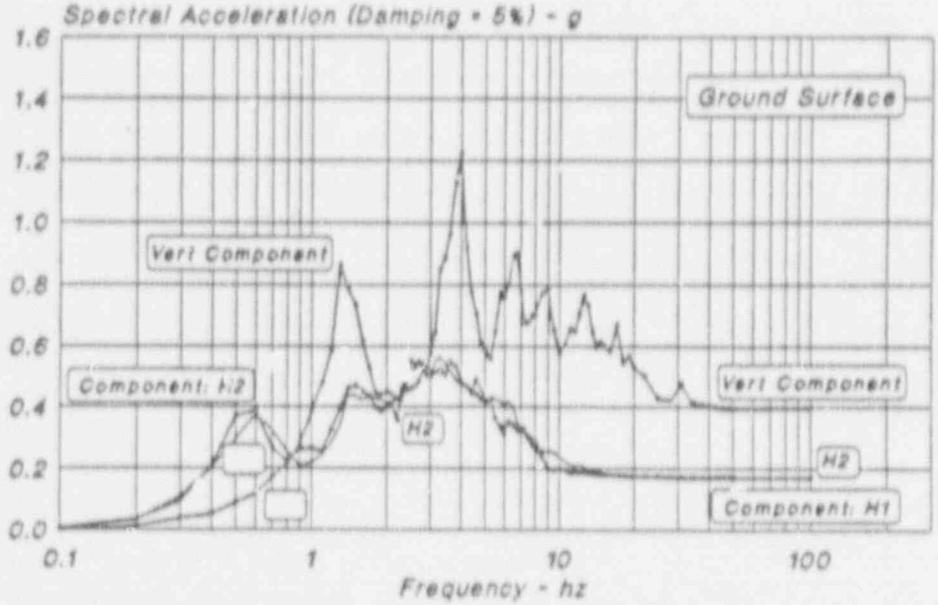

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Fig. C-18
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE D-1

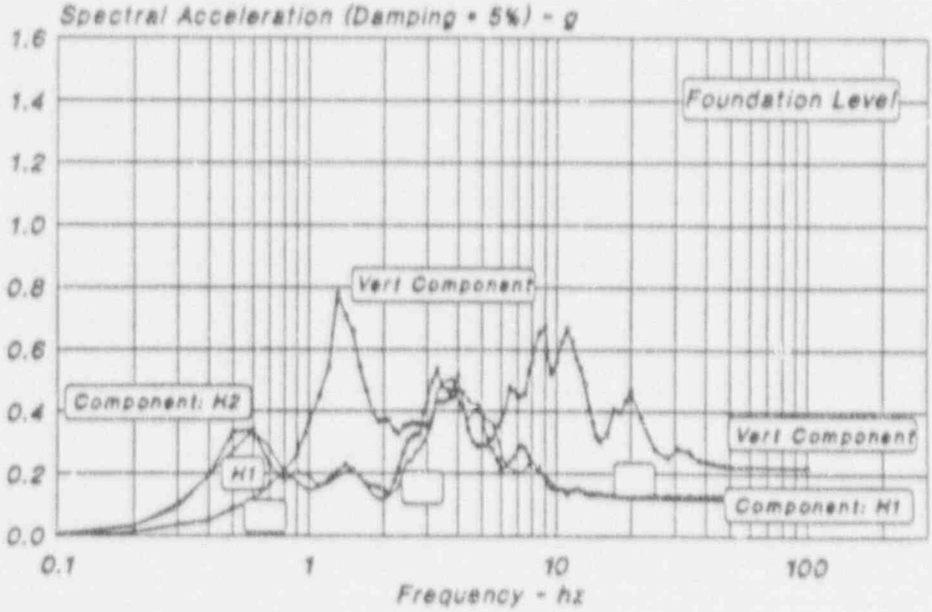

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Fig. C-19
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-1.5

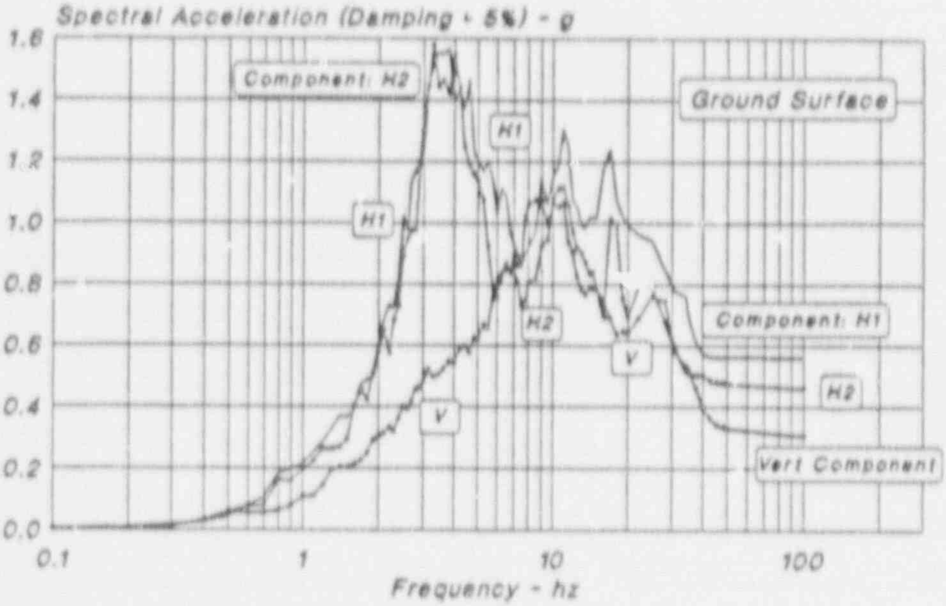

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Fig. C-20
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-1.5

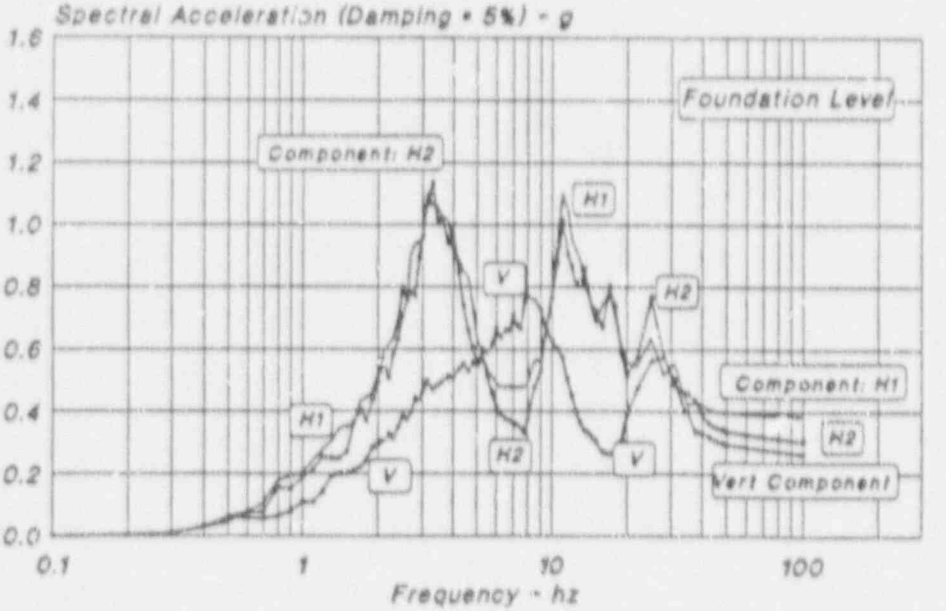

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Fig. C-21
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-3.5

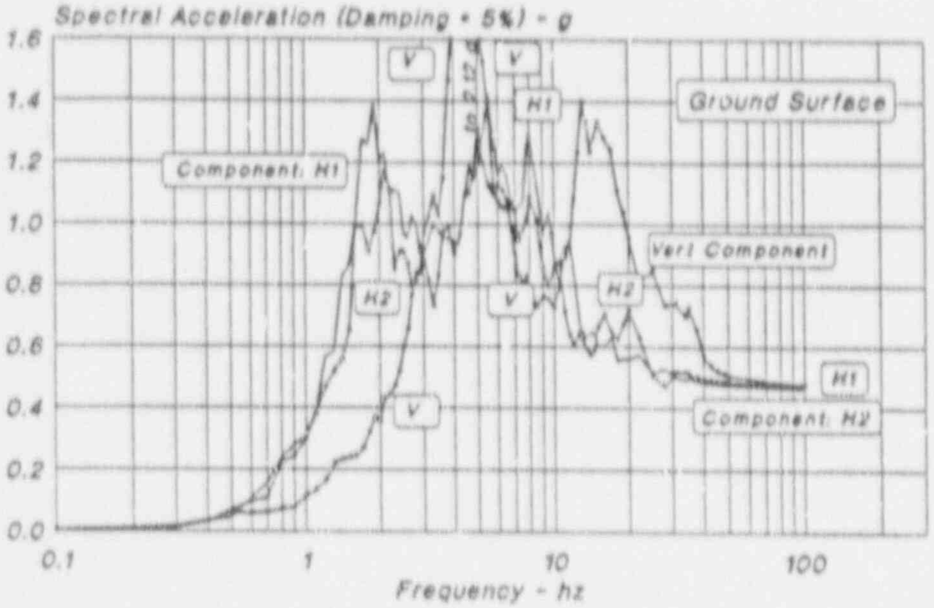

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Fig. C-22
CALCULA*ED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-3.5

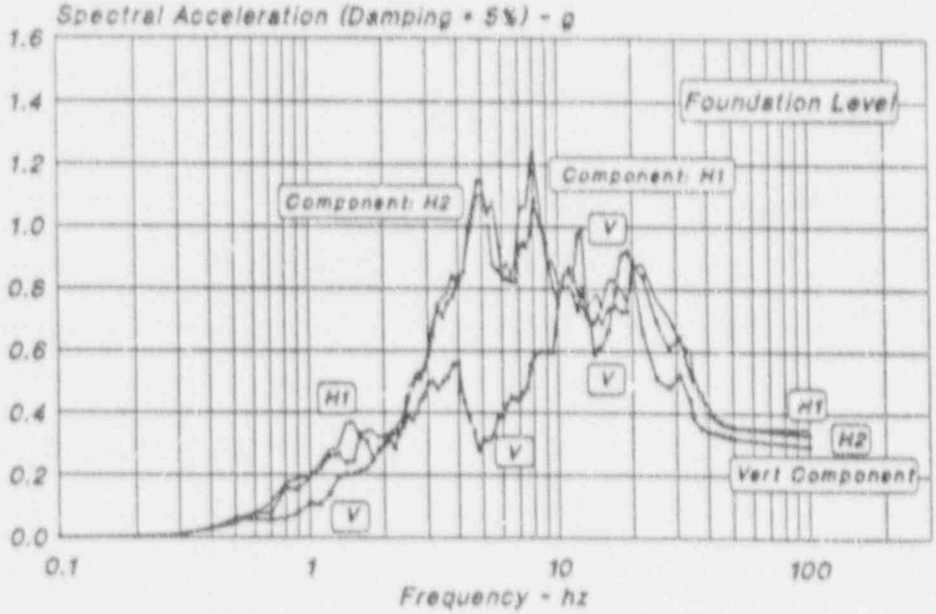

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Fig. C-23
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE C-1.5

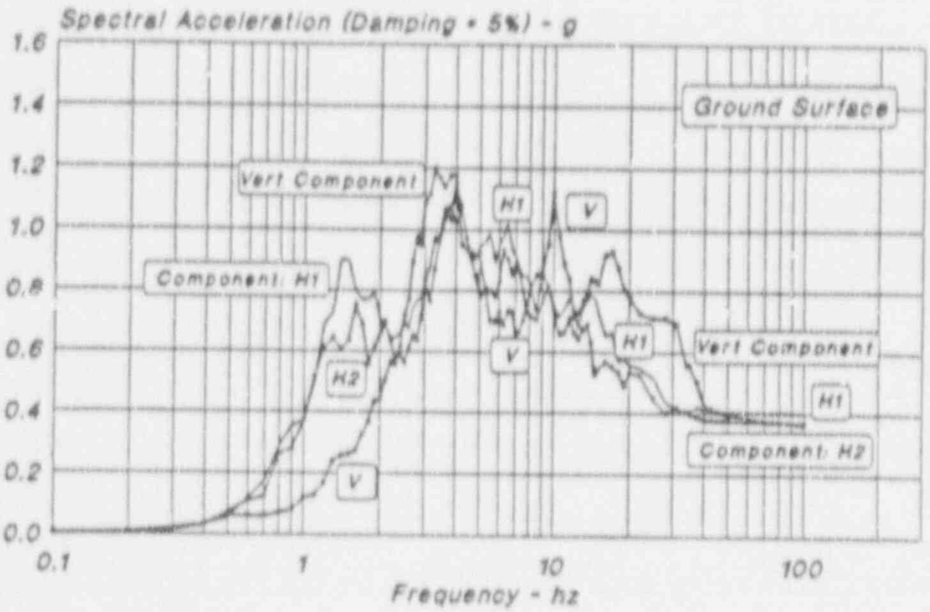

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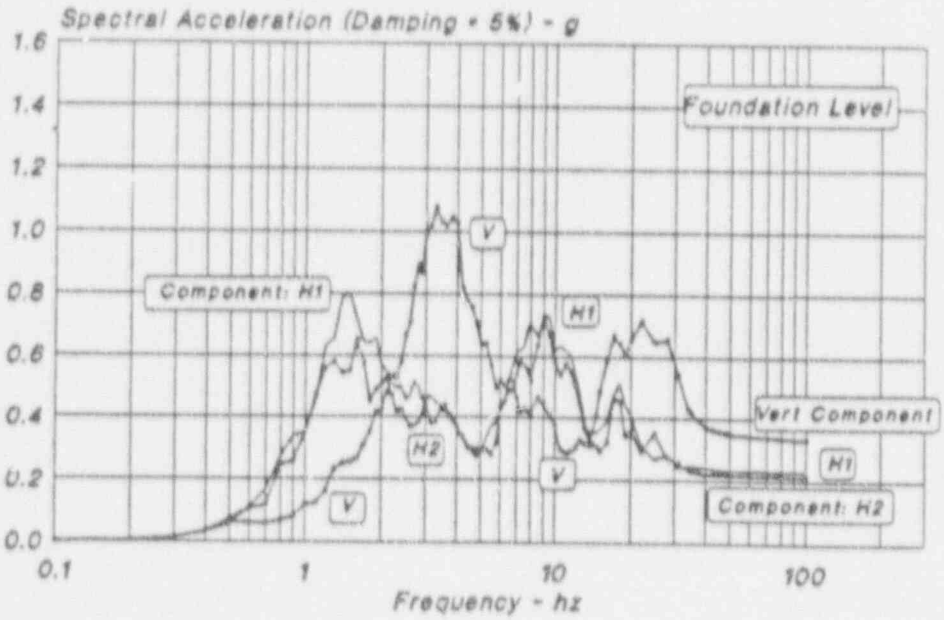
Fig. C-24
CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE C-1.5



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Question 220.5

Section 3.7.1.1 - The effect of differential seismic displacement on the equipment and supports is not described in Section 3.7.2.1 as stated in the last paragraph of Section 3.7.1.1. Provide a discussion of the effect.

Response 220.5

For analyses which are performed by the fully-coupled time-history method, the effect of differential support motions is inherently included by the methods described in Section 3.7.2. For subsystem response spectrum analyses, the effect of differential support motions is considered by the methods described in CESSAR-DC Section 3.7.3.1.

CESSAR-DC Section 3.7.1.1 will be corrected in a future amendment to make reference to Section 3.7.3.1.

RAI 220.5

3.7 SEISMIC DESIGN

3.7.1 SEISMIC INPUT

3.7.1.1 Seismic Input

This section discusses the seismic design parameters and methodologies being used for the design of those systems and subsystems important to safety and classified as Category I in Section 3.2.

The System 80+ Standard Design as defined by CESSAR-DC is not based on a specific site. Generic site conditions were selected to cover a range of possible conditions for the System 80+ sites. More specifically, sets of representative cases from each of four generic site categories were evaluated to create the ground surface and foundation level spectra shown in Figures 3.7-1 through 3.7-24. Out of 12 soil cases analyzed in Section 2.5.2, nine are used in the soil structure interaction (SSI) analysis. The three cases eliminated in the SSI analysis (A1, B3 and D1) were non-governing cases whose soil response levels were enveloped by other cases. See Section 2.5.2 for details of this analysis phase.

The effect of differential seismic displacement on the equipment and supports is included in the analysis as described in Section 3.7.~~2~~.1. 3 I

3.7.1.2 Design Time History

For the time history method of analysis, three design time histories are generated that are consistent with the design rock outcrop spectra at the free field. The characteristics of each time history are presented in Section 2.5.2.5.1. The response spectra plots for these time histories are shown in Figures 3.7-25 through 3.7-27.

3.7.1.3 Critical Damping Values

Damping values used for various nuclear safety-related structures systems and components are based upon Regulatory Guide 1.61 or ASME Code Case N-411-1 (See Figure 3.7-41). These values are expressed in percent of critical damping and are given in Table 3.7-1. When the response spectra method of analysis is used for piping, damping values are based on Code Case N-411-1.

Question 220.8

Section 3.7.1.2 - How were the time histories shown in Figures 3.7-25 through 3.7-27 developed? What rules were used to determine the compatibility of these time histories with the smoothed spectra? Identify the frequency intervals at which the spectra values were calculated. Are those smoothed spectra the rock outcrop spectra presented in Section 2.5?

Response 220.8

The time histories shown in Figs. 3.7-25 through 3.7-27 were generated using the program RASCAL (Silva and Lee, 1987). the frequencies used for calculating the spectrum are listed in Table 1 together with the calculated spectral ordinates for the target spectrum and the spectrum for the generated synthetic time history H1. (Note that the same frequencies were used for H2 and for the vertical components of the synthetic time histories). These spectra are plotted in Fig. 1.

The ratio of the spectral ordinates calculated for the synthetic time history divided by those for the target spectrum are listed in the fourth column of Table 1 and are plotted in Fig. 2. As can be noted from Fig. 2 and from Table 1, the spectrum for the synthetic time history is less than the target spectrum at the five frequencies listed below:

Frequency-Hz	Ratio
3	0.97
6.75	0.94
18	0.95
19	0.98
25	0.91

RAI 220.8

Table 1
Spectra for Target Spectrum And for Synthetic Time History

Frequency (Hz)	Target (g)	Time History (g)	Ratio
0.1		0.0037	
0.2	0.0270	0.0309	1.14
0.3	0.0500	0.0614	1.23
0.4	0.0730	0.0795	1.09
0.5	0.0960	0.1024	1.07
0.6	0.1150	0.1325	1.15
0.7	0.1340	0.1335	1.00
0.8	0.1530	0.1748	1.14
0.9	0.1719	0.1764	1.03
1.0	0.1909	0.2038	1.07
1.1	0.2099	0.2191	1.04
1.2	0.2289	0.2484	1.09
1.3	0.2479	0.2825	1.14
1.4	0.2669	0.3165	1.19
1.5	0.2859	0.3222	1.13
1.6	0.3048	0.3085	1.01
1.7	0.3238	0.3642	1.12
1.8	0.3428	0.3718	1.08
1.9	0.3618	0.3877	1.07
2.0	0.3808	0.3966	1.04
2.1	0.3998	0.4355	1.09
2.2	0.4188	0.4345	1.04
2.3	0.4377	0.4544	1.04
2.4	0.4567	0.4782	1.05
2.5	0.4757	0.5014	1.05
2.6	0.4947	0.5148	1.04
2.7	0.5137	0.5646	1.10
2.8	0.5327	0.5757	1.08
2.9	0.5517	0.5693	1.03
3.0	0.5706	0.5549	0.97
3.15	0.5991	0.6318	1.05
3.30	0.6276	0.6857	1.09
3.45	0.6349	0.6718	1.06
3.60	0.6369	0.6598	1.04
3.80	0.6395	0.6693	1.05
4.00	0.6419	0.6776	1.06
4.20	0.6442	0.6932	1.08

RAI 220.8

4.40	0.6464	0.6753	1.04
4.60	0.6484	0.7092	1.09
4.80	0.6505	0.6678	1.03
5.00	0.6524	0.6704	1.03
5.25	0.6547	0.6897	1.05
5.50	0.6569	0.7345	1.12
5.75	0.6589	0.7005	1.06
6.00	0.6609	0.6812	1.03
6.25	0.6629	0.7573	1.14
6.50	0.6647	0.7256	1.09
6.75	0.6665	0.6252	0.94
7.00	0.6682	0.7067	1.06
7.25	0.6698	0.7125	1.06
7.50	0.6714	0.6751	1.01
7.75	0.6730	0.7447	1.11
8.00	0.6745	0.8094	1.20
8.50	0.6773	0.6969	1.03
9.00	0.6800	0.7019	1.03
9.50	0.6826	0.7165	1.05
10.0	0.6850	0.7069	1.03
10.5	0.6873	0.7075	1.03
11.0	0.6895	0.7012	1.02
11.5	0.6915	0.7454	1.08
12.0	0.6935	0.7217	1.04
12.5	0.6955	0.6989	1.00
13.0	0.6973	0.7484	1.07
13.5	0.6991	0.7431	1.06
14.0	0.7008	0.7564	1.08
14.5	0.7024	0.7600	1.08
15	0.7040	0.7111	1.01
16	0.7071	0.7591	1.07
17	0.7099	0.7748	1.09
18	0.7126	0.6757	0.95
19	0.7152	0.7032	0.98
20	0.7176	0.7198	1.00
22	0.7220	0.7205	1.00
25	0.7281	0.6654	0.91
28	0.6248	0.6252	1.00
31	0.5321	0.5375	1.01
34	0.4480	0.4995	1.11
35	0.4216	0.4447	1.05
37.5	0.3588	0.4087	1.14

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40.0	0.3000	0.3954	1.32
42.5	0.3000	0.3606	1.20
45.0	0.3000	0.3530	1.18
47.5	0.3000	0.3468	1.16
50.0	0.3000	0.3428	1.14
100.0	0.3000	0.3004	1.00

RAI 220.8

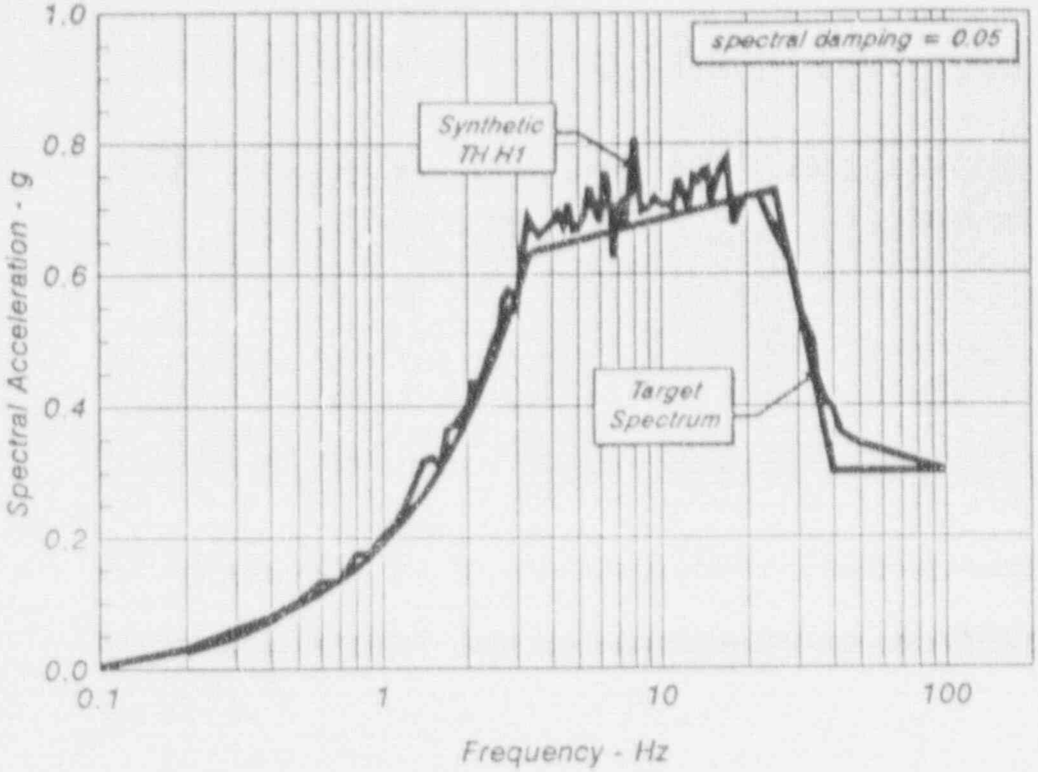


Fig. 1 Spectral Ordinates for the Target Spectrum and Spectral Ordinates for Synthetic Time History H1

IMI - 01/30/92

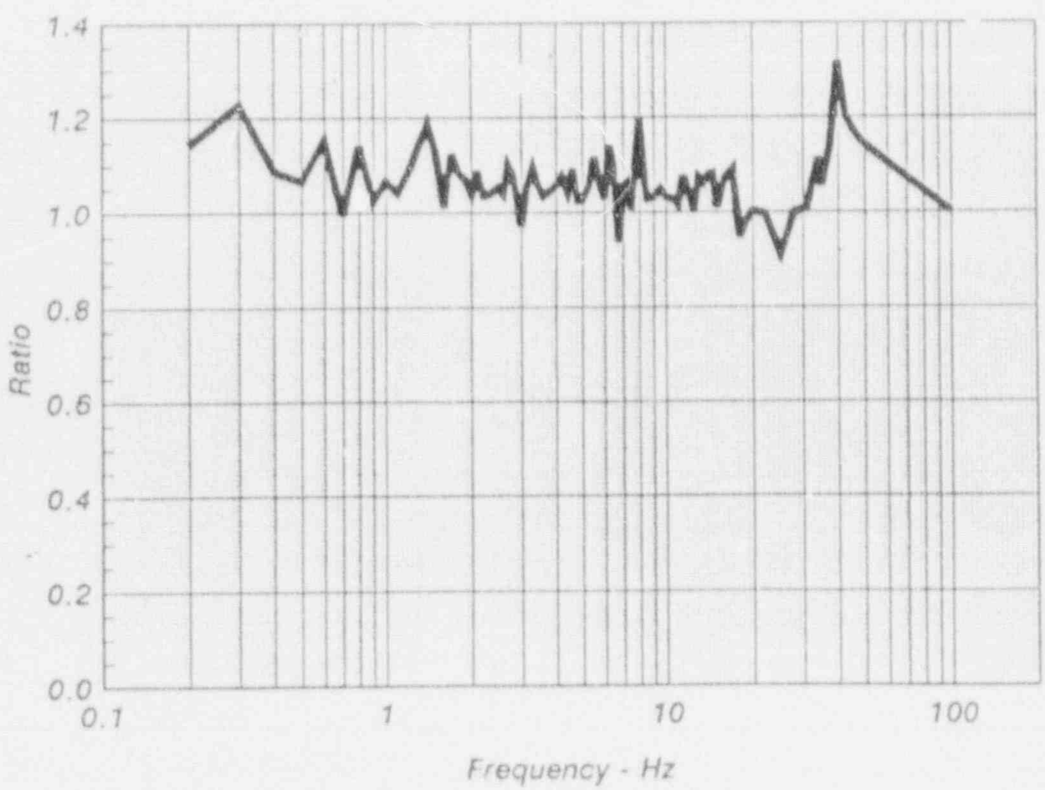


Fig. 2 Ratio of Spectral Ordinates for Synthetic Time History H1 Divided by the Spectral Ordinates for the Target Spectrum

IMI - 01/30/92

Question 22C.10

Section 3.7.1.4 - There is no Reference 21.

Response 220.10

The correct reference is #7. This correction will be made in a future amendment to CESSAR-DC.

RAI 220.10

3.7.1.4 Supporting Media for Seismic Category I Structures

Category I structures are founded directly on rock or competent soil. The foundation embedment depth for System 80+ standard plant is 52 feet (Reference ~~21~~ 7). The rock properties and the layering characteristics, including shear wave velocity, shear modulus, and density, are given in Section 2.5. The System 80+ Standard Plant is designed for the range of soil conditions discussed in Section 2.5 and shown in Appendix 3.7B.

3.7.1.4.1 Soil Structure Interaction (SSI)

Two different types of analysis methodologies are used for the seismic analyses. For the fixed-base cases, modal superposition time history analyses are performed using the rock outcrop motions as control motions. When a structure is supported on soil, the SSI is taken into account by coupling the structural model with the soil medium. To accomplish this, the methodology of the computer program SASSI (System for Analysis of Soil Structure Interaction, Reference 6) is used. Detailed methodology and results of the SSI analysis are presented in Appendix 3.7B.

3.7.2 SEISMIC SYSTEM ANALYSIS

3.7.2.1 Seismic Analysis Method

3.7.2.1.1 Seismic Category I Structures, Systems, and Components Other Than NSSS I

The Reactor Building (RB) is modeled as a multi-degree of freedom system for the seismic analysis. Figures 3.7-28 through 3.7-30 show typical sketches of the three structural components of the overall model - Internal Structure (IS), Shield Building (SB), and Steel Containment Vessel (SCV). Figure 3.7-31 is a schematic representation of the combined structural model of the RB. The RB is modeled as a lumped mass-spring model.

