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DRAWINGS CITED IN THIS CHAPTER*

*The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

DRAWING*	SUBJECT
M-5	General Arrangement Roof Plan Units 1 & 2
M-6	General Arrangement Main Floor At El. 451'-0" Units 1 & 2
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M-8	General Arrangement Grade Floor At El. 401'-0" Units 1 & 2
M-9	General Arrangement Floor Plan At El. 383'-0" Units 1 & 2
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M-12	General Arrangement Radwaste/Service Building Units 1 & 2
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M-14	General Arrangement Section "A-A" Units 1 & 2
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M-16	General Arrangement Section "C-C" and "D-D"
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M-35	Diagram of Main Steam System Unit 1
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DRAWING*	SUBJECT
M-66	Diagram of Component Cooling Water System Units 1 & 2
M-66A	Composite Diagram of Component Cooling Units 1 & 2 (Bryon)
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M-900	Outdoor Piping Essential Service Water at Cooling Tower (Byron)
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NCT-683-14H	General Installation Detail, ESW Cooling Tower Units 1 & 2 (Byron)
S-239	Essential Service Cooling Tower Foundation Plan (Byron)
S-241	Essential Service Cooling Tower Air Inlet Plan El. 875'-6"
S-243	Essential Service Cooling Tower Fill Support Beam Plan El. 888'-0" (Byron)
S-245	Essential Service Cooling Tower Distribution Support Beam Plan El. 901'-5" (Byron)
S-247	Essential Service Cooling Tower Roof Framing Plan El. 909'-6" (Byron)
S-249	Essential Service Cooling Tower Section 1-1 (Byron)
S-250	Essential Service Cooling Tower Sections & Details Sheet 1 (Byron)
S-259	Essential Service Cooling Tower Drainage Duct Plan, Section & Details (Byron)
S-1051	Containment Structural Acceptance Test Instrumentation

CHAPTER 3.0 - DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 Summary Description

The intent of the design of the Byron/Braidwood Stations is to conform to the Licensee's interpretation of the intent of Appendix A to 10 CFR 50. In this section, individual General Design Criteria are stated, the Licensee's interpretation is discussed, and reference is made to the specific portion of the UFSAR where further information is presented.

Based on the following discussions, the Licensee concludes that these stations fully satisfy and are in compliance with the NRC General Design Criteria.

3.1.2 Criterion Conformance

3.1.2.1 Group I - Overall Requirements

3.1.2.1.1 Evaluation Against Criterion 1 - Quality Standards and Records

"Structures, systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit."

RESPONSE

The total quality assurance program is described in Chapter 17.0 and consists of Topical Report NO-AA-10.

The detailed quality assurance program developed by the Licensee satisfies the requirements of Criterion 1.

3.1.2.1.2 Evaluation Against Criterion 2 - Design Bases for Protection Against Natural Phenomena

"Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed."

RESPONSE

The design of these stations conforms to the intent of Criterion 2. The historical records and other information influencing the selection of the design-basis natural phenomena are given in Sections 2.3, 2.4, and 2.5.

The design criteria to ensure that the Byron/Braidwood Stations can withstand the effects of natural phenomena, are given in Sections 3.3 through 3.11. The systems, components, and structures important to safety have been designed to accommodate, without loss of capability, effects of the design-basis natural phenomena along with appropriate combinations of normal and accident conditions.

3.1.2.1.3 Evaluation Against Criterion 3 - Fire Protection

"Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components."

RESPONSE

The design of the Byron/Braidwood Stations conforms to the intent of Criterion 3.

For further information, see Reference 1.

3.1.2.1.4 Evaluation Against Criterion 4 - Environmental and Missile Design Bases

"Structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss of coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit."

RESPONSE

Safety-related systems, components, and structures in this plant are designed to accommodate all normal or routine environmental conditions as well as those associated with postulated accidents (where appropriate). The design includes provisions to protect (by physical separation, barriers, or appropriate restraints) safety-related items from dynamic effects resulting from component failures, and specific credible external events and conditions.

The design criteria for these systems, components, and structures are discussed in the remainder of this chapter.

3.1.2.1.5 Evaluation Against Criterion 5 - Sharing of Structures, Systems, and Components

"Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units, is not significantly impaired by the sharing."

RESPONSE

Those systems, structures, and components important to safety shared by the two units are the ultimate heat sinks and the associated Byron makeup water systems; various heating, ventilating, and air conditioning systems within the shared auxiliary and fuel handling building; and a component cooling heat exchanger which can be valved to serve one unit or the other. These shared systems, structures, and components are more fully described elsewhere in this report. No safety-related systems, structures, or components are shared unless such sharing has been evaluated to ensure that there will be no significant adverse impact on safety functions.

3.1.2.2 Group II - Protection by Multiple Fission Product Barrier

3.1.2.2.1 Evaluation Against Criterion 10 - Reactor Design

"The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

"The reactor core and associated coolant, control, and protection systems are designed with adequate margins to: Preclude significant fuel damage during normal core operation and operational transients (Condition I)* or any transient conditions arising from occurrences of moderate frequency (Condition II)*.

"Ensure return of the reactor to a safe state following a Condition III* event with only a small fraction of fuel rods damaged although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

"Assure that the core is intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV)*."

NOTE: *Defined by ANSI N18.2-1973.

RESPONSE

Chapter 4.0 discusses the design bases and design evaluation of reactor components including the fuel reactor vessel internals and reactivity control systems. Details of the control and protection systems instrumentation design and logic are discussed in Chapter 7.0. This information supports the accident analyses of Chapter 15.0, which show that the acceptable fuel design limits are not exceeded for Condition I and II occurrences.

3.1.2.2.2 Evaluation Against Criterion 11 - Reactor Inherent Protection

"The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity."

RESPONSE

Prompt inherent nuclear feedback characteristics are assured in the reactor power operating range when considering the net effect of the reactivity coefficients for negative fuel temperature effect (Doppler defect) and the Technical Specifications limitation for maximum positive moderator coefficient as a function of reactor power. A negative Doppler coefficient is ensured by the use of low-enrichment fuel in the reactor. The Technical Specifications limitation for positive moderator temperature coefficient is ensured by administrative controls and limitations for dissolved neutron absorber concentration, burnable poisons, and control rod position limits.

These reactivity coefficients are discussed in Section 4.3.

3.1.2.2.3 Evaluation Against Criterion 12 - Suppression of Reactor Power Oscillations

"The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed."

RESPONSE

Power oscillations of the fundamental mode are inherently eliminated by the net effect of the reactivity coefficients for negative fuel temperature effect (Doppler defect) and the Technical Specifications limitation for maximum positive moderator coefficient as a function of reactor power. Oscillations due to xenon spatial effects in the radial, diametral, and azimuthal overtone modes are heavily damped due to the inherent design of the reactor, the nuclear fuel, and Technical Specifications operational limitations. Oscillations due to xenon spatial effects may occur in the axial first overtone mode. Oscillations due to xenon spatial effects in axial modes higher than the first overtone are also heavily damped due to the inherent design of the reactor.

Not exceeding fuel design limits by xenon axial oscillations is ensured by reactor trip functions, which use the measured axial power imbalance as an input.

Xenon stability control is discussed in Section 4.3.

Full-length control rods provide the capability of attenuating axial oscillations.

3.1.2.2.4 Evaluation Against Criterion 13 - Instrumentation and Control

"Instrumentation and control shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges."

RESPONSE

Instrumentation and controls are provided to monitor and control neutron flux, control rod position, temperatures, pressures, flows, and levels as necessary to assure that adequate plant safety can be maintained. Instrumentation is provided in the reactor coolant system, steam and power conversion system, the containment, engineered safety features systems, radiological waste systems and other auxiliaries. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the control room in proximity to the controls for maintaining the indicated parameter in the proper range.

The quantity and types of process instrumentation provided ensure safe and orderly operation of all systems over the full design range of the plant. These systems are described in Chapters 6.0 through 9.0 and in Chapters 11.0 and 12.0.

3.1.2.2.5 Evaluation Against Criterion 14 - Reactor Coolant Pressure Boundary

"The reactor coolant pressure boundary shall be designed, fabricated, and tested so as to have an extremely low probability of abnormal leakage or rapidly propagating failure, and of gross rupture."

RESPONSE

The reactor coolant system boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including all anticipated transients, and to maintain the stresses within applicable stress limits. See Sections 3.9 and 5.2 for details. Reactor coolant pressure boundary materials, selection, and fabrication techniques ensure a low probability of a gross break or significant leakage.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as a pipe break and seismic design as discussed in Sections 3.6 and 3.7. The system is protected from overpressure by means of pressure-relieving devices as required by applicable codes (see Subsection 5.2.2).

The reactor coolant system boundary has provisions for inspection, testing, and surveillance of critical areas to assess the structural and leaktight integrity. See Section 5.2 for details. For the reactor vessel, a materials surveillance program conforming to applicable codes is provided. See Section 5.3 for details.

3.1.2.2.6 Evaluation Against Criterion 15 - Reactor Coolant System Design

"The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

RESPONSE

The design pressure and temperature for each component in the reactor coolant and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

Additionally, reactor coolant pressure boundary components achieve a large margin of safety by the use of proven ASME materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and integrated hydrostatic testing of assembled components.

The effect of radiation embrittlement is considered in reactor vessel design, and surveillance samples monitor adherence to expected conditions throughout plant life.

Multiple safety and relief valves are provided for the reactor coolant system. The safety valves and their setpoints meet ASME criteria for overpressure protection. The ASME criteria have been shown to be satisfactory based on a long history of industry use. Chapter 5.0 discusses the reactor coolant system design.

Transient analyses are included in reactor coolant system design which conclude that design conditions are not exceeded during normal operation. Protection and control setpoints are based on these transient analyses. The design margin includes the effects of thermal lag, coolant transport times, pressure drops, system relief valve characteristics, and instrumentation and control response characteristics.

3.1.2.2.7 Evaluation Against Criterion 16 - Containment Design

"Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

RESPONSE

The containment design for the B/B Stations incorporates a post-tensioned concrete containment with a steel liner to enclose the nuclear steam supply system completely. The design criteria and methods of analysis for the containment structure are discussed in Subsection 3.8.1 and the functional design and testing provisions are described in Section 6.2.

3.1.2.2.8 Evaluation Against Criterion 17 - Electric Power Systems

"An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

"The onsite electric power sources, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

"Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

"Provisions shall be included to minimize the probability of losing electric power from any of the remaining sources as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power sources."

RESPONSE

Each unit of the Station has two separate diesel engine-driven generators to provide a-c electric power to two independent and redundant trains of engineered safety features. Each unit of the Station also has two separate batteries to provide d-c electric power to the two independent and redundant trains of engineered safety features.

The two offsite electric power system connections to the Stations are designed to provide access to a diversity of reliable power sources and are physically separate and electrically independent so that any single failure will affect only one supply and will not propagate to the alternate supply.

The preferred power system is considered as having three major sections, each of which must provide two physically separate and electrically independent circuit paths between the onsite power system and the transmission network (the transmission network excludes the station switchyard). The three sections are:

1. The transmission lines entering the station switchyard from the transmission network.

2. The station switchyard. (A common switchyard is allowed by GDC 17).

3. The overhead transmission lines, SATs, buses between the switchyard, and the onsite power system.

The Stations' auxiliary electric power system is designed to provide electrical isolation and physical separation of the redundant power supplies for station requirements which are important to nuclear safety. Means are provided for rapid location and isolation of system faults. Redundant loads important to plant safety are assigned to redundant and independent engineered safety feature system switchgear groups. A detailed discussion of these systems is presented in Chapter 8.0. The engineered safety features electrical systems are designed in accordance with IEEE Standards 279-1971 and 308-1974.

3.1.2.2.9 Evaluation Against Criterion 18 - Inspection and Testing of Electric Power Systems

"Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the conditions of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system."

RESPONSE

Provisions have been made in the design of offsite and onsite power systems for the inspection and testing of appropriate areas of the systems. Periodic tests can be made of major portions of the power sources and distribution systems under conditions simulating the design conditions. Analyses which demonstrate compliance with Criterion 18 are presented in Subsection 8.3.1.

3.1.2.2.10 Evaluation Against Criterion 19 - Control Room

"A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures”.

For accidents analyzed using alternative source term, radiation exposure limits are addressed in 10 CFR 50.67.

RESPONSE

The control room is designed following proven power plant design philosophy. All control stations, switches, controllers, and indicators necessary to operate and shut down the plant and to maintain safe control of the facility will be located in the control room.

The design of the control room permits safe occupancy during abnormal conditions. Shielding is designed to maintain tolerable radiation exposure levels in the control room under hypothetical accident conditions. The control room HVAC system provides the necessary environment for both the operators and the instrumentation. Refer to Subsections 6.4, 6.5.1, and 9.4.1 for details of system design. Makeup air to the control room system can be filtered through high efficiency particulate and charcoal filters, and provides pressurization to reduce the ingress of radioactive particles. The control room will be continuously occupied by qualified operating personnel under all operating and credible accident conditions.

Alternate local controls and instrumentation at locations outside the control room are provided to bring the plant to and maintain it in a hot shutdown condition. Cold shutdown from outside the control room is not contemplated. The control room has been designed to remain operable and habitable under extremely severe postulated events. Operators will not be forced to leave the control room under any credible circumstances.

Should it be postulated, through some nonmechanistic series of events, that control room evacuation is required, the plant can be brought to and maintained in a hot shutdown condition until control room conditions are restored. The plant could then be brought to cold shutdown or operation resumed, depending upon plant conditions, from the control room.

The potential capability for cold shutdown from outside the control room exists. Local controls on boration equipment and cooldown equipment are furnished. Some temporary control bypasses might be required to be used under procedural control for such an operation. Procedures for cold shutdown from outside the control room will not be normally available but would be developed if ever needed while the plant is in the hot shutdown condition since the details of such procedures would be dependent upon actual plant conditions at the time.

3.1.2.3 Group III - Protection and Reactivity Control Systems

3.1.2.3.1 Evaluation Against Criterion 20 - Protection System Functions

"The protection system shall be designed 1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and 2) to sense accident conditions and to initiate the operation of systems and components important to safety."

RESPONSE

A fully automatic protection system with appropriate redundant channels is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for all protection systems is in accordance with the intent of IEEE 279-1971 and IEEE 379-1972. The reactor protection system automatically initiates a reactor trip when any variable monitored by the system or combination of monitored variables exceeds the normal operating range. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the rod drive mechanisms of all the full-length rod cluster control assemblies. This causes the rods to insert by gravity, rapidly reducing the reactor power output. The response and adequacy of the protection system has been verified by analysis of anticipated transients.

The engineered safety features (ESF) actuation system automatically initiates emergency core cooling and other safeguards functions by sensing accident conditions using redundant analog channels measuring diverse variables. Manual actuation of safeguards may be performed where ample time is available for operator action. The ESF actuation system automatically trips the reactor on manual or automatic safety injection "S" signal generation.

3.1.2.3.2 Evaluation Against Criterion 21 - Protection System Reliability and Testability

"The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that 1) no single failure results in loss of the protection function, and 2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred."

RESPONSE

The protection system is designed for high functional reliability and inservice testability such that the requirements of Criterion 21 are satisfied.

Compliance with this criterion is discussed in detail in Subsections 7.2.2.2.3 and 7.3.2.2.5.

3.1.2.3.3 Evaluation Against Criterion 22 - Protection System Independence

"The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function."

RESPONSE

Protection system components are designed and arranged so that the environment accompanying any emergency situation in which the components are required to function does not result in loss of the safety function. Various means are used to accomplish this. Functional diversity has been designed into the system. The extent of this functional diversity has been evaluated for a wide variety of postulated accidents. Diverse protection functions will automatically terminate an accident before intolerable consequences can occur.

Automatic reactor trips are based upon neutron flux measurements, reactor coolant loop temperature measurements, pressure and level measurements, reactor coolant loop flow measurements, reactor coolant pump power underfrequency and undervoltage measurements, and steam generator level and pressure measurements. Trips may also be initiated manually or by safety injection signal. See Section 7.2 for details of the reactor trip system and Section

7.3 for details of the engineered safety features actuation system.

High quality components, conservative design and applicable quality control, inspection, calibration, and tests are utilized to guard against common-mode failure. Qualification testing is performed on the various safety systems to demonstrate functional operation at normal and postaccident conditions of temperature, humidity, pressure, and radiation for specified periods if required. Typical protection system equipment is subjected to type tests under simulated seismic conditions using conservatively large accelerations and applicable frequencies. The test results indicate no loss of the protection function. Refer to Section 3.10 for further details.

3.1.2.3.4 Evaluation Against Criterion 23 - Protection System Failure Modes

"The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced."

RESPONSE

The protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. Each reactor trip channel is designed on the deenergize-to-trip principle so that loss of power, disconnection, open-channel faults, and the majority of internal channel short-circuit faults cause the channel to go into its tripped mode. The protection system is discussed in Sections 7.2 and 7.3.

3.1.2.3.5 Evaluation Against Criterion 24 - Separation of Protection and Control Systems

"The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired."

RESPONSE

The protection system is separate and distinct from the control systems. Control systems may be dependent on the protection system in that control signals are derived from protection system measurements where applicable. These signals are transferred to the control system by isolation devices which are classified as protection components. The adequacy of system isolation has been verified by testing under conditions of postulated credible faults and conditions of credible interference on the control wiring in close proximity to the protection wiring in the racks. The failure of any single control system component or channel or the failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system which satisfies the requirements of the protection system. Distinction between channel and train is made in this discussion. The removal of a train from service is allowed only during testing of the train.

3.1.2.3.6 Evaluation Against Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

"The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods."

RESPONSE

The protection system is designed to limit reactivity transients so that fuel design limits are not exceeded. Reactor shutdown by full-length rod insertion is completely independent of the normal control function, since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. Thus, in the postulated accidental withdrawal (assumed to be initiated by a control malfunction), flux, temperature, pressure, level and flow signals would be generated independently. Any of these signals (trip demands) would operate the breakers to trip the reactor.

Analyses of the effects of possible malfunctions are discussed in Chapter 15.0. These analyses show that for postulated dilution during refueling, startup, or manual or automatic operation at power, the operator has ample time to determine the cause of dilution, terminate the source of dilution, and initiate reboration before the shutdown margin is lost. The analyses show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

3.1.2.3.7 Evaluation Against Criterion 26 - Reactivity Control System Redundancy and Capability

"Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions."

RESPONSE

Two reactivity control systems are provided. These are rod cluster control assemblies (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation the shutdown rod banks are fully withdrawn. The full-length control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks along with the full-length control banks are designed to shut down the operation with adequate margin under conditions of normal operation and anticipated operational occurrences, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses, and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The boron system will maintain the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCAs are presented in Chapter 4.0; their operation is discussed in Chapter 7.0. The means of controlling the boric acid concentration is described in Chapter 9.0. Performance analyses under accident conditions are included in Chapter 15.0.

3.1.2.3.8 Evaluation Against Criterion 27 - Combined Reactivity Control System Capability

"The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained."

RESPONSE

The facility is provided with means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. These means are discussed in detail in Chapters 4.0 and 9.0. Combined use of the rod cluster control system and the chemical shim control system permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. The single highest worth control cluster is assumed to be stuck full-out upon trip for this determination.

3.1.2.3.9 Evaluation Against Criterion 28 – Reactivity Limits

"The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither: 1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor 2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line break, changes in reactor coolant temperature and pressure, and cold water addition."

RESPONSE

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values that prevent a break of the reactor coolant system boundary or disruptions of the core vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The maximum positive reactivity insertion rates for the withdrawal of rod cluster control assemblies (RCCAs) and the dilution of the boric acid in the reactor coolant system are limited by the physical design characteristics of the RCCAs and of the chemical and volume control system. Technical specifications on shutdown margin and on RCCA insertion limits and bank overlaps as functions of power provide additional assurance that the consequences of the postulated accidents are no more severe than those presented in the analyses of Chapter 15.0. Reactivity insertion rates, dilution, and withdrawal limits are also discussed in Section 4.3. The capability of the chemical and volume control system to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 9.0.

Assurance of core cooling capability following Condition IV accidents, such as ejections, steamline break, etc., is given by keeping the reactor coolant pressure boundary stresses within faulted condition limits as specified by applicable ASME codes.

Structural deformations are also checked and limited to values that do not jeopardize the operation of necessary safety features.

3.1.2.3.10 Evaluation Against Criterion 29 - Protection Against Anticipated Operational Occurrences

"The protection and reactivity control system shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences."

RESPONSE

The protection and reactivity control systems are designed to assure extremely high probability of performing their required safety functions in any anticipated operational occurrences. Likely failure modes of system components are designed to be safe modes. Equipment used in these systems is designed, constructed, operated, and maintained with a high level of reliability. Loss of power to the protection system results in a reactor trip. Details of system design are covered in Chapter 7.0. Also refer to the discussions of Criteria 20 through 25.

3.1.2.4 Group IV - Fluid Systems

3.1.2.4.1 Evaluation Against Criterion 30 - Quality of Reactor Coolant Pressure Boundary

"Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage."

RESPONSE

Reactor coolant pressure boundary components are designed, fabricated, inspected and tested in conformance with ASME Boiler and Pressure Vessel Code, Section III. All components are classified according to ANSI N18.2-1973 and are accorded the quality measures appropriate to the classification. The design bases and evaluations of reactor coolant pressure boundary components are discussed in Chapter 5.0.

Leakage is detected by an increase in the amount of makeup water required to maintain a normal level in the pressurizer. The reactor vessel closure joint is provided with a temperature monitored leakoff between double gaskets. Leakage inside the reactor containment is drained to the reactor containment sump where it is monitored. Leakage is also detected by measuring the airborne activity in the reactor containment. Containment dry-bulb temperatures and pressure also provide indirect indication of leakage to the containment.

The reactor coolant pressure boundary leakage detection system is described in Subsection 5.2.5.

3.1.2.4.2 Evaluation Against Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

"The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions 1) the boundary behaves in a non-brittle manner, and 2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining 1) material properties, 2) the effects of irradiation on material properties, 3) residual, steady-state and transient stresses, and 4) size of flaws."

RESPONSE

Close control is maintained over material selection and fabrication for the reactor coolant system to assure that the boundary behaves in a non-brittle manner. The reactor coolant system materials which are exposed to the coolant are corrosion resistant stainless steel or Inconel. The nil ductility transition reference temperature (RT_{NDT}) of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests in accordance with 10 CFR 50, Appendix G, fracture toughness requirements. These tests ensure the selection of materials with adequate toughness properties and margins.

As part of the reactor vessel specification certain requirements which are not specified by the applicable ASME Codes are performed, as follows:

a. Ultrasonic testing - In addition to the straight beam code requirements, the performance of a 100% volumetric angle beam inspection of reactor vessel plate material and a post-hydro test ultrasonic map of all full-penetration welds in the pressure vessel are required. Cladding bond ultrasonic inspection to more restrictive requirements than those specified in the code are also required to preclude interpretation problems during inservice inspection.

b. Radiation Surveillance Program - In the surveillance programs, the evaluation of the radiation damage is based on preirradiation testing of Charpy V-notch and tensile specimens and postirradiation testing of Charpy V-notch, tensile, and 1/2 T (thickness) compact tension fracture mechanics test specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM E-195-1973, "Recommended Practice for Surveillance Tests for

Nuclear Reactor Vessels," and the requirements of 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements.

c. Reactor vessel core region material chemistry (copper, phosphorous and vanadium) is controlled to reduce sensitivity to embrittlement due to irradiation over the life of the plant.

The fabrication and quality control techniques used in the fabrication of the reactor coolant system are equivalent to those used for the reactor vessel. The inspections of reactor vessel, pressurizer, piping, pumps, and steam generator are governed by ASME code requirements. See Chapter 5.0 for details.

Allowable pressure-temperature relationships for plant heatup and cooldown rates are calculated using methods presented in the ASME Code, Section III, Appendix G, "Protection Against Non-Ductile Failure." The approach specifies that allowed stress intensity factors for all vessel operating conditions shall not exceed the reference stress intensity factor (K_{IR}) for the metal temperature at any time. Operating specifications include conservative margins for predicted changes in the material nil ductility transition reference temperatures (RT_{NDT}) due to irradiation.

3.1.2.4.3 Evaluation Against Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

"Components which are part of the reactor coolant pressure boundary shall 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."

RESPONSE

The design of the reactor coolant pressure boundary provides the capability for accessibility during service life to the entire internal surfaces of the reactor vessel, certain external zones of the vessel, including the nozzle to reactor coolant piping welds and the top and bottom heads, and external surfaces of the reactor coolant piping except for the area of pipe within the primary shielding concrete. The inspection capability complements the leakage detection systems in assessing pressure boundary component integrity. The reactor coolant pressure boundary will be periodically inspected under the provisions of the ASME Boiler and Pressure Vessel Code, Section XI.

Monitoring of changes in the fracture toughness properties of the reactor vessel core region plates, forgings, weldments and associated heat treated zones are performed in accordance with 10 CFR 50 Appendix H. Samples of reactor vessel plate materials are retained and cataloged should future engineering development show the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The observed shifts in RT_{NDT} of the core region materials with irradiation will be used to confirm the allowable limits calculated for all operational transients.

See the appropriate sections in Chapter 5.0 for further details on inspection and surveillance requirements.

3.1.2.4.4 Evaluation Against Criterion 33 - Reactor Coolant Makeup

"A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation."

RESPONSE

The chemical and volume control system provides a means of reactor coolant makeup to ensure appropriate makeup supply for small breaks as described in Subsection 9.3.4 and adjustment of the boric acid concentration. Makeup is added automatically if the level in the volume control tank falls below the preset level. The high-pressure centrifugal charging pumps provided are capable of supplying the required makeup and reactor coolant seal injection flow when power is available from either onsite or offsite electric power systems. These pumps also serve as high-head safety injection pumps. Functional reliability is assured by provision of standby components assuring a safe response to probable modes of failure. Details of system design are included in Section 6.3, with details of the electric power system included in Chapter 8.0.

3.1.2.4.5 Evaluation Against Criterion 34 - Residual Heat Removal

"A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

"Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

RESPONSE

The residual heat removal (RHR) system in conjunction with the steam and power conversion system, is designed to transfer the fission production decay heat and other residual heat from the reactor core within acceptable limits. The crossover from the steam and power conversion system to the residual heat removal system occurs at approximately 350°F and 360 psig.

Suitable redundancy at temperatures below approximately 350°F is accomplished with the two residual heat removal pumps (located in separate compartments with means available for draining and monitoring of leakage); the two heat exchangers; and the associated piping; cabling, and electric power sources. The residual heat removal system is capable of operating on either onsite or offsite electrical power.

Suitable redundancy at temperatures above approximately 350°F is provided by the four steam generators and attendant piping. Details of the system designs are given in Sections 5.4 and 9.2 and Chapter 10.0.

3.1.2.4.6 Evaluation Against Criterion 35 - Emergency Core Cooling

"A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

"Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

RESPONSE

An emergency core cooling system is provided to cope with any loss-of-coolant accident due to a pipe break. Abundant cooling water is available in an emergency to transfer heat from the core at a rate sufficient to maintain the core in a coolable geometry and to assure that cladding metal-water reaction is limited to less than 1%. Adequate design provisions are made to assure performance of the required safety functions even with a single failure.

Details of the capability of the systems are included in Section 6.3. An evaluation of the adequacy of the system functions is included in Chapter 15.0. Performance evaluations will be conducted in accordance with 10 CFR 50.46 and Appendix K.

3.1.2.4.7 Evaluation Against Criterion 36 - Inspection of Emergency Core Cooling System

"The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure integrity and capability of the system."

RESPONSE

Design provisions facilitate access to the critical parts of the injection nozzles, pipes, and valves for visual inspection for nondestructive inspection where such techniques are desirable and appropriate. The design is in accordance with ASME Section XI requirements.

The components outside the containment are accessible for leaktightness inspection during operation of the reactor.

Details of the inspection program of the emergency core cooling system are discussed in Subsection 6.3.4.

3.1.2.4.8 Evaluation Against Criterion 37 - Testing of Emergency Core Cooling System

"The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system."

RESPONSE

Each active component of the emergency core cooling system (ECCS) may be individually actuated on the normal power source or transferred to the emergency power source at any time during appropriate plant periodic tests.

Tests may be performed during shutdown to demonstrate proper automatic operation of the ECCS and an integrated system test can be performed during the late stages of reactor coolant system cooldown.

The details of the ECCS are included in Subsection 6.3.4.

3.1.2.4.9 Evaluation Against Criterion 38 - Containment Heat Removal

"A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

"Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

RESPONSE

Two diverse heat removal systems, each composed of redundant components, are provided: the containment spray system (two 100% capacity pumping systems) and the containment fan cooler system (four units provided, two required for accident heat removal). The containment spray system is described in Subsection 6.5.2 and the containment fan coolers are described in Subsection 6.2.2. The electric power provisions are described in Chapter 8.0.

3.1.2.4.10 Evaluation Against Criterion 39 - Inspection of Containment Heat Removal System

"The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system."

RESPONSE

The containment spray system's integrity will be verified by means of periodic testing and inspection as described in Subsection 6.5.2.4. Access has been provided for routine maintenance and inspections except for the spray ring headers. No provision has been made for inspecting the spray ring headers and nozzles. Provisions have been made to test these headers and nozzles periodically using smoke or other suitable means.

The fan cooler units are continually operating and provide continuous verification of operability and integrity. Access for routine maintenance and inspections has been provided as described in Subsection 6.2.2.4. The provisions for inspection and testing of the electric power supply for these systems are described in Chapter 8.0.

3.1.2.4.11 Evaluation Against Criterion 40 - Testing of Containment Heat Removal System

"The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system."

RESPONSE

The containment spray system's integrity will be verified by means of periodic miniflow and valve actuation testing as described in Subsection 6.5.2.4. Provisions have been made to test the spray ring headers and nozzles periodically. The fan cooler units are used in normal operation for continuous verification of operability and integrity.

The testing program and provisions for the normal and emergency power sources and the transfer capabilities will verify proper operation and are described in Chapter 8.0. The testing program and provisions for the essential service water system are described in Chapter 9.0. This system is used in normal operation, thereby verifying operability.

3.1.2.4.12 Evaluation Against Criterion 41 - Containment Atmosphere Cleanup

"Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

"Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure."

RESPONSE

The containment spray system is provided to control the concentration and quality of fission products in the containment following postulated accidents. The containment spray system consists of two independent subsystems, each supplied from separate ESF buses. The containment atmosphere mixing function of the combustible gas control system prevents local accumulation of combustible or detonable gases that could threaten containment integrity or equipment operating in a local compartment following a loss-of-coolant accident. The containment atmosphere mixing function is discussed in Subsection 6.2.5.2.3. The electric power provisions are described in Chapter 8.0.

3.1.2.4.13 Evaluation Against Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems

"The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems."

RESPONSE

The containment spray, combustible gas control, and post-LOCA purge systems are designed to permit periodic inspection of vital components as described in Subsections 6.5.2, 6.2.5, and 9.4.9, respectively. Access is provided to all active components for inspection and maintenance.

3.1.2.4.14 Evaluation Against Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

"The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems."

RESPONSE

The containment spray, combustible gas control, and post-LOCA purge systems are designed to permit periodic testing of vital components, as described in Subsections 6.2.5, 6.5.2, and 9.4.9. The containment spray system is also discussed in the response to Criterion 40. The containment spray system can be checked for leaktightness during miniflow testing. The hydrogen recombiners are designed to accommodate thermal testing. Provisions have been made for periodic pressure and functional testing for the combustible gas control system. The nonsafety grade post-LOCA purge system has provisions for appropriate testing to assure operability and integrity. Testability of the power sources is described in Chapter 8.0.

3.1.2.4.15 Evaluation Against Criterion 44 - Cooling Water

"A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

"Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

RESPONSE

The component cooling and essential service water systems provide appropriate cooling capacity for structures, systems, and components important to safety, and are designed with appropriate redundancy. A single failure can be accommodated without impairing the safety function of the systems. Appropriate leak detection capability is provided. These systems are described in Subsections 9.2.1 and 9.2.2. Electric power for the operation of each system may be supplied from offsite power sources, and as described in Chapter 8.0.

3.1.2.4.16 Evaluation Against Criterion 45 - Inspection of Cooling Water System

"The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system."

RESPONSE

The component cooling and essential service water system are designed to permit periodic inspection. Manholes, hand holes, inspection ports, and other design features are provided to facilitate inspection.

Leakage detection equipment is provided to monitor the integrity of the systems (Section 9.2).

For further information, see the following:

a. General Plant Description, Section 1.2;

b. Water Systems, Section 9.2; and

c. Initial Test Program, Chapter 14.0.

3.1.2.4.17 Evaluation Against Criterion 46 - Testing of Cooling Water System

"The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources."

RESPONSE

In addition to the provisions noted in the response to Criterion 45, provisions will be made for testing the actuation of the systems from both normal and emergency power sources, as described in Chapters 8.0 and 9.0.

3.1.2.5 Group V - Reactor Containment

3.1.2.5.1 Evaluation Against Criterion 50 – Containment Design-Basis

"The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservation of the calculational model and input parameters."

RESPONSE

The containment structure, including access openings and penetrations, is designed to withstand the peak accident pressure and temperature that could occur during the postulated design-basis loss-of-coolant accident. In addition to incorporating appropriate safety factors into this design, considerable allowances are also included for energy addition from sources which may not have been included in the postulated accident and for the limited experience in defining containment response.

For further details and discussion, see Subsections 3.8.1 and 6.2.1.

3.1.2.5.2 Evaluation Against Criterion 51 - Fracture Prevention of Containment Pressure Boundary

"The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws."

RESPONSE

The containment's ferritic materials are selected to ensure that their temperature under normal operating and testing conditions is at least 30°F above nil ductility transition temperature (NDTT). Detailed stress analyses have been made of the containment liner and liner anchors under normal and postulated accident conditions. Code allowable material discontinuities have been considered.

Further details regarding ferritic materials used in the containments and their design requirements are discussed in Section 3.8 and Subsections 6.1.1 and 6.2.1.

3.1.2.5.3 Evaluation Against Criterion 52 - Capability for Containment Leak Rate Testing

"The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure."

RESPONSE

The provisions for testing in conformation with this criterion, and the provisions for conformance with Appendix J to 10 CFR 50, dated August 25, 1971, are discussed in Subsection 6.2.6.1.

3.1.2.5.4 Evaluation Against Criterion 53 - Provisions for Containment Testing and Inspection

"The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows."

RESPONSE

The reactor containment design permits access to penetrations and other important areas for implementation of the surveillance program which is described in the Technical Specifications. Penetrations and resilient seals and bellows will be visually inspected and pressure tested for leaktightness periodically, according to the Technical Specifications.

3.1.2.5.5 Evaluation Against Criterion 54 - Piping Systems Penetrating Containment

"Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits."

RESPONSE

Piping systems penetrating the containment are equipped with isolation valving which provides an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and limits the leakage to within the applicable requirements of 10 CFR 20, 10 CFR 50 Appendix I, and 10 CFR 50.67. The types of valving used are discussed in Subsections 3.1.2.5.6, 3.1.2.5.7, and 3.1.2.5.8. A detailed discussion of containment isolation is given in Subsection 6.2.4. Instrument lines penetrating the containment are discussed in Chapter 7.0. Isolation of instrument lines complies with the intent of Regulatory Guide 1.11.

Test connections and pressurizing means are provided to test isolation valves for leaktightness. A detailed discussion of leakage testing is given in Subsection 6.2.6.

Containment isolation may be operator initiated or actuated via the containment and reactor coolant system leak detection and radiation monitoring provisions. A detailed discussion is given in Chapter 7.0.

Containment isolation capabilities regarding redundancy, reliability and performance are discussed in Subsection 6.2.4.

3.1.2.5.6 Evaluation Against Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

"Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

"Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

"Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs."

RESPONSE

Portions of the reactor coolant pressure boundary which penetrate the containment are provided with isolation valving of one of the following types:

a. One locked closed isolation valve inside and one locked closed isolation valve outside;

b. One automatic isolation valve inside and one locked closed isolation valve outside;

c. One locked closed isolation valve inside and one automatic isolation valve outside (a simple check valve will not be used as the automatic isolation valve outside); and

d. One automatic isolation valve inside and one automatic isolation valve outside (a simple check valve will not be used as the automatic isolation valve outside).

The valving used in a particular line depends on the lines functional typification which is defined and discussed in Subsection 6.2.4.

Isolation valves outside the containment are located as close to the containment wall as practical. Automatic isolation valves fail to the position of greatest safety. Valve locations and fail modes are discussed in detail in Subsection 6.2.4.

The appropriate design requirements which minimize the probability or consequence of a break of these lines or lines connected to them are discussed in Chapters 3.0 and 5.0.

3.1.2.5.7 Evaluation Against Criterion 56 - Primary Containment Isolation

"Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

"Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety."

RESPONSE

Lines which connect directly to the containment atmosphere and penetrate the containment are considered open systems within the containment and are equipped with isolation valving of one of the following types:

a. One locked closed isolation valve inside and one locked closed isolation valve outside;

b. One automatic isolation valve inside and one locked closed isolation valve outside;

c. One locked closed isolation valve inside and an automatic isolation valve outside (a simple check valve will not be used as the automatic isolation valve outside); and

d. One automatic isolation valve inside and one automatic isolation valve outside (a simple check valve will not be used as the automatic isolation valve outside).

The valving used in a particular line depends on the lines functional typification which is defined and discussed in Subsection 6.2.4.

Isolation valves outside the containment are located as close to the containment wall as practical. Automatic isolation valves are designed to fail in the position of greatest safety. Valve locations and fail modes are discussed in detail in Subsection 6.2.4.

3.1.2.5.8 Evaluation Against Criterion 57 - Closed System Isolation Valves

"Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as is practical. A simple check valve may not be used as the automatic isolation valve."

RESPONSE

Lines which penetrate the containment and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere are considered closed systems within the containment and are equipped with at least one containment isolation valve of one of the following types:

a. An automatic isolation valve (a simple check valve will not be used as this automatic valve);

b. A locked closed valve; or

c. A valve capable of remote manual operation.

This valve is located outside the containment and as close to the containment wall as is practical. Valve locations are discussed in detail in Subsection 6.2.4.

3.1.2.6 Group VI - Fuel and Radioactivity Control

3.1.2.6.1 Evaluation Against Criterion 60 - Control of Releases Radioactive Materials to the Environment

"The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment."

RESPONSE

An extensive system using demineralizers, filters and monitoring devices has been designed for liquid waste treatment and disposal. Gaseous wastes are processed by appropriate holdup. Solid wet wastes are processed in 55-gallon drums or in liners for eventual disposal in licensed burial grounds. Dry active wastes are placed in 55-gallon drums, C-vans, or liners and shipped to licensed burial grounds for disposal. The systems are sized to accommodate anticipated operational occurrences. The liquid waste management systems are discussed in Section 11.2 of this

UFSAR, the gaseous waste management system is discussed in Section 11.3, and the solid waste management system is discussed in Section 11.4.

3.1.2.6.2 Evaluation Against Criterion 61 - Fuel Storage and Handling and Radioactivity Control

"The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions."

RESPONSE

The fuel storage and handling systems and the radwaste systems are designed to conform to the intent of this criterion. Surveillance of safety-related items is accomplished by virtue of routine monitoring of the day-to-day operations of the systems.

Appropriate shielding, filtration, heat removal, and water inventory control provisions have been made for these systems. The spent fuel pool is so designed that no postulated accident could cause excessive loss of water inventory.

The design measures necessary to meet this criterion are described in Section 9.1 and Chapter 11.0 for the fuel storage and handling systems and the radwaste systems, respectively. Dry Cask Storage (DCS) is discussed in section 9.1.2.3.11.

3.1.2.6.3 Evaluation Against Criterion 62 - Prevention of Criticality in Fuel Storage and Handling

"Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

RESPONSE

Geometrically safe configurations are employed to preclude criticality in new and spent fuel storage facilities. The design measures necessary to conform to the intent of this criterion are described in Section 9.1.

3.1.2.6.4 Evaluation Against Criterion 63 - Monitoring Fuel and Waste Storage

"Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions."

RESPONSE

Monitoring systems are provided to alarm on excessive temperature or low-water level in the spent fuel pool. Radiation monitors and alarms are provided to warn personnel of an increase in the level of radiation. The design measures are described in Section 9.1 and Chapters 11.0 and 12.0.

3.1.2.6.5 Evaluation Against Criterion 64 - Monitoring Radioactivity Releases

"Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents."

RESPONSE

The reactor containment atmosphere is continuously sampled and monitored for radioactivity that may be released from normal operations (including anticipated operational occurrences) by a continuous air monitoring system that is located outside of the containment. A separate system is provided to enable collection and analysis of grab samples of the containment atmosphere during normal operations and accident conditions. GM type area monitors and high range ion chambers are provided in containment to measure radiation levels from normal and accident conditions.

Spaces containing components for recirculation of loss-of-coolant accident fluids and areas contiguous to the containment structure are monitored for airborne radioactivity by systems that sample and monitor the air exhausted from the associated areas. The systems consist of continuous air monitors, duct radiation monitors, and air samplers to enable collection of samples of exhaust air for laboratory analysis during normal and accident conditions.

Effluent discharge paths from the facility are continuously monitored for radioactivity during normal operations with continuous air and liquid monitoring systems. Sampling provisions are included to allow sample collection for analysis during normal operations and accident conditions. Extended range noble gas monitors are provided to allow continuous monitoring during accident conditions. A comprehensive environs monitoring program is provided to assess radioactivity releases. The systems provided for monitoring radioactive releases from the facility are described in Chapters 11.0 and 12.0. The environs monitoring program is described in Chapter 11.0, in the Offsite Dose Calculation Manual, and in the Environmental Report.

3.1.3 References

1. Exelon Generation Company, "Byron/Braidwood Stations Fire Protection Report in Response to Appendix A of BTP APCSB 9.5-1" (current amendment).

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 Safety Classification

Structures, systems, and components are classified for design purposes as either Safety Category I or Safety Category II. Piping and instrumentation diagrams (P&IDs) provided in this UFSAR show boundaries of classification, i.e., classification changes, on applicable drawings.

3.2.1.1 Safety Category I

Those structures, systems, and components important to safety that are designed to remain functional in the event of the safe shutdown earthquake (SSE) and other design-basis events (including tornado, probable maximum flood, operating basis earthquake - OBE, missile impact, or accident internal to the plant) are designated as Safety Category I. This category includes those structures, systems, and components whose safety function is to retain their own integrity and/or not constitute a hazard to the safety function of other Safety Category I structures, systems, and components.

Safety Category I structures, systems, and components are those necessary to assure:

a. the integrity of the reactor coolant pressure boundary,

b. the capability to shut down the reactor and maintain it in a safe shutdown condition, or

c. the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100 for accidents analyzed using TID-14844 and 10 CFR 50.67 for accidents analyzed using Regulatory Guide 1.183 (AST).

Safety Category I systems and components are not located within Safety Category II structures, except as noted in Table 3.2-1. Table 3.2-1 lists all plant structures and major components which are designated as Safety Category I. Systems or portions of systems, including piping, which are designated as Safety Category I are identified on the system P&IDs associated with that system. The division between Safety Category I and II portions of systems is in accordance with the intent of the requirements for seismic design classification.

Safety Category I systems or portions of systems and components meet the requirements of Appendix B to 10 CFR 50.

3.2.1.2 Safety Category II

Those structures, systems, and components which are not designated as Safety Category I are designated as Safety Category II. This category has no public health or safety implication.

Safety Category II structures, systems, and components are not specifically designed to remain functional in the event of the safe shutdown earthquake (SSE) or other design-basis events (including tornado, probable maximum flood, operating basis earthquake, missile impact, or an accident internal to the plant). A reasonable margin of safety is, however, considered in the design as dictated by local requirements. Many Safety Category II items in Category I buildings are supported with seismically designed supports. These items and their supports are not Safety Category I or Seismic Category I as defined by Regulatory Guide 1.29. Structures and major components not listed in Table 3.2-1 as Safety Category I are Safety Category II. Safety Category II systems or portions of systems and components do not follow the requirements of Appendix B to 10 CFR 50. The quality assurance standards for these systems and components follow normal industrial standards and any other requirements deemed necessary by the Licensee.

3.2.2 Quality Group Classification

The quality group classification is followed in which five quality groups (A, B, C, D, and G) are identified.

The following data indicates the overall correspondence between safety categories and quality groups and the general boundaries of systems to be considered part of each quality group.

QUALITY GROUP	SAFETY CATEGORY	GENERAL SYSTEM DESCRIPTION
A	I	Reactor coolant pressure boundary and extensions thereof.
B	I	Emergency core cooling, post-LOCA heat removal and cleanup, safe reactor shutdown and heat removal, portions of main steam and feedwater associated with containment isolation.
C	I	Cooling water and auxiliary feedwater systems or portions thereof that are designed for emergency core cooling; postaccident containment

QUALITY GROUP	SAFETY CATEGORY	GENERAL SYSTEM DESCRIPTION
		cooling; postaccident containment atmosphere cleanup; and residual heat removal from the reactor and spent fuel storage (including primary and secondary cooling systems).
C	I	Cooling water and seal water systems or portions of these systems that provide support for other systems and components important to safety.
C	I	Systems or portions of systems that are capable of being isolated from the reactor coolant pressure boundary during all modes of normal reactor operation by two valves which are either normally closed or capable of automatic closure.
C	I	Portions of the radioactive waste and other systems whose postulated failure would release radioactive isotopes that would result in a calculated offsite dose in excess of 0.5 rem to the whole body or its equivalent part (refer to Table 3.2-1).
D	II	Systems designed to ANSI B31.1.0 code criteria.
G	I	Safety Category I piping and/or components, non-ASME or under the jurisdiction of other codes. Diesel generator skid-mounted components, originally constructed to Safety Category I, Quality Group C requirements and reclassified in FSAR Amendment 49, are included in this classification (Byron only).

Table 3.2-1 lists all the Safety Category I components and the major Safety Category II components with their respective quality group classifications.

Table 3.2-2 identifies the applicable codes and standards used for each quality group.

Components which are not assigned specific quality group classifications are designed in accordance with normal industrial standards and good engineering practices.

Table 3.2-3 cross-references the ANS safety classifications to the classification designations established for Byron/Braidwood.

3.2.3 Risk Informed Categorization and Treatment

3.2.3.1 Introduction

As delineated in the Byron and Braidwood Station, Unit 1 and Unit 2 Operating Licenses (Reference 1), Byron and Braidwood have been approved to implement 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants," using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC- 3, and RISC-4 structures, systems, and components (SSCs) using:

> Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (AN0-2) passive categorization method to assess passive component risk for Class 2, Class 3, and non-Code class SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009.
>
> Exelon will complete the updated implementation items listed in Attachment 1 of Exelon letter to NRC dated September 13, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems and components (SSCs) subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on a method of categorizing SSCs according to their safety significance. For SSCs that are categorized as high safety-significant (HSS), existing treatment requirements are maintained or potentially enhanced. On the other hand, for SSCs categorized as low safety-significant (LSS) that do not significantly contribute to plant safety on an individual basis, the regulation allows an alternative risk-informed approach to treatment that provides a reasonable, although reduced, level of confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve focus on equipment that has high safety-significance, resulting in improved plant safety.

A risk-informed categorization process is employed to determine the safety-significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories, which are defined in 10 CFR 50.69(a), as follows:

RISC-1: Safety-related SSCs that perform safety-significant functions

RISC-2: Nonsafety-related SSCs that perform safety-significant functions

RISC-3: Safety-related SSCs that perform LSS functions

RISC-4: Nonsafety-related SSCs that perform LSS functions

3.2.3.2 SSC Categorization

The processes for categorization of RISC-1, RISC-2, RISC- 3, and RISC-4 SSCs is as outlined in the Byron and Braidwood Station, Unit 1 and Unit 2 Operating Licenses.

10 CFR 50.69 (f)(2) requires updating the UFSAR to reflect which systems have been categorized. The following table is revised as part of the periodic UFSAR update to reflect systems that have been categorized.

System Name	**System Designator**
None	N/A

3.2.3.3 SSC Treatment

3.2.3.3.1 Treatment of Component Categories

The programs or processes that implement the special treatment requirements are revised to recognize that the special treatments no longer apply to RISC-3 and RISC-4 SSCs. The programs or processes either allow continued application of the special treatments or acceptable alternative treatments, as applicable, to provide reasonable confidence that these SSCs would perform their safety-related function under design basis conditions.

For those components that are categorized as Low Safety Significant, 10 CFR 50.69 (b)(1) allows compliance with alternative requirements in lieu of the following special treatment requirements.

(i) 10 CFR part 21.

(ii) The portion of 10 CFR 50.46a(b) that imposes requirements to conform to Appendix B to 10 CFR part 50.

(iii) 10 CFR 50.49.

(iv) 10 CFR 50.55(e).

(v) The in-service testing requirements in 10 CFR 50.55a(f):

The in-service inspection and repair and replacement (with the exception of fracture toughness), requirements for ASME Class 2 and Class 3 SSCs in 10 CFR 50.55a(g); and the electrical component quality and qualification requirements in Section 4.3 and 4.4 of IEEE 279, and Sections 5.3 and 5.4 of IEEE 603-1991, as incorporated by reference in 10 CFR 50.55a(h).

(vi) 10 CFR 50.65, except for paragraph (a)(4).

(vii) 10 CFR 50.72.

(viii) 10 CFR 50.73.

(ix) Appendix B to 10 CFR part 50.

(x) The Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR part 50, for penetrations and valves meeting the following criteria:

(A) Containment penetrations that are either 1-in. nominal size or less, or continuously pressurized.

(B) Containment isolation valves that meet one or more of the following criteria:

(1) The valve is required to be open under accident conditions to prevent or mitigate core damage events;

(2) The valve is normally closed and in a physically closed, water-filled system;

(3) The valve is in a physically closed system whose piping pressure rating exceeds the

containment design pressure rating and is not connected to the reactor coolant pressure boundary; or

(4) The valve is 1-in. nominal size or less.

(xi) Appendix A to part 100, Sections VI(a)(1) and VI(a)(2), to the extent that these regulations require qualification testing and specific engineering methods to demonstrate that SSCs are designed to withstand the safe shutdown earthquake and operating basis earthquake.

Performance monitoring is being performed on all RISC-1, RISC-2, RISC-3, and RISC-4 SSCs and the station adjusts, as necessary, to either the categorization or treatment process so that the categorization process and results are maintained valid.

3.2.3.3.2 Enhanced Treatment of RISC-2 SSCs

10 CFR 50.69(d)(1) requires that the licensee ensure that RISC-1 and RISC-2 SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance.

3.2.4 References

1. Letter from J.S. Wiebe (NRC) to B.C. Hanson (Exelon Generation Company, LLC), "Braidwood 1 & 2, Byron 1 & 2 - Issuance of Amendments Nos. 198, 198, 204, and 204, Respectively, Regarding Adoption of Title 10 of the Code of Federal Regulations Section 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors' (CAC Nos. MG0201, MG0202, MG0203, and MG204; EPID L-2017-LLA-0285)," dated October 22, 2018.

TABLE 3.2-1

SAFETY CATEGORY AND QUALITY GROUP

CLASSIFICATION FOR STRUCTURES AND COMPONENTS

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
I. STRUCTURES			
1. Major Structures			
a. Reactor Containment, including penetrations	I	N/A	N/A
b. Reactor Containment Interior Structures	I	N/A	N/A
c. Auxiliary Building	I	N/A	N/A
d. Fuel Handling Building	I	N/A	N/A
e. Turbine Building	II	N/A	N/A
f. Solid Radwaste Storage Building	II	N/A	N/A
g. Service Building	II	N/A	N/A
h. Circulating Water Pump House	II	N/A	N/A
i. Main Steam Tunnels	I	N/A	N/A
j. River Screen House - Byron[1]	I	N/A	N/A
- Braidwood	II	N/A	N/A
k. Essential Service Water Discharge (Braidwood only)	I	N/A	N/A
l. Lake Screen House (superstructure) (Braidwood only)[2]	II	N/A	N/A
m. Lake Screen House (basemat, walls, and floor at El. 602 ft) (Braidwood only)	I	N/A	N/A
n. Essential Service Water Cooling Towers (Byron only)	I	N/A	N/A
o. Natural Draft Cooling Towers (Byron only)	II	N/A	N/A
p. Deep Well Enclosure (Byron only)	I	N/A	N/A
q. Old Steam Generator Storage Facility	II	N/A	N/A

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
2. Other Structures			
a. Those supporting or protecting safety-related items (Essential Cooling Pond - Braidwood)	I	N/A	N/A
b. Others (Cooling Pond - Braidwood)	II	N/A	N/A
c. Isolation Valve Room	I	N/A	N/A
d. Auxiliary Feedwater Tunnel	I	N/A	N/A
e. Spent Fuel Pool Concrete Structure	I	N/A	N/A
f. Spent Fuel Pool Liner	II	N/A	N/A
g. Refueling Cavity Concrete Structure	I	N/A	N/A
h. Refueling Cavity Liner	I	N/A	N/A
i. Reactor Vessel Nozzle Inspection Cavity Hatches	I	N/A	N/A
j. Containment Building Steel Liner	I	N/A	N/A
k. Refueling Water Storage Tank and Tank Foundation (except for small personnel hatch cover)	I	N/A	N/A
II. SYSTEMS AND COMPONENTS			
1. AB - Boric Acid Processing			
a. Boric Acid Tanks	I	C	N/A
b. Boric Acid Batching Tank	II	D	N/A
c. Boric Acid Transfer Pump	I	C	N/A
d. Boric Acid Transfer Pump Motor	II	N/A	Non-IE
e. Boric Acid Filter	I	C	N/A
f. Recycle Evaporator Feed Demineralizers	I	C	N/A
g. Recycle Hold-Up Tanks	I	C	N/A
h. Recycle Evaporator Feed Pumps	I	C	N/A
i. Recycle Evaporator Feed Pump Motors	II	N/A	Non-IE
j. Recycle Evaporator	I	C[3]	N/A
k. Recycle Evaporator (steam side)	II	D	N/A

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
l. Recycle Monitor Tanks	II	D	N/A
m. Recycle Evaporator Condensate Demineralizer	II	D	N/A
n. Recycle Evaporator Feed Filter	I	C	N/A

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
o. Recycle Evaporator Concentrates Filter	II	D	N/A
p. Recycle Evaporator Condensate Filter	II	D	N/A
q. Monitor Tank Pumps	II	D	N/A
r. Monitor Tank Pump Motors	II	N/A	Non-IE
s. Boric Acid Blender	I	C	Non-IE
t. Recycle Evaporator Concentrates Pumps	II	D	Non-IE
2. AC - Acid Feed and Handling (Except Boric Acid)	II	D	Non-IE
3. AF - Auxiliary Feedwater*			
a. Auxiliary Feedwater Pumps	I	C	N/A
b. Auxiliary Feedwater Pump Motors	I	N/A	IE
c. Auxiliary Feedwater Pump Diesel Engine Drives	I	G	IE
4. AN - Annunciator (Excluding Inputs)	II	N/A	Non-IE
5. AP - Auxiliary Power 480 Vac and above*			
a. All a-c auxiliary power equipment necessary for Category I items to perform their safety function including ESF switchgear, MCCS, transformers, buses and cables - which include the following:	I	N/A	IE
1. 4160-V ESF Buses: 141, 142, 241, 242			
2. 480-V ESF Buses: 131 X and Z, 132 X and Z, 231 X and Z, 232 X and Z (Byron only)			
3. 480-V ESF Buses: 131X, 132X, 231X, 232X (Braidwood only)			

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
4. 480-Vac Motor Control Centers fed from 480-V ESF Buses			
b. Other a-c auxiliary power equipment including unit auxiliary transformers, system auxiliary transformers and their low voltage connections to the ESF switchgear	II	N/A	Non-IE
6. AR - Area Radiation Monitoring			
a. Safety-Related Equipment	I	N/A	IE
b. All other equipment	II	N/A	Non-IE
7. AS - Auxiliary Steam (Including Heating Boiler)	II	D	Non-IE
8. BR - Boron Thermal Regeneration			
a. Moderating Heat Exchanger	I	C	N/A
b. Letdown Chiller Heat Exchanger			
Tube Side	I	C	N/A
Shell Side	II	D	N/A
c. Letdown Reheat Heat Exchanger			
Tube Side	I	B	N/A
Shell Side	I	C	N/A
d. Chiller Pumps	II	D	N/A
e. Chiller Surge Tank	II	D	N/A
f. Chiller	II	D	Non-IE
g. Thermal Regeneration Demineralizers	I	C	N/A
h. Chiller Pump Motors	II	N/A	Non-IE
9. CB - Condensate Booster	II	D	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
10. CC - Component Cooling*			
a. Component Cooling Heat Exchangers	I	C	N/A
b. Component Cooling Pumps	I	C	N/A
c. Component Cooling Surge Tanks	I	C	N/A
d. Component Cooling Pump Motors	I	N/A	IE
e. Containment Isolation	I	B	IE
11. CD - Condensate (Excluding Condensate Booster) (Including Makeup and Overflow)	II	D	Non-IE
12. CF - Chemical Feed and Handling (Hydrazine, Ammonia, Morphaline, Sulphite and other miscellaneous chemicals)	II	D	Non-IE
13. CO - Carbon Dioxide[9] (Includes Fire Protection and Generator Purge)	II	D	Non-IE
14. CQ - Code Call, Public Address, Telephone, Gate TV, Gate Operators, Evacuation Alarm, Station Security, etc.			
a. Electrical Penetrations	I	N/A	IE
b. All Other Components	II	N/A	Non-IE
15. CS - Containment Spray*			
a. Containment Spray Pumps	I	B	N/A
b. Containment Spray Pump Motor	I	N/A	IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
c. Spray Additive Tanks	I	B	N/A
d. Spray Eductors	I	B	N/A
16. CV - Chemical and Volume Control* (All except Boric Acid and Boron Thermal Regeneration)			
a. Regenerative Heat Exchangers			
Tube Side	I	B	N/A
Shell Side	I	B	N/A
b. Letdown Heat Exchangers			
Tube Side	I	B	N/A
Shell Side	I	C	N/A
c. Seal Water Heat Exchangers			
Tube Side	I	B	N/A
Shell Side	I	C	N/A
d. Excess Letdown Heat Exchangers			
Tube Side	I	B	N/A
Shell Side	I	B	N/A
e. Volume Control Tank	I	B	N/A
f. Reciprocating Charging Pumps	I	B	N/A
g. Reciprocating Charging Pump Motors	II	N/A	Non-IE
h. Centrifugal Charging Pumps	I	B	N/A
i. Centrifugal Charging Pump Motors	I	N/A	IE
j. Mixed Bed Demineralizers	I	C	N/A
k. Cation Bed Demineralizer	I	C	N/A
l. Resin Fill Tank	II	D	N/A
m. Chemical Mixing Tank	II	D	N/A
n. Letdown Reheat Heat Exchanger			
Tube Side	I	B	N/A
Shell Side	I	C	N/A

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
o. Reactor Coolant Filter	I	B	N/A
p. Seal Water Injection Filter	I	B	N/A
q. Seal Water Return Filter	I	B	N/A
r. Containment Isolation	I	B	IE
s. Letdown Booster Pump	II	D	N/A
t. Letdown Booster Pump Motor	II	N/A	Non-1E
17. CW - Circulating Water	II	N/A	Non-IE
18. CX - Computer (Excluding Inputs) and Supply Power	II	N/A	Non-IE
19. DC - Battery and D-C Distribution			
a. All d-c equipment necessary for Category I items to perform their function including batteries, chargers, distribution panels, and cables for these safety-related items, including the following:	I	N/A	IE
1. 125-Vdc Buses: 111, 112, 211, 212			
2. 125-Vdc Batteries and Chargers: 111, 112, 211, 212			
b. All other d-c equipment including batteries, chargers, distribution panels and cables for non-safety-related items	II	N/A	Non-IE

Table 3.2-1 (Cont'd)

20. DG - Diesel Generator*			
a. Diesel Engine-mounted components (including starting air, cooling water, lube oil and fuel oil support systems).	I	G	IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
b. Auxiliary skid-mounted components in starting air, cooling water, lube oil and fuel oil support systems with the exception of starting air compressors that are classified as Safety Category II, Quality Group D.	I I	G (Byron) C (Brwd)	IE
c. Air dryer package	II	D	Non-IE
21. DM - Drains, Miscellaneous Buildings, Floor and Roof Including Sump Pumps - Non-Radioactive (Crib House, Pumphouse)	II	N/A	Non-IE
22. DO - Diesel Fuel Oil* (Supply and Transfer)			
a. Diesel Fuel Oil Storage Tanks	I	C	N/A
b. Diesel Fuel Oil Transfer Pumps and in-line components	I I	G (Byron) C (Brwd)	N/A N/A
c. Diesel Fuel Oil Transfer Pump Motors	I	N/A	IE
d. Diesel Fuel Oil Unloading Pump Motors	II	D	Non-IE
23. DV - Feedwater Heater Miscellaneous Drains and Vents	II	D	Non-IE
24. EC - Chemical Cleaning, Equipment and Pipe	II	D	Non-IE
25. EF - Engineered Safety Features Logic Testing and Actuation*	I	N/A	IE
26. EH - Turbine EHC	II	D	Non-IE
27. EM - Environs Monitoring (Including Strong Motion - Seismic Instrumentation)	II	N/A	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
28. ES - Extraction Steam	II	D	Non-IE
29. EW - Welder Outlets	II	N/A	Non-IE
30. FC - Fuel Pool Cooling and Clean-Up (See Table 9.1-2)			
a. Spent Fuel Pit Heat Exchanger	I	C	N/A
b. Spent Fuel Pit Pump	I	C	N/A
c. Spent Fuel Pit Pump Motor	II	N/A	Non-IE
d. Skimmer Pump	II	D	N/A
e. Skimmer Pump Motors	II	N/A	Non-IE
f. Spent Fuel Pit Filter	II	D	N/A
g. Spent Fuel Pit Demineralizer	II	D	N/A
h. Refueling Water Purification Pump (one pump only)	I	C	N/A
i. Refueling Water Purification Pump Motor	II	N/A	Non-IE
31. FH - Fuel Handling and Transfer Nuclear			
a. New Fuel Storage Racks	I	N/A[4]	N/A
b. Spent Fuel Storage Racks	I	N/A[4]	N/A
c. Fuel Handling Building Crane	II	N/A[4,8]	Non-IE
d. Manipulator Crane	II	N/A[8]	Non-IE
e. Spent Fuel Bridge Crane	II	N/A[8]	Non-IE
f. New Fuel Elevator	II	N/A	Non-IE
g. Fuel Transfer System			
Fuel Transfer Tube & Flange	I	N/A	N/A
Conveyor System (Fuel Building Side)	II	N/A[8]	N/A

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
h. Remainder of System	II	N/A	N/A
32. FP - Fire Protection and Detection (Excluding CO_2 Systems)			
a. Seismic Qualified Areas	I	C	N/A
b. Centrifugal Pumps	II	D	Non-IE
c. Containment Isolation	I	B	IE
33. FW - Main Feedwater			
a. Outside Containment up to Isolation Valve	II	D	Non-IE
b. Inside Containment up to and including Isolation Valve	I	B	IE
34. GC - Generator Stator Cooling (Including Excitation Cubicle Cooling)	II	D	Non-IE
35. GD - Grounding and Cathodic Protection	II	N/A	Non-IE
36. GS - Turbine Gland Seal Steam	II	D	Non-IE
37. GW - Radioactive Waste Gas (Excluding Off-Gas) including the following:			
a. Waste Gas Compressor (See note 12)	II	D	Non-IE
b. Gas Decay Tanks	I	C	Non-IE
c. Gas Analyzer	II	D	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
38. HC - Hoists, Cranes, Elevators, and Manlifts (All except Fuel Handling and Transfer System)			
a. Containment Building Crane	II	N/A	Non-IE
b. All other equipment	II	N/A	Non-IE
39. HD - Feedwater Drains - Turbine Cycle	II	D	Non-IE
40. HT - Heat Tracing	II	N/A	Non-IE
41. HY - Hydrogen (for Turbine Generator)	II	D	Non-IE
42. IA - Instrument Air (Including the following)			
a. Instrument Air Receivers	II	D	Non-IE
b. Instrument Air Afterfilters	II	D	Non-IE
c. Instrument Air Dryers	II	D	Non-IE
d. Instrument Air Prefilters	II	D	Non-IE
e. Containment Isolation	I	B	IE
43. IC - Incore Flux Mapping			
a. Tubing from Reactor Vessel to Seal Table	I	A	N/A
b. Electrical Penetrations	I	N/A	IE
c. All other equipment and tubing	II	N/A	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
44A. IP - Instrument and Control Power* (Including Inverters, MG Sets)			
a. All equipment necessary for Category I items to perform their safety functions	I	N/A	IE
b. All other equipment	II	N/A	Non-IE
44B. IS - Industrial Security, Gate Operators, TV, etc.	II	N/A	Non-IE
45. IT - Incore Thermocouple System	I	N/A	IE
46. LL - Lighting	II	N/A	Non-IE
47. LV - Auxiliary Power, Low Voltage 120/208 V, Transformers, Distribution			
a. Electrical Penetrations	I	N/A	IE
b. Power Available Lights for Auxiliary Safeguards Cabinets	I	N/A	IE
c. All Other Components	II	N/A	Non-IE
48. MP - Main Power (Generator, Exciter Main Transformer, Bus Duct)	II	N/A	Non-IE
49. MS - Main Steam			
a. Inside Containment up to and including Isolation Valve	I	B	IE
b. Turbine Stop Valve Limit Switches (see Note 5)	I	N/A	IE
c. Other	II	D	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
50. NR - Neutron Monitoring System* (out of core)			
a. Detectors Not Required for Safety Functions	II	N/A	Non-IE
b. Other Instruments	I	N/A	IE
51. NT - Nitrogen			
a. Various containment electrical penetrations	I	B	IE
b. Others	II	D	Non-IE
52. OD - Equipment and Floor Oil Drain Disposal (Including Sump Pumps)	II	N/A	Non-IE
53. OG - Off-Gas (Including Hydrogen Recombiner)			
a. Hydrogen Recombiner	I	B	IE
b. All other equipment	II	D	Non-IE
c. Containment Isolation	I	B	IE
54. OH - Caustic Handling	II	N/A	Non-IE
55. OT - Bearing Oil Transfer and Purification (For Turbine-Generator and Turbine Drives)	II	D	Non-IE
56. PA - Auxiliary Control Equipment Room and Computer Room Panels and Cabinets			
a. For Safety-Related Equipment	I	N/A	IE
b. Others	II	N/A	Non-IE

TABLE 3.2-1 (Cont'd)

	PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
57.	PC - Primary Containment Isolation	I	B	IE
58.	PI - Control Rod Position Indication	II	N/A	Non-IE
	b. Others	II	N/A	Non-IE
60.	PM - Main Control Room Panels			
	a. For Safety-Related Equipment	I	N/A	IE
	b. Others	II	N/A	Non-IE
61.	PR - Process Radiation Monitoring			
	a. Safety-Related Equipment	I	N/A	IE
	b. All other equipment	II	D	Non-IE
	c. Containment Isolation	I	B	IE
62.	PS - Process Sampling Primary & Secondary System Including Chiller Equipment (Samp. Cond. & Monitoring Assemblies)			
	a. Primary Sampling Remote Air Operated Valves	I	B	Non-IE
	b. Primary Sampling Containment Isolation	I	B	IE
	c. All other equipment	II	D	Non-IE
63.	PW - Primary Water	II	D	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
64. RC - Reactor Coolant System (Not including Pressurizer System)			
a. Reactor Vessel	I	A	N/A
b. Steam Generator			
Tube Side	I	A	N/A
Shell Side	I	B	N/A
c. Reactor Coolant Loop Stop Valves	I	A	N/A
d. Pressure Boundary Piping and Fittings	I	A	N/A
e. Reactor Coolant Pump (RCP)			
RCP Casing	I	A	N/A
Main Flange	I	A	N/A
Thermal Barrier	I	A	N/A
Thermal Barrier Heat Exchanger	I	A	N/A
#1 Seal Housing	I	A	N/A
#2 Seal Housing	I	B	N/A
Pressure Retaining Bolting	I	A	N/A
f. RCP Motor (Refer to Section 5.4 for Safety Function)	I	N/A	Non-IE
65. RD - Control Rod Drive			
a. Full Length CRDM Housing	I	A	N/A
b. CRDM Head Adapter Plugs	I	A	N/A
c. Thermal Sleeves	I	N/A	N/A
d. Control Rod Drive Mechanism	I	N/A	N/A
e. Reactor Trip Switchgear	I	N/A	IE
f. All Other Components	II	N/A	Non-IE

TABLE 3.2-1 (Cont'd)

	PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
66.	RE - Reactor Building & Containment Equipment Drains to Radwaste (Including Reactor Coolant Drains and Pumps)			
	a. Reactor Coolant Drain Tank	II	D	N/A
	b. Reactor Coolant Drain Pumps	II	D	N/A
	c. Reactor Coolant Drain Pump Motors	II	N/A	Non-IE
	d. Containment Isolation	I	B	IE
67.	RF - Reactor Building & Containment Floor Drains to Radwaste (Including Sump Pumps)	II	D	Non-IE
	a. Containment Isolation	I	B	IE
68.	RH - Residual Heat Removal Pumps*			
	a. Residual Heat Removal Pump	I	B	N/A
	b. Residual Heat Removal Pump Motors	I	N/A	IE
	c. Residual Heat Exchangers			
	Tube Side	I	B	N/A
	Shell Side	I	C	N/A
69.	RP - Reactor Protection*	I	N/A	IE
70.	RY - Reactor Coolant Pressurizer System			
	a. Pressurizer	I	A	N/A
	b. Pressurizer Relief Tank	II	D	N/A
	c. Pressurizer Heaters	II	N/A	Non-IE
	d. Pressurizer Safety Valves	I	A	N/A
	e. Pressurizer Power-Operated Relief Valves	I	A	IE
	f. Containment Isolation	I	B	IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
71. SA - Service Air	II	D	Non-IE
a. Containment Isolation	I	B	IE
72. SD - Steam Generator Blowdown System			
a. Blowdown Condenser	II	D	N/A
b. Blowdown Condenser Pump	II	D	N/A
c. Blowdown Condenser Pump Motors	II	N/A	Non-IE
d. From Steam Generator to Containment Isolation Valves	I	B	IE
e. From Blowdown Lines to Wet Layup Spectacle Flanges (Unit 1 only)	I	B	N/A
f. From Wet Layup Spectacle Flanges to Wet Layup Pumps (Unit 1 only)	II	D	Non-IE
73. SH - Station Heating (Steam, Water, or Electrical) (Excluding Duct Air Systems)	II	D	Non-IE
74. SI - Safety Injection System*			
a. Refueling Water Storage Tank	I	N/A	N/A
b. Accumulators	I	B	N/A
c. Safety Injection Pumps	I	B	N/A
d. Safety Injection Pump Motors	I	N/A	IE
e. Containment Isolation	I	B	IE
f. Containment Sump Screen	I	N/A	N/A

TABLE 3.2-1 (Cont'd)

	PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
75.	SS - System Security (Automatic Dispatch)	II	N/A	Non-IE
76.	ST - Sewage Treatment	II	N/A	Non-IE
77.	SW - Screen Wash	II	D	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
78. SX - Essential Service Water System*			
a. Essential Service Water Pumps	I	C	N/A
b. Essential Service Water Pump Motors	I	N/A	IE
c. Essential Service Water Strainers	I	C	N/A
d. Essential Service Water Cooling Tower Make-Up Pump (Byron only)	I	C	N/A
e. Essential Service Water Cooling Tower Make-Up Pump Diesel Engine (Byron only)	I	G	IE
f. Essential Service Water Strainer Backwash Motor	II (BYR) I (BRW)	N/A N/A	Non-IE (BYR) IE (BRW)
g. Essential Cooling Tower Fan Motors (Byron only)	I	N/A	IE
h. Essential Service Water Bypass Valves 0SX162A-D (Byron only)			
1. Valve Body	I	C	N/A
2. Motor Operators	I	C	IE
i. Essential Service Water Discharge Extension Lines 0SX03EA and 0SX03EB (Braidwood only).	II	D	N/A
j. Containment Isolation	I	B[10]	IE
k. Essential Service Water Cooling Tower Make-Up Line Vacuum Breakers 0SX169A-F (Byron Only)	I	G	Non-IE
79. SY - Switchyard	II	N/A	Non-IE
80. TD - Turbine Drains and Vents	II	D	Non-IE
81. TE - Turbine Building Equipment Drains	II	D	Non-IE
82. TF - Turbine Building Floor Drains	II	N/A	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
83. TG - Turbine-Generator Auxiliaries and Miscellaneous Devices (Turning Gear, etc.)	II	D	Non-IE
84. TO - Turbine Oil (Bearing Oil and Seal Oil Systems furnished with Turbine-Generator)	II	D	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
85. TR - Treated Runoff (Industrial Wastewater Treatment)	II	D	Non-IE
86. TS - Turbine Supervisory	II	N/A	Non-IE
87. TW - Treated Water (Including Clarifier and Filtered Water, and Potable Water)	II	D	Non-IE
88. VA - Auxiliary Building HVAC*			
a. Supply, Exhaust,and Booster Fans	I	N/A	N/A
b. Supply, Exhaust, and Booster Fan Motors	I	N/A	IE
c. Cubicle Cooler Coils	I	C	N/A
d. Cubicle Cooler Fan Motors	I	N/A	IE
Other Cubicle Cooler Fan Motors	II	N/A	Non-IE
e. Controls and Instrumentation	I	N/A	IE
f. Exhaust Filters	I	N/A	N/A
g. Supply Filters	II	N/A	N/A
h. Supply Filter Plenums	I	N/A	N/A
89. VC - Control Room, Auxiliary Electric Equipment Room HVAC*			
a. Supply and Return Fans	I	N/A	N/A
b. Supply and Return Fans Motors	I	N/A	IE
c. Makeup Air Filter Package	I	N/A	IE
1. Motor, Flow Control, Electric Heater and Electric Heater Controls	I	N/A	IE
2. Instrumentation for indication and alarm	II	N/A	Non-IE
d. Cooling Coils	I	C	N/A

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
e. Supply Filters	I	N/A	N/A
f. Controls & Instrumentation	I	N/A	IE
g. Electric Heaters	II	N/A	Non-IE
h. Humidifiers	II	N/A	Non-IE
i. Charcoal Recirc Filter Instrumentation for Indication and Alarm	II	N/A	Non-IE
j. Ductwork and Dampers (Incl. Operators)	I	N/A	IE
k. Utility Exhaust Fans	II	N/A	Non-IE
l. Charcoal Recirc Filter	I	N/A	Non-IE
90. VD - Diesel-Generator Room Ventilation*			
a. Diesel-Generator Room Ventilation Fans & Motors	I	N/A	IE
b. Diesel-Generator Room Exhaust Fans & Motors	I[6]	N/A	IE
c. Controls and Instrumentation	I	N/A	IE
91. VE - Misc. Electric Equipment Room Ventilation* Including Controls and Instrumentation	I	N/A	IE
92. VF - Containment Building, Auxiliary Building Filtered Vents	II	D	Non-IE
93. VH - Pumphouse Ventilation	II	N/A	Non-IE
94. VI - Radwaste & Remote Shutdown Control Room HVAC	II	N/A	Non-IE
95. VN - Containment Building, & Auxiliary Building Non-Filtered Vents	II	D	Non-IE
96. VJ - Machine Shop Ventilation	II	N/A	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
97. VK - Switchyard Relay House HVAC	II	N/A	Non-IE
98. VL - Laboratory HVAC	II	N/A	Non-IE
99. VP - Primary Containment Ventilation*			
a. Reactor Containment Fan Coolers	I	N/A[4]	N/A
b. Reactor Containment Fan Coolers Motors	I	N/A	IE
c. CRDM Exhaust Fans	I	N/A	N/A
d. CRDM Exhaust Fans Motors	II	N/A	Non-IE
e. Reactor Cavity Vent Fans	II	N/A	N/A
f. Reactor Cavity Vent Fans Motors	II	N/A	Non-IE
g. CRDM Booster Fans	II	N/A	N/A
h. CRDM Booster Fans Motors	II	N/A	Non-IE
i. Containment Charc. Filter Unit[13]	II	N/A	N/A
j. Containment Charc. Filter Fan[13]	II	N/A	N/A
k. Cont. Charc. Filter Unit Fan Motor[13]	II	N/A	Non-IE
l. RCFC Ess. Service Water Coils	I	C[11]	N/A
m. RCFC Chill Water Coils	II	D	N/A
n. Ductwork Dampers and Supports			
1. Ductwork, Dampers and Supports Associated with RCFC	I	N/A	N/A
2. Ductwork, Dampers and Supports for Systems other than RCFC	II	N/A	N/A
q. RCFC Controls & Instrumentation			
1. Control Switches for RCFC Fans	I	N/A	IE
2. Vibration Switches for RCFC Motors	I	N/A	IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
3. Temperature Sensors/Monitor Upstream and Downstream of RCFC	II	N/A	Non-IE
r. RCFC Drain Pans and Channels Supporting Drain Pans	II	N/A	N/A
100. VQ - Primary Containment Purge			
a. Post-LOCA Purge Filter Unit	II	N/A	N/A
b. Purge Supply & Exh Fan	II	N/A	N/A
c. Purge Supply & Exh Fan Motor	II	N/A	Non-IE
d. Purge Supply & Exh Filters	II	N/A	N/A
e. Purge Controls & Instrumentation	II	N/A	Non-IE
f. Post-LOCA Purge Contr. & Instr.	II	N/A	Non-IE
g. Containment Isolation	I	B	IE
101. VS - Service Building, HVAC	II	N/A	Non-IE
102. VT - Turbine Building, HVAC	II	N/A	Non-IE
103. VV - Miscellaneous Ventilation	II	N/A	Non-IE
104. VW - Radwaste Facility Ventilation	II	N/A	Non-IE
105. VX - Switchgear Heat Removal*			
a. For Class IE Switchgear Rooms	I	N/A	IE
b. Other	II	N/A	Non-IE
c. Controls and Instrumentation	I	N/A	IE
106. WE - Aux. Bldg. Equip. Drain Radwaste Reprocessing & Disposal	II	D	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
107. WF - Auxiliary Building Floor Drain Radwaste Reprocessing & Disposal	II	D	Non-IE
108. WG - Gland Water	II	N/A	Non-IE
109. NOT USED			
110. WM - Makeup Demineralizer (Including Effluent and Flushing)			
a. Containment Isolation	I	B	IE
b. All Other Components	II	D	Non-IE
111. WO - Chilled Water*			
a. Control Room System			
1. Chilled Water and Service Water portions	I	C	IE
2. Refrigerant and Oil portions of Chiller Skids	I	G	1E
b. Remainder of System	II	D	Non-IE
c. Containment Isolation	I	B	IE
112. WS - Non-Essential Service Water	II	D	Non-IE
113. WW - Well Water (See Note 7 - Byron only)	II	D	Non-IE

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
114. WX - Solid Radwaste Reprocessing & Disposal (Wet and Dry) (Including Drumming and Resin Removal)			
a. Blowdown Demineralizers	II	D	N/A
b. Blowdown Prefilters	II	D	N/A
c. Blowdown Monitor Tanks	II	D	N/A
d. Blowdown Monitor Tank Pumps	II	D	N/A
e. Blowdown Monitor Tank Pump Motors	II	N/A	Non-IE
f. Concentrates Holding Tank (Byron)	II	D	N/A
g. Spent Resin Tank (Byron)	II	D	N/A
h. Low Activity Spent Resin Tank (Braidwood)	II	D	N/A
i. High Activity Spent Resin Tank (Braidwood)	II	D	N/A
j. Laundry Drain Tanks	II	D	N/A

TABLE 3.2-1 (Cont'd)

PRINCIPAL STRUCTURES, SYSTEMS, AND COMPONENTS	SAFETY CATEGORY	QUALITY GROUP	ELECTRICAL
k. Release Tank	II	D	N/A
l. Radwaste Evaporators	II	D	Non-IE
m. Waste Evaporator Monitor Tanks	II	D	N/A
n. Spent Resin Flushing Pump	II	D	N/A
o. Spent Resin Flushing Pump Motor	II	N/A	Non-IE
p. Liquid Radwaste Filters	II	D	N/A
115. WY - Laundry Equipment & Floor Drains Radwaste Reprocessing & Disposal	II	D	Non-IE
a. Laundry Drain Tank	II	D	N/A
116. WZ - Chemical Radwaste Reprocessing & Disposal	II	D	Non-IE
a. Chemical Drain Tank	II	D	N/A
117. Reactor Vessel or Core-Related			
a. Reactor Vessel Shoes and Shims	I	N/A	N/A
b. Control Rod Drive Mechanism Seismic Support Tie Rod Assemblies	I	N/A	N/A
c. Control Rod Guide Tubes	I	N/A	N/A
d. Reactor Vessel Internals	I	N/A	N/A
e. Full Length Control Rod Clusters	I	N/A	N/A
f. Burnable Poisons	II	N/A	N/A
g. Primary and Secondary Sources	II	N/A	N/A
h. Irradiation Sample Holder	I	N/A	N/A
i. Control Rod Drive Mechanism Dummy Can Assemblies	II	N/A	N/A

TABLE 3.2-1 (Cont'd)

NOTES

1. The Byron river screen house is not designed to withstand the probable maximum flood or design basis tornado because an alternate source of makeup water, described in Section 9.2, is available for the ultimate heat sink.