Further details of dynamic modeling of building structures for seismic analysis are described in Section 3.7.2.3. The horizontal models are analyzed for the plant E-W direction and N-S direction excitations and the vertical model for vertical excitation. The results are then combined as described in Section 3.7.2.6. The seismic analysis of the above systems is performed by one of the following methods:

Question 220.11

Section 3.7.2.1 - Provide discussion on how the effects of relative displacement among supports is considered in a seismic analysis. Discuss also the manner in which the foundation torsion and rocking are handled in a seismic analysis using input basemat motion derived from a soil-structure interaction analysis. (see also Comment 220.5)

Response 220.11

Relative displacement at supports is considered when different supports of a structure are excited by different input excitations. In the analysis of the coupled components of the RCS, excitations are input at a single point, the containment basemat. For the coupled components of the RCS the relative support displacements are inherently accounted for during the coupled analysis. The basemat motion derived from the soil-structure interaction analysis consists of six time histories per soil case, three linear and three rotational. For each soil case all six time history motions are applied at the containment basemat.

The calculated motions for input to subsequent subsystem analyses therefore include the motions caused by the foundation torsion and rocking.

Question 220.12

Section 3.7.2.1.1 - The mesh of the finite element model shown in Figure 3.7-30 seems coarse. Is there any parametric study performed to determine if stiffness characteristics of the containment vessel are properly represented by these coarse mesh finite elements?

Response 220.12

The finite element model of the Steel Containment Vessel (SCV) shown in Figure 3.7-90 was used only in the soil-structure interaction (SSI) analysis. Typically, stick models are used in SSI analyses of containment structures. However, in order to capture more accurately "breathing" modes and other higher order modes of the containment, a finite element model with shell elements was developed for the SCV.

A comparison of the SCV SSI model with a more detailed axisymmetric model was made in order to fine tune the natural frequencies and mass participation factors of the SCV SSI model. The axisymmetric model shown in Figure 220.12.1 was developed with the computer program ANSYS and was used as a benchmark for this fine tuning. The SCV SSI model properties were adjusted to get a match of dynamic properties with the more detailed ANSYS model.

RAI 22. '2

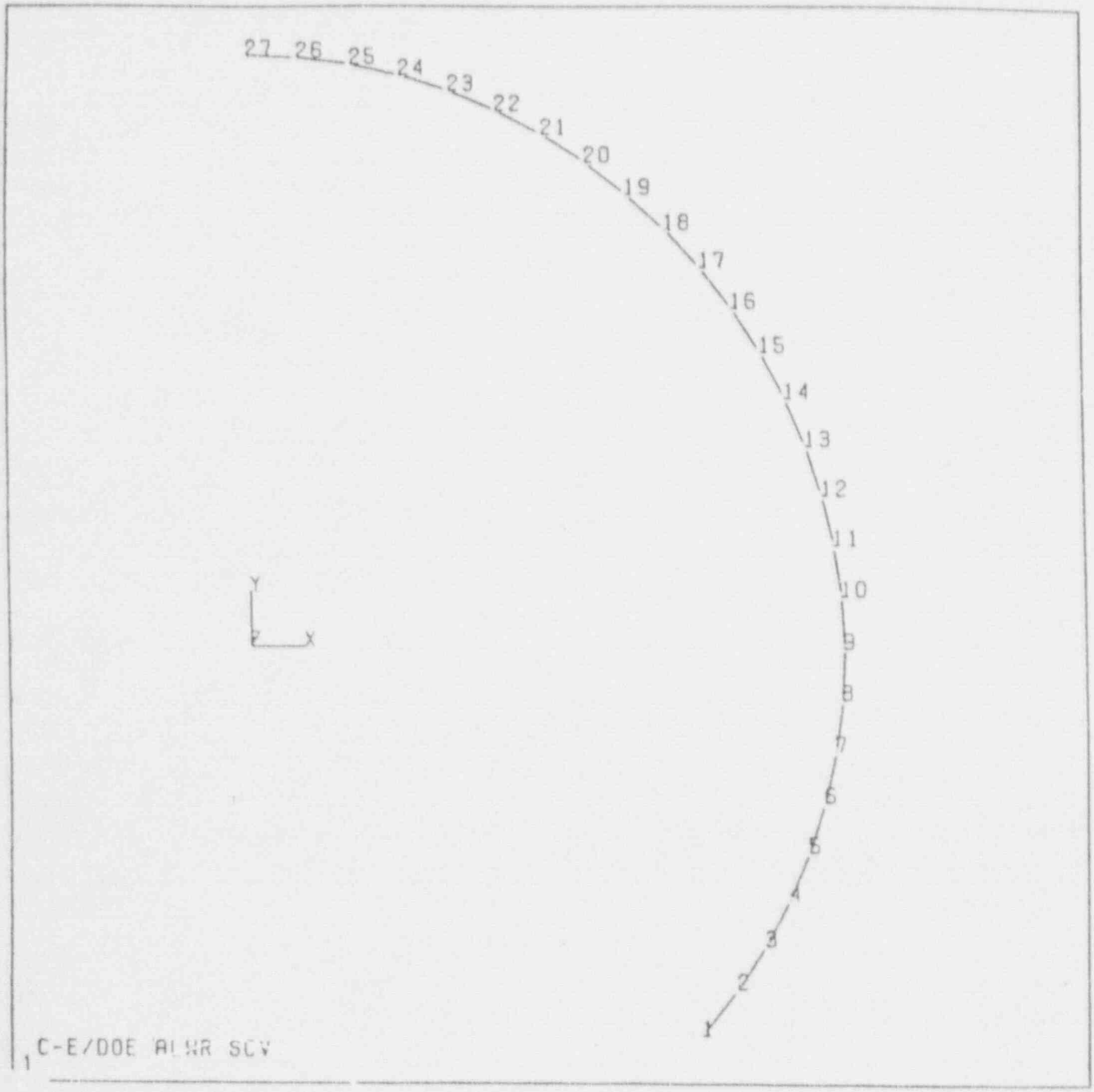


Figure 220.12-1 - Axisymmetric model of the SCV used in fine-tuning of the SSI model.

Question 220.13

Section 3.7.2.1.1 - This section did not present all Seismic Category I structures, systems, and components as the title indicated. List all Category I structures, systems, and components and identify the method of analysis for each.

Response 220.13

Seismic Category I structures, systems, and components are identified in CESSAR-DC Table 3.2-1. The methods of seismic analysis include response spectrum, and time history which are described in Section 3.7.2. An equivalent static load method of analysis is used for some subsystems as described in Section 3.7.3.5. The appropriate analysis method is used during the detailed design phase.

CESSAR-DC Section 3.7.2.1.1 will be revised in a future amendment to add a reference to Table 3.2-1.

RAI 220.13

3.7.1.4 Supporting Media for Seismic Category I Structures

Category I structures are founded directly on rock or competent soil. The foundation embedment depth for System 80+ standard plant is 52 feet (Reference 21). The rock properties and the layering characteristics, including shear wave velocity, shear modulus, and density, are given in Section 2.5. The System 80+ Standard Plant is designed for the range of soil conditions discussed in Section 2.5 and shown in Appendix 3.7B.

3.7.1.4.1 Soil Structure Interaction (SSI)

Two different types of analysis methodologies are used for the seismic analyses. For the fixed-base cases, modal superposition time history analyses are performed using the rock outcrop motions as control motions. When a structure is supported on soil, the SSI is taken into account by coupling the structural model with the soil medium. To accomplish this, the methodology of the computer program SASSI (System for Analysis of Soil Structure Interaction, Reference 6) is used. Detailed methodology and results of the SSI analysis are presented in Appendix 3.7B.

3.7.2 SEISMIC SYSTEM ANALYSIS

3.7.2.1 Seismic Analysis Method

3.7.2.1.1 Seismic Category I Structures, Systems, and Components Other Than NSSS I

SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS ARE IDENTIFIED IN TABLE 3.2-1.

~~The Reactor Building (RB) is modeled as a multi-degree of freedom~~ system for the seismic analysis. Figures 3.7-28 through 3.7-30 show typical sketches of the three structural components of the overall model - Internal Structure (IS), Shield Building (SB), and Steel Containment Vessel (SCV). Figure 3.7-31 is a schematic representation of the combined structural model of the RB. The RB is modeled as a lumped mass-spring model.

Further details of dynamic modeling of building structures for seismic analysis are described in Section 3.7.2.3. The horizontal models are analyzed for the plant E-W direction and N-S direction excitations and the vertical model for vertical excitation. The results are then combined as described in Section 3.7.2.6. The seismic analysis of the above systems is performed by one of the following methods:

Question 220.14

Section 3.7.2.1.1 - The statement "The horizontal models are analyzed for the plant E-W direction and N-S direction..." seems to indicate that two horizontal models are used for each building structure. Since the two horizontal directions are usually coupled, describe the reason for using two separate horizontal models. Describe also the differences in these two models and how the results of these two models are combined and used.

Response 220.14

There is only one horizontal model used in the SSI analysis of the Internal Structure. This horizontal model is a 3D model but was subjected to both the 0°-180° and 90°-270° motions independently. The results of the analyses were then combined with the vertical model results in the time domain. Section 3.7.2.1.1 is modified accordingly and will be included in a future amendment to CESSAR-DC.

3.7.1.4 Supporting Media for Seismic Category I Structures

Category I structures are founded directly on rock or competent soil. The foundation embedment depth for System 80+ standard plant is 52 feet (Reference 21). The rock properties and the layering characteristics, including shear wave velocity, shear modulus, and density, are given in Section 2.5. The System 80+ Standard Plant is designed for the range of soil conditions discussed in Section 2.5 and shown in Appendix 3.7B.

3.7.1.4.1 Soil Structure Interaction (SSI)

Two different types of analysis methodologies are used for the seismic analyses. For the fixed-base cases, modal superposition time history analyses are performed using the rock outcrop motions as control motions. When a structure is supported on soil, the SSI is taken into account by coupling the structural model with the soil medium. To accomplish this, the methodology of the computer program SASSI (System for Analysis of Soil Structure Interaction, Reference 6) is used. Detailed methodology and results of the SSI analysis are presented in Appendix 3.7B.

3.7.2 SEISMIC SYSTEM ANALYSIS

3.7.2.1 Seismic Analysis Method

3.7.2.1.1 Seismic Category I Structures, Systems, and Components Other Than NSSS I

The Reactor Building (RB) is modeled as a multi-degree of freedom system for the seismic analysis. Figures 3.7-28 through 3.7-30 show typical sketches of the three structural components of the overall model - Internal Structure (IS), Shield Building (SB), and Steel Containment Vessel (SCV). Figure 3.7-31 is a schematic representation of the combined structural model of the RB. The RB is modeled as a lumped mass-spring model.

Further details of dynamic modeling of building structures for seismic analysis is ~~are~~ described in Section 3.7.2.3. The horizontal models ~~are~~ analyzed for the plant 0°-180° ~~E-W~~ direction and 90°-270° ~~N-S~~ direction excitations and the vertical model for vertical excitation. The results are then combined as described in Section 3.7.2.6. The seismic analysis of the above systems is performed by one of the following methods:

Question 220.15

Section 3.7.2.1.1 - The statement "The seismic analysis of the above systems is performed by one of the following methods:" may not be appropriate because the combined reactor building model described in the first paragraph of this section is a SASSI model as shown in Figure 3.7-31, and SASSI methodology is different from those methods presented in Section 3.7.2.1.1.

Response 220.15

The SSI analysis was performed using the substructuring method and complex response (frequency domain analysis). CESSAR-DC will be revised in a future amendment to include the attached additional subsection:

3.7.2.1.1.3 Soil-Structure Interaction Analysis

The soil-structure interaction analyses were performed using the substructures method formulated in the frequency domain using the complex response method and the finite element technique. The methodology of the computer program SASSI was used with a modified approach to compute the impedance and scattering of the soil/foundation system. Appendix 3.7B describes the SSI analysis approach for the System 80+ structures. A summary of the method is provided below.

In the substructures method, the soil strata are analyzed first, in the frequency domain. The impedance and scattering properties of the soil-structure interface are computed next, and they are used as boundary conditions in a dynamic analysis of the structure with a loading that depends on the free-field conditions. The solution of the SSI problem is performed in three steps:

- Solution of the site response problem to determine the free-filed motions within the embedded part of the structure. For horizontal motions, vertically propagating S-waves are considered. For vertical motion, vertically propagating P-waves are considered.

- Solution of the impedance and scattering problem.

- Solution of the structural problem. This involves forming the complex stiffness matrices and load vector and solving the equations of motion for the final displacements and accelerations.

RAI 220.15

B. Direct Integration Method

In this method, direct integration of the equations of motion by either implicit or explicit methods of numerical integration are used to solve the equations of motion.

For commonly used implicit methods, ΔT is not larger than 1/10 of the shortest period of interest.

For explicit methods, the time step is also a function of the element size used in the model and is established on the basis of element size to ensure stability of the response.

Insert

3.7.2.1.2 Seismic Analysis Method for the NSSS

3.7.2.1.2.1 Introduction

The major components of the reactor coolant system are designed to the appropriate stress and deformation criteria of ASME Code, Section III, for the set of loadings included in the component design specification. The adequacy of seismic loadings used for the design of the major components of the reactor coolant system are confirmed by the methods of dynamic analysis employing time history and response spectrum techniques. The major components are the reactor vessel, the steam generators, the reactor coolant pumps, the reactor coolant main loop piping, the surge line and the pressurizer.

Detailed dynamic models of the building structures and the NSSS are generated. Based on these detailed models, equivalent, simplified dynamic models are developed. The simplified building and NSSS models are combined and translated into a form suitable for input to the SSI analysis code (see Section 3.7.1.4.1). A number of soil cases are modeled and the time history analyses are performed. The soil cases are chosen to envelope all potential building sites. The results of these analyses are contained in Appendix 3.7B. These results, the simplified building model(s), and the detailed NSSS model are used to perform the analysis discussed in Section 3.7.2.1.2.3. I

A composite three-dimensional lumped-mass model of the reactor vessel, the two steam generators, the four reactor coolant pumps, the pressurizer, and the interconnecting main loop piping is coupled with a three-dimensional lumped-mass model of the reactor building for performing the analysis of these dynamically coupled components of the reactor coolant system. In addition, the representation of the reactor vessel assembly used in this coupled model includes sufficient detail of the reactor internals to account for possible dynamic interaction between the reactor coolant system and internals. The seismic input excitation is the basemat acceleration time histories. The results of this

Question 220.16

Section 3.7.2.1.1 - Does the combined model shown in Figure 3.7-31 include any simplified building or NSSS models? What are these simplified models? What is the purpose of this combined model? Provide some analysis results of this combined model.

Response 220.16

The schematic of the model shown in Figure 3.7-31 represents the structural model of the Nuclear Island of the System 80+ Standard Design, which is used in the SSI analyses. The Nuclear Island model is composed of the models of the Internal Structure (IS), the Steel Containment Vessel (SCV), the Shield Building (SB) and the NSSS, together with the models of the Fuel Building, Control Building, Auxiliary Building and Diesel Generator Buildings (Nuclear Annex structures).

The IS and SB stick models were developed according to methodology describ in CESSAR Sections 3.7.2.3.4.1.1 and 3.7.2.3.4.1.2. No f ther simplification was performed on the stick models for se in the SSI analysis.

As described in Sect 3.7.2.3.4.1 of CESSAR-DC, the IS stick model was deve ed and fine tuned based on a detailed 3D finite element mod... The detailed 3D finite element model is shown in Figure 220.16-1, and it was developed according to the procedure described in Section 3.7.2.3.4.1.1 of CESSAR-DC. In a similar manner, the SB stick model was developed and fine tuned based on a detailed axisymmetric model of the SB, shown in Figure 220.16-2.

The IS and SB stick models are essentially co-axial (except for floor eccentricities in the IS). Since the IS and the SB are monolithically constructed, the stick models of these structures are connected with a rigid link at elevation +115 ft. The SCV is also connected to the IS with rigid links at elevation +91 ft. The NSSS model used in the SSI analysis is shown in Figure 220.16-3 and is connected to the IS stick model at several elevations using appropriate links whose properties depend on the flexibility of the connection. The NSSS SSI model is a simplified model of the detailed NSSS model shown in Figure 3.7-32 of CESSAR-DC. This simplified model is only used in the SSI analyses.

The Nuclear Annex structures are approximated by two-node/single element lumped parameter models. Mass and stiffness properties for these models represent a first

RAI 220.15

Insert

3.7.2.1.1.3 Soil-Structure Interaction Analysis

The soil-structure interaction analyses were performed using the substructures method formulated in the frequency domain using the complex response method and the finite element technique. The methodology of the computer program SASSI was used with a modified approach to compute the impedance and scattering of the soil/foundation system. Appendix 3.7B ~~described~~ describes in detail the SSI analysis approach for the System 80+ structures. A brief summary of the method is described below.

In the substructures method, the soil strata are analyzed first, in the frequency domain. The impedance and scattering properties of the soil-structure interface are computed next, and they are used as boundary conditions in a dynamic analysis of the structure with a loading that depends on the free-field conditions. The solution of the SSI problem ~~was~~ is performed in three steps:

- Solution of the site response problem to determine the free-filed motions within the embedded part of the structure. For horizontal motions, vertically propagating S-waves ~~were~~ are considered. For vertical motion, vertically propagating P-waves ~~were~~ are considered.
- Solution of the impedance and scattering problem.
- Solution of the structural problem. This ~~involved~~ involves forming the complex stiffness matrices and load vector and solving the equations of motion for the final displacements and accelerations.

Response 220.16 (cont'd)

order approximation of the dynamic properties of these structures. These models are included to represent the effects of structure-to-structure interaction as it pertains to the Nuclear Island response.

The fixed-base frequencies, modes and mass participation factors of the combined Nuclear Island model up to 30 Hz are provided in Reference 8 of CESSAR Section 3.7. The fundamental horizontal modes of the Nuclear Island model are at 5.53 and 5.58 Hz in the 90-270 and 0-180 directions, respectively. The fundamental vertical mode is at 11.79 Hz. The first eight modes and their associated participation factors are provided below (from Reference 8 of CESSAR Section 3.7).

RAI 220.16

Fixed-Base Modes and Frequencies of Nuclear Island
(all sticks combined)

Mode No.	Frequ. (Hz)	Cumulative Mass Part. Factors (%) 0-180	90-270	Vert.	Description of Mode
1	5.53	0.0	35.7	0.0	90-270, Main
2	5.58	28.9	35.7	0.0	0-180, Main
3	6.07	29.0	51.6	0.0	90-270, 2nd
4	6.15	48.6	51.6	0.0	0-180, 2nd
5	8.90	48.6	64.1	0.0	90-270, 3rd
6	9.79	70.6	64.1	0.0	0-180, 3rd
7	10.76	70.7	64.1	0.0	Local
8	11.79	70.7	64.1	24.5	Vertical, Main

RAI 220.16

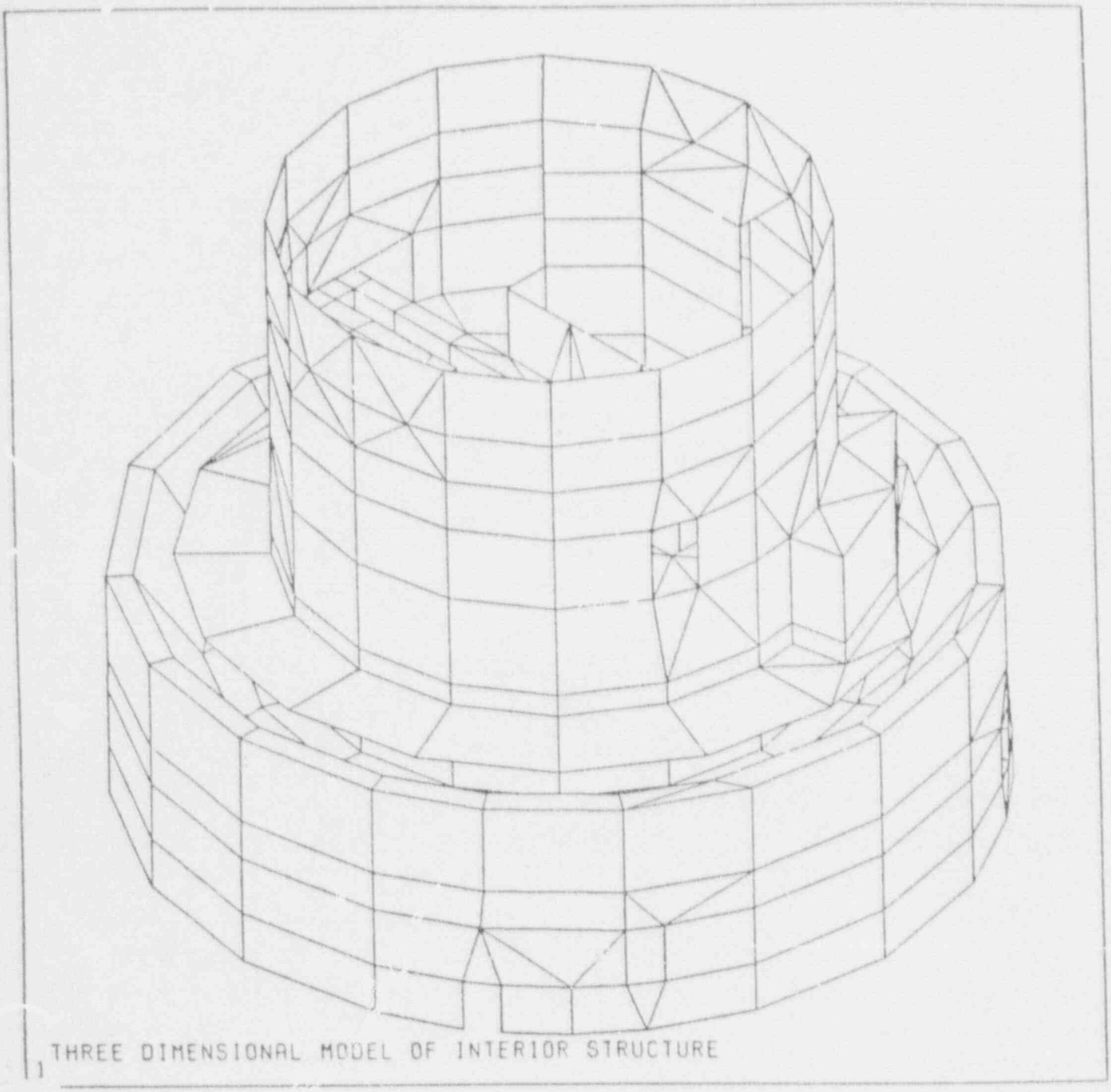


Figure 220.16-1 - Detailed 3D Finite Element Model of Internal Structure

RAI 220.16

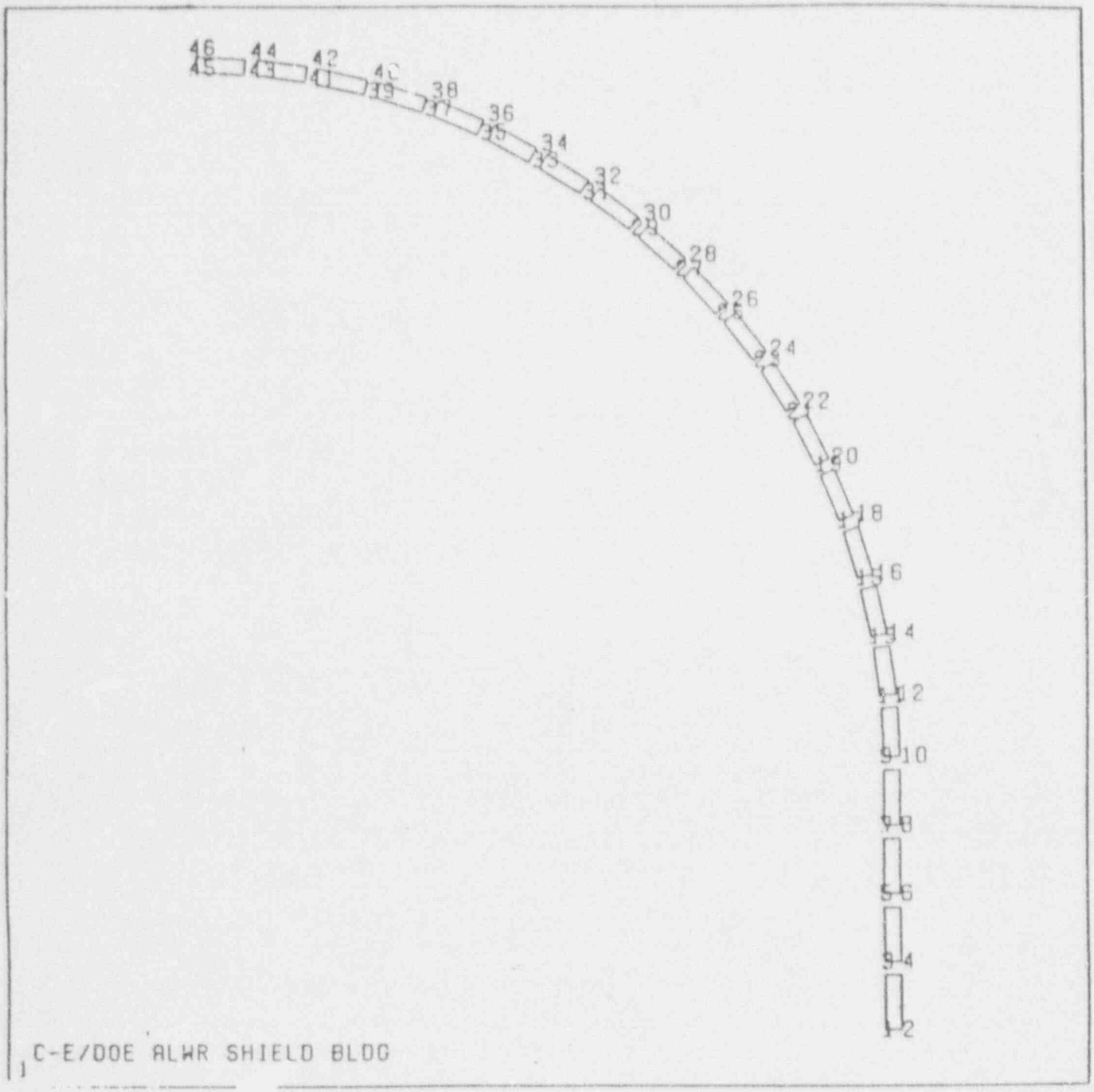


Figure 220.16-2 - Detailed Axisymmetric Model of Shield Building

RAI 220.16

Note: For illustrative purposes only; please see CESSAR-DC Figure 3.7-32 for appropriate definition of support restraints.

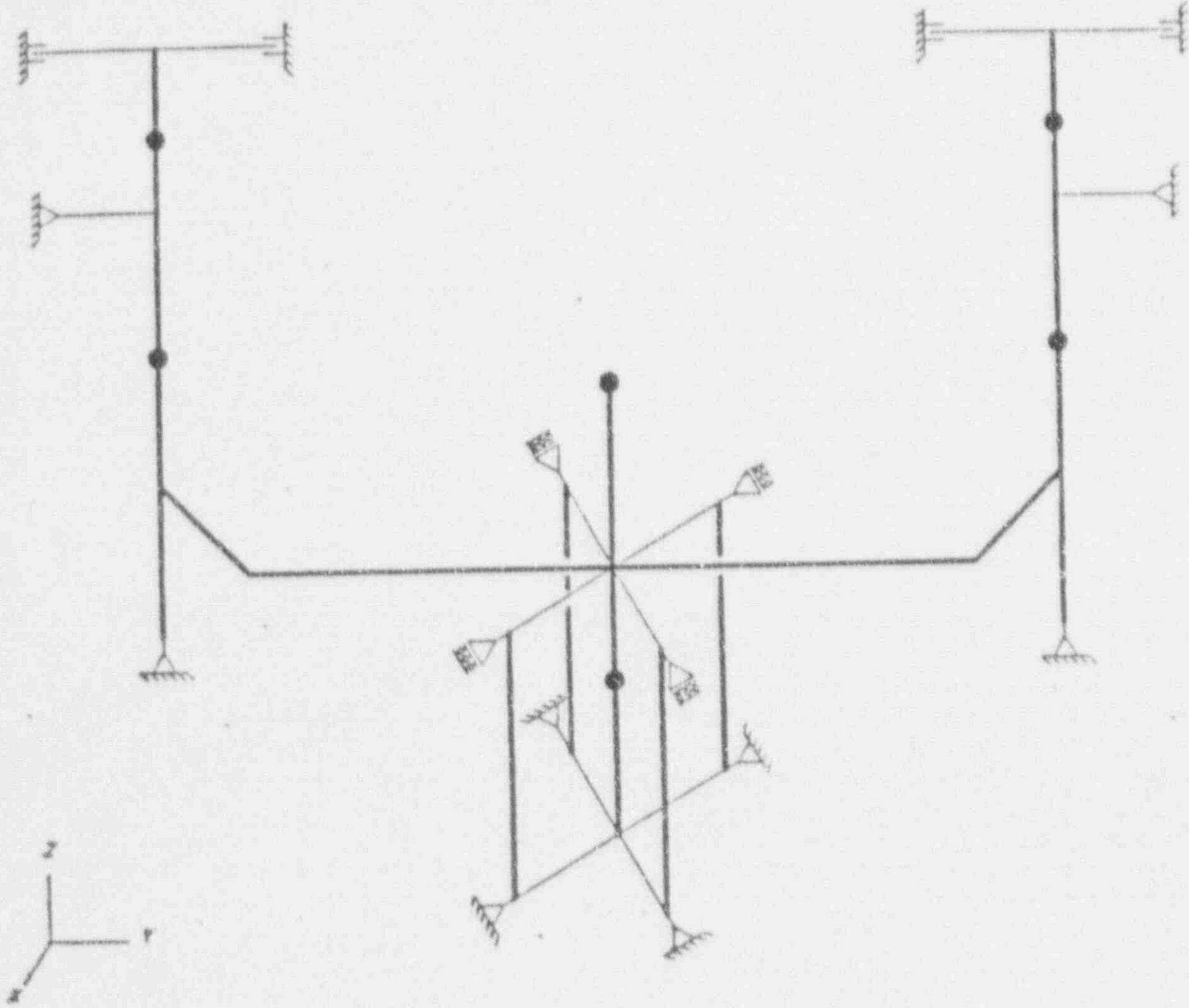


Figure 220.16-3 - Simplified NSSS Model for SSI Analysis

Question 220.17

Section 3.7.2.1.2.1 - The second paragraph of this section stated that "Detailed dynamic models of the building structures and the NSSS are generated. Based on these detailed models, equivalent, simplified dynamic models are developed.' List the structures that have a simplified model. Are these simplified models used anywhere else besides the SSI model discussed here and in Appendix 3.7B?

Response 220.17

As stated in Section 3.7.2.3.4.1 of CESSAR-DC, the modeling approach that was used for the Nuclear Island SSI model was to:

1. Develop a detailed 3D FEM of the IS. Use this model as the basis to develop and fine tune a stick model of the IS to be used in the SSI analyses.

2. Develop an axisymmetric model of the SB. Use this model as the basis to develop and fine tune a stick model of the SB to be used in the SSI analyses.

3. Develop the 3D model shown in CESSAR figure 3.7-30 for the SCV. Use this 3D model in the SSI analyses. (No stick model was developed for the SCV for SSI analysis).

4. Develop two node/single element lumped parameter models for the Nuclear Annex structures.

5. Develop a simplified model of the NSSS based on the detailed model shown in Figure 3.7-32 of CESSAR-DC. Use the simplified model in the SSI analyses of the combined Nuclear Island model.

The stick models were used only in the following two cases:

1. The stick models of the IS and the SB, the 3D model of the SCV and the simplified model of the NSSS were used in the SSI analyses of the combined Nuclear Island model.

2. The stick models of the IS and SB were also used as the support structures in the dynamic analysis of the detailed NSSS model.

Question 220.18

Section 3.7.2.1.2.1 - How many SASSI computer runs were performed for all soil conditions considered in Section 3.7.1.1? It is expected that there are a lot of time histories available because there are many soil cases and there are at least one P-wave and two S-waves. How were the basemat acceleration time histories obtained in these runs used for the analysis of various NSSS models described in Section 3.7.2.1.2.1 and Figures 3.7-32 through 3.7.39? Provide responses to these questions also for the combined model and the structures discussed in Section 3.7.2.1.1.

Response 220.18

The following soil cases and seismic events were performed to compute the SSI response of the Nuclear Island model:

CASE	SSE	OBE
B1	Yes	No
B1.5	Yes	No
B2	Yes	No
B3.5	Yes	Yes
B4	Yes	Yes
C1	Yes	No
C1.5	Yes	No
C2	Yes	No
C3	Yes	No

For each of the above cases, SASSI runs were performed according to the sequence shown in the flow diagram of Figure 220.18-1. For horizontal excitation, two runs were performed using vertically propagating S-waves, one with excitation in the 0-180 direction and one with excitation in the 90-270 direction. For vertical excitation, one run was performed using vertically propagating P-wave. For each of the three runs, at every node of the structure, response was obtained at all six degrees of freedom, i.e., three translational (X, Y, Z) and three rotational (XX, YY, ZZ). The response time histories at each degree of freedom were subsequently added algebraically in the time domain to obtain the combined response time histories due to the three excitations. Finally, the combined response time histories were used as input to generate response spectra at the selected building locations.

The combined response acceleration time histories at the center of the Nuclear Island basemat (three translational and three

Response 220.18

rotational) obtained from the SASSI analysis of each soil case are used as input to the seismic analysis of the detailed NSSS model (shown in Figure 3.7-32 of CESSAR-DC).

The rotational time histories are included in the NSSS analysis in order to properly account for the effects of rocking and torsion of the basemat. In this analysis, the detailed NSSS model is attached on a fixed-base stick model of the IS. The response acceleration time histories of the basemat are used as the base excitation of the IS stick model.

RAI 220.18

Figure 220.18-1 - SSI Analysis Flow Diagram

Question 220.19

Section 3.7.2.1.2.1 (third paragraph) - Is the three-dimensional lumped-mass model of the reactor building included in the NSSS model shown in Figure 3.7-32 a simplified model? Are the internal structure, steel containment, and shield building included in this reactor building model?

Response 220.19

The reactor buildin~ model included in the NSSS model includes simplifiea m.odels of the internal structure, the steel containment and the shield building. These simplified models have a total of 37 mass points with 210 DDOF.

Question 220.20

Section 3.7.2.1.2.1 - Are the effects of basemat rocking and torsion included in the analysis of the coupled model of NSSS and reactor building described in the third paragraph of this section? The third paragraph of Page 3.7-6 seems to indicate that three orthogonal direction of seismic excitations are applied without considering rocking and torsional motion of the basemat. How is the rocking and torsional effects of the basemat considered if the response spectrum method of analysis described in the fourth paragraph of Page 3.7-6 is used? (see also Comment 220.11)

Response 220.20

The rocking and torsion of the basemat are fully included in the analysis. All six time history motions, three linear and three rotational, are applied at the basemat.

For subsystem analyses using the response spectrum method, the input spectra used envelope all subsystem support locations. The effects of rocking and torsion are implicitly included because the spectra at the support points includes motions due to rocking and torsion and because differential support displacement effects are conservatively included using the ABSUM combination method.

Question 220.21

Section 3.7.2.1.2.3 - If response spectrum method is used (e.g., for surge line), will floor envelope spectra be used? How will the rocking effects be included? (see also Comments 220.11 and 220.20)

Response 220.21

If the response spectrum method is used to analyze the pressurizer, the spectrum used will envelope spectra of the hot leg nozzle, pressurizer nozzle, and building support locations. The effects of rocking and torsion are implicitly included because the spectra at the support points include motions due to rocking and torsion and because differential support displacement effects are conservatively included using the ABSUM combination method.

Question 220.22

Section 3.7.2.3 - How are the hydrodynamic effects of structures, systems, and components such as the IRWST considered in the seismic analysis? Discuss modeling of nonlinear characteristics of structures, systems, and components, such as gaps and 1-D tension elements, where applicable.

Response 220.22

The hydrodynamic effects of tanks such as the IRWST and the emergency feedwater tanks ar counted for in the overall seismic analysis by calcula ne mass assuming the tanks to be full. For the Soil St ure Interaction Analysis, this mass is applied at an elevation higher than the actual center of mass location. This method is used in the modal analysis due to the inability to incorporate nonlinear elements in the solution.

Refer to CESSAR-DC Section 3.7.3.14.1 for consideration of hydrodynamic effects and nonlinear characteristics in the seismic analysis of the reactor internals and core. With regard to other systems, structures, and components, consideration of hydrodynamic effects will be incorporated into the detailed design of these items.

Question 220.23

Section 3.7.2.3.4 - Second paragraph of Section 3.7.2.3.4 states that "Two independent models are used for the seismic analysis of the Internal Structure." The second paragraph of Section 3.7.2.3.4.1 states that "The modeling approach that is used for the Reactor Building (RB) structural model consists of developing a 3-D finite element model (FEM) of the internal structure (IS) and an axisymmetric FEM of the Shield Building (SB) and, based on the FEM models, developing equivalent 3-D lumped parameter stick models." How many stick models of the internal structure are available? What are the differences among these models? Explain also why an axisymmetric Shield Building FEM, which is 2-D, can be used to develop an equivalent 3-D stick model.

Response 220.23

There are two stick models of the IS: one which is used in the SSI analyses with horizontal excitation, and one which is used in the SSI analyses with vertical excitation. The only difference between these two models is the location of the center of rigidity of each of the beam elements of the stick model. This is because for horizontal analysis, the center of rigidity of the structural system between two adjacent floors is the shear center of that system, while for vertical analysis, the center of rigidity is the centroid. Therefore, the eccentricities of the center of mass of each floor to the centers of rigidity of the beam elements directly above and below that floor are different between the two IS stick models.

Although axisymmetric models are 2D models, by definition they constitute 3D representations of structures which exhibit axisymmetric properties. The axisymmetric model of the SB was used only as a benchmark in the development of the SB stick model. This model was not used as the starting point to develop the 3D stick model. The modal analysis of the axisymmetric model with a refined mesh provided accurate natural frequencies and mass participation factors of the SB. The SB stick model was then developed by: (1) calculating mass and stiffness properties of cylindrical concrete sections and (2) using the frequencies and mass participation factors from the axisymmetric analysis as benchmark values to fine tune the lumped masses and the stiffness properties of the stick beam elements.

Question 220.24

Section 3.7.2.8 - A statement was made that non-safety-related structures adjacent to safety-related-structures are designed so that their failure under SSE conditions will not cause the failure of the safety-related structures. Describe the design criteria and the analysis methods that will be applied to the non-safety-related structures to ensure protection of the safety-related-structures.

Response 220.24

Section 3.7.2.8 will be revised to include the following:

The following procedure is used to ensure that the failure of a Non-Seismic Category I (non-safety-related) structure under the effect of a seismic event does not impair the integrity of an adjacent Seismic Category I (safety-related) structure.

a. Sufficient separation between Non-Seismic Category I structure and Seismic Category I structure is maintained, or

b. The Non-Seismic Category I structure is designed to withstand the effect of the postulated SSE event, i.e., to maintain the structural integrity of the Non-Seismic Category I structure during and after the occurrence of the postulated SSE event, or

c. The Non-Seismic Category I structure is designed such that if the Non-Seismic Category I structure collapses, it will fall away from any Seismic Category I structure.

Letter ALWR-338
RAI 220.24

insert paragraph per attached page

B. In the analysis of complex system where closely spaced modal frequencies are encountered, the responses of the closely space modes are combined by the summation of the absolute values method and, in turn, combined with the responses of the remaining significant modes by the SRSS method. Modal frequencies are considered closely spaced when their difference is less than ±10 percent of the lower frequency.

3.7.2.8 Interaction of Non-Safety-Related Structures with Safety-Related Structures

When safety-related and non-safety-related structures are integrally connected, the non-safety-related structure is included in the model when determining the forces on safety-related structures. Such non-safety-related structures (as well as non-safety-related structures adjacent to safety-related structures) are designed so that their failure under SSE conditions will not cause the failure of the safety-related structures.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

To account for the expected variation in structural properties, dampings and other parameter variations, the peaks of floor response spectrum curves are broadened by ±15% and smoothed in accordance with Regulatory Guide 1.122. I

Soil property related spectrum peaks are further broadened, where required, to conservatively account for all potential variations of soil properties within the envelope of site conditions. Structures, systems and equipment are qualified to either 1) the envelope of the collective broadened spectra for all soil cases comprising the site envelope or 2) the broadened spectra for each of the soil cases which comprise the site envelope.

3.7.2.10 Use of Constant Vertical Static Factors

A constant seismic vertical load factor is not used for the seismic design of Seismic Category I structures, systems, components and equipment.

The safety-related structures, systems, and components are analyzed in the vertical direction using the methods described in Section 3.7.2.1. Based on the vertical seismic analysis, a vertical static factor is determined to design columns and shear walls. The vertical floor flexibilities are accounted for in the response spectra at each individual floor elevation of the building structures. The floor beams are designed statically for the acceleration value obtained per Reference 1.

CESSAR-DC Change

Section 3.7.2.8

Insert as second paragraph

The following procedure is used to ensure that the failure of a Non-Seismic Category I (non-safety-related) structure under the effect of a seismic event does not impair the integrity of an adjacent Seismic Category I (safety-related) structure.

a. Sufficient separation between Non-Seismic Category I structure and Seismic Category I structure is maintained, or

b. The Non-Seismic Category I structure is designed to withstand the effect of the postulated SSE event, i.e.,to maintain the structural integrity of the Non-Seismic Category I structure during and after the occurence of the postulated SSE event, or

c. The Non-Seismic Category I structure is designed such that if the Non-Seismic Category I structure collapses, it will fall away from any Seismic Category I structure.

Question 220.25

Section 3.7.3.12.1 - Specific items to be considered in the seismic design of buried piping, conduits and tunnels are provided in SRP Section 3.7.3.II.12. Provide the actual methods to be used to address these items at various sites.

Response 220.25

Refer to response to RAI 210.42.

Question 220.26

Page 3.7-34 - Provide document number and date of publication for some of the references listed. Reference 6 refers to a SASSI document dated April 1981. What is the date of the SASSI code used? Is it the latest version? If the SASSI code used is not the latest version, is there any validation performed against the latest version? In the latest version of SASSI code, no finite elements are needed in the soil layers except in the excavated volume of the foundation. Figure 3.7-31 shows that finite elements are used beneath the foundation.

Response 220.26

The version of the program SASSI that was used in the SSI analyses of the System 80+ was ABB Impell's version 2.0, dated March 1985. The following SASSI modules were part of the version 2.0 program:

° SITE
° COMBINE
° MOTION
° STRESS

To compute foundation impedances and scattering with an axisymmetric approach, SASSI was modified and enhanced by ABB Impell. Thus, two of the version 2.0 modules, HOUSE and ANALYS, were modified for the System 80+ project as version 3.0, and a new module, AXSYM, was developed as version 3.0. Figure 220.26-2 shows the sequential order of analysis using AXSYM to calculate impedances and scattering of an axisymmetric foundation.

The latest ABB Impell version of SASSI is version 4.0 June 1989. Version 4.0 has not been used in the System 80+ work. Version 4.0 was verified based on the verification plan of versions 2.0 and 3.0, and 100% compatibility of results exists between the versions in the solution of the same problem. Note that version 4.0 was not created to correct errors in Versions 2.0, but to provide program enhancements.

The axisymmetric approach to calculate impedances differs from the flexible volume method which is conventionally used with SASSI. Also, using the axisymmetric approach, the scattering matrices are computed explicitly. However, most of the other analysis features of SASSI are retained, i.e.:

° Frequency domain analysis with complex response.
° Substructuring method.
° Free-field representation by horizontal visco-elastic soil layers.

<u>Response 220.26</u>

- Solution of the site response problem.
- Three-dimensional modeling of structures.
- Evaluation of transfer functions at structure locations and computation of response in the time domain.

Using the axisymmetric approach, the near field soil is modeled with axisymmetric solid finite elements. The entire soil volume below the foundation is modeled using finite elements. At the bedrock elevation, fixed boundary conditions are applied, which is a conservative assumption. At the sides of the near field soil volume a transmitting boundary is established at each soil layer. The transmitting boundary is identical to the transmitting boundary used in the flexible volume method for the free field.

For rigid foundations, the impedances and scattering matrices are calculated about the center of the foundation mat. This is the attachment point of the superstructure model.

The axisymmetric approach was correlated to the flexible volume approach and was thoroughly verified as part of the SASSI verification plan for the System 80+ project. For axisymmetric foundations of arbitrary embedment, the axisymmetric approach and the flexible volume approach give identical results for impedances and scattering.

RAI 220.26

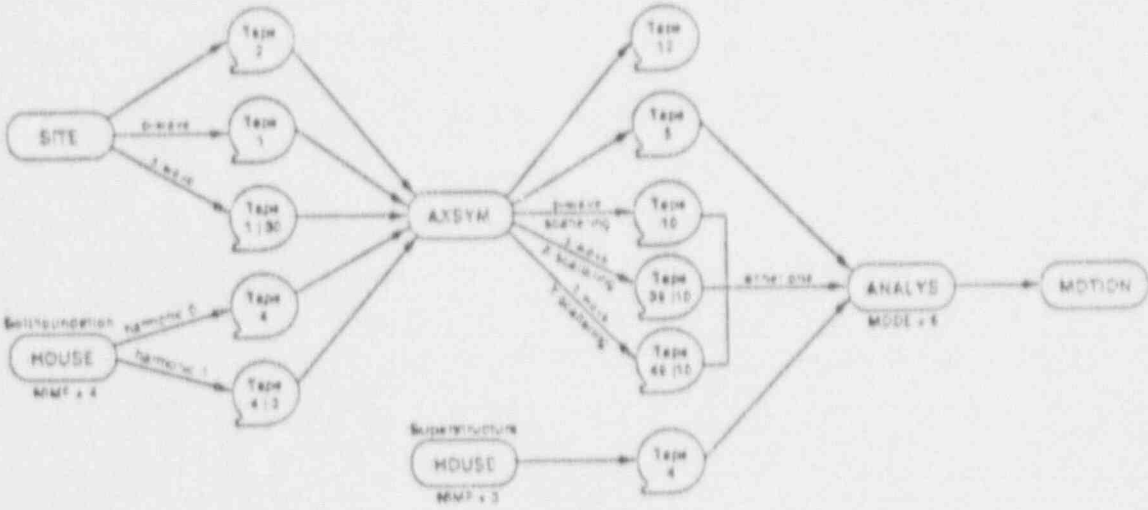


Figure 220.26-1 - Flow Diagram for SASSI Axisymmetric Solution

Question 220.27

Figure 3.7-28 shows the stick model of the internal structure for horizontal seismic analysis. Provide a figure of the stick model used for vertical seismic analysis. (see also Comment 220.24)

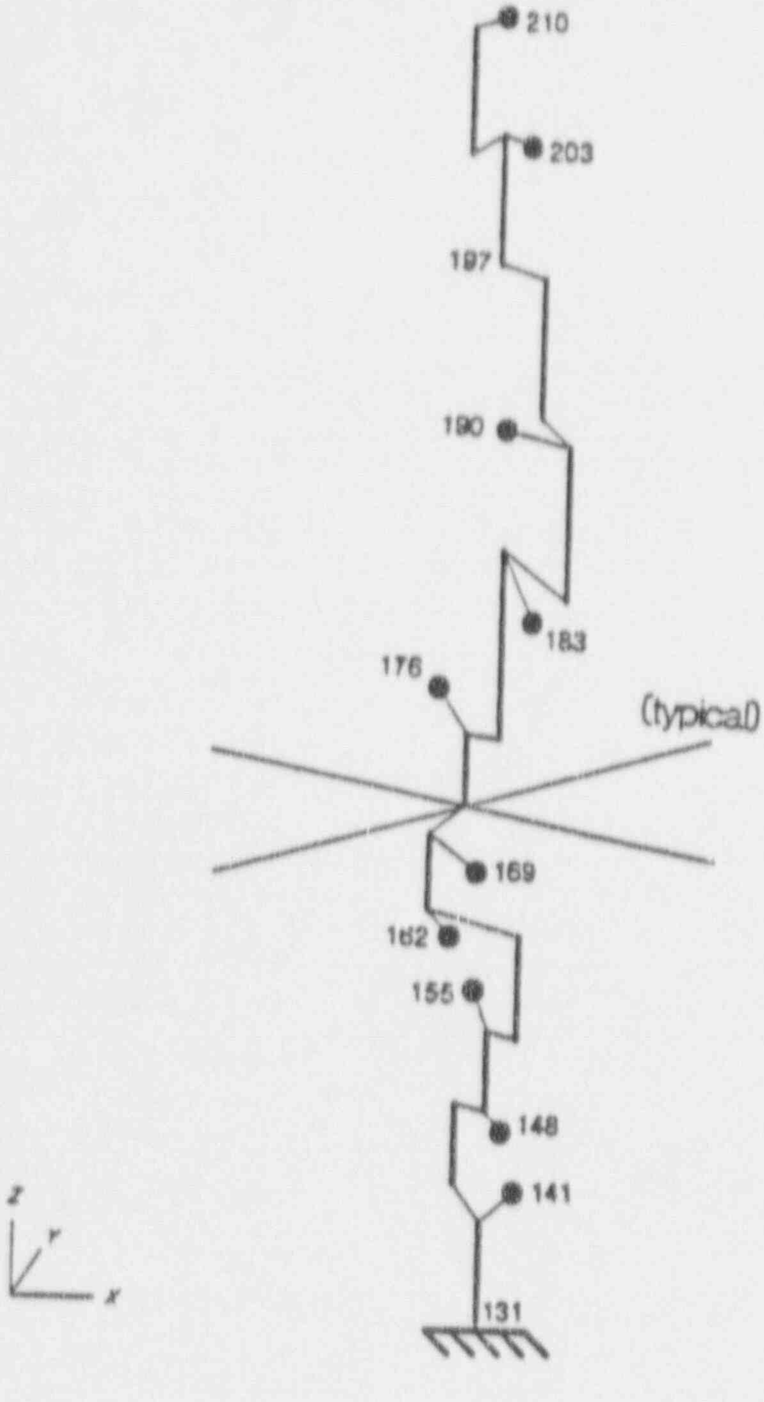
Response 220.27

The stick model shown in Figure 3.7-28 is typical of the horizontal and vertical model. The nodes shown represent the mass nodes which are at the same location in both models. The difference in the models is the x-y plane location of the sticks, which are at the center of rigidity. The sticks represent the stiffness between floor elevations. The horizontal model sticks are located at the shear center of the section. The vertical model sticks are located at the bending center of the section.

Figure 3.7-28 will be revised in a future amendment to CESSAR-DC to clarify this.

See also response to question 220.23. Note: The RAI references question 220.24. This appears to be in error.

RAI 220.27



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SYSTEM 80+™	STICK MODEL OF INTERNAL ~~STRUCTURE~~ ~~(FOR HORIZONTAL ANALYSIS)~~ DELETE	Figure 3.7-28

Question 220.28

Table 3.7A-2, Appendix 3.7A - Provide a discussion of how the design specification data were obtained.

Response 220.28

The design specification seismic loadings provided in Appendix 3.7A - Table 3.7A-2 are a compilation of responses based on past experience and conservative design analyses that reflect a wide range of site conditions and maximum ground accelerations. These design specification seismic loadings were developed and used in the System 80 design process to conservatively envelope expected seismic responses.

The design specification seismic loads identified in CESSAR-DC Table 3.7A-2 are to be used in the System 80+ design specifications. However, it should be noted that as a result of detail component design, design specification loads may require reduction, but in no case will specified for design loads be lower than maximum calculated values.

CESSAR-DC Table 3.7A-2 will be revised to clarify the intent of "specified for design" in a future amendment.

Question 220.29

Section 1.3, Appendix 3.7B - Justify the use of axisymmetric model for asymmetric structures and foundations surrounding the reactor building (Figure 1.2.1).

Response 220.29

The models of the annex structures were included in the Nuclear Island analysis in order to capture structure-to-structure interaction effects on the Nuclear Island seismic response. The Nuclear Annex structures were modeled with simplified 3D two node/single element lumped mass models founded on a common rigid basemat. Four such models were developed, one for each of the following structures: Auxiliary Building, Control Building, Fuel Building and Diesel Generator Building. The models were founded on an axisymmetric rigid basemat, with each model located at the center of the corresponding building. Since the Nuclear Island is much heavier than the adjacent structures, and structure-to-structure interaction is a secondary effect to the response of the Nuclear Island, modeling the foundation of the annex structures as axisymmetric and rigid is adequate for the purpose of generating floor response spectra for the Nuclear Island using SSI analysis.

Question 220.30

Section 1.3, Appendix 3.7B - What is the basis for considering the foundations as rigid?

Response 220.30

System 80+ was analyzed for a multitude of soil cases (soft soils, stiff soils and combinations of both), and therefore, any effects of the flexibility of the foundation to the response of the superstructure are included in the response of the superstructure for a softer soil case.

The Nuclear Island foundation has a minimum thickness of 10 ft. Furthermore, there are many 4 ft. vertical shear walls monolithically connected with the basemat which provide additional stiffness to the out of plane stiffness of the basemat. For the above reasons it is believed that modeling the Nuclear Island foundation as rigid is an appropriate representation for the purpose of generating floor response spectra. For evaluating basemat stresses or soil pressures, the flexibility of the basemat is explicitly included in the analysis.

Question 220.31

Section 1.4, Appendix 3.7B - Describe the dynamic analysis models of the structures adjacent to the reactor building. (see also Comment 220.30)

Response 220.31

As discussed in the response to RAI 220.29, simplified structural models for the adjacent structures were used, since the effects of structure-to-structure interaction are secondary. The Nuclear Annex structures are dynamically analyzed using refined stick models corresponding to the final configuration of each building for their individual response.

Question 220.32

Section 1.4, Appendix 3.7B - It is stated that "The input control motions were obtained from Reference 7 of Section 3.7." Describe the relationship of these input control motions with the time histories discussed in Section 3.7.1.2 and the unsmoothed spectra shown in Figures 3.7-1 through 3.7-24.

Response 220.32

The acceleration time histories that are provided as input control motions to the SASSI analyses are identical to the motions with unsmoothed spectra shown in Figures 3.7-1 to 3.7-24 of CESSAR-DC.

Question 220.33

Section 2.0, Appendix 3.7B - The second paragraph of this section states that "All analyses are three-dimensional with input excitation provided in three directions simultaneously." Can SASSI code used in analyzing CE System 80+ Standard Plant handle three directions of excitation simultaneously (i.e., applying one P-wave and two S-waves in one SASSI run)?

Response 220.33

In the SASSI code, the applications of all three ground control motions (two horizontal and one vertical) cannot be performed in one run. However, the procedure followed in the System 80+ Nuclear Island analysis involves three separate SASSI runs and algebraic addition of the response acceleration time histories, which, for linear elastic analysis, is equivalent to simultaneous application of the control motions in three directions. Refer to Response 220.18 for a description of the analysis procedure.

Question 220.34

Section 2.0, Appendix 3.7B - It is stated that the "cutoff frequencies were computed based on the dimensions of the soil discrimination." Describe how the soil layers were discredited to yield cutoff frequencies as low as 12 and 18 Hertz as shown in Table 3.7B-12.

Response 220.34

The discrimination of the soil media used in the SASSI SSI analyses is provided in Tables 3.7B-2 to 3.7B-10. These tables provide the thickness of each layer in each soil profile and its associated (strain-iterated) S-wave and P-wave velocities and S-wave and P-wave material damping ratios.

The thickness of the soil finite elements at a constant elevation is identical to the thickness of the free-field soil layer that corresponds to the same elevation, as shown in Figure 220.34-1.

As shown in tables 3.7B-2 to 3.7B-10, the selection of the thickness of the soil layers depend on the soil profile and the maximum frequency that can be transmitted through that soil profile. The selection of the cutoff frequencies transmitted through each profile was performed by examining the high-frequency content of motion in the free-field ground and foundation elevation response spectra. Cutoff frequencies were selected at the frequencies where the response spectra began to show no amplification, and essentially started to be "flat". Naturally, for stiff soil profiles, the cutoff frequency in the SSI analyses was 40 Hz, since this frequency could be transmitted adequately through the stiff soil layers, and since 40 Hz was the cutoff frequency for the rock outcrop response spectra. For soft and deep soil profiles lower frequencies were selected. As an example, for soil profiles C2 and C3, cutoff frequencies of 12 Hz and 18 Hz were selected, respectively, based on the ground and foundation spectra shown in Figures 3.7-12 to 3.7-15 for these two cases. It is observed that for soil case C2, there is essentially no amplification in horizontal motion beyond 12 Hz. Similarly, for soil case C3, there is no amplification in horizontal motion beyond 18 Hz.

It must also be noted that, for design purposes, the stiff soil profiles control the design loads at high frequencies. Therefore, it is justified to consider lower cutoff frequencies for the softer soil profiles.

Question 220.35

Section 2, Appendix 3.7B -

a) Give the details of benchmarking the SASSI code used in the SSI analysis.

b) Discuss the results of any analytical check made of the extent of reduction of ground motion from ground surface to the foundation level to satisfy the SRP requirement in this regard.

c) Compare the results of SSI analyses performed by using the SASSI code and a simplified analytical model of the containment structure.

d) Is the basemat flexibility modeled in the SSI analysis?

e) In the substructuring technique used in the SASSI code (which is based on the "flexible volume" method), the "scattering problem" need not be solved, as in the case of other substructuring methods of SSI analysis. The 3 steps involved in the flexible volume method are: site response, impedance analysis, and structural response analysis. It is stated in Appendix 3.7B, Paragraph 1.1, of CESSAR-DC, that the second step of the SASSI formulation is the solution of the impedance and scattering problem. Explain why the scattering matrix is calculated separately as stated in Paragraph 1.3 of Appendix 3.7B, and if the SASSI version used in System 80+ design is based on a method different from the "flexible volume" method.