2. The portion of the Braidwood lake screen house housing the essential service water intake is Safety Category I.

3. The steam side of the evaporator is Quality Group D, Safety Category II. Only those portions containing radioactive liquid are designated Quality Group C, Safety Category I.

4. Quality measures equivalent in intent to those in Quality Group C apply.

5. Applies only to turbine stop valve limit switches used for reactor trip inputs. Limit switches are located in the turbine building (Seismic Category II).

6. The diesel-generator room exhaust subsystem is Category I although it is required only to be Category II.

7. The portions of the Byron well water system which provide makeup water to the essential service water cooling towers have been qualified to withstand the design basis seismic event. The deep wells are used as an alternate source of makeup to the ultimate heat sink as described in Sections 2.4 and 9.2. New equipment which supports this function are procured safety related, non-ASME, Category IG.

8. Administrative procedures are provided and implemented to provide maintenance of these items. Activities performed via the procedures are independently monitored to assure implementation.

9. A seismic analysis has been performed for the carbon dioxide storage tank located near the Unit 1 Turbine Building elevator (Column K-16). The analysis concluded that a seismic event (safe shutdown earthquake and operating basis earthquake) will not affect the structural integrity of the tank.

10. Quality Group C piping was upgraded to Quality Group B standards. Refer to Subsection 6.2.2.4.1 and notes on Drawings M-42 Sheet 5A and M-126 Sheet 3.

11. In order to provide a level of quality equivalent to Quality Group B standards, additional ASME Section III Class 2 nondestructive examination was performed on the RCFC essential service water coils (Refer to Subsection 6.2.2.4.1).

12. The Waste Gas Compressor Heat Exchanger is Safety Category I Quality Group C.

13. Containment charcoal filter units at Braidwood Station have been abandoned in place.

*See Table 7.1-2.

TABLE 3.2-2

CODE REQUIREMENTS FOR COMPONENTS AND SYSTEMS

COMPONENT OR SYSTEM	QUALITY GROUP A	QUALITY GROUP B	QUALITY GROUP C	QUALITY GROUP D
Pressure Vessels* ***	ASME Boiler and Pressure Vessel Code Section III, Class 1	ASME Boiler and Pressure Vessel Code Section III, Class 2	ASME Boiler and Pressure Vessel Code Section III, Class 3	ASME Boiler and Pressure Vessel Code, Section VIII Div. 1
Piping ***	ASME Boiler and Pressure Vessel Code, Section III, Class 1	ASME Boiler and Pressure Vessel Code, Section III, Class 2	ASME Boiler and Pressure Vessel Code, Section III, Class 3	ANSI B31.1.0 Code for Pressure Piping
Pumps and Valves ***	ASME Boiler and Pressure Vessel Code, Section III, Class 1	ASME Boiler and Pressure Vessel Code, Section III, Class 2	ASME Boiler and Pressure Vessel Code, Section III, Class 3	ANSI B31.1.0 Code for Pressure Piping**
Low-Pressure Tanks ***	N/A	ASME Boiler and Pressure Vessel Code Section III, Class 2	ASME Boiler and Pressure Vessel Code, Section III, Class 3	American Petroleum Institute Recommended Rules for Design and Construction of Large Welded Low-Pressure Storage Tanks, API 620

* Containment vessel excluded.

** For pumps operating above 150 psi or 212°F, ASME Section VIII, Division 1, shall be used as a guide for calculating thickness of pressure retaining parts and in sizing cover bolting; below 150 psi and 212°F, manufacturer's standards for service intended will be used.

*** In certain limited cases, configurations exist with manual isolation valves installed inline with pressure relieving components. These manual valves are administratively controlled in the locked open position to ensure the relief capacity of the pressure relieving components is maintained.

TABLE 3.2-2 (Cont'd)

CODE REQUIREMENTS FOR COMPONENTS AND SYSTEMS

COMPONENT OR SYSTEM	QUALITY GROUP A	B	C	D
Atmospheric Storage Tanks	N/A	ASME Boiler and Pressure Vessel Code, Section III, Class 2	ASME Boiler and Pressure Vessel Code, Section III, Class 3	American Waterworks Association Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, AWWA-D100; Welded Steel Tanks for Oil Storage, API-650, or ANSI B96.1
Heat Exchangers	ASME Boiler and Pressure Vessel Code, Section III, Class 1	ASME Boiler and Pressure Vessel Code, Section III, Class 2	ASME Boiler and Pressure Vessel Code, Section III, Class 3	ASME Boiler and Pressure Vessel Code, Section VIII, Div. 1 and Tubular Exchanger Manufacturers Association (TEMA) Class C

TABLE 3.2-3

CROSS-REFERENCE BETWEEN ANS
SAFETY CLASSIFICATION AND BYRON/BRAIDWOOD
SAFETY CLASSIFICATION DESIGNATIONS

ANS	BYRON/BRAIDWOOD	
SAFETY CLASS	QUALITY GROUP	SAFETY CATEGORY
1	A	I
2	B	I
3	C	I
NNS*	D	II
Non-ASME**	G	I

* NNS – Non-Nuclear Safety

**Non-ASME – Safety-Related Non-ASME

3.3 WIND AND TORNADO LOADINGS

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

A design wind velocity of 85 mph, based upon a 100-year mean recurrence interval, is used in the design of Seismic Category I structures.

For Category II structures a design wind velocity of 75 mph is used, based upon a 50-year mean recurrence interval.

The vertical velocity distribution and gust factors employed for the wind velocities are based on Table 5 of Reference 1 for exposure Type C.

3.3.1.2 Determination of Applied Forces

The dynamic wind pressures are converted to an equivalent static force by considering appropriate pressure coefficients. The applied forces were derived in accordance with the provisions of Table 7, Reference 1, using external pressure coefficients, Cp of 0.8 and -0.5 for windward and leeward walls respectively, and -0.7 for side walls and roofs.

For structural shapes other than rectangular appropriate pressure coefficients are used in accordance with Reference 2.

3.3.2 Tornado Loadings

3.3.2.1 Applicable Design Parameters

The following are the parameters for the design-basis tornado (Reference 3):

Tangential velocity: 290 mph

Translational velocity: 70 mph

Radius of maximum rotational velocity from center of tornado: 150 feet

Pressure drop at the center of vortex: 3 psi

Rate of pressure drop: 2 psi/sec.

The characteristics and spectrum of design-basis tornado-generated missiles are found in Subsection 3.5.1.4.

The tornado parameters used in the probabilistic tornado missile risk analysis (TORMIS) described in Byron only Section 3.5.5 are found in Reference 5.

Load Factor

Since the postulated tornado loading is an extreme environmental condition with a very low probability of occurrence, a load factor of 1.0 is used.

3.3.2.2 Determination of Forces on Structures

The Category I structures which have wind tornado loads, design-basis tornado generated missiles, and/or combination of these loads addressed in their design are as follows:

a. containment building,

b. auxiliary building,

c. fuel handling building,

d. main steam tunnel,

e. auxiliary feedwater tunnel,

f. essential service water cooling tower (Byron),

g. essential cooling pond (Braidwood),

h. deep well enclosures (Byron),

i. lake screen house substructure (Braidwood),

j. isolation valve room, and

k. essential service water discharge (Braidwood).

Several individual essential service water cooling tower components (Byron) not fully protected from tornado generated missiles are addressed in the probabilistic tornado missile risk analysis (TORMIS) described in Byron only Section 3.5.5.

3.3.2.2.1 Transformation of Tornado Winds Into Effective Pressure

All tornado wind pressure and differential pressure effects are considered as static loads since the natural period of building structures and their exposed structural elements is very short compared to the rate of variation of the applied loads.

The effects of the design-basis tornado are translated into forces on structures with the use of a tornado model (Reference 4) that incorporates parameters defined in Subsection 3.3.2.1.

The tornado model considers a velocity distribution based on the following equations:

$$v(r) = V_c \frac{r}{R_c} + V_t \text{, for } \frac{r}{R_c} \leq 1 \qquad (3.3\text{-}1)$$

$$v(r) = V_c \frac{R_c}{r} + V_t \text{, for } \frac{r}{R_c} \geq 1 \qquad (3.3\text{-}2)$$

where:

$v(r)$ = wind velocity at radius r,

r = distance from the center of the tornado,

V_c = maximum tangential velocity,

R_c = distance from the center of the tornado, to the locus of the maximum wind velocity, and

V_t = translational velocity.

The distribution of the pressure drop with the radius from the tornado is as follows:

$$p(r) = 3.0\,[1 - 0.5\,(r/R_c)^2] \text{, for } \frac{r}{R_c} < 1 \qquad (3.3\text{-}3)$$

$$p(r) = 1.5\,(R_c/r)^2 \text{, for } \frac{r}{R_c} \geq 1 \qquad (3.3\text{-}4)$$

where:

$p(r)$ = pressure drop in psi.

The tornado velocity is converted into an equivalent static pressure using equations given in ANSI A58.1-1972 (Reference 1). Neither a "gust factor" nor any change in velocity with height is considered. Figure 3.3-1 shows the variation in wind velocity and differential pressure as per Equations 3.3-1, 3.3-2, 3.3-3, and 3.3-4. Figure 3.3-2 shows the windward and leeward wind pressure components of the tornado.

The load combination equation used for tornado load and tornado generated missiles is W_t = 448 psf + W_m. Figure 3.3-3 shows the resulting surface pressure when the effect of tornado wind and pressure drop components are added together.

The load combination equations as per SRP Section 3.3.2 using load parameters of UFSAR Section 3.3 are as follows:

1. $W_t = W_w$ i.e., W_t = 265 psf
2. $W_t = W_p$ i.e., W_t = 432 psf
3. $W_t = W_p = W_m$
4. $W_t = W_w + .5\, W_p$ i.e., W_t = 340.1 psf
5. $W_t = W_w + W_m$ i.e., W_t = 265 psf + W_m
6. $W_t = W_w + .5\, W_p + W_m$ i.e., W_t = 340.1 psf + W_m.

The equation (W_t = 448 psf + W_m) used in design is more conservative than the SRP equations above.

3.3.2.2.2 Venting of the Structure

Venting of concrete structures is not relied upon to reduce the differential pressure loadings. However, all siding and roof decking of the Turbine Building above the floor at elevation 451 feet 0 inch is designed and detailed to blow off at tornado pressures exceeding 105 psf. Above this pressure only bare framework is considered to be exposed to design-basis tornado loads.

3.3.2.2.3 Tornado Generated Missiles

The characteristics and spectrum of tornado generated missiles for the design-basis tornado are found in Subsection 3.5.1.4. The characteristics and spectrum of tornado generated missiles considered in the probabilistic tornado missile risk analysis (TORMIS) described in Byron only Section 3.5.5 are found in Reference 5. The procedures used for designing for the impactive dynamic effects of a point load resulting from tornado generated missiles are found in Subsection 3.5.3.

3.3.2.2.4 Tornado Loading Combinations

Refer to Tables 3.8-3 through 3.8-9 for the load factors and load combinations associated with tornado loading. In designing for the postulated design-basis tornado, the structure in consideration is placed in various locations of the pressure field to determine the maximum critical effects of shear, overturning moment, and torsional moment on the structure.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loadings

All non-safety-related structures which are connected to safety-related structures are designed to prevent collapse under the design tornado loading. The only exceptions are the fuel handling building train shed, the Essential Service Water Cooling Tower Security Booth Tower Walkway (applicable to Byron only) and Walkway Access Stair Tower (applicable to Byron only) and the equipment staging structures installed adjacent to the emergency hatches. The collapse of these structures under tornado loading does not affect the structural integrity of any safety-related structures.

All other non-safety-related structures are separated from safety-related structures by a distance exceeding the height of the non-safety-related structure. This ensures that the failure of non-safety-related structures will not affect safety-related structures. Missiles generated by the collapse of non-safety-related structures were evaluated to be less critical than those considered in Subsection 3.5.1.4 or were evaluated in Byron only Section 3.5.5.

3.3.3 References

1. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," American National Standards Institute, Inc., New York, New York, 1972.

2. "Task Committee on Wind Forces, Committee on Loads and Stresses, Wind Forces on Structures, Final Report," Paper No. 3269, Transactions, ASCE, Vol. 26.rg 1961.

3. USNR Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plant," April 1974.

4. J. D. Stevenson, "Tornado Design of Class I Structures for Nuclear Power Plants," Proceedings of Symposium on Structural Design of Nuclear Power Plant Facilities, University of Pittsburgh, December 1972.

5. Design Analysis ARA-002116, "Tornado Missile TORMIS Analysis of [Byron Generating Station] BGS," Revision 3.

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 Flood Protection

3.4.1.1 Flood Sources

The probable maximum flood (PMF) level for Byron Station, as defined in Subsection 2.4.3, is 708.3 feet above mean sea level. The elevation of the plant grade floor is 870 feet. The probable maximum flood level for Braidwood Station, as given in Subsection 2.4.8, is 598.17 feet above mean sea level. The plant grade floor elevation is 601 feet. In both cases, the probable maximum flood level is below the level of the plant grade floor, therefore, it will have no damaging effect on any safety-related structure, except the river screen house at the Byron Station which is discussed below.

The Byron river screen house is designed for the combined event flood. The combined event flood stage for the Rock River at the river screen house is 698.68 feet (refer to Subsection 2.4.3.7 for definition of combined event flood). The maximum wave runup plus setup is 4.71 feet. To prevent damage due to flood, the floor elevation is established at 702 feet and a 4-foot-high fire wall encloses the area where safety-related equipment is located.

The probable maximum precipitation at the plant sites causes only minor local flooding as discussed in Subsection 2.4.2.3. It has no appreciable effect on the perched groundwater elevation.

3.4.1.2 Safety-Related Systems

The list of safety-related systems and components is found in Section 3.2. All such systems located below grade in Seismic Category I structures are flood protected.

Byron Station is dependent upon the essential service water makeup subsystem which is required for safe shutdown (i.e., "hot" shutdown) to maintain an adequate volume of Category I storage as an auxiliary feedwater backup source in each cooling tower basin. The makeup system also assures that the cooling towers will continue to receive replenishment of blowdown, evaporative, and drift losses so that essential cooling capability is maintained. The combined event flood level is 698.68 feet as indicated in Subsection 2.4.3.9. Wave run ups are 2.77 feet for the significant wave and 4.71 feet for the maximum wave, so that the run up elevations are 701.45 feet and 703.39 feet, respectively. The makeup pump is at elevation 702 feet 0 inch. The engine is mounted on its subbase at elevation 703 feet 8 1/2 inches. The engine shaft centerline elevation is 705 feet 4 inches and the lower battery post elevation is approximately 703 feet 8 inches.

Braidwood Station is dependent only upon maintaining the undiked or higher water level in the essential cooling pond for safe shutdown. All safety-related systems, subsystems, and components other than buried out of door piping are located within the auxiliary and the containment buildings and their connecting pipe tunnels, and require no further flood protection.

The essential service water pumps are located in the auxiliary building basement (Elevation 330). The 1A and 2A pumps are located in one compartment and the 1B and 2B pumps are located in a separate adjacent compartment. Entrance to each compartment is via a watertight door. The doorways and penetrations into each compartment are watertight except for ventilation ducts connecting the essential service water compartments to the auxiliary building floor drain sump pump and to the auxiliary building equipment drain pump subcompartments. The lowest elevation of these ducts, at the penetration into the essential service water pump rooms, is approximately ten feet above the basement floor and is well above the design basis flood levels for the essential service water pump rooms. Therefore, flooding of one compartment due to a pipe failure will not cause flooding of the other train essential service water pump compartment and will not cause loss of function in either unit.

The leak detection sumps for each compartment are described in Subsection 9.2.1.2.4.

Each compartment contains an essential service water sump as shown in Drawing M-11. Each sump has two sump pumps. Since the essential service water is nonradioactive, the pumps normally discharge to the turbine building drain tank. However, at Braidwood, upon contamination or anticipation of contamination in the essential service water sump, local operator action may be taken to direct the flow to the auxiliary building floor drain tank.

Shown on the same drawing are subcompartments in which the auxiliary building floor drain sumps are located. Drainage from upper elevations is collected and pumped to the auxiliary building floor drain tank. A duplex pump is provided for each sump. The individual pumps are each rated 100 gpm.

Watertight doors have been provided at critical locations in the basement of the auxiliary building to protect safety-related equipment from flooding in different compartments.

3.4.1.3 Description of Structures

The structures that house safety-related equipment are the containment, auxiliary, fuel handling, river screen house at Byron Station; and the lake screen house at the Braidwood Station. These structures all have reinforced concrete walls below grade level.

The only exterior, personnel, or equipment access to these buildings is at grade level or above. All pipes penetrating the exterior walls are provided with watertight penetration sleeves. Water stops are provided in all horizontal and vertical construction joints in all exterior walls, as required. Pumps and drains are located throughout Seismic Category I structures, providing additional protection.

Additional information on structures that house safety-related equipment is as follows:

Byron Station

a. The river screen house is shown in Drawing M-20. Locations of exterior or access openings and penetrations that are below the design flood level of 702 feet are shown.

b. The essential service water cooling towers are shown in Drawings NCT-683-4H and –14H; Drawing M-900, and Drawing S-259(Byron), along with their associated auxiliary electrical equipment rooms. These cooling towers are well above design flood level.

c. The auxiliary building, containment buildings, and their connecting tunnels are shown in Drawings M-5 through M-18.

Braidwood Station

a. That portion of the lake screen house up to and above the exposed 30-inch essential service water lines and the traveling screens are seismically qualified. The spillway elevation is less than the 602 feet 0 inch elevation of the lake screen house operating floor.

b. The auxiliary building, containment building, and their connecting tunnels are shown in Drawings M-5 through M-18.

3.4.2 Analysis Procedures

Design-basis groundwater conditions are applied to all Seismic Category I structures, with the exception of the river screen house (Byron Station) and the lake screen house (Braidwood Station). For these structures design-basis flood conditions are applied. These conditions are set forth in Section 2.4.

The hydrostatic and hydrodynamic loads resulting from groundwater conditions are determined according to procedures described in Subsection 2.5.4.10.5 (Byron Station) and Subsection 2.5.4.10.1.3 (Braidwood Station). Wave loading for the river screen house (Byron Station) and the lake screen house (Braidwood Station) is determined according to provisions of the "Shore Protection Manual," Volumes I, II, III, U.S. Army Coastal Engineering Research Center, Department of the Army, Corps of Engineers, 1977. The properties of the waves considered are given in Subsection 2.4.3.9 (Byron Station) and Subsection 2.4.8.2.6 (Braidwood Station).

All loads are applied to the structures according to the categories and conditions given in Tables 3.8-9 and 3.8-10.

3.5 MISSILE PROTECTION

3.5.1 Missile Selection and Description

The systems located both inside and outside containment have been examined to identify and classify potential missiles. The basic approach is to assure design adequacy against generation of missiles, rather than allow missile formation and try to contain their effects.

3.5.1.1 Internally Generated Missiles (Outside Containment)

The principal design bases are that missiles generated outside of containment but internal to the plant site shall not cause loss of function of any design feature provided for either continued safe operation or shutdown during operating conditions, operational transients, and postulated accident conditions associated with the effects of missile formation. The seismic category and quality group classifications for these systems are identified in Section 3.2.

Equipment has been evaluated for potential missile sources. As a result of this review, the following information concerning potential missile sources and systems which require protection from internally generated missiles outside containment is provided.

Items outside containment which are required for safe shutdown of the reactor must be protected regardless of the missile source.

Systems and components essential for safe shutdown are located remotely, as much as possible and practical, from potential missile sources.

This is always true when the potential missile source is the result of a pressure boundary break requiring operation of the remotely located essential system. In cases where remote location cannot be achieved, physical separation has been employed (e.g., auxiliary feedwater piping is in a separate tunnel below the main feedwater tunnel separated by a structure designed to withstand the impact of a free end pipe "missile" whipping into it).

Redundant systems essential to a safe shutdown are physically separated at a sufficient distance so that the potential for a missile striking both is extremely unlikely.

3.5.1.1.1 Main Turbine and Diesel Generator Missiles

There are no internally generated missiles postulated outside the containment, other than the main turbine missiles described in Subsection 3.5.1.3 and the diesel generator missiles described below.

Failure of an emergency diesel generator may result in generation of internal missiles within the engine itself. Under a limited range of specific conditions, these missiles could penetrate the engine block and strike objects external to the engine. An assessment of these postulated missiles demonstrated that the missiles would not leave the diesel room and would not strike any components within the room which would adversely affect the operability of redundant systems or impair the capability to safely shutdown the plant.

3.5.1.1.2 Evaluation of Valve Stems and Bonnets

All valve stems are either backseated or have plugs, both of which would absorb the kinetic energy of a stem being thrown by impact with the bonnet gland, without exceeding the ultimate strength of retaining components.

Failure of bonnet flange bolts in such a way that the bonnet can become a missile is not credible since both the flanges and bolts/studs are designed to the safety factors inherent in Section III and the relatively higher potential failure mechanisms are such that missiles will not be generated.

The conclusion that valve stems and bonnets should not be considered a credible source of internally generated missiles capable of significantly jeopardizing plant safety is based on the low probability of such a failure and the inherent protection afforded by physical separation of redundant engineered safety feature equipment.

Two estimates of the probability of valve rupture are available from the reports of the Nuclear Plant Reliability Data (NPRD) and from WASH-1400. For the period from July 1974 to December 1977, NPRD reported 1057 failures of valves in 445 million service hours from a population of 32729 valves. None of the failures involved a rupture of the valve body or ejection of the stem, bonnet, or other part of the valve. Three valve body cracks were reported, one being a leak. From this data, it is concluded that an upper limit on the fraction of all valve failures which would be ruptures is 0.00066 to a 50% confidence, or 0.00283 to a 95% confidence. Thus, the upper limit on the probability of valve rupture is, from the same data, 1.57×10^{-9} per hour to 50% confidence, or 6.72×10^{-9} per hour to 95% confidence. WASH-1400 estimated the probability of external leak or rupture to be between 10^{-9} and 10^{-7} per hour. Thus, it is judged that the probability of 1.57×10^{-9} per hour (or 1.4×10^{-5} per year) is a reasonable upper limit.

It is necessary to consider the possible modes of rupture of valves to estimate the likelihood that, given a rupture, a missile would be ejected. As has been shown by NPRD data and general industry experience, rupture is most likely to take the form of a through-wall crack which is detected as a leak long before it could propagate into a serious loss of fluid or missile-generating failure. To be a source of a significant missile, such a crack would have to occur in the bonnet area of a valve, and would have to be a circumferential crack. With the above probability for any such rupture, it is not reasonably credible that such a particular crack could occur and remain undetected for a sufficient time to propagate into a missile generating condition.

The other possibility for valve rupture into a missile generating condition is failure of a bolted closure. Bolted closures, however, are designed to prevent leakage, a more stringent requirement than the prevention of rupture. By its nature, a properly designed bolted closure will leak, and be detected, far below the stress which could lead to a larger failure. A good discussion of the design of bolted closures is in ASME Boiler and Pressure Vessel Code, Section III, Article XII-1100. Bolted closures are also immune to failure analogous to crack propagation because of the independent bolts. Thus, in light of the above upper limit on valve rupture and the nature of bolted closures, such a failure does not appear to be a credible source of missiles.

Stem ejection is a possible source of missiles, but because the NPRD reports do contain such an event, the same upper limit on probability applies. Further, the stem is attached firmly to the valve internals as well as the driving and pressure retaining mechanism in the majority of large valves.

Even though it is not considered credible that a valve failure would lead to the generation of a missile capable of doing significant damage, the Byron and Braidwood Station design is such that a missile of this sort is most unlikely to cause damage which would prevent the safe and orderly shutdown of the affected unit. The separation of the redundant trains of engineered safety features which has resulted from consideration of pipe whip, jet impingement, and fire assures that this situation is very unlikely to occur.

3.5.1.1.3 Evaluation of Other Sources of Missiles

The following items suggested as potential missiles are not credible for the reasons stated:

a. Pressurizer heaters are potential missiles but inasmuch as they would be ejected in a downward direction, no damage to safety-related structures, systems, and components inside the containment would occur. A tabulation of the safety-related structures, systems, and components inside the containment that are required for safe shutdown is provided in Tables 3.5-15 and 3.5-16.

b. The pressurizer relief tank rupture discs are designed such that their failure will not result in the formation of missiles. With rupture, the disc will split into quadrants that will be retained by the disc circumference. The tank is located low in containment outside the secondary shield wall, and disc rupture will not cause failure to either the primary or secondary systems.

c. Instrument wells--All instrument wells forming part of the process piping pressure boundary are connected by a welded construction having a minimum of three times the structural strength required by the Code.

   In addition, those welds in systems in which a break could pose a threat to plant safety (ASME Class 1 and 2 systems) are inspected during fabrication. Postulated missiles resulting from instrument well ejections are not credible.

d. Broken piping--The design of all high and moderate energy piping is such that no free missiles will result as a consequence of postulated pressure boundary breaks (see Section 3.6).

e. Pump impellers--The pump casing will stop any internal missiles resulting from broken impellers.

f. Pump and valve motors--Any internal missiles from broken rotors will be stopped by the stator.

g. Motor-Generator set--The fabrication specifications of the motor-generator set flywheels control the material to meet ASTM-A533-7D, Grade B, Class I with inspections per MIL-I-45208A and flame cutting and machining operations governed to prevent flaws in the material. Nondestructive testing for nil ductility (ASTM-E-208), charpy V-notch (ATM-A593), ultrasonic (ASTM-A578 and A579) and magnetic particles (ASTM Section III, NB2545) is performed on each flywheel material lot. In addition to these requirements, stress calculations are performed consistent with guidelines of ASME Section III, Appendix A, to show the combined primary stresses due to centrifugal forces and the shaft interference fit shall not exceed 1/3 of the yield strength at normal operating speeds (1800 rpm) and shall not exceed 2/3 of the yield strength at 25 percent overspeed. However, no overspeed is expected for the following reason: the flywheel weighs approximately 1300 pounds and has dimensions of 35.26 inches in diameter times 4.76 inches wide. The flywheel mounted on the generator shaft which is directly coupled to the motor shaft is driven by a 200 hp, 1800 rpm synchronous motor. The torque developed by the motor is insufficient for overspeed. Therefore, there are no credible missiles from the MG sets.

Consideration has been given to secondary missiles, but there are none resulting from the missiles listed in Tables 3.5-15 and 3.5-16.

The analysis of the NSSS is conservative because the effects of gravity and friction were not considered. The velocity of potential missiles and the damage caused by missiles would be reduced by inclusion of these factors.

3.5.1.1.4 Analysis of Protective Features for Safety-Related Equipment

Safety-related systems in the auxiliary building are protected against damage from internally generated missiles by the separation, redundancy, and quality standards applied to the design of the Byron/Braidwood Stations. As an example of the approach used, the auxiliary feedwater system has been analyzed in detail and is described here.

The auxiliary feedwater system consists of the motor- and diesel-driven auxiliary feedwater pumps, associated intake and discharge piping, piping in the auxiliary building and auxiliary feedwater tunnel, and the system valves and instrumentation. The auxiliary feedwater piping exits the auxiliary feedwater tunnel and enters the main steam tunnel where it joins the main feedwater piping. On the pump suction side of the system, connections are made to the condensate storage tank as a primary source of feedwater and the essential service water system as a backup source of water.

The components which are postulated to fail resulting in missiles are pumps, pump drivers, valves, and instrument wells. None of these components are actually considered as a potential cause of missiles. However, the plant design incorporates additional mitigating features. Missiles will be postulated to demonstrate the additional margin in the design.

The following events will be postulated to create missiles:

a. Pump impeller failure,
b. Pump driver (motor or engine) failure,
c. Valve failure (valve stem ejection), and
d. Instrument well failure.

A fracture of the pump impeller could result in ejection of fragments. These fragments are not expected to penetrate the pump casing. However, if penetration of the casing is also postulated, the fragments would be stopped by walls of the auxiliary feedwater pump rooms. These rooms enclose both pumps with a 12-inch concrete wall separating the redundant pumps. Each room contains only piping and equipment for one auxiliary feedwater loop with the exception of one short length of loop A piping which travels for a short distance through one corner of pump room B. This pipe is routed against the upper wall of the compartment. The minimum distance between this line and the "B" pump or a "B" valve is 20 and 15 feet, respectively for Unit 1. For Byron, the condensate suction line for the 2A auxiliary feedwater pump is routed a short distance through the corner of the 2B pump room. The minimum distance between this line and the 2B pump or a "B" valve is 7 feet. For Braidwood, the pipe extends approximately 2 feet into the 2B pump room and is capped. For Byron and Braidwood, the condensate system is non-safety related and does not impact the safety related essential service water system, which also is a water supply for the 2A auxiliary feed pump. In addition, the "A" pump recirculation line on both units is routed through the "B" pump room for approximately 20 feet. Potential failure of the recirculation line is evaluated in UFSAR Section 10.4.9.3.1. Therefore, failure of a pump cannot affect the redundant train. In addition to the auxiliary feedwater system, other systems are located in these rooms. The essential service water system and the condensate system extend into these areas to supply water to the pumps. Damage to one of these lines could conceivably impair delivery of water to both trains. However, the service water and condensate piping are separated and in fact enter from opposite sides of the rooms. Check valves in conjunction with the normally open condensate system valves and the normally closed essential service water valves ensure that feedwater will be available in the event of damage to either the service water or condensate system. Fire protection, nonessential service water, and instrument air piping also pass through these areas. Damage to any of these systems will not impair safe shutdown of the plant.

The Byron/Braidwood Stations have one motor-driven auxiliary feedwater pump and one diesel-driven auxiliary feedwater pump per unit. No missiles are expected to result from failure of the motors or diesels. A fragmented rotor will be contained by the stator of the electric motor. Parts ejected following an internal failure of the diesel engine would be contained by the engine crankcase. In the unlikely event of fragments penetrating the stator or the crankcase, damage will be limited to the room enclosing the diesel. The loop "A" pipe mentioned above is located high, such that a fragment from the diesel would have to exit at a high angle. This is not considered credible. The protection afforded by the rooms and the separation of systems is described above.

To address the potential of damage from valve missiles, the Unit 2 valves will be described. There are no significant differences in the Unit 1 valves. Gate valves are used at the junction of the condensate system (Valves 2AF002A/B) and the essential service water system (Valves 2AF006A/B) with the auxiliary feedwater system. These valves are in the low pressure (pump suction) portion of the system and, therefore, no missiles are expected. Additionally, these valves are located in the auxiliary feedwater pump rooms (except for 1AF006A which is located outside the pump room) and thus maintain the separation of the redundant systems.

Gate valves are also installed on the outlet side of the pumps. Valve 2AF004A is just outside of pump room A and separated from all parts of the B loop by at least a 12 inch concrete wall. Valve 2AF004B is in pump room B and is on the opposite side of the diesel from the loop A pipe mentioned above. Both valves are oriented such that the stem is vertical and no safety system is directly above the valve.

For Byron valve 2AF036 is in pump room A and separated from all parts of the B loop by at least a 12 inch concrete wall and thus maintains the separation of the redundant systems. The valve is oriented such that the stem is horizontal and no safety system is directly in line with or potentially impacted by the valve stem. For Braidwood, valves 1/2AF036 are in pump room A and separated from all parts of the B loop by at least an 11 inch block wall and thus maintains the separation of the redundant systems.

The auxiliary feedwater lines are routed across the auxiliary building after exiting the pump rooms. The lines are not adjacent to any high energy lines. The 2A auxiliary feedwater pump safety-related discharge line is routed directly above the seismically supported safety-related essential service water line, which is routed to the suction of the 2A auxiliary feedwater pump, for approximately 20 feet. Both the auxiliary feedwater and the essential service water lines are seismically supported and are located outside of the auxiliary feedwater pump room to protect them from potential missiles from the 2B auxiliary feedwater pump or engine. The lines contain no valves or instrument wells until the pipes (now eight; two redundant loops per each steam generator) turn down at column row 26 just prior to entering the auxiliary feedwater tunnel. At this point is a bank of globe valves (2AF005A-H) which are used as control valves for the auxiliary feedwater flow. These valves are mounted in a vertical pipe run. The stems are oriented such that they do not point at any part of the auxiliary feedwater or other safety-related systems. The lines and valves are arranged at this point such that redundant lines (to the same steam generator) are separated by at least two other lines.

The auxiliary feedwater lines then travel the length of the auxiliary feedwater tunnel. There are no other valves or instruments in the line except for the globe isolation valves (Valve 2AF013A-H) which are located in the tunnel just before the lines penetrate the main steam tunnel and join the main feedwater line. These valves are all oriented such that the valve stems are vertical and no safety-related systems are routed above these valves. Check valves in the auxiliary feedwater system are not considered as potential missiles because there is no credible failure of a check valve which would create a missile.

The only instrument well in the high pressure portion of the system is the temperature sensor at the exit of the pump. Instrument wells are not expected to fail in a manner resulting in missiles because this attachment is in a Category I Safety Class C pipe and meets code standards. The well itself is relatively small and would not be expected to damage pipe and equipment. Even if a missile is postulated, the instrument wells are in the pump rooms close to the pumps and could not affect the redundant system.

The investigation has established that, even with unrealistically conservative postulation of missiles, the auxiliary feedwater system will not be susceptible to common mode failures and will not pose a danger to other safety-related systems because of internally generated missiles.

3.5.1.1.5 Protection From Falling Objects

All piping in the containment and the auxiliary building, including Category II piping, has been seismically supported. Therefore, no falling objects are postulated due to seismic events. A potential does exist for damage due to the dropping of an object being transported during maintenance or refueling activities. This potential has been addressed in accordance with the guidelines of NUREG-0612 in a separate heavy loads assessment. See Section 9.1.5 (Control of Heavy Loads) for additional details.

Secondary missiles are not considered as credible occurrences. As described in Subsection 3.5.1.1.2, there are very few credible primary missile sources. These have been considered in the design, and will strike barriers which will not result in secondary missiles. When one considers the low probability of primary missile generation, the low probability of a primary missile striking a potential secondary missile and the relatively lower energy necessarily associated with the secondary missile, it is concluded that no credible secondary missiles exist.

3.5.1.2 Internally Generated Missiles (Inside Containment)

The principal design bases are that missiles generated within the reactor containment, in coincidence with a loss-of-coolant accident, shall not cause loss of function of any redundant engineered safety feature. The seismic category and quality group classifications for NSSS and components can be found in Section 3.2.

Equipment inside containment has been evaluated for potential missile sources. As a result of this review, the following information concerning potential missile sources is provided.

The reactor seal ring is utilized only during refueling and is not stored on the vessel during power operation. It does not need to be considered a missile during a LOCA.

3.5.1.2.1 Missile Selection

In addition to the types of missiles not considered credible as described in Subsection 3.5.1.1, catastrophic failure of the reactor vessel, steam generators, pressurizer, reactor coolant pump casings, and piping leading to generation of missiles is not considered credible. Massive and rapid failure of these components is incredible because of the material characteristics, inspections, quality control during fabrication, erection and operation, conservative design and prudent operation as applied to the particular component. The reactor coolant pump flywheel is not considered a source of missiles for the reasons discussed in Subsection 5.4.1.5. Nuts and bolts are of negligible concern because of the small amount of stored elastic energy.

Components which, nevertheless, are considered to have a potential for missile generation inside the reactor containment, are the following:

a. control rod drive mechanism housing plug, drive shaft, and the drive shaft and drive mechanism latched together;

b. certain valves;

c. temperature and pressure sensor assemblies, and

d. pressurizer heaters.

Gross failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

a. Control rod drive mechanisms are shop-tested at 4105 psig.

b. The mechanism housings are individually hydrotested to 3107 psig after they are installed on the reactor vessel to the head adapters, and checked again during the hydrotest of the completed reactor coolant system.

c. The mechanism housings are made of Type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

However, it is postulated that the top plug on the control rod drive mechanism will become loose and it will be forced upward by the water jet. The following sequence of events is assumed: the drive shaft and control rod cluster are forced out of the core by the differential pressure of 2500 psig across the drive shaft. The drive shaft and control rod cluster, latched together, are assumed fully inserted when the accident starts. After approximately 12 feet of travel, the rod cluster control spider hits the underside of the upper support plate. Upon impact the flexure arms in the coupling joining the drive shaft and control rod cluster fracture, completely freeing the drive shaft from the control rod cluster. The control cluster would be completely stopped by the upper support plate; however, the drive shaft would continue to be accelerated upward to hit the missile shield provided.

Valve stems are not considered credible sources of missiles.

All the isolation valves installed in the reactor coolant system have stems with a back seat. This effectively eliminates the possibility of ejecting valve stems even if the stem threads fail.

Valves with nominal diameter larger than 2 inches have been designed against bonnet body connection failure and subsequent bonnet ejection by means of:

a. Using the design practice of ASME Boiler and Pressure Vessel Code, Section III, and

b. By controlling the load during the bonnet body connection stud tightening process.

Pressure retaining parts are designed per the Class 1 requirements established by the ASME Section III Code.

The proper stud torquing procedures limit the stress of the studs to the allowable limits established in the ASME Code.

This stress level is far below the material yield. The valves are hydrotested per the ASME Section III Code. The bodies and bonnets are volumetrically and surface tested to verify soundness.

Whereas valve missiles are not generally postulated due to the above discussion, exceptions are the valves in the region where the pressurizer extends above the operating deck. Valves in this region are the pressurizer safety valves, the motor-operated isolation valves in the relief line, the air-operated relief valves and the air-operated spray valves. Although failure of these valves should also be considered incredible, failure of the valve bonnet body bolts is, nevertheless, postulated and provisions are made to assure integrity of the containment liner from the resultant bonnet missile.*

The only credible source of jet-propelled missiles from the reactor coolant piping and piping systems connected to the reactor coolant system is that represented by the temperature and pressure sensor assemblies. The resistance temperature sensor assemblies can be of two types: "with well" and "without well." Two break locations have been postulated: around the weld (or thread) between the temperature element assembly and the boss for the "without well" element, and the weld (or thread) between the well and the boss for the "with well" element.

A temperature sensor is installed on the reactor coolant pumps close to the radial bearing assembly. A hole is drilled in the gasket and sealed on the internal end of a steel plate. In evaluating missile potential, it is assumed that this plate could break and the pipe plug on the external end of the hold could become a missile.

In addition, it is assumed that the welding between the instrumentation well and the pressurizer wall could fail and the well and sensor assembly could become a jet-propelled missile.

Finally, it is assumed that the pressurizer heaters could become loose and become jet-propelled missiles.

3.5.1.2.2 Missile Description

The control rod drive mechanism (CRDM) missiles are summarized in Table 3.5-1. The velocity of the missiles have been calculated by balancing the forces due to the water jet. No spreading of the water jet has been assumed.

The missile characteristics of the bonnets of the valves in the region where the pressurizer extends above the operating deck are given in Table 3.5-2a.

The missile characteristics of the piping temperature sensor assemblies are given in Table 3.5-2b. A 10 degree expansion half angle water jet has been assumed. The missile characteristics of

*To the extent practical, all valves are also oriented such that any missile will strike a barrier the piping pressure element assemblies are less severe than those of Table 3.5-2b.

The missile characteristics of the reactor coolant pump temperature sensor, the instrumentation well of the pressurizer, and the pressurizer heaters are given in Table 3.5-2c. A 10 degree expansion half angle water jet has been assumed.

3.5.1.3 Turbine Missiles

The turbine-generators at the Byron/Braidwood Stations are manufactured by the Westinghouse Electric Corporation. Each unit consists of four double-flow turbine cylinders: one high pressure, and three low pressure. The low pressure stages employ 40-inch last row blades. The rated speed of the turbine-generator is 1800 rpm.

The current approach to evaluating turbine missile protection focuses on the probability of turbine failure resulting in the ejection of turbine disc (or internal structure) fragments through the turbine casing (P_1). A risk assessment will be performed each refueling outage to ensure that the probability of a turbine missile, P_1, remains at an acceptably low value. Based on this low probability, the turbine missile hazard is not considered a design-basis event for these stations. The details of the approach to ensure turbine missile protection are provided in Section 10.2.3.

For details on turbine overspeed protection, valve testing, and turbine characteristics, refer to Subsection 10.2.2.

3.5.1.4 Missiles Generated By Natural Phenomena

Tornadoes are the only natural phenomenon occurring in the vicinity of the Byron/Braidwood Stations that can generate missiles. The characteristics of postulated design-basis tornado-generated missiles are given in Table 3.5-3. The impact velocities of these missiles resulting from the design-basis tornado (Subsection 3.3.2) are shown in Table 3.5-4. Missiles A, B, C, D, and E are considered at all elevations, and missiles F and G are postulated at elevations up to 30 feet above grade level. These missiles are assumed to be capable of striking in all directions.

The characteristics of tornado-generated missiles considered in the probabilistic tornado missile risk analysis (TORMIS) described in Byron only Section 3.5.5 are found in Reference 15.

3.5.1.5 Missiles Generated by Events Near the Site

Based on a review of the nearby industrial, transportation, and military facilities (as described in Section 2.2), it is concluded that there are no potential missiles resulting from accidental explosions in the vicinity of the site.

3.5.1.6 Aircraft Hazards

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3.5.1.5 Missiles Generated by Events Near the Site

As described in Section 2.2, an accidental explosion of TNT is not a credible event. Therefore, missiles due to a TNT accident are not a design basis load.

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3.5.2 Systems to be Protected

All systems and equipment which may require protection are listed in Table 3.2-1. Onsite storage locations for compressed gases are provided in Table 3.5-10. Table 15.1-2 must be evaluated for protection against missiles postulated in Section 3.5.

The following safety-related components are located outdoors, away from the main building complex, installed above grade and have missile protection to the extent indicated:

Byron Station

a. At the river screen house, the essential service water makeup pumps, and associated diesel-engine drives and fuel oil storage tanks are installed at elevation 702 feet 0 inch. The building does not protect the components from tornado missiles. Refer to Subsection 9.2.5.2 and Drawing M-20.

b. The mechanical draft fans and their respective electric motor drives are located at the essential service water cooling towers (SXCTs) (refer to Drawings NCT-683-4H and -14H). The fans and motors are not fully protected from missiles and are evaluated in Section 3.5.5. A combination of TORMIS analysis and tornado protection was used for the piping within the SXCTs.

c. The outside air intake openings for the SXCT ESF Switchgear rooms are not protected from a tornado missile and are evaluated in Section 3.5.5.

d. The onsite wells and pumps at Byron, although not safety-related, are each protected by missile-proof walls and roofs. The onsite wells supply makeup water to the SXCTs in the event that a tornado missile renders the essential service water makeup pumps inoperative. Missile protected check valves (0SX284A/B) are installed in the essential service water makeup lines to prevent back flow from the SXCT basins to the river screen house.

e. Safety-related electrical cables are adequately protected against tornado-generated missiles by the reinforced concrete ducts around them. Embedded conduits in the auxiliary building south wall and associated cable vaults supporting operation of the SXCTs and deep well pumps are evaluated in Section 3.5.5.

Braidwood Station

There are no safety-related components located outdoors at Braidwood Station.

All safety-related electrical components which are located outdoors are listed in Subsection 8.3.1.4.4 (Class 1E Equipment in Remote Structures).

All Category I buried pipes on Byron/Braidwood sites and the Category II (Non-Safety Related) Well Water (WW) piping from the onsite wells and pumps to the SXCTs at Byron have adequate soil cover for protection from tornado-generated missiles. These pipes are buried to depths greater than the required minimum depth of 4 feet 1 inch, determined using Young's method.

Safety-related HVAC system air intakes and exhausts are indicated on the plant arrangement Drawings M-5, M-6, M-14, M-15 and M-22-2.

Auxiliary Building and Containment Purge (VA, VQ)

Intakes

Intake louvers are shown as listed above. Protection is provided by missile walls.

Exhaust

The exhaust stacks are shown as listed above. Vertical stack connected to horizontal exhaust tunnel affords missile protection.

Diesel-Generator Room Intake (VD)

The diesel-generator room intake is shown as listed above. Protection is provided by missile walls.

Safety-related electrical cables are adequately protected against tornado-generated missiles by the reinforced concrete ducts around them.

Control Room Intake (VC)

The control room intake is shown as listed above. Protection is provided by missile walls. The Byron control room turbine building makeup air intakes are evaluated in Byron only Section 3.5.5.

3.5.3 Barrier Design Procedures

Two types of structural response to missile impact have been investigated for the design-basis tornado-generated missiles in Section 3.5.1.4:

a. local effect in the impacted area which includes estimation of the depth of penetration and, in the case of concrete barriers, the potential for secondary missiles by spalling or scabbing; and

b. overall response of the barrier which includes the calculation of deflection due to missile impact.

The design-basis tornado-generated missile velocities presented in Table 3.5-4 are based on TVA Topical Report TVA-TR74-1. Commonwealth Edison committed to design to these velocities during the PSAR review. The staff had accepted these velocities as indicated in Revision 0 of the Standard Review Plan 3.5.1.4.

Draft Rev. 1 of SRP 3.5.1.4 states: "At the operating license stage, applicants who were not required at the construction permit stage to design to the missile spectrum of Rev. 0 of this SRP and the corresponding velocity set, should show the capability of the existing structures and components to withstand at least missiles C and F of the Rev. 0 to this SRP." It can be noted from Table 3.5-4 that the horizontal velocity of the missile F, utility pole - used in the Byron/Braidwood design is larger than the velocity of 211 fps (i.e., 0.4 times the total tornado velocity) specified in Revision 0 missile spectrum. The velocity for the steel rod is however lower than the Revision 0 spectrum velocity of 317 fps.

(Braidwood only) The walls and roofs of structures protecting the safety-related systems and components from tornado-generated missiles are of reinforced concrete with minimum thickness of 24 and 14 inches respectively. The concrete used has a minimum cylinder strength of 3500 psi at 91 days.

(Byron only) The walls and roofs of structures protecting the safety-related systems and components from design-basis tornado-generated missiles are of reinforced concrete with thicknesses shown by analysis to prevent scabbing, spalling or penetration. The concrete used has a minimum cylinder strength of 3500 psi at 91 days.

The roof and wall thickness required to prevent back-face scabbing as a result of impact from the steel rod missile have been calculated using the modified National Defense Research council formula as 5.5 and 6.5 inches respectively. Since the minimum barrier thicknesses provided at Byron/Braidwood are much larger than the above values, it is concluded that the plant structures, systems and components are adequately protected against the Revision 0 spectrum of design-basis tornado-generated missiles.

Generally, all missiles (internal or external) are considered as impacting instantaneously with a very short rise time relative to the natural period of the impacting structure. Types of barriers designed to resist missile impact are:

a. Reinforced Concrete Barriers - The depth of penetration into a concrete barrier is calculated using the modified Petry equation (Reference 6). Concrete barriers are designed such that the missile penetrates no more than two-thirds of the thickness of the barrier thus preventing spalling or scabbing (Reference 6). The overall deformation of the panel is investigated using methods presented in Reference 7. Reference 7 presents an equation of motion which enables one to calculate an impact force time-history consistent with the calculated penetration depth. To establish the capacity of the barrier to absorb energy, the deflection due to static loads are first calculated, then the deflection due to missile impact is determined by integrating the equation of motion or by using a simplified expression adopted from the equation of motion. This is compared with the maximum allowable deflection (or allowable ductility ratio), in accordance with ACI-349 (Reference 10).

   The design of concrete members under impactive loads from design-basis tornado missiles is in compliance with Appendix A to SRP Section 3.5.3. Those members which see impactive and impulsive loads due to pipe breaks have been designed using a nonlinear analysis assuming hinge rotations not exceeding 0.07 radians. The experimental data on which this method is based is in Reference 13 (a PCA Bulletin).

b. Steel Plate Barriers - The thickness of steel plate required to resist the impacting design-basis missile is calculated using the Stanford Formula (Reference 8). The overall structural response, including structural stability and deformation is investigated using concepts and methods presented in Reference 9.

c. Control Rod Drive Missile Shield - A missile shield structure is provided over the control rod drive mechanisms to block missiles which might be associated with a fracture of the pressure housing of any mechanism. This missile shield is a reinforced steel structure attached to the reactor vessel head and located above the CRDMs. Each CRDM housing is terminated with a small tapered pin which penetrated the missile shield through a slightly larger diameter hole to direct the ejected CRDM missile into the shield. This prevents any missile from missing or ricocheting from the shield to strike the containment liner or other CRDM housings.

   The walls of the refueling cavity protect the CRDMs from missiles originating from the horizontal direction.

   Missile shield penetrations are given in Table 3.5-1 using the Ballistic Research Laboratories (BRL) formula for steel. The steel missile shield has an effective thickness of approximately 3 inches.

   For the case of housing plug and drive shaft impact, which is the design case, it is assumed that the plug partially perforates the missile shield. The drive shaft then hits the plug and further penetrates the steel missile shield. The resultant penetration into the shield is 0.773 inches. Therefore, the effective thickness of the steel missile shield is more than three times the combined penetration value for the design case.

   The CRDM missile shield is also designed to withstand the dynamic impact loads due to the missile and the water jet.

It is to be noted that the location of the secondary shield wall inside the containment structure is such that no potential missile will strike the containment liner.

3.5.4 Analysis of Missiles Generated by a Tornado

Effects of tornado missiles have been assessed for safety-related components located outdoors. These components are the SXCTs (Byron only), the emergency diesel generator exhaust stacks, the emergency diesel generator ventilation and combustion air intakes, the emergency diesel generator crankcase vents, and the main steam safety and power operated relief valve tailpipes (Braidwood only).

3.5.4.1 Essential Service Water Cooling Towers (Byron)

A temperature and inventory analysis of the UHS after the loss of SXCT fans due to tornado-generated missiles was performed. The analysis also considers out of service fans and postulated single failures. The number of fans lost due to tornado missiles is based on the results of the TORMIS analysis described in Section 3.5.5.

Based on the results of the TORMIS analysis, the deep well pumps remain available to provide makeup water if the SX makeup pumps are damaged during the tornado event.

The analysis was performed using SXCT performance curves generated using the method described in Section 9.2.5.3.1.1.2. Various outside air wet bulb temperatures were considered in the analysis. The results of the analysis are used to establish operating limits on the number of SXCT fans required to be operable based on the outside air wet bulb temperature and number of units operating. The analysis credits the following operator actions:

a. Manual initiation of the deep well pump(s) is assumed to occur 1.5 hours into the event,

b. Isolation of essential service water blowdown within two hours,

c. Isolation of the auxiliary feedwater telltale drains within two hours, and

d. Isolation of the SXCT riser leakoff drains within two hours

The analyses determined the SXCTs are capable of providing adequate heat removal and timely safe shutdown of both units.

3.5.4.2 Emergency Diesel Generator Exhaust Stacks

The diesel generator exhausts are completely protected up to the point where they penetrate the tornado proof concrete enclosure on the auxiliary building roof. Above this point, they are exposed for about 35 feet as they travel vertically. Analysis has established that the stacks can be damaged to the extent that the flow area is reduced to 50% of the original flow area without reducing the diesel power output (Braidwood only).

To prevent loss of diesel availability due to the exhaust stack damage, a rupture disc pressure relief device is installed on each diesel exhaust line. This relief device is located downstream of the silencer and inside the missile protection structure on the roof of the auxiliary building. Upon blockage of the stack, the rupture disc will open prior to backpressure increasing to the point that required diesel power is not available. The emergency diesel generators will therefore remain functional following any postulated tornado missile impact.

3.5.4.3 Emergency Diesel Generator Ventilation and Combustion Air Intakes and Crankcase Vents

Ventilation and combustion air for the emergency diesels is inducted through tornado proof intakes in the auxiliary building roof. The emergency diesel engine crankcase vents are exposed to tornado missiles. Reference 14 demonstrates that the crankcase vent lines can be blocked without adversely affecting the ability of the associated diesel to perform its design function.

3.5.4.4 Fuel Handling Building Railroad Freight Door

The railroad freight door is not designed to be tornado proof. In the event the door is missing or open, missiles would potentially enter the tunnel to the fuel handling building. To reach the fuel handling area, missiles would have to travel over 100 feet down the tunnel which is approximately 25 feet square. The two most vulnerable areas are the fuel pool heat exchangers on the lower level and fuel storage area on the upper level. After negotiating the tunnel, the missile would have to make a 90 degree turn and penetrate a wall to damage either of the heat exchangers (which are redundant) or make two 90 degree turns (up and right) to reach the fuel storage area. Based on this assessment, it is concluded that tornado missiles pose no hazard to the fuel handling building.

3.5.4.5 Main Steam Safety and Power Operated Relief Valve Tailpipes

The main steam safety valve 16 inch diameter 1/2 inch thick tailpipes extend approximately 1 foot above the MSSV roof and a maximum of 1.5 inches above a guard pipe. The guard pipe is a 20 inch diameter, 1/2 inch thick pipe. The combination of a short height above the roof and the installed guard pipe practically eliminates the possibility of a horizontal missile denting or crimping the MSSV tailpipes. The exhaust of the power operated relief valve is in a recessed area between the valve room upper roof and the containment wall, and is, therefore, protected from horizontal missiles.

The Byron main steam safety and power operated relief valve tailpipes are evaluated in Byron only Section 3.5.5.

3.5.4.6 Essential Service Water Discharge Extension Lines (Braidwood only)

The Non-Safety Related portions of the Essential Service Water (SX) discharge extension lines extend approximately 3 feet above lake level and they are attached to the safety related portion of the SX discharge piping at the discharge structure via a flanged connection. Analysis has demonstrated that the flange connection will fail and separate before the SX extension pipe stress reaches the yield stress of the pipe material, i.e. the extension pipe section will remain in the elastic behavior zone without plastic deformation due to a force generated by a horizontal missile. Therefore, the SX discharge flow path is not adversely impacted by a Tornado generated horizontal missile striking the exposed SX discharge extension lines.

3.5.5 Probabilistic Tornado Missile Risk Analysis

A probabilistic tornado missile risk analysis (Reference 15) was completed for Byron using the TORMIS computer code which is based on the NRC approved methodology detailed in References 16, 17 and 18. The TORMIS analysis was performed in accordance with the guidance described in the NRC TORMIS Safety Evaluation Report (Reference 19) and as clarified by Regulatory Issue Summary (RIS) 2008-14 (Reference 20).

3.5.5.1 Scope

The TORMIS analysis (Reference 15) includes plant components, identified as necessary to safely shutdown the plant and maintain a shutdown condition, located in areas not fully protected by missile barriers designed to resist impact from design-basis tornado missiles. The targets included in the TORMIS analysis are listed in Table 3.5-17 and additional details regarding targets (i.e., specific location and identification) are included in Reference 15, Volume 3.

3.5.5.2 Computer Codes

3.5.5.2.1 TORMIS

TORMIS (TORnado MISsile Risk Analysis Methodology Computer Code) uses a Monte Carlo simulation method that simulates tornado strikes on a plant. For each tornado strike, the tornado wind field is simulated, missiles are injected and flown (including vertical and near vertical missile impacts), and missile impacts on structures and equipment are analyzed. These models are linked to form an integrated, time-history simulation methodology.

By repeating these simulations, the frequencies of missiles impacting and damaging individual components (targets) and groups of targets are estimated. Statistical convergence of the results is achieved by performing multiple replications with different random number seeds.

3.5.5.2.2 TORRISK

TORRISK (TORnado RISK Analysis Methodology Computer Code) is a specialized version of TORMIS that produces tornado hazard curves distinct from the missile risk analysis features of TORMIS. TORRISK is a fast-running version of TORMIS and was spun-off in 1983 specifically for the purpose of tornado wind probability analysis for the different types of geometrical targets, like points, buildings, sites and transmission lines. TORRISK uses the same tornado input data as TORMIS and produces tornado wind hazard risks only. TORRISK produces a more accurate wind hazard curve than TORMIS since it is not encumbered with all of the TORMIS missile simulation variance reduction methods.

3.5.5.2.3 TORSCR

TORSCR is a FORTRAN computer code that is used to post-process TORMIS output files. Its primary function is to compute Boolean combinations of target hit and damage probabilities over multiple targets.

3.5.5.2.4 LS-DYNA

LS-DYNA is a nonlinear explicit finite element code for the dynamic analysis of structures. Since 1987, the LS-DYNA code has been extensively developed and supported by the Livermore Software Technology Corporation and is used for a wide variety of crash, blast and impact applications. LS-DYNA was used to develop missile threshold damage velocities for selected targets which are used as an input in the TORMIS model.

3.5.5.3 Analysis

The Byron TORMIS tornado missile risk analysis results show that the arithmetic sum of damage frequencies for all target groups affecting the individual units (i.e., Unit 1 plus common components and Unit 2 plus common components) are lower than the acceptable threshold frequency of 1.0E-06 per year established in SRP Section 2.2.3 and Reference 21.

The following limiting inputs and assumptions were used in the analysis (refer to Reference 15 for additional assumptions and engineering judgments used in the analysis):

a. A site specific tornado hazard curve and data set for Byron was developed using statistical analysis of the NOAA/National Weather Service Storm Prediction Center tornado data for the years 1950 thru 2013. The analysis utilizes the Enhanced Fujita (EF) scale wind speeds in the TORMUS simulations.

b. A TORMIS wind profile (#3) that adequately models increased near ground wind speeds

c. The missile characteristics and locations are based on a plant walk down survey and plant drawings. The plant walk down survey was performed during a unit outage to capture both non-outage and outage conditions during the survey. A stochastic (time-dependent) model of the missile population is implemented in TORMIS. The stochastic approach to the missile population varies the missile populations in each of the TORMIS replications to account for predictable changes in plant conditions (i.e., increased missiles during outages) and the randomness inherent in the total number of missiles present at the plant at any given time.

d. Finite element calculations were performed to provide the missile damage threshold velocity for each missile type to cause unacceptable crimping damage for the SXCT riser pipes, diesel driven auxiliary feedwater pump exhaust pipes and cover plates and the main steam power operated relief valve tailpipes.

e. For the UHS, one or two SXCT cells are assumed to be randomly out of service for maintenance. A postulated single failure of an electrical bus is assumed resulting in the loss of power to two additional SXCT cells. For the TORMIS analysis, success is defined as at least 3 of the remaining 5 cells surviving when one cell is out of service or 2 of the remaining 4 cells surviving when two cells are out of service.

The arithmetic sum of damage frequencies for all target groups affecting the individual units would exceed the acceptance criteria of 1.0E-06 per year per unit. Boolean combinations of targets were developed to aid in summarizing the results and understanding the effects of system redundancies. This approach yielded acceptable results. Boolean Logic is applied to target groups to account for redundancy in the structural or system design or TORMIS modeling of a component as multiple targets. With redundancy in the design, the system function could be met even with one or more individual targets damaged by postulated tornado missiles. The Boolean intersection operator was used for UHS targets to credit redundancy in the design. The logic is applied to each TORMIS simulated tornado to determine if the missile damage results in a loss of function of the target group.

There was a single change made to the TORMIS code of a purely "software" nature which was not related to the approved TORMIS physics engine and calculation approach; i.e., the dimensioned number of possible missile types was increased to 24 for evaluation of damage from missile velocity exceedance and pipe penetration pass through.

3.5.6 References

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19. NRC Safety Evaluation Report - "Electric Power Research Institute (EPRI) Topical Reports Concerning Tornado Missile Probabilistic Risk Assessment (PRA) Methodology," dated October 26, 1983 (ML080870291).

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21. Memorandum from Harold Denton, NRR Director to Victor Stello, Deputy Executive Director for Regional Operations and Generic Requirements, "Position of Use of Probabilistic Risk Assessment in Tornado Licensing Actions," dated November 7, 1983 (ML030020331).

TABLE 3.5-1

SUMMARY OF CONTROL ROD DRIVE MECHANISM MISSILE ANALYSIS

	POSTULATED MISSILE	WEIGHT (lb)	THRUST AREA (in^2)	EFFECTIVE IMPACT AREA (in^2)	IMPACT VELOCITY (fps)	KINETIC ENERGY (ft-lb)	PENETRATION (in.)
1.	Mechanism Housing Plug	50	4.91	0.87	40	1242	0.163
2.	Drive Shaft	165	2.40	3.56	100	25,620	0.773[2]
3.	Drive Shaft latched to Mechanism	1610	12.57	1.37	12	3,600	0.265

NOTES:

1. Ballistic Research Laboratories (for steel)

2. Assumes drive shaft impact drives housing plug further into missile shield.

TABLE 3.5-2a

VALVE - MISSILE CHARACTERISTICS

MISSILE DESCRIPTION	WEIGHT (lb)	FLOW DISCHARGE AREA (in^2)	THRUST AREA (in^2)	IMPACT AREA (in^2)	WEIGHT TO IMPACT AREA RATIO (psi)	VELOCITY (fps)
Safety Valve Bonnet (3" x 6" or 6" x 6")	350	2.86	80	24	14.6	110
3-inch Motor-Operated Isolation Valve Bonnet (plus motor and stem)	100	5.5	113	28	14.1	135
3 Inch Air-Operated Relief Valve Bonnet (plus stem)	75	1.8	20	20	3.75	115
4-Inch Air-Operated Spray Valve Bonnet	200	9.3	50	50	4	190

TABLE 3.5-2b

PIPING TEMPERATURE ELEMENT ASSEMBLY - MISSILE CHARACTERISTICS

1. For a tear around the weld between the boss and the pipe:

CHARACTERISTICS	"WITHOUT WELL"	"WITH WELL"
Flow Discharge Area	0.11 in^2	0.60 in^2
Thrust Area	7.1 in^2	9.6 in^2
Missile Weight	11.0 lb	15.2 lb
Area of Impact	3.14 in^2	3.14 in^2
$\left[\frac{\text{Missile Weight}}{\text{Impact Area}}\right]$	3.5 psi	4.84 psi
Velocity	20 ft/sec	120 ft/sec

2. For a tear at the junction between the temperature element assembly and the boss for the "without well" element and at the junction between the boss and the well for the "with well" element:

CHARACTERISTICS	"WITHOUT WELL"	"WITH WELL"
Flow Discharge Area	0.11 in^2	0.60 in^2
Thrust Area	3.14 in^2	3.14 in^2
Missile Weight	4.0 lb	6.1 lb
Area of Impact	3.14 in^2	3.14 in^2
$\left[\frac{\text{Missile Weight}}{\text{Impact Area}}\right]$	1.27 psi	1.94 psi
Velocity	75 ft/sec	120 ft/sec

TABLE 3.5-2c

CHARACTERISTICS OF OTHER MISSILES

POSTULATED WITHIN REACTOR CONTAINMENT

MISSILE CHARACTERISTICS	REACTOR COOLANT PUMP TEMPERATURE ELEMENT	INSTRUMENT WELL OF PRESSURIZER	PRESSURIZER HEATERS
Weight	0.25 lb	5.5 lb	15 lb
Discharge Area	0.50 in^2	0.442 in^2	0.80 in^2
Thrust Area	0.50 in^2	1.35 in^2	2.4 in^2
Impact Area	0.50 in^2	1.35 in^2	2.4 in^2
$\left[\frac{\text{Missile Weight}}{\text{Impact Area}}\right]$	0.5 psi	4.1 psi	6.25 psi
Velocity	260 fps	100 fps	55 fps

TABLE 3.5-3

DESIGN-BASIS TORNADO-GENERATED MISSILES AND THEIR PROPERTIES

MISSILE		WEIGHT (lb)	CROSS SECTION	LENGTH (ft)	HEAD ON CONTACT AREA (in^2)
A.	Wood	200	4 in. x 12 in.	12	48.00
B.	3-inch Schedule 40 steel pipe	78	3.5 in. OD	10	9.62
C.	Steel rod	8	1-in. diameter	3	0.79
D.	6-inch Schedule 40 steel pipe	285	6.625 in. OD	15	34.50
E.	12-inch Schedule 40 steel pipe	743	12.75 in. OD	15	127.70
F.	Utility pole	1,490	13.5 in. diameter	35	143.10
G.	Automobile	4,000	--	--	2,880.00

TABLE 3.5-4

IMPACT VELOCITIES OF DESIGN-BASIS TORNADO-GENERATED MISSILES

MISSILE	HORIZONTAL IMPACT VELOCITY* (fps)
A. Wood plank (4 in. x 12 in. x 12 ft, weight 200 lb)	368
B. Steel pipe (3 in. diameter, Schedule 40, 10 ft long, weight 78 lb)	268
C. Steel rod (1 in. diameter x 3 ft long, weight 8 lb)	259
D. Steel pipe (6 in. diameter, Schedule 40, 15 ft long, weight 285 lb)	230
E. Steel pipe (12 in. diameter, Schedule 40, 15 ft long, weight 743 lb)	205
F. Utility pole (13.5 in. diameter, 35 ft long, weight 1490 lb)	241
G. Automobile (frontal area 20 ft^2, weight 4000 lb)	100

* Vertical impact velocities are taken equal to 80% of the horizontal impact velocities.

TABLE 3.5-5

DATA FOR AIRCRAFT CRASH PROBABILITY ANALYSIS

AIRPORT	OPERATION MODE	DIRECTION OF TRAVEL	r (miles)	DEGREES	D(r,θ) (per miles^2)	R (accidents per operation)	N (operations per year)	A (miles^2)	NARD (accidents per year)

Security - Related Information Withheld Under 10 CFR 2.390

TABLE 3.5-6

IFR TRAFFIC COUNTS ON FEDERAL AIRWAYS

V156 AND V429 FOR CHICAGO CENTER

FISCAL YEAR: / PEAK DAY:	1975 9/12/74	1976 9/17/75
Number of flights past site on V156 on peak day	41	27
Number of flights past site on V429 on peak day	25	35
Total number of flights past site on peak day	66	62
Ratio of average day traffic count to peak day traffic count for Chicago section	0.70	0.67
Total number of flights past site on average day	46	42

Source: References 2 and 3 (FAA)

TABLE 3.5-7

EFFECTIVE TARGET AREAS, A_i

	EFFECTIVE TARGET AREA IN mi^2		
AIRCRAFT TYPE	FUEL HANDLING BUILDING	CONTAINMENT	TOTAL

Security - Related Information Withheld Under 10 CFR 2.390

TABLE 3.5-8

AIRCRAFT AVERAGE MOVEMENTS ON AIRWAYS, N_i

NUMBER OF FLIGHTS PAST SITE ON AVERAGE DAY (IFR)	TOTAL (IFR+VFR) PER DAY	SINGLE-ENGINE AIRCRAFT	TWIN-ENGINE AIRCRAFT	AIR CARRIER	MILITARY HELICOPTER	MILITARY JET
44	66	47	Pleasure - 0.8	2.6	2.8	2.0
			Business - 7.5			
			Air taxi - 3.7			

TABLE 3.5-9

RELATIVE FREQUENCIES OF ACCIDENTS

CATEGORY	PERCENTAGE FREQUENCY, C_i
General - Single-engine	51.4
General - Twin-engine:	
Pleasure	35.7
Business	3.3
Air taxi	7.2
Air carrier	0.2
Military helicopter	1.0
Military jet	1.0

TABLE 3.5-10

COMPRESSED GASES STORED ONSITE

GAS STORED	QUANTITY	PURPOSE	LOCATION STORED
CO_2	185,000 scf	Fire protection	Turbine building, elevation 401, columns K-L and rows 16-18
CO_2	37,000 scf	Fire protection	Byron river screen house
H_2	260,000 scf	Generator cooling; volume control tank blanketing	Outdoors
N_2 (liquid)	3110 gallons	Inerting; cover gas	Outdoors
N_2 (gas)	5550 scf	Inert gas blanketing; accumulator pressurization	Auxiliary building at elevation 364, column row Q, rows 10-12
Air	1200 scf	Motive fluid for operating pressurizer power-operated relief valves	Containment buildings outside missile barrier
N_2 (gas)	2000 scf(Byron) 2500 scf (Braidwood)	Maintenance activities of the valves inside the 1/2AD MSIV rooms	Outdoors (outside of each 1/2AD MSIV rooms)
N_2 (gas)	2000 scf(Byron) 2500 scf (Braidwood)	Maintenance activities of the valves inside the 1/2BC MSIV rooms	Outdoors (outside of each 1/2BC MSIV rooms)

TABLE 3.5-10

COMPRESSED GASES STORED ONSITE

GAS STORED	QUANTITY	PURPOSE	LOCATION STORED
N_2 (gas)	1380 scf(Byron) (max)	Cover gas for Unit Aux. Transformers (Unit 1)	Byron turbine Building, elevation 401, column L, row 1
N_2 (gas)	1380 scf(Byron)	Cover gas for Main Power Transformers and Unit Aux. Transformers (Unit 2)	Byron turbine Building, elevation 401, column L, row 34
Halon 1301	889 lb	Fire protection	Turbine building, elevation 468 at column rows 23 and L
Halon 1301	200 lb	Fire protection	Turbine building, elevation 439 at column rows 45 and H.9
Methane	600 scf	Laboratory use	Outdoors
Methane-argon	600 scf	Laboratory use	Outdoors

TABLE 3.5-10 (Cont'd)

GAS STORED	QUANTITY	PURPOSE	LOCATION STORED
Argon	600 scf	Laboratory use	Outdoors
Helium	1800 scf	Laboratory use Main Power Transformer gas analyzer	Outdoors
	2400 scf	Byron U1/U2 Unit Auxiliary/System Auxiliary Transformers	Outdoors
	1200 scf	Braidwood U1/U2 Unit Auxiliary Transformer Gas Analyzer	Outdoors
Propane	1800 scf	Laboratory use	Outdoors
Propane (Byron)	40 lb.	Aux. Boiler Ignition Systems	Aux Boiler Rooms 20 lb., Unit 1 20 lb., Unit 2
Emergency breathing air (Byron)	2,046 scf max	Emergency breathing air for main control room (back-up)	Turbine building, SCBA Storage Cage, elevation 401
Emergency breathing air (Byron)	1,056 scf (max)	Emergency breathing air for main control room	Main Control Room
Emergency breathing air (Braidwood)	3,500 scf (max)	Emergency breathing air for main control room (back-up)	Turbine building elevation 451

TABLE 3.5-10

COMPRESSED GASES STORED ONSITE

GAS STORED	QUANTITY	PURPOSE	LOCATION STORED
Emergency breathing air (Braidwood)	1,000 scf (max)	Emergency breathing air for main control room	Main Control Room
P-10 (gas) 90% Argon 10% Methane	1848 scf(Byron)	Whole Body Frisking Monitor	Byron Radwaste Building, elevation 401′ column G, row 37
P-10 (gas)	1848 scf(Byron)	Whole Body Frisking Monitors	Byron Auxiliary Building, elevation 426′ column M, row 22
P-10 (gas)	4312 scf(Byron)	Whole Body Frisking Monitors	Byron Auxiliary Building, elevation 401′ column L, row 20
P-10 (gas)	1232 scf(Byron)	Whole Body Frisking Monitors	Byron Unit 1 Containment Access Facility (CAF) entry vestibule
P-10 (gas)	1232 scf(Byron)	Whole Body Frisking Monitors	Byron Unit 2 CAF entry vestibule
P-10 (gas)	924 scf(Byron)	Whole Body Frisker	Byron Main Access Facility (MAF) elevation 401′ near exit turnstiles
P-10 (gas)	616 scf (Braidwood)	Whole Body Frisking Monitors	Braidwood Auxiliary Building, elevation 346′ column P, row 19
P-10 (gas)	616 scf (Braidwood)	Whole Body Frisking Monitors	Braidwood Auxiliary Building, elevation 364′ column S, row 18
P-10 (gas)	616 scf (Braidwood)	Whole Body Frisking Monitors	Braidwood Auxiliary Building, elevation 401′ column N, row 20

TABLE 3.5-10
Continue

COMPRESSED GASES STORED ONSITE

GAS STORED	QUANTITY	PURPOSE	LOCATION STORED
P-10 (gas)	616 scf (Braidwood)	Whole Body Frisking Monitors	Braidwood Auxiliary Building, elevation 426’ column U, row 18
P-10 (gas)	1848 scf (Braidwood)	Whole Body Frisking Monitors	Braidwood Turbine Building, elevation 401’ column K, row 22
P-10 (gas)	616 scf (Braidwood)	Whole Body Frisking Monitors	Braidwood Turbine Building, elevation 426’ column K, row 18
P-10 (gas)	616 scf (Braidwood)	Whole Body Frisking Monitors	Braidwood Rad Waste Building, elevation 401’
P-10 (gas)	308 scf (Braidwood)	Whole Body Frisking Monitors	Braidwood MAF Entry Vestibule
P-10 (gas)	616 scf (Braidwood)	Whole Body Frisking Monitors	Braidwood Unit 1 CAF Entry Vestibule
P-10 (gas)	616 scf (Braidwood)	Whole Body Frisking Monitors	Braidwood Unit 2 CAF Entry Vestibule
Air	375 scf* (Braidwood Unit 1)	1AF005A,B,C,D Operation	Braidwood Auxiliary Building elevation 364; column N, row 13
Air	375 scf* (Braidwood Unit 2)	2AF005A,B,C,D Operation	Braidwood Auxiliary Building elevation 364’, column N, row 24
Air	375 scf* (Braidwood Unit 1)	1AF005E,F,G,H Operation	Braidwood Auxiliary Building elevation 364; column N, row 13

TABLE 3.5-10

COMPRESSED GASES STORED ONSITE

GAS STORED	QUANTITY	PURPOSE	LOCATION STORED
Air	375 scf* (Braidwood Unit 2)	2AF005E,F,G,H Operation	Security - Related Information Withheld Under 10 CFR 2.390
Air	375 scf* (Byron Unit 1)	1AF005A,B,C,D (Operation)	
Air	375 scf* (Byron Unit 1)	1AF005E,F,G,H (Operation)	
Air	375 scf* (Byron Unit 2)	2AF005A,B,C,D (Operation)	
Air	375 scf* (Byron Unit 2)	2AF005E,F,G,H (Operation)	

*Based upon 33.4 ft^3 + 1%, 150 psig, 65°F

TABLE 3.5-11

Table 3.5-11 has been DELETED intentionally.

TABLE 3.5-12

Table 3.5-12 has been DELETED intentionally.

TABLE 3.5-13

Table 3.5-13 has been DELETED intentionally.

TABLE 3.5-14

Table 3.5-14 has been DELETED intentionally.

TABLE 3.5-15

SEISMIC CLASSIFICATION OF STRUCTURES, SYSTEMS, OR COMPONENTS INSIDE CONTAINMENT
REQUIRED FOR SAFE SHUTDOWN

	STRUCTURE, SYSTEM OR COMPONENT	LOCATION	APPLICABLE SEISMIC CATEGORY AND QUALITY GROUP CLASSIFICATION	SECTIONS IN UFSAR WHERE DESCRIPTIONS MAY BE FOUND
1.	Main steam system	Each loop compartment	IB	10.3
2.	Main feedwater system	Each loop compartment	IB	10.4.7
3.	Charging capability for R. C. pump seal injection for maintaining level in pressurizer and for increasing boron concentration following Xenon transient	Each loop compartment Note: For hot shutdown, the use of one or more reactor coolant pumps is recommended to equalize temperatures.	IB	9.3.4
4.	Component cooling water for R. C. pump thermal barrier, lube oil cooler reactor vessel support and penetration cooling	Each loop compartment	IC	9.2.2
5.	Essential service water cooled RCFC units	Outside Missile Barrier 1 in each quadrant	IC*	9.2.1.2 6.2.2.4.1

* In order to provide a level of quality equivalent to Quality Group B standards, additional ASME Section III Class 2 nondestructive examination was performed on the RCFC essential service water coils (Refer to Subsection 6.2.2.4.1).

TABLE 3.5-16

MISSILE PROTECTION OF STRUCTURES, SYSTEMS, OR COMPONENTS INSIDE CONTAINMENT REQUIRED FOR SAFE SHUTDOWN

	STRUCTURE, SYSTEM OR COMPONENT	REFERENCE DRAWINGS OR P&IDS APPLICABLE	IDENTIFICATION OF MISSILES TO BE PROTECTED AGAINST	MISSILE PROTECTION PROVIDED
1.	Main steam system	M-35	None	Lines are routed outside of missile barrier and above operating floor
2.	Main feedwater system	M-36, M-121	None	Lines are routed directly from the penetrations to the steam generators.
3.	Charging capability for R.C. pump seal injection for maintaining level in pressurizer, and for increasing Boron concentration following Xenon transient	M-64, M-64A	None	Lines are routed outside missile barrier to fullest possible extent
4.	Component cooling water for R.C. pump thermal barrier, lube oil cooler reactor vessel support and penetration cooling	M-66, M-66A	None	Line are routed outside missile barrier to fullest possible extent
5.	Essential service water cooled RCFC units	M-42, M-42A	None	Lines are routed outside of missile barrier

TABLE 3.5-17

TARGETS EVALUATED IN TORMIS ANALYSIS

TARGET	NUMBER OF TARGETS	FAILURE MODE(S)	NOTES
SXCT Riser Pipes	8	Perforation and Crimping	SXCT Cells A-H
SXCT Fan Motors and Power Feeds	8	Missile Hit	SXCT Cells A-H
SXCT Fan Gear Box Oil Level Gauges	8	Missile Hit	SXCT Cells A-H
SXCT Personnel Hatches	8	Perforation	SXCT Cells A-H
SXCT Fan Inspection Hatches	8	Perforation	SXCT Cells A-H
SXCT Fan Blades	8	Missile Hit	SXCT Cells A-H
SXCT Anti-Vortex Boxes and Trash Screens	2	Perforation	North and South
SXCT Switchgear Room Ventilation Louvers	4	Perforation	Division 11 (Bus 131Z), 12 (Bus 132Z), 21 (Bus 231Z) and 22 (Bus 232Z)
Diesel Driven Auxiliary Feedwater Pump Exhaust Pipes	2	Crimping	Unit 1 and Unit 2
Diesel Driven Auxiliary Feedwater Pump Exhaust Cover Plates	2	Crimping	Unit 1 and Unit 2
Steam Generator Power Operated Relief Valve Tailpipes	8	Pipe Penetration and Crimping	Unit 1 (4) and Unit 2 (4)

TABLE 3.5-17 (cont'd)

TARGET	NUMBER OF TARGETS	FAILURE MODE(S)	NOTES
Main Steam Safety Valve Tailpipes	40	Pipe Penetration	Unit 1 (20) and Unit 2 (20)
Deep Well Pump Enclosures	2	Spall	Pumps 0A and 0B
Embedded Conduits (Auxiliary Building South Wall)	4	Perforation	Division 11 (Bus 131Z), 12 (Bus 132Z, 21 (Bus 231Z) and 22 (Bus 232Z)
Cable Vaults - Division 11 (Bus 131Z), 12 (Bus 132Z), 21 (Bus 231Z) and 22 (Bus 232Z)	6	Spall	Division 11 (1G1), 12 (1H2 and 1J2), 21 (2G1) and 22 (2H2 and 2J2)
Auxiliary Building L Line Openings	2	Pipe Penetration	0A and 0B Main Control Room Turbine Building Makeup Air Intakes
Auxiliary Building L Line Openings	4	Pipe Penetration	Division 11, 12, 21 and 22, Miscellaneous Electrical Equipment Room Exhaust*
Non-ESF Switchgear Room Conduits	5	Perforation	Division 11 and 21 SXCT Power and Control Cables (Evaluated in Segments)

*In a limited number of cases, the exhaust path may be impacted. Therefore, manual action is relied on to restore ventilation. These manual actions entail simple activities to open doors (to provide an exhaust path) and restart supply fans.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED BREAK OF PIPING

To ensure safe and reliable operation of the Byron and Braidwood Stations, the possibility of high or moderate energy line breaks have been considered in the design. Systems which were considered for high energy piping failure are listed in Table 3.6-2.

Piping failures are postulated to occur in high and moderate energy fluid systems at locations defined using the criteria in subsection 3.6.2.1. In addition to the loss of fluid from the failed system, and the direct results of the pipe failure (i.e., pipe whip, fluid impingement, pressurization, environmental effects, water spray, flooding), a functional failure of any single active component is assumed except in those cases where the piping failure is in a dual purpose, moderate energy safety system. In these cases, the single active failure is assumed in any system other than the system which initially failed. A loss of offsite power is assumed to occur if the piping failure results in loss of offsite power or reactor trip.

Standard Review Plans (SRP) 3.6.1 and 3.6.2 were used as the basis for this study. SRP 3.6.1 includes Branch Technical Position (BTP) APCSB 3-1. Appendix B of the BTP, the attachment to letters sent to applicants and licensees by A. Giambusso in December 1972, and Appendix C to the BTP, the July 12, 1973 letter to applicants, reactor vendors and architect-engineers from J. F. O'Leary, provide the basis for identification of high energy line breaks and evaluation of their consequences.

High energy lines can be identified through the engineering controlled equipment/component database(s). Breaks have been postulated at the locations required by Branch Technical Position APCSB 3-1 for the purpose of assessing pipe whip, jet impingement, and pressurization effects. Temperatures in areas were calculated assuming the break occurs in the limiting location in the area. Locations of mitigating features such as pipe restraints and impingement shields are shown in Section 3.6. Drawings showing the location of high energy lines have been provided to the NRC ASB reviewer. These drawings also indicate location of subcompartment walls and pipe tunnels.

The effects of high and moderate energy line breaks inside containment have been assessed as described in Sections 3.6 and 6.2. The effects of high energy line breaks in the turbine building have been evaluated with respect to potential impact on safety-related equipment located in adjoining auxiliary building rooms. The results of this evaluation are described in Section 3.11. Other non-safety related areas were not investigated because damage to or failure of equipment in these areas will not affect plant safety.

The possible effects associated with the postulated break of piping considered are structural loads due to pressurization, increases in pressure and temperature which could affect environmental qualification of equipment, and damage due to pipe whip and jet impingement.

The methods used for protection against each postulated high energy piping failure are:

a. Provision of pipe whip restraints for postulated breaks in plant areas containing safety-related equipment, such that the whipping pipe cannot impact any nearby equipment.

b. Provision of deflectors in the path of effluent discharging from postulated breaks that would otherwise (a) impinge on safety-related equipment to the extent that a loss of function may result, or (b) impinge on equipment whose failure, in turn, may propagate such that a loss of function of safety-related equipment may result.

Areas of system piping where no breaks are postulated are:

a. The main steam piping from the containment penetration fluid head outboard weld, to the upstream weld of the main steam pipe to the main steam isolation valve, including the main steam relief valve header and branch piping to the main steam power operated relief valve and main steam safety valves. This includes approximately 65 feet of piping (20 feet of header and 45 feet of relief piping) for each steam generator.

b. The main feedwater piping from the downstream weld of the main feedwater pipe to the main feedwater isolation valve, to the containment penetration fluid head outboard weld, including the main feedwater isolation valve bypass line from its branch off the main feedwater line to the upstream weld of the line to the normally closed feedwater backpurge isolation valve. This includes approximately 25 feet of piping for each steam generator.

The design of the plant is such that given the above, and applying the load combinations as described in Section 3.9, the function of essential systems and components will not be damaged to the extent that safe shutdown capability is lost.

3.6.1 Postulated Piping Failures in Fluid Systems Outside the Containment

The following is a summary of applicable definitions; criteria employed; potential sources and locations of piping failures; identification of systems and components essential to safe plant shutdown; limits of acceptable loss of function or damage and effect on safe shutdown; habitability of critical areas following postulated piping breaks; and the impact of the plant design on inservice surveillance and inspection.

3.6.1.1 Design Bases

3.6.1.1.1 Definitions

Throughout this section, the following definitions apply:

a. Essential Systems and Components

   Systems and components required to shut down the reactor and mitigate the consequences of a postulated piping failure.

b. Fluid Systems

   High and moderate energy fluid systems that are subject to the postulation of piping failures against which protection of essential systems and components is needed.

c. High-Energy Fluid Systems

   Systems which are either in operation or maintained pressurized during normal plant conditions and meet either or both of the following requirements are called high-energy fluid systems:

   1. maximum operating temperature exceeds 200°F

   2. maximum operating pressure exceeds 275 psig.

d. Moderate-Energy Fluid Systems

   Systems which are either in operation or maintained pressurized(above atmospheric pressure) during normal plant conditions and meet the following requirements are called moderate energy fluid systems:

   1. maximum operating temperature is 200°F or less, and

   2. maximum operating pressure is 275 psig or less.

e. Normal Plant Conditions

   Plant operating conditions normally experienced during reactors startup, operation at power, hot standby, or reactor cooldown to cold shutdown condition.

f. Upset Plant Conditions

Plant operating conditions during system transients that may occur with moderate frequency during plant service life and are anticipated operational occurrences, but not during system testing.

g. Postulated Piping Failures

Longitudinal and circumferential breaks in high-energy fluid system piping and through-wall leakage cracks in moderate-energy fluid system piping.

h. S_h and S_a

Allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME Code, Section III. As defined in article NC/ND-3611.2 of the ASME Code, the allowable stress range, S_a is given by the following formula:

$$S_a = f(1.25\,S_c + 0.25\,S_h)$$

The stress range reduction factor (f) is set at 1.0 for thermal expansion loading conditions of less than or equal to 7,000 equivalent full temperature cycles, and at incrementally smaller values for loading conditions of greater than 7,000 cycles, as provided in Table NC/ND-3611.2(c)-1. In lieu of the Code-defined values for f, f may be calculated by $5.875/(N)^{0.2}$ for ASME Class 2 and 3 and ANSI B31.1 piping and components, where N is the number of equivalent full temperature cycles.

i. S_m

Design stress intensity as defined in Article NB-3600 of ASME Code, Section III.

j. Terminal Ends

Extremities of piping runs that connect to structures, large components (e.g., vessels, pumps) or pipe anchors that act as rigid constraints to piping movement including rotational movement from static or dynamic loading. A branch connection to a main piping run is a terminal end of the branch run.

Intersections of runs of comparable size and stability are not considered terminal ends when the piping stress analysis model includes both the run and branch piping and the intersection is not rigidly constrained to the building structure.

k. Leakage Crack

A theoretical opening in the piping system, the consequences of which are evaluated on the basis of pressure and temperature differential conditions, flooding effects, and wetting of all unprotected components within the compartment.

3.6.1.1.2 Criteria

Regulatory Guide 1.46 and the NRC's letter from A. Giambusso, dated December 1972, have been met for designs inside and outside the containment, respectively. By virtue of the Construction Permit date for this plant, the above is the required minimum.

Subsequent criteria, including that in the NRC's letter from J. F. O'Leary, dated July 1973, and Branch Technical Positions APCSB 3-1 and MEB 3-1, have been employed to the extent possible and practical, given the stage of design/construction.

The required protection has been provided by optimization of the plant layout to minimize the number of areas affected by piping failures and to locate systems and components used for safe shutdown such that unacceptable damage would not occur. In cases where separation of systems or physical barriers provided by plant structure were not sufficient to provide protection, special protective features such as pipe whip restraints and jet impingement shields were employed.

3.6.1.1.3 Identification of Systems Important to Plant Safety

Systems important to plant safety are listed in Table 3.6-1. For a given postulated piping failure, additional systems may be required (e.g., safety injection is required for a LOCA). Refer to Subsection 3.6.1.3 for a more detailed discussion of systems and components important to plant safety.

3.6.1.2 Description of Design Approach

3.6.1.2.1 Potential Sources and Locations of Piping/Environmental Effects

Potential sources of piping failures that are within or could affect Safety Category I structures are listed by system in Table 3.6-2. High energy lines can be identified through the engineering controlled equipment/component database(s).

Locations, orientations, and size of piping failures within high/moderate energy piping systems are postulated per the criteria given in Subsection 3.6.2.1. The dynamic effects of these postulated failures are accommodated by the methodology described in Subsections 3.6.2.2 through 3.6.2.5.

Pressure rise analyses are addressed in Subsection 3.6.1.3 Item a. There are no credible secondary missiles formed from the postulated break of piping.

Control room habitability is addressed in Section 6.4.

3.6.1.2.2 Impact of Plant Design for Postulated Piping Failures on Inservice Inspection

There are three areas of design necessitated for protection from piping failures which may interfere with inservice inspection as dictated by the ASME Boiler and Pressure Vessel Code, Section XI. They are:

a. physical separation of high/moderate energy piping in tunnels or behind barriers,

b. pipe whip restraints which may surround piping welds to be examined, and

c. impingement barriers which may interfere with weld examination or personnel/equipment access.

Design measures employed so that proper inservice inspection can be conducted are, respectively:

a. Tunnels containing Section III piping have been made to allow personnel/equipment access as needed.

b. Pipe whip restraints are of a bolted design which may be either removed from around the pipe or moved axially along the pipe to allow access to any welds.

c. Impingement/separation barriers are designed to minimize inservice inspection interference to the extent that is practical.

3.6.1.3 Safety Evaluation

In the design of this plant, due consideration was given to the effects of postulated piping breaks with respect to the limits of acceptable damage/loss of function to assure that even with a coincident loss of a single active component the remaining structures, systems, and components would be adequate to safely shut down the plant. The following is a summary of the structural, mechanical, instrumentation, electrical, and HVAC items that are deemed essential and, therefore, designed to remain functional against (1) a high energy line break with resulting whip, impingement, compartment pressurization and temperature rise, wetting of compartment surfaces, and flooding, or (2) a moderate energy through-wall leakage crack with resulting wetting of compartment surfaces and flooding.

a. Structural

All Safety Category I structures, listed in Table 3.2-1, remain functional with the exception of certain concrete block and partition walls in the auxiliary building which have not been specifically designed for loads resulting from piping failure because the failure of the wall will not cause damage to the extent that safe shutdown capability is affected. In the event walls were predicted to be loaded by postulated flooding, pressurization or jet impingement, either the walls were shown to be capable of withstanding the load or the potential effects of failure of the wall on safe shutdown components was assessed.

Pressurization and temperature rise studies for postulated breaks in all subcompartments containing normally operating high energy piping are given in Section 6.2 and Attachment A3.6 for inside and outside the containment, respectively. Flooding inside and outside containment is addressed in Attachment D3.6.

b. Mechanical

Table 3.6-3 lists all the mechanical systems which may be used for safe shutdown following any postulated pipe break. Note that all are seismically designed and are comprised of two full capacity, independent, redundant trains. In addition, many of the safety functions can be accomplished by two or more systems, allowing a diversity in safe shutdown procedures. For example, reactor coolant pump seal integrity is maintained if either seal injection flow (chemical and volume control system) or the thermal barrier cooling (component cooling system) is maintained. As another example, chemical shimming may be accomplished via the chemical and volume control system or the safety injection system.

It should also be noted that the essential systems are a function of the postulated initiating event. For any given event, only certain portions of an essential system may be required to achieve safe shutdown, dependent upon the postulated conditions and coincident failures.

The plant design is such that, whenever possible, all potentially essential systems are protected against loss of function resulting from any potential break. This cannot be attained when essential systems have direct communication with the postulated break (e.g., auxiliary feedwater connection to main feedwater or safety injection connection to reactor coolant). In these cases, the hydraulic design of the essential system is such that the "escaping" flow is not large enough to degrade the essential system flow below minimum requirements.

Due to influences on reactivity, cooling capability, etc., break propagation is further limited as defined by Westinghouse (Reference 6) and shown in Table 3.6-4. In addition, containment leakage is always limited to an acceptable level as described in Section 3.8.

Operation of the secondary side isolation valves is critical to the safety of the plant. Therefore, the piping in the isolation valve room areas is designed well within the stress levels set for postulated breaks. In addition, the boundaries of this room, consisting of the containment and a wall at the start of the main steam tunnel, are placed as close to the isolation valves as practical, to minimize the extent of piping in the area. The piping penetrations are designed to withstand the loadings of piping breaks outside this area without transferring enough strain to the isolation valves to render them inoperable. Refer to Subsection 3.8.2 for a description of their designs.

An assessment of the impact of flooding inside and outside containment resulting from failure of high or moderate energy line is included in Attachment D3.6. No potential flooding event affects the ability to bring the plant to a safe shutdown condition.

c. Instrumentation

Appendix B of Reference 7 lists the instrumentation required to sense critical breaks and automatically initiate protective actions to bring the plant to a safe shutdown. In some cases, instrumentation is set to initiate protective measures only when multiple reading is indicated from a number of redundant sensors (e.g., a "2 out of 4" logic). In these situations, the break may be allowed to render a sensor or sensors inoperable, with the additional sensor assumed inoperable due to a single unrelated active failure, so long as the required number of sensors necessary to signal and initiate protective measures remain.

For example in a "2 out of 4" logic, one sensor may be rendered inoperable as a consequence of the break, and the required minimum of "2 out of 4" would remain, assuming a single active failure in one sensor.

d. Electrical

Safety-related electrical components are located, to the extent possible, in areas which will not be affected by high or moderate energy line breaks. In areas such as the containment, where some electrical equipment must be located near high energy systems, redundant components are well separated to prevent failure of both trains from a common initiating event.

An equipment environmental qualification program was conducted to ensure that safe shutdown capability exists after postulated accidents (including a single-ended pipe break of high energy lines in the safety valve house). A list of Class 1E electrical equipment required to function under postulated accident conditions has been developed. Environmental zones, shown in Table 3.11-2, were reviewed to verify that worst case conditions of temperature, pressure, humidity, radiation and potential flooding consequences have been established. Location and categorization of Class 1E electrical equipment with respect to environmental zones have been completed. Equipment operating times have been determined. Finally, qualification test reports were accumulated and reviewed to ensure that the requirements of NUREG-0588 and IEEE-323 were satisfied.

As a result of this program, any Class 1E equipment needed for safe shutdown, which can be affected by the postulated accident environments, shall be qualified to withstand worst case environmental effects.

3.6.1.3.1 Environmental Qualification

A program to document the environmental qualification of electrical equipment was completed for Byron/Braidwood Stations. This program established that the equipment required to safely shut down the plant will be operable under potentially adverse environmental conditions.

One of the potential causes of severe environmental conditions is a break or crack in a high or moderate energy line. This could cause an increase in pressure, temperature, or humidity or a flooding condition in the area of the break.

The basic design of the Byron/Braidwood stations includes features to mitigate the impact of line breaks on the ability to safely shut the plant down. Some of the features are:

a. Essential safety systems are redundant or backed up by other safety systems;

b. The effectiveness of the redundancy is protected by separation of redundant systems to the greatest extent possible;

c. Walls and compartments have been included to both protect equipment and to isolate breaks;

d. Large high energy lines such as main steam, feedwater, and auxiliary steam partially or completely enclosed in protective tunnels in the auxiliary building;

e. Efforts have been made to minimize the number of high energy lines in areas containing safety related equipment and to minimize the size and length of high energy lines. For example, Byron/Braidwood uses motor and diesel driven auxiliary feedwater pumps rather than turbine driven pumps, thereby eliminating the associated high energy steamlines.

The zones identified in Subsection A3.6.1.1 for high energy line breaks analysis are included in the environmental zones. Table 3.11-2 has been updated to include these environmental conditions. The subcompartment transient conditions calculated in the pressurization analysis are used for qualification of equipment in the subcompartment required to safely shut down the plant following the postulated break.

The large general areas containing high energy lines are not subject to pressurization but the temperature in the area may be affected. The general areas were examined to locate limiting high energy lines and a conservative affected area was defined. Large areas separated from breaks by doorways or other restrictive passages were not evaluated because of the restricted flow and the relatively large areas which dilute the break flow. Only two areas were identified which contain high energy lines.

The areas identified as 4A, 4B, 10A, and 10B are actually interconnected. All are affected by breaks at various locations in a 3-inch letdown line in the chemical and volume control system. Orifices in the system limit the flow to a maximum of 120 gpm. The portion of the break fluid which flashes to steam will rise to the upper portions of Zone 4A/4B and flow out through openings into the upper levels of the auxiliary building. The break flow duration will be limited because two main control board alarms (high flow and high letdown heat exchanger outlet temperature) will immediately sound. The break will be isolable with containment isolation valves. As a result of the limited flow from this break and the dilution area which is extremely large, the temperature of the air in these zones will not exceed the maximum temperatures predicted during operating transients and an additional accident environment is not necessary. If the break is in the upper portion of Zone 10A/10B, the potential exists for heating a restricted area with no natural ventilation. None of the equipment in this area is required for safe shutdown following a letdown line failure. This scenario is discussed further in the Byron/Braidwood equipment qualification report.

The other area investigated was Zone 14 at elevation 401 feet. This open area contains a two inch auxiliary steam line. Failure of this line would release steam into the general area. The only equipment required for plant shutdown which could be affected are the boric acid transfer pump motors. The pumps are not required to bring the plant to a hot standby condition. Cold shutdown can be achieved by using water from the refueling water storage tank to increase the reactor coolant boron concentration, eliminating the need for the boric acid transfer pumps. Under certain conditions, required boration may be achieved using only the charging system. Under other conditions, reactor coolant letdown may also be required. Since a total loss of capability to charge or let down the reactor coolant system would not result from an auxiliary steamline break, cold shutdown capability will not be lost. Flow into adjacent areas would eventually occur but the dilution would be so great that the temperature of the adjacent areas would remain effectively unchanged. Table 3.11-2 has been updated to include the environmental conditions discussed here.

Moderate energy line breaks do not impact the equipment qualification parameters. For lines with operating temperatures significantly above the normal area temperature, the crack flow rate and potential for heat transfer has been checked to ensure that sufficient HVAC capability exists to prevent failure of required safety-related equipment.

The Turbine Building contains no safety-related components or other components required for safe shutdown of the Unit. However, there are adjacent rooms in the Auxiliary Building that contain such equipment and that communicate with the Turbine Building through ventilation openings. Therefore, the equipment in those adjacent rooms must be protected from or shown to be able to withstand the effects of HELB in the Turbine Building. The HELB mitigation strategy for the Auxiliary Building rooms involves (1) keeping the Turbine Building environment out of the Auxiliary Building rooms by means of HELB backdraft dampers; (2) configuring the fire dampers to close only in the event of a fire (thereby keeping them open during the HELB to allow the room ventilation exhaust path to remain open); and (3) automatically restoring room cooling (by installing auto-restart capability for the room ventilation fans.)

The following subsections of UFSAR Section 3.6 describe the approach used to evaluate the effects of high energy line breaks, including Turbine Building HELB. The resultant environmental conditions in the adjacent Auxiliary Building rooms have been determined per Reference 18. Due to the limited magnitude and short duration of the transient, the environmental parameters within these zones would not be significantly more severe than the environment that would occur during normal plant operation.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Break of Piping

Described herein are the design bases for locating breaks and cracks in piping inside and outside of containment, the procedures used to define the jet thrust reaction at the break location, the jet impingement loading criteria, and the dynamic response models and results.

Because of variations in requirements, techniques, and failure effects, high and moderate energy lines are addressed separately. Similarly, the pipe whip, subcompartment pressurization, and environmental analysis all have somewhat different approaches.

a. High Energy Line Analysis

Standard Review Plans 3.6.1 and 3.6.2 were followed in defining and identifying high energy lines. High energy lines are those larger than 1 inch diameter for which either:

1. The service temperature is greater than 200°F; or

2. The design pressure is greater than 275 psig.

Only a limited number of systems in the auxiliary building meet either of these criteria. The following systems have been identified as containing high energy lines in the auxiliary building:

Chemical and Volume Control	(CV)
Auxiliary Steam	(AS)
Steam Generator Blowdown	(SD)
Radioactive Waste Processing	(WX)
Boric Acid	(AB)
Main Steam	(MS)
Feedwater	(FW)
Auxiliary Feedwater	(AF)
Residual Heat Removal	(RH)
Safety Injection	(SI)

Systems which are normally not used or at reduced temperature and pressure are not necessarily required to be considered as high energy lines. A guideline has been established (Branch Technical Position MEB 3-1) that if the system is at high energy conditions less than 2% of the time, it may be considered a moderate energy line and its normal conditions applied to the line break analysis. On this basis, the last three systems (AF, RH, SI) are not considered as high energy systems. The Byron/Braidwood AF system is not used for normal startup as at some other plants. The only high energy line in the boric acid system is a steam supply line to the boric acid batching tank. This line is essentially a part of the auxiliary steam system and, as result, was not identified in Table 3.6-2.

Subcompartment pressurization is investigated for all lines with temperatures above 200°F. Lower temperatures lines do not have the potential for flashing to steam and thus will not increase the pressure of a subcompartment in the event of a break. Pressurization is of concern only in small subcompartments with relatively large high energy lines or subcompartments with limited pressure relief venting.

High energy lines below 200°F have only minor effects on the environmental conditions. The absence of steam and the ability to drain warm liquid from the break area limits the temperature rise from these breaks. The auxiliary building HVAC has sufficient capacity to accommodate these lower temperature breaks. Breaks of other high energy lines may influence the expected maximum temperature in some areas of the auxiliary building even if high pressures do not result. The auxiliary building contains several large areas with high energy lines that are not subject to pressurization but are investigated for environmental effects.

Certain postulated break locations in high energy piping systems are used to investigate the potential for damage due to pipe whip and jet impingement. The guidelines in Standard Review Plan 3.6.2 are used to determine the number and locations of the pipe breaks. Pipe restraints are added as required to prevent damage to structures and safety-related equipment.

The Turbine Building contains no safety-related components or other components required for safe shutdown of the Unit. However, there are adjacent rooms in the Auxiliary Building that contain such equipment and that communicate with the Turbine Building through ventilation openings. Therefore, the equipment in those adjacent rooms must be protected from or shown to be able to withstand the effects of HELB in the Turbine Building. Turbine Building HELBs were postulated in a manner that would produce the most challenging environmental conditions for the equipment in the adjacent Auxiliary Building rooms.

For the environmental analysis, numerous locations involving different Turbine Building elevations were considered to determine bounding conditions for the breaks. Break locations were chosen based on the resulting severity of the break and not on the potential to break (i.e., piping analysis results were not used to determine break locations). Per the UFSAR 15.1.5.2, the largest main steam (highest enthalpy) line break is 1.4 ft^2. This is based on the area of the integral flow restrictor in each of the four steam generators and flow losses between the steam generators and the Turbine Building and a Main Steam Isolation Valve to close (single failure). For liquid line breaks, the largest Feedwater line breaks and Heater Drain line breaks on different Turbine Building elevations were considered in the evaluation.

The pressures developed in the calculation are used as input for the qualification of the L-Line doors and dampers that separate the Turbine Building from the adjacent Auxiliary Building rooms. Additionally, the pressures internal to the rooms are used for qualifications of the divisional walls and doors between that separate them from each other.

The evaluations of flooding, pipe whip, and jet impingement effects for Turbine Building HELBs are discussed in Sections 3.6.2.b and 3.6.2.2.

b. Moderate Energy Line Breaks

Moderate energy lines are lines which operate at temperatures of 200°F or less and pressures of 275 psig or less. A break in a moderate energy line will not result in flashing of the liquid to steam and, as a result, has no potential for pressurization of areas. The relatively low temperature and reduced heat transfer effects of the liquid blowdown precludes significant temperature increases in the area of the break. The reduced break area applicable to these breaks and the absence of steam allows the auxiliary building HVAC to maintain temperatures within those specified in the environmental qualification program. The results of moderate energy line breaks are, therefore, confined to the physical effects of liquid discharge into the plant. Plant safety is affected only if equipment required to mitigate the break or to safely shut down the plant can be damaged by resultant flooding or water spray. Water spray was not found to affect plant safety because of the separation of redundant safe shutdown systems and components. Moderate energy line breaks do not result in pipe whip.

As an example, the auxiliary building basement at elevation 330 feet is designed to prevent loss of redundant trains of safety related equipment from the effects of a moderate energy line break. The basement is divided into two completely independent sections.

These sections are separated by a wall which has been designed to withstand the flooding. Each section contains redundant essential service water pumps which can supply both units. Therefore, flooding or spray from a break cannot affect the equipment in the other section of the basement and essential service water will be supplied to both units.

This separation is well documented in the Fire Protection Report. This report lists and locates equipment required for safe shutdown. When redundant safe shutdown systems are separated by fire walls or by more than 20 feet, spray from a crack in a moderate energy line would not impair the safe shutdown capability of the plant.

A moderate energy line break in the component cooling system was given special consideration because the component cooling system was not originally supplied with a Category I source of makeup water. A leak in this system could have theoretically drained the surge tanks resulting in damage to the component cooling pumps.

A significant leakage in the component cooling system is not expected. The system is a moderate energy, low pressure system and is not subject to severe loading. In the event the system is inoperable, the plant may be safely maintained in a hot shutdown condition until the component cooling system is restored.

If a crack is postulated in one of the large lines in the system, the level in the surge tank of the affected unit will drop. Demineralized water and primary water makeup is fed to the surge tank at preset level limits to maintain tank level in the normal range. Prior to reaching the pump trip setpoint, one or both trains of essential service water makeup MOVs will open to maintain the surge tank level and allow the component cooling water pumps to continue operation. Control Room annunciation is provided when essential service water makeup is fed to the component cooling water surge tank from either train. If the level reaches the low setpoint level, alarms will sound and the affected units component cooling pumps will be automatically tripped to prevent damage to the pumps.

If primary water, demineralized water or essential service water makeup is available, the component cooling pumps may be restarted and the unit operated normally while the leak is located and isolated. Otherwise, the reactor will be tripped because of the interruption of the component cooling to the reactor coolant pumps and the unit will be placed in a hot shutdown condition. Component cooling is not required to safely maintain the unit in hot shutdown mode. The component cooling system can be operated after a failure of the piping by maintaining sufficient surge tank makeup, and by closing the appropriate system valves to isolate the break location and maintain component cooling flow.

In the Turbine Building, numerous high energy lines are in the same area as the moderate energy lines. Because the environmental conditions from the high energy line breaks bound the environmental conditions from the moderate energy line breaks, the environmental impact analysis for Turbine Building HELBs bounds the effect of breaks in moderate energy lines.

Flooding in the Turbine Building would not adversely affect the equipment in the adjacent Auxiliary Building rooms. Numerous stairwells, grating areas, floor opening (e.g., pipe sleeves) and equipment hatches exist on all elevations of the Turbine Building. Therefore, water levels cannot develop any depth from which water could flow under doors into the adjacent Auxiliary Building rooms. Because the doors and dampers have been determined not to fail due to jet impingement, any leakage through these components would result in an inconsequential volume and level of water. Water that was not captured by the Turbine Building floor drain system would eventually reach the Turbine Building basement where flooding is bounded by a Circulating Water pipe break (UFSAR 10.4.5).

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

3.6.2.1.1 Reactor Coolant Loop Piping

Pipe failure protection is provided in accordance with the requirements of 10 CFR 50, Appendix A, General Design Criterion 4 (GDC4). The original design postulated pipe break locations in the reactor coolant loop are described in Reference 1. In accordance with the provisions of GDC4 (as revised per 52 FR 41288, October 27, 1987), the dynamic effects associated with postulated pipe breaks can be eliminated from the structural design basis if it is demonstrated that the probability of pipe rupture is extremely low.

Through the application of leak-before-break technology, the dynamic effects from postulated breaks in the reactor coolant loop primary piping, accumulator line piping, and reactor coolant loop bypass piping can be eliminated from the structural design basis, based on the evaluation presented in References 10 and 12. For Byron Units 1 and 2, and Braidwood Units 1 and 2, based on the evaluation presented in Reference 17, following application of the Mechanical Stress Improvement Process (MSIP) on all eight reactor coolant inlet/outlet nozzles, the leak-before-break analysis margins for the critical locations as documented in Reference 10 are still bounding. Approval of the elimination of breaks in Units 1 and 2 primary loop piping, accumulator line piping, and reactor coolant loop bypass piping is given in References 11 and 13. To provide the high margins of safety required by GDC-4, the nonmechanistic pipe rupture design basis is maintained for containment design and ECCS analyses, and the postulated pipe ruptures are retained for electrical and mechanical equipment environmental qualification.

3.6.2.1.2 Piping Other Than Reactor Coolant Loop Piping

This section applies to all high and moderate energy piping outside the reactor coolant pressure boundary and to any reactor coolant pressure boundary piping not covered in Section 3.6.2.1.1.

3.6.2.1.2.1 <u>High-Energy Fluid System Piping</u>

3.6.2.1.2.1.1 <u>Fluid System Piping not in the Containment Penetration Area</u>

a. Breaks in ASME Section III Class 1 piping are postulated at the following locations in each piping run or branch run:

   1. at terminal ends of the run;

   2. at intermediate locations between terminal ends where the primary plus secondary stress intensity range (including the zero load set) as calculated by equation (10) and either equation (12) or (13) in Paragraph NB-3653 of ASME Section III exceeds 2.4 S_m for transients resulting from normal and upset plant conditions; and

   3. at any intermediate locations between terminal ends where the cumulative usage factor derived from the piping fatigue analysis under the loadings resulting from plant normal, upset, and testing conditions and an OBE event exceeds 0.1.

b. With the exception of those portions of piping identified in Subsection 3.6.2.1.2.1.2, breaks in ASME Section III Class 2 and 3 piping and seismically analyzed and supported ANSI B31.1 piping are postulated at the following locations in each piping run or branch run:

   1. At terminal ends of the run.

   2. At each location where the stresses under the loadings resulting from normal and upset plant conditions and an OBE event as calculated by equations (9) and (10) in Paragraph NC-3652 of ASME Section III exceed 0.8 (1.2 S_h + S_a).

   3. As an alternate to (1) and (2), intermediate locations are assumed at each location of potential high stress or fatigue such as pipe fittings, valves, flanges and attachments.

c. Breaks in nonseismically qualified piping are postulated at the following locations in each piping run or branch run:

   1. At terminal ends of the run.

2. Intermediate locations are assumed at each location of potential high stress or fatigue such as pipe fittings, valves, flanges and attachments.

d. Leakage cracks in high energy ASME Section III Class 2 and 3 piping and seismically analyzed and supported ANSI B31.1 piping are postulated at locations where the stresses under the loadings resulting from normal and upset plant conditions and an OBE event as calculated by equations (9) and (10) in Paragraph NC-3652 of ASME Section III exceed 0.4 (1.2 S_h + S_a).

3.6.2.1.2.1.2 Fluid System Piping in Containment Penetration Areas

This section applies to the fluid system piping inside the isolation valve rooms, which includes the main steamlines and the feedwater lines, starting at the inside of the containment wall and extending to the first restraint outside the containment isolation valve.

3.6.2.1.2.1.2.1 Details of the Containment Penetration

Details of the containment penetrations are discussed in Subsections 3.8.1 and 3.8.2.

3.6.2.1.2.1.2.2 Break Criteria

Breaks are not postulated in the containment penetration area as defined above since the following design requirements are met:

a. The following design stress and fatigue limits are not exceeded for ASME Code Section III Class 2 piping and seismically qualified ANSI B31.1 piping:

1. The maximum stress ranges as calculated by the sum of Equations (9) and (10) in Paragraph NC-3652, ASME Code, Section III, under the loadings resulting from the normal and upset plant conditions (i.e., sustained loads, occasional loads, and thermal expansion) and an OBE event do not exceed 0.8 (1.2 S_h + S_a).

2. The maximum stress, as calculated by Equation (9) in Paragraph NC-3652 under the loadings resulting from internal pressure, dead weight, and a postulated piping failure of fluid systems piping beyond these portions of piping and excluding OBE, does not exceed 1.8 S_h. Primary loads include those which are deflection limited by whip restraints.

3. Following a piping failure outside the first pipe whip restraint, the formation of a plastic hinge is not permitted in the piping between the containment penetration and the first pipe whip restraint. Bending and torsion limiting restraints are installed, as necessary, at locations selected to optimize overall piping design, to prevent formation of a plastic hinge as just noted, to protect against the impairment of the leaktight integrity of the containment, to assure isolation valve operability and to meet the stress and fatigue limits in the containment penetration area.

b. Leakage cracks:

Per SRP 3.6.2, the break criteria of Subsection 3.6.2.1.2.1.2.2.a, paragraphs 1 and 2, also apply to the postulation of cracks in the penetration area in the region from the containment wall to and including the inboard or outboard isolation valves.

Leakage cracks in high energy ASME Section III Class 2 and 3 piping and seismically analyzed and supported ANSI B31.1 piping located in the containment penetration area, other than that piping described in the paragraph above, are postulated in accordance with Subsection 3.6.2.1.2.1.1. For the Main Feedwater and Main Steam lines, this includes the piping from the inboard weld of the FWIV/MSIV to the first restraint outside the isolation valve (i.e., in the MSIV room wall).

c. The number of circumferential and longitudinal piping welds and branch connections are minimized as far as practical.

d. The length of these portions of piping are reduced to the minimum length practical.

e. One hundred percent volumetric examination of full penetration process-piping butt welds, 6-inch nominal pipe size and greater, in the break exclusion area was performed as a baseline inspection before operation. During each inspection interval, process-piping welds in the break exclusion areas are subject to an examination program. In lieu of the requirements specified in NUREG 0800, EPRI Revised Risk-Informed Inservice Inspection Evaluation Procedure (Reference 14) and Extension of the EPRI Risk-Informed Inservice Inspection (RI-ISI) Methodology to Break Exclusion Region (BER) Programs (Reference 15) Topical Reports are used to establish the selection criteria and examination methods. The NRC approved the use of these alternate methods in Reference 16. The weld population subject to examination under the Risk-Informed BER program are non-exempted piping welds as determined in accordance with the rules of ASME Section XI, edition and addenda applicable to the existing inservice inspection program.

f. Access to process pipe welds within containment penetration sleeves is not provided since:

   1. There are no circumferential process pipe welds within containment penetration sleeves.

   2. Items a.2 and 3 of Subsection 3.6.2.1.2.1 cover break criteria for Class 1 piping, whereas, no Class 1 piping penetrates containment.

   3. There are no penetration sleeves to process pipe welds contained in piping covered in the augmented inservice inspection program. The containment penetrations for this piping are all Type I head fittings as shown in Figure 3.8-40.

3.6.2.1.2.2 Moderate-Energy Fluid System Piping

3.6.2.1.2.2.1 Moderate-Energy Fluid System Piping Outside Containment

a. Through-wall leakage cracks are postulated in Seismic Category I moderate-energy ASME Section III, Class 2 and 3 and seismically analyzed and supported ANSI B31.1 piping except where the maximum stress range is less than 0.4 ($1.2S_h$ + S_a). In unanalyzed moderate-energy ASME Section III Class 2 and 3 and ANSI B31.1 piping, this exception based on stress is not taken. The cracks are postulated individually at locations that result in the maximum effects from fluid spraying and flooding, with the consequent hazards or environmental conditions developed.

b. Through-wall leakage cracks instead of breaks are postulated in the piping of those fluid systems that qualify as high energy fluid systems for only short operational periods but qualify as moderate energy fluid systems for the major operational period.

   An operational period is considered "short" if the fraction of time that the system operates within the pressure-temperature conditions specified for high energy fluid systems is about 2 percent of the time that the system operates as a moderate energy fluid system.

3.6.2.1.2.2.2 Moderate-Energy Fluid System Piping Inside Containment

Through-wall leakage cracks are not postulated in moderate energy fluid systems inside containment because the flooding and water spray effects resulting from cracks is governed by the following:

a. Containment flooding is governed by a large loss of coolant accident which has been considered in the plant design.

b. Spray effects are considered in the equipment qualification program for safe shutdown equipment inside containment. "Chemical spray" qualification simulates containment spray.

3.6.2.1.2.3 Types of Breaks and Leakage Cracks in Fluid System Piping

3.6.2.1.2.3.1 Circumferential Pipe Breaks

Circumferential breaks are postulated in high-energy fluid system piping exceeding a nominal pipe size of 1 inch, at the locations specified in Subsection 3.6.2.1.2.1.

Where break locations are selected in piping without the benefit of stress calculations, breaks are postulated nonconcurrently at the piping welds to each fitting, valve, or welded attachment.

3.6.2.1.2.3.2 Longitudinal Pipe Breaks

The following longitudinal breaks are postulated in high-energy fluid system piping at the locations of the circumferential breaks specified in Subsection 3.6.2.1.2.3.1.

a. Longitudinal breaks in fluid systems piping and branch runs are postulated in nominal pipe size 4-inch and larger, where the maximum stress range exceeds 2.4 S_m for ASME Code, Section III, Class 1 piping and 0.8 (1.2 S_h + S_a) in ASME Code, Section III, Class 2 and 3 and seismically qualified ANSI B31.1 piping or where break locations are chosen per Subsection 3.6.2.1.2.1.1.

b. Longitudinal breaks are not postulated at:

   1. terminal ends; and

   2. locations chosen to meet the requirements of the minimum number of intermediate breaks as defined in Subsection 3.6.2.1.2.1.1 and Subsection 3.6.2.1.2.1.1 Item b.3.

c. Longitudinal breaks are assumed to result in an axial split without pipe severance. Splits are oriented (but not concurrently) at two diametrically-opposed points on the piping circumference such that the jet reaction causes out-of-plane bending of the piping configuration. Alternatively, a single split is assumed at the section of highest tensile stress as determined by detailed stress analysis.

d. If a postulated break location is at a nonaxisymmetric fitting (such as a tee or elbow), without the benefit of a detailed stress analysis, longitudinal breaks are postulated to occur:

1. Out of plane of an elbow oriented nonconcurrently at two diametrically-opposed points on the circumference in the middle of the elbow.

2. Out of plane of a tee oriented nonconcurrently at two diametrically-opposed points in the middle of the tee run section.

3.6.2.1.2.3.3 Through-Wall Leakage Cracks

The following through-wall leakage cracks are postulated in moderate energy fluid system piping at the locations specified in this position:

a. Cracks are postulated in fluid system piping and branch runs exceeding a nominal pipe size of 1 inch.

b. Fluid flow from a crack is based on a circular opening of area equal to that of a rectangle one-half pipe-diameter in length and one-half pipe wall thickness in width.

c. The flow from the crack is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments. Flooding effects are determined on the basis of a conservatively estimated time period required to effect corrective actions. Evaluation of jet impingement effects is not considered for postulated through-wall leakage cracks.

3.6.2.1.2.4 Definitions

Definitions are given in Subsection 3.6.1.1.

3.6.2.2 Analytical Methods to Define Forcing Functions and Response Models

3.6.2.2.1 Reactor Coolant Loop Piping

3.6.2.2.1.1 Dynamic Analyses

Following is a summary of the methods used to determine the dynamic response of the reactor coolant loop associated with postulated pipe breaks in the loop piping. Although the dynamic effects of postulated pipe breaks in the reactor coolant loop primary piping, accumulator line piping, and reactor coolant loop bypass piping can be eliminated from the structural design basis (see Subsection 3.6.2.1.1), the design verification of certain structures and components may retain the original pipe break loadings. For these cases, the following subsections describe the methods used in the analysis.

3.6.2.2.1.2 Time Functions of Jet Thrust Force on Broken and Intact Loop Piping

In order to determine the thrust and reactive force loads to be applied to the reactor coolant loop during the postulated loss-of-coolant accident (LOCA), it is necessary to have a detailed description of the hydraulic transient. Hydraulic forcing functions are calculated for the broken and intact reactor coolant loops as a result of a postulated LOCA. These forces result from the transient flow and pressure histories in the reactor coolant system. The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flow rates, and thermodynamic properties as a function of time. The second step uses the results obtained from the hydraulic analysis, along with input of areas and direction coordinates, and calculates the time history of forces at appropriate locations (e.g., elbows) in the reactor coolant loops.

The hydraulic model represents the behavior of the coolant fluid within the entire reactor coolant system. Key parameters calculated by the hydraulic model are pressure, mass flow rate, and density. These are supplied to the thrust calculation, together with plant layout information to determine the time-dependent loads exerted by the fluid on the loops. In evaluating the hydraulic forcing functions during a postulated LOCA, the pressure and momentum flux terms are dominant. The inertia and gravitational terms are taken into account in evaluation of the local fluid conditions in the hydraulic model.

The blowdown hydraulic analysis is required to provide the basic information concerning the dynamic behavior of the reactor core environment for the loop forces, reactor kinetics and core cooling analysis. This requires the ability to predict the flow, quality, and pressure of the fluid throughout the reactor system. The SATAN-IV Code (Reference 2) was developed with a capability to provide this information.

The SATAN-IV Code performs a comprehensive space-time dependent analysis of a LOCA and is designed to treat all phases of the blowdown. The stages are: (1) a subcooled stage where the rapidly changing pressure gradients in the subcooled fluid exert an influence upon the reactor coolant System and support structures, (2) a two phase depressurization stage, and (3) the saturated stage.

The code employs a one dimensional analysis in which the entire reactor coolant system is divided into control volumes. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. Pump characteristics, pump coastdown and cavitation, core and steam generator heat transfer including the W-3 DNB correlation in addition to the reactor kinetics are incorporated in the code.

The STHRUST computer program was developed to compute the transient (blowdown) hydraulic loads resulting from a LOCA.

The blowdown hydraulic loads on primary loop components are computed from the equation.

$$F = 144\,A \left[(P - 14.7) + \left(\frac{\dot{m}^2}{144\,\rho\,g\,A_m^2} \right) \right] \qquad (3.6\text{-}1)$$

The symbols and units are:

F	= Force, lb_f
A	= Aperture area, ft^2
P	= System pressure
M	= Mass flow rate, lb_m/sec
ρ	= Density, lb_m/ft^3
G	= Gravitational constant = 32.174 $ft\text{-}lb_m/lb_f - sec^2$
A_m	= Mass flow area, ft^2

In the model to compute forcing functions, the reactor coolant loop system is represented by a similar model as employed in the blowdown analysis. The entire loop layout is described in a global coordinate system. Each node is fully described by: (1) blowdown hydraulic information, and (2) the orientation of the streamlines of the force nodes in the system, which includes flow areas, and projection coefficients along the three axes of the global coordinate system. Each node is modeled as a separate control volume, with one or two flow apertures associated with it. Two apertures are used to simulate a change in flow direction and area. Each force is divided into its x, y, and z components using the projection coefficients. The force components are then summed over the total number of apertures in any one node to give a total x force, total y force, and total z force. These thrust forces serve as input to the piping/restraint dynamic analysis.

The STHRUST Code is described in Reference 3.

3.6.2.2.1.3 Dynamic Analysis of the Reactor Coolant Loop Piping Equipment Supports and Pipe Whip Restraints

The dynamic analysis of the reactor coolant loop piping for the LOCA loadings is described in Section 3.9.

3.6.2.2.2 Analytical Methods to Define Forcing Functions and Response Models for Piping Excluding Reactor Coolant Loop Piping

This section applies to all high energy piping outside the reactor coolant pressure boundary and to all reactor coolant pressure boundary piping, including the RCS bypass piping but excluding the reactor main coolant piping which connects the reactor vessel, the main coolant pumps, and the steam generators.

3.6.2.2.2.1 Determination of Pipe Thrust and Jet Loads

3.6.2.2.2.1.1 Circumferential Breaks

Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the broken piping sections unless physically limited by piping restraints, structural members, or piping stiffness. The dynamic force of the jet discharge at the break location is based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically determined thrust coefficient. Limited pipe displacement at the break location, line restriction flow limiters, positive pump controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of the jet discharge. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration and to cause pipe movement in the direction of the jet reaction.

3.6.2.2.2.1.2 Longitudinal Breaks

The dynamic force of the fluid jet discharge is based on a circular break area equal to the cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of jet discharge.

Piping movement is assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or piping stiffness.

3.6.2.2.2.1.3 Pipe Blowdown Force and Wave Force

The fluid discharge forces that result from either postulated circumferential or longitudinal breaks are calculated using a simplified one step forcing function methodology. This methodology is described in a Sargent & Lundy calculation procedure (Reference 5) and is based on the simplified methods described in ANSI 58.2 and in Reference 4.

When the simplified method discussed above leads to impractical whip restraint designs, then a more detailed computer solution which more accurately reflects the postulated pipe break event is used. The computer solution is based on the NRC's computer program, developed for calculating two-phase blowdown forces (Reference 9).

3.6.2.2.2.1.4 Evaluation of Jet Impingement Effects

The break locations defined for the pipe whip investigation were examined for jet impingement effects. The majority of locations had no effect on equipment required for safe shutdown. This was a result of the criteria used in design to maintain separation of redundant systems and the use of compartments to isolate high energy line break effects. Equipment which could be affected by jet impingement was analyzed and moved or protected if protection was required.

Jet impingement force calculations are required only if structures or components are located near postulated high energy line breaks and it cannot be demonstrated that failure of the structure or component will not adversely affect safe shutdown capability. The methodology used in the plant design when force calculations were found necessary is described in detail in Reference 5.

To confirm that the design approach for protection against jet impingement effects had been consistently applied throughout the design process, a thorough review of potential jet effects on safe shutdown components was completed in August 1984. A report (Reference 7) contains the results of this confirmatory review, and demonstrates that safe shutdown capability is not adversely affected by jet impingement. This effort utilized the most current information available as to the plant configuration and operating conditions. Recently, improved descriptions of steam and two-phase jet behavior were also incorporated into the review (Reference 8).

For Turbine Building HELBs, evaluations were performed in accordance with the methodologies described in the UFSAR to demonstrate that the L-line wall and components integral to the wall (doors and dampers) that separate the Turbine Building from the adjacent Auxiliary Building rooms can withstand the HELB jet forces.

The L-Line wall was determined to be not adversely affected by jet impingement due to the strength and thickness of the wall (a concrete re-enforced wall 42” in depth, and a safety related, Seismic Category I structure.) For doors and dampers, postulated jets were evaluated and found either (1) to not impact the doors or dampers, or (2) to result in forces that did not exceed the design pressures of the components, or (3) require shields to protect them from jet impingement.

The analysis utilized the guidance of Reference 8 to exclude targets greater than 10 pipe diameters from high energy lines. The use of Reference 8 was consistent with the NRC limitations on its use documented in Supplement 6 to the Byron Unit 1 Safety Evaluation Report (later made applicable to Byron Unit 2 and Braidwood Units 1 and 2.) For targets within 10 pipe diameters of high energy steam lines or high energy liquid lines that flash following the break, or that are near other high energy liquid lines, the evaluation utilized ANSI/ANS 58.2 to determine jet shapes and jet impingement loads. This is consistent with the methodology described in UFSAR Section 3.6.2 for determining jet loads.

3.6.2.2.2.2 Methods for the Dynamic Analysis of Pipe Whip

Pipe whip restraints provide clearance for thermal expansion during normal operation. If a break occurs, the restraints or anchors nearest the break are designed to prevent unlimited movement at the point of break (pipe whip). Two methods were used to analyze simplified models of the local region near the break and to calculate displacements of the pipe and restraint. These calculated displacements were then used to estimate strains in the pipe and the restraint.

An energy balance method was used to analyze carbon steel pipes since it was found possible to use a rigid-perfectly plastic moment-rotation law for pipes of this material with acceptable accuracy. The simplified models shown in Figure 3.6-15 were used to represent the local region near the break and to calculate the displacement of the pipe and the restraint when subjected to a suddenly applied constant force by the energy balance method. The restraint and structure resistances were assumed rigid perfectly plastic. Elastic effects increase the work done by the blowdown thrust. Since these effects are neglected in the rigid-plastic energy balance model, they were accounted for by increasing the gap between the pipe and the restraint by an empirical formula.

A finite difference model was used to analyze stainless steel pipes since it was found necessary to use a power law moment-curvature relationship for pipes of this material. The simplified models shown in Figure 3.6-16 were used to represent the local region near the break and to calculate the displacement in the restraint as well as the displacements and strains in the pipe.

3.6.2.2.2.2.1 Stages of Motion - Energy Balance Method

All references to points and lengths in this section can be found in Figure 3.6-15.

At the start of motion, the pipe is assumed fixed at point A. Physically, point A is an anchor, restraint, or elbow. In general, a hinge will form at some point B and outboard pipe segment BD will rotate as a rigid body until contact with the restraint is made at point C.

During the next stage of motion the hinge at B must move in order to satisfy the requirement that shear at a plastic hinge is zero. At the same time a hinge will form at the restraint (point C) if the plastic moment M_o is exceeded. Initially at contact, the force exerted on the pipe by the restraint is R, the restraint resistance. This force will remain constant as long as the restraint continues to deform.

If the structure resistance is $R_S < R$, at some point restraint deformation will stop while structure deformation (motion of point E) continues. The force on the pipe (and attached mass M) is the R_S. In any event, the moving hinge B will reach the fixed support at A before motion stops at C. In the final stage of motion hinges may exist at A and C until motion stops.

3.6.2.2.2.2.2 First Stage of Motion

The initial location of the hinge at B is determined by locating the point of zero shear and is given by:

$$L_Z = 1.5\left[1 + \left(1 + \frac{8\ M_t F}{3\ m\ M_0}\right)^{1/2}\right] \frac{M_o}{F}$$

where:

M_t	= tip mass (lbm),
F	= blowdown force (lb),
m	= mass of pipe/inch,
M_o	= plastic moment of pipe (in.-lb), and
L_z	= location in inches (Figure 3.6-17)

3.6.2.2.2.2.3 Second Stage of Motion (Moving Hinge)

Case 1. No hinge at restraint for the case when M_s (structural mass) is not accounted for (Figure 3.6-17).

After integrating, with respect to time, the equations for conservation of linear and angular momentum are:

$$P_1 t = I_1 \omega - C_1 \quad (3.6\text{-}3)$$

$$P_2 t = I_2 \omega - C_2 \quad (3.6\text{-}4)$$

where:

C_1 and C_2 are constants and are determined at t = 0
t = time of motion from B to present location,
I_1 = $1/2\ m\ L^2 + M_s(L-L_2) + M_t L$,
I_2 = $(1/6)\ mL^2 + (3\ L_2-L) + M_t\ L_2 L$,
P_1 = F - R
P = $F\ L_2 - M_o$, and
ω = $\dot{\theta}$ (radians/second).

From Equations 3.6-3 and 3.6-4:

$$\omega = \frac{P_1\ t + C_1}{I_1} \quad (3.6\text{-}5)$$

$$t = \frac{C_1\ I_2 - C_2\ I_1}{P_2\ I_1 - P_1\ I_2} \quad (3.6\text{-}6)$$

Equations 3.6-5 and 3.6-6 describe the second stage of motion.

Case 2. Hinge at restraint for the case when M_s (structural mass) is not accounted for (Figure 3.6-17).

For conservation of linear and angular moment a of the segments:

$$\omega = C_3 / (L - L_2)^3 \quad (3.6\text{-}7)$$

$$P_2 t + C_5 = M_{12} C_3 / (L - L_2)^2 + M_{11} \omega \quad (3.6\text{-}8)$$

$$P_1 t + C_4 = I_3 C_3 / (L - L_2)^2 + M_{12} \omega \quad (3.6\text{-}9)$$

where:

V = velocity of restraint = $\omega\ (L - L_2)$

C_3, C_4 and C_5 are constants and are determined at t = 0

I_3 = $1/2m(L+L_2) + M_t$,

M_{11} = $(1/3)mL_2^3 + M_t L_2^2$, and

M_{12} = $1/2mL_2^2 + M_t L_2$.

From Equations (3.6-8) and (3.6-9):

$$t = \frac{C_3(M_{12}2 - M_{11}I_3)/(L - L_2)^2 - (C_5 M_{12} - C_4 M_{11})}{(P_2 M_{12} - P_1 M_{11})} \qquad (3.6\text{-}10)$$

$$\omega = \frac{P_2 t + C_5 - (C_3 M_{12}/(L - L_2)^2)}{M_{11}} \qquad (3.6\text{-}11)$$

Equations 3.6-7, 3.6-10, and 3.6-11 describe the second stage of motion for hinge at restraint.

3.6.2.2.2.2.4 Third Stage of Motion (Hinge at Support)

From summation of moment about two hinges (at support and restraint) one gets:

$$K_{11}\ddot{\theta}_1 + K_{12}\ddot{\theta}_2 = FL - RL_1 - M_o \qquad (3.6\text{-}12)$$

$$K_{12}\ddot{\theta}_1 + K_{22}\ddot{\theta}_2 = FL_2 - M_o \qquad (3.6\text{-}13)$$

where:

K_{11} = $(1/3)mL^3 + M_t L^2$

K_{12} = $(1/2)mL^2 (L-L_2/3) + M_t LL_2$, and

K_{22} = $(1/3)mL_2^3 + M\ L_2^2$.

Equations 3.6-12 and 3.6-13 describe motion in the third stage.

3.6.2.2.2.2.5 Gap Increase to Account for Elastic Effects

It has been found by comparison with finite difference results that the neglect of elastic effects in the energy balance method can be compensated for by increasing the gap by an amount given by the following empirical formula:

$$g = 0.0025 \left(\frac{L}{D}\right) \frac{2L - 1}{(3 - L_2)\overline{L}_2} \; \frac{M_o}{F} \qquad (3.6\text{-}14)$$

where:

$$\overline{L}_2 = \frac{FL_2}{M_o} \; ; \; \overline{L} = \frac{FL_1}{Mo} \; ; \; D = \text{Pipe diameter (in.).}$$

Verification of the energy balance method by comparison with results obtained by finite difference calculations is documented in Tables 3.6-8 and 3.6-9 for a series of circumferential break models of the type shown in Figure 3.6-15 (item b). The tables compare restraint displacements given by the two methods. In all cases the bending strain in the pipe at the restraint as calculated by the finite difference program is less than half the strain at ultimate stress for this material.

3.6.2.2.2.2.6 Finite Difference Analysis

A finite difference formulation specialized to the case of a straight beam and neglecting axial inertia and large deflection effects is used for the analysis of pipe whip of stainless steel pipes. The dynamic analysis is performed by direct numerical time integration of the equations of motion.

The equations of motion are of the form:

$$h\,(p_k - m_k\,\ddot{y}_k) = -\,M_{k+1} + 2M_k - M_{k-1} \qquad (3.6\text{-}15)$$

where:

- h is the node spacing
- P_k is the externally applied lateral loads at node k
- m_k is the lumped mass at node k
- $\ddot{y}_k$ is the lateral deflection at node k

and

- M_k is the internal resisting moment in the beam at node k.

Power law moment-curvature relationship is assumed and the central difference approximation for the curvature,

$$1 / h^2(-Y_{k+1} + 2y_k - Y_{k-1})$$

is used.

A timewise central-difference scheme is used to solve the dynamic equations

$$y(t + \Delta t) = \Delta t^2 \ddot{y}(t) + 2y(t) - y(t - \Delta t) \qquad (3.6\text{-}16)$$

and for the first time step

$$y(\Delta t) = \Delta t^2 \ddot{y}(O) \qquad (3.6\text{-}17)$$

A time step equal to 1/10 the shortest period of vibration is used in the integration.

3.6.2.2.2.2.7 Elastic-Plastic Moment Curvature Law

The pipe is assumed to obey an elastic-strain hardening plastic moment-curvature law with isotropic strain hardening. The symbols used are defined as follows:

M = Moment
$\overline{M}$ = current yield moment
E = elastic modulus of material at temperature
I = moment of inertia
Z = EI
ϕ = Curvature
ϕ_c = M/Z = elastic curvature
$\Delta\phi_p$ = increment of plastic curvature
ϕ_p = $\Sigma|\Delta\phi|$ = effective plastic curvature
ϕ_o = $\Sigma\ \Delta\phi\rho$ = permanent set curvature

At the end of each integration step new values of ϕ are calculated at each node.

The known values of ϕ_p, ϕ_o, and $\overline{M}$ at the start of the step are used to calculated M, $\overline{M}$, and $\Delta\phi_p$ by the following procedure:

if $|\phi - \phi_o| < \overline{M}/Z$

$$M = Z\ (\phi - \phi_o)$$

and

$$\Delta\phi_p = 0$$

if $|\phi - \phi_o| > \overline{M}/Z$

$M = \overline{M} = F(|\phi - \phi_o| + \phi_p)$ sign $(\phi - \phi_o)$ and $\Delta\phi_p = \phi - \phi_o - \overline{M}/Z$

where $F(\phi) = K(\phi)n$.

3.6.2.2.2.2.8 Power Law Moment Curvature Relationship

The following stress strain law is assumed in the plastic range:

$$\sigma = K(\varepsilon)^n \tag{3.6-18}$$

The corresponding moment-curvature law is

$$M = K(\phi)^n \tag{3.6-19}$$

where:

$$K = \frac{2\sqrt{\pi}}{3+n}(R_o^{3+n} - R_i^{3+n})\frac{\Gamma(1/2n+1)}{\Gamma(1/2n+3/2)}\bar{K} \tag{3.6-20}$$

or, to a good approximation:

$$K = \frac{4\bar{K}}{3+n}(1 - .291n - .076\,n^2)(R_o^{3+n} - R_i^{3+n}) \tag{3.6-21}$$

in which:

R_o = pipe outside radius
R_i = pipe inside radius

In the elastic range the moment-curvature law is:

$$M = EI\phi \tag{3.6-22}$$

The transition from elastic to plastic behavior on initial loading occurs at:

$$\phi = \frac{(EI)}{K}^{\frac{1}{n-1}} \tag{3.6-23}$$

3.6.2.2.2.2.9 Strain Rate Effects

The effect of strain rate in carbon steel is accounted for by using a rate dependent stress strain law of the form

$$\sigma(\varepsilon,\dot{\varepsilon}) = \left[1 + \frac{\dot{\varepsilon}}{(40.4)}\right]^{1/5} G(\varepsilon)$$

where G(ε) is the static stress-strain relationship. For stainless steels, the effect of strain rate is less pronounced so that a 10% increase in yield and ultimate strengths is used. The selection of material properties is discussed in Attachment B3.6.

3.6.2.2.2.2.10 Restraint Behavior

The analysis is capable of handling the bilinear or power law restraint behavior as shown in Figure 3.6-18. The behavior of the restraint is unidirectional. The restraint unloads elastically only to zero state, being left with a permanent set, and reloads along the same curve as shown in Figure 3.6-18.