Response 220.35

a. The version of the SASSI program used in the System 80+ SSI analysis is extensively verified and valudated using three different methods of verfication and correlation:

- Correlation to results of problems with closed form solutions and published results, such as site response and response of simplified structural systems.

- Correlation to solutions of other well known SSI and soil analysis computer codes in the industry such as CLASSI and SHAKE.

- Correlation to experimental results, such as the Lotung Large Scale Experiment sponsored by the Electric Power Research Institute/Nuclear Regulatory Commission/Taiwan Power Company, and foundation vibration tests.

Response 220.35 (continued)

In all of the above benchmark problems, very good agreements were obtained between the SASSI solutions and the benchmark results. Furthermore, the SASSI methodology proved to be the best methodology in matching the Lotung results.

The complete verification and validation package for SASSI is available for audit by the NRC.

b. A check of the reduction of the free-field ground motion from the surface to the foundation level was made by computing the averages of the motions of all soil cases and subsequently dividing the average of the motions at the foundation level by the average of the surface motions. The computations were performed using as a basis the 5% damped spectra of the two horizontal motions, H1 and H2.

Figures 220.35.1 to 220.35.8 show the computed average motions and the ratios of foundation level to surface. As shown in Figure 220.35.4, the foundation motions corresponding to horizontal motion H1 are higher than 60% of the surface motions at all frequency ranges. Similar results are observed in Figure 220.35.8 for motion H2, except for the very low frequency range (frequencies <.03 Hz), which is of no structural significance, and the 1.5 Hz range, where the computed ratio is slightly less than 60%. At all other frequencies, the spectral ratios are between 0.6 and 0.9, conservatively satisfying the SRP requirements.

Therefore, based on this analytical check, the ground motions satisfy the SRP criterion for reduction in motion with depth.

c. A simplified three-dimensional model of the Nuclear Island is analyzed for one representative soil case and the results are compared to the results of the SASSI SSI analysis. Case C1.5 is selected because it represents a typical soil profile with low-strain shear wave velocity of 1000 ft/sec at the ground surface, which is gradually increasing with depth. Depth to bedrock is 200 ft.

For the purpose of this analysis, the simplified structural model of the Nuclear Island consists of the stick model of the IS coupled with simplified stick models of the SB and the SCV. The stick model corresponding to the horizontal SASSI analysis is utilized in the current study. For simplicity, the stick model of the IS is modified by lumping the NSSS mass at appropriate elevations. The stiffness of the NSSS model is not included in the simplified model. Each of the simplified sticks of the SB and the SCV

Response 220.35 (continued)

consists of a single beam element with a lumped mass. The fundamental frequency of the simplified stick models of the SB and the SCV are tuned to mat ı the fundamental frequency of the SB and the SCV, respectively.

The soil is modeled with frequency-independent soil springs and dampeners which are coupled with the structural stick model of the IS at its base (elevation +50 ft.). Soil springs are developed for the six rigid body degrees-of-freedom of the foundation (three translational and three rotatio al). The guidelines of Reference 220.35.1 are used to develop the properties of the soil springs and the dampers. Also, adjustment of the soil spring stiffness is performed, according to the guidelines of NRUEG/CR-1780, to account for the finite depth to bedrock (200 ft.).

To compute the soil spring stiffness, the foundation radius is taken as 110 ft. and the average strain-compatible shear modulus for the soil as 5000 ksf. The soil spring stiffnesses are listed below:

Horizontal Translation:		3.807E3 k/ft
Vertical Translation	:	7.140E3 k/ft
Rocking Rotation	:	3.330E10 k-ft/rad
Torsional Rotation	:	3.550E10 k-ft/rad

The lowest damping ratio obtained from the dampers is 31% and is associated with the horizontal translation. A uniform damping ratio of 31% is specified for all modes.

The free-field motion at foundation level of soil profile C1.5 is used as the control motion. The three acceleration time histories of profile C1.5 (two horizontal and the vertical) are applied simultaneously to the SSI soil-springs model. The time history analysis is performed by modal superposition. Response spectra at 5% damping are computed at the top of the IS for the horizontal (0°-180°) and vertical directions. Figures 220.35-9 to 220.35-11 show the correlation of the results using the two analytical procedures, SASSI and soil springs. The correlation of the results are shown for two different elevations of the Internal Structure and the basemat.

In the horizontal direction, the response from the SASSI model is, in general, higher than the soil springs model. This is due to the high damping ratio obtained by using the simplified soil spring approach. SASSI computes the radiation damping characteristics of the soil/structure system more realistically than the

RAI 220.35

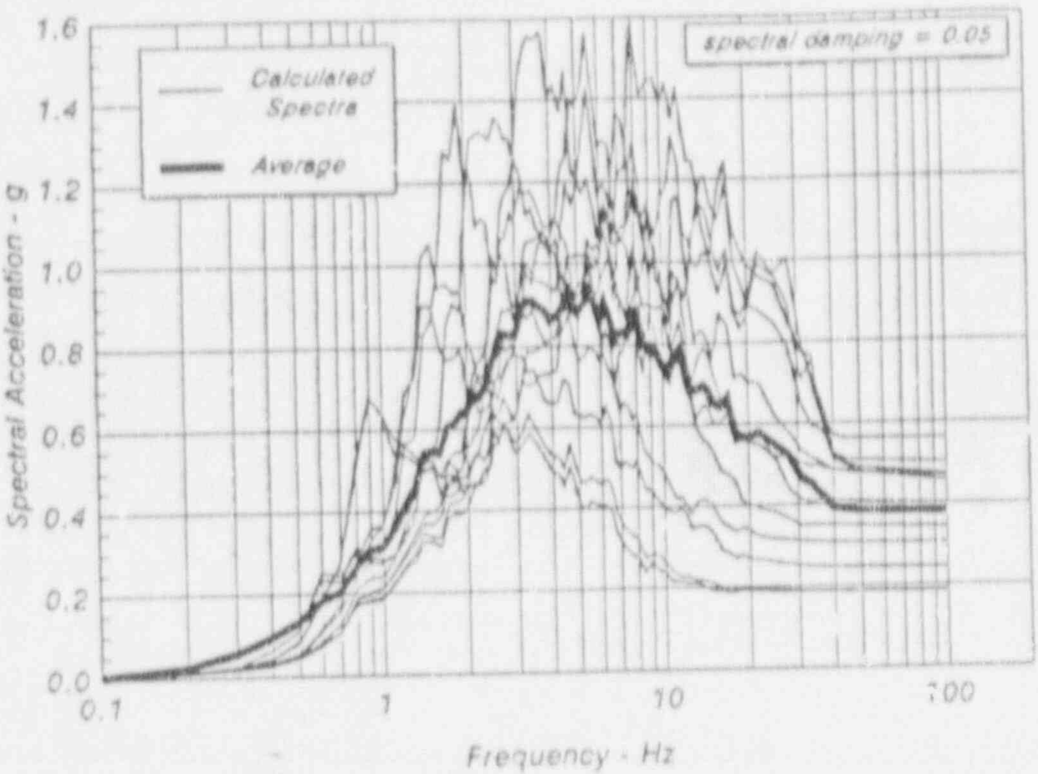


Figure 220.35.1 - Range and Average Spectral Ordinates Calculated at the Ground Surface Using Synthetic Time History H1

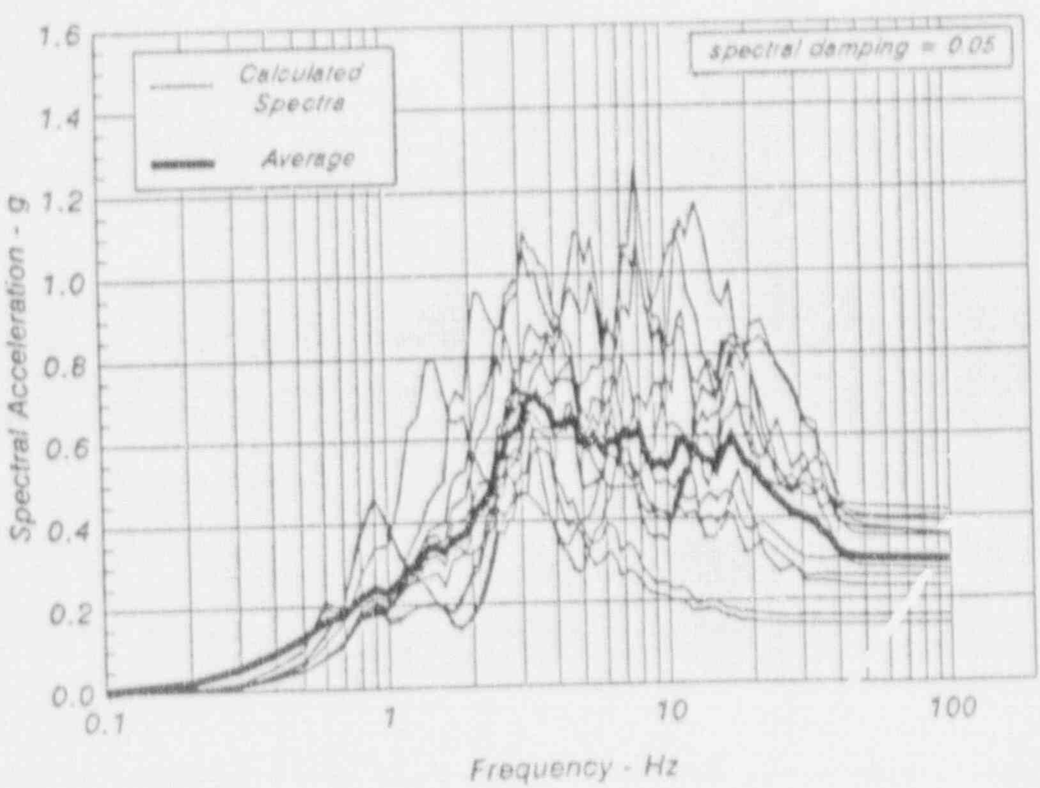


Figure 220.35.2 - Range and Average Spectral Ordinates Calculated at the Foundation Level Using Synthetic Time History H1

RAI 220.35

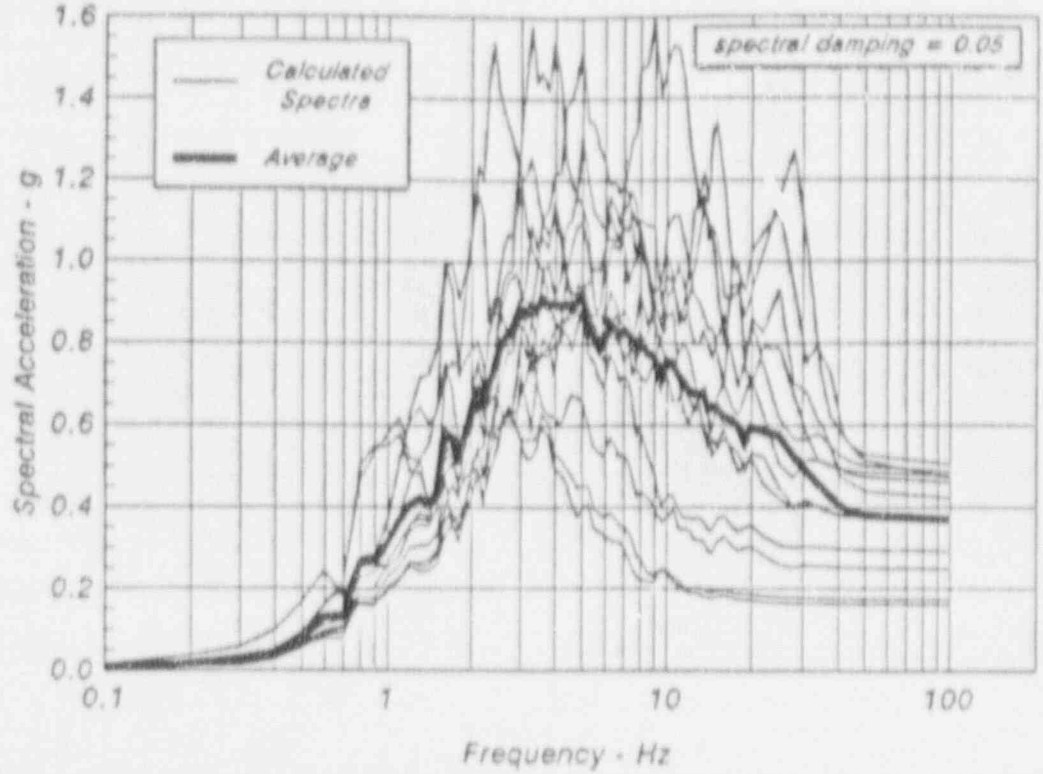


Figure 220.35.5 - Range and Average Spectral Ordinates Calculated at the Ground Surface Using Synthetic Time History H2

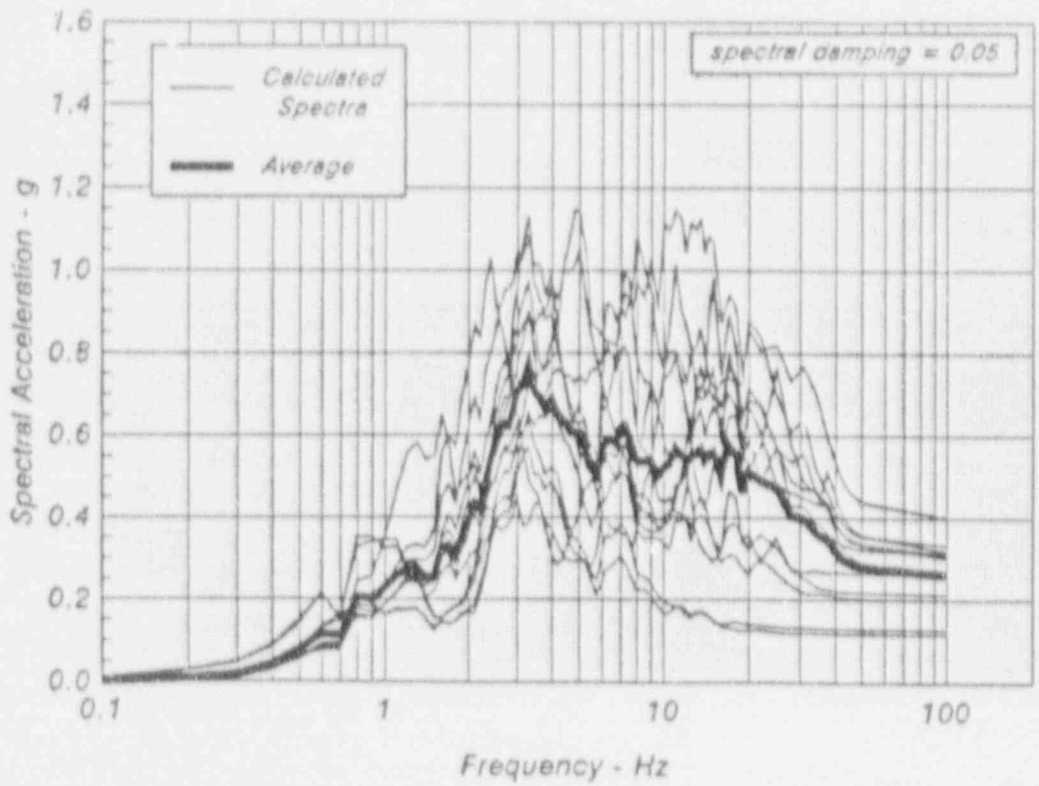


Figure 220.35.6 - Range and Average Spectral Ordinates Calculated at the Foundation Level Using Synthetic Time History H2

RAI 220.35

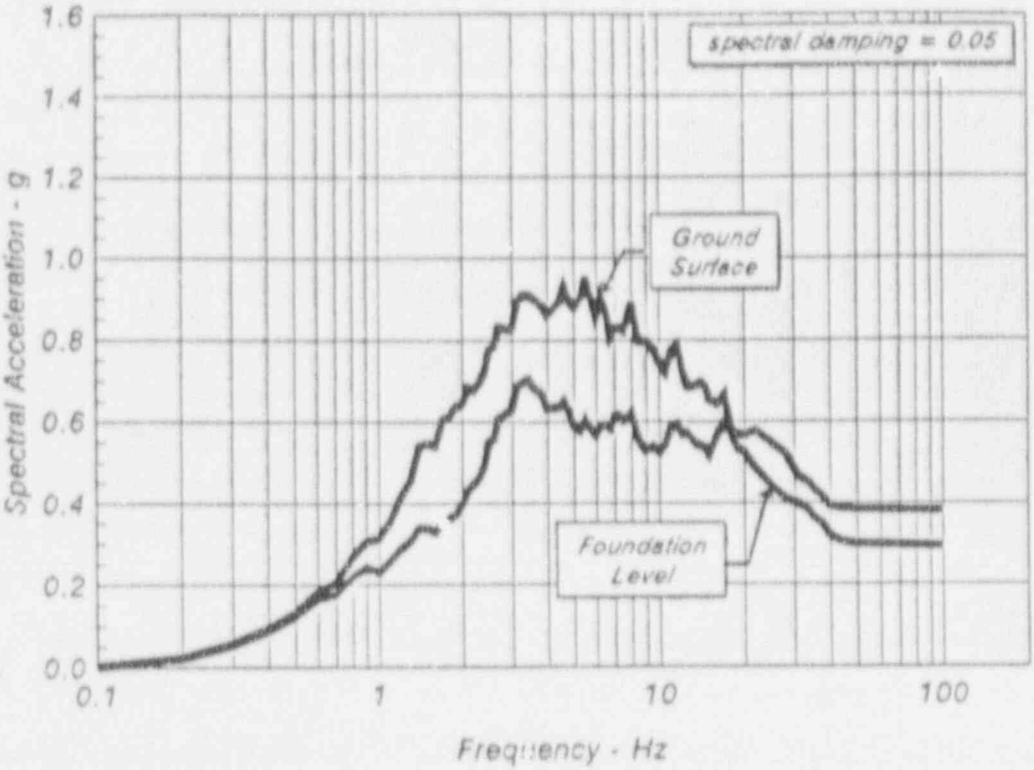


Figure 220.35.3 - Average Spectral Ordinates Calculated at the Ground Surface and at the Foundation Level Using Synthetic Time History H1

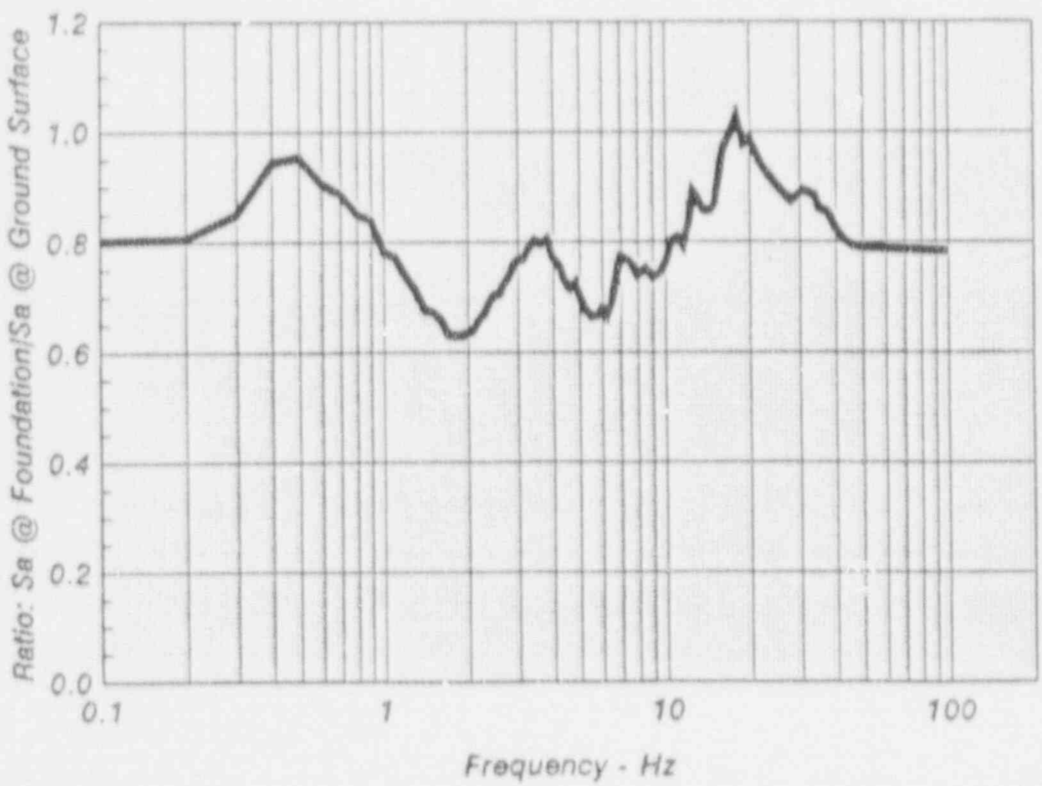


Figure 220.35.4 - Ratio of Spectral Ordinates Calculated at the Foundation Level Divided by those Calculated at the Ground Surface Using Synthetic Time History H1

RAI 220.35

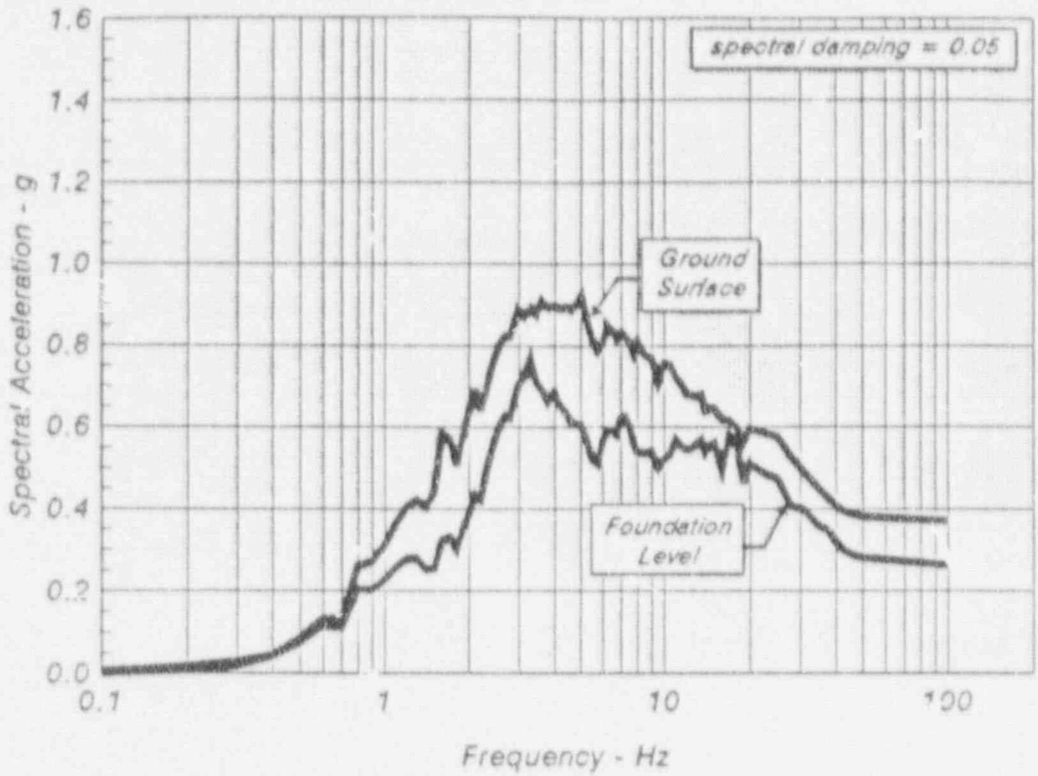


Figure 220.35.7 - Average Spectral Ordinates Calculated at the Ground Surface and at the Foundation Level Using Synthetic Time History H2

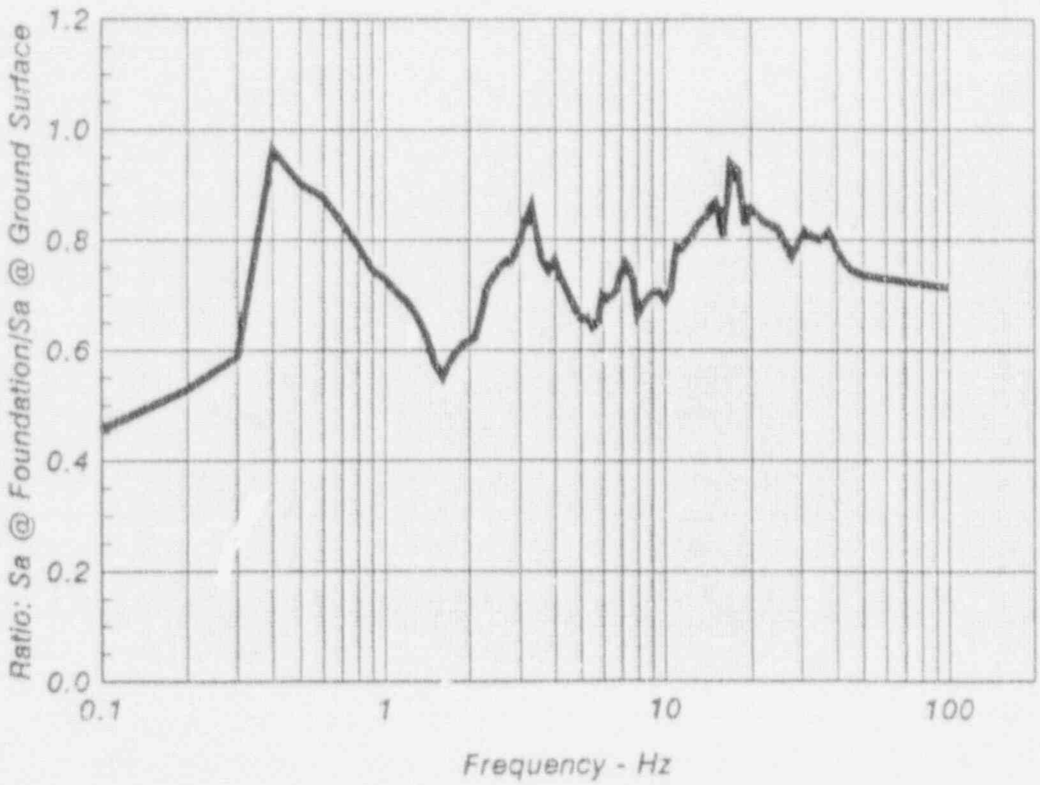


Figure 220.35.8 - Ratio of Spectral Ordinates Calculated at the Foundation Level Divided by those Calculated at the Ground Surface Using Synthetic Time History H2

Response 220.35 (continued)

simplified soil springs approach. Also, the SASSI methodology captures the effects of high frequency response, which the soil springs approach is unable to capture because of the high mass participation in the first modes of vibration of the soil/structure system. Furthermore, the response of the SASSI model is affected by the content of the free-field surface motion in the 3-4 Hz range, which the foundation level motion does nto contain.

The fundamental horizontal frequency of the soil springs model is at 2.45 Hz (in the 0-180 direction), which is near the frequency of the peak of the SASSI transfer function at the top of the Internal Structure (2.88 Hz).

In the veritcal direction, the response spectra of the two models are very similar.

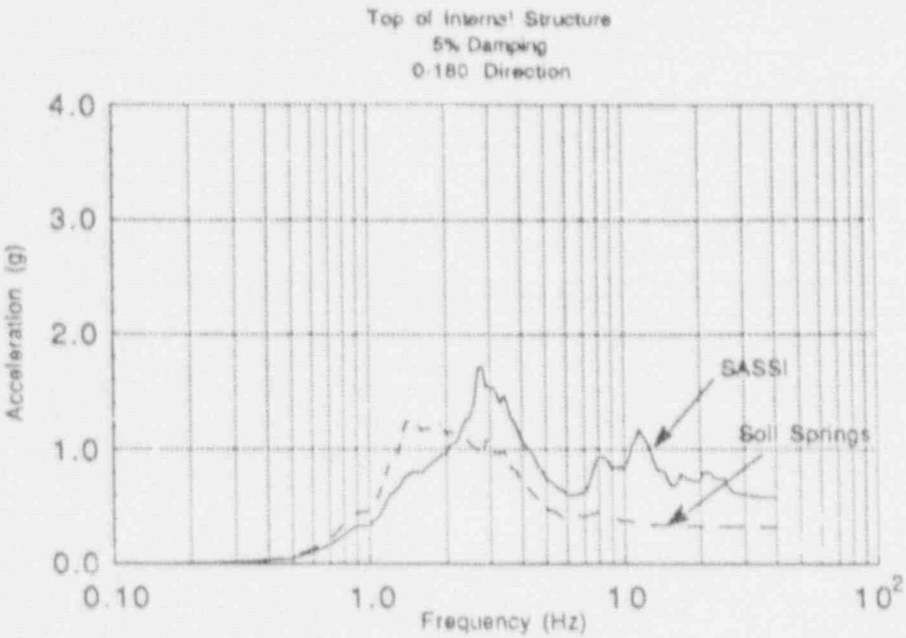
References:

220.35.1 American Society of Civil Engineers, "Seismic Analysis of Safety Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety Related Nuclear Structures", Publication No. ASCE 4-86, September 1986.

d. Refer to the response of RAI 220.30.

e. Refer to the response RAI 220.26.