3.6.2.2.2.3 Method of Dynamic Analysis of Unrestrained Pipes

The impact velocity and kinetic energy of unrestrained pipes is calculated on the basis of the assumption that the segments each side of the break act as rigid-plastic cantilever beams subject to piecewise constant blowdown forces. The hinge location is fixed either at the nearest restraint or at a point determined by the requirement that the shear at an interior plastic hinge is zero. The kinetic energy of an accelerating cantilever segment is equal to the difference between the work done by the blowdown force and that done on the plastic hinge. The impact velocity V is found from the expression for the kinetic energy:

$$KE = (1/2)\ M_{eg} V_I^2$$

where M_{eg} is the mass of the single degree of freedom dynamic model of the cantilever. The impacting mass is assumed equal to M_{eg}.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 Reactor Coolant Loop Pipe Whip Restraints and Jet Deflectors

As discussed in Subsection 3.6.2.1.1, the dynamic effects of postulated pipe breaks in the reactor coolant loop primary piping, accumulator line piping, and reactor coolant loop bypass piping can be eliminated from the structural design basis. Therefore, whip restraints or jet deflectors are not required.

3.6.2.3.2 Pipe Whip Restraints Inside Containment

This subsection applies to pipe whip restraints for all piping other than the reactor main coolant piping which connects the reactor vessel, the main coolant pumps, and the steam generators.

The methodology employed in the analysis of pipe whip is explained in detail in Subsection 3.6.2. Standard Review Plan

3.6.2 is followed. As discussed in the previous section, plant design features eliminate most pipe whip concerns.

Break locations have been defined for all high energy lines following the procedures in Standard Review Plan Section 3.6.2. Structural, piping, electrical and equipment target locations have been identified in the vicinity of the breaks and the potential for damage assessed. Restraints have been added where required to protect the plant structure or systems.

The main steam and feedwater systems are of significant concern due to the large size and high pressure.

In the remaining systems for which high energy line breaks must be postulated (CV, AS, SD, WX, AB systems), the lines in many cases are not highly stressed or do not have the potential of impacting safety systems.

3.6.2.3.2.1 General Description of Pipe Whip Restraints

Pipe whip restraints are designed and installed such that they do not offer thermal or seismic constraint/restraint to any piping. This is accomplished by providing adequate clearances and gaps to ensure that pipe whip restraints influence the piping only if a break should occur. Since all restraints are of an "unmovable" design and maximum piping temperatures are not in the creep range, the clearances and gaps established during installation will not change over the life of the plant. Therefore, there is no need for a procedure for ensuring that throughout the life of the plant, the restraints will not adversely affect the stresses in the pipes on which the restraints are installed.

Pipe whip restraints are provided to protect the plant against the effects of whipping during postulated pipe break. The design of pipe whip restraints is governed not only by the pipe break blowdown thrust, but also by functional requirements, deformation limitations, properties of whipping pipe and the capacity of the support structure. A pipe whip restraint consists of basically a ring around the pipe and components supporting the ring from the supporting structure. The diameter of the ring is established considering the pipe diameter, maximum thermal movement of pipe, thickness of insulation, and an additional 1/2 inch for installation tolerance. The restraint is designed for the impact force induced by the gap between the ring and the pipe.

This impact energy is usually too high for any elastic restraint system or support structure to absorb. Therefore energy absorbing measures designed by the energy balance approach (impact energy + external work = internal energy of pipe-restraint-structure system), are provided.

Pipe whip restraints on the Byron/Braidwood projects utilize a tension-compression system in which the legs of the restraints function as elements in a truss. The energy absorbing material is utilized only in taking compression loads in the restraint leg which is in compression under a given loading condition. The energy absorbing material (EAM) is not assumed to take any lateral load in the analysis of the restraints. However, during compression of the EAM in certain configurations, an angularity of load results. The effects of this angularity are considered to be minor.

3.6.2.3.2.2 Pipe Whip Restraint Components

Pipe whip restraints consist of the following components:

a. Energy Absorption Members - Members that under the influence of impacting pipes (pipe whip) absorb energy by significant plastic deformations (e.g., rods, and crushable honeycomb material).

b. Connecting Members - Those components which form a direct link between the pipe and the structure (e.g., ring and components other than energy absorption members).

c. Structural Attachments - Those fasteners which provide the method of securing the restraint connecting members to the structure (e.g., weld attachment).

d. Structural Components - Steel and concrete structures which ultimately carry the restraint load. Design criteria are specified in Section 3.8.

3.6.2.3.2.3 Design Loads

Restraint design loads, the reactions and the corresponding deflections are established using the criteria delineated in Subsection 3.6.2.2.2.2.

3.6.2.3.2.4 Allowable Stresses

The allowable stresses are as follows:

a. For energy absorption members - 0.95 Fy with 0.5 ε_u strain for steel in tension, where Fy is considered 15% higher than the Fy established according to the static test specified by ASTM and ε_u is the ultimate strain of steel at 0.16; and 6 ksi with 0.5 strain for crushable honeycomb in compression.

   The higher value for the allowable stress for energy absorbing tension steel members is comprised of the 10% dynamic increase factor in addition to a 5% increase factor for strain hardening effects.

This value is only 5% above the acceptable value (10%) which is given in Paragraph III.2.a of Standard Review Plan 3.6.2.

The energy balance method is used as the basis for pipe whip restraint analysis. The restraint resistance is assumed to be elastic-perfect plastic. In actuality, the material undergoes strain hardening much below 50% of the ultimate strain. The assumed 5% increase representing the strain hardening effect based on equivalent energy is a lower bound estimate and therefore conservative. Hence, the 5% increase in allowable stress above which is given in SRP Section 3.6.2 is acceptable.

The design of honeycomb material was based on energy absorption principles. The deflection is controlled by the design energy. The honeycomb material thickness is designed such that the strain under this deflection is less than approximately 50% of the ultimate strain and lies within the horizontal portion of the stress strain curve of the material. This ensures that the honeycomb material will not experience a deflection in excess of that defined by the horizontal portion of the load deflection curve.

Test specimens were taken from each lot of honeycomb material and precrushed to determine its actual dynamic crush strength and dynamic strain. The dynamic crush strength is maintained at ± 7% for at least 95% of the minimum usable strain. To ensure that energy absorption requirements are met, an adjusted cross-sectional area is determined based on the actual dynamic crush strength and dynamic strain.

b. For connecting member - 1.6 times the AISC allowable stress but not to exceed 0.95 Fcr where Fcr is Fy for bending and 0.55 Fy for shear, except for compression members, the allowable stress is 0.9 times the buckling stress Fbu as follows:

Fbu = 5/3 x Fa x DIF

where: 5/3 = Lower bound factor of safety in AISC for compression stress

Fa = AISC allowable compression stress

DIF = Dynamic increase factor = 1.1

c. For structural attachments and structural components - allowable stresses are the same as item b.

3.6.2.3.2.5 Design Criteria

The unique features in the design of pipe whip restraint components relative to the structural steel design are geared to the loads used and the allowable stresses. These are as follows:

a. Energy absorption members are designed for the reaction and the corresponding deflection established according to the pipe size and material and the blowdown force using the criteria delineated in Subsection 3.6.2.2.2.2.

b. Connecting members are designed for 1.25 times the reaction to ensure that the deflection required occurs in the energy absorption members instead of the connecting members.

c. The structural components and structural attachments are designed for 1.8 times the reaction. The 1.8 factor is the maximum dynamic load factor for 7% damping given in ASCE, Structural Design of Nuclear Plant Facilities, Volume 1-B, 1975, Page 1508.

3.6.2.3.2.6 Materials

The materials used are as follows:

a. For energy absorption members - ASTM A-193 Grade B7 for tension rods; and crushable honeycomb made of stainless steel for compression.

b. For other components - ASTM A-588, ASTM A572 Grade 50, and ASTM A36. Charpy tests are performed on materials subjected to impact loads and lamination tests are performed on members subjected to through thickness tension.

3.6.2.3.2.7 Jet Impingement Shields

The results of the HELB analysis of the as-built condition of piping outside containment have indicated that jet impingement shields are not required at Byron/Braidwood with the exception of shields for a small number of dampers in the boundary wall between the Turbine Building and the adjacent Auxiliary Building rooms that provide protection from Turbine Building HELBs.

3.6.2.3.3 Criteria for Protection Against Postulated Pipe Breaks in Reactor Coolant System Piping

A loss of reactor coolant accident is assumed to occur for a branch line break down to the restraint of the second normally open automatic isolation valve (Case II in Figure 3.6-23) on outgoing lines (Note: It is assumed that motion of the unsupported line containing the isolation valves could cause failure of the operators of both valves to function) and down to and including the second check valve (Case III in Figure 3.6-23) on incoming lines normally with flow. A pipe break beyond the restraint or second check valve will not result in an uncontrolled loss of reactor coolant if either of the two valves in the line close. Accordingly, both of the automatic isolation valves are suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint will not jeopardize the integrity and operability of the valves. Further, periodic testing capability of the valves to perform their intended function is essential. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV in Figure 3.6-23) a loss of reactor coolant accident is assumed to occur for pipe breaks on the reactor side of the valve.

Branch lines connected to the reactor coolant system are defined as "large" for the purpose of this criteria if they have an inside diameter greater than 4 inches up to the largest connecting line, generally the pressurizer surge line. A break of these lines results in a rapid blowdown from the reactor coolant system and protection is basically provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

Branch lines connected to the reactor coolant system are defined as "small" if they have an inside diameter equal to or less than 4 inches. This size is such that emergency core cooling system analyses using realistic assumptions show that no clad damage is expected for a break area of up to 12.5 in^2 corresponding to 4-inch inside diameter piping.

Engineered safety features are provided for core cooling and boration, pressure reduction, and activity confinement in the event of a loss of reactor coolant or steam or feedwater line break accident to ensure that the public is protected in accordance with 10 CFR 100 guidelines for accidents analyzed using TID-14844 or Regulatory Guide 1.183 for accidents using AST. These safety systems have been designed to provide protection for a reactor coolant system pipe break of a size up to and including a double ended break of the reactor coolant system main loop.

In order to assure the continued integrity of the vital components and the engineered safety systems, consideration is given to the consequential effects of the pipe break itself to the extent that:

a. The minimum performance capabilities of the engineered safety systems are not reduced below that required to protect against the postulated break.

b. The containment leaktightness is not decreased below the design value if the break leads to a loss of reactor coolant. (Note: The containment is here defined as the containment structure liner and penetrations and the steam generator shell, the steam generator steam side instrumentation connections, the steam, feedwater, blowdown, and steam generator drain pipes within the containment structure.)

3.6.2.3.3.1 Large Reactor Coolant System Piping

a. Propagation of damage resulting from a break of the main reactor coolant loop is permitted to occur but must not exceed the design basis for calculating containment and subcompartment pressure, loop hydraulic force, reactor internals reactor loads, primary equipment support loads, or ECCS performance.

b. Large branch line piping, as defined in Subsection 3.6.2.3.3, is restrained to meet the following criteria in addition to items a and b of Subsection 3.6.2.3.3.

    1. Propagation of the break is permitted to occur only within the limits of Table 3.6-4.

    2. Where restraints on the lines are necessary in order to prevent impact on and subsequent damage to the neighboring equipment or piping, restraint type and spacing are chosen such that a plastic hinge of the pipe at the two support points closest to the break is not formed.

3.6.2.3.3.2 Small Branch Lines

In the unlikely event that one of the small pressurized lines, as defined in Subsection 3.6.2.3.3, should fail and initiate a loss-of-coolant accident, the piping is restrained or arranged to meet the limits of Table 3.6-4 in addition to items a through b in Subsection 3.6.2.3.3.1.

3.6.2.3.3.3 Protective Provisions for Vital Equipment

In addition to pipe restraints, barriers and layout are used to provide protection from pipe whip, blowdown jet, and reactive forces.

Some of the barriers utilized for protection against pipe whip are the following. The secondary shield wall serves as a barrier between the reactor coolant loops and the containment liner. In addition, the refueling cavity walls, the operating floor, and the secondary shield wall, enclose each reactor coolant loop into a separate compartment, thereby preventing an accident which may occur in one loop from affecting another loop or the containment liner. The portion of the steam and feedwater lines within the containment have been routed behind barriers which separate these lines from all reactor cooling piping. The barriers described above will withstand loadings caused by jet forces and pipe whip impact forces.

Other than for the emergency core cooling system lines, which must circulate cooling water to the vessel, the engineered safety features are located outside the secondary shield wall. The emergency core cooling system lines which penetrate the secondary shield wall are routed around and outside the secondary shield wall to penetrate the secondary shield wall in the vicinity of the loop to which they are attached.

It has been demonstrated by Westinghouse Nuclear Energy System tests that lines hitting equal or larger size lines of same schedule will not cause failure of the line being hit e.g., a 1-inch line, should it fail, will not cause subsequent failure of a 1-inch or larger size line. The reverse, however, is assumed to be probable i.e., a 4-inch line, should it fail and whip as a result of the fluid discharged through the line, could break smaller size lines such as neighboring 3-inch or 2-inch lines. In this case, the total break area is less than 12.5 in^2.

Alternately, if the layout is such that whipping of the two free sections cannot reach equipment or other pipes for which protection is required, plastic hinge formation is allowed. As another alternative, barriers are erected to prevent the whipping pipe from impacting on equipment or piping requiring protection. Finally, tests and/or analyses are performed to demonstrate that the whipping pipe will not cause damage in excess of acceptable limits.

Whipping in bending of a broken stainless steel pipe section as used in the reactor coolant system does not cause this section to become a missile. This design basis has been demonstrated by Westinghouse Nuclear Energy Systems bending tests on large and small diameter, heavy and thin walled stainless steel pipes.

The methods described below are used in the Westinghouse design and verification of adequacy of primary reactor coolant loop components and supports. It is emphasized that these methods are used only to determine jet impingement loads on components and supports and are not used for design and checking of walls, barriers, cable trays, etc. Although the dynamic effects of postulated pipe breaks in the reactor coolant loop primary piping, accumulator line piping, and reactor coolant loop bypass piping can be eliminated from the structural design basis (see Subsection 3.6.2.1.1), the design verification of certain components and supports may retain the original jet impingement loadings. For these cases, the following subsection describes the methods used in the analysis.

The design-basis postulated pipe break locations for the reactor coolant loop piping are determined using the criteria given in Subsection 3.6.2. These design basis breaks are used here as the break locations for consideration of jet impingement effects on primary equipment and supports.

The dynamic analysis, as discussed in Subsection 3.6.2.2.1.3, is used to determine maximum piping displacements at each design-basis break location. These maximum piping displacements are used to compute the effective break flow area at each location. This area and break orientation are then used to determine the jet flow pattern and to identify any primary components and supports which are potential targets for jet impingement.

The jet thrust at the point of the break is based on the fluid pressure and temperature conditions occurring during normal (100%) steady-state operating conditions of the plant. At the point of the break, the jet force is equal and opposite to the jet thrust. The force of the jet is conservatively assumed to be constant throughout the jet flow distance. The subcooled jet is assumed to expand uniformly at a half-angle of 10° from which the area of the jet at the target and the fraction of the jet intercepted by the target structure can be readily determined.

The shape of the target affects the amount of momentum change in the jet and thus affects the impingement force on the target. The target shape factor is used to account for target shapes which do not deflect the flow 90° away from the jet axis.

The method used to compute the jet impingement load on a target is one of the following:

a. The dynamic effect of jet impingement on the target structure is evaluated by applying a step load whose magnitude is given by

$$F_j = K_o\ P_o\ A_{mB}\ RS$$

where:

F_j = jet impingement load on target

K_o = dimensionless jet thrust coefficient based on initial fluid conditions in broken loop

P_o = initial system pressure

A_{mB} = calculated maximum break flow area

R = fraction of jet intercepted by target

S = target shape factor.

Discharge flow areas for limited flow area circumferential breaks are obtained from reactor coolant loop analyses performed to determine the axial and lateral displacements of the broken ends as a function of time. A_{mB} is the maximum break flow area occurring during the transient, and is calculated as the total surface area through which the fluid must pass to emerge from the broken pipe. Using geometrical formulations, this surface area is determined to be a function of the pipe separation (axial and transverse) and the dimensions of the pipe (inside and outside diameter).

If a simplified static analysis is performed instead of a dynamic analysis, the above jet load (F_j) is multiplied by a dynamic load factor. For an equivalent static analysis of the target structure, the jet impingement force is multiplied by a dynamic load factor of 2.0. This factor assumes the target can be represented as essentially a one degree of freedom system and the impingement force is conservatively applied as a step load.

The calculation of the dimensionless jet thrust coefficient and break flow area is discussed in Subsection 3.6.2.5.

b. The dynamic effect of jet impingement is evaluated by applying the following time-dependent load to the target structure.

$$F_j = K\ P\ A_{mB}\ RS$$

where the system pressure P is a function of time; the jet thrust coefficient K is evaluated as a function of system pressure and enthalpy; and the break flow area A_{mB} is a function of time.

3.6.2.3.3.4 Pipe Restraints and Locations

Reactor coolant loop pipe restraints are discussed in Subsection 3.6.2.3.1.

3.6.2.3.3.5 Design Loading Combinations

As described in Section 3.9, the forces associated with the break of reactor piping systems are considered in combination with normal operating loads and earthquake loads for the design of supports and restraints in order to assure continued integrity of vital components and engineered safety features. Although the dynamic effects of postulated pipe breaks in the reactor coolant loop primary piping, accumulator line piping, and reactor coolant loop bypass piping can be eliminated from the structural design basis (see Subsection 3.6.2.1.1), the design verification of certain structures and components may retain the original pipe break loadings.

The stress limits for reactor coolant piping and supports are discussed in Section 3.9.

3.6.2.4 Guard Pipe Assembly Design Criteria

Guard pipe assemblies were utilized in the design of the Byron and Braidwood Stations for the recirculation sump piping and the fuel transfer tube. The guard pipes on these moderate energy lines are used to ensure containment integrity.

The guard pipe for the recirculation sump piping extends from the recirculation sump to the sump suction valve protection chamber. A seal ring exists between the guard pipe and the recirculation sump piping which serves as the containment boundary. The seal rings are subjected to Appendix J leakage testing as part of the containment integrated leak rate test. The section of guard pipe and seal ring that serve as the containment boundary are classified as ASME Section III, Class MC. The sump suction valve protection chamber and the section of guard pipe that extends beyond the containment boundary are classified as ASME Section III, Class 2.

The guard pipe for the fuel transfer tube extends along the length of the fuel transfer tube from the inside of containment, through the containment wall, to the outside of containment. The portion of the guard pipe from the containment liner of the 3'-6" wall, across the bellows towards the inside of containment, including the end flange of the tube on the inside of containment, then back towards the containment liner, serves as the containment boundary. This section of guard pipe is classified as ASME Section III, Class MC and is subjected to Appendix J leakage testing as part of the local leak rate testing program. The remainder of the guard pipe is maintained as ASME Section III, Class MC, but is not subject to hydrostatic testing or code stamping.

3.6.2.5 Dynamic Analysis Applicable to Postulated High Energy Pipe Break

3.6.2.5.1 Reactor Coolant Loops

a. The dynamic effects of postulated pipe breaks in the reactor coolant loop primary piping, accumulator line piping, and reactor coolant loop bypass piping can be eliminated from the structural design basis (see Subsection 3.6.2.1.1). The RHR line and pressurizer surge line connections remain as postulated break locations. These two locations are not eliminated by the reactor coolant loop or the accumulator line piping and reactor coolant loop bypass piping LBB analysis.

b. Design loading combinations and applicable criteria for ASME Class 1 components and supports are provided in Subsection 3.6.2.3.3.5. Pipe break loads include not only the jet thrust forces acting on the piping but also jet impingement loads on the primary equipment and supports.

c. The interface between Sargent & Lundy and Westinghouse concerning the design of the primary equipment supports and the interaction with the primary coolant loop is described in Subsection 3.9.3.4.4.1.

3.6.2.5.2 Postulated Breaks in Piping Other than Reactor Coolant Loop

The following material pertains to dynamic analyses completed for piping systems other than the reactor main coolant piping which connects the reactor vessel, the main coolant pumps, and the steam generators.

3.6.2.5.2.1 Implementation of Criteria for Defining Pipe Break Locations and Configurations

The locations and number of design basis breaks, including postulated break orientations, for the high energy piping systems are shown in Figures 3.6-25 through 3.6-99.

The above information was derived from the implementation of the criteria delineated in Subsection 3.6.2.1.

Stress levels and usage factors (usage factors for Class 1 piping only) for the postulated break locations are shown in Tables 3.6-11 and 3.6-12.

For Turbine Building HELBs, the selection of pipe break locations and configuration are described in UFSAR Sections 3.6.2.a for the environmental analysis, and 3.6.2.2 for the evaluation of pipe whip and jet impingement effects.

3.6.2.5.2.2 Implementation of Criteria Dealing with Special Features

Special protective devices in the form of pipe whip restraints and impingement shields are designed in accordance with Subsection 3.6.2.3.

Inservice inspection is discussed in Subsection 3.6.1.2.2.

3.6.2.5.2.3 Acceptability of Analyses Results

The postulation of break and crack locations for high and moderate energy piping systems and the analyses of the resulting jet thrust, impingement and pipe whip effects has conservatively identified areas where restraints, impingement shields, or other protective measures are needed and has yielded the conservative design of the required protective devices.

Results of jet thrust and pipe whip dynamic effects are given in Tables 3.6-13 and 3.6-14.

3.6.2.5.2.4 Design Adequacy of Systems, Components, and Component Supports

For each of the postulated breaks, the equipment and systems necessary to mitigate the consequences of the break and to safely shut down the plant (i.e., all essential systems and components) have been identified (Subsection 3.6.1). The equipment and systems are protected against the consequences of each of the postulated breaks to ensure that their design-intended functions will not be impaired to unacceptable levels as a result of a pipe break or crack.

When it became necessary to restrict the motion of a pipe which would result from a postulated break, pipe whip restraints were added to the applicable piping systems, or structural barriers or walls were designed to prevent the whipping of the pipe.

Design adequacy of the pipe whip restraints is demonstrated in Tables 3.6-13 and 3.6-14. Data in the tables was obtained through use of the criteria delineated in Subsection 3.6.2.1 through 3.6.2.3 inclusive.

The design adequacy of structural barriers, walls, and components is discussed in Section 3.8.

For Turbine Building HELBs, pipe whip as a result of a HELB is not a concern. There are no safety related components in the Turbine Building that are required for safe shutdown of the Unit that can be impacted by pipe movement (including jet thrust.) Additionally, if a pipe were to damage another high- or moderate-energy line, the pressure in the Turbine Building from the first break would have caused the dampers protecting the adjacent auxiliary building rooms to isolate. Therefore, a second break would not increase the environmental conditions in the rooms containing the safety-related equipment.

There are high- and moderate- energy piping subsystems in the vicinity of the L-Line wall separating the Turbine Building from the Auxiliary Building and the dampers and doors integral to L-Line wall. L-Line wall is a concrete re-enforced wall 42” in depth (a safety related, Seismic Category I structure). Although a pipe hitting the wall could cause surface damage to the concrete, the strength and thickness of the wall would prevent structural failure of the wall. For the dampers and doors integral to L-Line wall, the evaluation has determined that the doors and dampers would not be adversely impacted by pipe whip.

3.6.2.5.2.5 Implementation of the Criteria Related to Protective Assembly Design

Guard pipes or protective assembly designs were utilized in the design of the Byron and Braidwood Stations only for the containment penetrations for the fuel transfer tube and the recirculation sump piping. The guard pipes on these moderate energy lines are used to ensure containment integrity.

3.6.3 References

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ATTACHMENT A3.6

SUBCOMPARTMENT PRESSURIZATION

STUDIES OUTSIDE THE CONTAINMENT

TABLE OF CONTENTS

ATTACHMENT A3.6 - SUBCOMPARTMENT PRESSURIZATION STUDIES OUTSIDE THE CONTAINMENT

A3.6.1 Introduction

In accordance with the Nuclear Regulatory Commissions' Standard Review Plan, Subsection 6.2.1.2 (Subcompartment Analysis), a transient differential pressure response analysis is completed for all subcompartments containing high energy fluid lines. This attachment discusses the results of such analyses for the auxiliary building subcompartments.

If a high-energy fluid line is postulated to break within a subcompartment, the sudden discharge of this fluid will cause transient differential pressure across the walls of the subcompartment. Therefore, for a particular structural design, the results of a transient differential pressure and temperature response analysis should be an integral part of the structural design criteria.

Seven different subcompartments were analyzed (see Figures A3.6-1 through A3.6-25):

a. recycle waste evaporator rooms elevation 346 feet 0 inch, auxiliary building basement;

b. letdown reheat heat exchanger rooms and valve aisles at elevation 346 feet 0 inch, auxiliary building basement;

c. blowdown condenser room at elevation 364 feet 0 inch, auxiliary building upper basement;

d. letdown heat exchanger rooms and valve aisles at elevation 383 feet 0 inch, auxiliary building upper basement;

e. auxiliary steamline piping tunnel at elevation 394 feet 0 inch, auxiliary building upper basement (Braidwood only);

f. surface condenser room at elevation 401 feet 0 inch, auxiliary building ground floor (Braidwood only); and

g. radwaste evaporator room at elevation 414 feet 0 inch, auxiliary building ground floor (Braidwood only).

Pertinent high energy fluid lines, in which circumferential breaks were postulated, are listed in Figure A3.6-1 along with the initial operating conditions and computed choked or limited break mass discharge rates. At Byron, the steam supply to the auxiliary steamline piping tunnel, surface condenser rooms, and radwaste evaporator rooms has been permanently isolated in the turbine building by the installation of a blank plate.

At Braidwood, blank-off plates have been installed in the auxiliary steam piping in the Turbine Building isolating the steam from the subcompartments described in a, e, f and g above. Since the possibility exists that the blank-off plates could be removed in the future, the analysis for high energy line breaks in these subcompartments remains current and in place.

A3.6.1.1 <u>Subcompartment Pressurization</u>

For the purpose of protecting subcompartments from overpressurization, the CV, AS, SD, WX, MS, and FW systems were traced through the auxiliary building and all subcompartments containing high energy line were identified. The most severe break in the subcompartment was analyzed.

The main steam (MS) and feedwater (FW) systems are routed entirely in an enclosed tunnel in the auxiliary building. The limiting break in this tunnel is a main steamline break. Section C3.6 fully describes the subcompartment pressurization analysis of a break in this tunnel.

The remainder of the auxiliary building was surveyed level by level to identify all subcompartments which could be pressurized by high energy line breaks. Figures 3.6-100 through 3.6-104 identify all areas containing high energy lines. The identification of the limiting line in each zone is also included. The zone numbers do not correspond to environmental qualification zones (Section 3.11).

Figure 3.6-100 represents elevation 346 feet 0 inch. Zone 1, the recycle waste evaporator room, has been analyzed and the results are reported in this section. Zones 2 and 3, letdown reheat heat exchanger rooms and valve areas, have been analyzed and the results are reported in this section. The assessment in this section addressed Zone 3, the more limiting zone.

Figure 3.6-101 represents elevation 364 feet 0 inch. Zones 5A, 5B, 6A, 6B, 7A, and 7B, the positive displacement and centrifugal charging pump rooms, contain high pressure, low temperature lines. Failure of these lines (normal temperature of 115°F) will not cause pressurization or increase temperatures. Pipe whip and impingement are considered. Zones 9A and 9B contain portions of the steam generator blowdown system. Control valves upstream of these lines limit the blowdown flow and prevent the postulated breaks from impacting plant design. Zones 8A and 8B, blowdown condenser rooms, have been analyzed and the results are included in Subsection A3.6.4. Zones 11 and 12, blowdown condenser rooms, have been analyzed and the results are reported in Subsection A3.6.4.

Figure 3.6-102 represents elevation 383 feet 0 inch. Zones 11A, 11B, 11C, and 11D, letdown heat exchanger rooms, have been analyzed and the results are reported in Subsection A3.6.4. Zone 13, the auxiliary steamline piping tunnel (Braidwood only), has been analyzed and the results are reported in Subsection A3.6.4. At Byron, a blank plate has been installed in the steam supply to the auxiliary steamline piping tunnel (Zone 13). Zones 12A and 12B are very similar to Zones 11A through 11D in break size and subcompartment size and, therefore, the existing results are adequate. Zones 10A and 10B are large areas with only small high energy lines. The impact of a break in these areas is discussed in Subsection 3.6.1.3.1.

Figure 3.6-103 represents elevation 401 feet 0 inch. Zones 16A, 16B, and 16C, the surface condenser rooms (Braidwood only), have been analyzed and the results have been reported in Section A3.6. At Byron, a blank plate has been installed in the steam supply to the surface condenser rooms (Zones 16A, 16B, and 16C). Zone 14 is a large open area. The limiting high energy line break is a 2-inch auxiliary steamline. This event is discussed in Subsection 3.6.1.3.1.

Figure 3.6-104 represents elevation 426 feet 0 inch. Zones 18A, 18B, and 18C, radwaste evaporator rooms (Braidwood only), have been analyzed and the results are reported in Subsection A3.6.4. At Byron, a blank plate has been installed in the steam supply to the radwaste evaporator rooms (Zones 18A, 18B, and 18C).

A3.6.2 Basic Assumptions

Several assumptions were common to all analyses. They are the following:

a. The free volume of each subcompartment was used as the effective control volume.

b. Only one break of the specific high energy line is assumed for each subcompartment.

c. With regards to Standard Review Plan (SRP), Section 6.2.1.2 (II.5a, III), the doors to the compartments were assumed to begin to open linearly at 0.5 psi and fully opened at 1.0 psi differential pressure. It was further assumed that the doors were of the destructive blow-off type so that they remain open once they become partially or completely open.

d. Once the pipe break was initiated, the atmosphere in the subcompartment was homogeneously mixed by the WARLOC and COMPARE/MODl codes, keeping water as a fine mist with 100% liquid carryover.

e. A multiplier of 0.6 was applied to the vent critical flow path calculation between nodes in accordance with the acceptance criteria of SRP Section 6.2.1.2 (II.6).

f. The calculated transient differential pressure was multiplied by a 1.4 factor in accordance with the acceptance criteria of SRP Section 6.2.1.2 (II.7). (The multiplier of 1.4 was not applied for Cases c and f. since this calculation was performed after the completion of construction.)

g. Moody's critical flow method with a multiplier of 1.0 as stated in the acceptance criteria of SRP Section

6.2.1.2 (II.2) was used to compute the break mass discharge rate when a limited flow rate was not available. A circumferential break was postulated for each of the lines listed in Figure A3.6-1.

h. There were no cutoff valves associated with the postulated broken high energy lines.

i. In analyses 1, 2, 3, 4, and 6, the subcompartment door was the only vent flow path.

A3.6.3 Analytical Model

The control volume (node) and vent flow path simulation mode are depicted in Figure A3.6-2. This model was utilized for all analyses except those associated with the postulated 16-inch auxiliary steamline break at elevation 394 feet 0 inch, elevation 401 feet 0 inch, and elevation 414 feet 0 inch.

The transient differential pressure and temperature time histories in each subcompartment were determined with the Sargent & Lundy WARLOC computer program (09.8.038-4.0) and the NRC's COMPARE/MODl code. These codes are multicell thermal-hydraulic transient analysis codes capable of predicting the short-term and long-term containment and associated subcompartment pressure and temperature response to an accidental (or abnormal) transient such as a design-basis loss-of-coolant accident or high energy pipe line break.

Figures A3.6-3, A3.6-6, A3.6-9, A3.6-12, and A3.6-19 (Braidwood only) list the dimensions and initial conditions of the control volumes and flow paths used for each analysis as input to the WARLOC and COMPARE codes.

A3.6.4 SUMMARY OF RESULTS

Figures A3.6-2 through A3.6-14 and Figures A3.6-18 through A3.6-25 (Braidwood only) depict the initial input, dimensions, and computed pressure-temperature time histories for each analysis. Following are brief summaries:

a. recycle waste evaporator room at elevation 346 feet 0 inch, peak transient differential pressure of 1.2 psid, maximum transient temperature of 295°F;

b. letdown reheat heat exchanger room at elevation 346 feet 0 inch, peak transient differential pressure of 0.7 psid, maximum transient temperature of 210°F;

c. blowdown condenser room at elevation 346 feet 0 inch, peak transient differential pressure of 2.54 psid, maximum transient temperature of 212°F;

d. letdown heat exchanger room at elevation 383 feet 0 inch, peak transient differential pressure of 0.7 psid, maximum transient temperature of 175°F;

e. auxiliary steamline piping tunnel at elevation 394 feet 0 inch, peak transient differential pressure of 1.85 psid, maximum transient temperature of 300°F (Braidwood only), at Byron, a blank plate has been installed in the steam supply to this zone such that the high temperature and pressure described is no longer possible;

f. surface condenser room at elevation 401 feet 0 inch, peak transient differential pressure of 1.25 psid, maximum transient temperature of 295°F (Braidwood only), at Byron, a blank plate has been installed in the steam supply to this zone such that the high temperature and pressure described is no longer possible; and

g. radwaste evaporator room at elevation 414 feet 0 inch, peak transient differential pressure of 1.8 psid, maximum transient temperature of 300°F (Braidwood only), at Byron, a blank plate has been installed in the steam supply to this zone such that the high temperature and pressure described is no longer possible.

A3.6.5 Conclusions

All of the results are conservative with respect to the assumption of not having cutoff valves in the postulated broken lines. However, the structural designs, in every case, could accommodate the postulated differential pressure transient or were modified accordingly.

The structural integrity of all mentioned subcompartments can be maintained in the very unlikely event of a circumferential break of any of the high energy lines in the auxiliary building.

ATTACHMENT B3.6

SELECTION OF PIPE MATERIAL PROPERTIES FOR USE IN PIPE WHIP ANALYSIS

ATTACHMENT B3.6

SELECTION OF PIPE MATERIAL PROPERTIES FOR USE IN PIPE WHIP ANALYSIS

A substantial amount of elevated temperature test data for A106 Grade B carbon steel is given in Reference 1. Material property values based on this data are used.

Since little test data is available for TP304, TP304L, and TP316 stainless steels ASME Code specified values are used with the realization that they are very conservative.

The power law stress strain relationship is used for all steels.

$$\sigma(\varepsilon) = K(\varepsilon)^n \tag{1}$$

The effect of strain rate in carbon steels is accounted for (as suggested in Reference 2) by modifying Equation 1 as follows:

$$\sigma(\varepsilon, \dot{\varepsilon}) = \left[1 + \left(\frac{\varepsilon}{D}\right) 1 / \rho\right] K\varepsilon^n \tag{2}$$

Where $D = 40.4\ \text{sec}^{-1}$

$\rho = 5$

This modification has been widely used - see for example References 2 and 3. For stainless steels the effect of strain rate is less pronounced (Reference 4) so that the use of a 10% increase in yield and ultimate strengths is used.

<u>A106 Grade B Material Properties at 600°F</u>

The results of tests on 71 specimens, in the temperature range of interest, are given in Reference 1. Twenty were tested at 600°F, 22 at room temperature, and the rest at temperatures between 200°F and 585°F. Yield stress (1) was shown to decrease - ultimate stress to increase - with increasing temperature. The minimum yield stress of any of the 71 specimens tested was 31.7 ksi, the average for the 20 tested at 600°F was 36.28 ksi with a sigma value of 3.67 ksi. The minimum ultimate stress value for all specimens was 64.4 ksi, the average for the 22 tested at room temperature 71.79 ksi with a sigma value of 4.72 ksi. The strength coefficient K and the hardening exponent n can be evaluated from the equations.

$$\sigma y = K(.002)^n \qquad (3)$$
$$\sigma u = Kn^n$$

Values of K and n obtained in this way are given in Table 1 below:

	Yield Stress (Ksi)	Ultimate Stress (Ksi)	K (Ksi)	n
Minimum	31.7	64.4	86.445	0.16142
Mean-Sigma	32.61	67.08	90.204	0.16371
Mean	36.28	71.79	95.971	0.15653

Table 1 Material Properties A106 GrB at 600°F based on data from Reference 2.

The mean-sigma values of K=90.204 Ksi and n = 0.16371, or more conservative values, are used for all temperatures 600°F and below.

(1) 0.2% offset values were measured

References:

1. R. J. Eiber, et al. Investigation of the Initiation and Extent of Ductile Pipe Rupture, Battelle Memorial Institute, Report BMI-1866, July 1969

2. S. R. Bodner and P. S. Symonds, "Experimental and Theoretical Investigation of the Plastic Deformation of Cantilever Beams Subjected to Impulsive Loading", JAM, December 1962

3. J. C. Anderson and A. K. Singh, "Inelastic Response of Nuclear Piping Subjected to Rupture Forces," ASME Paper No. 75-PVP-21

4. C. Albertini and M. Montagnani, "Wave Propagation Effects in Effects in Dynamic Loadings," Nuclear Engineering and Design, 37 115-124, 1976.

ATTACHMENT C3.6

MAIN STEAMLINE BREAK IN MAIN STEAM TUNNEL

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ATTACHMENT C3.6 - MAIN STEAMLINE BREAK IN MAIN STEAM TUNNEL

I. Introduction

One of the design criteria for the main steam tunnel and valve room subcompartments is to retain functional integrity indefinitely, that is, to have the capability of withstanding peak transient differential pressures under a postulated accident mode.

It was the purpose of this study to determine the transient pressure response in the Unit 1 and Unit 2 main steam tunnels and the associated safety valve rooms in the first and second quadrants at the time of a sudden and complete circumferential failure of a main steamline.

Six break locations for Unit 1 and four for Unit 2 were considered. The common locations are the lower valve room just downstream of the isolation valve; the main steam tunnel just outside the valve room in the first quadrant; the main steam tunnel between the first and second quadrants; and the main steam tunnel just outside the valve room in the second quadrant. Additionally, Unit 1 was evaluated for two breaks in the first quadrant between the valve room and the turbine building opening.

Qualification tests have been conducted for the components in the safety valve house. The components include the main steam and main feedwater isolation valves, the main steam power-operated relief valve, and the main steam safety valves. These tests conservatively applied aging, radiation, seismic, and worst case environmental (temperature, pressure, and humidity) loading to the components, and showed that loss of function did not occur.

The portion of the main steam and main feedwater pipe in the tunnel between the safety valve house and the turbine building meets the guidelines of Branch Technical Position APSCB3-1. A special pipe whip restraint is located around each pipe as it passes through the wall separating the isolation valve room from the main steam tunnel. This restraint limits the amount of strain that can be transmitted to the isolation valves from any pipe break in the tunnel to a level which will not interfere with the proper functioning of the isolation valves.

The safety valve house, the steam tunnel, and the compartment between the containment and the safety valve house all have the same basis for design. These compartments have been designed for pressurization, impingement, and temperature as specified in Table 3.8-10, load combinations 8, 13, and 14.

An assumed pipe crack or break in the tunnel, isolation valve room , or safety valve house cannot cause structural failure. The subcompartment pressurization analysis is included in this attachment. The methods used to calculate the pressure buildup in subcompartments outside the containment are the same as those used for subcompartments inside the containment.

II. Analysis

A. Description of the Computer Code

The analysis was carried out by using the RELAP code, which is a multicell thermal-hydraulic transient analysis computer program.

The basis for the computer code is a network of fluid control volumes (fluid nodes) and fluid paths (interconnecting control volumes) for which the conservation equations of mass, momentum, and energy are solved in space and time. Superimposed on the network are computer subroutines which permit physical modeling of the reactor system, the containment, plant subcompartments, safeguard fluid systems and the pipe break flow.

B. Simulation of the System

1. Assumptions

The following are the major assumptions used in this study:

a. The initial conditions in the steam tunnel and the valve rooms are 14.7 psia of pressure at a temperature of 90°F and a relative humidity of 30%.

b. Only one break occurred per analysis.

c. The Moody choked-flow calculation was used with a multiplier of 0.6 as required by the NRC for choked-flow check between nodes.

d. Homogeneous fine mist for the steam/water-air system in the control volumes with complete liquid carryover was used to produce a conservative solution.

e. The length of a flow path connecting any two control volumes was taken as the distance between the centroids of these volumes.

f. The area of a flow path is the effective area (i.e., the cross-sectional area of the path excluding areas occupied by grating, pipes, louvers, etc.).

g. Mass and energy release rate for a postulated main steamline break is included in Tables 4 and 5 for Unit 1 and Unit 2 respectively.

h. The doors and HVAC louvers/panels in the upper chambers of the valve rooms are initially assumed closed or intact. A differential pressure equal to 1.5 psi will blow open the doors and panels to atmosphere.

2. Analytical Model

To determine the transient pressures and temperatures in the main steam tunnel and the safety valve rooms after a sudden failure of a main steamline, the main steam tunnel was simulated by five control volumes connected by flow paths. The area of each flow path is equivalent to the net area of the steam tunnel.

The subcompartments of the valve room in each quadrant were represented by four control volumes connected by flow paths. The area of each flow path was equivalent to the total vent areas between subcompartments.

Figures 1 and 2 depict a plan of the system and the flow diagram of the analytical model used in the study, respectively.

Tables 1, 2 and 3 give the dimensions of the control volumes and flow paths, while Tables 4 and 5 show the blowdown rates and properties versus time from a postulated main steamline break as provided by Framatome Technologies International for Unit 1 (Reference 2) and Westinghouse for Unit 2, respectively.

III. Results and Conclusions

A. Unit 1 Results

The peak nodal differential pressures, which represent the difference between steam tunnel/valve room nodes and the surrounding areas, are presented in Table 1 and Figures C3.6-3 through C3.6-6.

The peak differential pressures across internal walls and floors are shown in Table 6 and Figures C3.6-7 and C3.6-8.

1. Pipe Break in the Main Steam Tunnel

Five locations of a postulated main steamline break were considered in the main steam tunnel. The first and second locations were just outside the valve rooms in the first and second quadrants (Nodes 6 and 8), respectively. The third location was in the steam tunnel between the first and second quadrants (Node 7). The last two break locations are in the tunnel leading to the entrance to the turbine building (Nodes 13 and 14).

Figures C3.6-4 through C3.6-6 show the differential pressures in the control volumes directly affected by the line breaks in the tunnel.

2. Pipe Break in Valve Room

A pipe break in the lower chamber of the valve room in the second quadrant (Node 5) was evaluated to provide the most conservative differential pressure.

Figure C3.6-3 shows the differential pressures in the control volumes directly affected by the line break in the valve room.

Tables 1 and 6 provide a summary of the peak pressures used in the qualification of the structure (References 3 and 4).

B. Unit 1 Conclusions

The integrity of the Unit 1 main steam tunnel and valve rooms in both the first and second quadrants can be maintained during a postulated main steamline break. These differential pressures, modified by dynamic load factors, were used to qualify the subject walls for the tunnels (Reference 3) and valve rooms (Reference 4).

C. Unit 2 Results

1. Pipe Break in the Main Steam Tunnel

Three locations of a postulated main steamline break were considered in the main steam tunnel. The first and second locations were just outside the valve rooms in the first and second quadrants (Nodes 6 and 8), respectively. The third location was in the steam tunnel between the first and second quadrants (Node 7).

Figures 10 through 18 show the differential pressures in the control volumes directly affected by the line break.

2. Pipe Break in Valve Room

A pipe break in the lower chamber of the valve room in the second quadrant (Node 5) was considered to give the most conservative differential pressure.

Figures 7 through 9 show the differential pressures in the affected control volumes after a line break.

Table 2 gives a summary of the peak pressures to the valves used in the design of the structure.

D. Unit 2 Conclusions

The integrity of the main steam tunnel, the auxiliary feedwater tunnel, and the valve rooms in both the first and second quadrants can be maintained during a postulated main steamline break.

IV. References

1. Calculation 3C8-0282-001, Revision 3.
2. NDIT 960136, "Steam Generator Replacement Project: Transmittal of Steam Line Break Mass and Energy for Steam Tunnel Pressure Analysis," dated September 16, 1996.
3. Calculation 5.6.1-BYR96-233/5.6.1-BRW-96-604, Revision 0.
4. Calculation 5.6.3-BYR96-234/5.6.3-BRW-96-608, Revision 0.

TABLE 1

UNIT 1 SUBCOMPARTMENT NODAL DESCRIPTION

MAIN STEAM LINE BREAK IN UNIT 1 MAIN STEAM TUNNEL OR VALVE ROOMS

VOLUME NO.	DESCRIPTION	HEIGHT ft	CROSS-SECTIONAL AREA ft^2	VOLUME ft^3	INITIAL CONDITIONS TEMP. °F	PRESS. psia	HUMID.* %	DBA BREAK CONDITIONS BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS DIFF. psid	DESIGN PEAK PRESS DIFF. psid	DESIGN MARGIN %
1	Atmosphere	$5x10^3$	$1x10^3$	107	90	14.7	30	-	-	-	-	-	-	-
2	2nd Quad-rant Upper Valve Chamber	12.33	133.25	4895.7	90	14.7	30	5	Main Steam	1.4	Double-ended Guillo-tine	15.3	26.2	71
3	2nd Quad-rant Upper Valve Chamber	12.33	183.7	4895.7	90	14.7	30	5	Main Steam	1.4	Double-ended Guillo-tine	15.3	26.2	71
4	2nd Quad-rant Lower Valve Chamber	24.00	213.0	6007.0	90	14.7	30	5	Main Steam	1.4	Double-ended Guillo-tine	17.7	26.2	48
5	2nd Quad-rant Valve Chamber	24.00	213.0	6007.0	90	14.7	30	5	Main Steam	1.4	Double-ended Guillo-tine	17.7	28.6	62

* Relative humidity.

TABLE 1 (Cont'd)

VOLUME NO.	DESCRIPTION	HEIGHT ft	CROSS-SECTIONAL AREA ft^2	VOLUME ft^3	INITIAL CONDITIONS TEMP. °F	PRESS. psia	HUMID.* %	DBA BREAK CONDITIONS BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS DIFF. psid	DESIGN PEAK PRESS DIFF. psid	DESIGN MARGIN %
6	2nd Quad-rant Main Steam Tunnel	19.00	317.0	13695.0	90	14.7	30	6	Main Steam	1.4	Double-ended Guillo-tine	16.0	26.5	66
7	Main Steam Tunnel	19.00	203.00	34865.0	90	14.7	30	7	Main Steam	1.4	Double-ended Guillo-tine	15.2	21.3	40
8	1st Quad-drant Steam Tunnel	20.00	432.0	35016.0	90	14.7	30	8	Main Steam	1.4	Double-ended Guillo-tine	8.3	11.2	35
9	1st Quad-rant Upper Valve Chamber	12.33	133.25	4895.7	90	14.7	30	5**	Main Steam	1.4	Double-ended Guillo-tine	15.3	26.2	71
10	1st Quad-rant Upper Valve Chamber	12.33	183.7	4895.7	90	14.7	30	5**	Main Steam	1.4	Double-ended Guillo-tine	15.3	26.2	71

* Relative humidity.
** Differential pressures calculated for 1st quadrant valve chambers are also applicable to corresponding 2nd quadrant valve chambers.

TABLE 1 (Cont'd)

VOLUME NO.	DESCRIPTION	HEIGHT ft	CROSS-SECTIONAL AREA ft^2	VOLUME ft^3	INITIAL CONDITIONS TEMP. °F	PRESS. psia	HUMID.* %	DBA BREAK CONDITIONS BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS DIFF. psid	DESIGN PEAK PRESS DIFF. psid	DESIGN MARGIN %
11	1st Quad-rant Valve Chamber	24.00	213.0	6007.0	90	14.7	30	5**	Main Steam	1.4	Double-ended Guillo-tine	17.7	28.6	62
12	1st Quadrant Lower Valve Chamber	24.00	213.0	6007.0	90	14.7	30	5**	Main Steam	1.4	Double-ended Guillo-tine	17.7	28.6	62
13	1st Quad-rant Main Steam Tunnel	29.00	432.0	17388.4	90	14.7	30	13	Main Steam	1.4	Double-ended Guillo-tine	10.7	13.2	23
14	1st Quad-rant Main Steam Tunnel	19.00	280.0	13529.9	90	14.7	30	14	Main Steam	1.4	Double-ended Guillo-tine	11.3	***	***

* Relative humidity.
** Differential pressures calculated for 1st quadrant valve chambers are also applicable to corresponding 2nd quadrant valve chambers.
*** The calculated peak differential pressure in Node 14 has been evaluated to be within plant design basis code allowable stresses (Reference 3).

TABLE 2

UNIT 2 SUBCOMPARTMENT NODAL DESCRIPTION

MAIN STEAM LINE BREAK IN UNIT 2 MAIN STEAM TUNNEL OR VALVE ROOMS

					INITIAL CONDITIONS			DBA BREAK CONDITIONS				CALC.	DESIGN	
VOLUME NO.	DESCRIPTION	HEIGHT ft	CROSS-SECTIONAL AREA ft^2	VOLUME ft^3	TEMP. °F	PRESS. psia	HUMID.* %	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	PEAK PRESS DIFF. psid	PEAK PRESS DIFF. psid	DESIGN MARGIN %
1	Atmosphere	$5x10^3$	$1x10^3$	10^7	90	14.7	30	-	-	-	-	-	-	-
2	2nd Quadrant Upper Valve Chamber	12.33	133.25	4895.7	90	14.7	30	5	Main Steam	1.4	Double-ended Guillotine	17.4	26.2	51
3	2nd Quadrant Upper Valve Chamber	12.33	183.7	4895.7	90	14.7	30	5	Main Steam	1.4	Double-ended Guillotine	17.4	26.2	51
4	2nd Quadrant Lower Valve Chamber	24.00	213.0	6007.0	90	14.7	30	5	Main Steam	1.4	Double-ended Guillotine	17.4	26.2	51
5	2nd Quadrant Valve Chamber	24.00	213.0	6007.0	90	14.7	30	5	Main Steam	1.4	Double-ended Guillotine	19.7	28.6	45

* Relative humidity.

TABLE 2 (Cont'd)

VOLUME NO.	DESCRIPTION	HEIGHT ft	CROSS-SECTIONAL AREA ft^2	VOLUME ft^3	INITIAL CONDITIONS TEMP. °F	PRESS. psia	HUMID.* %	DBA BREAK CONDITIONS BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS DIFF. psid	DESIGN PEAK PRESS DIFF. psid	DESIGN MARGIN %
6	2nd Quad-rant Main Steam Tunnel	19.00	317.0	13695.0	90	14.7	30	6	Main Steam	1.4	Double-ended Guillo-tine	16.4	26.5	61
7	Main Steam Tunnel	19.00	203.00	34865.0	90	14.7	30	7	Main Steam	1.4	Double-ended Guillo-tine	15.5	21.3	38
8	1st Quad-drant Steam Tunnel	20.00	432.0	35016.0	90	14.7	30	8	Main Steam	1.4	Double-ended Guillo-tine	8.8	11.2	28
9	1st Quad-rant Upper Valve Chamber	12.33	133.25	4895.7	90	14.7	30	5**	Main Steam	1.4	Double-ended Guillo-tine	17.4	26.2	51
10	1st Quad-rant Upper Valve Chamber	12.33	183.7	4895.7	90	14.7	30	5**	Main Steam	1.4	Double-ended Guillo-tine	17.4	26.2	51

* Relative humidity.
** Differential pressures calculated for 1st quadrant valve chambers are also applicable to corresponding 2nd quadrant valve chambers.

TABLE 2 (Cont'd)

VOLUME NO.	DESCRIPTION	HEIGHT ft	CROSS-SECTIONAL AREA ft^2	VOLUME ft^3	INITIAL CONDITIONS TEMP. °F	PRESS. psia	HUMID.* %	DBA BREAK CONDITIONS BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS DIFF. psid	DESIGN PEAK PRESS DIFF. psid	DESIGN MARGIN %
11	1st Quadrant Valve Chamber	24.00	213.0	6007.0	90	14.7	30	5**	Main Steam	1.4	Double-ended Guillotine	19.7	28.6	45
12	1st Quadrant Lower Valve Chamber	24.00	213.0	6007.0	90	14.7	30	5**	Main Steam	1.4	Double-ended Guillotine	19.7	28.6	45
13	1st Quadrant Main Steam Tunnel	29.00	432.0	17388.4	90	14.7	30	8	Main Steam	1.4	Double-ended Guillotine	8.9	13.2	48
14	1st Quadrant Main Steam Tunnel	19.00	280.0	13529.9	90	14.7	30	8	Main Steam	1.4	Double-ended Guillotine	5.9	10.3	75

* Relative humidity.
** Differential pressures calculated for 1st quadrant valve chambers are also applicable to corresponding 2nd quadrant valve chambers.

TABLE 3

SUBCOMPARTMENT VENT PATH DESCRIPTION

MAIN STEAM LINE BREAK IN MAIN STEAM TUNNEL OR VALVE ROOM

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW* CHOKED‡ UNCHOKED	AREA† ft^2	LENGTH† ft	HYDRAULIC DIAMETER ft	HEAD LOSS, K FRICTION K, ft/d	TURNING LOSS, K	EXPAN-SION, K	CONTRAC-TION, K	TOTAL
1	14	1	Main Steam Tunnel to Turbine Building Unchoked	270.8	28.0	13.9	-	-	-	-	1.0656
2	2	5	2nd Quadrant Upper Valve Chamber to Lower Valve Chamber Unchoked	121.0	16.4	5.7	-	-	-	-	1.5207
3	3	4	2nd Quadrant Upper Valve Chamber to Lower Valve Chamber Unchoked	121.0	16.4	5.7	-	-	-	-	1.5207
4	6	4	2nd Quadrant Lower Valve Chamber to Main Steam Tunnel Unchoked	73.0	17.2	4.5	-	-	-	-	1.5685
5	6	5	2nd Quadrant Lower Valve Chamber to Main Steam Tunnel Choked (5)	73.0	17.2	4.5	-	-	-	-	1.5685

* See Figures 1 and 2.
† Length/area is the inertial term input directly into RELAP4/MOD5.
‡ Number in parentheses indicates the volume number of the break location which caused choke flow in the vent. For break locations not indicated, unchoked flow had occurred for the vent.

TABLE 3 (Cont'd)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW* CHOKED‡ UNCHOKED	AREA† ft^2	LENGTH† ft	HYDRAULIC DIAMETER ft	HEAD LOSS, K FRICTION K, ft/d	TURNING LOSS, K	EXPAN-SION, K	CONTRAC-TION, K	TOTAL
6	5	4	Openings Between 2nd Quadrant Lower Valve Chambers Unchoked	100.0	16.1	7.1	-	-	-	-	1.5071
7	6	7	2nd Quadrant Main Steam Tunnel to Main Steam Tunnel Unchoked	199.8	102.0	13.3	-	-	-	-	2.1860
8	7	8	Main Steam Tunnel to 1st Quadrant Main Steam Tunnel Unchoked	199.8	132.4	13.3	-	-	-	-	2.7530
9	12	9	1st Quadrant Upper Valve Chamber to Lower Valve Chamber Unchoked	121.0	16.4	5.7	-	-	-	-	1.5207
10	9	10	Openings Between 1st Quadrant Upper Valve Chambers Unchoked	100.0	11.2	4.5	-	-	-	-	1.5120
11	11	10	1st Quadrant Upper Valve Chamber to Lower Valve Chamber Unchoked	121.0	16.4	5.7	-	-	-	-	1.5207

* See Figures 1 and 2.
† Length/area is the inertial term input directly into RELAP4/MOD5.
‡ Number in parentheses indicates the volume number of the break location which caused choke flow in the vent. For break locations not indicated, unchoked flow had occurred for the vent.

TABLE 3 (Cont'd)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW* CHOKED‡ UNCHOKED	AREA† ft^2	LENGTH† ft	HYDRAULIC DIAMETER ft	HEAD LOSS, K: FRICTION K, ft/d	TURNING LOSS, K	EXPANSION, K	CONTRACTION, K	TOTAL
12	12	11	Openings Between 1st Quadrant Lower Valve Chambers / Unchoked	100.0	16.1	7.1	-	-	-	-	1.5071
13	8	13	1st Quadrant Main Steam Tunnel / Unchoked	373.6	42.0	17.6	-	-	-	-	2.2600
14	13	14	1st Quadrant Main Steam Tunnel / Unchoked	270.8	42.0	13.9	-	-	-	-	2.2600
15	8	11	1st Quadrant Lower Valve Chamber to Main Steam Tunnel / Unchoked	73.0	17.2	4.5	-	-	-	-	1.5685
16	8	12	1st Quadrant Lower Valve Chamber to Main Steam Tunnel / Unchoked	73.0	17.2	4.5	-	-	-	-	1.5685
17	2	3	Openings Between 2nd Quadrant Upper Valve Chambers / Unchoked	100.0	11.2	4.5	-	-	-	-	1.5120

* See Figures 1 and 2.
† Length/area is the inertial term input directly into RELAP4/MOD5.
‡ Number in parentheses indicates the volume number of the break location which caused choke flow in the vent. For break locations not indicated, unchoked flow had occurred for the vent.

TABLE 3 (Cont'd)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW* CHOKED‡ UNCHOKED	AREA† ft^2	LENGTH† ft	HYDRAULIC DIAMETER ft	HEAD LOSS, K FRICTION K, ft/d	TURNING LOSS, K	EXPANSION, K	CONTRACTION, K	TOTAL
18	2	1	HVAC Panels in 2nd Quadrant Upper Valve Chambers Choked (5, 6)	51.3	15.2	5.9	-	-	-	-	2.900
19	3	1	Door and HVAC Panels in 2nd Quadrant Upper Valve Chambers Choked (5, 6)	75.8	25.3	5.4	-	-	-	-	2.900
20	9	1	HVAC Panels in 1st Quadrant Upper Valve Chamber Choked (5)§	51.3	15.2	5.9	-	-	-	-	2.900
21	10	1	Door and HVAC Panels in 1st Quadrant Upper Valve Chamber Choked (5)[4]	75.8	25.3	5.4	-	-	-	-	2.900
22(a) **	0	5	Main Steam Line Break in Node 5 Fill	1.0	0.0	0.0	-	-	-	-	0.000
22(b)[5]	0	6	Main Steam Line Break in Node 6 Fill	1.0	0.0	0.0	-	-	-	-	0.000

* See Figures 1 and 2.
† Length/area is the inertial term input directly into RELAP4/MOD5.
‡ Number in parentheses indicates the volume number of the break location which caused choke flow in the vent. For break locations not indicated, unchoked flow had occurred for the vent.
§ Choking results for 2nd quadrant valve room are applied to 1st quadrant valve room.
** Four cases were considered each having a different break location.

TABLE 3 (Cont'd)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW* CHOKED‡ UNCHOKED	AREA† ft²	LENGTH† ft	HYDRAULIC DIAMETER ft	HEAD LOSS, K FRICTION K, ft/d	TURNING LOSS, K	EXPAN-SION, K	CONTRAC-TION, K	TOTAL
22(c) **	0	7	Main Steam Line Break in Node 7 / Fill	1.0	0.0	0.0	-	-	-	-	0.000
22(d)§	0	8	Main Steam Line Break in Node 8 / Fill	1.0	0.0	0.0	-	-	-	-	0.000

* See Figures 1 and 2.
† Length/area is the inertial term input directly into RELAP4/MOD5.
‡ Number in parentheses indicates the volume number of the break location which caused choke flow in the vent. For break locations not indicated, unchoked flow had occurred for the vent.
§ Choking results for 2nd quadrant valve room are applied to 1st quadrant valve room.
** Four cases were considered each having a different break location.

TABLE 4

UNIT 1 BLOWDOWN RATES AND ENTHALPY FOR MAIN STEAMLINE BREAK

TIME (sec)	FLOW (lb/sec)	ENTHALPY (Btu/lb)
0.0	14,189	1,024.9
0.02	14,189	1,024.9
0.04	13,883	1,116.4
0.06	12,901	1,119.3
0.08	12,479	1,124.6
0.10	12,342	1,133.0
0.12	12,250	1,134.0
0.14	12,093	1,128.3
0.16	11,827	1,119.4
0.18	11,434	1,111.2
0.20	11,022	1,107.3
0.22	10,642	1,105.7
0.24	10,315	1,105.5
0.26	10,041	1,105.9
0.28	9,810	1,106.3
0.30	9,608	1,106.4
0.32	9,424	1,106.4
0.34	9,255	1,106.3
0.36	9,097	1,106.4
0.38	8,954	1,106.8
0.40	8,827	1,107.7
0.42	8,728	1,110.1
0.44	8,633	1,109.1

TABLE 4 (Cont'd)

TIME (sec)	FLOW (lb/sec)	ENTHALPY (Btu/lb)
0.46	8,558	1,112.7
0.48	8,522	1,116.4
0.50	8,512	1,119.5
0.52	8,523	1,122.7
0.54	8,553	1,125.9
0.56	8,598	1,128.8
0.58	8,652	1,131.3
0.60	8,709	1,133.3
0.62	8,765	1,134.8
0.64	8,813	1,135.8
0.66	8,852	1,136.5
0.68	8,879	1,136.9
0.70	8,894	1,137.2
0.72	8,898	1,137.3
0.74	8,892	1,137.4
0.76	8,878	1,137.5
0.78	8,875	1,138.5
0.80	8,846	1,137.5
0.82	8,816	1,137.9
0.84	8,788	1,138.3
0.86	8,761	1,138.5
0.88	8,733	1,138.7
0.90	8,705	1.138.9
0.92	8,679	1,139.1

TABLE 4 (Cont'd)

TIME (sec)	FLOW (lb/sec)	ENTHALPY (Btu/lb)
0.94	8,655	1,139.3
0.96	8,632	1,139.5
0.98	8,611	1,139.6
1.00	8,591	1,139.8
1.02	8,573	1,139.9
1.04	8,557	1,140.1
1.06	8,542	1,140.2
1.08	8,529	1,140.3
1.10	8,518	1,140.4
1.12	8,508	1,140.5
1.14	8,499	1,140.5
1.16	8,492	1,140.5
1.18	8,486	1,140.4
1.20	8,481	1,140.3
1.22	8,477	1,140.2
1.24	8,473	1,140.0
1.26	8,471	1,139.7
1.28	8,468	1,139.4
1.30	8,466	1,139.0
1.32	8,465	1,138.6
1.34	8,463	1,138.1
1.36	8,462	1,137.5
1.38	8,462	1,136.9
1.40	8,461	1.136.2

TABLE 5

UNIT 2 BLOWDOWN RATES AND ENTHALPY FOR MAIN STEAMLINE BREAK

TIME (sec)	FLOW (lb/sec)	ENTHALPY (Btu/lb)
0.0	11,000	1,195.4
2.0	10,434	1,195.1
4.0	9,608	1,196.9
6.0	9,017	1,197.7
8.0	8,613	1,199.4
10.0	9,318	1,199.8
10.1	2,098	1,201.1
20.0	1,993	1,199.2
30.0	1,879	1,208.1
50.0	1,625	1,206.1
75.0	1,064	1,203.0
100.0	814	1,201.5

TABLE 6

SUMMARY OF UNIT 1 PEAK PRESSURES
BETWEEN VALVE ROOM AND MAIN STEAM TUNNEL

WALL LOCATION BETWEEN NODES		DELTA P ACROSS	PEAK PRESSURE (psid)
2-3	9-10	Vertical Wall	5.48
3-4	10-11	Horizontal Floor	12.50
4-5	11-12	Vertical Wall	13.20
2-5	9-12	Horizontal Floor	12.50
4-6 11-8	5-6 12-8	Main Steam Tunnel Vertical Wall	14.40

ATTACHMENT D3.6

FLOODING

ATTACHMENT D3.6 - FLOODING

D3.6.1 Flooding Inside Containment

The containment is provided with a system of floor drains and sumps. The sumps are equipped with level alarms to monitor excess leakage. Flooding is not postulated except in LOCA events when the leakage is large and the drainage system will be closed by the containment isolation system.

D3.6.1.1 Assumptions for Flooding Inside Containment

The limiting break for calculating containment flood level is a double-ended pump suction (DEPS) break. Potential sources of flood water are the RCS water, refueling water storage tank (RWST), containment spray additive tank, and SI accumulators. The maximum flood level inside containment occurs just after all the available water mass (RCS, RWST, SI accumulators, and CS additive tank) has been added into containment. The most limiting single failure is the inability to isolate the RWST during switchover from the injection phase to the recirculation mode (failure to close SI8812A, SI8812B, CS001A, or CS001B). This allows the entire volume of the RWST to gravity feed to the containment sump. This scenario is more limiting than the loss of a single containment spray pump.

Some equipment and structures are assumed to displace flood water. Areas of limited accessibility are assumed to remain dry. These areas include the reactor cavity and annulus. A portion of the flood water is assumed to be trapped in the refueling cavity and various sumps, drains, and trenches. During the break, containment temperature and pressure responses are lowest. This minimizes water trapped in the containment atmosphere and maximizes the flood water.

D3.6.1.2 Results and Conclusions

The maximum flood level in the containment is conservatively predicted to be less than the evaluated flood level of 6 feet 3 inches above the floor and is documented by the reference in Section D3.6.3. All equipment which is flood sensitive and required after a LOCA is located above this evaluated level.

D3.6.2 Auxiliary Building Flooding

Flooding in the auxiliary building is much more complex than the containment because of the numerous structural divisions in the auxiliary building and because of the diversity of flooding sources and scenarios. An extensive analysis of the potential flooding sources and the result of the subsequent water levels has been completed. This study established that flooding as a result of pipe failure will not prevent safe shutdown of the plant.

D3.6.2.1 Method of Auxiliary Building Flooding Analysis

The auxiliary building was divided into 217 areas for Byron and 219 areas for Braidwood covering both the Unit 1 and Unit 2 sides of the plant. A limiting break was defined in each area using the guidelines in Section 3.6. Flow from isolable breaks was assumed to continue for 30 minutes before isolation.

Fluid is removed from areas either by floor drains or by flow through doors and openings. Doors were assumed to remain closed to calculate the maximum flood in the area containing the break and assumed open to check on potential flooding of adjacent areas. Flooding on upper levels drains into the lower level of the plant by way of stairways and hatches.

A complete review of the non-Seismic Category I piping system failure inside the auxiliary building was done to demonstrate that safety-related equipment will be protected from flooding. There are 20-inch nonessential service water supply and return lines in the auxiliary building. The routing of the lines has been evaluated from the standpoint of potential for flooding or impacting safety-related equipment. In those areas where such potential exists, the supports for these lines have been seismically qualified. The same is true of primary water lines that draw reactor grade makeup water from outdoor storage tanks and condensate lines that provide Category II sources of makeup water to the auxiliary feedwater pumps.

D3.6.2.2 Results

Most of the areas defined in the upper levels of the plant have predicted maximum flood levels of 4 inches or less. Design of the plant eliminates any damage to equipment with this flood level. Predicted flood levels are higher in some areas, especially in the lower elevations. Due to the conservative assumption that the doors remain closed, some subcompartments are unrealistically predicted to fill with water. Even with these conservative results, the ability to safely shut down the plant will be unimpaired. In the lower areas, equipment is elevated or enclosed in waterproof areas as required.

D3.6.2.3 Conclusions

Flooding as a result of high and moderate energy line failure will not adversely affect the safe shutdown capability of the plant.

D3.6.3 References

Sargent & Lundy, "Containment Flood Level," Calculation ATD-0111.

TABLE 3.6-1

SYSTEMS IMPORTANT TO PLANT SAFETY

Auxiliary Feedwater
Residual Heat Removal
Component Cooling
Essential Service Water
Essential Service Water Makeup (Byron only)
Chemical and Volume Control

Note: For given postulated events, further additional systems may be required (e.g., safety injection required for a LOCA). See Subsection 3.6.1.3.

TABLE 3.6-2

SYSTEMS WHICH CONTAIN HIGH OR MODERATE ENERGY FLUID

DURING NORMAL OPERATION

HIGH ENERGY	SYSTEM ACRONYM	MODERATE ENERGY
Auxiliary Steam	AS	Boric Acid and Boron Recycle
Chemical and Volume Control	CV	Boron Thermal Regeneration
Main Feedwater	FW	Chemical and Volume Control
Main Steam	MS	Chemical Feed
Radioactive Waste Processing	WX	Component Cooling
Reactor Coolant	RC	Containment Air Monitoring/Sampling
Steam Generator Blowdown	SD	Diesel Fuel Oil
		Essential Service Water
Pressurizer	RY	Fire Protection
Safety Injection Accumulators	SI	H_2, N_2, and CO_2 Systems
		Instrument and Service Air
		Nonessential Service Water
		Radioactive Waste Processing
		Residual Heat Removal
		Spent Fuel Pit Cooling
		Station Heating
		Steam Generator Blowdown
		Waste Gas

Note: 1. Systems shown are either totally or partially high/moderate energy during normal operation. High/moderate energy lines can be identified through the engineering controlled equipment/component database(s).

2. The above listing reflects criteria with regard to systems operated infrequently, such as the Residual Heat Removal System.

3. The auxiliary feedwater system is not utilized for normal startup and shutdown of the unit. It is classified as a moderate-energy system.

TABLE 3.6-3

ESSENTIAL SYSTEMS

VERSUS

TYPE OF POSTULATED PIPING BREAK

TYPE	POSTULATED PIPING BREAK	ESSENTIAL SYSTEMS
I	Large Reactor Coolant Break (Piping Larger than 4 inches)	Safety Injection Residual Heat Removal Chemical and Volume Control Containment Spray Auxiliary Feedwater Component Cooling Essential Service Water Essential Service Water Makeup (Byron only)
II	Small Reactor Coolant Break (Piping Equal to or Smaller than 4 inches)	Safety Injection Residual Heat Removal Chemical and Volume Control Containment Spray Auxiliary Feedwater Component Cooling Essential Service Water Essential Service Water Makeup (Byron only)
III	Critical Secondary Side Break (Main steam, feedwater, or steam generator blowdown inside containment)	Safety Injection Residual Heat Removal Chemical and Volume Control Containment Spray Auxiliary Feedwater Component Cooling Essential Service Water Essential Service Water Makeup (Byron only)

TABLE 3.6-3 (Cont'd)

TYPE	POSTULATED PIPING BREAK	ESSENTIAL SYSTEMS
IV	Noncritical Secondary Side Break (Main steam, feedwater, or steam generator blowdown outside isolation valve room)	Residual Heat Removal Chemical and Volume Control Auxiliary Feedwater Component Cooling Essential Service Water Essential Service Water Makeup (Byron only)
V	Breaks other than Types I through IV	Residual Heat Removal Chemical and Volume Control Auxiliary Feedwater Component Cooling Essential Service Water Essential Service Water Makeup (Byron only)

TABLE 3.6-4

BREAK PROPAGATION LIMITS

VERSUS

TYPE OF POSTULATED PIPING FAILURE

(In addition to not propagating to essential systems listed in Table 3.6-3, postulated piping breaks are further limited as follows:)

TYPE	POSTULATED PIPING BREAK	CANNOT PROPAGATE TO
I	Large Reactor Coolant Break (Piping larger than 4 inches)	a) Secondary Side b) Unaffected loops c) Over 20% more than the original break area within the affected loop
II	Small Reactor Coolant Break (Piping equal to or smaller than 4 inches)	a) Secondary Side b) Unaffected loops c) Unaffected legs within the affected loop d) Over 12.5 in^2 in total area within the affected leg*

*The only exception to this rule is that any high head injection line break cannot propagate to any other high head injection line, even in the same leg.

TABLE 3.6-4 (Cont'd)

TYPE	POSTULATED PIPING BREAK	CANNOT PROPAGATE TO
III	Critical Secondary Side Break (Main Steam, feedwater, or steam generator blowdown inside the containment)	a) Reactor Coolant b) Secondary side lines from unaffected steam generators, the unaffected steam generators themselves, and their drain piping
IV	Noncritical Secondary Side Break (main steam, feedwater, or steam generator blowdown outside isolation valve room)	a) Reactor Coolant b) Any other piping in the main steam tunnel**
V	Breaks other than Types I through IV	a) Reactor Coolant b) Secondary Side

**This requirement is a limiting condition from the pressurization design of the main steam tunnel.

TABLE 3.6-5

Deleted

Pages 3.6-53 through 3.6-56 have been deleted intentionally.

TABLE 3.6-8

ENERGY BALANCE VS FINITE DIFFERENCE ANALYSIS*
(30-inch pipe)

RESTRAINT DISPLACEMENT (in.)

L_1 (in.)	L_2 (in.)	ENERGY BALANCE	FINITE DIFFERENCE
99.54	66.34	3.36	3.01
149.30	66.34	5.17	4.93
431.21	66.34	6.41	6.76
87.06	58.04	3.14	2.15
145.01	58.04	4.43	4.59
435.30	58.04	5.91	5.85
99.51	49.75	2.76	2.39
447.77	49.75	5.15	5.31

Pipe OD = 30 in. F = 607.2 kips
Pipe wall = 1.1619 in. R = 1062.6 kips
Pipe weight = 34.7 lb/in. GAP = 4 in.

Plastic moment used in Energy Balance analysis = 50349 in kips.

Material properties used in finite difference analysis.

Modulus of elasticity = 26100 ksi

Power law constants
K = 90.277 ksi n = 0.1648 in $\sigma = K(\varepsilon)^n$.

*A106 Grade B carbon steel pipe at 550°F.

TABLE 3.6-9

ENERGY BALANCE VS FINITE DIFFERENCE ANALYSIS*
(12-inch pipe)

RESTRAINT DISPLACEMENT (in.)

L_1 (in.)	L_2 (in.)	ENERGY BALANCE	FINITE DIFFERENCE
81.51	54.4	4.10	3.46
122.28	54.4	5.39	5.58
380.53	54.4	7.93	6.75
82.47	47.1	3.28	3.10
223.82	47.1	6.55	6.79
388.74	47.1	7.06	6.20

Pipe OD = 12.75 in. F = 95.3 kips
Pipe wall = 0.95607 R = 166.78 kips
Pipe weight = 14.08 lb/in. GAP = 4 in.

Plastic moment used in Energy Balance analysis = 6908 in. kips.

Material properties used in finite difference analysis.

Modulus of elasticity = 26100 ksi
Power law constants:
K = 90.277 ksi n = 0.1648 n $\sigma = K(\varepsilon)^n$.

* A106 Grade B carbon steel pipe at 550°F.

Table 3.6-10 has been deleted intentionally.

TABLE 3.6-11

CALCULATED STRESS AND CUMULATIVE USAGE FACTORS FOR

POSTULATED BREAK POINTS

(For ASME Sec. III Class 1 Piping Systems)

PIPING SYSTEM LINE NUMBER(S)	BREAK ID	CALCULATED STRESS NORMAL & UPSET PLANT CONDITIONS (psi)	2.4(S_m) (psi)	CUMULATIVE USAGE FACTOR
RCS BYPASS Loop 1				
	See stress analysis report for Subsystem RC-01			
RCS BYPASS Loop 2				
	See stress analysis report for Subsystem RC-02			
RCS BYPASS Loop 3				
	See stress analysis report for Subsystem RC-03			
RCS BYPASS Loop 4				
	See stress analysis report for Subsystem RC-04			

TABLE 3.6-11 (Cont'd)

PIPING SYSTEM		CALCULATED STRESS		
LINE NUMBER(S)	BREAK ID	NORMAL & UPSET PLANT CONDITIONS (psi)	2.4(S_m) (psi)	CUMULATIVE USAGE FACTOR
RCS SURGE LINE				
	See stress analysis report for Subsystem RY-05			
SAFETY INJECTION Loop 1				
	See stress analysis report for Subsystem RH-02			
SAFETY INJECTION Loop 2				
	See stress analysis report for Subsystem SI-11			
SAFETY INJECTION Loop 3				
	See stress analysis report for Subsystem RH-02			
SAFETY INJECTION Loop 4				
	See stress analysis report for Subsystem SI-10			
RESIDUAL HEAT REMOVAL				
	See stress analysis report for Subsystem RH-02			

TABLE 3.6-12

CALCULATED STRESSES FOR POSTULATED BREAK POINTS

(For ASME Sec. III Class 2&3 and ANSI B31.1 Piping Systems)

PIPING SYSTEM LINE NUMBER(S)	BREAK ID	CALCULATED STRESS NORMAL & UPSET PLANT CONDITIONS (psi)	$0.8(1.2S_b+S_A)$ (psi)
FEEDWATER Loop 1			
	See stress analysis report for Subsystem FW-02		
FEEDWATER Loop 2			
	See stress analysis report for Subsystem FW-02		
FEEDWATER Loop 3			
	See stress analysis report for Subsystem FW-04		
FEEDWATER Loop 4			
	See stress analysis report for Subsystem FW-05		

TABLE 3.6-12 (Cont'd)

PIPING SYSTEM LINE NUMBER(S)	BREAK ID	CALCULATED STRESS NORMAL & UPSET PLANT CONDITIONS (psi)	$0.8(1.2S_b+S_A)$ (psi)
FEEDWATER IN M.S. TUNNEL			
	See stress analysis report for Subsystem FW-01		
MAIN STEAM Loop 1			
	See stress analysis report for Subsystem MS-05		
MAIN STEAM Loop 2			
	See stress analysis report for Subsystem MS-06		
MAIN STEAM Loop 3			
	See stress analysis report for Subsystem MS-07		
MAIN STEAM Loop 4			
	See stress analysis report for Subsystem MS-08		

TABLE 3.6-12 (Cont'd)

PIPING SYSTEM LINE NUMBER(S)	BREAK ID	CALCULATED STRESS NORMAL & UPSET PLANT CONDITIONS (psi)	$0.8(1.2S_b+S_A)$ (psi)
MAIN STEAM IN M.S. TUNNEL			
	See stress analysis report for Subsystem MS-01		
SAFETY INJECTION Loop 1			
	See stress analysis report for Subsystem RH-02		
SAFETY INJECTION Loop 2			
	See stress analysis report for Subsystem SI-13		
SAFETY INJECTION Loop 3			
	See stress analysis report for Subsystem RH-02		
SAFETY INJECTION Loop 4			
	See stress analysis report for Subsystem SI-14		

TABLE 3.6-13

RESULTS OF DYNAMIC ANALYSES FOR POSTULATED PIPE BREAKS - RESTRAINED PIPE - INSIDE CONTAINMENT

PIPING SYSTEM							RESTRAINT INFORMATION			
POSTULATED BREAK ID	RESTRAINT ID	F_{imp} (kips)	T_{imp} (sec.)	F_{FINAL} (kips)	GAP (inches)	TIP DISPLACEMENT (inches)	DEFLECTION (inches)	PEAK DYNAMIC LOAD (kips)	STRAIN (in/in)	ALLOWABLE STRAIN (in/in)
FEEDWATER*										
C55	FWR-1	176	.001	75	1.463	6.33	.179	631	.026	.5
C50	FWR-9	191	.008	142	2.642	5.85	1.314	390	.07	.08
C61	FWR-10	176	.001	75	1.526	4.88	.1964	513	.078	.5
C56	FWR-19	191	.008	142	2.00	3.84	1.03	414	.07	.08

* See Figures 3.6-47 and 3.6-48.

TABLE 3.6-13 (Cont'd)

PIPING SYSTEM							RESTRAINT INFORMATION			
POSTULATED BREAK ID	RESTRAINT ID	F_{imp} (kips)	T_{imp} (sec.)	F_{FINAL} (kips)	GAP (inches)	TIP DISPLACEMENT (inches)	DEFLECTION (inches)	PEAK DYNAMIC LOAD (kips)	STRAIN (in/in)	ALLOWABLE STRAIN (in/in)
MAIN STEAM**										
C1	MSP-1	660	.003	460	6.38	14.60	4.4	1121	.054	.08
C9 (Unit 1)	MSP-8	660	.003	460	2.18	14.52	8.413	842	.062	.08
C9 (Unit 2)	MSP-8	660	.003	460	6.573	14.99	4.486	1122	.035	.08
C17	MSP-16	775	.003	539	2.57	8.13	3.45	1076	.05	.08
C25	MSP-23	775	.003	539	3.95	10.92	4.34	1071	.075	.08

** See Figure 3.6-56 through 3.6-59

TABLE 3.6-13 (Cont'd)

PIPING SYSTEM							RESTRAINT INFORMATION			
POSTULATED BREAK ID	RESTRAINT ID	F_{imp} (kips)	T_{imp} (sec.)	F_{FINAL} (kips)	GAP (inches)	TIP DISPLACEMENT (inches)	DEFLECTION (inches)	PEAK DYNAMIC LOAD (kips)	STRAIN (in/in)	ALLOWABLE STRAIN (in/in)
SAFETY INJECTION***										
C500	SI1R-10B	100	.0004	35	1.578	2.86	0.896	98	.077	.08
C501	SI1R-30	102	.0013	7	3.76	9.15	.16	145	.0074	.08
C518	SI3R-640A	107	.0004	58	1.814	3.96	2.033	121	.467	.5
C519	SI3R-655	102	.0015	7	1.10	6.20	2.1	95	.24	.5
C507	SI4R-15B	108	.0010	8	0.688	2.81	1.86	123	.485	.5
C507	SI4R-35	102	.0013	7	3.812	11.62	0.294	124	.033	.08
C513	SI9R-475B	108	.0010	9	0.53	6.90	6.18	74	.049	.08
C513	SI9R-495	102	.0015	9	3.168	8.45	1.299	125	.004	.08

*** See Figures 3.6-88 through 3.6-91.

TABLE 3.6-13 (Cont'd)

PIPING SYSTEM							RESTRAINT INFORMATION			
POSTULATED BREAK ID	RESTRAINT ID	F_{imp} (kips)	T_{imp} (sec.)	F_{FINAL} (kips)	GAP (inches)	TIP DISPLACEMENT (inches)	DEFLECTION (inches)	PEAK DYNAMIC LOAD (kips)	STRAIN (in/in)	ALLOWABLE STRAIN (in/in)
RCS SURGE LINE*										
C400/C403	RY-2	-	-	350	1.53	11.60	3.70	469	0.370	0.50
C400/C403	RY-3	-	-	350	1.44	5.73	0.36	527	0.049	0.50
C400/C403	RY-4	-	-	350	1.25	6.00	0.46	570	0.058	0.50
C400/C403	RY-5	-	-	350	2.25	6.00	0.21	399	0.026	0.50
C400/C403	RY-6	-	-	350	1.13	8.05	0.90	368	0.276	0.50
C400/C403	RY-7	-	-	350	1.06	5.62	0.15	458	0.039	0.50
C400/C403	RY-8	-	-	350	1.25	12.95	2.51	527	0.228	0.50

* See Figure 3.6-77

TABLE 3.6-13 (Cont'd)

PIPING SYSTEM							RESTRAINT INFORMATION			
POSTULATED BREAK ID	RESTRAINT ID	F_{imp} (kips)	T_{imp} (sec.)	F_{FINAL} (kips)	GAP (inches)	TIP DISPLACEMENT (inches)	DEFLECTION (inches)	PEAK DYNAMIC LOAD (kips)	STRAIN (in/in)	ALLOWABLE STRAIN (in/in)
RCS BYPASS*										
C144	RC1-1	87	.002	42	0.5	1.33	0.227	140	.0016	.08
C149	RC1-4	73	.010	15	3.48	17.57	2.01	74	.045	.08
C150	RC2-1	87	.002	42	0.5	1.19	.08	180	.0021	.08
C155	RC2-4	73	.010	15	3.712	17.66	1.254	78	.03	.08
C156	RC3-1	87	.002	42	0.5	1.24	.081	176	.003	.08
C161	RC3-4	73	.010	15	3.86	18.50	2.91	78	.003	.08
C162	RC4-1	87	.002	42	0.5	1.17	.079	180	.002	.08
C167	RC4-4	73	.010	15	3.75	15.87	2.85	76	.004	.08

* See Figures 3.6-64 through 3.6-67.

TABLE 3.6-13 (Cont'd)

PIPING SYSTEM							RESTRAINT INFORMATION			
POSTULATED BREAK ID	RESTRAINT ID	F_{imp} (kips)	T_{imp} (sec.)	F_{FINAL} (kips)	GAP (inches)	TIP DISPLACEMENT (inches)	DEFLECTION (inches)	PEAK DYNAMIC LOAD (kips)	STRAIN (in/in)	ALLOWABLE STRAIN (in/in)
RESIDUAL HEAT REMOVAL*										

No Longer Applicable

NOTES:

(1) Restraints may be loaded by more than one break. Only the break that results in the most severe restraint loading is shown.

(2) When $F_{FINAL} \geq F_{imp}$, F_{FINAL} is used for the entire loading duration.

(3) For tension member where A193B7 bar is used, the allowable strain = 0.08 in/in (see Subsection 3.6.2.3.2).

(4) For compression members where a honeycomb material is used, the allowable strain = 0.5 in/in (see Subsection 3.6.2.3.2).

TABLE 3.6-14

RESULTS OF DYNAMIC ANALYSES FOR POSTULATED PIPE BREAKS - RESTRAINED PIPE - OUTSIDE CONTAINMENT

PIPING SYSTEM							RESTRAINT INFORMATION			
POSTULATED BREAK ID	RESTRAINT ID	F_{imp} (kips)	T_{imp} (sec.)	F_{FINAL} (kips)	GAP (inches)	TIP DISPLACEMENT (inches)	DEFLECTION (inches)	PEAK DYNAMIC LOAD (kips)	STRAIN (in/in)	ALLOWABLE STRAIN (in/in)
MAIN STEAM IN TUNNEL (1MSOCC-32-3/4)										

The results of the HELB analysis of the as-built condition of piping outside containment have indicated that pipe whip restraints outside containment are not required at Byron/Braidwood.

3.7 SEISMIC DESIGN

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra

The site response spectra, which are defined at the ground surface, are given in Subsection 2.5.2 and are shown in Figures 2.5-40 and 2.5-41 for the Byron site and in Figures 2.5-47 and 2.5-48 for the Braidwood site. Foundation level response spectra and time histories were generated by a deconvolution procedure described in Subsection 3.7.1.2. The maximum horizontal and vertical ground accelerations at the foundation level are 20% of gravity for the safe shutdown earthquake (SSE) and 9% of gravity for operating basis earthquake (OBE). The comparisons between the free field seismic design motion applied at the surface and the corresponding foundation (rock) spectra for 2%, 3%, 4%, 5%, and 7% damping ratios are shown in Figures 3.7-1 through 3.7-20 for the Byron site and in Figures 3.7-21 through 3.7-40 for the Braidwood site.

During the review of the FSAR for an Operating License, the Byron/Braidwood seismic design was reevaluated using the Regulatory Guide 1.60 spectra without the application of a deconvolution analysis. Attachment 3.7A contains the specific NRC questions/responses on seismic design. These questions and responses document the historical evolution of certain aspects of the Byron/Braidwood seismic design. Attachment 3.7A also provides the details and results of this reevaluation. It is concluded that the present seismic design of Byron/Braidwood is conservative. Based on the reevaluation described in Attachment 3.7A, the Byron/Braidwood seismic design basis is acceptable and will therefore be used for all future seismic evaluations.

3.7.1.2 Design Time-History

The following two step procedure is used for generating the foundation (rock) level response spectra and time histories.

Step 1 - Generation of Spectrum Consistent Time-History

The north-south and vertical components of the 1940 El Centro Earthquake records were modified using the RSG program (see Appendix D for a description of RSG) such that the response spectra generated using these synthetic records match closely the site response spectra for the horizontal and vertical directions.

One hundred and seventeen periods lying between 0.02 to 2.0 seconds were considered in generating the response spectra from the modified synthetic records. These periods are spaced as shown below:

Period Range (seconds)	Number of Equally Spaced Periods Considered
0.02 - 0.1	17
0.1 - 0.5	80
0.5 - 2.0	20

The comparison of response spectra obtained from horizontal and vertical synthetic time histories and the corresponding design spectra for 0.2g ground acceleration is presented in Figures 3.7-41 through 3.7-50 for 2, 3, 4, 5, and 7% damping ratios. The synthetic time histories were then scaled to the required ground surface response spectra acceleration levels.

Step-2 - Generation of Foundation Motion

The soil-rock profile above the foundation was modeled as a one-dimensional continuous shear layer system. For Byron Station an average 16 feet of soil overburden and 22 feet of rock were included in the above model, while for Braidwood a total of 38 feet of soil overburden above foundation was modeled. The compatible time-history obtained in Step 1 was scaled to maximum surface ground acceleration levels given in Subsection 2.5.2. The scaled time-histories were applied at the ground surface and the foundation (rock) motion was obtained using the SHAKE program. The description of the SHAKE program is given in Appendix D.

The strain-dependent dynamic soil properties used in the analysis are shown in Figures 2.5-83 and 2.5-84 for the Byron site and in Figures 2.5-77 through 2.5-80 for the Braidwood site.

3.7.1.3 Critical Damping Values

The damping values (expressed as a percentage of critical) used in the analysis of various Category I structures, systems, and components are listed in Table 3.7-1.

3.7.1.4 Supporting Media for Seismic Category I Structures

The description of the supporting media for each Category I structure is given in Table 3.7-3. The table includes the depth of soil over bedrock, foundation embedment depth, size of the structural foundation, and total structural height for each Category I structure. The soil properties and soil layering characteristics are given in Section 2.5.

3.7.2 Seismic System Analysis

3.7.2.1 Seismic Analysis Methods

The seismic analysis was performed using modal superposition method. The displacements and accelerations of mass points were determined by the response spectrum method while the response spectra at various floor elevations for subsystem analysis were generated by the time-history method.

Dynamic modeling of the building structures is described in Subsection 3.7.2.3. The computer program DYNAS (Dynamic Analysis of Structures) was used to analyze the Seismic Category I building structures. The description of this program is presented in Appendix D.

Figures 3.7-51 through 3.7-53 show typical sketches of the horizontal seismic model for Byron/Braidwood Stations. Rigid slabs at various floor elevations are connected by shear wall springs. The containment in Figure 3.7-53 is modeled as a lumped mass-spring model. The horizontal models were analyzed for X (N-S direction) and Y (E-W direction) excitations and the results were combined as described in Subsection 3.7.2.6.

As in the horizontal analysis, both response spectrum and time history methods of analysis were performed on the vertical model also using DYNAS.

Because of different subsurface conditions at the two plant sites, two separate seismic analyses were performed and the enveloped results were used in the design of both the plants.

3.7.2.2 Natural Frequencies and Response Loads

Structural frequencies and participation factors obtained from the seismic analysis of the auxiliary-fuel handling building complex are tabulated in Table 3.7-4 for the Byron and Braidwood Stations. Those for the containment structure are tabulated in Table 3.7-5.

Seismic response loads for safe shutdown earthquake for major Category I shear walls and the containment wall are provided in Tables 3.7-7 through 3.7-10 for the Braidwood Station. The seismic responses for similar shear walls at the Byron Station are provided in Tables 3.7-11 through 3.7-13.

The response spectra at major plant equipment elevations for Category I structures are shown in Figures 3.7-61 through 3.7-76.

3.7.2.3 Procedures Used for Modeling

3.7.2.3.1 Designation of Systems Versus Subsystems

The calculation of the dynamic response of a nuclear power plant subject to an earthquake loading can be divided into two categories. The first is the seismic main structural system and the second is the seismic subsystem. The seismic main structural system category refers to the analysis of major buildings and structures which house and/or support Category I systems. The seismic subsystem category refers to smaller Category I structures, systems, and components.

The major structures which were analyzed in the main structural system analysis are:

a. auxiliary fuel handling building complex,

b. containment outer shell,

c. containment inner structure,

d. essential service water cooling towers (Byron),

e. river screen house (Byron), and

f. lake screen house (Braidwood).

3.7.2.3.2 Decoupling Criteria for Subsystems

All subsystems such as equipment and piping were decoupled from the floor they are supported on, as the structural mass of the supporting floor is large compared to the subsystem masses. However, their masses were included with the structural mass of the supporting floor slabs in the system model. In the containment inner structure model, the nuclear steam supply system (NSSS) was decoupled since it is basically supported from the containment base mat. Appropriate masses of the NSSS were lumped at the elevations of the lateral support points for the horizontal analysis.

Specific ratios between subsystem mass and system mass, R_m, or between fundamental frequencies of subsystem and system, R_f, were not used in subsystem decoupling since no quantitative criteria were available at the time the seismic model was generated. All subsystem mass was lumped at the appropriate location in the seismic model.

The subsystems generally have a small mass ratio (where R_m is less than 0.01) or frequencies away from resonance with the system (where $0.8 \geq R_f \geq 1.25$). Examples of large decoupled subsystems and approximate values for mass and frequency ratio for each subsystem are provided in Table 3.7-14.

The reactor pressure vessel (RPV), which has the largest R_m of the decoupled subsystems, was decoupled and its mass lumped at the support elevation on the basis of its rigid behavior vertically and its near rigid behavior horizontally. The mass and frequency ratios for the RPV do not meet the decoupling criteria of SRP 3.7.2.

The adequacy of decoupling the reactor pressure vessel was demonstrated by performing a response spectrum analysis and comparing the design basis forces with the forces from a model containing the RPV coupled to the containment inner structure in accordance with the SRP. The member forces of the coupled model were lower than the design basis model, ranging from 14% to 62% with an average change of 35%.

In addition, Westinghouse has performed a time history analysis using Westinghouse equipment models coupled to the Sargent & Lundy inner structure model. This analysis has shown that the resulting forces (the reactions of the NSSS supports) were lower than the forces obtained from a response spectrum analysis.

3.7.2.3.3 Lumped Mass Considerations

The criteria used to ensure that an adequate number of masses and degrees of freedom were employed in the dynamic modeling to determine the response of all Seismic Category I structures and equipment are given below. These criteria comply with SRP 3.7.2. Typical sketches of the mathematical models are given in Subsection 3.7.2.1.

For the shear structure system of the auxiliary fuel handling building two models were used, a model for the horizontal excitation and a model for the vertical excitation. In the horizontal model, every major slab level is represented in the model with two orthogonal horizontal translational dynamic degrees of freedom and a rotational dynamic degree of freedom about the vertical axis. In the vertical model, every major slab is represented in the model by at least one vertical dynamic degree of freedom, and every fundamental flexural slab frequency is represented with one vertical dynamic degree of freedom.

The mass points in the auxiliary fuel handling building model are considered at each slab elevation. In the Braidwood seismic model, there are a total of 21 major slabs. In the Byron model, a total of 18 slabs were considered since some of the slabs below grade are supported on rock. Additional degrees of freedom will not significantly affect the response of the shear structure. Additional mass joints located at midstory, for instance, for the sole purpose of obtaining more degrees of freedom will not affect the response because the mass of the shear wall is much less than the mass which would be lumped at the slab.

In general, each mass point has six degrees-of-freedom. However, in a shear structure system with rigid concrete slabs interconnected with shear walls, the predominant deformation is shear deformation under horizontal seismic excitation. Consequently, the relative rotation of the slabs about horizontal axes do not cause significant deformations. However, due to the asymmetrical mass-stiffness distribution, rotation of the slabs about a vertical axis may be significant. Therefore, in the horizontal model three degrees of freedom, two horizontal and a rotation about vertical axis, are considered for each slab.

The dynamic behavior of a building in the vertical direction is a function of the wall and column axial stiffness, the floor system flexural stiffness and the mass distribution. A plane frame model was used to simulate the behavior of the auxiliary fuel handling building for the vertical excitation. Appropriate masses were lumped at the wall slab junction at the midpoint of the slab. The predominant deformations are the axial deformation of the walls and the transverse deformation of the slabs, therefore, only the vertical degree-of-freedom was considered.

For the analysis of the containment, a beam or stick model with discrete masses was utilized to determine the seismic response. The stick model utilizes enough masses so there is a frequency match between the predominant frequencies of the thin shell model used for the design of the containment and the predominant frequencies of the stick model.

The dynamic characteristics for the containment structure stick models indicate the number of dynamic degrees of freedom meet the guidelines set forth in SRP 3.7.2. In the horizontal model of the containment shell, there are four modes with frequencies less than 33 cps and 26 dynamic degrees of freedom. In the vertical model, there is one mode with a frequency less than 33 cps and 13 dynamic degrees of freedom. In the horizontal model of the containment internal structure, there are eight modes with frequencies less than 33 cps and 60 dynamic degrees of freedom. In the vertical model, there are also eight modes with frequencies less than 33 cps and 27 dynamic degrees of freedom.

3.7.2.3.4 Modeling for Three Component Input Motions

As discussed in Subsection 3.7.2.3.3, two independent models, one in horizontal and the other in vertical direction are used. The horizontal and vertical models can be decoupled, since the response due to horizontal excitation in the vertical direction is negligible. In horizontal analysis of an asymmetrical structure, the seismic model is analyzed along the two principal axes of the structure and the results from the two analyses are combined on a square root of the sum of the squares basis. For symmetrical structures the model is analyzed along any one principal axis, since the response along both the principal axes is the same.

3.7.2.4 Soil/Structure Interaction

In Subsection 3.7.1.4, the supporting media for various seismic Category I structures has been given. For Byron Station, all the structures are supported on rock directly except for the river screen house. For structures founded on rock, the rock (foundation) motion was used directly to excite the fixed base model. For the Byron river screen house, soil-structure interaction was included in both horizontal and vertical directions. The horizontal soil-structure interaction was done using a finite element model. The criteria used in the finite element modeling and the general procedure was the same as described in Sargent & Lundy Report SL-3026, dated May 9, 1973 (Reference 7) except that the design spectrum compatible time-history was applied at the free field ground surface level. The report presents a general procedure for a coupled three-dimensional soil-structure interaction analysis using modal synthesis and finite element techniques. The strain-dependent soil properties used are discussed in Section 2.5. For vertical excitation, the soil column under the structure was modeled by a number of axial spring elements and masses, each of which represents a soil layer and a stiffness equivalent to the axial stiffness of that soil layer.

For Braidwood Station, the containment foundation is supported on rock. However, the auxiliary-fuel handling building complex is founded partly on bedrock and partly on soil. To account for the effect of soil beneath the slab, the structure was modeled by introducing shear springs and associated mass representing the soil between the slab and bedrock. The soil-structure model was analyzed using the foundation spectra at the bedrock.

3.7.2.5 Development of Floor Response Spectra

The time-history method of analysis is used to generate the floor response spectra. The spectra are generated at approximately 40 points in the period range of 0.02 to 2.0 seconds including all the structural periods of the seismic model. As discussed in Subsection 3.7.2.3.4 the horizontal and vertical models were decoupled and the floor response was obtained due to two separate analyses. For horizontal analysis, the response spectrum is generated for each floor along the two principal axes of the structure. In vertical analysis, the response spectra are generated for the slabs as well as at discrete mass points at the wall/slab junction for the design of subsystems supported off the wall.

The spectra were generated for 1, 2, and 4% of critical damping for OBE and for 2, 3, 4, 5, and 7% damping for SSE. The spectra were also generated for the damping values defined in ASME Code Case N-411 for OBE and SSE.

The peaks of the response spectra were widened as described in Subsection 3.7.2.9.

3.7.2.6 Three Components of Earthquake Motion

Seismic response resulting from analysis of systems due to three components of earthquake motions were combined in the following manner:

$$R = \sqrt{R_x^2 + R_y^2 + R_z^2} \qquad (3.7\text{-}1)$$

where:

- R = design seismic response,
- R_x = maximum seismic response due to horizontal earthquake motion along the x-axis,
- R_y = maximum seismic response due to vertical earthquake motion along the y-axis, and
- R_z = maximum seismic response due to horizontal earthquake motion along the z-axis.

R_x, R_y, and R_z are maximum, codirectional seismic responses of interest (strain, displacement, stress, moment, shear, etc., or their interaction coefficients) due to earthquake excitations in x, y, and z directions, respectively.

3.7.2.7 Combination of Modal Responses

When a response spectrum method of analysis is used to analyze a system, the maximum response (accelerations, shears, and moments) in each mode is calculated independent of time; whereas, actual modal responses in different modes do not occur simultaneously. Based on References 8 and 9 the final response R was computed as:

$$R = \left[\sum_{k=1}^{N} \sum_{\ell=1}^{N} R_k \; R_\ell \; \epsilon_{k\ell} \right]^{1/2} \qquad (3.7\text{-}2)$$

in which: $$\epsilon_{k\ell} = \left\{ 1 + \left[\frac{\left(\omega'_k - \omega'_\ell\right)}{\left(\beta'_k \; \omega_k + \beta_\ell \; \omega_\ell\right)} \right]^2 \right\}^{-1}$$

$$\omega'_k = \omega_k \left[1 - \left(\beta'_k\right)^2 \right]^{1/2}$$

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

where ω_k and β_k are the modal frequency and damping in the kth mode, respectively, and t_d is the duration of the earthquake.

For the time-history method of seismic analysis, the displacements, accelerations, shears, and moments due to each mode were added algebraically at each instant of time to obtain the final response.

When using independent support motion spectra, as defined in Subsection 3.7.3.1.2, the modal responses are combined by the square root sum of the squares.

3.7.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

When Seismic Category I and non-Category I structures are integrally connected, the non-Category I structures are included in the model when determining the forces on Seismic Category I structures.

Non-Category I structures integrally connected to or located in the close vicinity of Category I structures are designed for Category I loads in order to prevent their failure on Category I structures or evaluated for the effect of the failure of the non-Category I structure on the Category I structure when subjected to Category I loads. Table 3.7-15 describes such non-Category I structures and elements and shows loads for which they are designed.

The design and construction of non-Category I structures complies with the NRC Regulatory Staff position regarding interaction of non-Category I structures with Category I structures, as given in SRP Section 3.7.2.II.8. In particular, the design of the turbine building substructure and superstructure used the same SSE loading combinations and design allowables as were used in Category I design. Therefore, the turbine building has the same margin of safety as the Category I structures. The material suppliers and contractors for the construction of the turbine building were the same as for the construction of the Category I structures. Construction personnel have monitored the construction work and have ensured quality control. The quality of the construction is reflected in the average actual concrete strengths. The design requirement for the concrete compressive strength is 3500 psi. The Byron site was constructed with an average concrete strength of 5265 psi; Braidwood with an average of 5369 psi. These strengths were achieved in both the Category I and Category II structures.

Based on the equivalent margins of safety provided in the design of the turbine building and the Category I structures, and the quality control provided in the construction, the integrity and functionality of the essential service water piping has been assured. Quality control documentation for the construction of the turbine building basement is available at each plant site.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

To account for the expected variation in structural properties, damping and soil properties, the peaks of various floor response spectra curves were widened by 15% on the period scale to either side of the peak for horizontal as well as vertical components.

3.7.2.10 Use of Constant Vertical Static Factors

The Seismic Category I structures, systems, and components were analyzed in the vertical direction using the methods described in Subsection 3.7.2.1. No vertical static factors were used for the vertical analysis of Seismic Category I structures. However, each individual floor framing beam of the buildings was designed statically for 1.5 times the acceleration value corresponding to the fundamental frequency of the beam from the applicable wall response spectrum.

The above method is in conformance with SRP 3.7.2. Although the SRP states that a factor of 1.5 should be applied to the peak acceleration of the applicable floor response spectra, deviation from this approach is allowed with justification. The SRP permits the use of any rational and justifiable equivalent static load method. Section II.1.b.(3) of SRP 3.7.2 applies only to the design of floor-attached structures, equipment, and components and is based on a static load method which involves no analysis, i.e., no frequency calculation or modeling of the component. The equivalent static load design method stated above for the design of floor framing is a more comprehensive and realistic method. It involves modeling each floor framing member, determination of the fundamental frequency of the member, and consideration of source of seismic excitation and includes the effect of higher mode participation. The adequacy and conservatism of this method has been evaluated by comparison of results with a dynamic analysis for a typical floor framing. Results have been previously published in Reference 15.

The justification for using the wall response spectrum instead of the floor response spectrum is based on the fact that the floor framing members are supported by steel columns. Columns are included in the vertical seismic model with the walls. Seismic response of floor framing members is given by the response of the supporting columns. Column response is given by the applicable wall spectra. Therefore, it is appropriate to use the wall response spectra for the design of floor framing members. These wall response spectra adequately account for the amplification effect of any structures or components attached to the wall.

Since the floor framing member is modeled as a single degree of freedom system, an amplification factor of 1.5 is used to account for higher mode participation. This factor is a conservative value. A typical steel floor framing member behaves close to a single degree of freedom system, in which higher mode participation is insignificant. Use of 1.5 as the amplification factor for flexible beams having frequencies lower than 33 Hz will ensure the design adequacy of the equivalent static load method used herein.

3.7.2.11 Method Used to Account for Torsional Effects

The floor slabs in building structures, along with the heavy equipment resting on them, have asymmetric mass-stiffness distribution. Therefore, the slabs will rotate about their vertical axes when these structures are subjected to lateral seismic loads. This torsional response was accounted for in the horizontal building model by including a torsional degree-of-freedom in each slab.

In addition, to account for "accidental torsion", a seismic load corresponding to an eccentricity of ±5% of the maximum building dimension at each level has been applied to all Category I structures other than the auxiliary/fuel handling building. The Byron/Braidwood auxiliary/fuel handling building is a Category I structure interconnected with the turbine building which is a non-Category I structure. The plant layout of the auxiliary, fuel handling and turbine buildings is shown in Figure 3.7-57. Major slabs are continuous throughout both Category I and non-Category I buildings; the maximum building dimension is 875 feet. However, the auxiliary building, with a maximum length of 413 feet, provides the main shear resisting components.

The horizontal seismic models include the large eccentricities corresponding to the distribution of mass and stiffness for this structure. The eccentricities between the mass centroid for a slab and the center of rigidity of its supporting shear walls results in an average torsional moment of 8% of the maximum building dimension times the story shear for the three major slabs in the Byron station. Similarly an average torsional moment of 11% occurs in the Braidwood station. The amount of torsion considered in the design of the Byron/Braidwood auxiliary fuel handling turbine building is considerably larger than the 5% minimum required by the Uniform Building Code. It is unlikely that these eccentricities would increase significantly due to any changes other than major changes to the plant structure since the weight of the permanent structural elements accounts for a substantial percentage of the total mass. Since the plant construction is complete, such a change is not feasible. Therefore, it is not appropriate to include arbitrary torsion greater than what already exists as a result of actual eccentricities in the auxiliary fuel handling turbine building. It should be noted that the shear walls of the auxiliary fuel handling building have been evaluated considering an additional 5% of the design shear force. When using the actual material strengths, the evaluation has shown that all walls maintain stress levels within the design basis allowables.

3.7.2.12 Comparison of Responses

Table 3.7-6 shows shear forces obtained from response spectrum method of analysis tabulated against those obtained from time-history method of analysis. The values indicated are for major shear walls of the auxiliary building at elevation 401 feet 0 inch. It can be concluded from the table that the shear forces obtained from the two methods of analysis are comparable.

3.7.2.13 Methods for Seismic Analysis of Dams

This section is not applicable since there are no Seismic Category I dams at the Byron and Braidwood stations.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

In the design of shear walls for overturning moments, all the walls along one vertical plane spanning between floors are grouped together in the horizontal seismic model. Modal shear forces are integrated over the height of the structure to determine the overturning moments for each mode at each slab horizontal elevation. The maximum probable overturning moment in each direction is then obtained by the double sum method as described in Subsection 3.7.2.7. Due to vertical excitation, the effect of reduced dead load is considered in design.

3.7.2.15 Analysis Procedure for Damping

The damping values associated with various elements of the system (as given in Table 3.7-1) were used for all modes. Since there are no structural models with different element damping characteristics, no composite modal damping calculations were required for a normal mode solution.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Seismic Analysis Methods

Each pipeline is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear and axial deformations as well as changes in stiffness due to curved members. Next the mode shapes and the undamped natural frequencies are obtained. The dynamic response of the system is calculated by using the response spectrum method of analysis. When the piping system is anchored and/or supported at points with different excitations, the response spectrum analysis is performed using the enveloped response spectra of all response spectra which apply.

As an alternative, the seismic response of a piping system can be calculated by using the independent support motion response spectrum method of analysis, with the following rules for response combination:

a. Group responses for each direction shall be combined by the absolute sum method.

b. Modal and directional responses shall be combined by the SRSS method.

3.7.3.1.1 Seismic Analysis Methods (Westinghouse)

This subsection describes the seismic analysis methods performed for safety-related components and systems supplied by Westinghouse.

Those components and systems that must remain functional in the event of the SSE (Seismic Category I) are identified by applying the criteria of Subsection 3.2.1. This equipment is classified into three types according to its dynamic characteristics. The analysis methods used for this equipment also depended on these classifications.

The first type is flexible equipment. This equipment is characterized by several modes in the frequency range that could produce amplification of the base input motion. The components which are classified as flexible equipment, i.e., with more than one mode below 33 Hz, are the steam generators, reactor coolant pumps, pressurizers, control drive mechanisms, reactor internals, and fuel. Dynamic analyses were performed for these components using modal analysis techniques with either the response spectrum method, integration of the uncoupled modal equations of motion, or by direct integration of the coupled differential equation of motion. Details of the methods used for these analyses are described in Subsections 3.7.3.1.1.1 through 3.7.3.1.1.5.

The second classification is rigid equipment. This equipment has a fundamental natural frequency that is sufficiently high (greater than 33 Hz) so that base input motions are not amplified. Such equipment is particularly suitable for static analysis as described in Subsection 3.7.3.1.1.6.

Finally, the third type of equipment is classified as limited flexible, with only one predominate mode in the frequency range subject to possible amplification of the input motion. The fundamental mode of this type of equipment is basically a translational bending mode at a frequency less than 33 Hz. The second mode is usually a rocking mode with a frequency greater than 33 Hz. Because of the simple response characteristics of the equipment, dynamic analysis techniques that account for multiple mode effects and closely spaced modes are not required. Therefore, this equipment was evaluated using static analysis methods as described in Subsection 3.7.3.1.1.6.

3.7.3.1.1.1 Dynamic Analysis - Mathematical Model

The first step in any dynamic analysis is to model the structure or component, i.e., convert the real structure or component into a system of masses, springs, and dashpots suitable for mathematical analysis. The essence of this step is to select a model so that the displacements obtained will be a good representation of the motion of the structure or component. Stated differently, the true inertia forces should not be altered so as to appreciably affect the internal stresses in the structure or component. Some typical modeling techniques are presented in Reference 3.

Equations of Motion

Consider the multi-degree-of-freedom system shown in Figure 3.7-54. Making a force balance on each mass point r, the equations of motion can be written in the form:

$$m_r \ddot{y}_r = \overset{i}{\Sigma}\, c_{ri} \dot{u}_i + \overset{i}{\Sigma}\, k_{ri} u_i = 0 \qquad (3.7\text{-}3)$$

where:

m_r = the value of the mass or mass moment of rotational inertia at mass point r

$\ddot{y}_r$ = absolute translational or angular acceleration of mass point r

c_{ri} = damping coefficient - external force or moment required at mass point r to produce a unit translational or angular velocity at mass point i, maintaining zero translational or angular velocity at all other mass points. Force or moment is positive in the direction of positive translational or angular velocity

$\dot{u}_i$ = translational or angular velocity of mass point i relative to the base

k_{ri} = stiffness coefficient - the external force (moment) required at mass point r to produce a unit deflection (rotation) at mass point i, maintaining zero displacement (rotation) at all other mass points

Force (moment) is positive in the direction of positive displacement (rotation)

u_i = displacement (rotation) of mass point i relative to the base

As an example, note that Figure 3.7-54 does not attempt to show all of the springs (and none of the dashpots) which are represented in Equation 3.7-3.

Since:

$$\ddot{y}_r = \ddot{u}_r + \ddot{y}_s \qquad (3.7\text{-}4)$$

where:

$\ddot{y}_s$ = absolute translational (angular) acceleration of the base

$\ddot{u}_r$ = translational (angular) acceleration of mass point r relative to the base

$$m_r\ddot{u}_r + \overset{i}{\Sigma}\ c_{ri}\dot{u}_i + \overset{i}{\Sigma}\ k_{ri}u_i = -m_r\ddot{y}_s \qquad (3.7\text{-}5)$$

For a single degree-of-freedom system with displacement u, mass m, damping c, and stiffness k, the corresponding equation of motion is:

$$m\ddot{u} + c\dot{u} + ku = -m\ddot{y}_s \qquad (3.7\text{-}6)$$

3.7.3.1.1.2 Modal Analysis

Natural Frequencies and Mode Shapes

The first step in the modal analysis method is to establish the normal modes, which were determined by eigen solution of Equation 3.7-5. The right hand side and the damping term are set equal to zero for this purpose as illustrated in Reference 4 (Pages 83 through 111). Thus, Equation 3.7-5 becomes:

$$m_r\ddot{u}_r + \overset{i}{\Sigma}\ k_{ri}u_i = 0 \qquad (3.7\text{-}7)$$

The equation given for each mass point r in Equation 3.7-7 can be written as a system of equations in matrix form as:

$$[M]\{\ddot{\Delta}\} + [K]\{\Delta\} = 0 \qquad (3.7\text{-}8)$$

where:

[M]	=	mass and rotational inertia matrix
$\{\Delta\}$	=	column matrix of the general displacement and rotation at each mass point relative to base
[K]	=	square stiffness matrix
$\{\ddot{\Delta}\}$	=	column matrix of general translational and angular accelerations at each mass point relative to the base, $d^2\{\Delta\}/dt^2$.

Harmonic motion is assumed and the $\{\Delta\}$ is expressed as:

$$\{\Delta\} = \{\delta\}\sin\omega t \qquad (3.7\text{-}9)$$

where:

$\{\delta\}$	=	column matrix of the spatial displacement and rotation at each mass point relative to the base
$\{\omega\}$	=	natural frequency of harmonic motion in radians per second

The displacement function and its second derivative are substituted into Equation (3.7-8) and yield:

$$\{K\}\{\delta\} = \omega^2 [M]\{\delta\} \qquad (3.7\text{-}10)$$

The determinant $|[K] - \omega^2[M]|$ is set equal to zero and is then solved for the natural frequencies. The associated mode shapes are then obtained from Equation 3.7-10. This yields n natural frequencies and mode shapes where n equals the number of dynamic degrees of freedom of the system. The mode shapes are all orthogonal to each other and are sometimes referred to as normal mode vibrations. For a single degree-of-freedom system, the stiffness matrix and mass matrix are single terms and the determinant $|[K] - \omega^2[M]|$ when set equal to zero yields simply:

$$k - \omega^2 m = 0$$

Or $\qquad (3.7\text{-}11)$

$$\omega = \sqrt{\frac{k}{m}}$$

where ω is the natural angular frequency in radians per second.

The natural frequency in Hz is therefore:

$$f = \frac{1}{2\pi}\sqrt{\frac{k}{m}} \qquad (3.7\text{-}12)$$

To find the mode shapes, the natural frequency corresponding to a particular mode, ω_n, can be substituted in Equation 3.7-10,

<u>Modal Equations</u>

The response of a structure or component is always some combination of its normal modes. Good accuracy can usually be obtained by using only the first few modes of vibration. In the normal mode method, the mode shapes are used as principal coordinates to reduce the equations of motion to a set of uncoupled differential equations that describe the motion of each mode n. These equations may be written as (Reference 4, Pages 116 through 125).

$$\ddot{A}_n = + \omega_n p_n \dot{A}_n + \omega_n^2 A_n = - r_n \ddot{y}_s \qquad (3.7\text{-}13)$$

where the modal displacement or rotation, A_n, is related to the displacement or rotation of mass point r in mode n, u, by the equation:

$$u_{rn} = A_n \phi_{rn} \qquad (3.7\text{-}14)$$

where:

ω_n	=	natural frequency of mode in radians per second
p_n	=	critical damping ratio of mode n
r_n	=	modal participation factor of mode n given by:

$$r_n = \frac{\sum^{n} m_r \phi'_{rn}}{\sum^{n} m_r \phi_{rn}^2} \qquad (3.7\text{-}15)$$

where:

ϕ'_{rn}	=	Value of ϕ_{rn} in the direction of the earthquake

The essence of the modal analysis lies in the fact that Equation 3.7-13 is analogous to the equation of motion for a single degree-of-freedom system that will be developed from Equation 3.7-6. Dividing Equation 3.7-6 by m gives:

$$\ddot{u} = \left(\frac{c}{m}\right)\dot{u} + \left(\frac{k}{m}\right)u = -\ddot{y}_s \qquad (3.7\text{-}16)$$

The critical damping ratio of a single degree of freedom system, p, is defined by the equation:

$$p = \frac{c}{c_c} \qquad (3.7\text{-}17)$$

where the critical damping coefficient is given by the expression:

$$c_c = 2\,m\,\omega \qquad (3.7\text{-}18)$$

Substituting Equation 3.7-18 into Equation 3.7-17 and solving for c/m gives:

$$\frac{c}{m} = 2\omega p \qquad (3.7\text{-}19)$$

Substituting this expression and the expression for k/m given by Equation 3.7-11 into Equation 3.7-16 gives:

$$\ddot{u} + 2\omega p\dot{u} + \omega^2 u = -\ddot{y}_s \qquad (3.7\text{-}20)$$

Note the similarity of Equations 3.7-13 and 3.7-20. Thus each mode may be analyzed as though it were a single degree-of-freedom system and all modes are independent of each other. By this method a fraction of critical damping, i.e., c/c_c, may be assigned to each mode and it is not necessary to identify or evaluate individual damping coefficients, i.e., c. However, assigning only a single damping ratio to each mode has a drawback. Normally, there are two ways used to overcome this limitation when considering a slightly damped structure (e.g., steel) supported by a massive moderately damped structure (e.g., concrete).

The first method is to develop and analyze separate mathematical models for both structures using their respective damping values. The massive moderately damped support structure is analyzed first. The calculated response at the support points for the slightly damped structures is used as a forcing function for the subsequent detailed analysis. The second method is to inspect the mode shapes to determine which modes correspond to the slightly damped structure and then use the damping associated with the structure having predominant motion.

3.7.3.1.1.3 Response Spectrum Analysis

The response spectrum is a plot showing the variation in the maximum response (Reference 5, Pages 24 through 51) (displacement, velocity, and acceleration) of a single degree-of-freedom system versus its natural frequency of vibration when subjected to a time-history motion of its base. Examples of response spectra are shown in Figures 3.7-55 and 3.7-56.

The response spectrum concept can be best explained by outlining the steps involved in developing a spectrum curve. Determination of a single point on the curve requires that the response (displacement, velocity, and acceleration) of a single degree-of-freedom system with a given damping and natural frequency is calculated for a given base motion. The variations in response are established and the maximum absolute value of each is plotted as an ordinate with the natural frequency used as the abscissa. The process is repeated for other assumed values of frequency in sufficient detail to establish the complete curve. Other Curves corresponding to different fractions of critical damping are obtained in a similar fashion. Thus, the determination of each point of the curve requires a complete dynamic response analysis, and the determination of a complete spectrum may involve hundreds of such analyses. However, once a response spectrum plot is generated for the particular base motion, it may be used to analyze each structure and component with that base motion. The spectral acceleration, velocity, and displacement are related by the equation:

$$S_{an} = \omega_n S_{vn} = \omega_n^2 S_{dn} \qquad (3.7\text{-}21)$$

There are two types of response spectra that must be considered. If a given building is shown to be rigid and to have a hard foundation, the ground response spectrum or ground time-history is used. It is referred to as a ground response spectrum. If the building is flexible and/or has a soft foundation, the ground response spectrum is modified to include these effects. The response spectrum at various support points must be developed. These are called floor response spectra. The specific response spectra used are discussed in Subsection 3.7.1 and 3.7.2.5.

3.7.3.1.1.4 Integration of Modal Equations

This method can be separated into the following two basic parts:

a. Integration procedure for the uncoupled modal Equation 3.7-13 to obtain the modal displacements and accelerations as a function of time.

b. Using these modal displacements and accelerations to obtain the total displacements, accelerations, forces, and stresses.

Integration Procedure

Integration of these uncoupled modal equations is done by step-by-step numerical integration. The step-by-step numerical integration procedure consists of selecting a suitable time interval, Δ_+, and calculating modal acceleration, $\ddot{A}_n$, modal velocity, $\dot{A}_n$, and modal displacement, A_n, at discrete time stations Δ_+ apart, starting at t = 0 and continuing through the range of interest for a given time-history of base acceleration.

Total Displacements, Accelerations, Forces, and Stresses

From the modal displacements and accelerations, the total displacements, accelerations, forces, and stresses can be determined as follows:

a. Displacement of mass point r in mode n as a function of time is given by Equation 3.7-14 as:

$$u_{rn} = A_n\phi_{rn} \qquad (3.7\text{-}22)$$

with the corresponding acceleration of mass point r in mode n as:

$$\ddot{u}_{rn} = \ddot{A}_n\phi_{rn} \qquad (3.7\text{-}23)$$

b. The displacement and acceleration values obtained for the various modes are superimposed algebraically to give the total displacement and acceleration at each time interval.

c. The total acceleration at each time interval is multiplied by the mass to give an equivalent static force. Stresses are calculated by applying these forces to the model or from the deflections at each time interval.

3.7.3.1.1.5 Integration of Coupled Equations of Motion

The dynamic transient analysis is a time-history solution of the response of a given structure to known forces and/or displacement forcing functions. The structure may include linear or nonlinear elements, gaps, interfaces, plastic elements, and viscous and Coulomb dampers. Nodal displacements, nodal forces, pressure, and/or temperatures may be considered as forcing functions. Nodal displacements and elemental stresses for the complete structure are calculated as functions of time.

The basic equations for the dynamic analysis are as follows:

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F(t)\} \qquad (3.7\text{-}24)$$

where the terms are as defined earlier and (F(t)) may include the effects of applied displacements, forces, pressures, temperatures, or nonlinear effects such as plasticity and dynamic elements with gaps. Options of translational accelerations input to a structural system and the inclusion of static deformation and/or preload may be considered in the nonlinear dynamic transient analysis. The option of translational input such as uniform base motion to a structural system is considered by introducing an inertia force term of -[M] z to the right hand side of the basic Equation 3.7-24, i.e.,

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F\} - [M]\{\ddot{z}\} \qquad (3.7\text{-}25)$$

The vector $\ddot{z}$ is defined by its components $\ddot{z}_i$ where i refers to each degree-of-freedom of the system. $\ddot{z}_i$ is equal to a_1, a_2, or a_3 if the i-th degree-of-freedom is aligned with the direction of the system translational acceleration a_1, a_2, or a_3, respectively. $\ddot{z}_i = 0$ if the i-th degree-of-freedom is not aligned with any direction of the system translational acceleration. Typical application of this option is a structural system subjected to a seismic excitation of a given ground acceleration record. The displacement {x} obtained from the solution of Equation 3.7-25 is the displacement relative to the ground.

The option of the inclusion of initial static deformation or preload in a nonlinear transient dynamic structural analysis is considered by solving the static problem prior to the dynamic analysis. At each stage of integration in transient analysis, the portion of internal forces due to static deformation is always balanced by the portion of the forces which are statically applied. Hence, only the portion of the forces which deviate from the static loads will produce dynamic effects. The output of this analysis is the total result due to static and dynamic applied loads.

One available method for the numerical integration of Equations 3.7-24 and 3.7-25 is the Newmark Beta integration scheme proposed by Chan, Cox, and Benfield (Reference 6). In this integration scheme, Equations 3.7-24 and 3.7-25 are replaced by:

$$\frac{1}{(\Delta t)^2}[M]\{x_{n+2} - 2x_{n+1} + x_n\} + \frac{1}{2(\Delta t)}\{x_{n+2} - x_n\}[C] \qquad (3.7\text{-}26)$$

$$+ [K]\{\beta x_{n+2} + (1-2\beta)\ x_{n+1} + \beta x_n\}$$

$$= \{\beta F_{n+2} + (1-2\beta)\ F_{n+1} + \beta F_n\}$$

n, n+1, n+2 = past, present, and future (updated) values of the variables

β = parameter to be selected on the basis of numerical stability and accuracy

[F] = the total right hand side of the equation of motion (Equation 3.7-24 or 3.7-25)

Δ_t = $t_{n+2} - t_{n+1} = t_{n+1} = t_n$

The value of β is chosen equal to 1/3 in order to provide a margin of numerical stability for nonlinear problems. Since the numerical stability of Equation 3.7-26 is mostly determined by the left-hand side terms of that equation, the right-hand side terms were replaced by F_{n+2}. Furthermore, since the time increment may vary between two successive time substeps, Equation 3.7-24 may be modified as follows:

$$\frac{2}{(\Delta t + \Delta t_1)}[M]\left\{\frac{x_{n+2} - x_{n+1}}{\Delta t} - \frac{x_{n+1} - x_n}{\Delta t_1}\right\} + \frac{1}{(\Delta t = \Delta t_1)}[C]$$

$$\{x_{n+2} - x_n\} + \frac{1}{3}[K]\{x_{n+2} + x_{n+1} + x_n\} = \{F_{n+2}\} \qquad (3.7\text{-}27)$$

By factoring x_{n+2}, x_{n+1}, and x, and rearranging terms, Equation 3.7-28 is obtained as follows:

$$\{C_5[M] + C_3[C] + (1/3)[K]\}\{x_{n+2}\} = \{F_{n+2}\} +$$

$$\{C_7[M] - (1/3)[K]\}\{x_{n+1}\} \qquad (3.7\text{-}28)$$

$$+ \{-C_2[M] + C_3[C] - (1/3)[K]\}\{x_n\}$$

where:

$$C_2 = \frac{2}{\Delta t_1(\Delta t = \Delta t_1)}$$

$$C_3 = \frac{1}{\Delta t + \Delta t_1}$$

$$C_5 = \frac{2}{\Delta t(\Delta t + \Delta t_1)}$$

$$C_7 = C_2 + C_5$$

The above set of simultaneous linear equations is solved to obtain the present values of nodal displacements (x_t) in terms of the previous (known) values of the nodal displacements. Since [M], [C], and [K] are included in the equation, they can also be time or displacement dependent.

3.7.3.1.1.6 Static Analysis - Rigid and Limited Flexible Equipment

Rigid equipment and limited flexible equipment as defined in Subsection 3.7.3.1.1 are generally analyzed using the static analysis method. This technique involves the multiplication of the total weight of the equipment or component member by a specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient was established on the basis of the excitation level that the component was expected to experience in the plant.

For rigid equipment, the seismic acceleration coefficients were compared with the high frequency (greater than 33 Hz) acceleration levels for the applicable response spectra developed for the plant to confirm the design analysis. The seismic acceleration coefficients for limited flexible equipment are compared with the acceleration levels from the applicable response spectra at the calculated fundamental natural frequency of the component. If the design seismic acceleration coefficients for either rigid or limited flexible equipment are exceeded by the actual plant acceleration levels, the design analysis is performed again at the actual level to confirm the equipment adequacy.

3.7.3.1.2 Differential Seismic Movements of Interconnected Supports

Systems that are supported at points which undergo certain displacements due to a seismic event are designed to remain capable of performing their Seismic Category I functions. The displacements, obtained from a time-history analysis of the supporting structure, cause moments and forces to be induced into the piping system. Since the resulting stresses are self-limiting, it is justified to place them in the secondary stress category. Therefore these stresses exhibit properties much like a thermal expansion stress and a static analysis is used to obtain them.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Piping

During the plant life five Operating Basis Earthquakes (OBE) are assumed for piping subsystem fatigue analysis. Ten maximum stress cycles per earthquake are considered for a total of 50 maximum stress cycles for the 5 OBEs.

3.7.3.2.2 Equipment

In case of single frequency sinusoidal input or sine-beat input, a dwell test is performed at each of the equipment's natural frequencies for a period of 30 seconds or 200 cycles, whichever is smaller.

In case of random frequency test, the test is performed for a period of 30 seconds.

Since the issuance of IEEE 344-1975, some equipment is tested for five OBE loadings, followed by one SSE loading with a test period of 30 seconds for each loading.

3.7.3.2.3 Equipment Supplied by Westinghouse

For each OBE the system and component will have a maximum response corresponding to the maximum induced stresses. The effect of these maximum stresses for the total number of OBE's must be evaluated to assure resistance to cyclic loading.

The OBE is conservatively assumed to occur 20 times over the life of the plant. The number of maximum stress cycles for each occurrence depends on the system and component damping values, complexity of the system and component, duration and frequency contents of the input earthquake. A precise determination of the number of maximum stress cycles can only be made using time-history analysis for each item which is not feasible. Instead, a time-history study has been conducted to arrive at a realistic number of maximum stress cycles for all Westinghouse systems and components.

To determine the conservative equivalent number of cycles of maximum stress associated with each occurrence, an evaluation was performed considering both equipment and its supporting building structure as single degree-of-freedom systems. The natural frequencies of the building and the equipment are conservatively chosen to coincide. The damping in the equipment and building are equivalent to the damping values in Table 3.7-2.

The results of this study indicate that the total number of maximum stress cycles in the equipment having peak acceleration above 90% of the maximum absolute acceleration did not exceed eight cycles. If the equipment was assumed to be rigid in a flexible building, the number of cycles exceeding 90% of the maximum stress was not greater than three cycles.

This study was conservative since it was performed with single degree-of-freedom models which tend to produce a more uniform and unattenuated response than a complex interacted system. The conclusions indicate that 10 maximum stress cycles for flexible equipment (natural frequencies less than 33 Hz) and 5 maximum stress cycles for rigid equipment (natural frequencies greater than 33 Hz) for each of 20 OBE occurrences should be used for fatigue evaluation of Westinghouse systems and components.

3.7.3.3 Procedure Used for Modeling

Procedures used for modeling safety-related components and systems within Westinghouse's scope are discussed in Subsection 3.7.3.1.1.1.

Rigid valves (i.e., with natural frequencies greater than 33 Hz) are included in the piping system model as lumped masses on rigid extended structures. If it is shown, by test or analysis, that a valve is not rigid (one or more natural frequencies below 33 Hz), then a multimass, dynamic model of the valve, including the appropriate stiffnesses, is developed for use in the piping system model. The valve model used in the piping analysis is constructed such that its calculated frequencies correspond to those obtained by test.

For Class 1 piping systems analyzed by Westinghouse, the actual calculated stiffness of pipe supports are included in the model of the piping system. In the modeling of Class 2 and 3 piping systems analyzed by Westinghouse, pipe supports are represented as minimum rigid elements for which the stiffness is predetermined based upon the particular support type.

For Class 1, 2, and 3 piping systems analyzed by Sargent & Lundy, pipe supports are modeled as infinitely rigid elements.

For future modifications, a support may be modelled as a rigid boundary or with an applicable stiffness value as deemed appropriate for the corresponding design.

The mathematical model used for the dynamic analyses of the reactor coolant system is shown in Figure 3.9-1.

3.7.3.3.1 Modeling of the Piping System

The continuous piping system is modeled as an assemblage of beams. The mass of each beam is lumped at nodes which are connected by weightless elastic members, representing the physical properties of each segment. The pipe lengths between mass points are not greater than the length which would have a natural frequency of 33 Hz when calculated as a simply supported beam. All concentrated weights on the piping system such as main valves, relief valves, pumps, and motors are modeled as lumped masses. The torsional effects of the valve operators and other equipment with offset center of gravity with respect to centerline of the pipe is included in the analytical model.

3.7.3.3.2 Field Location of Supports and Restraints

The field location of seismic supports and restraints for Seismic Category I piping and piping systems components is selected to satisfy the following two conditions:

a. The location selected must furnish the required response to control strain within allowable limits.

b. Adequate building strength for attachment of the components must be available.

The final location of seismic supports and restraints for Seismic Category I piping, piping system components, and equipment, including the placement of snubbers, is checked against the drawings and instructions issued by the engineer. An additional examination of these supports and restraints devices is made to ensure that the location and characteristics of these supports and restraining devices are consistent with the dynamic and static analyses of the systems.

For Byron Units 1 and 2 portions of the auxiliary building Seismic Category I piping and all containment building piping, with the exception of containment spray rings, was designed by Westinghouse. The remainder of Seismic Category I piping, including the location of supports and restraints, is designed by Sargent & Lundy. For Braidwood Units 1 and 2, all Seismic Category I piping, including the location of supports and restraints, was designed by Sargent & Lundy (with the exception of the NSSS piping which was designed by Westinghouse). The field location of supports and restraints is done only for Seismic Category II piping, 4-inch nominal pipe size and smaller, and 200°F and colder.

3.7.3.4 Basis for Selection of Frequencies

The basis for the selection of forcing frequencies is presented in the seismic qualification criteria. All frequencies in the range of 1 to 33 hertz are considered in the analysis and testing of the components and their supporting structures.

Three ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible, as follows:

a. If the equipment is rigid relative to the structure, the maximum acceleration of the equipment mass approaches that of the structure at the point of equipment support. The equipment acceleration value in this case corresponds to the low period region of the floor response spectra.

b. If the equipment is very flexible relative to the structure, the internal distortion of the structure is unimportant, and the equipment behaves as though supported on the ground.

c. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

d. In the case of the equipment and the support being flexible and the equipment having dominant frequencies within 1/2 to twice the support dominant frequency appropriate loads are obtained from an analysis of the equipment/support system and area used in the analysis of the equipment and the support. This is in compliance with SRP Section 3.7.3.II.4.

Also, as noted in Subsection 3.7.3.2, rigid equipment/support systems have natural frequencies greater than 33 hertz.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

Balance of Plant

No static load method is utilized in the seismic analyses of piping systems. However, in the seismic analyses of equipment, the equivalent static load method is used if the equipment is not rigid and a dynamic analysis is not performed.

If the fundamental natural period (FNP) is known, the static seismic coefficient is equal to 1.5 times the g level corresponding to the equipment FNP in the applicable response spectrum curves (RSC). If the FNP is unknown, the static coefficient is equal to 1.5 times the peak g level in the applicable RSC.

The equivalent seismic static load is the product of the equipment mass and the static seismic load coefficient and is applied at the center of gravity.

NSSS

The static load equivalent or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration coefficient. The magnitude of the seismic acceleration, coefficient is established on the basis of the expected dynamic responses characteristics of the component. Components which can be adequately characterized as single degree of freedom systems are considered to have a modal participation factor of one. Seismic acceleration coefficients for multi-degree of freedom systems which may be in the resonance region of the amplified response spectra curves are increased by 50% to account conservatively for the increased modal participation.

3.7.3.6 Three Components of Earthquake Motion

Seismic responses resulting from analysis of subsystems due to three components of earthquake motions are combined in the same manner as the seismic response resulting from the analysis of building structures (Subsection 3.7.2.6).

The following description is applicable to safety-related components and systems within Westinghouse's scope

The seismic design of the piping and equipment includes the effect of the seismic response of the supports, equipment, structures and components. The system and equipment response is determined using three earthquake components, two horizontal one vertical. The design ground response spectra, specified in Subsection 3.7.1, are the bases for generating these three input components. Floor response spectra are generated for two perpendicular horizontal directions, (i.e., N-S, E-W) and the vertical direction. System and equipment analysis is performed with these input components applied in the N-S, E-W and vertical directions. The damping values used in the analysis are those given in Table 3.7-2.

In computing the system and equipment response by response spectrum modal analysis the methods of Subsection 3.7.2.7 are used to combine all significant modal responses to obtain the combined unidirectional responses.

The combined total response is then calculated using the square root of the sum of the squares formula applied to the resultant unidirectional responses. For instance, for each item of interest such as displacement, force, stresses, etc., the total response is obtained by applying the above described method. The mathematical expression for this method (with R as the item of interest) is:

$$R_C = \left[\sum_{T=1}^{3} R_T^2 \right]^{1/2} \qquad (3.7\text{-}29)$$

where:

$$R_T = \left[\sum_{i=1}^{N} R_{Ti}^2 \right]^{1/2}$$

where:

R_C = total combined response at a point

R_T = value of combined response of direction T

R_{Ti} = absolute value of response for direction T, mode i

N = total number of modes considered

The subscripts can be reversed without changing the results of the combination.

Again, for the case of closely spaced modes, R_T in Equation 3.7-30 shall be replaced with R_T as given by Equation 3.7-48 in Subsection 3.7.3.7.

3.7.3.7 Combination of Modal Responses

When a response spectrum method of analysis is used to analyze a subsystem, the maximum response (accelerations, shears, and moments) in each mode is calculated independent of time; whereas, actual modal responses are nearly independent functions of time and maximum responses in different modes do not occur simultaneously. It has been shown that the probable maximum response is about equal to the square root of the sum of the squares of the modal maxima. This square root criterion is used in combining the modal responses in the response spectrum method of analysis. The method of analysis described in this subsection conforms to the guidance of Regulatory Guide 1.92.

The final response, R, is computed as the square root of the sum of the squares of individual modal responses, R. Thus

$$R = \left[\sum_{k=1}^{N} R_k^2 \right]^{1/2}$$

If the frequencies of the subsystems are well separated, the SRSS method (Equation 3.7-31) gives acceptable results, however, where the structural periods are not well separated, the coupling between close modes may be considered based on References 8 and 9. The final response would then be computed as

$$R = \left[\sum_{\ell=1}^{N} \sum_{k=1}^{N} \left| R_k \, R_\ell \varepsilon_{k\ell} \right| \right]^{1/2} \tag{3.7-32}$$

where:

$$\varepsilon_{k\ell} = \left\{ 1 + \left[\frac{\omega'_k - \omega'_\ell}{\beta'_k \omega_k + \beta'_\ell \omega_\ell} \right]^2 \right\}^{-1} \tag{3.7-33}$$

in which:

$$\omega'_k = \omega_k \left[1 - \left(\beta'_k \right)^2 \right]^{1/2} \tag{3.7-34}$$

$$\beta'_k = \beta_k + \frac{2}{t_d \omega_k} \tag{3.7-35}$$

ω_k and β_k are the modal frequency and damping in the k^{th} mode, respectively and t_d is the duration of the earthquake.

In the double sum method of modal combination, a modified damping factor, instead of an uncorrected value, should be used for the damped frequency to evaluate the correlation coefficient of the closely space modes.

Equations 8 through 11 of Regulatory Guide 1.92 are based on a study by Rosenblueth and Elorduy (Reference 8). Referring to that paper, for a single-degree-of-freedom system governed by the equation of motion

$$\ddot{q}\,(t) + 2\,\xi_1\,\omega_1\,\dot{q}(t) + w_1^2\,q(t) = \ddot{x}(t)\,, \tag{3.7-36}$$

the correction factor for damping can be expressed as (Equation 4 of Reference 8)

$$\frac{E(Q)}{E(Q\circ)} = (1 + \xi_1\,\omega_1\,s / 2)^{1/2} \tag{3.7-37}$$

where S is the duration of a segment white noise excitation. $E(Q)$ and $E(Q\circ)$ are the expected value of the damped and the undamped systems.

The maximum response of a system to a transient disturbance of form $\ddot{x}(t) = f(t)\,W(t)$ can be expressed (Equation 8 of Reference 8)

$$Q^2 = \alpha \int_{\circ}^{\infty} \psi_q{}^2 dt \qquad (3.7\text{-}38)$$

The transfer function $\psi_q(t)$ for the deformation of the system expressed by Equation 3.7-36 is (Equation 10.3 of Reference 16)

$$\psi_q(t) = \frac{-1}{\omega_1{}'} \exp\left(-\xi_1\omega_1 t\right) \sin \omega_1{}' t \qquad (3.7\text{-}39)$$

where

$$\omega_1{}' = \omega_1 \quad 1 - \xi_1{}^2 \qquad (3.7\text{-}40)$$

when q is the pseudovelocity of a single-degree system, the second member in Equation 3.7-36 gives 1/2 $\xi_1\omega_1$.

In order to adjust the percentage of damping to coincide with the expected response, Rosenblueth suggested the use of a modified damping factor (Equation 9 of Reference 8)

$$\xi_1{}' = \xi_1 + 2/\omega_1 s \qquad (3.7\text{-}41)$$

in the system's natural modes of vibration.

In other words, the uncoupled equation of motion of a multi-degree-of-freedom system should be adjusted as

$$\ddot{q}_i(t) + 2\,\omega_i \xi_i' \;\dot{q}_i(t) + \omega_i{}^2 \; q_i(t) = a_i \ddot{x}(t) \qquad (3.7\text{-}42)$$

and the transfer function is given by (Equation 10.3 of Reference 16):

$$\psi_q(t) - \sum_i \frac{ai}{\omega_1{}'} \exp\left(-\xi_i'\omega_i t\right) \sin \omega_i' t \qquad (3.7\text{-}43)$$

where

$$\omega_i{}' = \omega_i \;\; 1 - \xi_i'{}^2 \qquad (3.7\text{-}44)$$

Note here that the modified damping factor ξ_i' and not the uncorrected damping value ξ_i, is used in Equation 3.7-44 for the damped frequency of the adjusted system.

The final solution is then obtained based on the transfer function of Equation 3.7-43 but not Equation 3.7-39, as

$$Q^2 = \sum_i Q_i^2 + \sum_{i \neq} \sum_j Q_i Q_j / (1 + \varepsilon_{ij}{}^2) \qquad (3.7\text{-}45)$$

where:

$$\Sigma_{ij} = \left|\omega_i' - \omega_j'\right| / \left(\xi_i' \omega_i + \xi_j' \omega_j\right) \qquad (3.7\text{-}46)$$

$$\omega_i' = \omega_i \sqrt{1 - \xi_i'^2} \qquad (3.7\text{-}47)$$

Thus a modified damping factor should be used for the damped frequency. Amin and Gungor (Reference 17) and Singh, Chu and Singh (Reference 9) also used the modified damping factor for the damped frequency in their computation of the correlation coefficient of closely spaced modes.

For a lightly damped system and an earthquake duration of 10 seconds as in the Byron/Braidwood design basis, the damped frequency based on an uncorrected damping factor are approximately the same. However, on a theoretical basis, the modified damping factor should be used for the damped frequency in the evaluation of the correlation of the closely spaced mode response. For a 10-hertz system with 2% damping and 10-second earthquake duration, the damped frequency using modified and uncorrected damping factors is 9.9973 hertz and 9.9980 hertz respectively. Thus the use of either modified or uncorrected damping factor does not affect the results.

The following description is applicable to safety-related components and systems within Westinghouse's scope.

The total unidirectional seismic response is obtained by combining the individual modal responses utilizing the square root of the sum of the squares method. For systems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. The groups of closely spaced modes are chosen such that the difference between the frequencies of the first mode and the last mode in the group does not exceed 10% of the lower frequency. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root of the sum of the squares of all modes the product of the responses of the modes in each group of closely spaced modes and a coupling factor.

This can be represented mathematically as:

$$R_T^2 = \sum_{i=1}^{N} R_i^2 + 2 \sum_{j=1}^{S} \sum_{K=M_j}^{N_{j-1}} \sum_{\ell=k+1}^{N_j} R_k R_\ell \varepsilon_{k\ell} \qquad (3.7\text{-}48)$$

where:

R_T	= total unidirectional response
R_{i1}, R_{k1}, $R_{c\ell}$	= absolute value of response of mode i, k, and ℓ, respectively
N	= total number of modes considered
S	= number of groups of closely spaced modes
M_j	= lowest modal number associated with group j of closely spaced modes
N_j	= highest modal number associated with group j of closely spaced modes
$\varepsilon_{k\ell}$	= coupling factor with

$$\varepsilon_{k\ell} = \left\{1 + \left[\frac{\omega'_k - \omega'_\ell}{\left(\beta'_k\omega_k + \beta'_\ell\omega_\ell\right)}\right]^2\right\}^{-1} \qquad (3.7\text{-}49)$$

and

$$\omega'_k = \omega_k\left[1 - \left(\beta'_k\right)^2\right]^{1/2} \qquad (3.7\text{-}50)$$

$$\beta'_k = \beta_k + \frac{2}{\omega_k t_d} \qquad (3.7\text{-}51)$$

where:

ω_k	= frequency of closely spaced mode K
β_k	= fraction of critical damping in closely spaced mode K
t_d	= duration of the earthquake

The method presented in Equation 3.7-48 and in associated Equations 3.7-49 and 3.7-50 to calculate the effects of closely spaced modes has been utilized by Westinghouse for several years in a number of applications and has been accepted by the NRC as an acceptable alternative to Regulatory Guide 1.92, one case of such acceptance was on the RESAR-414 application. Therefore, it is not necessary to present further justification of Equation 3.7-48.

An example of this equation applied to a system can be supplied with the following considerations. Assume that the predominant contributing modes have frequencies as given below:

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

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

S = 2 number of groups of closely spaced modes

M_1 = 2 lowest modal number associated with group 1

N_1 = 4 highest modal number associated with group 1

M_2 = 6 lowest modal number associated with group 2

N_2 = 7 highest modal number associated with group 2

N = 8 total number of modes considered

The total response for this system is, as derived from the expansion of Equation 3.7-48:

$$R_T^2 = [R_1^2 + R_2^2 + R_3^2 + ... + R_8^2] + 2\ R_2\ R_3\ \varepsilon_{23} + 2\ R_2\ R_4\ \varepsilon_{24} + 2\ R_3\ R_4\ \varepsilon_{34} + 2\ R_6\ R_7\ \varepsilon_{67} \quad (3.7\text{-}52)$$

For subsystems analyzed using the damping values defined in ASME Code Case N-411, an acceptable alternative modal combination method is the Ten Percent methods as defined in the NRC Regulatory Guide 1.92.

When using independent support motion response spectra as defined in Subsection 3.7.3.1.2, the modal responses are combined by the square root sum of the squares.

3.7.3.8 Analytical Procedures for Piping

3.7.3.8.1 Introduction

All Seismic Category I piping is analyzed for seismic effects by a dynamic response spectrum method of analysis.