RAI 220.35



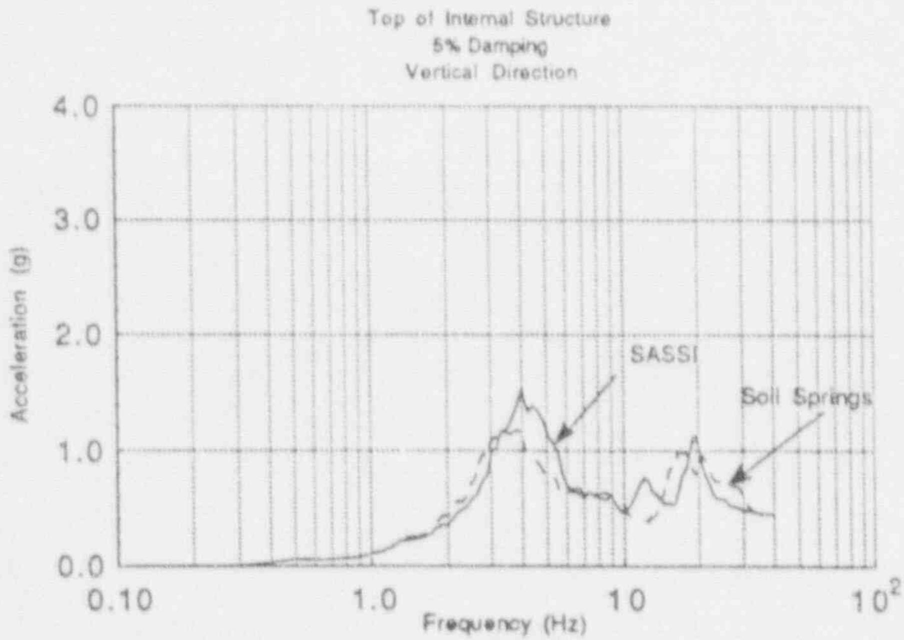


Figure 220.35-9 - Correlation of SASSI vs. Soil Springs (Top of Internal Structure)

RAI 220.35

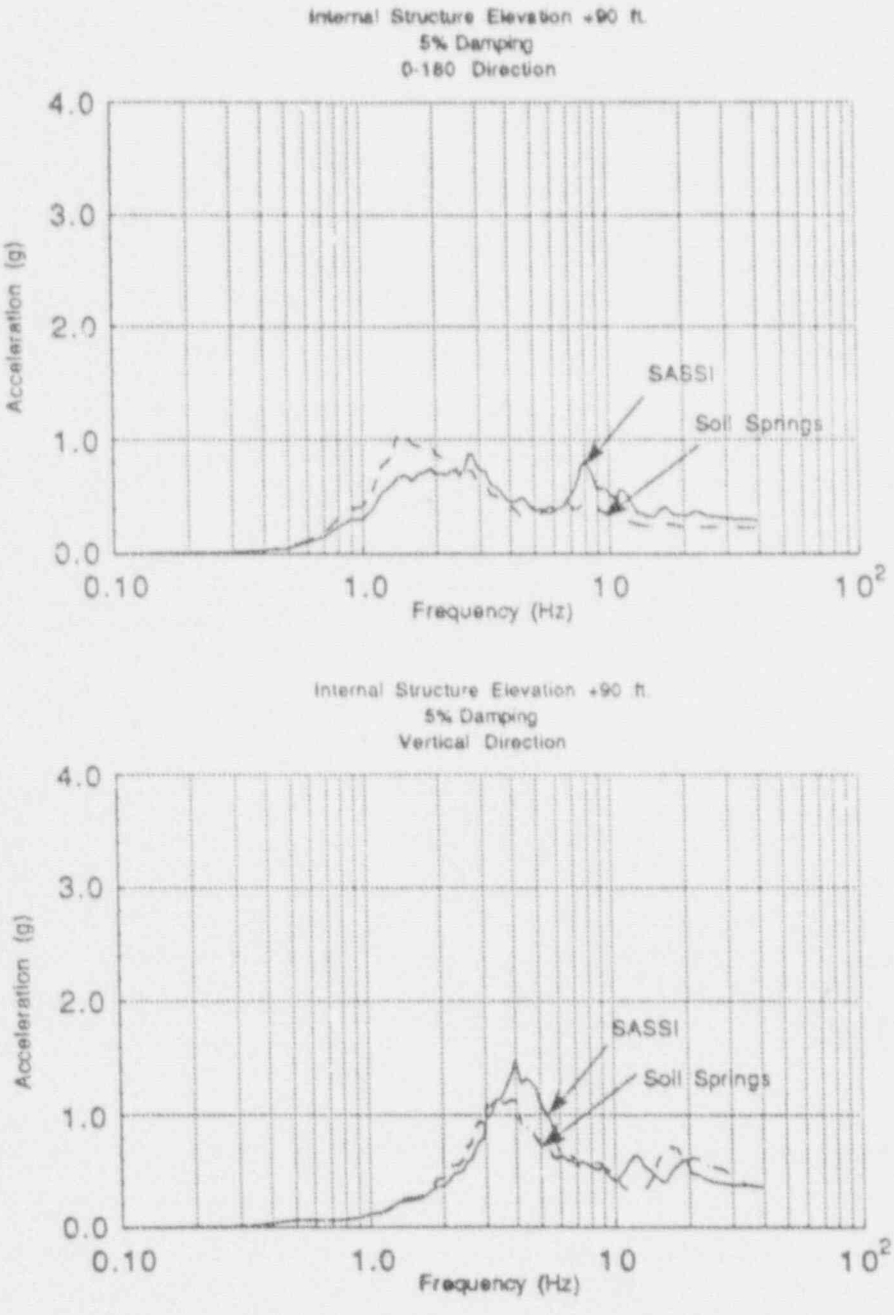


Figure 220.35-10 - Correlation of SASSI vs. Soil Springs (Internal Structure, Elevation +90 ft.)

RAI 220.35

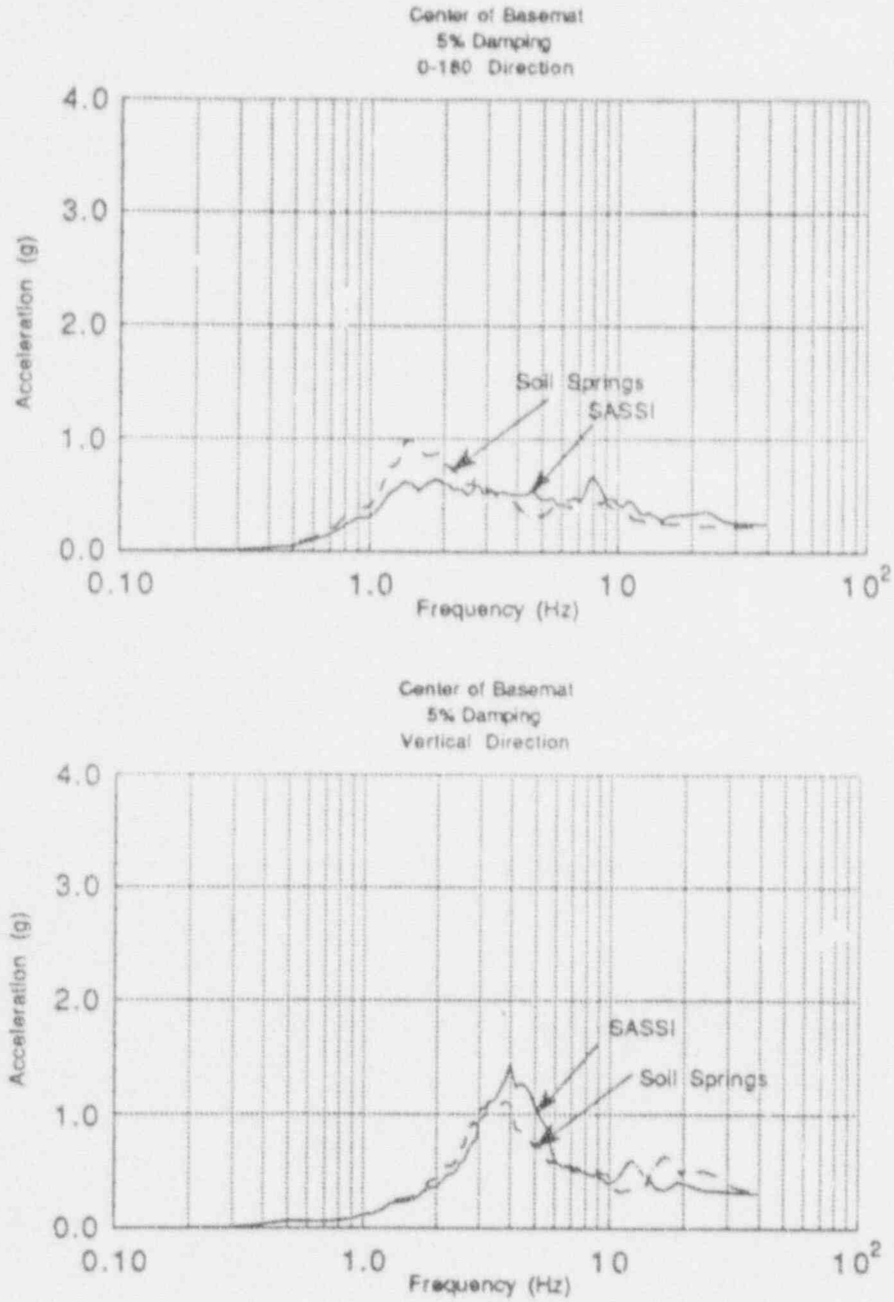


Figure 220.35-11 - Correlation of SASSI vs. Soil Springs (Basemat)

Question 220.36

Section 2, Appendix 3.7B - Discuss if and how the effects of the following phenomena on the structural response are considered in the various cases of the SSI analysis:

a) Basemat uplift (including the impact of time phasing of ground motion on uplift effects).

b) Ground motion incoherence.

c) Structure-to-structure interaction.

Response 220.36

a) To evaluate whether uplift of the basement occurs during the SSE event, the results from the linear SSI response of soil case B3.5 with the common basemat configuration were utilized. Soil case B3.5 produces the worst case vertical spectra, therefore, conclusions based on this study envelop all the soil cases examined. The selection of the common basemat results is justified, since the final configuration of the Nuclear Island is founded on a common basemat.

The loads considered in the uplift analysis are:

° Dead load of Nuclear Island and Annex Structures, (D)
° Bouyant force due to embedment, (B)
° Earthquake loads, (E)

The load combination that determines whether uplift occurs is: E + B - D.

The dead loads and the bouyant force correspond to the loads and geometry of the final design of the System 80+ Nuclear Island and Annex Structures. The earthquake accelerations correspond to the results of soil case B3.5 with common basemat, which is documented in CESSAR-DC, Appendix 3.7B.

The net acceleration at the edge of the foundation (outermost point) in the 0°-180° direction was checked for uplift. To include time phasing effects, the vertical response acceleration due to rocking (about the 90°-270° axis) and the vertical response acceleration due to the vertical motion were added algebraically in the time domain. The maximum vertical acceleration at the edge of the foundation was calculated as 0.360 g (upward). The contribution of the rocking component in the net upward vertical acceleration is approximately 5%. The equivalent net acceleration due to dead load and bouyant force was calculated as 0.557 g (downward). Therefore, no uplift occurs.

Response 220.36 Cont.

b. Results of research studies that have been conducted in the past with recorded data have shown that, in general, due to the lack of coherence of free-field ground motions over horizontal distances, it appears that due to kinematic interaction a large foundation would experience average translation motions that are reduced from the free-field motions (Reference 220.36.1). This phenomenon, defined as "base-averaging" effect, is more pronounced in the high frequency spectral range.

   Similar conclusions are noted by research studies based on numerical determinations of both random and deterministic effects of incoherence on foundation response (Reference 220.36.2). These studies indicate that the spatial variation of the ground motion produces a reduction in translational components of the foundation response at high frequencies and creation of rocking and torsional components.

   In the System 80+ seismic SSI analyses, no reduciton of the translation components of motion was performed to account for the effects of ground motion incoherence, which adds to the conservatism of the System 80+ analysis. In addition, the multitude of soil cases that were considered in the System 80+ SSI analyses of the Nuclear Island covers a broad range of rocking and torsional response, thus, conservatively accounting for any creation of rocking and torsional components in the motion due to ground motion incoherance.

   Reference 220.36.1: NUREG/CR-3805, "Engineering Characterization of Ground Motion", August 1986.

   Reference 220.36.2: Luco, J. E., and Wong, H. L., "Response of a Rigid Foundation to a Spatially Random Ground Motion", Earthquake Engineering and Structural Dynamics, Vol. 14. 1986.

c. The interaction of the Internal Structure, the Steel Containment Vessel and the Shield Building with the adjacent Nuclear Annex structures is included in the SSI analysis. Refer to RAI 220.29 for a description of the models of the Nuclear Annex structures.

Question 220.37

Section 3.0, Appendix 3.7B - There is no Reference 25 in Section 3.7. Provide the document referenced as Reference 25.

Response 220.37

The correct reference is #9. This correction will be included in a future amendment to CESSAR-DC.

RAI 220.37

constitute an upper bound for the majority of the frequencies, the application of these factors results in adequate and slightly conservative OBE seismic loads for design purposes.

Structure	Direction	Factors
PGC Foundation	X, Y, Z	0.4 (all frequencies)
IS (all elevations)	X, Y, Z	0.45 (all frequencies)
SB (all elevations)	X, Y, Z	0.45 (all frequencies)
SCV (all elevations)	X, Y	0.40 for frequencies $\leq$ 5 Hz 0.45 for frequencies $\geq$ 5 Hz
SCV (all elevations)	Z	0.40 for frequencies $\leq$ 10 Hz 0.65 for frequencies $\geq$ 10 Hz

3.0 SSI ANALYSIS COMMON BASEMAT CASE

To evaluate the impact of the modification from a dual foundation to a common basemat for all PGC structures, one critical SSI case is reanalyzed with a common basemat foundation. I

Case B3.5 is selected for the "common basemat" analysis because, when the RB is coupled with the B3.5 soil profile, it is subjected to high accelerations which result in critical spectral peaks. This response is observed in the "dual foundation" results.

To reanalyze the B3.5 case, the PGC model is modified as shown in Figure 3.7B-25. The foundation is modeled as a continuous rigid basemat with rigid sidewalls which are in direct contact with the side soil. The adjacent-to-the-RB structures are connected to the center of the common basemat with rigid links.

9

The results of the "common basemat" analysis are documented in detail in Reference 25 of Section 3.7. A comparison of response spectra at selected locations is shown in Figures 3.7B-26 to 3.7B-33, as follows:

Figure	Building	Location	Direction
3.7B-26	RB	Fdtn (Node 131)	X (0-180)
3.7B-27	RB	Fdtn (Node 131)	Z (vertical)
3.7B-28	IS	Top (Node 210)	X (0-180)
3.7B-29	IS	Top (Node 210)	Z (vertical)
3.7B-30	SB	Top (Node 125)	X (0-180)
3.7B-31	SB	Top (Node 125)	Z (vertical)
3.7B-32	SCV	Top (Node 61)	X (0-180)
3.7B-33	SCV	Top (Node 61)	Z (vertical)

REFERENCES FOR SECTION 3.7

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2. Fritz, R.J., "The Effect of Liquids on the Dynamic Motions of Immersed Solids", Journal of Engineering for Industry, Paper No. 71-VIB-100.

3. McDonald, C.K., "Seismic Analysis of Vertical PVMPs Enclosed in Liquid Filled Containers", ASME Paper No. 75-PVP-56.

4. Pahl, P.J., "Modal Response on Containment Structures", Seismic Design for Nuclear Power Plants, MIT Press, Cambridge, Mass.

5. Forsberg, K., "Axisymmetrical and Beam-Type Vibrations of These Cylindrical Shells", AJAA Journal, Volume 7, February 1969.

6. Lysmer, J., Tabatabaie, M., Tajirian, F., Vahdani, S., Ostadan, F., "SASSI - A System for the Analysis of Soil-Structure Interaction", Report No. UCB/GT/81-02, Univ. of California, Berkeley, April, 1981.

7. Idriss, I.M., "Earthquake Ground Motions - Selection of Control Motion and Development of Generic Soil Sites".

8. ABB Impell Report No. 01-8503-1784, "Seismic Analysis of the Reactor Building of the System 80+ Certified Design".

9. Impell Corporation, Calculation No. ALWR-2, "SSI Analysis of Case B3.5 with Common Basemat", Job No. 8503-003-1355, Revision 6.

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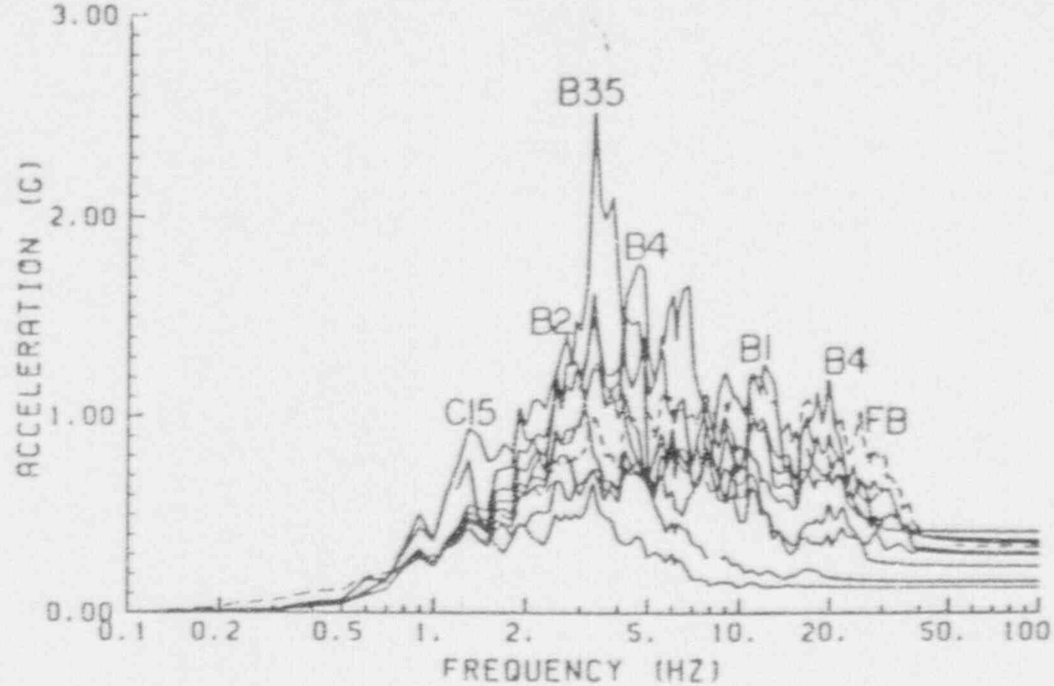
Question 220.38

Figure 3.7B-5 has the same title as Figure 3.7B-6. Provide correct titles.

Response 220.38

Figure 3.7B-5 should read "(..,0°-180° direction)", not "(.., vertical direction)". The Figure 3.7B-6 title is correct.

The title to Figure 3.7B-5 will be corrected in a future amendment to CESSAR-DC.



SYSTEM 80+™

COMPARISON OF 2% H+V RESPONSE SPECTRA
(SSE, RB FDTN, NODE 131, ~~VERTICAL~~ DIRECTION)

0-180

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Figure
3.7B-5

RAI 220.38

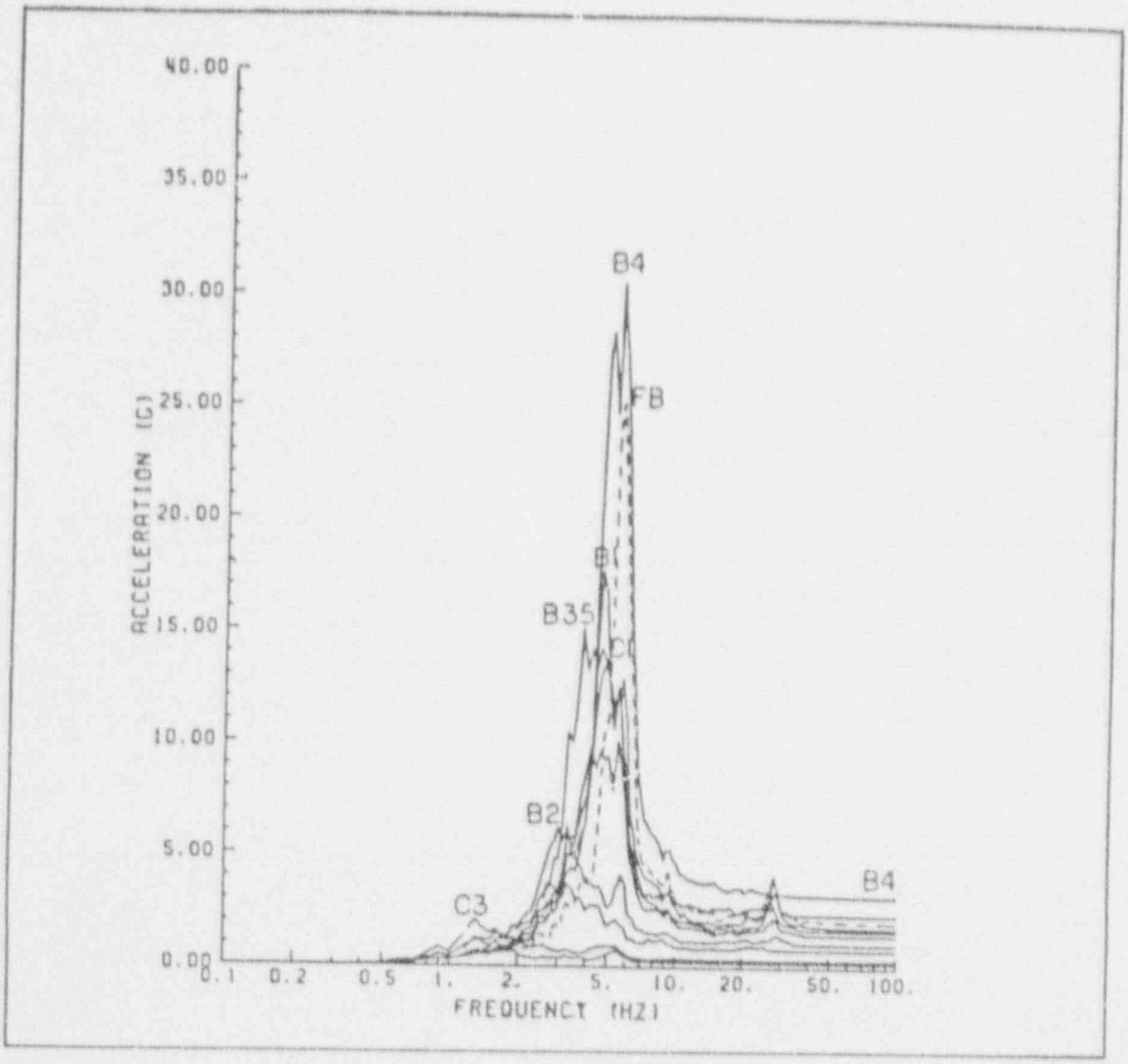
Question 220.39

The Figure 3.7B-11 title should read "(.., 0-180 direction)", not "(.., vertical direction)". The title for Figure 3.7B-12 is correct.

Response 220.39

The Figure 3.7B-11 title should read "(..., 0°-180° Direction)", not "(..., Vertical Direction)". The title for Figure 3.7B-12 is correct. The title to Figure 3.7B-11 will be corrected in a future amendment to CESSAR-DC.

RAI 220.39



0-180

Amendment I
December 21, 1990

SYSTEM 80+™	COMPARISON OF 2% H+V RESPONSE SPECTRA (SSE, SCV, NODE 61, VERTICAL DIRECTION)	Figure 3.7B-11

The <u>Question 220.40</u>

Section 3.7.4 - The staff is in the process of revising RG 1.12. The objective of this revision is to have new plants equipped with state-of-the-art solid state digital seismic instrumentation. The new guidance is scheduled to be published for public comments by January 1992. Provide a discussion of such proposed changes and how such changes will be implemented for CE 80+ plants.

<u>Response 220.40</u>

Combustion Engineering will comment on the proposed revisions to USNRC Regulatory Guide 1.12, "Instrumentation for Earthquakes" when they are published for public comment and will assess the need for revision to CESSAR-DC at that time. C-E recognizes the requirement to rapidly assess any earthquake motion at a plant site to determine if the design operating basis earthquake has been exceeded, thus requiring plant shutdown and inspection. Seismic instrumentation will be installed at System 80+ plants which will provide this rapid assessment capability and which will be based on the state-of-the-art knowledge and experience.

Question 220.41

Section 3.8.2.1.2 - It is stated that "No shear connectors are provided between the containment plate and shield building foundation or base slab of internal structures." Is there any study performed in the seismic PRA analysis (Section 4.3, Appendix B) to determine if the internal structure and the steel containment will be stable for all levels of ground motion including earthquake intensities much more severe than the SSE? If there is a relative movement between the concrete and the steel plate in this region, how is this relative movement considered in this study and in the determination of component and structural fragilities?

Response 220.41

For design basis loading cases, including SSE and OBE, calculations were performed to investigate the possibility of relative movement between 1) the steel containment plate and the lower dish concrete structure (foundation) and2) the internal structure concrete and the steel containment plate. From these calculations it was determined that no relative movement occurs in any of the cases and that the factors of safety against sliding are satisfied.

There has not been any corresponding study performed for earthquake intensities much more severe than the SSE in the seismic PRA analysis. Earthquake levels at which instability might occur, and determination of component and structural fragilities, are site specific and not available at time of design certification.

Question 220.45

Section 3.8.2.4 - It is stated that "These methods are described in Article NE-3222 of the ASME Code and ASME Code Case N-284." Clarify this statement because Article NE-3222 of the ASME Code (1989) and Code Case N-284 do not provide descriptions of a three-dimensional finite element bifurcation analysis. Clarify also the statement that "Code Case acceptability is in concurrence with RG 1.84." Justify the use of Code Case N-284 for the CE System 80+ spherical containment shell, which is asymmetric with large openings.

Response 220.45

Article NE-3222.1(a)(2) states that "classical (linear) analysis reduced by margins which reflect the difference between theoretical and actual load capacities" is one acceptable method for determining the maximum buckling stress values to be used for the evaluation of instability. Article-1300 of Code Case N-284, "Stress Analysis Procedures", states that "the shell anlaysis may be performed by the axisymmetric shell of revolution method of 1310 or by alternate methods. The more elaborate, three-dimensional thin shell analysis method of -1320 may be used, if the vessel geometry and/or the magnitude of any attached masses are such that axisymmetric shell of revolution analysis is not appropriate". Thus, a three-dimensional bifurcation analysis is permitted by the Code and Code Case. For a shell whose geometry has large openings such as the System 80+™ containment, the three-dimensional analytical approach is definitely more appropraite.

Regulatory Guide 1.84 states that Code Case N-284 is acceptable for use "subject to the following condition in addition to those conditions specified in the Code Case: Prior to implementation of the Code Case, the applicant must demonstrate to the satisfaction of the NRC Staff (via Safety Analysis Report) that any axisymmetric techniques that are proposed will be applicable to a vessel having large asymmetric openings and that the overall margin used to prevent shell buckling is acceptable." The analysis of the System 80+™ containment, vessel uses three-dimensional analysis techniques consistent with the Code Case so justification of axisymmetric techniques is not necessary.

Adequate margin is provided by satisfying the requirements of NE-3222. The Code Case is used for its description of

Response 220.45 (cont.)

acceptable analytical techniques and its definition of an acceptable capacity reduction factor for external pressure loads. Within the strict limitations of the Regulatory Guide (and the Code Case itself) the Code Case is acceptable for use in the analysis of the System 80+™ containment.

Question 220.46

Section 3.8.2.4 - Provide justification for neglecting large openings in calculating the maximum pressure capacity of the containment vessel using axisymmetric finite element model.

Response 220.46

The analysis performed to calculate the maximum pressure capacity of the steel containment vessel uses nonlinear elastic-plastic solution techniques. Analyses performed by Duke Power Company on Catawba Nuclear Station for this loading condition considered the area around the equipment hatch and upper airlock in a separate three-dimensional model. The results of these analyses indicated that with the openings reinforced as required by Section III, Subsection NE of the ASME Code the ultimate pressure capacity is the same as that predicted by axisymmetric analysis.

Question 220.47

Section 3.8.2.4 - It is stated that "The stresses in the containment vessel due to combustible gas loadings... this model is similar to those used for the seismic analysis and buckling evaluation." What are the differences between the models used to calculate the stresses in the containment vessel due to combustible gas loadings and those for buckling evaluation and seismic analysis?

Response 220.47

The same three-dimensional model is used for the combustible gas analysis as is used for the seismic analysis and buckling evaluation.

Section 3.8.2.4 will be revised to clarify this information.

as a three dimensional thin shell using the finite element method of analysis. The stresses and deflections produced in the shell under the applied loads are calculated with the ANSYS computer program (Reference 2). The ANSYS mathematical model used to represent the containment vessel is shown in Figure 3.8-3.