3.7.3.8.2 Dynamic Analysis

Each pipeline is idealized as a mathematical model consisting of lumped masses connected by elastic members. Appendages having significant dynamic effects on the piping system, such as motors attached to motor-operated valves, are included in the model. Using the elastic properties of the pipe, the stiffness matrix for the piping system is determined. This includes the effects of torsional, bending, shear, and axial deformations, as well as change in stiffness due to curved members. Next, the frequencies and mode shapes for all the significant modes of vibrations are calculated. After the frequency is determined for each mode, the corresponding horizontal and vertical spectral accelerations with appropriate damping are read from the appropriate response spectrum curves. For each mode, the inertia forces, moments, displacements and accelerations are determined due to excitation in each of the three directions (two horizontal and one vertical). The modal responses in each of these directions are combined by the double sum technique or the ten percent method of the NRC Regulatory Guide 1.92. When using an independent support motion response spectrum, the modal responses shall be combined by the SRSS method (Subsection 3.7.3.7). Finally, the stresses are determined by taking the square root of the sum of the squares (SRSS) of the individual responses in the three directions. Horizontal and vertical earthquake excitations are assumed to occur simultaneously. All of the calculations outlined in this subsection are performed by using computer programs GAPPIPE, OPTPIPE, PIPSYS, or WESTDYN for the dynamic analysis of a three-dimensional piping system. (For a description of GAPPIPE, OPTPIPE, PIPSYS, and WESTDYN see Appendix D.)

The relative displacement between anchors corresponding to the elevation of seismic supports and the reactor pressure vessel at the elevation of the nozzles is determined from the dynamic analysis of the structures and vessel. The results of the relative anchor-point displacement are used as input of PIPSYS program for a static analysis to determine the additional stresses due to relative anchor-point displacements.

3.7.3.8.3 Allowable Stresses

Allowable stresses in the piping caused by an earthquake are in accordance with Section III of the ASME Code. Allowable stresses in the earthquake restraint components, such as shock suppressors, are in accordance with any additional stress limits that may have been established by ASME Section III at the time the restraint components were purchased. The stresses resulting from the piping supports relative movements are added to the thermal stresses of the piping.

3.7.3.8.4 Piping Supplied by NSSS Vendor

The Class 1 piping systems are analyzed to the rules of the American Society of Mechanical Engineers (ASME Code, Section III, NB-3650. When response spectrum methods are used to evaluate piping systems supported at different elevations, the following procedures are used. The effect of differential seismic movement of piping supports is included in the piping analysis according to the rules of the ASME Code, Section III, NB-3653. According to ASME definitions, these displacements cause secondary stresses in the piping system. The response quantity of interest induced by differential seismic motion of the support is computed statically by considering the building response on a mode-by-mode basis.

In the response spectrum dynamic analysis for evaluation of piping systems supported at different elevations, the most severe floor response spectrum corresponding to the support locations is used or the independent support motion response spectrum method is used as defined in subsection 3.7.3.1. The NSSS Vendor does not have in their scope of analysis any piping systems interconnected between buildings.

3.7.3.9 Multiple Supported Equipment Components with Distinct Inputs

When the equipment or component is supported at points with different elevations, the envelope of these elevation response spectra is used for the seismic qualification of the equipment.

For subsystems or components supplied by the NSSS Vendor, the following description is applicable.

When response spectrum methods are used to evaluate reactor coolant system primary components interconnected between floors, the procedures of the following paragraphs are used. There are no NSSS components which are connected between buildings. No primary component of the reactor coolant system is supported at more than two floor elevations.

A dynamic response spectrum analysis is first made assuming no relative displacement between support points. The response spectra used in this analysis are the most severe floor response spectra.

Secondly, the effect of differential seismic movement of components interconnected between floors is considered statically in the integrated system analysis and in the detailed component analysis. The results of the building analysis are reviewed on a mode-by-mode basis to determine the differential motion in each mode. Per ASME Code rules, the stress caused by differential seismic motion is clearly secondary for piping (NB-3650) and component supports (NF-3231). For components, the differential motion will be evaluated as a free end displacement, since, per NB-3213.19, examples of a free end displacement are motions "that would occur because of relative thermal expansion of piping, equipment, and equipment supports, or because of rotations imposed upon the equipment by sources other than the piping".

The effect of the differential motion is to impose a rotation on the component from the building. This motion, being a free end displacement and being similar to thermal expansion loads, will cause stresses which will be evaluated with ASME Code methods including the rules of NB-3227.5 used for stresses originating from restrained free end displacements.

The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses.

3.7.3.10 Use of Constant Vertical Static Factors

In general, Seismic Category I subsystems are analyzed in the vertical direction using the methods specified in Subsection 3.7.3.1. No vertical static factors are used for subsystems.

3.7.3.11 Torsional Effects of Eccentric Masses

All concentrated loads in the piping system, such as valves and valve operators are modeled as massless members with the mass of the components lumped at its center of gravity. A rigid member is modeled connecting the center of gravity to the piping so that the torsional effects of the eccentric masses are considered.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

During an earthquake, buried structures such as piping and tunnels respond to various seismic waves propagating through the surrounding soil as well as to the dynamic differential movements of the buildings to which the structures are connected. The various waves associated with earthquake motion are P (compression) waves, S (shear) waves, and Rayleigh waves. The stresses in the buried structure are governed by the velocity and angle of incidence of these traveling waves. However, the wave types and their directions during earthquake are very complex. For design purposes, expressions for upper bound stresses as given in the published results of Newmark (Reference 10), Yeh (Reference 11), and Shah and Chu (Reference 12) were used.

The criteria used to analyze seismic effects on buried Category I piping systems and electrical ducts are described below and agree with the criteria of SRP 3.7.3. Wave motion is assumed to be propagated in one direction without interference from other waves in other directions, and is assumed to act in the worst direction on piping components. It should be noted that no flange connections are used for buried piping components.

The following relationships for the maximum axial strain in the structural element are used:

When particle displacement is along the direction of propagation of wave,

$$\varepsilon_m = \frac{V_m}{C_p}$$

When particle displacement is perpendicular to the direction of propagation of wave,

$$\varepsilon_m = \frac{V_m}{2C_s}$$

ε_m = maximum axial strain in the homogeneous element,

V_m = maximum particle velocity,

C_p = compressive wave velocity,

C_s = shear wave velocity.

When a displacement Δ is imposed, in this medium, on the flexible element, the shear and moment induced in the element are given by the following expressions:

$$S = \frac{K\Delta}{\lambda}$$

$$M = \frac{K\Delta}{2\lambda^2}$$

where:

Δ = $\frac{\varepsilon_m L_m}{2}$

K = $K_o b$

λ = $4\sqrt{K/(4EI)}$

L_m = Maximum slippage length.

K_o = Modulus of subgrade reaction for soil.

b = Width of the element on elastic foundation.

I = Moment of inertia.

E = Modulus of elasticity of the element.

Δ = Displacement.

Since all buried essential service water piping falls under subsection ND of ASME B&PV Code, Section III, the following stress limits are met:

Stresses due to sustained loads	$\leq 1.0\ S_h$
Stresses due to occasional loads (OBE)	$\leq 1.2\ S_h$
Stresses due to occasional loads (SSE)	$\leq 1.8\ S_h$
Stresses due to bending moments caused by soil settlement and/or overburden pressure	$\leq 3.0\ S_c$

For all buried concrete electrical duct runs associated with the essential service water system, the design is in accordance with ACI-318-71 requirement.

The design of the main steam/auxiliary feedwater tunnel and the refueling water tunnel conform to the same load combinations and design allowables as the Category I structures other than the containment.

For Byron, the essential service water pipeline was encased in concrete. This concrete encasement was designed to span the 50 foot-diameter design-basis sinkhole described in Subsection 2.5.4.10.4. The concrete encasement was designed such that the design-basis sinkhole could occur at any portion along the pipeline route.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

The seismic induced effects of Category II piping systems on Seismic Category I piping are accounted for by including in the analysis of the Seismic Category I piping a length of the Category II systems, to the first anchor beyond the point where the change in category occurs.

3.7.3.14 Seismic Analysis for Reactor Internals

Fuel assembly grid impact loads and component stresses induced by horizontal seismic disturbances are analyzed through the use of finite element computer modeling. The time history floor response based on a standard seismic time-history normalized to SSE level is used as the seismic input. The reactor internals and the fuel assemblies are modeled as spring and lumped mass systems or beam elements. The component seismic response of the fuel assemblies is analyzed to determine design adequacy. The detailed discussion of the analysis performed for the standard, optimized, and VANTAGE 5 fuel assembly designs is contained in the References 2, 13, 14, 18, and 19.

The CRDMs are seismically analyzed to confirm that system stresses, under the combined loading conditions described in Subsection 3.9.1, do not exceed allowable levels as defined by the ASME Code, Section III for "Upset" and "Faulted" conditions. The CRDM is mathematically modeled as a system of lumped and distributed masses. The model is analyzed under appropriate seismic excitation and the resultant seismic bending moments along the length of the CRDM are calculated. The corresponding stresses are then combined with the stresses from the other loadings required and the combination is shown to meet ASME Code, Section III requirements.

The damping values given in Table 3.7-2 are used for the systems analysis of Westinghouse equipment. These are consistent with recommended damping values for seismic design except in the case of the primary coolant loop system components and large piping (excluding reactor pressure vessel internals) for which the damping values of 2% and 4% are used as established in testing programs reported in Reference 1. The damping values for control rod drive mechanisms (CRDMs) and the fuel assemblies of the nuclear steam supply system, when used in seismic system analysis, are in conformance with the values for welded and/or bolted steel structures (as appropriate). For reactor internals analysis, Westinghouse uses 2% damping for OBE and 4% damping for SSE as given by Regulatory Guide 1.61.

The damping values used in component analysis of CRDMs and their seismic supports were developed by testing programs performed by Westinghouse. These tests were performed during the design of the CRDM support; the support was designed so that the damping in Table 3.7-2 could be conservatively used in the seismic analysis. The CRDM support system is designed with plates at the top of the mechanism and gaps between mechanisms. These are encircled by a box section frame which is attached by tie rods to the refueling cavity wall. The test conducted was on a full size CRDM complete with rod position indicator coils, attachment to a simulated vessel head, and variable gap between the top of the pressure housing support plate and a rigid bumper representing the support. The internal pressure of the CRDM was 2250 psi and the temperature on the outside of the pressure housing was 400°F.

The program consisted of transient vibration tests in which the CRDM was deflected as specified initial amount and suddenly released. A logarithmic decrement analysis of the decaying transient provides the effective damping of the assembly. The effect on damping of variations in the drive shaft axial position, upper seismic support clearance, and initial deflection amplitude was investigated.

The upper support clearance had the largest effect on the CRDM damping with the damping increasing with increasing clearance.

With an upper clearance of 0.06 inches, the measured damping was approximately 8%. The clearances in a typical upper seismic CRDM support is a minimum of 0.10 inch. The increasing damping with increasing clearances trend from the test results indicated that the damping would be greater than 8% for both the OBE and the SSE based on a comparison between typical deflections during these seismic events to the initial deflections of the mechanisms in the test. Component damping values of 5% are, therefore, conservative for both the OBE and the SSE.

These damping values are used and applied to CRDM component analysis by response spectra techniques.

3.7.3.15 Analysis Procedure for Damping

In instances of the equipment supplied by Westinghouse, either the lowest damping value associated with the elements of the system is used for all modes, or an equivalent modal damping value is determined by testing programs such as was done for the reactor coolant loop (Reference 1).

3.7.4 Seismic Instrumentation

3.7.4.1 Instrumentation for Earthquakes

Seismic instrumentation is necessary to determine the seismic response of nuclear power plant features to permit comparison of such response with that used as the design basis.

The seismic instrumentation utilizes two types of sensor-recorders with an analysis capability available in the control room area. The location and function of these seismic devices were selected to provide for the determination of seismic event loads into the structures via computerized analysis programs. The system is maintained during a loss of ac power by a backup dc power supply. Loss of ac power operation is alarmed locally and in the main control room.

3.7.4.2 Location and Description of Instrumentation

3.7.4.2.1 Time-History Accelerograph

The time-history accelerograph is the central recording unit for the seismic monitoring instrumentation. There are eight time-history accelerographs utilized as part of the seismic instrumentation.

The time-history accelerograph located in the Unit 1 auxiliary electric equipment room is a central recorder which receives inputs from five other time-history accelerographs. Each of the five accelerographs are associated with a corresponding accelerometer, each of which measures the absolute acceleration as a function of time in three orthogonal directions; these directions coincide with the major axes of the analytical model of the structure.

These accelerometers are placed at the following locations:

1. in the free field at site coordinates 41+00E, 27+00N
2. on the containment building foundation slab at elevation 377 feet and azimuth 145 degrees,
3. on the containment shell wall at elevation 502 feet and azimuth 145 degrees,
4. on the containment refueling floor at elevation 426 feet, and
5. on the floor, elevation 426 feet, in the counting room in the auxiliary building.

Two time-history accelerographs and their sensors are located at the river screen house. One sensor is located at the foundation and the second sensor is located at elevation 702 feet. The response spectrum can be determined using a dedicated computer.

3.7.4.2.2 Peak Accelerographs

A triaxial peak recorder which measures the absolute peak acceleration in three orthogonal directions coinciding with the major axes of the analytical model is provided at each of the following locations:

a. on the accumulator tank located at elevation 426 feet in the containment building;

b. on the safety injection piping at elevation 421 feet in the containment building;

c. on the essential service return piping at elevation 346 feet in the auxiliary building.

3.7.4.2.3 Response Spectrum Analyzer

This unit determines the variation in the maximum response of a single degree-of-freedom system versus its natural frequency vibration when subjected to a time-history motion of its base. The response spectrum analyzer is a computer which determines the response spectrum of the event, compares it to the design response spectra of the plant, and indicates whether the event exceeded the OBE or SSE criteria.

Inputs for calculating the peak acceleration vs. frequency are measured at the two locations listed below:

a. on the base slab of the containment building, elevation 377 feet. This location serves the dual purpose of monitoring the base slab response and the support motion of reactor equipment recorded from the accelerometer through the time-history accelerograph.

b. on the floor, elevation 426 feet, in the counting room in the auxiliary building.

The seismic monitoring instrumentation provides an alarm in the Unit 1 auxiliary electrical equipment room in the event that preset acceleration limits are exceeded.

Regulatory Position C.1.c(3) of Regulatory Guide 1.12, "Instrumentation for Earthquakes," Revision 1 suggests a separate triaxial response spectrum recorder capable of measuring both horizontal motions and the vertical motion to be provided at the foundation of an independent Seismic Category I structure where the response is different from that of the reactor containment structure. Exception has been taken to this guidance. Except for the river screen house, all the structures are founded on rock, and will have the same foundation response as the containment structure. The river screen house is founded on about 80 feet of soil. In the seismic analysis of the river screen house, a soil structure interaction analysis was performed for both horizontal and vertical directions. The results show that the response at the foundation of the river screen house is lower than that of the containment building. Therefore, a separate triaxial response spectrum recorder is not needed.

These locations are chosen to allow meaningful correlation between the recorded accelerations and those calculated using the analytical model of the structure.

The specifications of the response spectrum analyzer, including dynamic range, frequency ranges, and damping, satisfy the guidance of Positions C.4 and C.5 of Regulatory Guide 1.12, Revision 1.

3.7.4.3 Control Room Operator Notification

The centrally located seismic indication and recording equipment in the Unit 1 auxiliary electrical equipment room is the source of operator information concerning the acknowledgment of an earthquake. An acceleration of 0.02g in any direction for the free field sensor triggers the system, initiates recording of the event, and provides an alarm in the main control room.

The system uses input from the containment building foundation and auxiliary building counting room accelerometers to determine if OBE/SSE limits have been exceeded. A comparison is made between the recorded data from these accelerometers and predefined acceleration limits contained in the response spectrum analyzer. An alarm in the Unit 1 auxiliary electrical equipment room indicates when the OBE/SSE limits have been exceeded. In accordance with 10 CFR 100 Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants", the reactor is shutdown if the vibratory ground motion or evaluated data exceeds OBE values. Data from the other sensor locations is also available for review after the seismic event.

3.7.4.4 Comparison of Measured and Predicated Responses

The computer program which evaluates the time-history data computes the maximum response accelerations at various points of the model. The observed response spectra can be compared with the reference response spectra. Agreement between the observed response spectra and the computed response spectra from the time-history inputs demonstrates the adequacy of the analytical model. The magnitude of actual forces at various structural locations can then be compared to design values to authenticate the capability of the plant to continue operation without undue risk to the health and safety of the public.

3.7.4 Seismic Instrumentation

3.7.4.1 Instrumentation for Earthquakes

Seismic instrumentation is necessary to determine the seismic response of nuclear power plant features to permit comparison of such response with that used as the design basis.

The seismic instrumentation utilizes two types of sensor-recorders with an analysis capability available in the control room area. The location and function of these seismic devices were selected to provide for the determination of seismic event loads into the structures via computerized analysis programs. The system is maintained during a loss of ac power by a backup dc power supply. Loss of ac power operation is alarmed locally and in the main control room.

3.7.4.2 Location and Description of Instrumentation

3.7.4.2.1 Time-History Accelerograph

There are six time-history accelerographs utilized as part of the seismic instrumentation.

Five accelerographs are connected to a central Network Control Center (NCC), located in the Unit 1 auxiliary electric equipment room. These accelerographs are associated with a corresponding accelerometer, each of which measures the absolute acceleration as a function of time in three orthogonal directions; these directions coincide with the major axis of the analytical model of the structure. Depending on their location, some accelerographs are located adjacent to the accelerometer in the same housing, while others are remote-mounted (away from the accelerometer) and located in auxiliary electric equipment room next to the NCC.

These accelerometers whose accelerographs are connected to the NCC are placed at the following locations:

1. in the free field at site coordinates 34+15E, 38+01S,

2. on the containment building foundation slab at elevation 377 feet and azimuth 145 degrees,

3. on the containment shell wall at elevation 502 feet and azimuth 145 degrees,

4. on the containment refueling floor at elevation 426 feet, and

5. on the floor, elevation 426 feet, in the counting room in the auxiliary building.

The sixth time-history accelerograph and its accelerometer is provided at elevation 335 feet 0 inch in the auxiliary building on the column row 18 wall adjacent to the column line “L” wall. This accelerograph is a self-contained unit independent of the other accelerograph and is not connected to the central NCC. The response spectrum for this location can be determined using the dedicated computer and analysis program for this accelerograph.

3.7.4.2.2 Peak Accelerographs

A triaxial peak recorder which measures the absolute peak acceleration in three orthogonal directions coinciding with the major axes of the analytical model is provided at each of the following locations:

a. on an accumulator tank located at elevation 426 feet in the containment building;

b. on the safety injection piping at elevation 421 feet in the containment building;

c. on the essential service water return piping at elevation 346 feet in the auxiliary building.

3.7.4.2.3 Response Spectrum Analyzer

The response spectrum analyzer is a laptop computer which determines the response spectrum of the event, compares it to the design response spectra of the plant, and indicates whether the event exceeded the OBE criteria.

Inputs for calculating the response spectrum are measured at the free field. In addition, the cumulative absolute velocity (CAV) is also calculated at this location and is part of the OBE exceedance determination. For each component of ground motion, the CAV is calculated as follows:

a. The absolute acceleration (g units) time-history is divided into 1-second intervals

b. Each 1-second interval that has at least 1 exceedance of 0.025g is integrated over time

c. All the integrated values are summed together to arrive at the CAV

The CAV is considered exceeded if any CAV calculation is greater than 0.16 g-second.

The seismic monitoring instrumentation provides an alarm in the Unit 1 auxiliary electrical equipment room in the event that preset acceleration limits are exceeded.

Regulatory Position C.1.2.4 of Regulatory Guide 1.12, "Nuclear Power Plant Instrumentation for Earthquakes," Revision 2 recommends providing a triaxial time-history accelerograph at the foundation of an independent Seismic Category I structure where the response is different from that of the reactor containment structure. Exception has been taken to this guidance. All the structures are founded on rock, and will have the same foundation response as the containment structure. Therefore, a separate triaxial time-history accelerograph on an independent Seismic Category I structure foundation where the response is different from that of the containment structure is not provided.

These locations are chosen to allow meaningful correlation between the recorded accelerations and those calculated using the analytical model of the structure.

The specifications of the accelerometers, accelerographs, NCC, and laptop, including the dynamic range, frequency, ranges, damping, triggers, and indication, satisfy the guidance of Positions C.4, C.6, and C.7 of Regulatory Guide 1.12, Revision 2.

3.7.4.3 Control Room Operator Notification

The centrally located seismic NCC in the Unit 1 auxiliary electrical equipment room is the source of operator information concerning the acknowledgment of an earthquake. An acceleration of 0.01g in any direction for the free field or containment foundation sensor triggers the system, initiates recording of the event, and provides an alarm in the main control room.

The system uses input from the free field sensor to determine if the OBE was exceeded. A comparison is made between the response spectrum of the event and the design response spectra. In addition, the CAV is calculated. When the response spectrum of the event exceeds the design response spectra and the CAV is exceeded, the OBE is exceeded and the reactor is shut down. An alarm in the Unit 1 auxiliary electrical equipment room indicates when the OBE limits have been exceeded. Data from the other sensor locations is also available for review after the seismic event.

3.7.4.4 Comparison of Measured and Predicated Responses

The computer program which evaluates the time-history data computes the maximum response accelerations at various points of the model. The observed response spectra can be compared with the reference response spectra. Agreement between the observed response spectra and the computed response spectra from the time-history inputs demonstrates the adequacy of the analytical model. The magnitude of actual forces at various structural locations can then be compared to design values to authenticate the capability of the plant to continue operation without undue risk to the health and safety of the public.

3.7.4.5 Post-Earthquake Actions

Information on the seismic instrumentation's characteristics and relevant data is maintained at the station. The actions taken immediately after an earthquake are those specified in Regulatory Position C.2 of Regulatory Guide 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions," March 1977.

A log is maintained at the station that allows traceability of the date/time of data collection and identification of the configuration and component information of the sensor from which the data was collected. Trained station personnel collect the instrument data in accordance with established procedures.

If both the response spectrum check and the CAV check are exceeded, the OBE is exceeded and the plant is required to shut down. If either check is not exceeded, then the OBE is not exceeded. If only one check can be performed, the other check is assumed to be exceeded. A determination of whether the OBE has been exceeded is performed even if the plant automatically shuts down as a result of the earthquake. If the response spectrum and CAV cannot be obtained because of a malfunction, the interim guidelines of Regulatory Guide 1.166, Appendix A are used to determine whether the OBE has been exceeded. Additional walkdowns are performed within eight hours of an earthquake to determine if any damage has occurred. If no damage is found during the walkdowns and the OBE was determined not to have been exceeded, the plant may continue running or may restart if it was shut down.

All pre-shutdown inspections are to be performed with existing plant procedures, which conforms to Regulatory Position C.6 of Regulatory Guide 1.166.

In order to restart after a seismic event, plant procedures conforming to Regulatory Position C.1 of Regulatory Guide 1.167 are provided. The plant must also be returned to its current licensing basis, in accordance with Regulatory Position C.2 of Regulatory Guide 1.167.

3.7.5 References

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17. M. Amin and I. Gungor, "Random Vibration in Seismic Analysis - An Evaluation," ASCE National Structural Engineering Meeting, Baltimore, Md., April 1971.

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19. S. L. Davidson (ed.), "Reference Core Report, VANTAGE 5 Fuel Assembly," WCAP 10444-P-A, September 1985.

20. B&W, Calculation 222-7720-B162,"Damping Values for Steam Generator Shell Internals," Revision 0, April 1997.

21. B&W, Calculation 222-7720-B163,"Damping Values for Feedwater Header and Primary Separators and Deck Assembly," Revision 0, May, 1997.

TABLE 3.7-1

DAMPING VALUES
(Percent of Critical)

ITEM	OPERATING BASIS EARTHQUAKE	SAFE SHUTDOWN EARTHQUAKE
STRUCTURE		
Welded steel frame structures	2.0	4.0
H.S. bolted steel frame structure	4.0	7.0
Bolted and riveted steel frame structure	4.0	7.0
Prestressed concrete structures	2.0	5.0
R-C shear wall structure	4.0	7.0
Equipment (steel assembly)	2.0	3.0
Piping (diameter ≤ 12 inches)*	1.0	2.0
Piping (diameter > 12 inches)*	2.0	3.0

* Alternative damping values per ASME Code Case N-411 may be used in lieu of the values shown above. If used, the conditions of NRC Regulatory Guide 1.84 must also be met.

TABLE 3.7-2

DAMPING VALUES USED FOR SYSTEMS ANALYSES
(Performed by Westinghouse)

ITEM	DAMPING** (Percent of Critical) UPSET CONDITION (OBE)	FAULTED CONDITION (SSE, DBA)
Primary coolant loop system components and large piping*	2	4
Small piping	1	2
Welded steel structures	2	4
Bolted and/or riveted steel structures	4	7
Unit 1 SG Lower Shell Internals***	5	7
Unit 1 SG Feedwater Header***	4	5
Unit 1 SG Primary Deck***	4	7

* Large piping refers only to piping greater than or equal to 12 inches in diameter and attached to the main reactor coolant loop piping or to the primary equipment of the reactor coolant system. Large piping that is not attached to the main reactor coolant loop piping or to the primary equipment of the reactor coolant system uses damping values of 2% for the upset condition and 3% for the faulted condition.

** Alternative damping values per ASME Code Case N-411 may be used in lieu of the values shown above. If used, the conditions of NRC Regulatory Guide 1.84 must also be met.

*** Damping values are in accordance with Regulatory Guide 1.61 and experimental data (see References 20 and 21 and ASME Code Case N-411).

TABLE 3.7-3

SUPPORTING MEDIA FOR SEISMIC

CATEGORY I STRUCTURES

STRUCTURE	DEPTH OF SOIL OVER BEDROCK (ft)	EMBEDMENT DEPTH OF FOUNDATION (ft)	SIZE OF STRUCTURAL FOUNDATION (ft)	TOTAL STRUCTURAL HEIGHT (ft)
Containment	0	38	157 (Diameter)	237
Auxiliary-Fuel Handling Building	0	Varies Max. 70 Min. 0	167 x 462	169
River Screen House	83	22	130 x 72	80
Essential Service Cooling Tower	0	9	172 x 42	60

Note: All dimensions are approximate.

TABLE 3.7-3

SUPPORTING MEDIA FOR SEISMIC

CATEGORY I STRUCTURES

STRUCTURE	DEPTH OF SOIL OVER BEDROCK (ft)	EMBEDMENT DEPTH OF FOUNDATION (ft)	SIZE OF STRUCTURAL FOUNDATION (ft)	TOTAL STRUCTURAL HEIGHT (ft)
Containment	0	38	157 (Diameter)	225
†Auxiliary Fuel Handling Building Complex	Varies Max. 36 Min. 0	Varies Max. 70 Min. 0	167 x 462	169
Lake Screen House	10	37	193 x 116	77

† See Figures 2.5-16, 2.5-26, and 3.8-45.

TABLE 3.7-4

SUMMARY OF NATURAL FREQUENCIES AND PARTICIPATION

FACTORS FOR AUXILIARY-FUEL HANDLING BUILDING

MODE	PERIOD (sec)	FREQUENCY (c/s)	PARTICIPATION FACTORS NORTH-SOUTH	EAST-WEST
1	1.0187	1.0	-.0	11.3
2	.6563	1.5	1.4	-.4
3	.4680	2.1	-12.3	-.1
4	.4408	2.3	2.4	-4.1
5	.3727	2.7	.6	33.2
6	.3727	2.7	-1.5	12.1
7	.3598	2.8	-6.2	-1.6
8	.2992	3.3	1.9	-8.5
9	.2631	3.8	-7.1	-3.4
10	.2537	3.9	20.6	-2.5
11	.2537	3.9	-10.9	-4.8
12	.2098	4.8	-23.9	-1.2
13	.2098	4.8	-13.1	2.2
14	.1869	5.3	-2.6	87.0
15	.1674	6.0	-3.2	-5.5
16	.1577	6.3	45.1	-6.4
17	.1534	6.5	46.9	.6
18	.1459	6.9	-4.4	-5.8
19	.1243	8.0	63.2	1.8
20	.0905	11.0	1.3	14.0
21	.0901	11.1	7.4	-25.6
22	.0810	12.3	2.6	.9
23	.0708	14.1	.3	-42.5
24	.0640	15.6	-17.9	-6.1
25	.0620	16.1	-27.4	-1.2
26	.0536	18.7	-26.8	3.5
27	.0519	19.3	1.4	7.8
28	.0509	19.6	-19.6	-4.9
29	.0494	20.2	.5	19.7
30	.0417	24.0	16.8	1.4
31	.0395	25.3	-1.8	10.8
32	.0379	26.4	10.4	-3.8
33	.0361	27.7	-2.0	-17.0
34	.0332	30.1	-13.3	.2
35	.0288	34.7	11.3	.6
36	.0282	35.4	-.5	-1.8
37	.0276	36.3	-5.5	.4
38	.0263	38.1	-.8	-3.2
39	.0260	38.5	17.1	2.3
40	.0259	38.6	0.0	-7.8

TABLE 3.7-4

SUMMARY OF NATURAL FREQUENCIES AND PARTICIPATION

FACTORS FOR AUXILIARY-FUEL HANDLING BUILDING

MODE	PERIOD (sec)	FREQUENCY (c/s)	PARTICIPATION FACTORS NORTH-SOUTH	PARTICIPATION FACTORS EAST-WEST
1	1.0187	1.0	-.0	-11.5
2	.6562	1.5	-1.5	.1
3	.4904	2.0	-21.3	-5.5
4	.4679	2.1	-13.0	-.2
5	.4410	2.3	-2.6	5.9
6	.4334	2.3	3.5	-32.9
7	.3762	2.7	.1	46.2
8	.3749	2.7	-2.0	-4.5
9	.3597	2.8	-6.5	-1.9
10	.2992	3.3	2.0	-9.9
11	.2643	3.8	9.1	21.4
12	.2549	3.9	30.3	-10.1
13	.2540	3.9	3.9	14.1
14	.2357	4.2	-13.6	-86.0
15	.2116	4.7	-39.6	41.0
16	.2097	4.8	-10.5	6.5
17	.1901	5.3	6.1	77.7
18	.1660	6.0	1.0	.7
19	.1559	6.4	26.6	-3.5
20	.1509	6.6	-114.7	.8
21	.1371	7.2	-22.2	-1.7
22	.1209	8.2	-39.2	-21.4
23	.1031	9.7	-31.7	29.9
24	.0956	10.5	-8.2	-36.6
25	.0903	11.1	3.5	1.7
26	.0817	12.2	-.9	.7
27	.0751	13.3	-32.3	-.6
28	.0686	14.6	24.2	2.5
29	.0645	15.5	10.3	-.5
30	.0626	16.0	11.0	-3.5
31	.0579	17.3	1.5	-.5
32	.0524	19.1	-.0	7.6
33	.0477	21.0	7.1	5.5
34	.0431	23.2	3.2	-4.2
35	.0431	23.2	.5	12.1
36	.0419	23.9	2.0	-5.2
37	.0392	25.5	.3	-.7
38	.0337	29.7	4.3	-.1
39	.0331	30.2	2.1	10.3
40	.0325	30.8	-3.5	8.1

TABLE 3.7-5

SUMMARY OF NATURAL FREQUENCIES AND PARTICIPATION

FACTORS FOR CONTAINMENT BUILDING

MODE	PERIOD (sec)	FREQUENCY (c/s)	PARTICIPATION FACTOR
1	.2865	3.5	-39.7
2	.0834	12.0	18.4
3	.0494	20.3	6.3
4	.0366	27.3	-8.2

TABLE 3.7-6

COMPARISON OF RESPONSES FOR TYPICAL

SHEAR WALLS FOR SSE CONDITION

WALL NUMBER	SHEAR DUE TO RESPONSE SPECTRUM METHOD OF ANALYSIS (kips)	SHEAR DUE TO TIME HISOTRY METHOD OF ANALYSIS (kips)
708134	20816	22348
708135	10028	9347
708136	7911	7434
708137	5725	5889
708138	10604	9977
708139	11891	10197
708140	10613	9887
708142	5725	5889
708145	10063	9014
708146	12689	11600

Layout of Walls

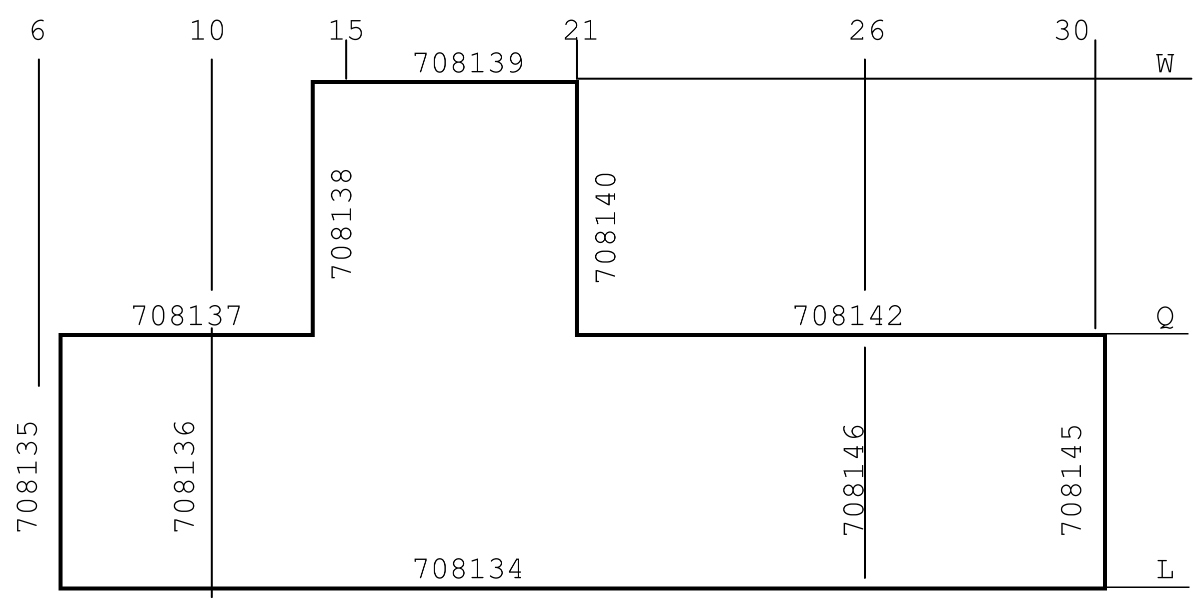

TABLE 3.7-7

BRAIDWOOD STATION
SEISMIC RESPONSE (SSE) FOR WALLS AND COLUMN ROW L
AUXILIARY-FUEL HANDLING BUILDING

ELEVATION (ft)	SHEAR (kips)	MOMENT (k-ft)
477	5431	43447
451	10373	303750
439	38237	752265
426	41385	1287695
401	55558	2664158
383	15454	2195645
367	19419	3235847
346	40482	3901814

TABLE 3.7-8

BRAIDWOOD STATION
SEISMIC RESPONSE (SSE) FOR WALLS AND COLUMN ROW Q
AUXILIARY-FUEL HANDLING BUILDING

ELEVATION (ft)	SHEAR (kips)	MOMENT (k-ft)
451	3183	82769
439	10640	209498
426	11829	362897
401	12748	676959
383	4272	744808
367	9620	907897
346	6598	1018056

TABLE 3.7-9

BRAIDWOOD STATION
SEISMIC RESPONSE (SSE) FOR WALLS AND COLUMN ROW 30
AUXILIARY-FUEL HANDLING BUILDING

ELEVATION (ft)	SHEAR (kips)	MOMENT (k-ft)
451 ft 0 in	3214	83570
439 ft 0 in	10992	207808
426 ft 0 in	11487	355556
401 ft 0 in	15361	729298
376 ft 5 in	11042	1037071

TABLE 3.7-10

SEISMIC RESPONSE (SSE) FOR CONTAINMENT BUILDING

ELEVATION (ft)	SHEAR (kips)	MOMENT (k-ft)
538	9882	467045
522	11588	702589
502	13283	962085
484	14308	1218187
466	15238	1485301
448	16085	1762123
436	16707	1953632
424	17160	2145458
414	17516	2310119
404	17788	2475429
394	18000	2642556
384	18146	2811127
377	18220	2880694

TABLE 3.7-11

BYRON STATION
SEISMIC RESPONSE (SSE) FOR WALLS AND COLUMN ROW L
AUXILIARY-FUEL HANDLING BUILDING

ELEVATION (ft)	SHEAR (kips)	MOMENT (k-ft)
477	2706	21646
451	5093	149645
439	15565	328238
426	16835	544536
401	20816	1045220
383	8784	1195894
367	8489	1342584
346	9687	1497007

TABLE 3.7-12

BYRON STATION
SEISMIC RESPONSE (SSE) FOR WALLS AND COLUMN ROW Q
AUXILIARY-FUEL HANDLING BUILDING

ELEVATION (ft)	SHEAR (kips)	MOMENT (k-ft)
451	1739	45221
439	5612	111813
426	6243	192590
401	5725	331660
383	1958	565962
367	3628	429779
346	1369	451838

TABLE 3.7-13

BYRON STATION
SEISMIC RESPONSE (SSE) FOR WALLS AND COLUMN ROW 30
AUXILIARY-FUEL HANDLING BUILDING

ELEVATION (ft)	SHEAR (kips)	MOMENT (k-ft)
451 ft 0 in	3401	62419
439 ft 0 in	8286	159401
426 ft 0 in	8721	271789
401 ft 0 in	10597	528867
376 ft 6 in	9040	844375

TABLE 3.7-14

MASS RATIOS AND FREQUENCY RATIOS
FOR LARGE DECOUPLED SUBSYSTEMS

SUBSYSTEM	R_m	R_f
Containment polar crane	0.10	0.55
Steam generator:‡		
Upper lateral support	0.09	1.30
Lower lateral support	0.05	1.56
RPV	0.40	2.14
Fuel handling crane	0.04	0.58

* R_m and R_f values are only applicable to Unit 2. Also, RCS loop analysis of Unit 1 by Framatome Technologies utilizes coupled analysis.

TABLE 3.7-15

CATEGORY II STRUCTURES TO BE DESIGNED

FOR CATEGORY I LOADS

CATEGORY II STRUCTURES	INTERCONNECTED CATEGORY I STRUCTURES	CRITICAL CATEGORY II ITEMS DESIGNED FOR CATEGORY I LOADS	LOADS
1. Turbine Building	Auxiliary Building	1. Concrete floors 2. Columns 3. Crane girders 4. Roof girders 5. Vertical and horizontal bracing 6. Roof trusses 7. Purlins required for lateral support of roof girders 8. Tie rods 9. Connections to Auxiliary Bldg. at L row 10. Shear walls 11. Mat	SSE and Tornado
2. Containment Building Buttress and Dome Enclosure	Containment Building	1. Embedments to the Containment	SSE and Tornado
3. Train Shed	Fuel Handling Building	See Note 1	See Note 1
4. Security Booth Walkway and Walkway Access Stair Tower (Byron only)	Essential Service Water Cooling Tower	See Note 2	See Note 2

NOTES: 1. Failure of the train shed under Safety Category I loadings will have no detrimental effect on adjacent Safety Category I structures. The two roof girders from the train shed that frame into the Fuel Handling Building will be permitted to fail. Consequences of

TABLE 3.7-15

CATEGORY II STRUCTURES TO BE DESIGNED

FOR CATEGORY I LOADS

their failure will be less critical than the failure due to missiles identified in Subsection 3.5.1.4.

2. Failure of the walkway and walkway access stair tower under Safety Category I loadings will have no detrimental effect on adjacent Safety Category I structures (Essential Service Water Cooling Tower). Consequences of their failure will be less critical than the failure due to missiles identified in Subsection 3.5.1.4.

ATTACHMENT 3.7A

REEVALUATION AND VALIDATION OF THE BYRON/BRAIDWOOD SEISMIC DESIGN BASIS

ATTACHMENT 3.7A

TABLE OF CONTENTS

ATTACHMENT 3.7A

3.7A.1 INTRODUCTION

This section describes the Questions and Responses documenting the historical evaluation of the Byron/Braidwood seismic design. It also validates the current seismic design basis which uses the deconvolution analysis to generate response spectra at the foundation level.

The Construction Permit for Byron and Braidwood was based on OBE and SSE levels that were 0.09g and 0.20g, respectively, at the foundation. The foundation level spectrum shape was obtained through a deconvolution analysis with the wide band Regulatory Guide 1.60 spectra defined at grade level. Mean soil properties were used in the deconvolution analysis.

Subsequent to the issuance of the Construction Permit and during the review of the FSAR for an Operating License, several aspects of the seismic design were reevaluated. The most notable seismic concern was the reevaluation of the Byron/Braidwood design using the Regulatory Guide 1.60 spectra without the application of a deconvolution analysis. The Marble Hill design basis, which was a replicate of the Byron/Braidwood design but had implemented Regulatory Guide 1.60 spectra, was used for comparison in the design assessment.

The reevaluation of the Byron/Braidwood seismic design took place over several years. It included responses to many NRC questions during the review of the FSAR. In order to maintain the chronological perspective of the individual assessments requested by each NRC question, the individual questions and their specific responses have been kept intact in this attachment. The five remaining sections of this attachment have been organized according to topics. NRC questions with Applicant responses for each specific topic have been grouped into the appropriate section. All tables and figures associated with each response have been left intact. A brief description to each question and response is also provided.

3.7A.2 BYRON/BRAIDWOOD GENERAL SEISMIC REEVALUATION

During the review of the FSAR, the NRC disallowed the use of the deconvolution analysis to generate the response spectra at the foundation that had been approved during the PSAR review.

Question 130.6 requested a seismic analysis based on Regulatory Guide (R.G.) 1.60 spectra. The response provides a comparison of the Byron/Braidwood design basis and the R.G. 1.60 spectra. It also provides a historical review of the seismic issue and a justification for areas of nonconformance to R.G. 1.60 spectra. Question 130.60a requested a comparison of the structural responses of Category I structures and the design parameters used as the design basis with those which would have been obtained using the R.G. 1.60 response spectra. The response spectra provides a reassessment of the structures in accordance with the question.

The Response to Question 130.6a references responses to Questions 110.68 and 110.70. These responses are provided in this section for completeness.

QUESTION 130.6

"The seismic analysis was performed by the response spectrum method. However, the response spectra at the foundation level generated by the synthetic time history have displayed a significant dip over a large range of frequencies as compared with the design response spectra in R.G. 1.60 (Figures 3.7-1 through 3.7-40). The use of such unconservative response spectra is unacceptable to the staff. The deconvolution procedure as described in the FSAR is not appropriate for the Byron/Braidwood sites due to the shallow soil overburden (16 ft to 38 ft) on bedrock. Therefore, it is requested that the analysis shall be based on R.G. 1.60 free field surface design response spectra applied at the foundation level and the design time history shall generate response spectra envelope the R.G. 1.60 design response spectra at the foundation level."

RESPONSE

1. Introduction

The seismic design process involves various steps. These include (1) determination of g level; (2) specification of the shape of the design response spectra and design time-history; (3) analysis to obtain design response spectra at the base mat elevation; (4) modeling of the structure; (5) calculation of structural response and floor response spectra; (6) specification of load factors, load combinations, factors of safety, and allowable stresses; and (7) designs of components to the combined effects of seismic and other loads. The overall safety of the plant is a function of the design parameters assumed at each stage. The margin in design for the various stages may vary, but good engineering design requires that the overall design be conservative.

In this response, the background to the present design bases is presented to highlight the bases of the present design criteria. The conservatism associated with the design "g" levels, with the design response spectrum when the effect of earthquake wave passage is considered, with the use of elastic analysis and low damping values, and the use of minimum yield/ultimate strength for design are quantified to show that any reduction in response due to the use of the deconvolution analysis is more than compensated for by the margins in design introduced by the conservative definitions of other seismic design parameters. To comply with the NRC request, the structural responses and floor response spectra obtained by applying the Regulatory Guide 1.60 spectra at the foundation levels are also presented.

Based on a composite evaluation of the above information it is concluded that the present design of Byron/Braidwood is conservative.

2. Background to Present Design Criteria

In the present Byron/Braidwood design, a wide band Regulatory Guide 1.60 spectrum is specified at the grade elevation. One dimensional deconvolution analysis is used to compute the foundation elevation spectra. The structural response and the floor response spectra are computed using the deconvolved foundation level spectra.

In the PSAR, it was our evaluation that for the Byron and Braidwood sites a 0.06 g OBE and a 0.12 g SSE level are conservative design bases. The NRC staff stated in Question 2.5.63 that the OBE and SSE levels should be 0.1 g and 0.20 g, respectively. In the ensuing discussions with the NRC staff, it was agreed that the OBE and SSE spectra at the foundation elevation will have 0.09 g and 0.20 g rigid period accelerations, respectively. The foundation level spectrum shape was to be obtained through a SHAKE (Reference 1) deconvolution analysis with the wide band Regulatory Guide 1.60 spectra defined at the grade elevation. Consistent with the practice at that time (1974), mean soil properties were used in the deconvolution analysis. The PSAR was amended in November 1974 to reflect the above design bases. In December 1975, the construction permit was issued by the NRC. In September 1976, the NRC requested additional information, stating that:

> "The current NRC staff position is that when the design response spectra are defined for the free field and applied at the finished grade level of the site, the SHAKE computer program is acceptable for deconvolution analysis to obtain a time history at the base of the idealized soil profile provided that appropriate soil properties, and variations thereof, are used in the analysis.
>
> "In view of the uncertainty and variability of soil properties, the response spectra at the base of the soil-structure interaction system should envelop all response spectra of those deconvolved time histories within the range of variable soil properties, and should not be less than 60 percent of the free field surface spectra."

A reply to the above NRC concern was submitted on December 9, 1976 (Reference 2). In the reply it was stated:

> "We found that a variation in soil properties of ±20% and a strict adherence to the requirements of Standard Review Plan 3.7.1 that the foundation spectrum be no less than 60% of the surface spectrum at any point would cause an increase in the design forces for Category I structures. Most of the increase in forces is due to the rather arbitrary 60% limit and not due to the ±20% soil property variation.
>
> "There are several areas of conservatism in the seismic analysis for Byron/Braidwood. Areas such as the methods used for the determination of the maximum ground acceleration for the SSE and the OBE have a considerable amount of conservatism. The use of the wide band response spectrum and the corresponding synthetic time history that envelopes the spectrum is another factor which results in higher forces than actual. Various items in the modeling and analysis, such as lower damping values, three simultaneous spatial components of equal strength, not accounting for the traveling nature of seismic waves are all areas of conservatism which are built into the analysis.
>
> "We have also reviewed the conservatism in many of the assumptions and methodology used in the design, compared the actual material strength obtained in the field with the design strength used, and have concluded that the increases in the design forces are more than compensated by these conservatisms. Therefore, the overall safety margin of the stations is not affected.

Since our response was accepted by the NRC and no further information was requested, the design and construction of the Byron/Braidwood plant proceeded, based on the design criteria as contained in the PSAR. At the present time, the structural design and construction of the plant structures are complete. The remaining electrical and mechanical components and equipment are either on site or at advanced stages of fabrication and qualification.

It is evident from the above that the present Byron/ Braidwood design criteria were appropriately judged to be conservative by the NRC staff in 1974 and again in 1976. It is also clear that the judgment was based on an overall evaluation of the seismic design process. We feel that none of the parameters have changed since then to alter this conclusion.

3. <u>Conservative Selection of Design Earthquakes</u>

For the selection of design earthquakes, the maximum historical random earthquake of the entire seismotectonic province is assumed to occur at the site even if there is no history of seismic activity in the site vicinity. This is a very conservative assumption. In addition, the staff has required that a VII-VIII intensity earthquake be considered. In the case of the Byron/Braidwood sites, it is our conclusion that the maximum random earthquake should be of MM Intensity VII. Our reasons for this have been documented in detail in the Byron/Braidwood PSAR. Using MM Intensity VII, the Trifunac & Brady relationship gives a maximum acceleration of 0.13 g. A more recent NRC-sponsored study (Reference 3), performed by Computer Services Corporation (CSC), which was based on much more exhaustive data, yields an acceleration of 0.085 g for the United States sites. Even for an Intensity VII-VIII earthquake, the CSC study gives only an 0.11 g level for United States sites. On the basis of the above reasoning, the value of 0.20 g for SSE is higher than necessary, and a value of 0.12 g, as proposed during the PSAR review stage, is more than appropriate.

For the OBE, the design acceleration is 0.09 g. The bases used for this acceleration are extremely conservative when compared with the more recent projects. A seismic risk analysis for the Byron/Braidwood Stations showed that the return period for an Intensity VI earthquake would be 2150 years. This return period is high when compared to the return period used in the Koshkonong (1000 years) project, which is more recent. The return periods for Intensities IV and V at the Byron/Braidwood site would be 322 years and 833 years, respectively. These return periods are more comparable to the Koshkonong project. Thus, a more appropriate OBE intensity for Byron/Braidwood would be IV or at most V.

The acceleration values obtained from Trifunac & Brady and the CSC relationships for these Intensities are shown in Table Q130.6-1.

It can be concluded from Table Q130.6-1 that 0.06 g is a more reasonable acceleration level for the OBE for the Byron/Braidwood design. The 0.06 g level for OBE was proposed in the initial PSAR submittal.

Based on the above discussion, levels of 0.06 g for OBE and 0.12 g for SSE can be considered appropriate but conservative design bases. Figures Q130.6-1 and Q130.6-2 provide a comparison of the Byron/Braidwood deconvolved design spectra to the 0.06 g OBE and 0.12 g SSE Regulatory Guide 1.60 spectra for horizontal and vertical motions, respectively. The comparison shows that the Byron/Braidwood design bases envelop the Regulatory Guide spectra. Thus, the seismic forces obtained by applying a 0.06 g OBE and 0.12 g SSE Regulatory Guide 1.60 spectra would be smaller than those presently considered in the design for the containment, containment internal structures, and the auxiliary fuel handling building complex. Table Q130.6-2 shows the comparison for overturning moment and base shear force for the containment shell structure. The total shear force and overturning moment from the regulatory guide input are lower than those used for design. Figures Q130.6-9 through Q130.6-56 provide a comparison of the present design floor response spectra with those obtained using 0.06 g OBE and 0.12 g SSE Regulatory Guide 1.60 spectra. OBE spectra are for 1% oscillator damping, whereas SSE spectra are presented for 2% oscillator damping. The comparison is provided for both horizontal and vertical responses in the containment and the auxiliary building complex. Table Q130.6-3 lists the location and elevations for the spectra comparison. The comparison shows that the present design spectra are higher and, except for a few isolated instances, they envelop those obtained using the Regulatory Guide 1.60 spectra.

From the above discussions, it can be concluded that the Byron/Braidwood design is conservative. Any reduction in response due to deconvolution is more than compensated by the extremely conservative specification of the OBE and SSE levels.

4. Design Spectra Considering Effect of Foundation Size

The observation has frequently been made that structures on large foundations appear to respond with less intensity to earthquakes than do smaller structures and, more specifically, than does free field instrumentation. Researchers who have attempted to give a rational explanation for this behavior have concluded that during an earthquake, not all particles under a large building foundation describe the same motion simultaneously; thus the relatively rigid structure-foundation system tends to average the ground motion, resulting in a reduced effective input excitation and consequently less damage.

In a report to the NRC dated September 1976 and entitled "A Rationale for Development of Design Spectra for Diablo Canyon Reactor Facility," Dr. Newmark investigated the effect of foundation size on design spectrum (Reference 4). His recommended reduced effective inputs, and the earthquake wave transit times, τ, for various structures, are given in Table Q130.6-4.

It is recognized that the reduced effective spectra were developed by Dr. Newmark for the Diablo Canyon site and for a near-field earthquake on a rock site. However, it is our evaluation that the concept and the methodology proposed by Dr. Newmark are also applicable to the Byron/Braidwood seismic design. Our evaluation is based on a comparison of the seismic design parameters for the two plants. The Byron/Braidwood and Diablo Canyon building sizes and rock site conditions are comparable; wave transit times of 0.04 second for the containment and 0.067 second for the auxiliary-turbine building complex are appropriate. For the Byron/Braidwood site, the maximum historical earthquake of the entire seismotectonic province is assumed to occur at the site. Thus, the earthquake is, by definition, a near-field earthquake and the reduced effective spectra due to wave transit times can be constructed using the reduction factors recommended by Dr. Newmark. It is possible that ground motions at the Byron/Braidwood site may occur due to seismic activity at distances greater than those considered for the reduced effective spectra at the Diablo Canyon plant. However, the ground motions in such an event are likely to be smaller than the design basis ground motions and would not control the design.

Figures Q130.6-3 and Q130.6-4 present a comparison of the 0.09 OBE and 0.20 SSE Regulatory Guide 1.60 spectra, the deconvolved Byron/Braidwood design bases spectra, and the reduced effective spectra (denoted as "τ" spectra on the figures) for the containment and the auxiliary-turbine building complex for OBE and SSE, respectively. The hatched area shows the frequency region where the Byron/Braidwood spectra are exceeded by the reduced effective spectra. It can be observed that for the auxiliary-turbine building complex the Byron/Braidwood design spectra envelop the reduced effective spectra. For the containment building, the Byron/Braidwood spectra do not fully envelop the reduced effective spectra; however, at the predominant structural period of 0.287 seconds, the Byron/Braidwood spectra are higher. Thus, it can be concluded that when the effect of foundation size on the design spectra is considered, the present Byron/Braidwood seismic design is conservative.

5. <u>Conservatism in Analysis</u>

As in the seismic analysis of any complex structure, several conservative assumptions are used in the Byron/Braidwood design. Many of these assumptions are regulatory requirements; others were necessary to simplify the analysis. These assumptions do provide additional margins of safety. They are briefly described below.

The time-history used for Byron/Braidwood (B/B) deconvolution analysis has a response spectrum which is 5% to 20% higher than the Regulatory Guide 1.60 spectrum in the range of significant structural frequencies (4-12 Hz). This is shown in Figures Q130.6-5 and Q130.6-6 for 4% and 7% damping, respectively.

In the B/B design, the two horizontal and the vertical simultaneous components of earthquake motion are assumed to have the same maximum accelerations as required by Regulatory Guide 1.60. However, recorded earthquake motions show that the three components do not have the same accelerations. Studies presented in References 5 and 6 indicate that a 1.0:0.87:0.70 ratio for the three components is more appropriate. Dr. Newmark, in a report (Reference 7) prepared for the NRC, recommends that the vertical acceleration be 2/3 of the horizontal.

In the seismic modeling of the containment and the auxiliary/fuel handling/turbine building below the grade level, the effect of the soil or rock on the sides of the exterior walls was neglected in computing the responses. However, the walls were designed for dynamic earth pressures. Consideration of the side soil/rock effect would tend to reduce the overturning moment on the shear walls and the foundation mat.

The maximum seismic response of the structure is strongly influenced by the energy absorption characteristics or damping of the structure. Low values of damping result in higher responses and are thus conservative. In the B/B design the damping values recommended in Regulatory Guide 1.61 were used. Newmark and Hall (Reference 8) in their recent report NUREG-0098, prepared for the NRC Systematic Evaluation Program, have recommended higher and more realistic damping values. A comparison of the Regulatory Guide 1.61 damping values and the NUREG-0098 damping values is provided in Table Q130.6-5. Note that the NUREG damping values are higher and thus would lead to lower responses.

In considering the response of nuclear power plant structures to seismic motions, one must take into account the implications of various levels of damage short of impairment of the safety, and definitely short of the collapse of the structure. Some elements of plant structures must remain elastic or nearly elastic in order to perform their allocated safety function. However, in many instances, a purely linear elastic analysis may be unreasonably conservative when one considers that even up to the near yieldpoint range there are nonlinearities of amounts sufficient to reduce the required design levels significantly. Moreover, limited yielding of a structure may reduce the response of equipment located in the structure below those levels of response that would be excited were the structure to remain elastic. The concept of ductility factors (Reference 9) is a simple but effective means of accounting for small excursions into the inelastic range. A ductility of 1.3 for concrete and 3.0 for steel members was proposed for the Diablo Canyon Power Plant docket and for the NRC Systematic Evaluation Program Seismic Criteria (Reference

9). Use of these ductility factors on B/B would result in a 10%-50% reduction in design response computed using an elastic analysis. Based on the above discussion, it can be concluded that there are several areas of major conservatism in the B/B seismic design, and due consideration should be given to these factors when reviewing the B/B seismic design.

6. Conservatism in Material Strength

   The compressive strength of concrete obtained from the cylinder tests exceeds the value used in design. The actual strength for the reinforcement steel and structural steel also exceeds those used in design. Table Q130.6-6 compares the values used in the design to those obtained by tests. The actual strength is the mean value obtained from the concrete cylinder test report summaries and a sampling of certified material test reports for reinforcement and structural steel for the B/B project. It shows that the actual strength exceeds the design strength by 12% to 50%, adding proportionally to the design margins.

7. Response Due to 0.09 g OBE and 0.20 g SSE Regulatory Guide 1.60 Spectrum

   To comply with the NRC request, forces, moments, and floor response spectra obtained by applying 0.09 g OBE and 0.20 g SSE Regulatory Guide 1.60 spectra at the foundation level are compared to the corresponding B/B design forces, moments, and floor response spectra. The comparison is provided for the containment and the auxiliary building. The forces, in many instances, increase when the Regulatory Guide 1.60 spectra are applied. However, these increased forces should be judged against the conservatism in the B/B design as discussed previously.

   a. Containment Forces

      Table Q130.6-7 presents a comparison between forces and moments for the containment shell and base mat between current B/B values and those obtained by applying the Regulatory Guide 1.60 spectra at the base level. The base mat is designed to resist overturning moments from the containment shell and the containment internal structures. The magnitude of these moments affects the area of the base mat which is uplifted (see Figure Q130.6-7) and the design moments in the mat (see Figure Q130.6-8). The increased overturning results in the engagement of the reactor cavity as a rotational key producing large meridional membrane forces, whereas for the current B/B design these forces are negligible. Note that in the seismic modeling of the containment below the grade level, the effect of the soil or rock on the sides of external walls was neglected in computing the responses. Consideration of the side/rock effect would tend to reduce the overturning moments and meridional membrane forces.

b. Containment Internal Structures

1) Reinforced Concrete

A review of the internal concrete structures including the refueling pool walls, primary shield wall, secondary shield wall, and enclosure walls was made. The seismic design is controlled by forces generated from horizontal SSE spectra. The lowest horizontal frequency for the internal structures model is 9.8 cps. Figure Q130.6-81 shows that for 9.8 cps and higher frequencies, the B/B design spectra envelop the Regulatory Guide 1.60 spectra. The resulting forces for the Regulatory Guide 1.60 spectra are consequently lower than those used in the present B/B design.

2) Structural Steel

A summary of the structural steel beams showing the percent increase in force for OBE and SSE conditions due to the application of Regulatory Guide 1.60 spectra is presented in Table Q130.6-8. The table is based on a representative sample comprised of all beams at elevation 426 feet 0 inch. A total of 108 beams were reviewed, out of an estimated 740 per unit. Note that the increase in forces in 100 of the 108 beams is less than 20%.

Structural steel columns are seismically designed by amplifying the permanent loads on the columns in proportion to the zero period acceleration of the wall response spectra at that elevation. The minimum values for g used for design are 0.5 and 0.9 for OBE and SSE, respectively. The maximum values for the zero period acceleration of the wall spectra at various elevations for the Regulatory Guide 1.60 foundation elevation definition are 0.26 for OBE and 0.42 for SSE. Comparing these g values shows that the forces are consequently larger for the present B/B design than for the Regulatory Guide 1.60 foundation elevation definition.

c. Auxiliary-Fuel Handling Building Complex

1) Reinforced Concrete

The areas of the base mat found to have increased forces as a result of Regulatory Guide 1.60 spectra are indicated by the cross-hatched areas shown in Figure Q130.6-8. The values shown are the percent increase in force over the design force.

The comparison of all the shear wall forces from the B/B design basis with those resulting from implementation of the Regulatory Guide 1.60 spectra was made. A summary of this comparison, showing the percent change in seismic force due to Regulatory Guide 1.60 spectra is shown in Table Q130.6-9. Note that for the SSE excitation, the increase in forces in 234 out of 272 shear walls is less than 20%.

The remaining concrete structural components, including columns, beams, and slabs, have no increase in seismic forces due to the implementation of Regulatory Guide 1.60 spectra. These components are essentially rigid and are located at elevation 401 feet 0 inch or below. A comparison of the vertical spectra in Figures Q130.6-59 and Q130.6-83 shows that the B/B spectra envelop the Regulatory Guide 1.60 spectra.

2) Structural Steel

A summary of the structural steel beams for the auxiliary-fuel handling building complex showing the percent increase in force due to the application of Regulatory Guide 1.60 spectra is presented in Table Q130.6-10. A representative sample comprised of those beams at elevations 451 feet 0 inch and 426 feet 0 inch has been reviewed. The 230 beams reviewed are representative of all beams. Note that for the SSE excitation the increase in forces in 224 of the 230 beams is less than 20%.

The criteria for the design of the structural steel columns in the auxiliary-fuel handling complex are the same as those for the containment building. The minimum value for g used for design is 0.26 for OBE and 0.68 for SSE. The maximum value for various elevations, based on a Regulatory Guide 1.60 spectra foundation elevation definition is 0.23 for OBE and 0.46

for SSE. Comparing these "g" values shows that the forces are consequently larger for the present B/B design than for Regulatory Guide 1.60.

d. Floor Response Spectra

Figures Q130.6-57 through Q130.6-104 provide a comparison of the present design floor response spectra with those obtained by using 0.09 g OBE and 0.20 g SSE Regulatory Guide 1.60 spectra at the foundation elevation. The comparison is provided for both horizontal and vertical response for the containment and auxiliary building. Table Q130.6-11 lists the locations and elevations for the spectra comparison.

8. Summary

In the present Byron/Braidwood design, a wide band Regulatory Guide 1.60 spectrum is specified at the grade elevation. One-dimensional deconvolution analysis is used to compute the foundation elevation spectra. The structural response and floor response spectra are computed using the deconvolved foundation level spectra. In the question, the NRC staff states that the deconvolution procedure is not appropriate for the Byron/Braidwood sites and that analysis should be based on a Regulatory Guide 1.60 spectra applied at the foundation elevation.

In this response, the background to the present design is presented to highlight the bases of the present criteria. The conservatism associated with the design "g" levels, with the design response spectrum when the effect of earthquake wave passage is considered, with the use of elastic analysis and low damping values, and the use of minimum yield/ultimate strength for design are quantified to show that any reduction in response due to the use of the deconvolution analysis is more than compensated for by the margins in design introduced by the conservative definition of other seismic design parameters. To comply with the NRC request, the structural responses and floor response spectra obtained by applying the Regulatory Guide 1.60 spectra at the foundation levels are also presented.

Based on the composite elevation of the above information, it is concluded that the present design of Byron/Braidwood is conservative.

References

1. P. B. Schnabel, J. Lysmer, and H. B. Seed, "SHAKE - A Computer Program for Earthquake Response Analysis of Horizontally-Layered Sites," Earthquake Engineering Research Center, Report No. EERC 72-12, University of California, Berkeley, 1972.

2. Submittal to the NRC dated December 9, 1976, entitled "Byron and Braidwood Stations Units 1&2 - Additional Information on Seismic Design Analysis," NRC Dockets 50-454/455 and 50-456/457.

3. J. R. Murphy and L. J. O'Brien, "Analysis of a Worldwide Strong Motion Data Sample to Develop an Improved Correlation Between Peak Acceleration, Seismic Intensity, and Other Physical Parameters," NUREG-0402, report prepared for the NRC by Computer Services Corporation, January 1978.

4. N. M. Newmark, "A Rationale for Development of Design Spectra for Diablo Canyon Reactor Facility," report to the USNRC, September 1976.

5. A. S. Bartu, "Discussion on Seismic Design Spectra for Nuclear Power Plants," Journal of the Power Division, ASCE, December 1974.

6. J. Penzien and M. Watabe, "Characteristics of 3-Dimensional Earthquake Ground Motions," Earthquake Engineering and Structural Dynamics, Vol. 3, 1975.

7. W. J. Hall, B. Mohraz, and N. M. Newmark, "Statistical Studies of Vertical and Horizontal Earthquake Spectra," Report NUREG-0003, prepared for USNRC, January 1976.

8. N. M. Newmark and W. J. Hall, "Development of Criteria for Seismic Review of Selected Nuclear Power Plant," Report NUREG/CR-0098, prepared for NRC, May 1978.

9. N. M. Newmark and E. Rosenblueth, "Fundamentals of Earthquake Engineering," Prentice-Hall, Inc., 1971.

10. NRC Summary of Meeting held February 4, 1977 to discuss the Diablo Canyon Seismic Design Re-evaluation.

TABLE Q130.6-1

ACCELERATION-INTENSITY RELATIONSHIP

INTENSITY	TRIFUNAC & BRADY	CSC
VI	0.065 g	0.05 g
V	0.0325 g	0.029 g
IV	0.0165 g	0.0165 g

TABLE Q130.6-2

COMPARISON OF CURRENT B/B CONTAINMENT

FORCES WITH 0.12 g REGULATORY GUIDE 1.60 SSE

ITEM	FORCE OR MOMENT (SSE) B/B DESIGN	NRC REGULATORY GUIDE 1.60 (0.12 g)
Total overturning moment at base of shell	$4,540,000^{ft-k}$	$3,156,000^{ft-k}$
Total shear at base of shell	$26,500^{k}$	$18,420^{k}$

TABLE Q130.6-3

LOCATIONS FOR SPECTRA COMPARISON -

B/B DESIGN VS. 0.06 g OBE AND 0.12 g SSE

BUILDING	ELEVATION BYRON/BRAIDWOOD	EARTHQUAKE	DIRECTION	FIGURE NUMBER
Auxiliary & Containment	330;374	OBE	Horizontal EW;NS	Q130.6-9
Auxiliary & Containment	330;374	OBE	Vertical	Q130.6-10
Auxiliary (wall)	346;364 383;401	OBE	Vertical	Q130.6-11
Auxiliary (slab)	346;364 383;401	OBE	Vertical	Q130.6-12
Auxiliary	401	OBE	Horizontal EW	Q130.6-13
Auxiliary	401	OBE	Horizontal NS	Q130.6-14
Auxiliary;Turbine; Heater Bay	426	OBE	Horizontal NS	Q130.6-15
Auxiliary;Turbine; Heater Bay	426	OBE	Horizontal EW	Q130.6-16
Auxiliary (wall)	426;439 451	OBE	Vertical	Q130.6-17
Auxiliary (slab)	426;451 439	OBE	Vertical	Q130.6-18

TABLE Q130.6-3 (Cont'd)

BUILDING	ELEVATION BYRON/BRAIDWOOD	EARTHQUAKE	DIRECTION	FIGURE NUMBER
Auxiliary;Turbine Heater Bay	451	OBE	Horizontal NS	Q130.6-19
Auxiliary;Turbine Heater Bay	451	OBE	Horizontal EW	Q130.6-20
Auxiliary	477	OBE	Horizontal NS	Q130.6-21
Auxiliary	477	OBE	Horizontal EW	Q130.6-22
Auxiliary (slab)	467;477	OBE	Vertical	Q130.6-23
Auxiliary (wall)	467;477 473;485	OBE	Vertical	Q130.6-24
Containment	424;436	OBE	Vertical	Q130.6-25
Containment	424;436	OBE	Horizontal NS	Q130.6-26
Containment	496	OBE	Horizontal NS;EW	Q130.6-27
Containment	496	OBE	Vertical	Q130.6-28
Containment Inner Structure	426	OBE	Horizontal NS	Q130.6-29
Containment Inner Structure	426	OBE	Horizontal EW	Q130.6-30
Containment Inner Structure (wall)	412;426	OBE	Vertical	Q130.6-31

TABLE Q130.6-3 (Cont'd)

BUILDING	ELEVATION BYRON/BRAIDWOOD	EARTHQUAKE	DIRECTION	FIGURE NUMBER
Containment Inner Structure (slab)	390;401 412;426	OBE	Vertical	Q130.6-32
Auxiliary & Containment	330;374	SSE	Horizontal	Q130.6-33
Auxiliary & Containment	330;374	SSE	Vertical	Q130.6-34
Auxiliary (wall)	346;383 364;401	SSE	Vertical	Q130.6-35
Auxiliary (slab)	346;383 364;401	SSE	Vertical	Q130.6-36
Auxiliary	401	SSE	Horizontal NS	Q130.6-37
Auxiliary	401	SSE	Horizontal EW	Q130.6-38
Auxiliary;Turbine Heater Bay	426	SSE	Horizontal NS	Q130.6-39
Auxiliary;Turbine Heater Bay	426	SSE	Horizontal EW	Q130.6-40
Auxiliary (wall)	426;439 451	SSE	Vertical	Q130.6-41
Auxiliary (slab)	426;439 451	SSE	Vertical	Q130.6-42

TABLE Q130.6-3 (Cont'd)

BUILDING	ELEVATION BYRON/BRAIDWOOD	EARTHQUAKE	DIRECTION	FIGURE NUMBER
Auxiliary;Turbine Heater Bay	451	SSE	Horizontal NS	Q130.6-43
Auxiliary;Turbine Heater Bay	451	SSE	Horizontal EW	Q130.6-44
Auxiliary (wall)	467;477 473;485	SSE	Vertical	Q130.6-45
Auxiliary (slab)	467;477	SSE	Vertical	Q130.6-46
Auxiliary	477	SSE	Horizontal NS	Q130.6-47
Auxiliary	477	SSE	Horizontal EW	Q130.6-48
Containment	424;436	SSE	Horizontal NS;EW	Q130.6-49
Containment (wall)	424;436	SSE	Vertical	Q130.6-50
Containment	496	SSE	Horizontal NS;EW	Q130.6-51
Containment (wall)	496	SSE	Vertical	Q130.6-52
Containment Inner Structure	426	SSE	Horizontal NS	Q130.6-53
Containment Inner Structure	426	SSE	Horizontal EW	Q130.6-54
Containment Inner Structure (wall)	412;426	SSE	Vertical	Q130.6-55
Containment Inner Structure (slab)	390;412 401;426	SSE	Vertical	Q130.6-56

TABLE Q130.6-4

EARTHQUAKE WAVE TRANSIT TIME AND

PEAK GROUND ACCELERATION

STRUCTURE	τ (sec)	PEAK GROUND ACCELERATION (g)	REDUCTION FACTOR
Small Structures	0.00	0.75	1.00
Containments	0.04	0.60	0.80
Auxiliary Building	0.052	0.55	0.73
Turbine Building	0.067	0.50	0.67

TABLE Q130-6.5

COMPARISON OF REGULATORY GUIDE 1.61

AND NUREG-0098 DAMPING VALUES

STRUCTURE OR COMPONENT	REGULATORY GUIDE 1.61 DAMPING FOR SSE	NUREG-0098 DAMPING AT OR JUST BELOW YIELD
Piping	2-3	2 to 3
Welded Steel	4	5 to 7
Prestressed Concrete		
(a) Without complete loss in prestress	5	5 to 7
(b) With no prestress left	7	7 to 10
Reinforced Concrete	7	7 to 10
Bolted Steel Structures	7	10 to 15

TABLE Q130.6-6

MATERIAL STRENGTH

MATERIAL	DESIGN STRENGTH (psi)	ACTUAL MEAN STRENGTH (psi)
Concrete (f'_c)	5,500 3,500	6,935 5,265
Reinforcement (fy)	60,000	67,000
Structural Steel (fy)	36,000 50,000	43,200 56,000

TABLE Q130.6-7

COMPARISON OF B/B CONTAINMENT DESIGN FORCES

TO THOSE FROM REGULATORY GUIDE 1.60 SPECTRA

DESCRIPTION	B/B (SSE)	REGULATORY GUIDE 1.60 (0.2 g)
Total overturning moment at base of shell	$4,540,000^{ft-k}$	$5,260,000^{ft-k}$
Total shear at base of shell	$26,500^{k}$	$30,700^{k}$
Net tensile membrane force in shell	27 k/ft	72 k/ft
Bending moment in base mat	$6,650^{ft-k/ft}$	$9,513^{ft-k/ft}$
Net membrane tensile force in reactor cavity wall	NA	1,335 k/ft

TABLE Q130.6-8

CONTAINMENT BUILDING STRUCTURAL BEAMS COMPARISON OF FORCES

BETWEEN B/B DESIGN BASIS FORCE AND REGULATORY GUIDE 1.60

% CHANGE IN DESIGN BASIS FORCE		NUMBER OF BEAMS	
		OBE	SSE
< 0		84	88
0 - 10		12	8
10 - 20		4	8
20 - 30		--	4
30 - 40		8	--
> 40		--	--
	TOTAL	108	108

NOTE

1. All 108 beams reviewed for elevation 426 feet 0 inch.

2. % increase does not necessarily reflect a state of stress in the beam.

TABLE Q130.6-9

AUXILIARY BUILDING-FUEL HANDLING BUILDING COMPLEX

SHEAR WALLS - COMPARISON BETWEEN B/B DESIGN BASIS

SEISMIC FORCES AND REGULATORY GUIDE 1.60

% CHANGE IN DESIGN BASIS FORCES		NUMBER OF SPRINGS OBE	NUMBER OF SPRINGS SSE
< 0		122	153
0 - 10		25	51
10 - 20		19	30
20 - 30		8	7
30 - 40		50	8
40 - 50		33	6
> 50		15	17
	TOTAL	272	272

NOTE

Percent increase does not necessarily reflect a state of stress in the wall.