Seismic stresses and deflections are calculated using the response spectrum method. The frequencies of vibration and corresponding mode shapes are determined using the normal mode method. Modal responses are combined as described in Regulatory Guide 1.92 (Reference 15). The appropriate damping level for the applied response spectra is defined in Regulatory Guide 1.61 (Reference 14).

C. Buckling

The critical buckling stresses in the containment vessel are determined by applying the appropriate safety factors and capacity reduction factors to the results of a three-dimensional linear bifurcation analysis using an ANSYS finite element model similar to that constructed for the static and dynamic analyses. These methods are described in Article NE-3222 of the ASME Code and ASME Code Case N-284 (Reference 5). Code Case acceptability is in concurrence with Regulatory Guide 1.84 (Reference 18). I

D. Ultimate Capacity

The maximum pressure capacity of the containment vessel is evaluated by a large displacement elastic-plastic nonlinear analysis. The vessel is modeled with axisymmetric shell finite elements using the ANSYS computer program. The ANSYS model is shown in Figure 3.8-4.

The stresses in the containment vessel due to combustible gas loadings are calculated using a static linear elastic analysis. The vessel is represented by a three-dimensional thin shell finite element model with the ANSYS computer program. This model is ~~similar to~~ the same as those used for the seismic analysis and buckling evaluation.

E. Nonaxisymmetric and Localized Loads

The containment is not divided into compartments (see Section 6.2.1.2) so there are no nonaxisymmetric loads applied to the containment vessel during a Design Basis Accident.

Question 220.48

Section 3.8.2.5 - It is implied here that the safety factors for buckling for various levels of loading combinations are in accordance with NE-3222 of the ASME Code. NE-3222 provides a number of options for determining the buckling stress values. Provide information on what procedure is used for the SCV of the CE System 80+ design.

Response 220.48

The safety factors used in the evaluation of instability for the System 80+™ containment vessel are determined from NE-3222. Article NE-3222.2(a) requires the maximum buckling stress to be set to one-third of the critical buckling stress determined by the method in NE-3222.1(a)(2) for Levels A and B. Thus the critical buckling stress divided by the maximum buckling stress for these load levels must have a safety factor of 3.0 (including the effects of capacity reduction factors.) The safety factors for Levels C and D are obtained by dividing the Level A and B safety factor by 1.20 and 1.50 respectively (this is the same as multiplying the maximum buckling stress by the same values as stated in NE-3222.2(b) and (c). This results in permissible safety factors of 2.5 for Level C and 2.0 for Level D.

Question 220.49

Section 3.8.2.7 - Provide information on the preoperational structural integrity testing of the SCV. Steel containment vessels of operating plants do not have any benchmark measurements against which their structural behavior can be assessed. The staff believes that such measurements are useful in validating the analytical methods and in assessing the SCV performance after it has gone through significant degradation. Provide information on taking such measurements (chains and deflections) at critical locations during SCV initial pressure testing.

Response 220.49

Preoperational structural integrity testing data will be taken as described in CESSAR-DC Section 14.2.12.1.130, Containment Integrated Leak Rate Test and Structural Integrity Test, which states that "The readings of strain gauges, load cells and deflection rods will be recorded at selected pressure levels."

Question 220.51

Section 3.8.4.4 - Is the dynamic effect of pressure loads considered for both design basis and severe accident conditions? Some of the pressure loads reach close to their peak values in a very short time after the accidents.

Response 220.51

The dynamic effect of pressure loads for postulated pipe breaks not eliminated by leak-before-break evaluations is considered for design basis conditions. Deterministic evaluations for dynamic effects for severe accident conditions are not required.

Question 220.52

GSI 103, "Design for Probable Maximum Precipitation" - See comment 3.11.1 on Table 20-1, and 220.2 on Probable Maximum Flood. Resolution of these comments will resolve the GSI with respect to CE System 80+ Design.

Response 220.52

Responses to questions/comments 311.1 on Table 2.0-1, and RAI 220.2 with regard to Probable Maximum Flood have been provided. Therefore, GSI 103, "Design for Probable Maximum Precipitation," is resolved.

Question 220.52 is incorrect in referencing comment 3.11.1 and Table 20.1. It is believed that the correct references are 311.1 and Table 2.0-1.

Question 220.53

USI A-40, "Seismic Design Criteria" - Resolution of comments on Section 2.5 and 3.7 will resolve the USI with respect to CE System 80+ Design.

Response 220.53

Combustion Engineering agrees with the above staff position. C-E believes that resolution of all RAIS on CESSAR-DC Sections 2.5 and 3.7 has been provided. Therefore, USI A-40, "Seismic Design Criteria" as it relates to the System 80+ design is resolved.

Question 220.54

GSI B-05, "Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments.

- Ductility of Two-Way Slabs and Shells - The stated resolution is acceptable when modified to include the provision 10 and 11 of RG 1.142.

- Buckling Behavior of Steel Containments - Resolution of comments of Section 3.8.2 of the DC document will resolve this concern with respect to CE System 80+ Design.

Response 220.54

The stated resolution of GSI B-05, "Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments," will be revised to include Provisions 10 and 11 of Req. Guide 1.142.

Responses to comments on CESSAR-DC Section 3.8.2 have been provided for resolution of GSI B-05 with respect to Buckling Behavior of Steel Containments.

(Reference 4) does not provide detailed guidance on the treatment of buckling of steel containment vessels for such loading conditions.

Moreover, this Code does not address the asymmetrical nature of the containment shell due to the presence of equipment hatch openings and other penetrations. Regulatory Guide 1.57 recommends a minimum factor of safety of two against buckling for the worst loading condition provided a detailed rigorous analysis, considering in-elastic behavior, is performed.

On the other hand, the 1977 Summer Addendum of the ASME Code 1
permits three alternate methods, but requires a factor of safety between 2 and 3 against buckling, depending upon applicable service limits.

ACCEPTANCE CRITERIA

The acceptance criterion for Concern 1 is that analysis methods used for two-way reinforced concrete slabs adequately address dynamic loading in biaxial membrane tension, flexure, and shear that occur due to a HELB or LOCA.

The acceptance criterion for Concern 2 is that all applied loads must be adequately addressed by the steel containment vessel design.

Insert: (Including Provisions 10 and 11 of Regulatory Guide 1.142, (Reference 5))

RESOLUTION

With respect to Concern 1 of this issue, the System 80+ containment design utilizes the methods outlined in Appendix C of ACI 349-85 (Reference 3) for treating the impactive and impulsive loads associated with a HELB or LOCA (ACI 349-85 is identified in CESSAR-DC, Section 3.8). In addition to these concrete design methods, the System 80+ Standard Design containment piping analysis uses the Leak-Before-Break (LBB) methodology described in CESSAR-DC Section 3.6, thereby reducing the number of situations in which these loadings occur.

With respect to Concern 2 of this issue, the System 80+ Standard Design steel containment vessel satisfies the design requirements set forth in Section III of the ASME Code. The vessel is not subjected to the unsymmetrical dynamic pressure loading described above because of the layout and design of the Reactor Building.

The inside of the containment building is not divided into compartments and thus no unsymmetrical dynamic pressure loading on the steel containment is generated. (The unsymmetrical loading is a consequence of differential pressures within and outside the inside compartments adjacent to containment.)

In addition, a three-dimensional linear bifurcation analysis with appropriate capacity reduction factors is used to arrive at the actual safety factor for the stability analysis as described in CESSAR-DC, Section 3.8.2, and thus satisfies all the current design requirements.

Since the System 80+ containment design is based upon ACI 349-85, which establishes methods by which the above loading conditions for Concern 1 of this issue are addressed, and the steel containment design meets the requirements of the ASME Code for Concern 2 of this issue, both concerns are fully resolved for the System 80+ Standard Design. I

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues", U.S. Nuclear Regulatory Commission, December 1989.

2. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)", U. S. Nuclear Regulatory Commission, June 1978.

3. ACI 349-85, "Code Requirements for Nuclear Safety Related Structures", American Concrete Institute, 1985

4. ASME Boiler and Pressura Vessel Code, Section III, Division I, Subsection NE, American Society of Mechanical Engineers, 1986

Add 5. Regulatory Guide 1.142 - "Safety Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments) U.S. Nuclear Regulatory Commission, October 1981. Rev. 1

Question 220.56

GSI 119.3 "Decoupling the OBE from the SSE"

Per SECY-90-016, the staff agrees that the OBE should not control the design of safety systems. The staff will consider decoupling the OBE from the SSE on a design-specific basis for advanced reactors pending a revision of 10 CFR Part 100, Appendix A. For the CE System 80+ design certification, a revision of 10 CFR Part 100, Appendix A may not support the certification schedule specified in SECY-91-161 which assumed that resolution of the OBE/SEE issue would be resolved on a case-by-case basis.

For the resolution of GSI 119.3, CE has proposed an OBE of 0.1g peak ground acceleration which is one third of the System 80+ SSE (0.3g). Consequently, this proposed resolution is a departure from existing regulations since 10 CFR Part 100, Appendix A calls for a maximum vibratory ground acceleration of the OBE shall be at least one-half the maximum vibratory ground acceleration for the SSE. As such, CE should request an exemption under 10 CFR Part 50.12 for regulatory relief from the applicable provisions of 10 CFR Part 50, Appendix A, GDC 2 and 10 CFR Part 100, Appendix A. In support of the exemption, CE should provide the criteria used for establishing 0.1g OBE point as appropriate for the CE System 80+ design.

In addition, what parameters determine the point, following an OBE event, where seismic inspection activities would be performed for a System 80+ plant in order to verify that structures, systems, and components designed to withstand the effects of an OBE are in an acceptable condition (e.g., within applicable stress and deformation limits) for continued operation? Also, what inspection criteria requested above will be incorporated into the resolution of USI A-17 for the systems interactions program cited in RAI 440.127?

Response 220.56

Combustion Engineering believes that it is inappropraite to request an exemption under 10 CFR Part 50.12 for regulatory relief from the applicable provisions of 10 CFR Part 50, Appendix A, GDC 2 and 10 CFR Part 100, Appendix A at this time. The pending revision to 10 CFR Part 100, Appendix A for application to future nuclear power plants may no longer necessitate such an exemption. Further, it is our understanding that the pending revision to 10 CFR Part 100, Appendix A and/or new regulatory guides to be issued in conjunction with this revision will address shutdown and inspection criteria following a seismic event.

C-E discussed the above in a meeting on January 21 and 22, 1992 in Windsor with Thomas Murley, Director of NRR, William Russell, the senior technical manager in NRR, and other memebers of their technical staff. It is our understanding that additional guidance regarding this request for exemption will be provided by the staff. C-E is taking no action pending receipt of this additional guidance.

For your information, see also the response to RAI 220.40 regarding C-E's position on shutdown and inspection requirements following a seismic event.

Question 230.2

Section 2.5 - Provide technical basis for selecting 5000 ft./sec. bedrock shear wave velocity. Shear wave velocities of 9000 to 12000 ft./sec. are not uncommon in some eastern U.S. sites. Higher bedrock shear wave velocity may produce higher motion at the ground surface. This is particularly true if there is a high shear wave velocity contrast between the rock and the overlying soil.

Response 230.2

A bedrock shear wave velocity of 5,000 fps was selected as a reasonable upper range value. In addition, this bedrock shear wave velocity together with the range of velocities assigned to the cases considered appeared to provide a significant range of velocity variations and contrast.

The effects of having a bedrock shear wave velocity of 2,500 fps, 6,000 fps and 8,000 fps is examined by re-evaluating the response at the ground surface and at the foundation level using soil profiles B-2 and C-2.

The response at the ground surface of soil profile B-2 is shown in Figure 1a for all four bedrock shear wave velocities. The corresponding response at the foundation level (52 ft below ground surface) is presented in Figure 1b. The results of Figures 1a and 1b indicate that varying the bedrock shear wave velocity from 5,000 to 8,000 fps results in only small variations in spectral ordinates of the motions calculated at the ground surface and the foundation level over almost the entire frequency range. Only in a very narrow frequency range of 2 $\pm$ 0.3 Hz, far removed from building structure frequencies where amplification occurs, does the variation in response appear to be significant. Using a rock shear wave velocity of 2,500 fps, however, results in a significant decrease in the spectral ordinates at the ground surface and the foundation level in soil profile B-2.

Figures 2a and 2b show the spectral ordinates for the motions calculated at the ground surface and at the foundation level in soil profile C-2 using rock shear wave velocities of 2,500, 5,000, 6,000 and 8,000 fps. Essentially identical spectral ordinates are obtained when the rock shear wave velocity is varied from 5,000 to 8,000 fps. The spectral ordinates obtained using a rock shear wave velocity of 2,500 fps are slightly lower than those obtained with the higher rock shear wave velocity.

Thus, it is concluded that the use of 5,000 fps for the bedrock shear wave velocity is sufficiently adequate to represent all cases.

RAI 230.2

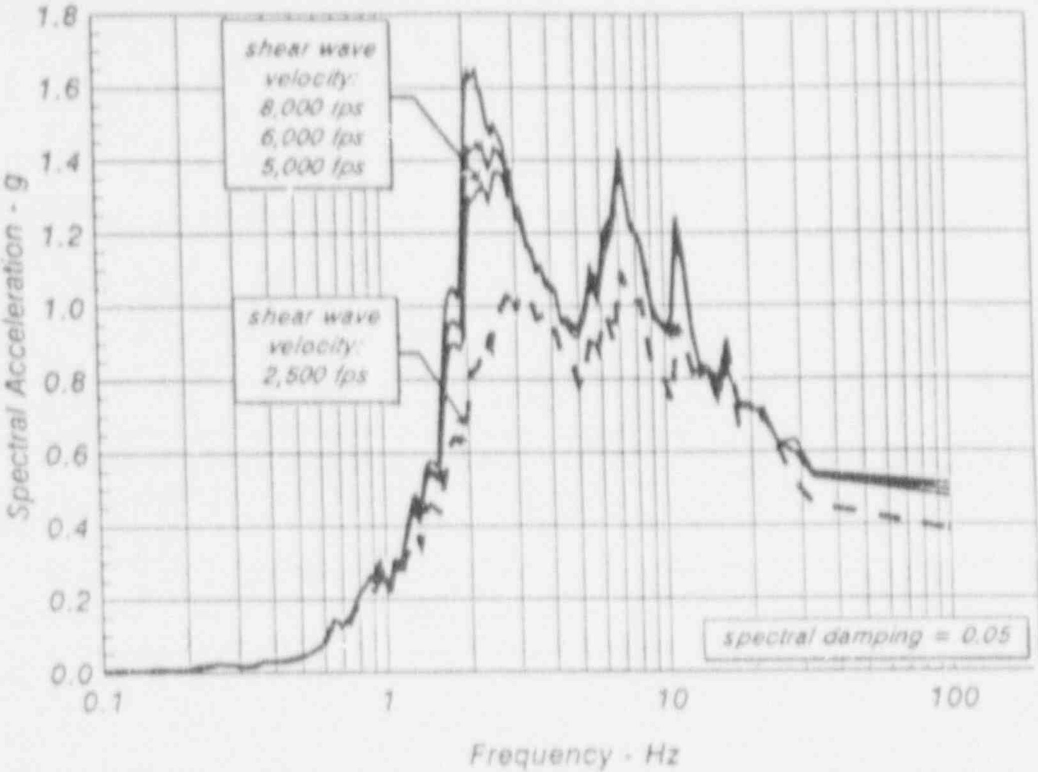


Fig. 1a Influence of Bedrock Shear Wave Velocity on Calculated Spectral Ordinates at the Ground Surface - Case B-2

Fig. 1b Influence of Bedrock Shear Wave Velocity on Calculated Spectral Ordinates at the Foundation Level - Case B-2

RAI 230.2

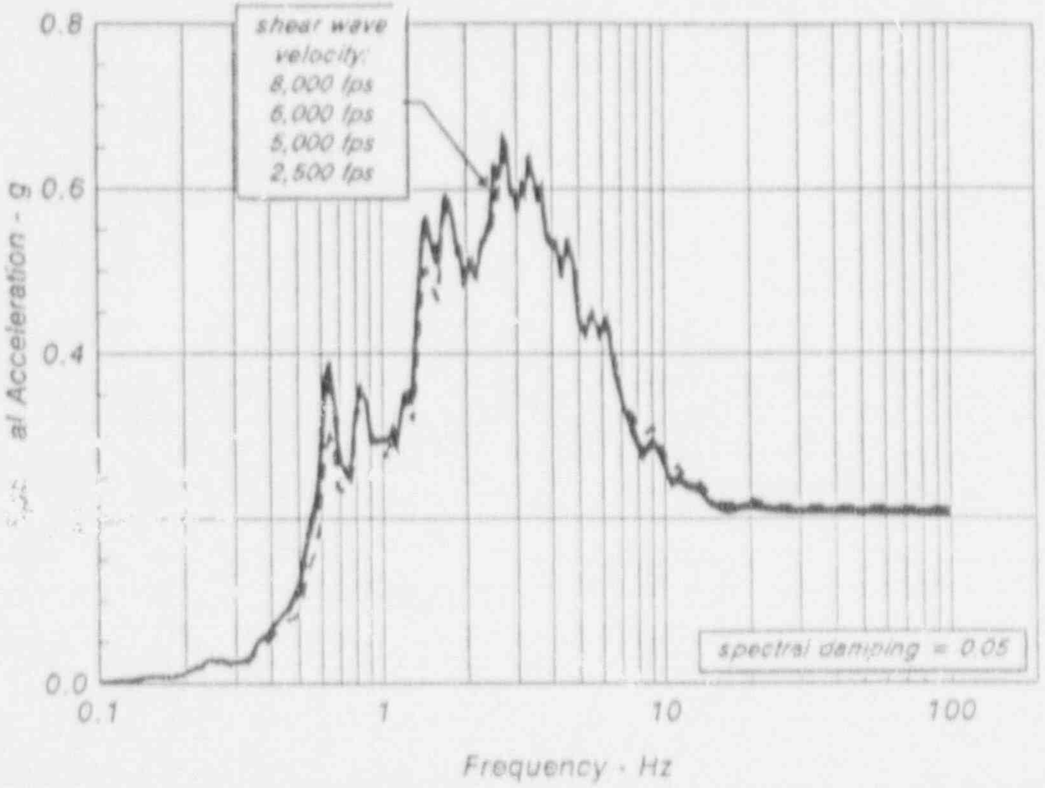


Fig. 2a Influence of Bedrock Shear Wave Velocity on Calculated Spectral Ordinates at the Ground Surface - Case C-2

IMI - 12/30/91

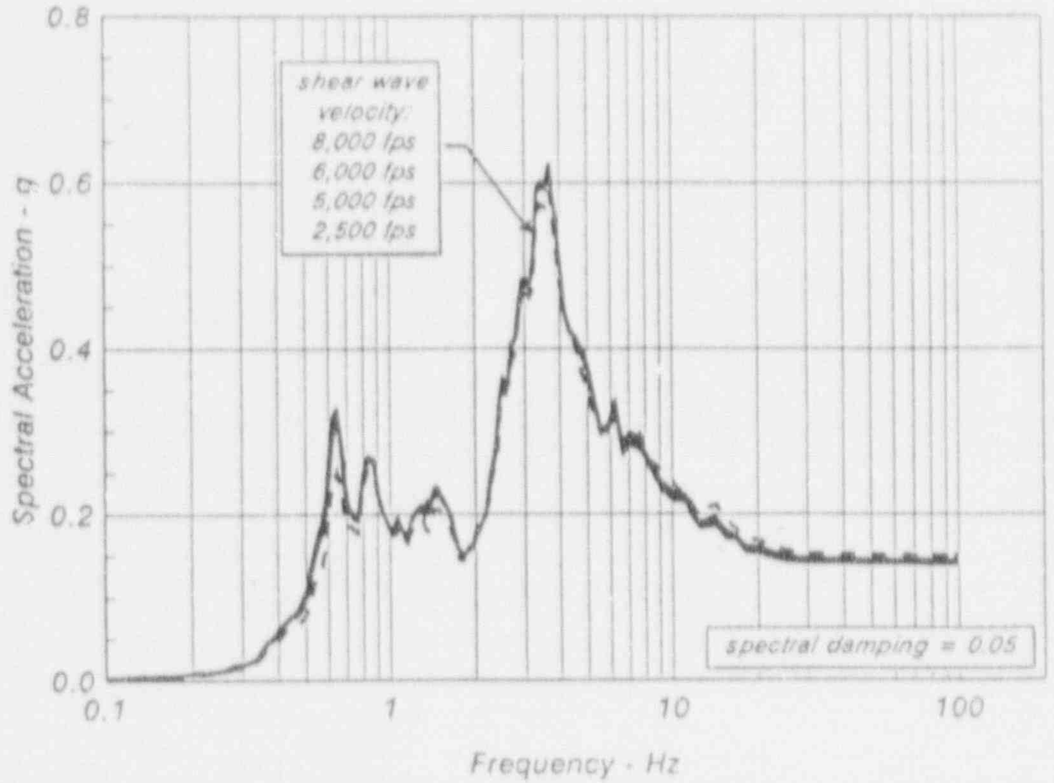


Fig. 2b Influence of Bedrock Shear Wave Velocity on Calculated Spectral Ordinates at the Foundation Level - Case C-2

IMI - 12/30/91

Question 230.3

Section 2.5 - It is stated that all rock sites (with no soil deposits below the foundation level) are acceptable. In case the bedrock is available at or near ground surface, say at a depth of 5 or 10 ft., what is the method of achieving the minimum embedment depth of 52 ft.? In such cases, indicate how the control motion at foundation basemat level will be defined, and what assumptions will be used in designing the structures.

Response 230.3

The structures will be founded at a depth of 52 ft. below finished grade. This will require excavation using appropriate construction procedures (including ripping or blasting in rock, as required) to reach this embedment depth.

The control motion will be defined at the ground surface and will be obtained using the same procedures described in CESSAR-DC. In fact, Case A-1 was included to consider the possibility raised by this question.

Question 230.5

Section 2.5.2.5.1 - Discuss the adequacy of the duration of the time histories.

Response 230.5

The significant duration of the time histories is assessed using the procedure originally proposed by Trifunac and Brady (1975) and used by Dobry, et al (1978) to derive the following equation for the median duration of earthquake ground motions at rock sites:

$$\mathrm{Log}\ (D) = 0.43*M - 1.83$$

in which Log is logarithm to base 10, D is significant duration in seconds and M is earthquake magnitude.

The above expression was derived based on the time required for the buildup of the integral

$$\int_0^t a^2(t)dt$$

in which a(t) is the acceleration time history. Arias (1969) showed that this integral is a measure of the energy of the accelerogram, and defined the intensity of the entire record by the following expression:

$$I_A = \frac{\pi}{2g}\int_0^{t_f} a^2(t)dt$$

in which I_A is the Arias' intensity and t_f is the total duration. Husid (1969) proposed the use of the normalized variable h(t):

$$h(t) = \frac{I_A(t)}{I_A(t_f)} = \frac{\int_0^t a^2(t)dt}{\int_0^{t_f} a^2(t)dt}$$

Thus, h(t) = 0 at the beginning of the record and = 1 (or 100 percent) at the end of the record. The plot of h(t) Vs t is known as the Husid plot.

Significant duration, D, as defined by Trifunac and Brady (1975) is the time interval needed for h(t) to build up from 5 to 95 percent. This definition was used by Dobry, et al (1978) to derive the equation given above for D as a function of M.

A plot of the synthetic acceleration time history H1 used in CESSAR-DC is shown in Fig. 1 and the corresponding Husid plot for this time history is presented in Figure 2. Similar plots were obtained for synthetic time histories H2 and V.

Using the same definition for duration as that selected by Dobry, et al (1978), the following values of significant duration are obtained for H1, H2 and V:

Synthetic Time History	Husid function, h(t)--percent	Time at which H(t) occurs--sec	Significant Duration--sec
	5	4.40	
Horizontal--H1	95	23.99	19.59
	5	3.72	
Horizontal--H2	95	23.55	19.83
	5	2.75	
Vertical--V	95	24.38	21.63

Thus, the significant duration for the synthetic time histories selected for CESSAR-DC is of the order of 20 seconds. This would correspond to a magnitude of about 7 1/4 using the equation relating significant duration and earthquake magnitude. Hence the duration of the selected time histories is adequate.

RAI 230.5

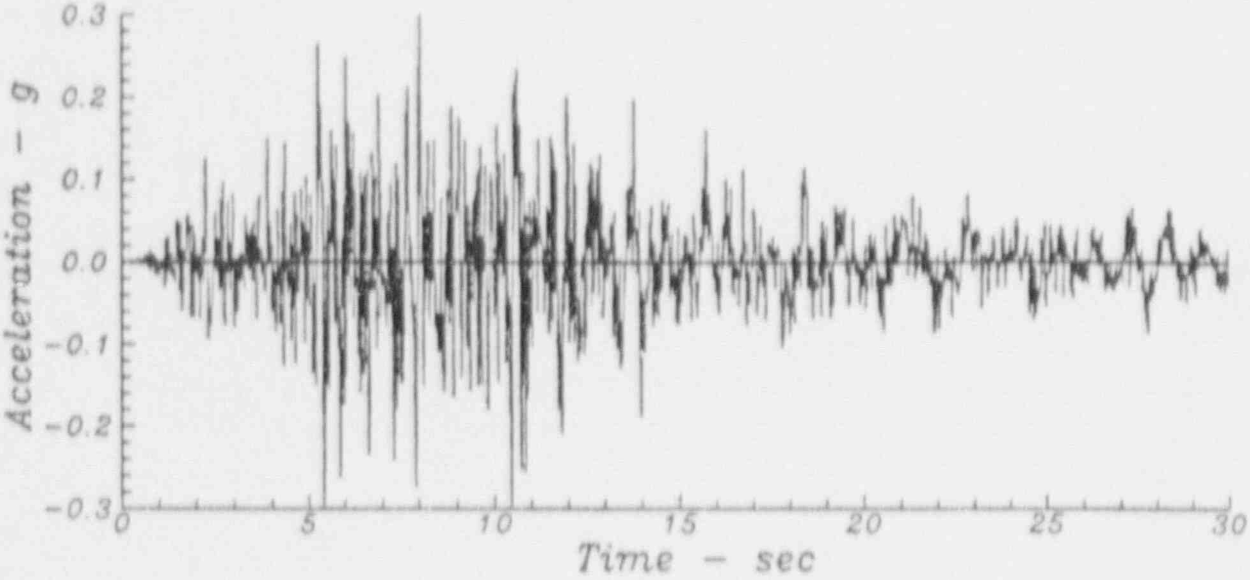


Fig. 1 Synthetic Acceleration Time History H1

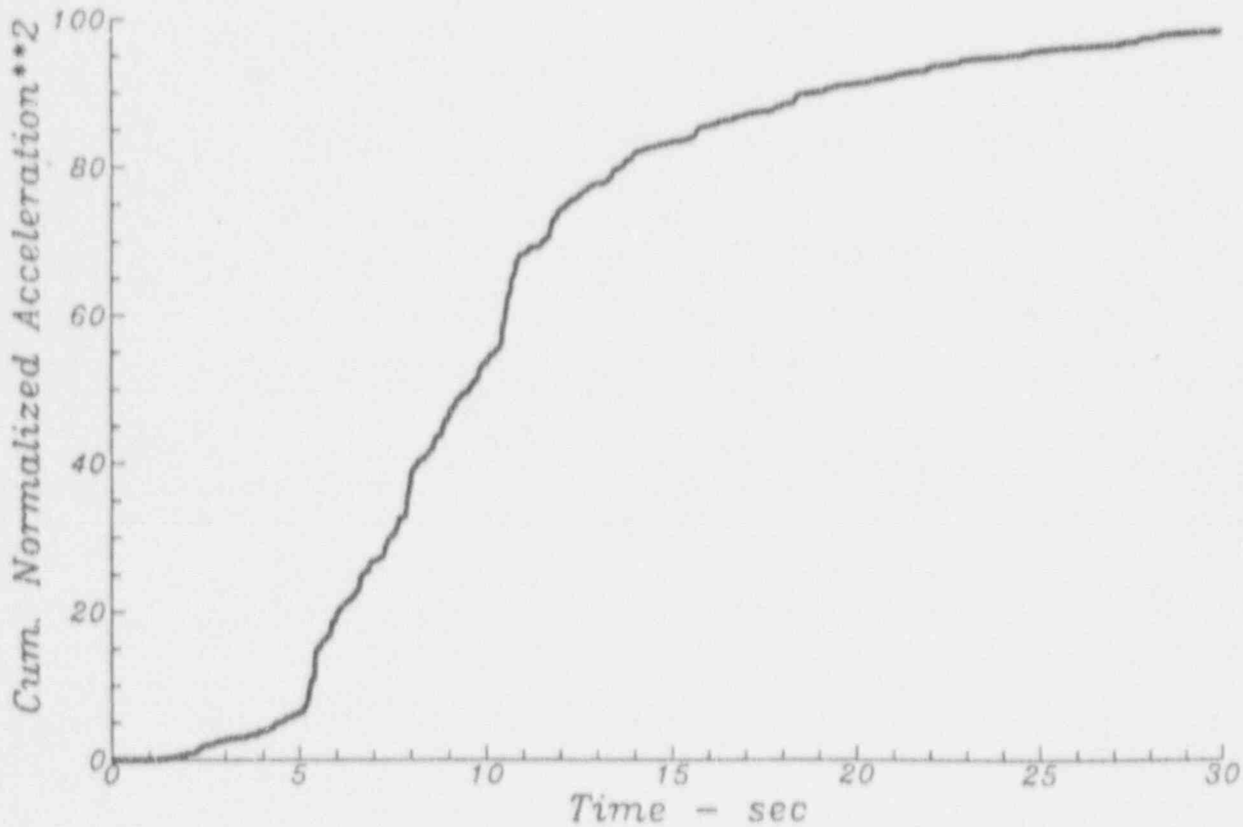


Fig. 2 Husid Plot for Synthetic Time History H1

Question 230.7

Section 2.5.2.5.2 - Why are the upper bounds (non-conservative) of the shear modulus shear strain relationship (Figure 2.5-3) published by Seed and Idriss selected?