TABLE Q130.6-10

AUXILIARY-FUEL HANDLING BUILDING COMPLEX STRUCTURAL

STEEL BEAMS - COMPARISON OF FORCES BETWEEN B/B DESIGN

BASIS FORCES AND REGULATORY GUIDE 1.60

% CHANGE IN DESIGN BASIS FORCES		NUMBER OF BEAMS	
		OBE	SSE
< 0		148	132
0 - 10		61	85
10 - 20		16	7
20 - 30		5	6
> 30		---	---
	TOTAL	230	230

NOTE

1. Beams located at elevation 426 feet 0 inch and 451 feet and 0 inch in the auxiliary building.

2. Percent increase does not necessarily reflect a state of stress in the beam.

TABLE Q130.6-11

LOCATIONS FOR SPECTRA COMPARISON -

B/B DESIGN VS. 0.09 g OBE and 0.20 g SSE

BUILDING	ELEVATION BYRON/BRAIDWOOD	EARTHQUAKE	DIRECTION	FIGURE NUMBER

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE Q130.6-11 (Cont'd)

BUILDING	ELEVATION BYRON/BRAIDWOOD	EARTHQUAKE	DIRECTION	FIGURE NUMBER

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE Q130.6-11 (Cont'd)

BUILDING	ELEVATION BYRON/BRAIDWOOD	EARTHQUAKE	DIRECTION	FIGURE NUMBER

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE Q130.6-11 (Cont'd)

BUILDING	ELEVATION BYRON/BRAIDWOOD	EARTHQUAKE	DIRECTION	FIGURE NUMBER

Security - Related Information Figure Withheld Under 10 CFR 2.390

QUESTION 130.6a*

"We have reviewed your response to Question 130.6 and we conclude that it is not adequate and not acceptable for the following reasons:

1) Selection of SSE and OBE Design Earthquakes

A considerable portion of our response is based on the conservatism you feel is in the SSE and OBE design earthquakes. You also presented arguments for reducing the design earthquake to those originally proposed in the PSAR (zero period acceleration of 0.06g for OBE and 0.12 for SSE).

These values have been subsequently increased to 0.09g and 0.20g respectively and rationale for the Regulatory staff position was stated in the Question 2.5.63. Furthermore, on the basis of further investigation, the staff came to the conclusion that the deconvolution procedures are not acceptable and that the Regulatory Guide 1.60 Design Response Spectra should be applied at the foundation level.

2) Effect of Foundation Size on Design Spectra

The response suggests that the design spectra can be reduced based on previous studies performed by Dr. Newmark for the Diablo Canyon Site. These studies justify reduced effective spectra as a result of considering the effect of foundation size on design spectrum. You pointed out in the response that the reduced effective spectra were developed for the specific site of the Diablo Canyon Plant and the basic reason for its acceptance was the postulated near-field earthquake. Since the Byron/Braidwood sites are located in an entirely different tectonic province, the argument which was used in the case of Diablo Canyon application cannot be applied to the subject sites.

3) Conservatism in Analysis

The staff does agree that three components of earthquake motion are probably not the same acceleration. The magnitude of the actual acceleration of each component should be found by means of a 3-dimensional analysis. It is the position of the staff that the response spectrum for vertical motion can be taken as 2/3 of the response spectrum for horizontal motion for the Western United States only. For other locations, the vertical response spectrum should be the same as that given in Regulatory Guide 1.60 (see enclosure).

*QUESTION 130.6a is a restated version of NRC QUESTION 130.6.

As far as the damping values are concerned, the referenced report, NUREG-CR 0098 was developed for a specific purpose of evaluating seismic risk of nuclear plants which are already operating. The damping values contained in that report cannot be applied in licensing of new plants.

The response claims that the elastic analysis which is used in design of new plants may be unreasonably conservative. In view of the fact that there is a lot of safety-related equipment which might produce catastrophic consequences in case of excessive deformation of supporting members, this position of the Regulatory staff is not unreasonable. You neglected to mention in your response that the referenced criteria for the Diablo Canyon plant stipulate that the ductility of 1.3 for concrete and 3 for steel are for turbine building and intake structure. These structures are non-Category I per se and the only reason that they have been reviewed by the staff was that in certain locations they are housing some safety-related equipment. Thus the criteria which are applicable to these two structures cannot be automatically applied to all Category I structures.

Evaluation of Structures using 0.09g OBE and 0.20g SSE Regulatory Guide 1.60 Spectrum

The evaluation of structures using the Regulatory criteria provided in the response have been reviewed. It is recognized that there is a general increase in the stress level of many structural members. We find, however, that without re-analysis of the affected structures and determination of the shear forces and moments imposed by the new loads the evaluation cannot be considered to be conclusive. You are, therefore, requested to compare the structural responses of Category I structures and the design parameters (bending moments, shears and axial loads) actually used in design of Byron/Braidwood plant with those which would have been obtained if the criteria stated in Question 130.06 were used."

RESPONSE

Introduction

The design of the Byron/Braidwood structures and components required for safe shutdown was reassessed during Regulatory Guide 1.60 input. This comprehensive review was made for the SSE condition (0.20 ZPA) to ensure adequate safety. The reassessment basis was agreed upon in the meeting at Bethesda on February 26, 1981 with the NRC, Commonwealth Edison Company, and Sargent & Lundy. The Marble Hill design response spectra was utilized to determine the effects on Byron/Braidwood of the Regulatory Guide 1.60 seismic event. Where structural elements were identified as unique on Marble Hill, they were reassessed for Byron/Braidwood using the actual material strengths.

The Marble Hill design is based on Regulatory Guide 1.60 spectra for an SSE event of 0.20g. The Marble Hill structures are a replicate of the Byron/Braidwood structures. Structural element properties as shown in Figures Q130.6a-1 and Q130.6a-2 are the same for Byron/Braidwood and Marble Hill. For these reasons, the Marble Hill design forces and spectra were used as the reassessment basis for Byron/Braidwood. Although there are a few unique structural features on Marble Hill, the effect of these features is negligible for purposes of seismic analysis.

Comparison of the Byron/Braidwood design basis model with the Marble Hill design basis model indicates that the overall geometry mass, stiffness, and dynamic characteristics are equivalent. The best measure of model equivalence is shown in the comparison spectra given in attached Addendum. Response spectra generated using the Byron/Braidwood design basis model with the same analysis parameters used in the Marble Hill analysis for comparison to the Marble Hill design basis response spectra is also provided in the attached Addendum.

The containment building seismic models used for the Marble Hill are identical to the models used for the Byron/Braidwood Stations.

Reassessment of Structures

The Byron/Braidwood structures were reviewed against the Marble Hill structures by comparing the respective design drawings. All unique elements due to the seismic design were identified.

The unique Byron/Braidwood elements were reviewed for the Marble Hill SSE loading conditions using average actual material strengths. For average actual material strengths, the values given in Table Q130.6a-1 were used.

The actual concrete strengths were obtained from the concrete cylinder test results reported by the onsite independent testing agency. The actual steel strengths were obtained from the certified material test results submitted by the material suppliers.

Containment Building

Areas where Marble Hill had unique features were identified and a comparison of the forces were tabulated for these areas. These areas include the base mat reinforcing, the vertical post-tensioning tendons, and the reinforcing steel at the main steam penetrations.

The Marble Hill and Byron/Braidwood design basis was based on conservative assumptions as is appropriate for initial design purposes.

Conservatisms were identified in the design basis analysis and the analysis was refined accordingly. The results of the assessment based on reanalysis show that stresses are below the design basis allowables.

Refinement to the analysis include the following:

1. The overall containment overturning moment, axial force and shear used in the assessment were obtained using a shell analysis rather than the beam analysis.

2. The stiffness of the elements used to represent the reactor cavity wall was refined to account for initial concrete cracking due to shrinkage, thermal effects and restraint.

Containment Internal Structures

Concrete

Review of the containment internal concrete structures reveals no unique concrete elements. Therefore, the stresses in the containment internal concrete structures are within design basis allowables.

Structural Steel Columns

Review of the containment structural steel columns reveal no unique column sections. Therefore, the stresses are within the design basis allowables.

Structural Steel Beams

Eighty-three beams of 740 total beams per unit have unique design due to SSE forces. Reassessing these beams using the average actual material strength indicates a stress level below yield strength.

Auxiliary/Fuel Handling Building

Base Mat

The auxiliary/fuel handling building base mat contains 17 unique finite elements which represent less than 1% of the total base mat area. (Refer to Figure Q130.6a-3 for unique areas.) Reassessing these elements using the actual average material strength for the concrete and reinforcing steel indicates a stress level below yield strength.

Shear Walls

Assessment of all shear walls based on Regulatory Guide 1.60 SSE spectra generated loads revealed 27 of the 65 total shear walls in the auxiliary/fuel handling building to have an increase in SSE force. Using the average actual material strengths for the reinforcing steel, the vertical reinforcement for all walls maintains a stress level less than the yield strength. The horizontal reinforcing steel also maintained a stress level less than the yield strength for all walls except for one. This single case has a stress level of 3.5% above the yield strength.

Concrete Beams, Slabs, and Piers

There are no unique concrete beams, slabs, and piers between Byron/Braidwood and Marble Hill. Therefore, all stresses are within the design basis allowables.

Structural Steel Columns

Of the 100 structural steel columns in the auxiliary/fuel handling building, there are 46 columns with at least one unique section. Using the average actual material strength, the stress level did not exceed the yield strength.

Structural Steel Beams

A review of the 3,400 structural steel beams in the auxiliary/fuel handling building revealed 242 unique beams. Reassessing these beams using the average material strengths indicates a stress level below yield strength.

Essential Service Water Cooling Tower (Byron)

The essential service water cooling tower is unique to Byron Station. The SSE forces were generated using Regulatory Guide 1.60 input. Reassessment using these forces indicates that stress levels do not exceed the design basis allowables.

Lake Screen House (Braidwood)

The lake screen house is unique to the Braidwood Station. Seismic forces were generated using Regulatory Guide 1.60 input. These forces were lower than the original SSE forces used in the design and, therefore, all stresses remain within the design basis allowables.

Electrical Raceways and Supports

The assessment of raceways indicated no unique design features when compared with Marble Hill.

Raceway supports on four different elevations in the auxiliary building were chosen for reassessment. These elevations have the largest increase in acceleration values in the frequency range of the raceway support. One hundred and twenty-two hangers, representing 15 different types, were chosen and analyzed using Marble Hill response spectra. The results of reanalysis are given in Table Q130.6a-2.

The six hangers with a ductility ratio (actual stress/yield stress) greater than one have overstresses in one of their members giving a ductility ratio ranging from 1.04 to 1.33 with an average of 1.17. These members will not collapse under an SSE event.

Conduit Supports

Six conduit supports are unique to Marble Hill. These supports were reassessed and it was determined that Byron/Braidwood supports are within the design basis allowables.

HVAC Ducts and Supports

The assessment of HVAC ducts indicate no unique design features when compared with Marble Hill.

HVAC duct supports at elevation 477 feet 0 inch in the auxiliary building were reassessed using the Marble Hill spectra. This elevation was chosen because it has the largest increase in acceleration values in the frequency range of the HVAC duct supports. Based on a refined analysis of 347 supports, it was found that stresses do not exceed the yield strength, except for twelve cases. These twelve supports will not collapse under an SSE event.

Piping Systems, Supports and In-line Valves

Refer to the responses to Questions 110.65, 110.66, 110.67, 110.68, and 110.70. These questions comprise the reassessment for the piping systems, supports, and in-line valves.

Equipment

Due to the different methodologies used in reassessing the safe shutdown equipment, the equipment was divided into two groups: NSSS equipment and balance of plant equipment.

NSSS Equipment

Introduction

A reassessment was performed to determine if sufficient margin exists in the NSSS supplied mechanical equipment to withstand a change in the Byron seismic design basis. The investigation was performed for both primary mechanical equipment and auxiliary mechanical equipment. A comparison was made of the generic seismic qualification levels for the equipment with the applicable Marble Hill response spectra. These comparisons indicate that all of the equipment has sufficient design margin to be qualified with no modification to the Marble Hill response spectra.

Generic Seismic Analysis of Mechanical Equipment

The NSSS supplied mechanical equipment that was included in this study is listed in Table Q130.6a-3. This list contains both primary and auxiliary equipment. All of this equipment was designed and analyzed for use in many plants. As part of the original design, seismic qualification loads were established that provide sufficient margin so that the equipment can be used in plants with both low and high seismic designs. The amount of margin in the design depends on the type of analysis used to qualify the component.

Margin for Primary Mechanical Equipment

Comparisons were made of the generic seismic response spectra with the Marble Hill response spectra for all of the primary equipment identified in Table Q130.6a-3. An example of the comparison for the reactor vessel internals is shown in Figure Q110.70-1. This figure is typical of all the primary components comparisons and represents the component with the smallest amount of margin between the generic response spectrum and the actual plant response spectrum. Based on these comparisons, it is clear that the response spectra in this reassessment does not significantly change the margin in the components generic design.

Margin for Auxiliary Mechanical Equipment

All of the auxiliary mechanical equipment was qualified using "g" levels. For the equipment listed in Table Q130.6a-3, the pumps were qualified for 2.1g along each of the two principal horizontal axes and 2g vertically. The tanks and heat exchangers were qualified for 1.5g in each of the two principal horizontal axes and 1.5g in the vertical axes.

All of the pumps included in this study were demonstrated to be rigid by either analysis or test and an investigation of the seismic accelerations for Marble Hill spectra above 33 Hz shows the highest acceleration to be 0.9g. This is well below the 2.1g acceleration used for qualification of the pumps and, therefore, all the pumps are acceptable.

The tanks and heat exchangers all have one modal frequency below 33 Hz. A comparison was made between the qualification levels for these components and the Marble Hill seismic levels. The comparison was made at the calculated component natural frequency with the applicable spectra for the location of the component in the plant. The highest seismic acceleration spectra was found to be 1.1g; this is within the design acceleration for the component of 1.5g. As a result of these comparisons, it was demonstrated that all of the auxiliary mechanical equipment have sufficient margin.

Summary

A reassessment of the seismic design margin of the NSSS supplied mechanical equipment was performed. The investigation compared the components seismic design levels to the Marble Hill levels. The reassessment demonstrated that all of the components have sufficient design margin.

BOP Equipment

The method used to reassess the BOP safe shutdown equipment is as follows:

1. Compare the seismic accelerations, from the appropriate Marble Hill spectra utilizing Regulatory Guide 1.61 damping values against the seismic accelerations used in the existing qualification report. In some cases, the seismic accelerations in the qualification reports exceeded the Marble Hill accelerations. This result can be attributed to one or more of the following:

   a. Byron spectra exceeding the Marble Hill spectra for the equipment frequency.

   b. The vendor performed a generic qualification.

   c. Vendor enveloped several spectra for equipment located on various elevations.

2. In cases where the Marble Hill levels exceeded the levels in the qualification report, stresses, deflections and margins were calculated for the Marble Hill accelerations by scaling the calculated seismic stresses and deflections by the required Marble Hill to Byron acceleration ratio.

3. Reference response to Question 110.68 for the effect of the Marble Hill spectra on safety-related small piping and instrumentation.

Summary

Table Q130.6a-4 contains the results. Regulatory Guide 1.61 damping values were used in the reanalysis reassessment. In addition, it should be noted that the resultant stresses were compared to the code allowable stress, not against the failure stress of the material. The results show the safe shutdown equipment can safely withstand the required seismic event.

RESPONSE TO NRC ENCLOSURE I "INFORMATION REQUIRED FOR REEVALUATION"

1. Item: It is requested that the applicant present the information on the respective stress components due to LOCA, if applicable, and SSE and the relation of each with the specified allowable values.

Response: The LOCA load condition is only applicable to the containment structures. Reassessment of the containment structure has shown that the stresses are within the design basis allowables. This reassessment included the combined effect of the LOCA and SSE load conditions.

2. Item: Document the following:

    a. The use of Marble Hill plant seismic analysis as the basis for comparison with the Byron/ Braidwood plant. In your discussion, provide a comparison of the mathematical models for key structures for the two plants which show dynamic parameters such as stiffness, periods, moduli of elasticity, Poisson's ratio and masses.

    b. Describe any difference between the two plants in terms of construction materials, quality control, construction techniques, etc.

    c. Describe the detailed procedures as to how the Marble Hill seismic responses will be used in computing stress levels for different load combinations of Byron/Braidwood structural members. In addition, describe the criteria for selection of members to be evaluated.

Response: a. The Marble Hill design is based on Regulatory Guide 1.60 spectra for an SSE event of 0.20g. The Marble Hill structures are a replicate of the Byron/Braidwood structures. For these reasons, the Marble Hill design forces and spectra were used as the reassessment basis for Byron/Braidwood. Although there are a few unique structural features on Marble Hill, the effect of these features is negligible for modeling purposes.

Comparison of the Byron/Braidwood design basis model with the Marble Hill design basis model indicates that the overall geometry, mass, stiffness, and dynamic characteristics are equivalent. The best measure of model equivalence is documented in the comparison spectra shown in attached Addendum. Response spectra were generated using the Byron/Braidwood design basis model with the same analysis parameters used in the Marble Hill analysis for comparison to the Marble Hill design response spectra. For completeness, the Byron/Braidwood design basis spectra is also provided in attached Addendum.

The Containment Building seismic models used for Marble Hill are identical to the models used for Byron/Braidwood Stations.

b. Equivalent material and construction specifications are used on the two plants.

c. The Byron/Braidwood and Marble Hill structures were compared using their respective design drawings. All unique elements due to seismic design were identified and reassessed using Marble Hill forces and Byron/Braidwood actual material strengths. All design basis loading combinations are the same for Byron/Braidwood and Marble Hill.

3. <u>Item</u>: Provide a comparison of the floor response spectra used in design of structures at key locations for each of the safety-related structures of the Byron/Braidwood and Marble Hill plants.

<u>Response</u>: A comparison of the SSE response spectra is provided in attached Addendum.

4. <u>Item</u>: In order to assess the actual margins in the design for the OBE loads, compare for the key member and the stress levels resulting from the original Byron/Braidwood seismic analysis with those resulting from the use of Marble Hill loads.

<u>Response</u>: Reassessment based on OBE loads was done on a few randomly selected structural elements which were reassessed earlier for increased SSE loads. Marble Hill used a zero period acceleration of 0.08g OBE, and Byron/Braidwood used 0.09g OBE; therefore, the Marble Hill loads were factored by 0.09/0.08 to determine the Byron/Braidwood OBE loads.

Containment Building

Review of the containment structure for Marble Hill loads factored by the ratio between Byron/Braidwood and Marble Hill OBE level indicates that the containment is within design allowable stresses.

Eight structural beams in the containment building were reassessed. Using the average actual material strength, all the beam stresses are within the design basis allowables.

Auxiliary/Fuel Handling Building

Eleven structural steel beams were selected from the auxiliary/fuel handling building. Using the average actual material strength, a factor of safety between 1.42 to 1.61 was maintained against yield strength.

Ten structural steel columns were reassessed for the auxiliary/fuel handling building. Using the average actual material strength, a factor of safety between 2.43 to 3.35 was maintained against yield strength.

Twenty-one shear walls in the auxiliary/fuel handling building were reassessed. Based on the average actual material strength of the reinforcing steel, a factor of safety for the horizontal reinforcing steel between 1.47 to 3.45 was maintained against yield strengths. A factor of safety in the vertical reinforcing steel between 1.61 to 6.58 was also maintained. The stresses in these walls do not exceed the yield strength.

From the finite element model of the auxiliary/fuel handling building base mat, ten elements were chosen in the critical SSE area. Using the average actual material strengths for both concrete and reinforcing steel, the stresses of all ten elements did not exceed the yield strength.

5. Item: Ductility under normal conditions shall be one. Exceptions will be considered in special situations for SSE load and where modifications are considered impractical by the Applicant, and the Applicant's judgment is confirmed by the staff, provided that safety is assured. Floor response spectra will be computed on the basis of elastic analysis.

Response: Ductility is the ratio between the actual stress to the yield stress. Throughout the reassessment of the structure, a ductility ratio less than or equal to one has been maintained. One isolated case for a shear wall where the ductility ratio exceeded one has been indicated. This case is the result of the SSE load condition. Under the OBE condition, the ductility ratio does not exceed one.

Seismic responses were completed on the basis of elastic analysis.

6. Item: Material Properties

a. Concrete

As-built compressive strength of concrete, f'_c may be used in the reevaluation. The as-built strength of concrete shall be demonstrated by the Applicant through submittal of test data. The average compressive strength, established by the tests, can be used as the "as-built concrete strength." The scope and the extent of tests performed shall be evaluated and approved by the staff.

b. Steel

Both reinforcing and structural steel yield stresses, f_y, will be taken as the average of actual test values. In no case will the yield strength value used in strength computations be taken as greater than 70% of the corresponding tested average ultimate strength value. The scope and the extent of the test program and the resulting test data shall be reviewed and approved by the staff.

Response: The average actual material strengths used have been given in Table Q130.6a-1. The actual concrete strength was obtained from the concrete cylinder test results reported by the onsite independent testing agency. The actual steel strengths were obtained from the material certification reports submitted by the steel supplier. The average actual yield strengths have been compared to the average actual ultimate strengths and it was found that the yield does not exceed 70% of the ultimate.

7. Item: Analysis Procedures

a. Regulatory Guide 1.61 damping values will be used.

b. Accidental torsion will be considered by including an additional eccentricity in the mathematical models of 5% of the building dimension in the direction perpendicular to the applied loads.

c. Stability requirements as stated in the Standard Review Plan Section 3.8.5 must be met.

Response: a. Damping values used in the reassessment conform to Regulatory Guide 1.61.

b. The horizontal seismic model for the auxiliary/ fuel handling/turbine buildings include eccentricities corresponding to the distribution of mass and stiffness for Byron/Braidwood. The eccentricities between the mass centroid for a slab and the center of rigidity of its supporting shear walls result in an average torsional moment of 8% of the maximum building dimension times the story shear, for the major slabs in the model.

It is unlikely that these eccentricities would increase significantly due to any changes other than major changes to the plant structure since the weight of the permanent structural elements accounts for a large percentage of the total mass.

c. Stability requirements of the Standard Review Plan Section 3.8.5 have been met.

Summary

The design of the Byron/Braidwood structures and components required for safe shutdown has been reassessed for SSE loads based on Regulatory Guide 1.60. This reassessment has shown that the design of the Byron/Braidwood plant is conservative. Byron/Braidwood seismic design basis ensures that the integrity and functionality of the safety-related structures are maintained.

TABLE Q130.6a-1

MATERIAL STRENGTH

MATERIAL	DESIGN STRENGTH (psi)	AVERAGE ACTUAL STRENGTH (psi)
Concrete	3,500	5,265
	5,500	6,935
Reinforcing Steel (f_y)	60,000	67,000*
Structural Steel (f_y)	36,000	43,200*
	50,000	56,000*

*Value does not exceed 70% of the actual average ultimate strength.

TABLE Q130.6a-2

RACEWAYS SUPPORTS SUMMARY

ELEVATION	NUMBER OF HANGERS REVIEWED	NUMBER OF SUPPORTS WITH DUCTILITY RATIO ≤1.0	NUMBER OF SUPPORTS WITH DUCTILITY RATIO >1.0
477 ft-0 in.	55	49	6
451 ft-0 in.	30	30	0
401 ft-0 in.	10	10	0
426 ft-0 in.	27	27	0

TABLE Q130.6a-3

NSSS MECHANICAL EQUIPMENT

I. Primary Mechanical Equipment

Reactor Pressure Vessel
Reactor Vessel Internals
Reactor Coolant Pump
Steam Generator Loop
Pressurizer

II. Auxiliary Mechanical Equipment

A. Pumps

Centrifugal Charging Pumps
RHR Pumps
Boric Acid Pumps
Component Cooling Pump
Lube Oil Pumps for Centrifugal Charging Pump
Essential Service Water Pump Motor
Auxiliary Feedwater Pump Motor

B. Tanks and Heat Exchangers

Boric Acid Tank
RHR Heat Exchanger
Component Cooling Heat Exchanger

TABLE Q130.6a-4

BOP EQUIPMENT

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE Q130.6a-4 (Cont'd)

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
Aux. Building Supply Fans	Aux. 451 ft	0.81	0.81	0.67	2	>33	0.73	0.66	0.91	no	17%
Aux. Building Exhaust Fans	Aux. 475 ft	0.96	0.96	0.67	2	>33	0.92	0.65	0.85	no	
ESS Service Water Pump Room Coolers	Aux. 330 ft	2.30	2.30	2.30	2	32.7	0.21	0.21	0.21	yes	
ESS Service Pump Room Fans	Aux. 330 ft				Tested					yes	
RHR Pump Room Cubicle Coolers	Aux. 346 ft	2.30	2.30	2.30	2	32.7	0.27	0.28	0.65	yes	
RHR Pump Room Cubicle Fans	Aux. 346 ft				Tested					yes	
Centrifugal Charging Pump Room Coolers	Aux. 364 ft	2.30	2.30	2.30	2	32.7	0.30	0.26	0.65	yes	
Centrifugal Charging Pump Room Fans	Aux. 364 ft				Tested					yes	
Diesel-Driven Aux. Feed Pump Room Cubicle Cooler	Aux. 383 ft	2.00	2.00	2.00	2	>33	0.38	0.36	0.65	yes	

TABLE Q130.6a-4 (Cont'd)

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
Diesel-Driven Aux. Feed Pump Room Cubicle Fans	Aux. 383 ft				Tested					yes	
ESS Switchgear Room Fans	Aux. 326 ft	0.65	0.65	0.94	2	>33	0.67	0.50	0.91	no	631%
Diesel-Generator Room Exhaust Fans	Aux. 401 ft	0.52	0.52	0.84	2	>33	0.48	0.38	0.65	yes	
Misc. Elect. Equip. Room Vent Fan	Aux. 451 ft	0.89	0.89	0.95	2	>33	0.75	0.66	0.91	yes	
Governor				Tested at 2% Damping.						yes	
Fuel Oil Transfer Pump (Aux.)		0.34	0.34	0.84		>33	0.47	0.38	0.65	no	+66%
Diesel-Driven Aux. Feedwater Pump Batteries 1AF01EA-A 1AF01EA-B 1AF01EB-A 1AF01EB-B	Aux. 383 ft	Replicate equipment generically qualified to levels exceeding the required level.									

TABLE Q130.6a-4 (Cont'd)

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA			DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA			MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
		H_1	H_2	VERT			H_1	H_2	VERT		
DC Distribution Bus and Panels 1DC05E 1DC06E 1DC05EA 1DC06EA	Aux. 451 ft	Tested.									
Storage Batteries (Byron only) 1DC01EA 1DC01EB 1DC02EA 1DC02EB	Aux. 451 ft	Replicate equipment qualified to envelop B/B and MH spectra.							yes		
Storage Batteries (Braidwood only) 1DC01E 1DC02E	Aux. 451 ft	Replicate equipment qualified to envelop B/B spectra.							yes		
Battery Chargers 1DC03E 1DC04E	Aux. 451 ft	Replicate equipment tested to envelop B/B and MH spectra.							yes		
Switchgear 4160V 0CC01E 1AP05E 1AP06E 2PA05E	Aux. 426 ft	Tested.									
Switchgear 2AP06E	Aux. 426 ft	Tested.									

TABLE Q130.6a-4 (Cont'd)

		DESIGN BASIS SPECTRA			DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA			MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
EQUIPMENT	LOCATION	H_1	H_2	VERT			H_1	H_2	VERT		
Pump Batteries	Aux.	Tested.									
480V	426 ft										
1AP10E	Clg.										
1AP12E	Twr.										
1AP98E	874 ft										
1AP99E											
Motor Control	Aux.	Tested.									
Centers	Bldg.										
1AP21E	401 ft										
1AP21E-A	426 ft										
1AP22E											
1AP23E											
1AP24E											
1AP25E											
1AP26E											
1AP27E											
1AP28E											
1AP28E-A											
1AP30E											
1AP32E											
1AP92E											
1AP93E											
2AP22E											

TABLE Q130.6a-4 (Cont'd)

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H1	DESIGN BASIS SPECTRA H2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H1	MARBLE HILL SPECTRA H2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
Main Control Boards 0PM02J 1PM01J 1PM04J 1PM05J 1PM06J 2PM06J	Aux. 451 ft	Replicate equipment qualified to envelop B/B and MH spectra.								yes	
Process Protection System Cabinets 1PA01J 1PA02J 1PA03J 1PA04J	Aux. 451 ft	Qualification Spectra envelop Required Spectra								yes	
Solid State Protection System Panels 1PA09J 1PA10J	Aux. 451 ft	Qualification Spectra envelop Required Spectra								yes	
Safeguards Test Cabinets 1PA27J 2PA27J	Aux. 451 ft	Qualification Spectra envelop Required Spectra								yes	

TABLE Q130.6a-4 (Cont'd)

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
Control System Cabinet 1PA33J 1PA34J	Aux. 451 ft	Tested								yes	
Misc. Elect. Equip. Room Vent Fan	Aux. 451 ft	0.90	0.90	0.95	2	33	0.73	0.66	0.91	yes	
Misc. Elect. Room Exhaust Fan	Aux. 451 ft	0.89	0.89	0.67	2	33	0.73	0.66	0.91	no	402%
Aux. Feed Pump Diesel Drive and Controls	Aux. 383 ft	Tested								yes	
Chilled Water Pump Motor	Aux. 383 ft-0 in.	1.12	1.29	5.2	2	129 Hz	(EW) 1.0g	(NS) 1.0g	3.9g	yes	
Refrigeration Units	Aux. 383 ft-0 in.	0.95g	0.95g	1.5g	2	24.3 Hz	(EW) 0.70g	(NS) 0.85g	1.3g	yes	
Cooling Coil Cabinet Units	Aux. 463 ft-5 in.	2.3	2.3	3.3	2	30.2 Hz	(EW) 0.90g	(NS) 0.67g	0.85g	yes	
Cooling Tower Fan Blades	ESWCT 909 ft			(0.75g) 0.70g	2	6.2 Hz			1.25g	no	186%
				0.78g	2	21.6 Hz			0.64g	yes	

TABLE Q130.6a-4 (Cont'd)

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	H_2	VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	H_2	VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
Cooling Tower Gear Box	ESWCT 909 ft	1.74g (1.5X 1.16)	1.35g (1.5X 0.90)	2.04g (1.5X 1.36)	2	45.9 Hz	0.4g	0.29g	0.50g	yes	
Cooling Tower Fan Motor	ESWCT 909 ft	0.68g	0.68g	0.68g	2	33 Hz	0.50	0.40	0.52	yes	
Cooling Tower Internals ------------											
Lintels	ESWCT 888 ft			0.28g	2	13.99 Hz			0.73g	no	+54%
Extren Beams	ESWCT 901 ft	0.65	1.10	0.78	2	22.6	0.88	1.70	0.65	no	
Mechanical Equipment Supports	ESWCT 909 ft-6 in.	0.43	0.55	0.60	2	33	0.40	0.55	0.50	yes	
Fan Drive Shaft	ESWCT 909 ft-6 in.	0.33	0.38	0.41	2	32	0.41	0.59	0.43	no	+3695%
Gear Reducer Piping	ESWCT 909 ft-6 in.	Dynamic analysis performed. Marble Hill spectra are higher for some frequencies.								no	+723%
Cooling Tower Piping	ESWCT 901 ft	There are 40 natural frequencies below 33 Hz. MH spectra are higher for many frequencies.								no	+22%
Anchor Bolts and U-Bolts	ESCWT 901 ft	Used loads from piping analysis (above).								no	+6.4%

TABLE Q130.6a-4 (Cont'd)

DIESEL EQUIPMENT

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
Main Engine Structure	Aux. 401 ft	1.55			2	24.15	0.67			yes	
		0.93			2	33.0	0.47			yes	
				0.85	2	50.58			0.62	yes	
				0.953	2	30.71			0.66	yes	
			1.047		2	30.74		0.39		yes	
			0.917		2	33.35		0.38		yes	
			0.80		2	39.50		0.35		yes	
				1.025	2	27.63			0.67	yes	
		1.187			2	28.43		0.43		yes	
		0.928			2	33.10		0.38		yes	
		0.80			2	39.50		0.35		yes	
Fuel Oil Filter and Strainer	Aux. 401 ft	0.80	0.80	0.84	2	64.6	0.40	0.34	0.62	yes	
Fuel Injection Pump and Nozzle	Aux. 401 ft	1.78	2.04	1.16	2	>50	1.66	1.08	0.65	yes	
Fuel Transfer Pump	Aux. 401 ft	1.26	1.71	0.71	2	>50	0.60	0.54	0.66	yes	
18 in. Exhaust Manifold Expansion Joint	Aux. 401 ft	2.52	1.89	1.20	2	39.7	1.66	1.08	0.85	yes	
Air Intake Filter Silencer	Aux. 426 ft	0.80	0.80	0.83	2	33.2	0.67	0.52	0.91	no	+43%

TABLE Q130.6a-4 (Cont'd)

DIESEL EQUIPMENT

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
Exhaust Silencer	Aux. 477 ft	0.80	0.80	0.83	2	37.3	0.92	0.65	0.83	no	+3.1%
Lube Oil Relief Valve	Aux. 401 ft	Resultant H = 5.0		5.0	2	>50	Resultant H = 5.0		0.85	yes	
Jacket Water Circ. Pump	Aux. 401 ft	0.50	0.50	1.00	2	>33	0.47	0.38	0.65	yes	
Jacket Water Pump Engine Driven	Aux. 401 ft	1.20	1.18	0.94	2	>50	0.85	0.76	0.74	yes	
Lube Oil Filters	Aux. 401 ft	0.80	0.80	0.85	2	>50	0.47	0.38	0.65	yes	
Lube Oil Pump	Aux. 401 ft	0.89	0.89	3.18	2	H_1 >50	0.40	0.38	0.09	yes	
						H_2 >50	0.47	0.38	1.5	yes	
							SRSS	SRSS	SRSS		
						v >17	0.62	0.54	1.64	yes	
Turbo Lube Oil Filter	Aux. 401 ft	5.0	5.0	5.2	2	n/a	1.80	1.58	5.10	yes	
Jacket Water and Lube Oil Heaters	Aux. 401 ft	0.80	0.80	0.85	2	>33	0.47	0.38	0.65	yes	
Lube Oil Circ. Pump	Aux. 401 ft	0.38	0.38	0.84	2	>33	0.47	0.38	0.65	no	+15%

TABLE Q130.6a-4 (Cont'd)

DIESEL EQUIPMENT

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA			DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA			MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
		H_1	H_2	VERT			H_1	H_2	VERT		
Lube Oil Circ. Pump Motor	Aux. 401 ft	1.70	1.20	5.20	2	>33	0.47	0.38	0.65	yes	
Starting Air* Relief Valve	Aux. 401 ft	3.0	3.0	3.0	2	>50	0.47	0.38	0.65	yes	
Starting Air Separator	Aux. 401 ft	0.80	0.80	0.845	2	30.8-H >33-H_2+V	0.50	0.38	0.65	yes	
Starting Air* Dryer Skid	Aux. 401 ft	5.00	5.00	5.30	2	8.6-H_1 8.53-H_2 11.0-v	0.80	1.10	3.50	yes	
		0.80	0.80	0.85		51.9-H_1 33.2-H_2 37.2-v	0.40	0.34	0.64	yes	
		0.80	0.80	0.85		86.3-H_1 73.8-H_2 112.0-v	0.40	0.34	0.62	yes	
Refrigerated Dryer	Aux. 401 ft	Tested (Byron only)									
Starting Air* After Cooler	Aux. 401 ft		1.2		2	25.7		0.57		yes	
		1.5				24.4	0.68			yes	
		0.8	0.8	0.85		>33	0.47	0.38	0.65	yes	
Jacket Water Thermo Valve	Aux. 401 ft	5.0	5.0	5.0	2	>33	0.53	0.48	0.89	yes	
Lube Oil Thermo Valve	Aux. 401 ft	5.0	5.0	5.0	2	>33	0.63	0.42	0.65	yes	

*Not applicable to diesel-generators with membrane type air dryers.

TABLE Q130.6a-4 (Cont'd)

DIESEL EQUIPMENT

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
Intercooler	Aux. 401 ft	2.17	2.57	1.41	2	>33	0.94	0.41	0.86	yes	
3, 4, and 6 inch Valves Ball & Water Sphere Jamesbury Valves	Aux. 401 ft	3.0	3.0	3.0	2	>33	0.40	0.40	0.85	yes	
							0.81	0.93	1.14	yes	
							0.81	0.40	0.85	yes	
							0.46	2.16	1.03	yes	
Gate Valves	Aux. 401 ft	5.0	5.0	5.0	2	n/a	1.14	1.51	1.44	yes	
Standpipe	Aux.		1.75		2	22.3		0.84		yes	
	401 ft	1.59				23.7	0.68			yes	
			0.80	0.85		45.5		0.35		yes	
						>45			0.63	yes	
Fuel Oil Relief Valve	Aux. 401 ft	5.0	5.0	5.0	2	>50	0.43	0.03	0.03	yes	
Starting Air Compressor Motor	Aux. 401 ft	2.52	2.52	1.68	2	>33	1.14	1.51	1.44	yes	
Lube Oil Cooler	Aux.	1.20	1.20	1.20	2	25.35	0.68	0.59	0.76		
	401 ft	0.80	0.80	0.80		>33	0.48	0.38	0.65	yes	

TABLE Q130.6a-4 (Cont'd)

DIESEL EQUIPMENT

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H1	DESIGN BASIS SPECTRA H2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H1	MARBLE HILL SPECTRA H2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
Level Switch On	Aux.				2	Support					
Standpipe	401 ft					22.3		1.45			
						23.7	1.12				
						>33			0.65	yes	
		3.42ZPA		1.1ZPA		Level					
			3.02ZPA			Switch					
Jacket Water	Aux.		0.88		2	21.9		0.84			
Heater System	401 ft	0.727				23.6	0.68				
			0.40			55.7		0.34			
				0.85		>50			0.65	yes	
Jacket Water	Aux.		1.60		2	15.5		0.90			
Cooler System	401 ft	1.01				20.5	0.95				
Piping			0.73			23.5		0.72			
		0.55				26.4	0.62				
			0.49			28.2		0.43			
			0.46			30.7		0.40			
		0.43				33.3	0.47				
		0.41				35.0	0.45				
		0.40				36.3	0.44			no	34
Lube Oil Strainer	Aux.		0.68		2	24.2	0.68				
Piping	401 ft	0.40				42.6	0.4				
			0.40	0.85		>50		0.34	0.65	yes	

TABLE Q130.6a-4 (Cont'd)

DIESEL EQUIPMENT

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
Lube Oil Cooler Piping	Aux. 401 ft		1.60		2	14.9		0.90			
			0.78			23.2		0.80			
		0.50				27.8	0.58				
			0.49			28.4		0.43			
		0.47				29.9	0.48				
		0.45				31.2	0.47				
			0.42			34.3		0.37			
		0.40				36.4	0.42				
				0.85		>40			0.65	no	40
Lube Oil Heater Piping	Aux. 401 ft		0.46		2	30.7		0.40			
			0.44			32.0		0.38			
		0.40				39.6	0.41				
				0.85		>40			0.65	no	30
Lube Oil Drain Line	Aux. 401 ft	0.50	0.46	0.86	2	52.7	0.43	0.37	0.66	yes	
Lube Oil Dump Valve	Aux. 401 ft	1.96	1.96	1.96	2	>33	0.40	0.38	0.85	yes	
Starting Air Tank Assembly	Aux. 401 ft		1.27		2	27.3		0.47			
		0.85				35.0	0.45				
			0.80			39.4		0.36			
				0.87		>39.4			0.66	yes	

TABLE Q130.6a-4 (Cont'd)

DIESEL EQUIPMENT

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
3, 5, and 6-inch Check Valves	Aux. 401 ft	3.0	3.0	3.0	2	>50	0.40 1.68 1.60 0.40	0.40 1.19 1.03 0.40	0.85 0.85 0.85 0.85	yes	
Starting Air Compressor	Aux. 401 ft	1.10	1.70	1.70	2	46	0.61	0.80	0.90	yes	
Intercooler Water Piping	Aux. 401 ft	2.17	1.41	1.56	2	44.2	0.94	0.41	0.86	yes	
Jacket Water Cooler	Aux. 401 ft	0.95 0.87 0.80	0.95 0.87 0.80	0.95 0.87 0.80	2	29.6 31.4 >33.1	0.50 0.49 0.47	0.41 0.39 0.38	0.66 0.65 0.65	yes	
Generator	Aux. 401 ft	0.80	0.80	0.85	2	50-x,y 31,4-2	0.49	0.38	0.65	yes	
AC. Outlet Box	Aux. 401 ft			Tested at 2% Damping						yes	
Starting Air Relief Valve	Aux. 401 ft	5.0	5.0	5.0		>50	0.47	0.38	0.65	yes	
Thermo Control Valve	Aux. 401 ft	0.47	1.00	1.04		34.8	2.6	14.8	22.0	yes	

TABLE Q130.6a-4 (Cont'd)

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
ESF Sequencing + Actuation Cabinets 1PA13J 1PA14J 2PA13J 2PA14J	Aux. 451 ft	Tested and Analyzed									
Remote Shutdown Panels 1PL04J 1PL05J 1PL06J 2PL04J 2PL05J	Aux. 383 ft	Tested and Analyzed								yes	
Annunciator Input Cabinets 1PA31J 1PA32J 2PA31J 2PA32J	Aux. 451 ft	Tested								yes	
HVAC Local Control Panels OVA01JA 0VA01JB	Aux. 467 ft	0.65	0.65	0.68	2	>33	0.87	0.66	0.85	no	+4.4%

TABLE Q130.6a-4 (Cont'd)

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
HVAC Local Control Panels 1VA01J 1VA02J 1VA03J 1VA04J 1VA10J 1VA11J 1VA13J	Aux. 401 ft 426 ft 451 ft	Tested									
HVAC Local Control Panels 1VD01JA 1VD01JB	Aux. 401 ft	0.38	0.38	0.84	2	>33	0.47	0.38	0.65	no	34%
HVAC Local Control Panels 1VE01J 1VX01J 1VX02J	Aux. 451 ft	0.70	0.70	0.95	2	>33	0.72	0.65	0.90	no	77%
Aux. Feed Pump Startup Panel 1AF01J	Aux. 383 ft	Replicate equipment generically qualified to levels exceeding the required levels.									
DC Fuse Panel 1DC10J 1DC11J	Aux. 451 ft	Replicate equipment tested to envelop B/B and MH spectra.								yes	

TABLE Q130.6a-4 (Cont'd)

EQUIPMENT	LOCATION	DESIGN BASIS SPECTRA H_1	DESIGN BASIS SPECTRA H_2	DESIGN BASIS SPECTRA VERT	DAMPING FACTOR (%)	NATURAL FREQ (Hz)	MARBLE HILL SPECTRA H_1	MARBLE HILL SPECTRA H_2	MARBLE HILL SPECTRA VERT	MH SPECTRA ENVELOPED? Y/N	IF NO, GIVE TOTAL MARGIN
Diesel-Generator Control Panels 1PL07J 1PL08J	Aux. 401 ft	Tested									
Local Instrument Panels 1PL50J 1PL67J	Cont. 377 ft										
1PL75J 1PL52J	Aux. 364 ft	0.97	0.97	1.25	2	33	0.40	0.36	0.75	yes	
1PL79JB 1PL77JC 1PL84JA 1PL85JB	Safety Valve Room 377 ft										

QUESTION 110.68

> "Provide a discussion on the effect of the new seismic response spectra on the safety-related small piping and instrumentation design. Specifically address how the seismic support spans as specified in the Byron/Braidwood 'Small Piping Design Standard' are affected by the new seismic response spectra."

RESPONSE

All the instrumentation required for safe shutdown was qualified to a spectra enveloping both the design spectra and the Marble Hill spectra. Hence, the new spectra had no effect.

In the small piping design procedure, the support spans were governed by the piping stress allowables. In tabulating the support reactions by dynamic analysis, a spectra enveloping both the design and Marble Hill spectra was used. Hence, the new spectra had no effect.

QUESTION 110.70

"Indicate an assessment of the effects of the new seismic response spectra on Reactor Pressure Vessel Internals."

RESPONSE

Figure Q110.70-1 shows the comparison of the generic response spectra to which the reactor pressure vessel internals were qualified versus the design basis and Marble Hill spectra.

The generic response spectra is much greater than the Marble Hill spectra. Hence, the Marble Hill spectra has no effect on the reactor pressure vessel internals.

3.7A.3 Byron/Braidwood Assessment of Piping Systems, Components, and Supports Using the New Seismic Response Spectra

NRC Question 110.66 requested an assessment of the various safety-related piping, components, and supports to meet the new seismic response criteria per Regulatory Guide 1.60. The Applicant used the Marble Hill spectra for the evaluation. The Marble Hill design is a replicate of the Byron/Braidwood design. The Applicant provided a justification for any site unique structural differences at critical sections of each plant.

QUESTION 110.66

"Provide an assessment of those piping systems, components, and supports identified above for the new seismic response spectra. If the applicant chooses to use the Marble Hill spectra for its evaluation of piping and supports, its use must be justified with respect to the structural differences at critical sections of the Byron/Braidwood and Marble Hill plants.

a. For piping, provide a table showing the selected critical locations, the design basis seismic stress, the new seismic stress, the total stress (pressure, weight, and seismic), and the allowable stress.

b. For valves, provide a similar table showing the acceleration values for the design basis, the new seismic design, and the allowable accelerations.

c. For supports, provide a similar table showing the support loads for the design basis, the new seismic design, and the support allowable load.

d. Provide a qualitative measure of the overall margin to failure for each of the above components."

RESPONSE

As the Marble Hill design basis is replicate to the Byron and Braidwood design and had implemented Regulatory Guide 1.60 spectra, it was used in the assessment of the Byron/Braidwood piping systems, components, and supports. The structural differences at critical sections of the Byron/Braidwood and Marble Hill plants are as follows.

Methodology

Based on the Marble Hill design, the stress level in structural elements unique to Marble Hill as compared to Byron/Braidwood were identified. The overstressed elements were reviewed against the actual average material strength. Reassessed elements having a safe shutdown earthquake (SSE) stress level less than the yield limit of the material will be considered acceptable due to the localized nature of the forces. Elements still overstressed will be reviewed on a case by case basis in order to determine their impact on the safe shutdown of the plant.

Reassessment of Safety-Related Structures

Containment Building Structure

The initial assessment of the containment structure was based on a comparison of the Byron/Braidwood design with that of Marble Hill. Areas where Marble Hill had unique design features were identified and force comparisons tabulated.

The Marble Hill and Byron/Braidwood design bases, however, were based on conservative assumptions as is appropriate for initial design purposes and, therefore, the force comparison does not reflect the adequacy of the structure. Subsequent to the initial assessment, the following conservatisms were identified and removed and the structure reanalyzed:

a. The overall containment overturning moment, axial force and shear used in the reassessment were obtained using a shell model rather than the beam model previously used.

b. The stiffness of the elements used to represent the reactor cavity wall was modified to account for initial concrete cracking due to shrinkage, thermal effects, and restraints.

c. Full hydrostatic uplift forces included in the design basis base mat analysis were excluded from the present analysis. This is a more realistic assumption due to the uplift of the base mat during a seismic event.

d. The factor applied to the effect of a single horizontal excitation for base mat analysis to account for the effect of three components of excitation was reduced to 1.05 from the 1.10 used in the design basis analysis. The factor 1.10 is based on a study performed during the design basis analysis and it included a 5% to 8% safety margin.

The results of the containment structure assessment based on the refined analysis show no overstress in the containment structure.

Containment Internal Structures

Concrete

No unique design features were located.

Steel Columns

No unique design features were located.

Steel Beams

Eighty-three beams of approximately 740 per unit have a unique design due to seismic forces. These beams were assessed against the Marble Hill SSE response spectra and the results tabulated as follows.

Acceptance Criteria	Number of Beams
Lesser of 0.95 F_y or 1.6 AISC Allowable where F_y = 36 ksi	75
Lesser of 0.95 F_y or 1.6 AISC Allowable where F_y = 42.3* ksi	6
1.0 F_y where F_y = 42.3* ksi	2
	83

*Actual average material strength for A36 structural steel.

Auxiliary/Fuel Handling Building

Base Mat

The results of the assessment are tabulated as follows.

Acceptance Criteria	Number of Mat Areas (Finite Elements)
ACI Allowable F'_c = 3,500; F_y = 60,000	10
ACI Allowable F'_c = 5,265*; F_y = 67,000*	7
	17

*Actual average material strength for concrete and Rebar.

Shear Walls

The results of the assessment are as follows:

Acceptance Criteria	Number of Walls: Vertical Steel	Number of Walls: Horizontal Steel
ACI (0.9 F_y) $F'_c = 3,500$; $F_y = 60,000$	45	64
ACI (0.9 F_y) $F'_c = 3,500$; $F_y = 67,000$*	19	0
1.0 F_y $F'_c = 3,500$; $F_y = 67,000$*	1	0
1.035 F_y (M.F. = 0.966) $F'_c = 3,500$; $F_y = 67,000$*	0	1
	65	65

*Actual average material strength for Rebar.

Concrete Beams and Slabs

No unique design features were located.

Steel Columns

The results of the assessment are as follows:

Acceptance Criteria	Number of Columns
AISC Allowable $F_y = 50$	62
AISC Allowable $F_y = 56$*	4
	66

*Actual average material strength for A577 structural steel.

Steel Beams

The results of the assessment are as follows:

Acceptance Criteria	Number of Beams
Lesser of 0.95 F_y or 1.6 AISC Allowable $F_y = 36$	189
Lesser of 0.95 F_y or 16 AISC Allowable $F_y = 42.3$*	52
1.0 F_y $F_y = 42.3$*	1
	242

*Actual average material strength for A36 structural steel.

a. In reassessing the piping systems, all the piping subsystems or problems were identified and grouped according to system as shown in Table Q110.66-1. From the 319 piping problems listed, a representative sample of 40 piping problems were selected using the following criteria:

1. Selected all piping problems with Level C design basis levels "Equation 9 NB/NC-3600" greater than 80% of the allowable limits.

2. Selected piping problems ranging in size from 3/4 inch to 48 inches nominal diameter.

3. Selected piping problems from a range of building elevations to provide a diversity of response spectra input.

4. Selected at least one piping problem from each system required for safe shutdown.

Methodology

In assessing the piping problems, the problems were analyzed for at least 30 modes or a range of frequency covering 33 Hz. The damping valves used were in accordance with Regulatory Guide 1.61 and the resultant stresses for service Level C were compared against the allowable stress limits of Articles NB/NC-3600 of ASME PBC Code Section III.

Table Q110.66-2 contains a comparison of the highest stress points in a given piping problem, the function of the line, location, elevation, and allowable stress limits.

b. A total of 89 valves were located in the 40 selected piping subsystems. These valves were assessed as follows:

   1. 61 valves had to meet allowable acceleration limits in each mutually perpendicular direction (X,Y,Z).

   2. 28 valves had to meet allowable acceleration limits in horizontal and vertical direction.

   Table Q110.66-3 contains a comparison of the inline valve accelerations using the design basis and Marble Hill response spectra for the SSE event. The valve types, sizes, and allowable accelerations are also shown in Table Q110.66-3.

c. There are a total of 875 supports on the 40 piping problems selected. Of these 875, only 242 supports had load increases. Table Q110.66-4 contains the comparison of the support loads resulting from the dynamic analysis using the design basis and Marble Hill response spectra for the SSE event. Also included is the support number and type, and the maximum load carrying capacity of the support for service Level C.

d. The difference between the results of the dynamic analysis using the design basis and the Marble Hill response spectra for the SSE event is minimal. Figures Q110.66-1 through Q110.66-4 provide a qualitative measure of the overall margin to failure for the piping problems, valves, and supports based on the Marble Hill spectra.

TABLE Q110.66-1

BREAKDOWN OF PIPING SUBSYSTEMS*

SYSTEM	NUMBER OF PIPING PROBLEMS
Main Steam System	5
Main Feedwater System	18
Auxiliary Feedwater System	15
Emergency Diesel Generator System	20
Component Cooling System	64
Essential Service Water System	52
Chemical and Volume Control System	70
Borated Water System	3
Residual Heat Removal System	20
Reactor Coolant System	30
Chilled Water System	22
TOTAL	319

* Piping systems required for safe shutdown of the plant assuming the occurrence of an safe shutdown earthquake event.

TABLE Q110.66-2

STRESS LEVELS FOR ASME CLASS PIPING

SUB-SYSTEM	LINE	CODE CLASS	BLDG. ELEVATION	PIPE ELEMENT	LEVEL C EQ. 9 STRESS[1] (KS1) DESIGN BASIS	LEVEL C EQ. 9 STRESS[1] (KS1) MARBLE HILL	LEVEL C STRESS LIMIT [2] (KS1)
1AF-02	Aux. Feedwater Pump Discharge	2	Aux/364	Reducer	20.5	19.5	27.0
1AF-05	Aux. Feedwater Supply to Steam Generator	2	Aux/383	Elbow	16.4	16.8	27.0
1AF-06	Aux. Feedwater Pump Discharge	2	Aux/364	Valve End	11.9	10.3	27.0
1AF-07	Aux. Feedwater Supply to Steam Generator	2	Aux/401	Valve End	21.3	22.5	27.0
1CC-42	Component Cooling Return Piping	2	Aux/383	Elbow	7.5	5.4	27.0
1DO-16	Transfer Pump Discharge	2	Aux/383	Str. Pipe	5.9	6.0	27.0
1DO-17	Transfer Pump Supply	2	Aux/383	Eq. Conn.	14.3	15.4	27.0

(1) Operating Pressure and Deadweight and SSE.
(2) 2.25 S_m for Class 1, 1.8 S_h for non-Class 1, defined at operating temperature.

TABLE Q110.66-2 (Cont'd)

SUB-SYSTEM	LINE	CODE CLASS	BLDG. ELEVATION	PIPE ELEMENT	LEVEL C EQ. 9 STRESS(1) (KS1) DESIGN BASIS	MARBLE HILL	LEVEL C STRESS LIMIT (2) (KS1)
1FW-12	Main Feedwater Valve Bypass Line	2	Aux/383	Struc. Anchor	19.6	19.7	27.0
1FW-16	Main Feedwater Valve Bypass	2	Aux/383	Valve End	14.8	12.7	27.0
1SX-02	Essential Service Water Return Headers	2	Aux/346	Elbow	10.6	10.5	27.0
1SX-05	Essential Service Water	2	Aux/383	Elbow	14.3	15.2	27.0
1SX-13	Essential Service Water Return Piping	2	Aux/401	Elbow	11.8	12.5	27.0
1SX-34	Essential Service Water Return Piping	2	Aux/401	Struct. Anchor	26.1	25.7	27.0
1SX-63	Essential Service Water Return Piping	2	Aux/401	Struct. Anchor	19.2	20.5	27.0

(1) Operating Pressure and Deadweight and SSE.
(2) 2.25 Sm for Class 1, 1.8 Sh for non-Class 1, defined at operating temperature.

TABLE Q110.66-2 (Cont'd)

SUB-SYSTEM	LINE	CODE CLASS	BLDG. ELEVATION	PIPE ELEMENT	LEVEL C EQ. 9 STRESS(1) (KS1) DESIGN BASIS	LEVEL C EQ. 9 STRESS(1) (KS1) MARBLE HILL	LEVEL C STRESS LIMIT (2) (KS1)
2SX-03	Essential Service Water Return Piping	2	Aux/401	Struct. Anchor	6.3	6.6	27.0
1W0-32	Chilled Water	2	Aux/451	Eq. Conn.	7.3	9.9	27.0
----	Reactor Coolant Loop	1	Cont.				
	Hot Leg			Elbow	31.0	29.7	39.6
	Crossover Leg			Elbow	32.3	29.8	29.6
	Cold Leg			Weld	36.1	26.3	39.6
1MS06	Main Steam	2	Cont. 386 ft to 464 ft	Elbow	15.7	15.5	27.0
1FW04	Feedwater	2	Cont.	Elbow 390 ft to 407 ft	8.4	8.8	17.0
1RH02	RHR/SI System	1	Cont. 393 ft	Elbow	45.0(3)	35.6	38.0
				Tee	39.1	27.8	38.0
		2		Anchor	17.1	31.0	33.3

(1) Operating Pressure and Deadweight and SSE.
(2) 2.25 Sm for Class 1, 1.8 Sh for non-Class 1, defined at operating temperature.
(3) Designed to Level D stress Limit (=51 KS1).

TABLE Q110.66-2 (Cont'd)

SUB-SYSTEM	LINE	CODE CLASS	BLDG. ELEVATION	PIPE ELEMENT	LEVEL C EQ. 9 STRESS[1] (KS1) DESIGN BASIS	MARBLE HILL	LEVEL C STRESS LIMIT [2] (KS1)

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE Q110.66-2 (Cont'd)

SUB-SYSTEM	LINE	CODE CLASS	BLDG. ELEVATION	PIPE ELEMENT	LEVEL C EQ. 9 STRESS[1] (KS1) DESIGN BASIS	LEVEL C EQ. 9 STRESS[1] (KS1) MARBLE HILL	LEVEL C STRESS LIMIT[2] (KS1)

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE Q110.66-2 (Cont'd)

SUB-SYSTEM	LINE	CODE CLASS	BLDG. ELEVATION	PIPE ELEMENT	LEVEL C EQ. 9 STRESS[1] (KS1) DESIGN BASIS	MARBLE HILL	LEVEL C STRESS LIMIT [2] (KS1)
1CV60	Excess Letdown	1	Cont. 415 ft	Weld	26.4	22.4	38.3
		2		Weld	18.7	22.6	28.6
1CC25	Component	2	Cont.	Weld	14.7	20.6	33.3
1CC28	Cooling		396 ft	Weld	23.5	19.3	33.3
1SD04	SG Blowdown	2	Cont. 387 ft to 401 ft	Straight Pipe	18.4	14.1	27.0
1SI02	Safety Injection	2	Cont. 420 ft	Straight Pipe	9.1	10.9	28.6
1SI04	Accumulator System	1	Cont. 393 ft to 429 ft	Tee	30.5	29.4	45.0
		2		Elbow	21.2	17.6	33.3
1SI09	Accumulator	1	Cont. 393 ft to 429 ft	Tee	29.4	31.1	45.0
		2		Weld	22.5	17.6	33.3

(1) Operating Pressure and Deadweight and SSE.
(2) 2.25 Sm for Class 1, 1.8 Sh for non-Class 1, defined at operating temperature.

TABLE Q110.66-3

VALVE ACCELERATIONS ON ASME PIPING

SUB-SYSTEM	VALVE SIZE/TYPE	DESIGN BASIS SSE			MARBLE HILL SSE			ALLOWABLE SSE		
		H_1	H_2	V	H_1	H_2	V	H_1	H_2	V
1AF-02	3 in. Control (A.O)	0.564	1.333	0.955	0.562	1.185	0.718	3.0	2.5	3.0
	3 in. Control (A.O)	0.906	1.264	1.059	0.679	1.090	0.774	3.0	2.5	3.0
	3 in. Control (A.O)	1.393	1.575	0.992	1.061	1.272	0.753	3.0	2.5	3.0
	3 in. Control (A.O)	1.545	1.057	1.211	1.029	0.833	0.869	3.0	2.5	3.0
1AF-05	6 in. Check	1.718	1.836	1.001	1.661	1.857	1.042	3.0	2.5	3.0
	6 in. Gate	1.406	1.857	0.756	1.436	1.921	0.744	3.0	2.5	3.0
	6 in. Control	1.459	1.983	0.566	1.520	2.051	0.504	3.0	2.5	3.0
	6 in. Gate	1.578	2.111	0.777	1.633	2.177	0.675	3.0	2.5	3.0
	6 in. Check	1.354	1.073	1.853	1.307	1.194	1.955	3.0	2.5	3.0
	4 in. Globe	1.198	0.987	1.263	1.195	1.093	1.271	3.0	2.5	3.0
	4 in. Check	1.315	1.534	1.269	1.250	1.561	1.302	3.0	2.5	3.0
	4 in. Globe	1.678	1.467	0.887	1.592	1.370	0.896	3.0	2.5	3.0
1AF-06	6 in. Check	0.559	0.573	1.183	0.506	0.573	1.212	3.0	2.5	3.0
	6 in. Gate (M)	0.569	1.149	1.149	0.516	0.580	1.160	3.0	2.5	3.0
	4 in. Control (A.O)	0.500	0.634	0.922	0.455	0.626	0.777	3.0	2.5	3.0
	6 in. Gate (M)	1.216	1.005	1.561	1.098	0.924	1.269	3.0	2.5	3.0
	4 in. Nozl Check	0.956	0.869	1.539	0.897	0.829	1.365	3.0	2.5	3.0
	4 in. Globe (MO)	1.143	0.997	0.882	0.883	0.973	0.780	3.0	2.5	3.0
	4 in. Nozl Check	1.102	0.452	0.914	0.931	0.436	0.671	3.0	2.5	3.0
	4 in. Globe Iso (MO)	1.341	0.448	0.876	1.034	0.452	0.635	3.0	2.5	3.0
1AF-07	6 in. Check	1.109	0.953	0.869	0.960	0.846	0.670	3.0	2.5	3.0
	6 in. Gate (M)	0.833	0.717	0.911	0.726	0.649	0.728	3.0	2.5	3.0
	6 in. Control (AO)	0.598	0.481	0.851	0.530	0.430	0.646	3.0	2.5	3.0
	6 in. Gate (M)	0.714	0.570	1.117	0.624	0.501	0.825	3.0	2.5	3.0

TABLE Q110.66-3 (Cont'd)

SUB-SYSTEM	VALVE SIZE/TYPE	DESIGN BASIS SSE			MARBLE HILL SSE			ALLOWABLE SSE		
		H_1	H_2	V	H_1	H_2	V	H_1	H_2	V
1AF07	6 in. Check	1.126	0.880	2.444	0.924	0.710	1.534	3.0	2.5	3.0
	4 in. Globe (MO)	1.702	1.103	1.338	1.035	0.787	0.865	3.0	2.5	3.0
	4 in. Check	0.856	0.641	0.936	0.719	0.547	0.685	3.0	2.5	3.0
	4 in. Globe (MO)	1.568	0.663	0.855	1.171	0.564	0.622	3.0	2.5	3.0
1CC-42	3 in. Gate (M)	0.760	0.375	1.034	0.535	0.339	0.608	2.12	2.12	2.0
	3 in. Gate (M)	0.430	1.943	1.470	0.816	1.051	0.798	2.12	2.12	2.0
1DO-16	3 in. Relief	0.478	0.853	0.657	0.497	0.841	0.600			
1DO-17	3 in. Gate	0.277	0.347	0.825	0.293	0.341	0.608	3.0	2.5	3.0
	3 in. Gate	0.258	0.312	0.816	0.289	0.324	0.592	3.0	2.5	3.0
	3 in. Gate	0.945	0.453	0.857	0.819	0.412	0.739	3.0	2.5	3.0
	3 in. Gate	0.289	0.477	0.812	0.315	0.419	0.623	3.0	2.5	3.0
1FW-12	3 in. Check	1.499	1.220	1.377	1.303	1.071	1.151	3.0	2.5	3.0
	3 in. Control	0.987	0.857	0.990	0.874	0.789	0.803	3.0	2.5	3.0
	3 in. Gate	1.482	1.229	0.696	1.232	1.066	0.517	3.0	2.5	3.0
	3 in. Gate	0.886	0.849	0.561	0.763	0.744	1.465	3.0	2.5	3.0
	3 in. Control	0.476	0.552	0.949	0.423	0.504	0.705	3.0	2.5	3.0
	3 in. Gate	0.878	0.876	0.976	0.729	0.786	0.814	3.0	2.5	3.0
1SX-02	42 in. Butterfly	0.244	0.436	0.842	0.269	0.412	0.609	3.0	2.5	3.0
	42 in. Butterfly	0.230	0.389	0.833	0.238	0.390	0.602	3.0	2.5	3.0
	42 in. Butterfly	0.412	0.799	0.983	0.363	0.696	0.758	3.0	2.5	3.0
	42 in. Butterfly	0.274	0.644	1.012	0.290	0.554	0.738	3.0	2.5	3.0

TABLE Q110.66-3 (Cont'd)

SUB-SYSTEM	VALVE SIZE/TYPE	DESIGN BASIS SSE			MARBLE HILL SSE			ALLOWABLE SSE		
		H_1	H_2	V	H_1	H_2	V	H_1	H_2	V
1SX-02	42 in. Butterfly	0.525	0.591	1.157	0.454	0.459	0.077	3.0	2.5	3.0
	42 in. Butterfly	0.570	0.602	1.295	0.559	0.470	0.978	3.0	2.5	3.0
	24 in. Butterfly	0.323	0.309	0.991	0.321	0.321	0.839	3.0	2.5	3.0
	24 in. Butterfly	0.798	0.604	1.164	0.699	0.529	1.004	3.0	2.5	3.0
	0.75 in. Globe	0.672	0.284	1.171	0.630	0.305	0.845	3.0	2.5	3.0
	24 in. Butterfly	0.599	0.396	0.966	0.593	0.370	0.714	3.0	2.5	3.0
	30 in. Butterfly	0.304	1.253	0.896	0.328	0.937	0.650	3.0	2.5	3.0
1SX-05	16 in. Butterfly	0.518	0.449	1.036	0.468	0.639	0.875	3.0	2.5	3.0
1SX-13	10 in. Butterfly	0.768	0.450	1.685	0.737	0.569	2.055	3.0	2.5	3.0
	10 in. Butterfly	0.802	0.451	1.534	0.788	0.568	1.899	3.0	2.5	3.0
	10 in. Butterfly	0.887	0.453	1.480	0.890	0.565	1.732	3.0	2.5	3.0
	10 in. Butterfly	1.162	0.753	1.189	1.255	0.796	1.149	3.0	2.5	3.0
1WO-32	3 in. Globe	0.746	0.641	1.191	0.725	0.861	1.043	3.0	2.5	3.0
	3 in. Globe	0.850	0.925	1.162	0.906	1.149	1.032	3.0	2.5	3.0
	3 in. Globe	0.862	1.137	1.033	0.921	1.624	0.881	3.0	2.5	3.0
	3 in. Globe	0.829	1.1669	1.324	0.793	2.252	1.254	3.0	2.5	3.0

TABLE Q110.66-3 (Cont'd)

SUBSYSTEM	LINE	VALVE SIZE AND TYPE	DESIGN BASIS SSE (g) H	DESIGN BASIS SSE (g) V	MARBLE HILL SSE ACCELERATIONS (g) H	MARBLE HILL SSE ACCELERATIONS (g) V	ALLOWABLE ACCELERATIONS (g) H	ALLOWABLE ACCELERATIONS (g) V
1RH02	RHR System	12 in. Gate	2.7	2.0	2.1	1.5	3.0	2.0
		12 in. Gate	4.2	0.7	3.3	0.9	3.0	2.0(1)
		12 in. Gate	4.6	2.6	3.8	1.9	3.0	2.0(1)
		12 in. Gate	2.0	1.9	2.4	1.4	3.0	2.0
1SI04	Accumulator System	10 in. Gate	3.2	1.8	3.5	1.9	4.2	3.0
1SI09	Accumulator System	10 in. Gate	2.3	0.9	2.1	0.7	4.2	3.0
1CV06	CVCS	3 in. Gate	2.9	1.4	2.6	1.6	3.0	2.0
		3 in. Globe	3.6	1.9	2.8	2.3	6.0	4.0
1CV60	Excess Letdown	1 in. Globe	0.1	0.1	0.9	0.5	3.0	2.0
		1 in. Globe	0.1	0.1	0.5	0.6	3.0	2.0
1CC25	Component Cooling	2 in. Globe	2.3	2.1	2.8	2.4	8.5	4.0
		2 in. Globe	2.4	1.4	3.7	1.7	8.5	4.0
		4 in. Gate	2.0	1.5	2.5	1.4	3.0	2.0
1RC21	Sample Line	3/4 in. Globe	0.1	0.0	0.7	0.5	3.0	2.0
1CV36	RCP Seal	3/4 in. Globe	2.36	3.52	2.2	3.27	8.5	4

(1) Higher limits to be qualified.

TABLE Q110.66-3 (Cont'd)

SUBSYSTEM	LINE	VALVE SIZE AND TYPE	DESIGN BASIS SSE (g) H	DESIGN BASIS SSE (g) V	MARBLE HILL SSE ACCELERATIONS (g) H	MARBLE HILL SSE ACCELERATIONS (g) V	ALLOWABLE ACCELERATIONS (g) H	ALLOWABLE ACCELERATIONS (g) V
1CC21	Component	3 in. Globe	1.6	1.3	0.8	0.6	6	4
1CC22	Cooling	3 in. Globe	4.1	1.9	3.0	0.9	6	4
1CC23		3 in. Globe	4.2	2.9	1.7	1.7	6	4
		3 in. Globe	4.7	3.6	2.2	2.3	6	4
		3 in. Control	3.3	0.6	1.9	0.3	3(1)	2
1SX06	Service	10 in. Butterfly	2.4	0.38	2.42	1.31	3	2
	Water	10 in. Butterfly	0.62	0.38	1.65	1.16	3	2
		10 in. Butterfly	0.16	0.34	0.35	1.24	3	2
		10 in. Butterfly	0.66	0.26	1.55	1.0	3	2
1SX08	Service	10 in. Butterfly	1.12	0.3	1.66	1.16	3	2
	Water	10 in. Butterfly	1.68	0.54	2.11	1.29	3	2
		10 in. Butterfly	1.14	0.42	2.26	1.43	3	2
		10 in. Butterfly	1.46	0.68	1.61	1.38	3	2

(1) Higher limits to be qualified.