Response 230.7

The upper range modulus reduction curve published by Seed and Idriss (1970) was selected for use in the ground analyses for CESSAR-DC. This modulus reduction curve was selected because it represents a reasonable upper range for most cohesionless soils. The possibility that the modulus reduction curve can be higher (ie, less modulus reduction as a function of shear strain) was more than accommodated by the very large range in shear wave velocity (ie, shear modulus at very small shear strain) assigned to the soil profiles considered in CESSAR-DC. Similar arguments pertain to the need to use a lower (ie, more modulus reduction as a function of shear strain) modulus reduction curve.

Available modulus reduction curves for cohesionless as well as cohesive soils are presented in Figure 1; the term PI shown in Figure 1 is the plasticity index. Note that the lower range of the modulus reduction curve orginally published by Seed and Idriss (1970) is not shown in Figure 1 and that the curve labeled PI<10 is approximately equal to the average modulus reduction curve published by Seed and Idriss for sands.

The effects of the modulus reduction curve on response were evaluated by calculating the response of Cases B-1, B-2 and C-3 using the three modulus reduction curves shown in Fig. 2. Synthetic time history H1 was used in these calculations.

The results of these response calculations are presented in Figs. 3, 4 and 5 for Cases B-1, B-2 and C-3, respectively. The upper part of each figure shows the variations of peak horizontal accelerations with depth and the spectral ordinates calculated at the ground surface are shown in the lower part of each figure. These results indicate that "conservatism" is not necessarily obtained by selecting either the lower or the higher modulus reduction curve. Nevertheless, the wide range of soil depths and shear wave velocities used in System 80+ design as described in CESSAR-DC covers the potential variations indicated by the results presented in Figures 3, 4 and 5.

RAI 230.7

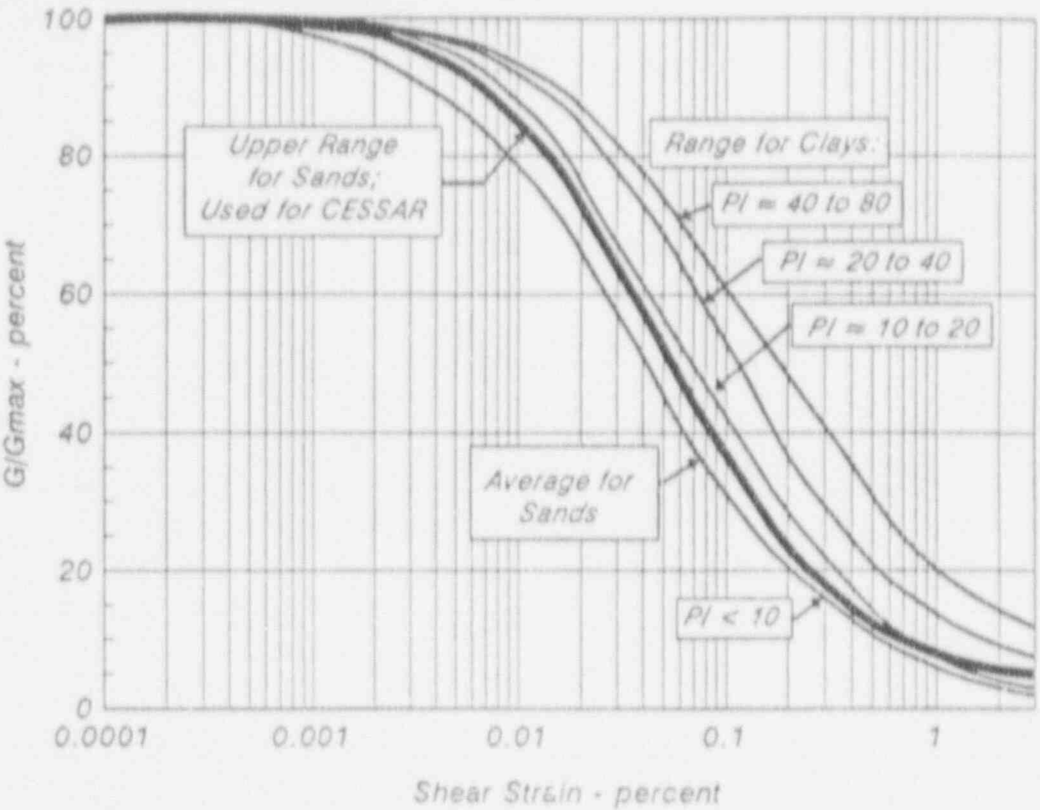


Fig. 1 Range of Modulus Reduction Curves for Clays and for Sands

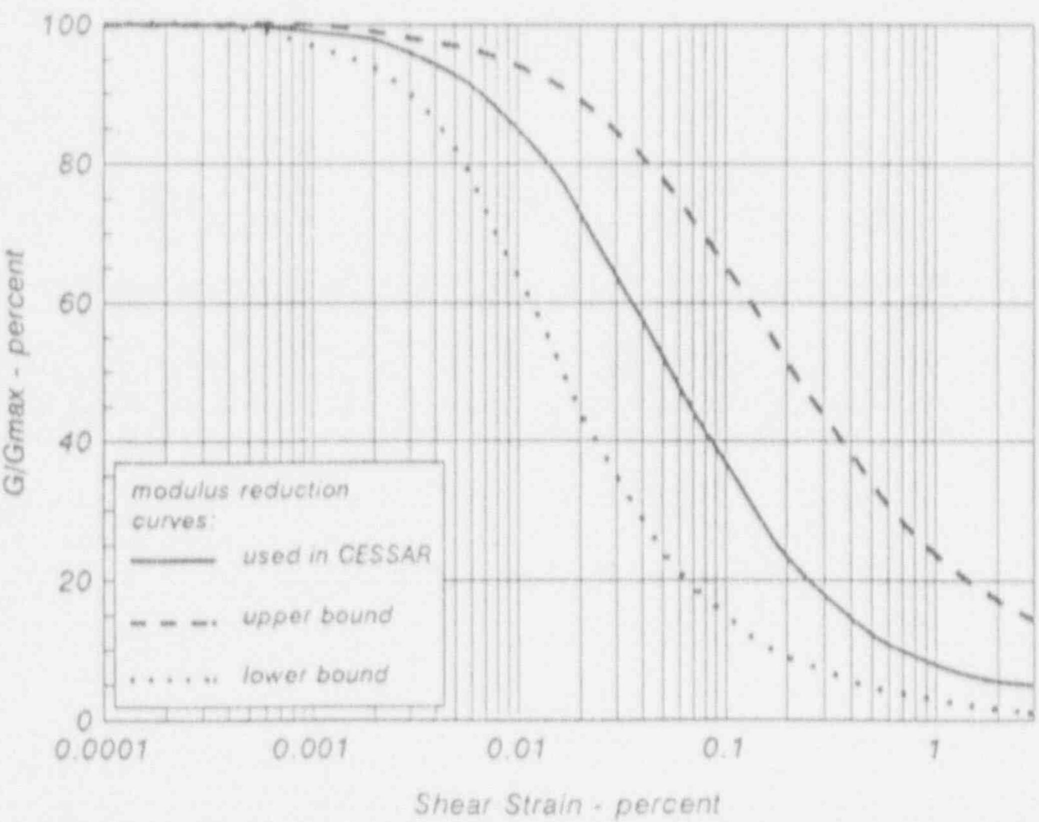


Fig. 2 Modulus Reduction Curves Used in Present Evaluations

RAI 230.7

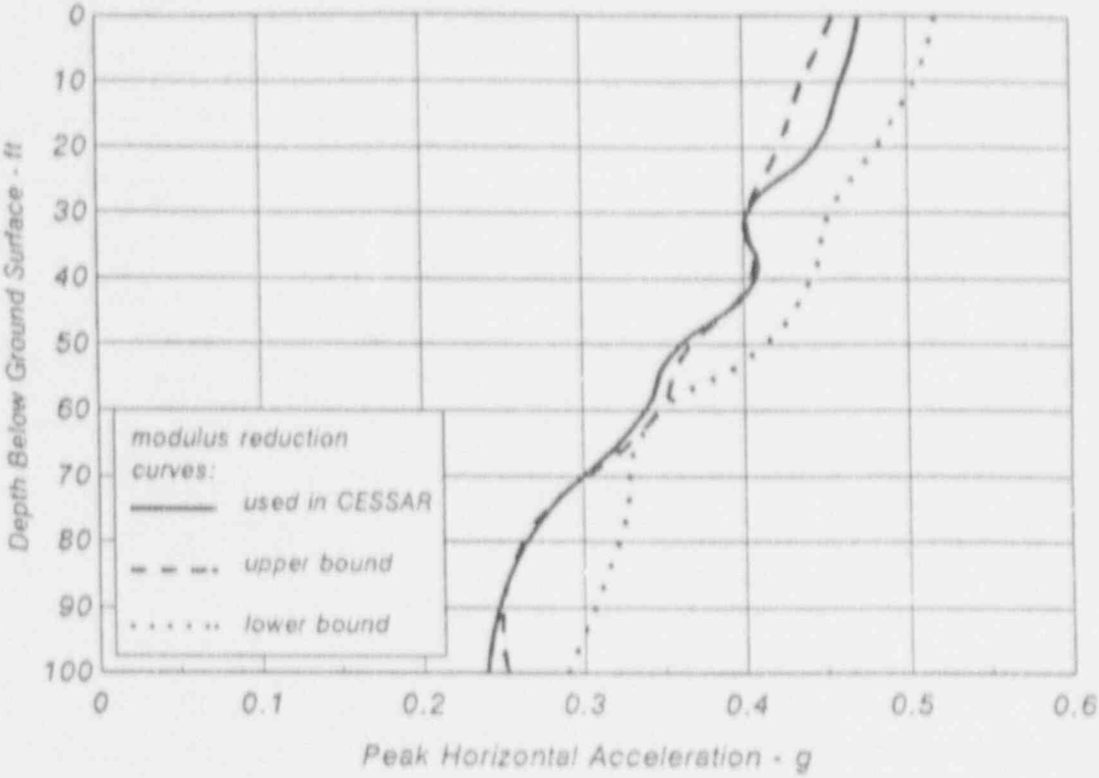


Fig. 3a Effects of Modulus Reduction Curves on Peak Horizontal Accelerations for Case B-1

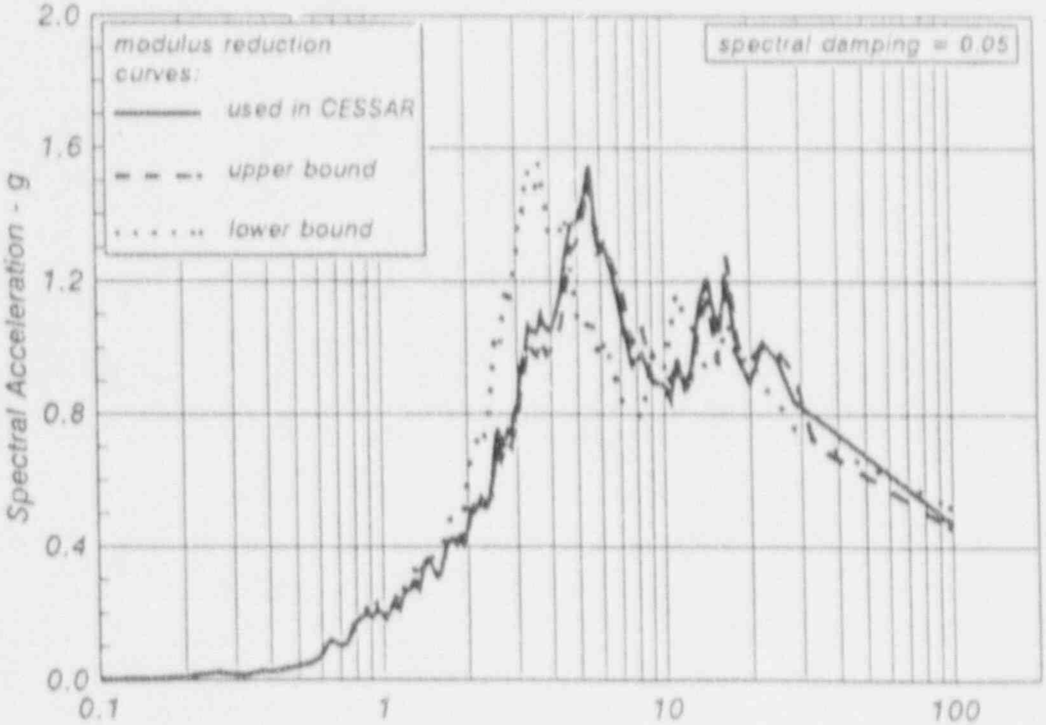


Fig. 3b Effects of M-dulus Reduction on Spectral Ordinates of Calculated Motions at Ground Surface for Case B-1

RAI 230.7

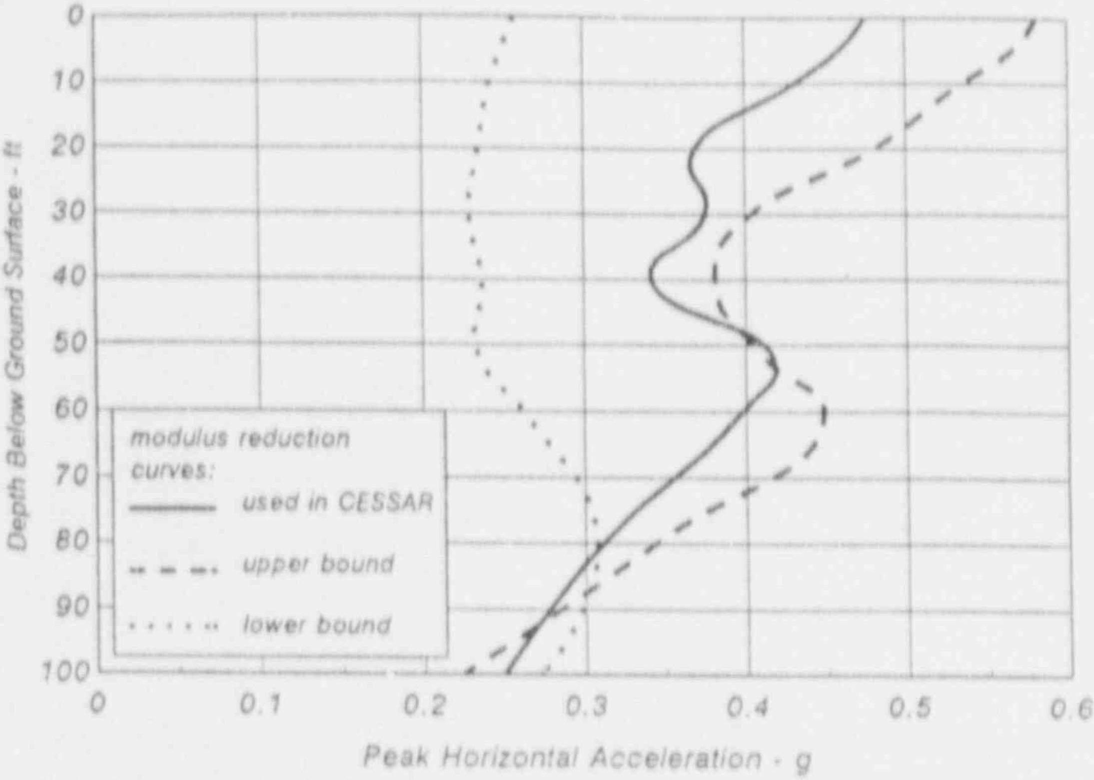


Fig. 4a Effects of Modulus Reduction Curves on Peak Horizontal Accelerations for Case B-2

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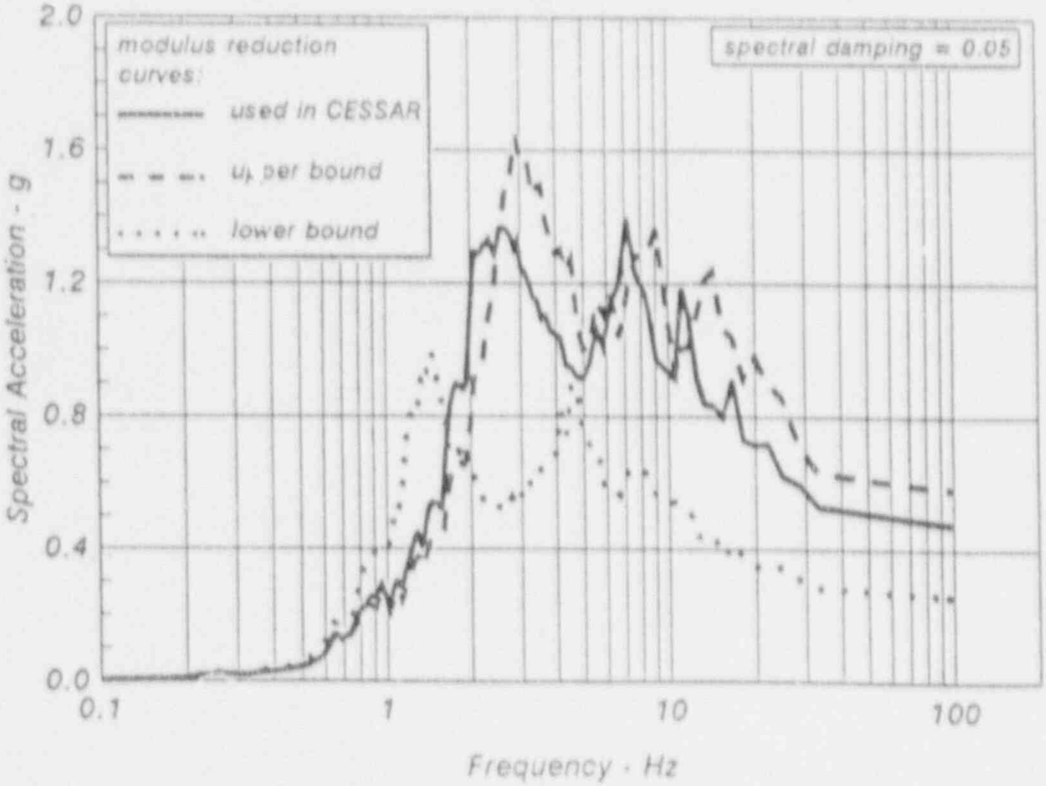


Fig. 4b Effects of Modulus Reduction on Spectral Ordinates of Calculated Motions at Ground Surface for Case B-2

IMI - 12/30/91

RAI 230.7

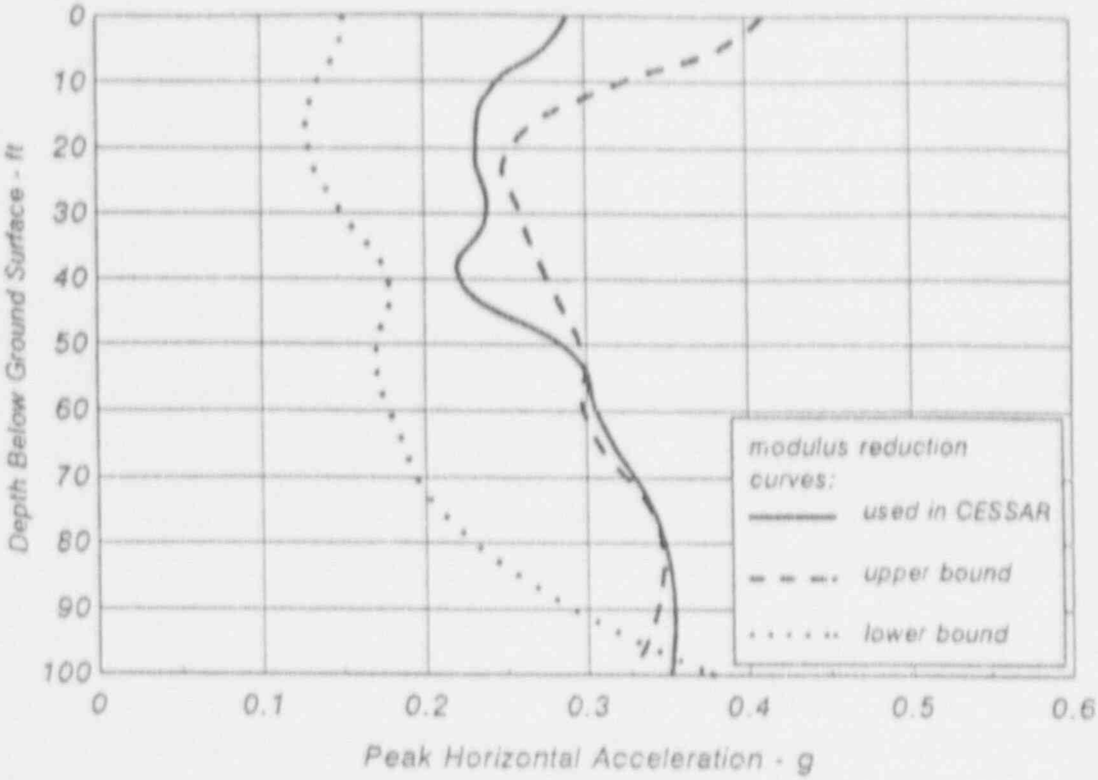


Fig. 5a Effects of Modulus Reduction Curves on Peak Horizontal Accelerations for Case C-3

IMI - 12/30/91

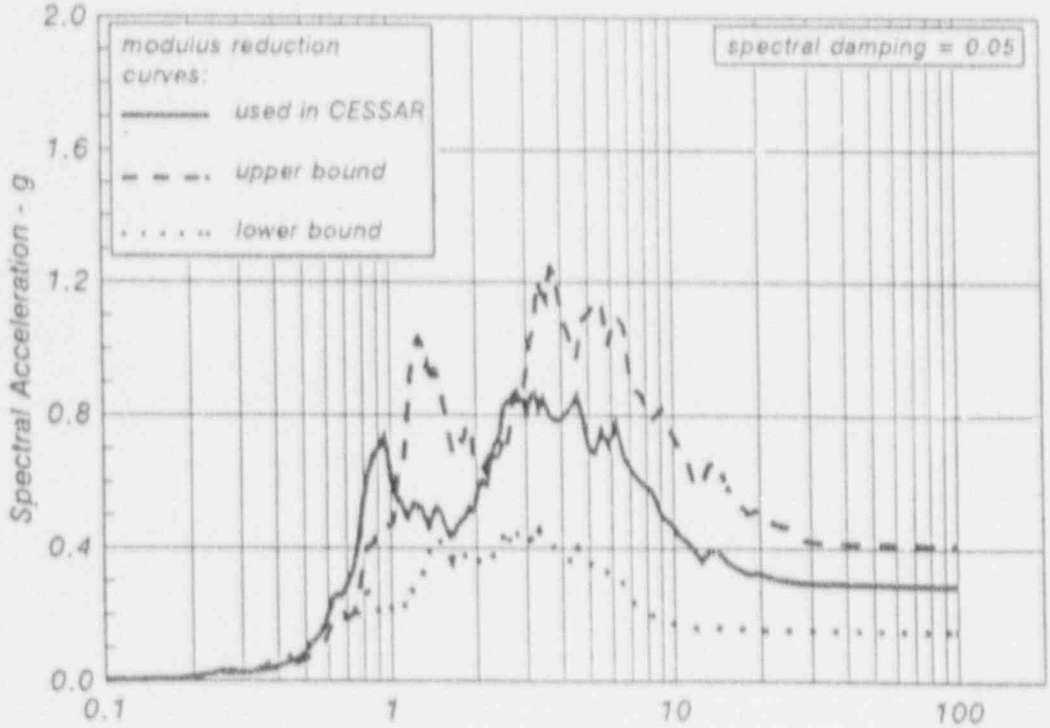


Fig. 5b Effects of Modulus Reduction on Spectral Ordinates of Calculated Motions at Ground Surface for Case C-3

IMI - 12/30/91

Question 230.9

Section 2.5.2.8.1 - Provide explanation for the use of 1/3 factor in estimating damping values associated with the propagation of P-waves.

Response 230.9

Very little work has been done regarding energy disipation associated with the propagation of P-waves. Most, if not all, response calculations completed to date have been done considering that these damping values are identical to those associated with the propagation of shear waves. To provide reasonably conservative vertical ground motions at soil sites, a factor 1/3 was used to estimate the damping values associated with the propagation of P-waves.

From theoretical considerations, it can be shown that the relationship between Q_p and Q_s can be approximated by the following expression:

$$\frac{v_p^2}{Q_p} = \frac{4}{3}\frac{v_s^2}{Q_s}$$

Since damping ratio is inversely proportional to Q, the above equation can be rewritten in terms of the damping ratios λ_p and λ_s as follows:

$$\lambda_p = \frac{4}{3}\lambda_s\left(\frac{v_s}{v_p}\right)^2$$

For $v_s/v_p = 2$, the relationship between λ_p and λ_s is 1/3. This was used as a guide (and not as a basis) for selecting the factor to obtian λ_p from λ_s.

The reasonableness of this selection was tested by examining the recordings obtained at soil sites during the 1979 Imperial Valley eqrthquake. The results presented in Figs. 3.7-1 through 3.7-24 show similar trends to those obtained for the recordings in the aforementioned earthquake.

Question 252.02

Section 1.8 Regulatory Guides

Table 1.8-1 lists applicable Regulatory Guides addressed in the System 80+ design; however, this table is not as comprehensive as Tables B.1-1 and B.1-2 in Appendix B to Chapter 1 of the EPRI ALWR Evolutionary Plant Requirements Document. The EPRI tables list not only applicable Regulatory Guides, but also CFR sections, GDCs, SRPs, Branch Technical Positions, SECY papers, NUREG reports, and NRC memorandums. The staff recommends that CE expands Table 1.8-1 in the same manner.

Response 252.02

Combustion Engineering does not believe that a listing of "SECY papers, NUREG reports, and NRC memoranda" in CESSAR-DC would contribute materially to the description of the System 80+ design or to its review per the Standard Review Plan. Those documents and "Regulatory Guides, CFR Sections, and GDCs" are referenced in the Standard Review Plan and, as appropriate, in the text of CESSAR-DC.

Question 252.16

10 CFR Part 50, Appendix A, General Design Criterion 32 requires, in part, that components which are part of the reactor coolant pressure boundary be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity and (2) an appropriate material surveillance program for the reactor pressure vessel. The information provided in CESSAR-DC Section 5.2.4 does not provide sufficient detail to ensure that GDC 32 will be met. The staff's position is that all System 80+ components should be designed to be inspectable.

The staff requires for the reactor coolant pressure boundary that a Preservice Inspection (PSI) Program Plan be submitted for review. Provide a schedule defining when the entire PSI Program Plan will be completed and submitted for review. The PSI Program Plan should include reference to ASME Code Section XI Edition and Addenda that will be used for the selection of components for examination, lists of the components subject to examination, a description of the components exempt from examination by the applicable Code, examination methods, extent of examination, and the examination isometric drawings.

Plans for preservice examination of the reactor pressure vessel welds should address the degree of compliance with RG 1.150.

Response 252.16

The requested Preservice Inspection (PSI) Plan outline will be part of the Operational Support Information (OSI) Program. The information in CESSAR-DC Section 5.2.4 is to demonstrate that the System 80+ design meets applicable ASME Code and regulatory requirements.

Question 252.17

The information provided in CESSAR-DC Section 5.2.4 does not address Code relief requests. All preservice examination requirements defined in Section XI of the ASME Code that have been determined to be impractical for System 80+ should be identified and a supporting technical justification should be provided. The relief requests should include at least the following information:

1. For ASME Code Class 1 components, provide a table similar to IWB-2500 confirming that either the entire Section XI preservice examination can be performed on the component or relief is requested with a technical justification supporting your conclusion.

2. Where relief is requested for pressure retaining welds in the reactor vessel, identify the specific welds that will not receive a 100 percent preservice ultrasonic examination and estimate the extent of the examination that will be performed.

3. Where relief is requested for piping system welds, Examination Category B-J, provide a list of the specific welds that will not receive a complete Section XI preservice examination including drawing or isometric identification number, system, weld number, and physical configuration (e.g., pipe-to-nozzle weld). Estimate the extent of the preservice examination that will be performed. When the volumetric examination is performed from one side of the weld, discuss whether the entire weld volume and the heat-affected zone (HAZ) and base metal on the far side of the weld will be examined. State the primary reason that a specific examination is impractical (e.g., support of component restricts access, fitting prevents adequate ultrasonic coupling on one side, component-to-component welds prevents ultrasonic examination). Indicate any alternative or supplement examinations to be performed and methods of fabrication examination.

Response 252.17

Preparation of Code relief requests requires final pipe routing, piping/piping support erection, and as-procured vendor information. This information is not available for design certification.

CESSAR-DC, Section 5.2.4 provides preservice examination requirements for Class 1 components as defined in Section XI of the ASME Boiler and Pressure Vessel Code. Where ASME Code Section XI examination requirements are impractical, Code

Response 252.17 (Cont'd)

relief requests are prepared for submission to the NRC, with technical justification and alternative examinations, if any. These requests are prepared subsequent to detailed pipe routing and design.

Question 252.18

Similar to RAI 252.16, the staff finds the information provided in Section 6.6 is not sufficient to ensure that certain Class 2 and 3 systems have been designed to permit (1) appropriate periodic inspection of important component parts to assure system integrity and capability and (2) appropriate periodic pressure testing to assure the structural integrity of their components. The staff's position is that all System 80+ components should be designed to be inspectable.

The staff requires for Class 2 and 3 components that a Preservice Inspection (PSI) Program Plan be submitted for review. Provide a schedule defining when the entire PSI Program Plan will be completed and submitted for review. The PSI Program Plan should include reference to the ASME Code Section XI Edition and Addenda that will be used for the selection of components for examination, lists of the components subject to examination, a description of the components exempt from examination by the applicable Code, examination methods, extent of examination, and the examination isometric drawings.

Response 252.18

The requested Preservice Inspection (PSI) Plan outline will be part of the Operational Support Information (OSI) Program. The information in CESSAR-DC Section 6.6 is to demonstrate that the System 80+ design meets applicable ASME Code and regulatory requirements.

Question 252.19

The information provided in CESSAR-DC Section 6.6 does not address Code relief requests. All preservice examination requirements defined in Section XI of the ASME Code that have been determined to be impractical for System 80+ must be identified and a supporting technical justification should be provided. The relief requests should include at least the following information:

1. For ASME Code Class 2 and 3 components, provide a table similar to IWC-2500 and IWD-2500 confirming that either the entire Section XI preservice examination can be performed on the component or relief is requested with a technical justification supporting your conclusion.

2. Where relief is requested for piping system welds, Examination Category C-F-1 or C-F-2, provide a list of the specific welds that will not receive a complete Section XI preservice examination including drawing or isometric identification number, system, weld number, and physical configuration. Estimate the extent of the preservice examination that will be performed. When the volumetric examination is performed from one side of the weld, discuss whether the entire weld volume and the heat-affected zone (HAZ) and base metal on the far side of the weld will be examined. State the primary reason that a specific examination is impractical (e.g., support of component restricts access, fitting prevents adequate ultrasonic coupling on one side, component-to-component welds prevents ultrasonic examination). Indicate any alternative or supplement examinations to be performed and methods of fabrication examination.

Response 252.19

Preparation of Code relief requests requires final pipe routing, piping/piping support erection, and as-procured vendor information. This information is not available for design certification.

CESSAR-DC, Section 6.6 provides preservice examination requirements for Class 2 and 3 components as defined in Section XI of the ASME Boiler and Pressure Vessel Code. Where ASME Code Section XI examination requirements are impractical, Code relief requests are prepared for submission to the NRC, with technical justification and alternative examinations, if any. These requests are prepared subsequent to detailed pipe routing and design.

Question 252.20

Discuss the relationship between the inspections, tests, and acceptance criteria in your Preservice Inspection (PSI) Program and the inspections, tests, (analyses), and acceptance criteria (ITAAC) required by 10 CFR Part 52. Specifically address to what extent the PSI program will be included in the ITAAC for the certified design and to what extent it will be included in the ITAAC to be submitted with the combined operating license (COL) application.

Response 252.20

The preservice inspection program is discussed in the responses to RAIs 210.51 and 252.16 - 252.19. As indicated, much of the detailed information for a preservice inspection program depends on specific components that will not be available until actual equipment procurement. Information available prior to design certification will, of course, be identified and included in the preservice inspection program as part of the larger Operational Support Information program. These preservice inspection requirements will be included in the ITAAC documentation if they are necessary to demonstrate Tier 1 design features or characteristics. The ITAAC program is currently underway and the scope of Tier 1 features is being developed consistent with NRC/industry discussions.

Question 270.42

Section 3.8.2.1.2 - One of the highly stressed regions of the containment shell under seismic and thermal conditions is in or near the transition region (Sheet 3 of 3, Figure 3.8-1). Provide quantitative information to show that buckling and over-stress will not occur in this region.

Response 270.42

Service Load Analysis and Results

The containment is analyzed for the loads and loading combinations shown in CESSAR-DC Tables 3.8-1 and 3.8-2. The stress intensities calculated in these analyses must be less than the allowable stress intensities per ASME Code Article NE-3221. The allowable stress intensity values are tabulated for the containment vessel material in Attachment 1. The containment vessel is modeled using the finite element method of structural analysis. The finite element model is shown in Attachment 2.

Stress intensities are calculated at the midsurface and extreme fibers of the finite elements in the structural model as appropriate for the load combination being evaluated. Loading combinations for the testing, design Level A, Level B and Level D conditions are assembled. Level C conditions are not considered since this loading level has higher allowable stress intensities but smaller loads than Level D conditions. Only the load combinations containing accident loads are examined for Levels A, B and D since the stress intensities from these combinations are much higher than for those combinations containing normal operating loads. Seismic stress intensities are calculated performing a modal analysis of the model and applying response spectra at the base. Soil case B4 produced the highest stress intensities for the seismic loading.

The maximum stress intensities calculated for the various load combinations are summarized in Attachment 3. The calculated stress intensities are less than the allowable stress intensities for all load combinations in the transition region.

Stability Analysis

The stability analysis is preformed using the methods described in Article NE-3222 of the ASME Code and ASME Code Case N-284. The buckling capacity of the sphere is determined by performing a classical linear bifurcation analysis of the same three dimensional thin shell finite element model used in the service load analysis.

All load combinations are evaluated for buckling considerations. However, many of these combinations need not be analyzed because the containment vessel will not be in a compressive stress state and hence will not buckle. Preliminary buckling analysis showed that stability need only be evaluated when the containment vessel experiences a vacuum or external pressure load.

The load case with the OBE load is checked against Level B allowable safety factors and the load case with the SSE load is checked against Level D allowable safety factors. The buckled mode shapes for both loading combinations shows that the containment vessel buckles in the upper reaches of the dome away from any of the openings. The stress fields at this location consist of unequal biaxial compressive membrane stresses with a small amount of shear stress. The principal stresses are both compressive but unequal. ASME Code Case N-284 does not specifically address this condition so the capacity reduction factor of 0.124 recommended for equal biaxial compressive stresses is used. The load factor (eigenvalue) calculated by ANSYS is multiplied by this capacity reduction factor to determine the safety factor for the load combination under consideration.

Load Combination	Capacity Reduction Factor	ANSYS Load Factor	Safety Factor
1 (OBE)	0.124	24.506	3.039
2 (SSE)	0.124	19.231	2.385

The required safety factor for Level B is 3.0 and for Level D is 2.0 so the containment vessel design is acceptable for stability considerations per the ASME Code.

Attachment 1

RAI Response 270.42

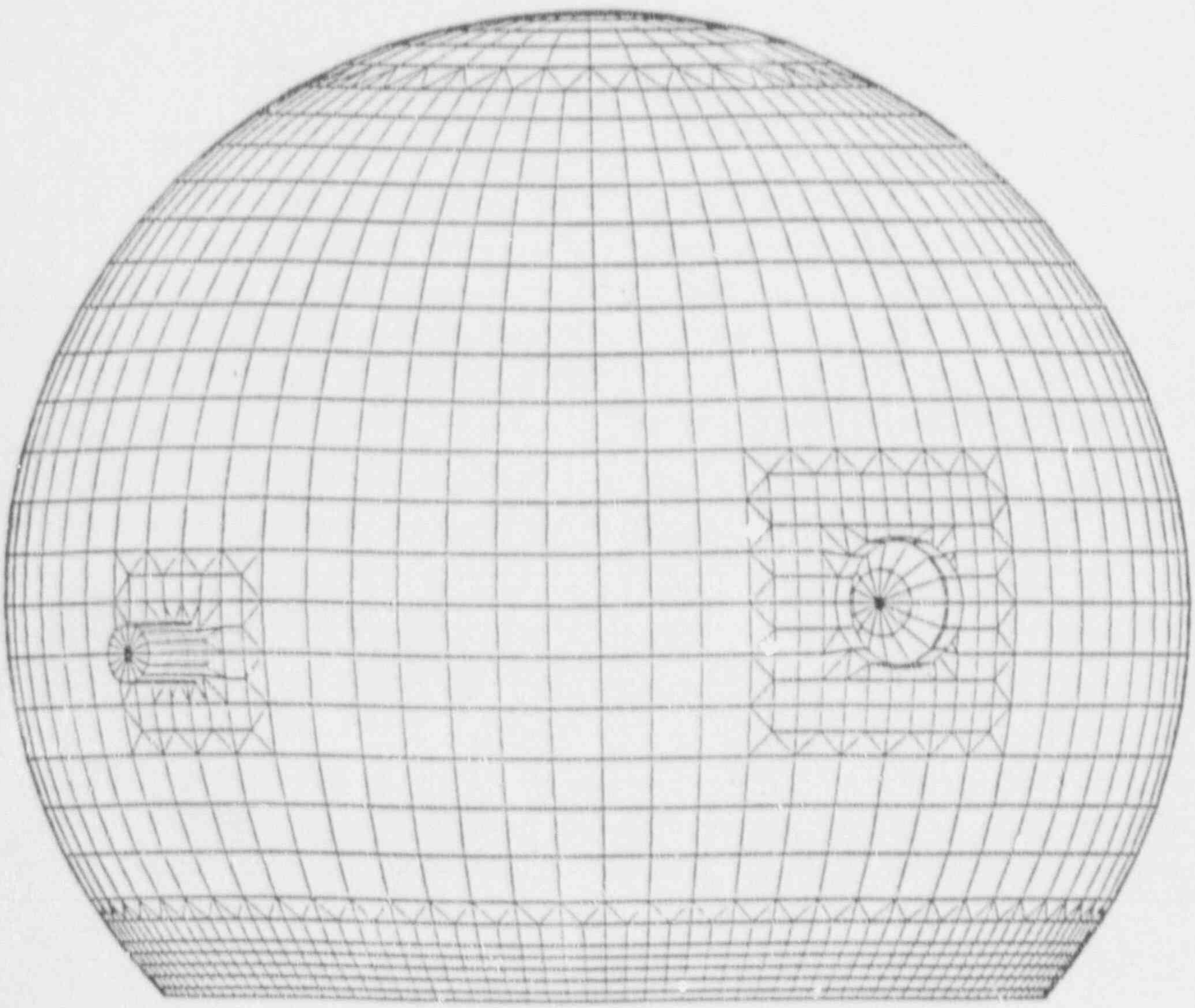
Containment Allowable Stress Intensities for SA537 Class 2 Steel

Load Categories		Primary Stresses Gen. Mem. P_m	Local Mem. P_L	Bending & Local Mem. $P_b + P_L$	Primary & Secondary $P_L + P_b + Q$
Testing Condition	Pneumatic	44250	67650	67850	N/A
Design Condition		22000	33000	33000	N/A
Level A Service Limit		22000	33000	33000	60100
Level B Service Limit		22000	33000	33000	60100
Level C Service Limit	Not integral and Continuous	22000	33000	33000	60100
	Integral and Continuous	52460	78720	78720	N/A
Level D Service Limit	Not Integral and Continuous	52480	78720	78720	N/A
	Integral & Elastic An.	47600	71400	71400	N/A
	Cont. Inelastic An.	47600	47600	47600	

NOTE: All values are given in pounds per square inch.

Attachment 2

RAI Response 270.42



FINITE ELEMENT MODEL STEEL CONTAINMENT VESSEL

Service Load Analysis Stress Intensities

Load Combinations	Containment Shell	
	Primary	Primary and Secondary
Testing	24614	N/A
Design	21184	N/A
Level A	21184	28896
Level B	21775	28650
Level D	30620	N/A

Notes:

1) All stress intensities are in pounds per square inch.

2) All stress intensities are calculated using the preliminary design pressure of 53 psig with the exception of the Level B stress intensities. The Level B stress intensities are calculated at the final design pressure of 49 psig.

Question 270.43

Section 3.8.2.1.2 - It is stated that "In the transition region, compressible material is provided as shown in Figure 3.8-1 to eliminate excessive bearing loads on the concrete...". Explain why compressive material is not provided on the other side of the containment shell? What is this compressive material? Describe how the mechanical properties of the compressive material for the static and dynamic conditions are obtained. What are the uncertainties associated with these mechanical properties? Will this area be accessible for periodic inspection?

Response 270.43

Compressible material is provided only on the exterior of the containment vessel because the primary purpose of this material is to reduce the local bending stresses caused by thermal and internal pressure loads. The material properties are those of self-expanding cork provided by W. R. Grace and Company to Duke Power Company for use in the design of the cancelled Cherokee Nuclear Station. Properties for static loading conditions were determined from load deflection data provided by W. R. Grace and Company. Properties for dynamic conditions were not obtained but should have negligible impact on the design as the compressible material is located where the containment vessel is fully restrained by the concrete structures of the Reactor Building. The procurement specification will define mechanical property requirements, including tolerances, for this material consistent with the properties used in the structural evaluations. The area at the transition region of the containment vessel on both the inside and outside of the vessel will be available for inspection over the life of the plant.

Question 270.44

Section 3.8.2.4 - It is stated that "The critical buckling stresses in the containment vessel are determined by applying the appropriate safety factors and capacity reduction factors to the results of a three-dimensional linear bifurcation analysis using an ANSYS finite element model similar to that constructed for the static and dynamic analysis." Provide figures or sketches showing the finite element model for the buckling analysis and describe the differences between this model and the model for the static and dynamic analyses.

Response 270.44

The buckling analysis model is summarized in the response to RAI 220.42. The same three-dimensional model is used for both the static and dynamic analysis as well as the buckling analysis (Reference Figure 3.8.3).

CEASSAR-DC Section 3.8.2.4 will be revised in a future submittal.

as a three dimensional thin shell using the finite element method of analysis. The stresses and deflections produced in the shell under the applied loads are calculated with the ANSYS computer program (Reference 2). The ANSYS mathematical model used to represent the containment vessel is shown in Figure 3.8-3.

Seismic stresses and deflections are calculated using the response spectrum method. The frequencies of vibration and corresponding mode shapes are determined using the normal mode method. Modal responses are combined as described in Regulatory Guide 1.92 (Reference 15). The appropriate damping level for the applied response spectra is defined in Regulatory Guide 1.61 (Reference 14).

C. Buckling

The critical buckling stresses in the containment vessel are determined by applying the appropriate safety factors and capacity reduction factors to the results of a three-dimensional linear bifurcation analysis using ~~an~~ the ANSYS finite element model ~~similar to that~~ constructed for the static and dynamic analyses. These methods are described in Article NE-3222 of the ASME Code and ASME Code Case N-284 (Reference 5). Code Case acceptability is in concurrence I
with Regulatory Guide 1.84 (Reference 18).

D. Ultimate Capacity

The maximum pressure capacity of the containment vessel is evaluated by a large displacement elastic-plastic nonlinear analysis. The vessel is modeled with axisymmetric shell finite elements using the ANSYS computer program. The ANSYS model is shown in Figure 3.8-4.

The stresses in the containment vessel due to combustible gas loadings are calculated using a static linear elastic analysis. The vessel is represented by a three-dimensional thin shell finite element model with the ANSYS computer program. This model is similar to those used for the seismic analysis and buckling evaluation.

E. Nonaxisymmetric and Localized Loads

The containment is not divided into compartments (see Section 6.2.1.2) so there are no nonaxisymmetric loads applied to the containment vessel during a Design Basis Accident.

Question 311.1

Table 2.0-1

Precipitation - The EPRI-ALWR requirement document (RD) prescribes the maximum rainfall rates of 19.4 in/hr., and 6.2 in/5min. Justify the use of 10 in./hr. for the System 80+ design. Also, at some locations in the continental U.S., the 50 lbs./sq. ft. snow loading may not be adequate for 100 year occurrence interval. Provide the basis for the general use of the 50 lbs./sq. ft. load.

Soil Properties -

- The minimum shear wave velocity of soil is stated as 500 ft./sec. (fps). Figure 2.5-2 shows that the minimum shear wave velocity of soil ranges from about 550 fps at the ground surface to about 850 fps at a depth of 200 ft. At its meetings with the staff in October 1990, CE stated that the minimum value of shear wave velocity will be 700 fps and that the required soil strength will be governed by static analysis (e.g., settlement of structures). EPRI ALWR Requirements Documents assumes a minimum shear wave velocity of 100 fps. Justify the use of a lower shear wave velocity and explain the difference in your presentation.

- What is the postulated margin against liquefication potential at site specific and generic SSE?

Response 311.1

Precipitation

The use of 10 in. per hour for the maximum rainfall rate was based on Revision 0 of the EPRI Requirements Document. The values of 19.4 in. per hour and 6.2 in. per 5 min. were incorporated into Revision 1 of the EPRI Requirements Document. The System 80+ design will comply with the Revision 1 values and CESSAR-DC Table 2.0-1 will be revised in a future amendment to reflect this change.

Concerning the use of 50 pounds per square foot for snow loading; this value was chosen based on the EPRI requirements and was deemed suitable to envelope the majority of possible plant sites within the continental United States. Plant sites with possible snow loadings in excess of the 50 pounds per square foot parameter will be evaluated on a site specific basis.

Response 311.1 Cont.

Minimum Shear Wave Velocity

The minimum shear wave velocity used is 500 fps; there was a slight shift in the plot in Fig. 5.2-2 which resulted in an apparent minimum shear wave velocity of about 550 fps instead of the intended minimum velocity of 500 fps. A corrected Fig. 2.5-2 is attached.

The range of shear wave velocities was selected to bracket potential shear wave velocities at many sites where a nuclear plant may be constructed. Thus, values of shear wave velocities below those that might exist at nuclear power plant sites as well as variations in wave velocities with depth that are not particularly likely to exist were selected. These variations produced conservative ground motions at both the ground surface as well as at the foundation level.

The EPRI ALWR Requirements Document assumed a minimum shear wave velocity of 1,000 fps. The range of shear wave velocities selected for CESSAR-DC includes all the possibilities considered in the EPRI ALWR Requirements Document. As noted above, however, it was considered useful to bracket a wider range of shear wave velocities for CESSAR-DC and hence the choice of 500 fps was selected as the minimum value.

Margin Against Liquefaction Potential

The margin against liquefaction will be determined on a site-specific basis taking into account the nature (including type, denseness, fines content...etc.) of the soils at the site and the depth of the water table together with the level of shaking associated with the site specific SSE. A margin of about 1.5 will be required when the median cyclic resistance of the soils (as may be determined from cyclic tests, from SPT blow count, from CPT ...etc) is used in the liquefaction evaluation and a minimum margin of about 1.1 will be required when the median minus one standard deviation cyclic resistance of the soils is used.

TABLE 2.0-1

(Sheet 1 of 2)

ENVELOPE OF PLANT SITE DESIGN PARAMETERS

Ground Water	
Maximum Level:	2 feet below grade(1)
Flood (or Tsunami) Level(2)	
Maximum Level:	1 ~~foot~~ below grade
Precipitation (for Roof Design)	
Maximum rainfall rate:	19.4 in/hr. and 6.2 in/5 min.(8) ~~10 in/hr~~
Maximum snow load:	50 lb/sq. ft.
Design Temperatures	
Ambient	
1% Exceedance Values	
Maximum:	100°F dry bulb 77°F coincident wet bulb
Minimum:	-10°F
0% Exceedance Values (Historical Limit)	
Maximum:	115°F dry bulb 82°F coincident wet bulb
Minimum:	-40°F
Emergency Cooling Water Inlet:	95°F
Condenser Cooling Water Inlet:	≤100°F
Extreme Wind	
Basic Wind Speed:	100 mph(3) / 130 mph(4)
Tornado(5)	
Maximum tornado wind speed:	330 mph
Translational velocity:	70 mph
Radius:	150 ft
Maximum atmosphere P:	2.4 psid
Missile spectra:	per SRP 3.5.1.4 Spectrum II

TABLE 2.0-1 (Cont'd)

(Sheet 2 of 2)

ENVELOPE OF PLANT SITE DESIGN PARAMETERS

Soil Properties

Minimum Bearing Capacity (demand):	15 ksf (static)	H
Minimum Shear Wave Velocity:	500 ft/sec(6)	
Liquefaction Potential:	None (at site-specific SSE level)	D

Seismology

OBE Peak Ground Acceleration (PGA):	0.10 g (7)	
SSE PGA:	0.30 g (7)	
SSE Response Spectra:	Section 3.7.1	
SSE Time History:	Section 3.7.1	H

1. Site will be dewatered to elevation 40+0 with permanent, safety grade dewatering system.

2. Probable maximum flood level (PMF), as defined in ANSI/ANS-2.8, "Determining Design Basis Flooding at Power Reactor Sites."

3. 50-year recurrence interval; value to be utilized for design of non-safety-related structures only.

4. 100-year recurrence interval; value to be utilized for design of safety-related structures only.

5. 10,000,000-year tornado recurrence interval, with associated parameters based on the NRC's interim position on Regulatory Guide 1.76. Pressure effects associated with potential offsite explosions are assumed to be non-controlling for the design.

Add

6. Site profiles are given in Section 2.5.

7. The control motion is defined at a hypothetical rock outcrop.

8. Maximum value for 1 hour 1 sq. mile PMP with ratio of 5 minutes to 1 hour PMP of .32 as found in National Weather Service Publication HMP No. 52

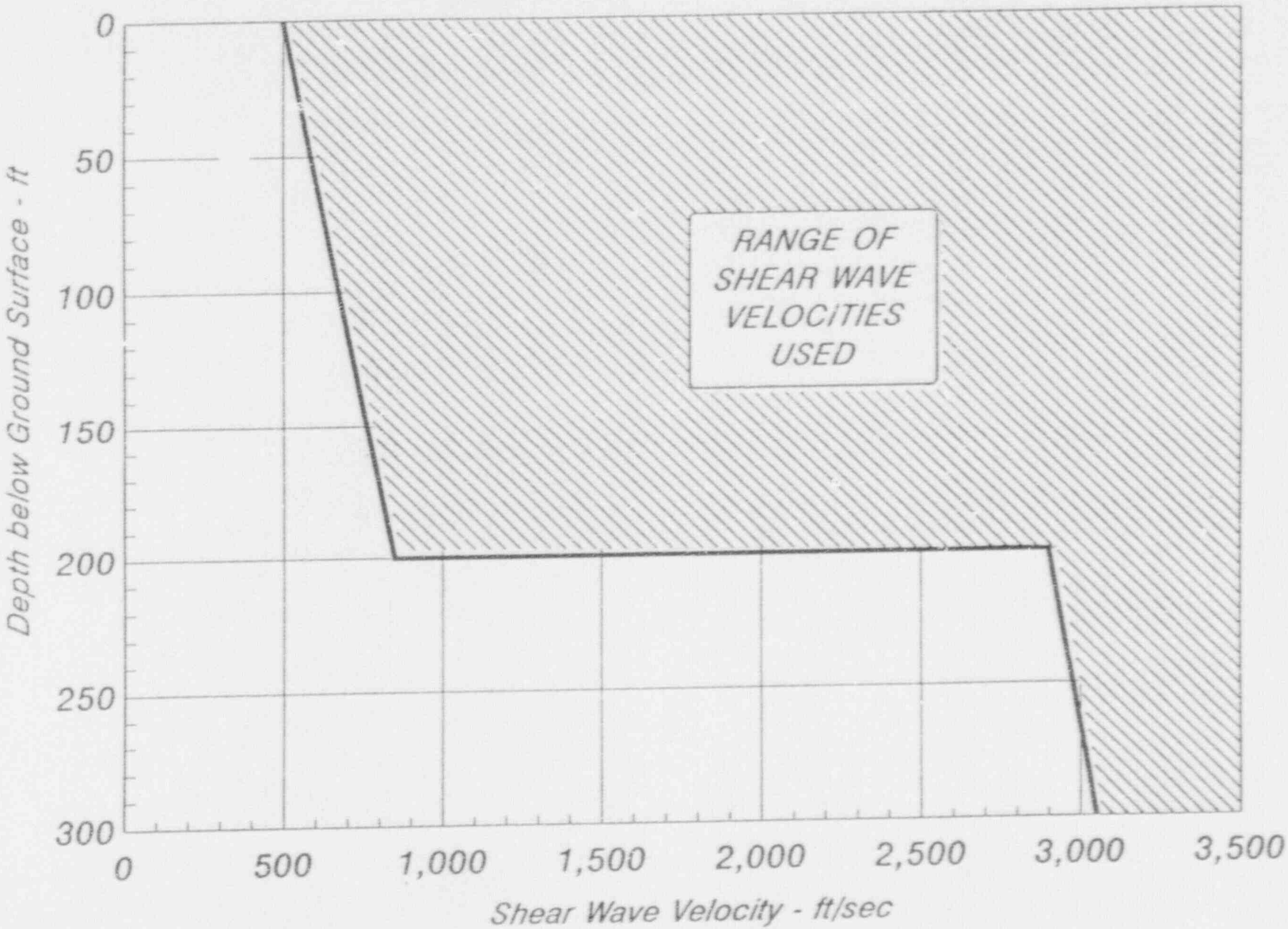


Fig. 2.5-2 RANGE OF SHEAR WAVE VELOCITIES
FOR ALL CASES CONSIDERED

IMI - 12|30|91

RAI 511.1

Question 311.2

Section 2.2.1 - The minimum distances stated in this Section may in many cases be inadequate to ensure that the aircraft hazard is insignificant at the site. For example, if a large airport is 10 miles from the site, the site analysis is likely to reveal high probability of aircraft hazard at the plant site. Provide justification for the stated distances.

Response 311.2

CESSAR-DC Section 2.2.1 is intended to provide site acceptance criteria that meet the relative requirements of 10 CFR Part 100, Paragraph 100.10 as it relates to the site location insuring a low risk of public exposure. With general guidance provided in USNRC Standard Review Plan 3.5.1.6 "AIRCRAFT HAZARDS", "This requirement is met if the probability of aircraft accidents resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is less than about 10^{-7} per year."

CESSAR Section 2.2.1 will be revised to include restrictions on the projected annual number of operations and/or flights as follows:

A site is acceptable for the System 80+ plant without further review if the distances from the plant meet the following requirements:

1. The plant-to-airport distance D is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance D is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2,

2. The plant is at least 5 statute miles form the edge of military training routes, including low-level training routes, except for those associated with a usage greater than 1000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation,

3. The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or airport.

If the above site proximity acceptance criteria are not met, or if sufficiently hazardous military activities are identified, a detailed review of the aircraft hazards must be performed to qualify a specific site for the System 80+ plant.

Attachment to ALWR-336

Response to RAI 311.2.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION AND MILITARY FACILITIES

Industrial, transportation and military hazards are discussed below.

Replace this section w/info. on next page.

2.2.1 AIRCRAFT HAZARDS

~~The System 80+ plant will not be sited in regions of high aircraft traffic. It will be sited at least 5 miles from small commercial airports, 10 miles from large airports, 5 miles from military training routes, and 2 miles from the nearest airway, holding pattern, or approach pattern. After an actual site location is selected, the aircraft impact potential will be re-evaluated by the site operator to verify that siting criteria are satisfied.~~ H

2.2.2 TRANSPORTATION

The offsite power transmission lines will be separated such that a single transportation accident event does not result in a loss of all offsite power sources.

A closed-cycle cooling system (the ultimate heat sink, Section 9.2.5) which provides the source of cooling water for all safety-related plant systems and components during all modes of operation is incorporated in the System 80+ Standard Design to eliminate the potential impacts on plant operations from boat or barge accident events.

Plant security and other barriers will be used at a System 80+ plant site to preclude transportation events from having an impact on plant operations.

After a specific site is chosen, transportation accidents will be evaluated by the site operator to ensure that these types of accidents will not have an impact on the plant.

2.2.3 OTHER INDUSTRIAL HAZARDS ON AND OFF SITE

2.2.3.1 Hazardous Material Releases On Site

Operating practices that tightly control the use, transportation, and storage of hazardous materials on site will be followed for System 80+ facilities.

Chemical quantities will be such that no adverse impact on the operation of the plant could occur.

The use of hazardous materials will be limited to those applications where no viable alternatives exist.

RAI 311.2

Replace as Section 2.2.1

A site is acceptable for the System 80+^{tm} without further review if the distances from the plant meet the following requirements:

1. The plant-to-airport distance D is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance D is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2.

2. The plant is at least 5 statute miles from the edge of military training routes, including low-level training routes, except for those associated with a usage greater than 1000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation.

3. The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or airport.

If the above site proximity acceptance criteria are not met, or if sufficiently hazardous military activities are identified, a detailed review of the aircraft hazards must be performed to qualify a specific site for the System 80+^{TM} plant.