TABLE Q110.66-4

PIPING SUPPORTS

SUBSYSTEM	SUPPORT	DESIGN BASIS LOAD	MARBLE HILL LOAD	PERCENT INCREASE	SUPPORT LOAD CAPACITY
1AF-02	1AF02006X	649	677	4.3	8000
	1AF02016X	476	786	65.5	8000
	1AF02009X	665	743	11.2	8000
	1P Guide	1840	1884	2.4	-
	1AF02013X	539	606	12.4	8000
1AF-05	1AF05061R	4748	4780	0.67	6000
	1AF05002X	729	762	4.5	9600
	1AF05058R	595	659	10.7	2250
	1AF05004R	1525	1570	2.9	2410
	1AF05006R	588	607	3.2	1500
	1AF05007R	472	496	5.1	1500
	1AF05008R	406	407	0.24	1500
	1AF05010R	296	301	1.7	1500
	1AF05012R	430	454	5.6	1500
	1AF05014R	361	382	5.8	1500
	1AF05015R	368	377	2.4	1500
	1AF05017R	363	378	4.1	1500
	1AF05018R	363	372	2.4	1500
	1AF05020R	363	376	3.6	1500
	1AF05021R	363	389	7.2	1500
	1AF05023R	364	374	2.7	1500
	1AF05025R	363	366	0.8	1500

Notes: R - Rigid in Y-Direction.
X - Rigid in Horizontal Direction.
S - Snubber
G - Guides

TABLE Q110.66-4 (Cont'd)

SUBSYSTEM	SUPPORT	DESIGN BASIS LOAD	MARBLE HILL LOAD	PERCENT INCREASE	SUPPORT LOAD CAPACITY
1AF-05	1AF05026R	376	393	4.5	1500
	1AF05028R	397	422	6.3	1500
	1AF05029R	387	408	6.4	1500
	1AF11002X	3853	2871	0.6	9600
	1AF11003X	3023	3068	1.5	9600
	1AF11021R	580	650	12.1	1500
	1AF11005X	764	817	6.9	9600
	1AF11006X	864	900	4.3	9600
	1AF05072R	1625	1648	1.4	2410
	1AF05040R	404	442	9.4	1500
	1AF05045R	362	421	16.3	1500
	1AF05048R	361	390	8.0	1500
	1AF05049R	369	392	6.2	1500
	1AF05051R	377	407	7.9	1500
	1AF05053R	376	403	7.2	1500
	1AF05065R	380	470	23.7	870
	1AF12003X	553	568	2.7	9600
	1AF12002X	544	557	2.4	9600
	1AF12001X	506	2109	316.7	9600
	1AF12005X	1360	1458	7.2	9600
	1AF12004R	532	543	2.1	1500
	1AF12006X	590	720	22.0	9600
	1AF05060S	NOT AVAILABLE			
1AF-05	1AF05063S	3936	4035	2.5	8610
	1AF05064S	2428	2510	3.4	8610
	1AF05009S	4134	4231	2.3	8610
1AF-06	1AF06001R	1407/-61	1415/-69	0.6	2410
	1AF06013R	920/-142	937/-159	1.8	NON-STANDARD
	1AF06017R	820	828	0.9	930
	1AF06025X	736/-661	972	32	9600
	1AF06026X	494/-580	618	4.5	9600

TABLE Q110.66-4 (Cont'd)

SUBSYSTEM	SUPPORT	DESIGN BASIS LOAD	MARBLE HILL LOAD	PERCENT INCREASE	SUPPORT LOAD CAPACITY
	1AF06032R	524	525	0.2	930
	1AF06014R	589	627	6.5	1500
	1AF06028X	530/-352	566	6.8	9600
1AF-07	1AF07026X	372	394	5.9	9600
	1AF07009X	630	639	1.4	9600
	1AF07016R	729/-161	749	2.7	1500
	1AF07024X	335	378	12.8	9600
1DO-16	1DO16004X	68/-1163	76/-1171	0.7	NON-STANDARD
	1DO16006G	41/-20	51/-31	26.8	240
	1DO16006G	50/-119	51/-121	1.6	3010
	1DO16008G	44/-50	47/-53	6.0	240
		155/-93	155/-93	-	3010
1DO-17	1DO17002X	650	714	9.8	3010
	1DO17008G	212	224	5.6	6020
		175	165		480
1SX-02	1SX02031R	32740	33045	0.93	33500
	1SX02015S	33906	34172	0.080	50000
1SX-05	1SX05009X	6481	6528	0.72	9600
	1SX00510X	4953	5039	1.73	9600
	1SX05011X	6007	6113	1.76	9600
1SX-13	1SX13001X	3334	3758	12.7	4500
	1SX13002R	3261	3632	11.4	2710/ROD
	1SX13014R	4283	4813	12.4	3770/ROD
	1SX13016X	2216	2250	1.5	4500
	1SX13021R	1794	1905	6.2	1810/ROD

TABLE Q110.66-4 (Cont'd)

SUBSYSTEM	SUPPORT	DESIGN BASIS LOAD	MARBLE HILL LOAD	PERCENT INCREASE	SUPPORT LOAD CAPACITY
1SX-13	1SX13011X	1592	1629	2.3	3240
	1SX13020X	3955	4140	1.7	2710/ROD
	1SX13019R	1806	1869	3.4	1130/ROD
	1SX13018X	3139	3459	10.2	4500
1SX-34	1SX34001X	1731	1769	2.2	2250
	1SX34004X	840	951	13.2	11630
	1SX34007R	1437	1591	10.7	2250
	1SX34012X	542	721	33.0	1500
	1SX34008R	1922	2374	23.5	2710
	1SX34009X	594	831	39.9	4500
	1SX34013X	927	1210	30.5	4500
	1SX34010R	3773	5144	36.3	11630
	1SX34011X	1268	1714	35.2	11630
1SX-63	1SX63001X	1110	1199	8.0	1500
	1SX63002R	3102	3231	4.1	4500
	1SX63003X	2018	2129	5.5	4500
	1SX63004X	1507	1579	4.8	2250
	1SX63005R	1733	1787	3.1	2710
	1SX6300SX	942	945	0.21	1500
	1SX63007R	1671	1726	3.3	4500
	1SX63008X	627	655	4.5	1500
1WO-32	1WO32002X	1495	1531	2.4	NON-STANDARD
	1WO32993R	938	956	0.95	1502
	1WO32004X	126	134	6.3	3710
	1WO32005R	986	994	0.81	1502
	1WO32006X	304	373	22.7	3710
	1WO32008X				
	1WO32007X	980	1001	2.1	1500
	1WO32010X	326	431	32.2	NON-STANDARD
	1WO32012X	370	456	23.2	3710

TABLE Q110.66-4 (Cont'd)

SUBSYSTEM	SUPPORT	DESIGN BASIS LOAD	MARBLE HILL LOAD	PERCENT INCREASE	SUPPORT LOAD CAPACITY
	1WS32013R	662	670	1.2	1502
	1WO32014X	986	1034	4.8	3710
	1WO32015R	572	591	3.3	870
	1WO32016R	327	355	8.6	3710
	1WO32017R	874	925	5.8	1500
	1WO32018X	265	277	4.5	870
	1WO32020X	342	361	5.5	870
	1WO32023X	538	575	6.9	870
	1WO32037X	1164	1170	0.5	6000
	1WO32032R	701	720	2.7	1502
	1W032038X	397	401	2.5	3710
	1WS32039R	994	1043	4.9	2407
	1WO32040R	573	599	4.5	NON-STANDARD
	1WO32042X	569	601	5.6	870
	1WO32044R	1189	1221	2.7	2407
	1WO32045X	223	281	26.0	6000

TABLE Q110.66-4 (Cont'd)

SYSTEM	SUPPORT NUMBER	TYPE (S,R,F)	DESIGN BASIS LOAD	MARBLE HILL LOAD	CAPACITY
1SI04	SI4-004S	S	3.1	3.8	8.61
	ADD-SNUB	S	8.9	9.5	NSC (New load accepted)
	SI4-009S	S	9.2	9.7	13.96
	ADD-Y-SN	S	10.2	10.7	NSC (New load accepted)
	ADD-H-SN	S	4.6	5.5	NSC (New load accepted)
	SI4-017S	S	4.7	4.9	8.61
1SI09	SI9-020S	S	9.2	10.7	70.35
	SI9-021S	S	6.1	7.3	8.61
	ADD-2-SK	R	7.2	7.7	NSC (New load accepted)
	SI9-024S	S	12.9	15.8	20.1
	ADD-X-DS	S	8.0	9.2	NSC (New load accepted)
	SI9-015S	S	2.9	3.1	8.61
	SI9-14R	F	5.1	5.5	NSC (New Load accepted)
	SI9-04S	S	6.0	7.4	8.61
1CV06	1CV06-15	R	1.1	1.2	24.84
	1CV06-14	S	1.2	1.4	20.1
	1CV06-08	S	2.0	2.3	8.61
	1CV06-10	S	1.0	1.2	2.067
1CV02	1CV02003	S	2.1	2.5	8.6
	1CV02010	S	1.2	1.4	2.1
	1CV02005	S	0.5	0.6	0.6
	1CV02111	F	2.6	2.9	NSC (New load accepted)
1RH02	RHR062R	F	15.5	26.7	NSC (New load accepted)
	RHR058S	S	18.4	18.5	20.1
	ADDY	R	9.0	9.5	NSC (New load accepted)

TABLE Q110.66-4 (Cont'd)

SYSTEM	SUPPORT NUMBER	TYPE (S,R,F)	DESIGN BASIS LOAD	MARBLE HILL LOAD	CAPACITY
1CV09	005	S	0.3	0.4	0.9
	045	S	0.4	0.5	0.5
	047	S	0.2	0.3	0.9
	013	R	0.5	0.6	2.3
	014	R	0.2	0.3	1.2
	016	R	0.2	0.3	1.5
	019	R	0.3	0.4	2.3
	020	R	0.4	0.5	1.3
	030	S	0.4	0.5	2.1
	024	F	0.4	0.6	NSC (New load accepted)
	035	S	0.2	0.3	0.5
	036	R	0.4	0.5	0.9
	039	R	0.1	0.2	0.9
	REIRC02ACA	F	0.4	0.5	NSC (New load accepted)
1CV60	VALVE 1	R	0.2	0.3	NSC (New load accepted)
1SD04	004S	S	0.18	0.23	0.5
	RE	F	0.25	0.26	NSC (New load accepted)
1SI02	ANCHOR 1	F	7.0	7.1	NSC (New load accepted)
1CC25	006R	F	1.4	1.6	NSC (New load accepted)
	010R	F	0.7	1.0	NSC (New load accepted)
	013R	F	1.0	1.1	NSC (New load accepted)
	016X	R	2.5	2.8	6.0
	017R	R	1.0	1.2	1.5
	009X	R	0.7	0.9	0.87
	1RB-68	F	2.4	2.7	NSC (New load accepted)
1MS06	0045B	S	47.2	48.1	On Hold
	001X		41.6	44.3	On Hold

TABLE Q110.66-4 (Cont'd)

SYSTEM	SUPPORT NUMBER	TYPE (S,R,F)	DESIGN BASIS LOAD	MARBLE HILL LOAD	CAPACITY
1SX08	SX08001R	R	6.9	9.0	9.6
	002R	R	6.0	7.2	9.6
	003R	R	4.7	6.3	6.0
	004R	R	3.0	4.6	6.0
	005R	R	4.6	5.7	6.0
	019X	R	5.2	6.4	14.1
	006R	R	3.4	5.0	NSC (New load accepted)
	015X	R	2.9	3.6	14.1
	007R1	R	4.3	5.6	NSC (New load accepted)
	008R1	R	4.8	7.0	NSC (New load accepted)
	009R1	R	4.0	6.7	NSC (New load accepted)
	010R1	R	8.9	10.5	NSC (New load accepted)
	011R1	R	5.1	10.5	NSC (New load accepted)
	013R1	R	4.6	12.0	NSC (New load accepted)
	014R1	R	3.6	4.3	NSC (New load accepted)
	021S	S	2.3	2.8	8.6
	020R	R	9.7	11.5	19.2
	017S	S	1.1	1.4	2.1
	016X	R	2.1	2.4	14.1
	025R	R	8.0	9.2	19.2
1SD03	1SD03008S	S	0.2	0.3	0.5
	019R	R	0.3	0.4	0.9
	013X	S	0.3	0.4	0.5
	014X	R	0.1	0.2	0.9
1CC21, 22, 23	1CC22005	R	0.9	1.1	NSC (New load accepted)
	023	R	0.3	0.5	1.5
	034	F	0.4	0.5	NSC (New load accepted)
1SX06	SX0601R	R	3.2	4.3	4.5
	02R	R	2.4	3.8	NSC (New load accepted)
	03R	R	5.1	6.6	NSC (New load accepted)

TABLE Q110.66-4 (Cont'd)

SYSTEM	SUPPORT NUMBER	TYPE (S,R,F)	DESIGN BASIS LOAD	MARBLE HILL LOAD	CAPACITY
	04R	R	6.7	13.6	NSC (New load accepted)
	05R	R	5.3	7.4	NSC (New load accepted)
	06R	R	4.8	7.6	NSC (New load accepted)
	07R	R	4.8	7.9	NSC (New load accepted)
	08R	R	4.3	7.0	NSC (New load accepted)
	09R	R	4.8	6.8	NSC (New load accepted)
	20R	R	4.7	6.8	NSC (New load accepted)
	11R	R	2.9	3.2	3.0 (Faulted 4.0)
	14X	R	3.0	4.4	NSC (New load accepted)
	15R	R	1.0	1.6	1.7
	34R	R	3.6	6.1	5.0 (Faulted 7.1)
	36S	S	3.0	4.3	8.6
	35X	R	3.6	4.5	6.0
	37X	R	1.9	2.6	3.7
	22R	R	5.7	6.4	6.6
	24X	R	1.2	1.8	7.8
	23X	R	3.3	4.7	7.8
	25X	R	1.3	1.7	3.1
	27X	R	0.4	0.6	3.7
	26X	R	2.5	2.8	3.7
	16R	R	5.3	6.4	6.6
	18R	R	3.4	7.8	8.0 (Faulted 8.0)
	17X	R	3.2	5.6	14.0
	19X	R	1.2	2.5	2.3 (Faulted 3.0)
	20X	R	0.6	1.0	2.2
	21X	R	2.6	3.5	6.0
	28R	R	4.5	6.5	6.2 (Faulted 8.8)
	29X	R	4.2	5.2	7.8
	30X	R	4.5	5.7	7.8
	31X	R	3.4	3.9	7.8
	33X	R	4.2	4.6	7.8
	32X	R	2.6	3.0	7.8

TABLE Q110.66-4 (Cont'd)

SYSTEM	SUPPORT NUMBER	TYPE (S,R,F)	DESIGN BASIS LOAD	MARBLE HILL LOAD	CAPACITY
1RY05	1RY05003	S	32.6	38.5	70.4
	008	R	21.8	22.0	37.7
	010	S	17.3	18.3	20.1
1CV03	1CV03002	S	1.3	1.5	2.1
	015	S	0.8	0.9	0.9
	011	R	0.9	1.1	4.5

3.7A.4 Byron/Braidwood Detailed Review of the Seismic Design of One Piping Subsystems

NRC Question 110.67 requested a detailed comparison of the design basis spectra versus the new (R.G. 1.60) seismic spectra for one selected piping subsystem. The response evaluated the 15X-13 piping problem for the essential service water system. The comparison between the Marble Hill (Regulatory Guide 1.60) spectra and the Byron/Braidwood design basis spectra is provided.

QUESTION 110.67

> "Provide a detailed discussion of one of the selected piping subsystems and include a comparison of the design basis spectra used versus the new seismic spectra applicable to the subsystem showing those frequencies where the new seismic spectra are not bounded by the design basis spectra. Include a discussion on the modal frequencies and participation factors of the selected piping subsystem. Show what the resulting effects of those frequencies where the new seismic spectra are not bounded by the design basis spectra are on the piping stresses, support loads, and valve accelerations."

RESPONSE

Piping problem 1SX-13, Figure Q110.67-1, was selected for the detailed discussion for the following reasons:

a. It spans through three different auxiliary building elevations.

b. It contains four valves at different locations.

c. The Marble Hill response spectra is not bounded by the design basis spectra at the higher frequencies. See Figures Q110.67-2, Q110.67-3, and Q110.67-4.

d. It is a 10-inch pipe, which is a good representative size for a study of this nature.

Table Q110.67-1 contains the dynamic characteristics of the piping system, specifically the modal periods and participation factors of piping subsystem 1SX-13.

Table Q110.67-2 contains the highest participation factors for four modes in each direction, the corresponding modal periods and acceleration values for both the design basis and the Marble Hill response spectra.

Table Q110.67-3 is a listing of the seismic stresses using the design basis response spectra.

Table Q110.67-4 is a listing of the seismic stresses using the Marble Hill response spectra.

Table Q110.67-5 provides a comparison of the valve accelerations resulting from both analyses versus the allowable acceleration values.

Table Q110.67-6 provides a comparison of the support loads resulting from both analyses versus the allowable load carrying capacity of the supports for service Level C limits.

The results of these comparisons indicate there is enough margin left in the pipe stresses, valve accelerations, and support loads for service C limits using the Marble Hill response spectra.

TABLE Q110.67-1

DYNAMIC CHARACTERISTICS OF THE PIPING SYSTEM

4391-00 B/B-1 1SX-1

MODE	MODAL PERIODS (SEC)	PARTICIPATION FACTORS X	Y	Z
1	0.18823	1.047	0.194	-2.111
2	0.14355	0.794	0.104	-1.497
3	0.10304	2.024	-0.179	-1.027
4	0.07878	1.317	0.678	-0.315
5	0.06669	-0.293	1.865	-0.231
6	0.05508	-1.950	-0.880	-0.816
7	0.04535	0.566	0.490	0.890
8	0.03987	-0.003	1.233	0.680
9	0.03529	-0.810	0.165	-0.356
10	0.03362	-1.534	-0.828	0.393
11	0.03265	0.194	-0.641	0.961
12	0.03072	-0.429	0.544	0.363
13	0.02786	0.013	1.281	-0.061
14	0.02517	1.685	-1.432	0.395
15	0.02333	-0.248	-0.475	0.251
16	0.02169	0.121	0.804	-1.048
17	0.02020	-0.942	-0.032	0.078
18	0.01887	-0.274	-0.351	-1.353
19	0.01702	0.144	0.994	0.116
20	0.01633	0.283	0.344	0.235
21	0.01622	0.335	-1.019	0.021
22	0.01557	-0.393	-0.520	0.107
23	0.01447	0.027	0.497	0.626
24	0.01387	1.103	0.744	0.177
25	0.01339	0.469	-0.464	0.665
26	0.01314	1.132	-0.162	0.685
27	0.01295	-0.192	0.033	0.405
28	0.01248	0.559	-1.797	0.207
29	0.01211	0.013	-1.282	0.386
30	0.01196	-0.255	0.402	-0.055

TABLE Q110.67-2

PARTICIPATION FACTORS, MODAL PERIODS, AND ACCELERATION VALUES

FOR DESIGN BASIS AND MARBLE HILL SPECTRA

MODE	MODAL PERIOD	PARTICIPATION FACTOR	DESIGN BASIS ACCELERATION (g)	MARBLE HILL ACCELERATION (g)
X-DIRECTION				
3	0.10304	2.024	1.50	1.49
6	0.05508	-1.950	0.85	0.90
10	0.03362	-1.534	0.48	0.52
14	0.02517	1.685	0.48	0.52
Y-DIRECTION				
5	0.06669	1.865	3.00	4.20
8	0.03987	1.233	1.00	1.10
13	0.02786	1.281	0.95	0.90
14	0.02517	-1.432	0.95	0.85
Z-DIRECTION				
1	0.18823	-2.111	1.90	2.15
2	0.14355	-1.497	1.60	1.60
16	0.02169	-1.048	0.42	0.61
18	0.01887	-1.353	0.42	0.61

TABLE Q110.67-3

SEISMIC STRESSES USING DESIGN BASIS SPECTRA

NODE	TYPE	FACTOR I	STRESS IN PSI	DISPLACEMENTS IN INCHES X	Y	Z	AXIS
5	3	1.00	2418.	.000	.000	.000	GLOB
10A	9	2.61	6297.	.000	.000	.000	GLOB
10B	9	2.61	4753.	.026	.018	.004	GLOB
13	1	1.00	872.	.122	.030	.009	GLOB
15A	9	2.61	5105.	.241	.044	.023	GLOB
15B	9	2.61	5734.	.256	.045	.023	GLOB
17	1	1.00	2591.	.235	.030	.023	GLOB
18	1	1.00	2539.	.215	.020	.000	SKEW
20	1	1.00	2445.	.168	.000	.023	GLOB
23	1	1.00	2333.	.087	.038	.023	GLOB
25	1	1.00	4354.	.000	.068	.023	GLOB
33	7	1.90	6819.	.078	.071	.023	GLOB
35	7	1.90	5745.	.098	.069	.023	GLOB
40	7	1.90	4830.	.121	.065	.023	GLOB
45	7	1.90	4439.	.142	.061	.023	GLOB
53	7	1.90	7416.	.195	.049	.023	GLOB
55A	9	2.61	11459.	.211	.045	.023	GLOB
55B	9	2.61	12100.	.207	.022	.024	GLOB
57	1	1.00	4106.	.207	.000	.100	GLOB
60	1	1.00	2956.	.207	.031	.290	GLOB
65	1	1.00	2690.	.208	.049	.658	GLOB
70A	9	2.61	6764.	.208	.043	.946	GLOB
70B	9	2.61	6599.	.000	.037	.929	SKEW
78	1	1.00	1517.	.114	.037	.474	GLOB
80A	9	2.61	11560.	.023	.037	.061	GLOB
80B	9	2.61	14156.	.000	.001	.000	GLOB
85	4	2.10	11543.	.000	.000	.000	GLOB
90A	9	2.61	3539.	.171	.053	.018	GLOB
90B	9	2.61	2983.	.142	.047	.020	GLOB
95A	9	2.61	3742.	.043	.006	.020	GLOB

TABLE Q110.67-3 (Cont'd)

NODE	TYPE	FACTOR I	STRESS IN PSI	DISPLACEMENTS IN INCHES X	Y	Z	AXIS
95B	9	2.61	4068.	.025	.000	.021	GLOB
100A	9	2.61	4108.	.018	.002	.021	GLOB
100B	9	2.61	4625.	.000	.005	.022	GLOB
105	1	1.00	1432.	.069	.008	.022	GLOB
108	1	1.00	1880.	.119	.000	.000	SKEW
110	1	1.00	1951.	.138	.020	.022	GLOB
115A	9	2.61	4008.	.129	.037	.022	GLOB
115B	9	2.61	3313.	.113	.035	.020	GLOB
120A	9	2.61	3833.	.019	.015	.001	GLOB
120B	9	2.61	4768.	.000	.000	.000	GLOB
125	3	1.00	1793.	.000	.000	.000	GLOB
130A	9	2.61	6507.	.072	.073	.026	GLOB
130B	9	2.61	5563.	.098	.066	.031	GLOB
135	1	1.00	1868.	.107	.051	.031	GLOB
140	7	1.90	4328.	.118	.026	.031	GLOB
145A	9	2.61	6618.	.120	.018	.031	GLOB
145B	9	2.61	6753.	.109	.000	.037	GLOB
150	1	1.00	1759.	.109	.021	.149	GLOB
155	1	1.00	1763.	.109	.032	.326	GLOB
160A	9	2.61	4640.	.109	.027	.461	GLOB
160B	9	2.61	3807.	.100	.022	.444	GLOB
165	1	1.00	2796.	.000	.022	.300	SKEW
170A	9	2.61	6574.	.015	.022	.036	GLOB
170B	9	2.61	8632.	.000	.000	.000	GLOB
175	4	2.10	7279.	.000	.000	.000	GLOB
178	1	1.00	0.	.131	.043	.021	GLOB

TABLE Q110.67-3 (Cont'd)

NODE	TYPE	FACTOR I	STRESS IN PSI BRANCH	RUN	DISPLACEMENTS IN INCHES X	Y	Z
30	WDT	1.97	5520.	8040.	.063	.073	.023
30	UFT	5.28	14821.	21589.	.063	.073	.023
30	RFT	5.28	14821.	21589.	.063	.073	.023
50	WDT	1.97	4907.	7135.	.180	.053	.023
50	UFT	5.28	13175.	19158.	.180	.053	.023
50	RFT	5.28	13175.	19158.	.180	.053	.023
93	WDT	1.97	1.	2145.	.131	.043	.020
93	UFT	5.28	4.	5761.	.131	.043	.020
93	RFT	5.28	4.	5761.	.131	.043	.020

TABLE Q110.67-4

SEISMIC STRESSES USING MARBLE HILL SPECTRA

NODE	TYPE	FACTOR I	STRESS IN PSI	DISPLACEMENTS IN INCHES X	Y	Z	AXIS
5	3	1.00	2367.	.000	.000	.000	GLOB
10A	9	2.61	6166.	.000	.000	.000	GLOB
10B	9	2.61	4697.	.025	.018	.004	GLOB
13	1	1.00	860.	.119	.030	.009	GLOB
15A	9	2.61	5090.	.236	.043	.022	GLOB
15B	9	2.61	5695.	.252	.044	.023	GLOB
17	1	1.00	2637.	.233	.030	.023	GLOB
18	1	1.00	2683.	.213	.020	.000	SKEW
20	1	1.00	2793.	.168	.000	.023	GLOB
23	1	1.00	2415.	.088	.044	.023	GLOB
25	1	1.00	4482.	.000	.081	.023	GLOB
33	7	1.90	7181.	.083	.087	.023	GLOB
35	7	1.90	6046.	.104	.085	.023	GLOB
40	7	1.90	5048.	.129	.081	.023	GLOB
45	7	1.90	4599.	.151	.077	.023	GLOB
53	7	1.90	8058.	.206	.064	.023	GLOB
55A	9	2.61	12231.	.224	.059	.023	GLOB
55B	9	2.61	12819.	.219	.028	.025	GLOB
57	1	1.00	4522.	.219	.000	.106	GLOB
60	1	1.00	3390.	.220	.040	.307	GLOB
65	1	1.00	2984.	.220	.058	.698	GLOB
70A	9	2.61	7170.	.220	.046	1.003	GLOB
70B	9	2.61	7048.	.000	.040	.985	SKEW
78	1	1.00	1616.	.120	.040	.502	GLOB
80A	9	2.61	12255.	.025	.040	.065	GLOB
80B	9	2.61	15010.	.000	.001	.000	GLOB
85	4	2.10	12238.	.000	.000	.000	GLOB
90A	9	2.61	4211.	.182	.068	.018	GLOB
90B	9	2.61	3047.	.151	.059	.020	GLOB
95A	9	2.61	4059.	.045	.007	.020	GLOB

TABLE Q110.67-4 (Cont'd)

NODE	TYPE	FACTOR I	STRESS IN PSI	DISPLACEMENTS IN INCHES X	Y	Z	AXIS
95B	9	2.61	4385.	.026	.000	.020	GLOB
100A	9	2.61	4354.	.018	.002	.021	GLOB
100B	9	2.61	4825.	.000	.005	.021	GLOB
105	1	1.00	1459.	.068	.008	.022	GLOB
108	1	1.00	1865.	.117	.000	.000	SKEW
110	1	1.00	1917.	.136	.020	.021	GLOB
115A	9	2.61	3969.	.126	.036	.021	GLOB
115B	9	2.61	3299.	.111	.034	.020	GLOB
120A	9	2.61	3802	.018	.015	.001	GLOB
120B	9	2.61	4706.	.000	.000	.000	GLOB
125	3	1.00	1769.	.000	.000	.000	GLOB
130A	9	2.61	6830.	.078	.088	.026	GLOB
130B	9	2.61	5926.	.105	.080	.031	GLOB
135	1	1.00	1990.	.113	.063	.031	GLOB
140	7	1.90	4399.	.122	.032	.031	GLOB
145A	9	2.61	6674.	.123	.022	.031	GLOB
145B	9	2.61	6833.	.112	.000	.036	GLOB
150	1	1.00	1820.	.112	.022	.150	GLOB
155	1	1.00	1835.	.112	.033	.329	GLOB
160A	9	2.61	4783.	.112	.028	.465	GLOB
160B	9	2.61	3872.	.102	.022	.447	GLOB
165	1	1.00	3074.	.000	.022	.302	SKEW
170A	9	2.61	6637.	.016	.022	.036	GLOB
170B	9	2.61	8713.	.000	.000	.000	GLOB
175	4	2.10	7252.	.000	.000	.000	GLOB
178	1	1.00	0.	.140	.054	.021	GLOB

TABLE Q110.67-4 (Cont'd)

		FACTOR	STRESS IN PSI		DISPLACEMENTS IN INCHES		
NODE	TYPE	I	BRANCH	RUN	X	Y	Z
30	WDT	1.97	5730.	8447.	.066	.088	.023
30	UFT	5.28	15386.	22682.	.066	.088	.023
30	RFT	5.28	15386.	22682.	.066	.088	.023
50	WDT	1.97	5809.	7917.	.191	.068	.023
50	UFT	5.28	15598.	21258.	.191	.068	.023
50	RFT	5.28	15598.	21258.	.191	.068	.023
93	WDT	1.97	1.	2180.	.139	.054	.020
93	UFT	5.28	4.	5854.	.139	.054	.020
93	RFT	5.28	4.	5854.	.139	.054	.020

TABLE Q110.67-5

VALVE ACCELERATION ON ASME PIPING

SUB-SYSTEM	VALVE SIZE/TYPE	DESIGN BASIS SSE			MARBLE HILL SSE			ALLOWABLE SSE		
		H_1	H_2	V	H_1	H_2	V	H_1	H_2	V
1SX-13	10 in. Butterfly	0.768	.450	1.685	0.737	.569	2.055	3.0	2.5	3.0
	10 in. Butterfly	0.802	.451	1.534	0.788	.568	1.899	3.0	2.5	3.0
	10 in. Butterfly	0.887	.453	1.480	0.890	.565	1.732	3.0	2.5	3.0
	10 in. Butterfly	1.162	.753	1.189	1.255	.796	1.149	3.0	2.5	3.0

TABLE Q110.67-6

PIPING SUPPORTS

SUBSYSTEM	SUPPORT	DESIGN BASIS LOAD	MARBLE HILL LOAD	PERCENT INCREASE	SUPPORT LOAD CAPACITY
1SX-13	1SX13011X	1592	1692	2.3	3240
	1SX13020X	3955	4140	1.7	2710/ROD
	1SX13019R	1806	1869	3.4	1130/ROD
	1SX13018X	3139	3459	10.2	4500
	1SX13001X	3334	3758	12.7	4500
	1SX13002R	3261	3632	11.4	2710/ROD
	1SX13014R	4283	4813	12.4	3770/ROD
	1SX13016X	2216	2250	1.5	4500
	1SX13021R	1794	1905	6.2	1810/ROD

3.7A.5 Byron Site Specific Seismic Concerns

In NRC Question 130.9, it was requested that the soil-structure interaction evaluation for the Byron river screen house include the half-space lumped spring and mass method as well as the finite element method used for the Byron design. The response provided the soil-structure interaction responses using the half-space lumped parameter approach.

NRC Question 130.9a requested a quantitative assessment of the impact of enveloping the most severe responses of the two methods used in the response to Question 130.9 on the design of the Byron river screen house. The response to Question 130.9a outlined the assessment and identified possible modifications to the superstructures.

QUESTION 130.9

"The river screen house at the Byron station is founded on soil. Soil-structure interaction was performed by using the finite element method. It is the staff's position that the methods for implementing the soil-structure interaction analysis should include both the half space lumped spring and mass representation and the finite element approaches. Category I structures, systems and components should be designed to responses obtained by any one of the following methods:

a. Envelope of results of the two methods,

b. Results of one method with conservative design consideration of impact from use of the other method.

c. Combination of a. and b. with provisions of adequate conservatism in design.

"Therefore, we request you to compare the responses obtained by the half space (lumped parameter) approach to those obtained by the finite element approach at a few typical locations.

"Floor response spectra should be provided at least at the base mat, an intermediate elevation and an upper elevation. For the lumped parameter representation, the variation of soil properties should be considered."

RESPONSE

The soil-structure interaction responses are generated using the half-space lumped parameter approach. A soil shear modulus corresponding to a low value (10^{-4}%) of strain is used in the analysis. The lower and upper bounds of the soil properties are considered (see Figure 2.5-89) of the Byron FSAR). The soil properties under the river screen house vary with depth. The average soil properties shown in Table Q130.9-1 were used in the elastic half-space soil spring analysis. The translational, rocking, and vertical soil-spring constants are computed based on these soil properties and formulas provided in Reference 1. The numerical values of soil spring constants are shown in Table Q130.9-2.

The horizontal floor response spectra at the base mat (el. 664 ft), an intermediate elevation (el. 702 ft), and the roof (el. 744 ft) are presented in Figures Q130.9-1 through Q130.9-6 for the OBE and in Figures Q130.9-7 through Q130.9-12 for the SSE. The response spectra are generated for 1% damping for OBE and 2% damping for SSE. The dot-dash line represents the design basis, using the finite element method (FEM). The solid line and dash line are those obtained by the half-space solution using the lower and upper bounds, respectively, of the soil properties. Table Q130.9-3 lists the comparison of the selected shear wall forces obtained using the soil spring soil-structure interaction (SSI) response spectra and the finite element SSI response spectra. Figures Q130.9-13 and Q130.9-14 provide the locations of the shear walls. The shear walls were evaluated to these higher forces and the stresses were found to be within the allowable.

The vertical soil-structure interaction analysis using the half-space approach has also been performed. The vertical floor response spectra at the base mat (elevation 664 ft), an intermediate elevation (elevation 702 ft), and the roof (elevation 744 ft) are presented in Figures Q130.9-15 through Q130.9-17 for the OBE and in Figures Q130.9-18 through Q130.9-20 for the SSE. The response spectra are generated for 1% damping for OBE and 2% damping for SSE. The dot-dash line represents the design basis, using the finite element method (FEM). The solid line and dash line are those obtained by the half-space solution using the lower and upper bounds, respectively, of the soil properties.

The higher responses, when the soil spring method is used, can be attributed directly to the conservative assumptions which are made in the analysis as compared to those used in the finite element method: (1) the river screen house is a deeply embedded structure, yet the soil spring method used required that the embedment be neglected; (2) the soil properties vary with depth, as shown in Figure 2.5-89 of the Byron FSAR: an average soil layer property was used in the soil spring method; (3) the soil-shear modulus corresponding to low shear strain and no material damping were used in the soil spring method, while more realistic strain-dependent shear modulus and damping values were used in the finite element method; and (4) in the present B/B seismic design criteria, Regulatory Guide 1.60 spectra are obtained at the surface and deconvolution analysis is used to obtain foundation elevation motions in the free field. These free field motions were used in the finite element SSI analysis; in the present soil spring analysis, the Regulatory Guide 1.60 spectra are applied at the foundation elevation.

In view of these assumptions, we feel that the soil spring SSI results are overly conservative.

REFERENCES

1. "Analysis for Soil-Structure Interaction Effects for Nuclear Power Plants," Report by the Ad-Hoc Group on SSI of the Structural Division of ASCE, November 1978.

TABLE Q130.9-1

AVERAGE SOIL PROPERTIES AT THE RIVER SCREEN HOUSE SITE

	SHEAR MODULUS (K/ft^2)	POISSON'S RATIO	WEIGHT DENSITY (kip/ft^3)
Lower Bound	3324	0.42	0.123
Upper Bound	4627	0.42	0.123

TABLE Q130.9-2

SPRING AND DASHPOT CONSTANTS NUMERICAL VALUES

PARAMETER*	UNITS	(UPPER BOUND)	(LOWER BOUND)
Translational Stiffness, K_x	Lb/in.	10.994×10^7	7.898×10^7
Translational Stiffness, K_y	Lb/in.	10.944×10^7	7.898×10^7
Vertical Stiffness, K_z	Lb/in.	14.350×10^7	10.310×10^7
Rocking Stiffness, K_ψ^x	Lb-in/rad	8.106×10^{13}	5.823×10^{13}
Rocking Stiffness, K_ψ^y	Lb-in/rad	3.127×10^{13}	2.246×10^{13}
Translational Damping, C_x	Lb-sec/in.	3.259×10^6	2.763×10^6
Translational Damping, C_y	Lb-sec/in.	3.259×10^6	2.763×10^6
Vertical Damping, C_z	Lb-sec/in.	6.278×10^6	4.510×10^6
Rocking Damping, C_ψ^x	Lb-in-sec/rad	1.459×10^{12}	1.263×10^{12}
Rocking Damping, C_ψ^y	Lb-ins-sec/rad	3.935×10^{11}	3.335×10^{11}

*The foundation shape and coordinate axes considered in the analysis are shown below:

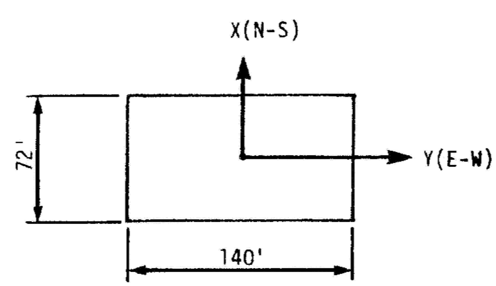

TABLE Q130.9-3

COMPARISON OF SHEAR WALL FORCES FROM

FINITE ELEMENT AND SOIL SPRING APPROACHES

	OBE		SSE	
SHEAR WALL SPRING NO.	FEM* (kips)	SSM** (kips)	FEM (kips)	SSM (kips)
X-1011	154	271	287	497
X-1012	198	356	391	670
X-1013	33	61	73	120
X-1014	33	61	73	120
X-1015	88	163	198	325
X-1016	28	53	65	106
X-1017	28	53	65	106
X-1018	46	86	108	175
Y-1021	9	16	22	35
Y-1022	9	16	22	35
Y-1023	102	185	250	403
Y-1024	62	114	155	247
Y-1025	63	115	155	250
Y-1026	168	311	424	670
Y-1027	170	314	424	680

* Finite element method
** Soil spring method

QUESTION 130.9a*

"For the river screen house at the Byron station there is a marked increase in the response spectra for most of the frequencies of interest in structural design. The technical position of the Regulatory staff is that the results of the two methods, i.e., the half space and the finite element method should be enveloped in order to be used in the design. This position is stated in the enclosure and designated as Method 3(a). As an alternate solution, the staff would find acceptable the two other options which are designated in the Enclosure as Methods 3(b) and 3(c). You are requested to perform a seismic analysis using one of the above noted three options, quantitatively assess its impact on structural design of the river screen house at the Byron plant and submit the results for our review."

RESPONSE

To comply with the staff position, all structures and components of the Byron river screen house will be qualified to the envelope of the responses based on the half-space and finite element methods for soil-structure interaction (SSI) analysis. Any modifications resulting from this reanalysis will be completed prior to plant operation. Consistent with the NRC staff position on Byron seismic reevaluation, requalification of the Byron river screen house structures using the enveloped spectra will be limited to the SSE load combinations only.

The details of the finite element method used for the SSI analysis are provided in Subsection 3.7.2.4.

For the half-space approach, the Regulatory Guide 1.60 spectrum, normalized to 0.2g, was used as input to the soil-structure model. Frequency-dependent soil impedance functions were computed based on the method proposed by Luco (Reference 1). Two sets of soil properties, as presented in Table Q130.9a-1, were used. The soil properties in Set 1 are consistent with those used in the finite element method analysis. Set 2 corresponds to 40% of the geophysical soil shear modulus and represents an upper bound estimate of the soil shear modulus under the SSE excitation at the Byron site. The 40% value is an upper bound soil property of 10^{-3} inch/inch strain. The coupled soil structure system was analyzed in the frequency domain using the complex frequency response method.

*QUESTION 130.9a is a restated version of NRC QUESTION 130.9.

Modifications to the superstructure resulting from the reanalysis will consist of the following items as required:

a. Vertical bracing will be added.

b. Existing connections for the vertically braced column rows will be reinforced.

c. Cover plates will be added to floor and roof beams.

REFERENCES

1. J. E. Luco, "Vibrations of a Rigid Disc on a Layered Visoelastic Medium," from "Nuclear Engineering and Design," Vol. 36, pp. 325-340, 1976.

TABLE Q130.9a-1

SSE SOIL PROPERTIES USED FOR HALF-SPACE SSI ANALYSIS

LAYER NUMBER	DEPTH TO TOP OF LAYER (ft)	LAYER THICKNESS (ft)	SHEAR MODULUS (ksf) SET 1	SET 2
1	0	16	399.0	1850.0
2	16	18	760.0	1850.0
3	34	24	1219.0	1850.0
4	58	32	1997.0	1850.0
5	90	half-space	45000.0	45000.0

3.7A.6 Braidwood Site Specific Seismic Concerns

NRC Question 362.11 challenged the adequacy of the seismic design basis for all Category I structures not founded on rock. The response noted that although the lake screen house rested on 10 feet of hard glacial till there was no appreciable amplification between the rock and the top of the till.

NRC Question 130.56 requested a comparison of the half-space lumped parameter method with the finite element method for determining responses in the lake screen house. The response provided the comparison via reference to Question 130.6a contained in Section 2 of this attachment.

QUESTION 362.11

> "The safe shutdown earthquake for the Braidwood Station site is based on the postulated occurrence of a maximum Modified Mercalli intensity VIII (body wave magnitude about 5.8) earthquake near the site (see Byron Station Safety Evaluation Report (SER), NUREG-0876). The staff's position, as stated in the Byron Station SER, is that a Regulatory Guide 1.60 spectrum with a high frequency anchor of 0.20g at the foundation level of structures founded on rock is an adequately conservative representation of the vibratory ground motion from this size earthquake. Soils can amplify vibratory ground motion. The amplitude and frequency of the amplified motion is a function of the physical properties of the material and its thickness. For all Category I structures not founded on rock, demonstrate the adequacy of the design basis by directly calculating a site-specific response spectrum and/or by calculating the amplification of an appropriate rock spectrum resulting from the presence of the soil.
>
> "We recommend that the details of the study planned in response to this question be discussed with the staff prior to work initiation."

RESPONSE

For the Braidwood site, the design response spectra, which is defined at the ground, is a 0.26g Regulatory Guide 1.60 spectra as shown in Figure 2.5-47. Foundation level response spectra and time histories were generated by deconvolution as described in Subsection 3.7.1. The maximum horizontal and vertical ground accelerations at the foundation level is 0.2g for SSE.

In response to Question 130.6a, all structures and equipment required for cold shutdown were reevaluated for a 0.2g Regulatory Guide 1.60 spectra specified at the foundation elevation of all Category I structures. The founding conditions of the various Category I structures and justification for the use of 0.2g Regulatory Guide 1.60 spectra at the foundation elevation is as follows.

Lake Screen House

The lake screen house rests on 10 feet of hard glacial till. The shear wave velocity of the till material is 2,400 ft/sec. The shear wave velocity of the underlying rock is 3,200 ft/sec. Because of the high soil column frequency (V_s/4H=60 Hz) and the low velocity contrast (3,200/2,400=1.33) between the rock and the soil medium, there will be no appreciable amplification of motion between the rock and the top of the till in the critical frequency range of 1 to 20 Hz. Thus, the use of the 0.2 g Regulatory Guide 1.60 spectrum at the foundation elevation of the lake screen house for reevaluation in response to Question 130.6a is justified.

Containment Structure

The containment structure is founded on rock.

Auxiliary Building - Fuel Handling Building Complex

The auxiliary-fuel handling building complex is founded on rock with only a small percentage of the foundation at the periphery overlying soil (see Figures 2.5-16 and 3.8-45). The major slabs of the structure are continuous diaphragms and monolithically connect the portions on soil to the portions on the rock foundation. Thus, the structure is essentially founded on rock and was analyzed as such.

QUESTION 130.56

"Conflicting information is provided in Table 3.7-3 of Section 3.7 and Section 3.8.5.1.5 of the FSAR which state that the Braidwood lake screen house rests on rock and glacial till respectively. If the information contained in Section 3.8.5.1.5 is correct, describe the method of analysis used to account for the soil-structure interaction.

"It is the staff's position that the methods implementing the soil-structure interaction analysis for structure situated on soil should include both the half space lumped spring and mass representation and the finite element approaches. Category I structures, systems and components should be designed to responses obtained by the appropriate methods described in the Standard Review Plan, Section 3.7.2.II.4.

"Therefore, you are requested to compare the responses obtained by the half space (lumped parameter) method to those obtained by the finite element approach at the following locations:

"(a) Base mat
(b) El. 588'-0"
(c) El. 602'10"
(d) Roof.

"In both analyses the variation of soil properties should be considered.

"If the lake screen house is supported by bedrock, please confirm that the information provided in Section 3.8.5.1.5 on the same subject is incorrect and make the necessary FSAR corrections."

RESPONSE

The Braidwood lake screen house foundation rests on 10 feet of glacial till underlain by rock. Due to the shallow depth (10 feet) of the till and a high shear wave velocity of 2400 ft/sec, the soil structure interaction effects are not significant (see the response to Question 362.11).

The original design for structures and components was based on a fixed base analysis using the deconvoluted rock time history; the comparison between the design spectra and the spectra from the deconvoluted rock time history is shown in Figures 3.7-21 through 3.7-40.

Furthermore, as a result of Question 130.6a, a reanalysis was performed on a fixed base analysis using the design time history. The main slabs and walls and the major equipment are designed for the envelope of the two analyses results.

Table 3.7-3 has been revised to indicate there is 10 feet of hard glacial till between the rock and the lake screen house foundation.

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 Concrete Containments

Wherever reference is made to "ASME Boiler and Pressure Vessel Code (B&PV) Section III, Division 2," this refers to "Proposed Standard Code for Concrete Reactor Vessels and Containments," issued for interim trial use and comment, April 1973, by the American Society of Mechanical Engineers (ASME).

3.8.1.1 Description of the Containment

3.8.1.1.1 General

The containment structure is a prestressed concrete shell structure made up of a cylinder with a shallow dome roof and flat foundation slab. The cylindrical portion is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. There are three buttresses equally spaced around the containment and each horizontal tendon is anchored at buttresses 240° apart, bypassing the intermediate buttress. The dome post-tensioning system is made up of three groups of tendons oriented 120° to each other and anchored at the vertical face of the dome ring. The entire structure is lined on the inside with steel plate which acts as a leaktight membrane.

The containment completely encloses the entire pressurized water reactor, steam generators, reactor coolant loops, and portions of the auxiliary and engineered safety features systems. It ensures that leakage of radioactive material to the environment does not cause the dose limits of 10 CFR 50.67 to be exceeded. The general configuration and dimensions of the reactor containment structure are shown in Figure 3.8-1. Details of the slab-cylinder intersection, cylinder-dome intersection, buttress, and equipment hatch penetration are shown in Figures 3.8-2, 3.8-3, 3.8-4, 3.8-5, and 3.8-6 respectively.

The containment has the following dimensions:

a. thickness of base slab - 12 feet,

b. diameter of base slab - 157 feet,

c. inside diameter of containment - 140 feet,

d. inside height of containment - 222 feet,

e. thickness of containment wall - 3 feet 6 inches, and

f. dome thickness - 3 feet.

3.8.1.1.2 Base Foundation Slab

The base foundation slab is conventionally reinforced with high strength reinforcing steel. A continuous access gallery is provided beneath the base slab for access to the vertical tendons. The top of the base slab, within the containment, is lined with a steel liner plate to provide a leaktight membrane.

3.8.1.1.2.1 Reinforcing Layout

The base mat is reinforced in both radial and hoop directions. Figures 3.8-7 and 3.8-8 show bottom and top reinforcing plan views of the base slab.

3.8.1.1.2.2 Liner Plate and Anchorage

The steel liner plate is 1/4-inch thick and is anchored by structural steel rolled sections embedded in the concrete and welded to the liner plate (Figure 3.8-2).

3.8.1.1.2.3 Anchorage of Interior Structure Through Liner Plate

Vertical support columns for the pressurizer, steam generators, and reactor coolant-pumps are anchored through the liner plate and into the base slab, as shown in Figure 3.8-9. Also shown in Figure 3.8-9 is the detail used to transfer uplift forces from internal concrete walls into the base slab. Tension loads are transferred from typical wall reinforcement to dowels that are attached to a thickened liner plate using Cadweld sleeves. The tension load is then transferred to steel rods which carry the load to a bearing plate embedded in the base slab. Leak test chambers are provided to check the welds connecting the liner plate insert to 1/4 inch plate for leaktightness before and after concrete is poured between elevations 374 feet and 377 feet.

3.8.1.1.3 Containment Wall

3.8.1.1.3.1 General

The containment cylindrical wall has a constant thickness of 3.5 feet starting from the base slab elevation of 374 feet to the dome springline at elevation 555 feet 3-3/8 inches. The wall has been thickened locally around main steam penetrations, personnel lock, and equipment hatch. Containment reinforcing consists primarily of hoop and meridional steel. Prestressing tendons are arranged in hoop and meridian directions.

3.8.1.1.3.2 Reinforcing Layout

Continuous hoop and meridian reinforcement is placed at the outside face of the cylindrical wall. Similar reinforcement has also been provided at the inside face where the cylindrical wall intersects with the base slab or dome ring and in the area where polar crane brackets are embedded in the containment wall. Where transverse shear reinforcing is required, Number 7 ties have been provided. Figures 3.8-2, 3.8-10, and 3.8-3 show the details of wall reinforcement.

3.8.1.1.3.3 Prestressing Tendon Layout

The containment wall is prestressed using 201 hoop and 162 vertical unbonded tendons. Each hoop tendon is anchored at buttresses 240° apart bypassing the intermediate buttress. The hoop tendons are arranged in the wall between elevation 374 feet 0 inch and 562 feet 0 inch. Figures 3.8-11 and 3.8-12 show typical tendon layout. The buttress details and the anchorage of hoop tendons are shown in Figure 3.8-4.

Vertical tendons are anchored at the underside of the base slab at elevation 362 feet and at the top of the dome ring at elevation 579 feet 0 inch. The anchorage zones for all the tendons have been provided with additional reinforcing to account for transverse tensile stresses resulting from anchorage forces reacting on the concrete.

3.8.1.1.3.4 Liner Plate Details and Anchorage

The 1/4-inch liner plate is attached to the containment wall by means of 3 by 2 by 1/4-inch vertical angles spaced horizontally every 15 inches. Additional horizontal stiffeners are provided to permit the liner to serve as formwork for the containment wall. Figure 3.8-13 shows typical liner plate anchorages.

3.8.1.1.3.5 Penetrations

3.8.1.1.3.5.1 General

Access to the interior of the containment is provided through two personnel locks. One of these penetrates the dished door of the equipment hatch and the other is on the side opposite the equipment hatch at grade level. The equipment hatch permits transfer of equipment into and out of the containment. In addition to these access openings the other major penetrations provided in the containment wall are those required for main steam and feedwater lines. The containment wall is also penetrated by various process pipe lines and electrical penetration assemblies. Figure 3.8-14 shows the location of various penetrations in the containment wall. The size and type of penetrations are listed in Table 3.8-1.

Typical reinforcing around a penetration is shown in Figure 3.8-15. The type, size, and location of the penetration as well as any load that may be imposed by the penetration determines whether any additional reinforcing is required. To provide for continuity, tendons are deflected around the penetrations.

3.8.1.1.3.5.2 Main Steam Penetrations and Personnel Lock

The shell wall around the main steam penetration and emergency personnel lock has been thickened to 4 feet 6 inches. Additional reinforcing has been provided to account for stress concentrations due to the openings and pipe support reactions. The tendons are deflected around the penetrations. Figures 3.8-16, 3.8-17, and 3.8-18 show the details of reinforcement around the main steam penetration and the emergency personnel lock.

3.8.1.1.3.5.3 Equipment Hatch

The wall around the equipment hatch has been thickened to 7 feet. Figures 3.8-5 and 3.8-6 show the details of reinforcing provided around the opening. The tendons are deflected around the equipment hatch as shown in Figure 3.8-19.

3.8.1.1.3.6 Crane Bracket

To support the polar crane girder, brackets have been provided spaced 10° apart with respect to the centerline of the containment and embedded into the containment wall. Forces acting on the wall due to bracket loads have been accounted for in providing additional reinforcing in the wall. A 1-1/2-inch thick liner plate insert has been provided in the bracket area. The spacing of vertical liner stiffeners in the vicinity of thick insert plate has been designed to limit excessive strains in the adjoining 1/4-inch liner plate. Figure 3.8-10 shows reinforcing provided in the containment wall at the bracket area.

3.8.1.1.4 Dome and Dome Ring

3.8.1.1.4.1 General

The roof of the containment structure is made up of a 3-foot thick shallow ellipsoidal dome. The containment wall has been thickened at the top to serve as a dome ring. The inside of the dome is lined with a steel liner plate to provide leaktightness. The dome concrete is poured in two layers. For the first layer of approximately 8 inches, the liner plate with its stiffeners serves as a formwork. For the second layer, the first layer and the liner plate act as a composite section providing the formwork support.

3.8.1.1.4.2 Reinforcing Layout

The dome has been reinforced in two directions. Orthogonal grid type reinforcing has been provided within a radius of 50 feet from the apex of the dome. For the remaining portion, radial and hoop reinforcing has been provided. Radial ties have also been provided over the entire dome to account for radial tension due to prestressing tendons. Figures 3.8-3 and 3.8-20 show the layout of reinforcing in the dome.

3.8.1.1.4.3 Prestressing Tendons

Three groups of tendons oriented 120° to each other have been provided in the dome. In each group there are 40 tendons spaced equally on a horizontal projection. Bearing plates for anchorage of the tendons are placed on wedge shaped pockets located on the vertical face of the dome ring. Figures 3.8-11 and 3.8-21 show the layout and the anchorage of the dome tendons.

3.8.1.1.4.4 Liner Plate Details and Anchorage

Radial and hoop stiffeners have been provided to attach the 1/4-inch liner plate to the concrete dome. However, in the central portion having a diameter of 15 feet, stiffeners are placed to form a rectangular grid. Figures 3.8-22 and 3.8-23 show the arrangement of dome liner stiffeners.

3.8.1.2 Applicable Codes, Standards, and Specifications

This section lists codes, specifications, standards of practice, regulatory guides, and other accepted industry guidelines which are adopted to the extent applicable, in the design and construction of the containment. The codes, standards, and specifications are listed and discussed in Table 3.8-2 and given a reference number (see Appendix A for regulatory guides). The applicable codes, standards, and specifications for the containment are 1 through 23.

3.8.1.3 Loads and Load Combinations

Table 3.8-3 lists load combinations used in the design of the containment shell and base slab. A description of load categories and a definition of loads are given in Table 3.8-4.

The load categories defined in this section include any condition encountered during construction, testing, and in the normal operation of a nuclear power plant, as well as the conditions resulting from the single failure of the reactor coolant system plus those extreme environmental conditions postulated during the life of the facility and certain combinations thereof.

Loads analyzed include both static and transient loads. The transient thermal gradient described in Subsection 3.8.1.4.7 occurs as the containment wall heats up gradually after a LOCA in response to elevated containment atmospheric temperatures. Therefore, it is treated as a static load.

The seismic loads on the structure are transient and are determined from a dynamic analysis as indicated in Subsection 3.7.2.

LOCA pressures are transient; however, LOCA pressures are considered as a static loading because the rate of pressurization is gradual as shown in Figures 6.2-1 through 6.2-6a. Pipe loads resulting from pipe breaks are transient. In design the bounding values of these loads are calculated on the basis of the collapse mechanism of the pipe (e.g., the pipe's plastic moment).

These loads and load combinations meet the requirements of Section CC-3000 of the ASME B&PV Code, Section III, Division 2.

However, there are a number of minor differences between Table 3.8-3 and some of the loading cases of Section CC-3000 of the code. These differences are explained in the following paragraphs.

The differences between Table 3.8-3 of the UFSAR and Table CC-3200-1 of the Summer 1973 Proposed Section III, Division 2 issued for trial use and comment, or Table CC-3230-1 of the 1977 edition of Section III, Division 2 are more a matter of nomenclature than of real differences. A review of the definition of load in Table 3.8-4 and Division 2, Section 3000, reveals equivalence between definition of the individual loads to be considered, as shown in Figure 3.8-85.

The internal flooding load H_a, is the only load specified in the 1977 and later editions of Division 2 that was not defined in earlier editions of Division 2 nor in the Byron/Braidwood design specification as a design-basis event. However, the containment has been evaluated for flooding loads and found adequate. Flooding loads evaluation include flooding inside of the containment to a water level 6 ft 3 in. above the base slab and all other applicable loads specified in Division 2 Table CC-3230-1 for the abnormal/severe environment load combinations.

The construction load case reflects an approach that is more conservative than that required by the original version of Division 2. The load factor of 0.75 is used for all loads but the prestress loads and the resulting stresses compared with the normal values. Under the original Division 2, allowable stresses could be increased by 1/3 when a load combination included wind and/or earthquake. This corresponds to employing a load factor of 0.75 for all elements of the load. The stresses due to wind during construction are small enough that the treatment of this aspect of the problem is not significant.

The use of a load factor of 1.0 for the accident pipe reactions, R_a, is in keeping with the original Division 2.

Consistent with the original Division 2, earthquake and wind for the abnormal/ severe environmental condition are not combined. This is not consistent with the 1977 version, however, it is apparent by inspection that the SSE forces are greater than wind forces such that the full earthquake alone rather than one-half SSE in combination with wind controls the design.

To summarize, the load combinations employed in the design of the Byron/Braidwood containment are consistent with the original version of Division 2, and for all practical purposes are consistent with the more recent 1977 version.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 General

The containment was analyzed using computer programs which are available in the Sargent & Lundy program library. These programs have all been validated by comparing results for selected problems with their closed-form solutions, when available, or by comparing the solution of a given problem with the solution of the same problem obtained from one or more previously validated programs. These programs have been used very effectively on similar containments and have been found to be appropriate for containment analysis. A more detailed description of the various programs named in these paragraphs can be found in Appendix D.

Throughout the analysis, the following areas of the containment have been given special attention:

a. the intersection between the base slab and the cylinder;

b. the intersection between the cylinder, dome ring, and dome;

c. the stresses around large penetrations;

d. the polar crane bracket area;

e. the behavior of the base slab relative to the underlying foundation material;

f. the stresses due to transient temperature gradients in the liner plate and concrete;

g. penetrations and points of concentrated loads; and

h. the buttresses.

The design and analysis procedure is in compliance with the requirements of Article CC-3000 of the ASME B&PV Code, Section III, Division 2.

The Unit 1 containment structure was also analyzed using Bechtel computer program BSAP to assess the effects of a temporary construction opening created in the containment wall to accommodate the steam generator replacement activities. A three-dimensional, finite-element model of the containment structure was used in the analysis. Special attention to the area of the opening was given to assess the state of stress in the containment wall during construction and after restoration of the temporary opening. A more detailed description of BSAP is included in Appendix D.

3.8.1.4.2 Containment

To account for the effects of axisymmetric loads such as dead loads, pressure, prestress forces and thermal loads, the containment was analyzed by two methods.

The first method of analysis used the thin-shell program SOR III. SOR III is a thin-shell-of-revolution program which permits consideration of elastic boundary conditions at the end boundaries and at boundaries between individual shells making up the complete shell. The complete containment is modeled including the dome, dome ring, and cylinder. The boundary between the base mat and the wall is assumed as fixed. This is justified in view of large relative stiffness of the 12 feet thick base mat which is founded on rock. The model used for this program is shown in Figure 3.8-24.

The loads applied to the shell model are centerline loads; therefore, consideration was given to the shift of the load from the actual place of application to the centerline of the shell. Figures 3.8-25 through 3.8-32 show the results of SOR III analysis of the containment shell. The values of moments and forces for the various load cases are indicated on these figures.

In the second method, the finite element program DYNAX was used to analyze the containment as a thick-shell-of-revolution, using quadrilateral finite elements including elements which represent the steel liner. Figure 3.8-33 shows the analytical model comprising the cylindrical shell, the dome ring and the dome. There is no stiffness attributed to the liner elements for any of the applied loads, except for the analysis of the effect of thermal liner expansion on the containment structure.

The rationale for employing both a thick-shell and a thin-shell analysis is as follows. In general, the shell diameter-to-thickness ratio is over 40 and thus falls well within the accepted definition of thin-shell theory. The thin-shell analysis is much more expedient and efficient in terms of the time and level of effort required. However, it is known that a thin-shell analysis does not adequately predict behavior in the vicinity of irregular transitions in thickness. A thick-shell model was prepared in order to ascertain the distance from ring beam and base mat at which thin-shell theory is applicable and to predict the stress state for design in the region in which the thin-shell theory is not applicable.

The thick-shell study was carried out using five rectangular elements through the shell thickness in the finite-element model, as shown in Figure 3.8-33. The computer program used for the analysis was DYNAX. The SOR III computer program was used for thin-shell analysis. This assumes a linear variation of stress through the shell thickness. This is also a finite-element program. Analytical results of both models are compared in Table 3.8-14 for the same internal pressure of 50 psi.

The correlation of forces given by the two methods is very good, except near the thickened wall, as expected. The various sections at which forces are compared are shown in Figure 3.8-86.

3.8.1.4.3 Base Slab

The base slab was analyzed using the SLSAP-1 (nonlinear) program. Figure 3.8-34 shows the analytical model comprising the base slab supported on rock foundation. For axisymmetric loads only one quarter of the slab is modeled, whereas for nonaxisymmetric loads (seismic) one-half of the slab is modeled. Foundation springs are modeled as nonlinear elements to the extent that they have compressive stiffness only and have no effect in uplift regions of the base slab.

A rotational spring is introduced at the top nodes of the containment wall as boundary elements. Therefore, the top nodes of the wall are free to displace but are restrained against rotation. In the analysis, advantage was taken of symmetry and a finite-element model prepared of one-half of the base mat. The in-plane wall stiffness of the containment wall was represented by a finite-element model of a portion of the wall 50 feet high. Rotational springs were employed at the top of the 50-foot height to represent the flexural rigidity of the portion of the shell omitted. The major portion of the shell's restraint of distortion at the periphery of the base mat is due to the membrane shear stiffness of the walls. Therefore, if the membrane shear deformations become small in the portion of the wall height employed to represent the shell, a sufficient height of wall has been employed to represent the effects of the entire containment. If the membrane shear deformations vanish, the displacement of points at a given elevation of the shell wall lie in a plane surface, i.e., a plane surface in the undeformed state remains a plane surface in the deformed state.

To verify that a 50-foot height of the containment wall adequately represents the membrane shear stiffness of the containment wall, the vertical displacements 50 feet above the base mat were compared as shown in Figure 3.8-87 with the position of a plane surface containing the displacements at the symmetry plane and orthogonal to the plane of symmetry. The load case considered included both accident internal pressure and earthquake. In the plot, the vertical ordinate is the displacement, and the horizontal ordinate is the angular distance to the node point at which the displacement occurs. Vertical displacements plotted in this fashion are defined by a cosine function. The solid line cosine function in the figure represents the displaced plane section. The broken line curve represents the displacements taken from the finite-element model.

It is apparent that 50 feet above the base mat the vertical displacements become essentially planar. This indicates that shear deformations are small and the use of the 50-foot wall height is justified.

3.8.1.4.4 Analysis of Areas Around Large Penetrations

The containment analyses using DYNAX and SOR III neglect the effects of the penetrations within the containment wall. To determine the local effects at larger penetrations such as the equipment hatch and main steam pipes, the areas around these penetrations were modeled by a finite element program, PLFEM-II. The element nodes lie along the centerline of the containment wall, thus accounting for curvature of the wall. The size of the model was chosen in order that the boundary conditions are compatible with those of an undisturbed cylinder. The change in thickness of the containment wall around the equipment hatch was represented by a change in element thickness.

3.8.1.4.5 Analysis of Areas Around Crane Brackets

The containment wall around the crane brackets was analyzed for the effects of local bracket loads. The wall was modeled for the axisymmetric finite element program DYNAX. The size of the elements was kept small in the vicinity of the brackets to account for the sharp gradients of the loads. The bracket loads induce in the wall meridian and hoop forces and moments as well as radial shears. For these nonaxisymmetric loads equivalent Fourier expansion with suitable number of harmonics was used.

3.8.1.4.6 Containment Liner

Stress in a typical liner panel prior to buckling of any panel is determined from the strain imposed on the liner by prestress, creep, shrinkage, and liner thermal strain restrained by the surrounding containment wall.

The liner anchorage system is analyzed using the computer program LAFD (Appendix D) which calculates force and deflection at anchorage points. The following cases, considered to produce the worst possible loading conditions on the anchorage system, are included in the analysis:

a. Case I - an initial inward deflection of 1/16 inch;

b. Case II - lower yield bound and 15% decrease in plate thickness of buckled panel;

c. Case III - upper yield bound and 15% increase in plate thickness in stable liner panels; and

d. Case IV - anchor spacing doubled to simulate failed or missing anchor (zipper effect). This case considers the postbuckling strength of this panel to be zero.

The anchor is designed so that if failure were to occur, it would be in the anchor and not in the liner.

3.8.1.4.7 Thermal Analysis

The containment was analyzed for both steady-state and transient thermal gradients.

The steady-state gradients are applied to each design section along with any appropriate axial forces and moments due to mechanical loads acting simultaneously with the thermal loads. The stresses in the concrete and reinforcing then are found by using TEMCO (Appendix D) which takes into account the extent of cracking of the section.

The moment resulting from thermal gradient is the only stress resultant that is permitted to change due to cracking. All other forces and moments are obtained from the various programs assuming the concrete to be a homogeneous material of appropriate stiffness.

For the transient gradient, an equivalent linear gradient is found by summing moments about the centerline of the section. The section is analyzed for this equivalent gradient by the same procedure used for the steady-state gradients.

3.8.1.4.8 Effects of Losses of Prestress

The effects of elastic shortening, creep and shrinkage of the concrete as well as friction and relaxation losses in the tendons have been included in the prediction of stress losses in the tendons and strains imposed on the steel liner. For the purposes of design, values for the parameters involved were assumed on the basis of published data and test results obtained during the construction of other containments. The assumptions concerning the values of the parameters were verified by means of laboratory and field tests during construction.

3.8.1.4.9 Buttress Analysis

The buttresses anchoring the hoop prestressing tendons were analyzed as a plane strain problem. A horizontal cross section was modeled by the finite element program PLFEM-II using quadrilateral elements. The model started at the centerline of the buttress and extended around the containment wall to the point where the boundary conditions were compatible with those of an undisturbed cylinder.

The increase in stiffness of the containment wall due to the buttress was also investigated.

3.8.1.4.10 Additional Reinforcing in Tendon Anchorage Zones

Additional reinforcing has been provided at all tendon anchorage zones located on the three buttresses and the dome ring. This reinforcing is designed and located as per requirements of ASME B&PV Code Division 2, Section III and is provided to resist bursting forces which exceed 0.10 times the seating force of the tendon required by code. The resulting reinforcing steel stress is less than 0.5 times the specified yield strength of the reinforcement.

3.8.1.5 Structural Acceptance Criteria

The acceptance criteria stated in this section is in full compliance with Article CC-3000 of the ASME B&PV Code, Section III, Division 2 and exceptions to this code are listed in the Standard Review Plan. The margin of safety implied by the use of the Code is best defined by the committee reports that lead to this code.

In the analysis of a reinforced concrete section, the strain in the reinforcing steel and concrete is assumed to be directly proportional to the distance from the neutral axis. The stress in the steel is limited to 90% of the yield stress. The tensile strength of the concrete is not relied upon to resist flexure or membrane tension.

3.8.1.5.1 Service Load Allowable Stresses

The design for the service load combinations in Table 3.8-3 was performed using the following allowable stresses:

Concrete
Compression:

a. Membrane compression $0.3f'_c$

b. Membrane plus flexural compression $0.45f'_c$

c. Local compression $0.6f'_c$

d. Compression under the tendons' end anchor bearing plates $0.6\, f_{ci} \sqrt[3]{\frac{A_1}{A_2}}$ but not to exceed f'_c

Radial Shear:

The design for radial shear is in accordance with Chapter 11 of ACI-318, with the following modifications. A 55% reduction factor is used on the permissible shear stress carried by the concrete calculated in accordance with Section 11.4 and the permissible stress in the shear reinforcement is $0.5f_y$ rather than f_y.

Tangential Shear

Prestressing force provided is such that under service load combinations tangential shear does not result in any principal tension.

Reinforcing Steel

a. Tension $0.5\ f_y$

b. Compression (load-resisting) $0.5\ f_y$

Tendon Stresses

a. Tension during stressing. $0.8\ f_{pu}$

b. Tension after anchoring $0.7\ f_{pu}$

3.8.1.5.2 Factored Load Allowable Stresses

The design for the factored load combinations in Table 3.8-3 was performed using the yield limit criteria given in this paragraph. The yield limit strength of the structure is defined for this design as the upper limit of elastic behavior of the effective load-carrying materials. The allowable stresses for this limit are defined as follows:

Concrete
Compression:

a. Membrane compression $0.6\ f'_c$

b. Membrane plus flexural compression $0.75\ f'_c$

c. Local compression $0.9\ f'_c$

Reinforcing Steel

a. Tension $0.9\ f_y$

b. Compression (load-carrying) $0.9\ f_y$

Radial Shear:

The design for radial shear is in accordance with Chapter 11 of ACI-318, using Section 11.4 to determine the permissible shear stress carried by the concrete. Although Section 11.4 is for nonprestressed concrete members, its use is conservative and is explained as follows. Prestressed concrete sections typically have higher concrete shear strength than nonprestressed concrete sections. Several critical sections in the containment structure were evaluated for radial shear using both Section 11.4 and Section 11.5 of ACI-318-71 and it was found that the allowable concrete shear strength per Section 11.5 was, in general, at least 50% higher than that allowed under the provisions of Section 11.4. Therefore, Section 11.4 was used for reasons of conservatism and expeditious design. The term N/A is taken as the computed membrane stress occurring simultaneously with V_u.

Tangential Shear:

The principal stresses resulting from tangential shear stresses and membrane stresses have been computed for all load combinations. Principal tension less than $3\sqrt{f'_c}$ is assumed to be resisted by concrete. For principal tension greater than $3\sqrt{f'_c}$ reinforcing is provided to carry the total tensile force.

The use of $3\sqrt{f'_c}$ as a limiting value for principal tensile stresses resulting from tangential shear stresses and membrane stresses is not an exception to the code and is more conservative than the value of $4\sqrt{f'_c}$ suggested by the NRC in the Standard Review Plan (p. 3.8.1-9).

The ASME B&PV Code, Section III, Division 2, does state that concrete tensile strength shall not be relied upon to resist flexural and membrane tension. However, the term membrane stress, defined in CC-3136.1, refers to hoop or meridional stress, not principal stress, as is pointed out in footnote 1. Similarly, the term bending stress should be interpreted to mean the hoop and meridional bending stress.

In light of the definition of the terms, the intent of the code is to prohibit use of concrete tensile strength to resist hoop and meridional tensile membrane and flexural stresses. The design has been accomplished in accordance with this requirement.

Since Division 2 does not address the topic of tangential shear in prestressed concrete containments, a criterion was determined based on a review of general usage, existing codes and the NRC recommendation. The use of the tensile strength of concrete to resist principal tensile stresses is universally employed although not always expressed in these terms. In fact, tensile capacity of concrete is required for reinforced concrete to exist as a structural concept. Most limitations on shear stresses in concrete are actually indirect limitations on principal tensile stresses. Thus, recognition of shear strength in concrete not provided with shear reinforcement constitutes acceptance of tensile strength in the concrete. Since all codes recognize some degree of shear strength in concrete without shear reinforcement, there is a precedent for taking advantage of the tensile strength of concrete. Precedent can also be cited for direct recognition of concrete tensile strength. For example, ACI 318-1977, Section 15.11, permits flexural tensile concrete stresses in unreinforced footings and pedestals of $5\varphi \sqrt{f'_c}$, and Chapter 18 permits extreme fiber stresses in tension ranging from $3 \sqrt{f'_c}$ to $12 \sqrt{f'_c}$ in prestressed members. Thus it is apparent that there is no conceptual objection to utilizing the tensile strength of concrete per se.

Therefore, the NRC recommendation presented with regard to CC-3411.5 in the Standard Review Plan, Chapters 18 and 19 of ACI 318-1971, and past containment practice were reviewed and a conservative criterion selected. The most conservative value of tensile strength for prestressed concrete given in Chapter 18 of ACI 318-1971, $3 \sqrt{f'_c}$, was employed as the measure of tensile strength to resist tensile principal stresses due to tangential shear.

The allowable stress parameters for Subsection 3.8.1.5.1 and this subsection are defined as:

f'_c = Specified minimum compressive strength of concrete.

f_{ci} = Compressive strength of concrete at time of initial prestress.

A_1 = Maximum concrete surface area perpendicular to the tendon axis geometrically similar to and concentric with the contact area of the end anchor bearing plate, which does not overlap corresponding areas of adjacent tendon anchorages.

A_2 = Total surface area of end-anchor bearing plate neglecting the loss of area from the tube.

f_{pu} = Ultimate strength of prestressing steel.

f_y = Minimum guaranteed reinforcing steel yield strength.

Interaction diagrams based on these acceptance criteria are generated using the computer program COLID (Appendix D) at the sections illustrated in Figure 3.8-35, Sheet 1. On the interaction diagrams (Figure 3.8-35, Sheets 2 through 27), load levels resulting from the analysis described in Subsection 3.8.1.4 are plotted for various significant loading combinations. For each combination, the variation in moment due to temperature changes are included. These diagrams clearly illustrate the relationship of containment capacity to the imposed loads.

3.8.1.5.3 Allowable Stresses and Strains for Liner and Anchorages

Table 3.8-5 gives the allowables for the containment liner and liner anchorages for mechanical as well as self-limiting loads. The allowable stresses and strains for the liner plate are limited to values that have been shown to provide leaktight vessels.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

3.8.1.6.1 General

Materials and quality control requirements for containment elements that serve pressure vessel functions are listed in Appendix B. The physical properties of these materials are listed in appropriate section of Appendix B.

3.8.1.6.2 Post-Tensioning Sequence

Design of the prestressing system includes full provision for the following required stressing sequence which is adhered to in order to minimize unbalanced loads and differential stresses in the structure; this sequence is based on the consideration that entire primary containment structure is complete before post-tensioning (for exception see b. below), and that stressing of each tendon is done in a single stage.

The following stressing sequence was used:

a. The vertical and dome stressing are independent, and were performed without regard to sequence between them.

b. Dome stressing was performed before the closing of construction openings in some cases, and after the closing of the latter in other cases.

c. Horizontal wall tendons were stressed last.

The vertical wall tendons were stressed from the top and were performed by using a minimum of three jacks spaced evenly about the structure. Stressing positions were alternated to prevent concentrations of multiple stressed tendons adjacent to multiple unstressed tendons.

Dome stressing was performed by using a minimum of six jacks simultaneously with two jacks working on a single tendon in each of the three azimuths of the three-way roof system. Dome stressing positions were alternated to prevent large concentrations of stressed tendons.

One complete ring of three horizontal tendons is stressed before moving to another level. Stressing operations are alternated, progressing from top or bottom, taking every third ring of tendons on each successive trip. For restoration of the Unit 1 containment opening following steam generator replacement, the vertical and horizontal tendons were stressed in a continuous sequence.

3.8.1.7 Testing and Inservice Surveillance Requirements

3.8.1.7.1 Code Compliance Requirements

There are two basic structural tests that are performed to check the containment integrity: 1) a structural acceptance test, and 2) a leak rate test. In addition, inservice testing of the containment is performed to provide continuing check on structural adequacy. Descriptions of the inservice surveillance programs have been incorporated into the Pre-Stressed Concrete Containment Tendon Surveillance Program.

3.8.1.7.2 Preoperational Testing

3.8.1.7.2.1 Preoperational Leak Rate Testing

A discussion of preoperational leak rate testing is contained in Subsection 6.2.6.1.

3.8.1.7.2.2 Structural Acceptance Test

The structural acceptance test was performed after the containment was complete with liner, concrete structures, all electrical and piping penetrations, equipment hatch, and personnel lock in place. The structural acceptance test did conform to the requirements of the ASME Code Section III, Division 2, ACI 359 Article CC-6000.

Corrections will be made in the deflection measurements to account for the discrepancies due to the effects of creep, shrinkage, temperature, and variation in modulus of elasticity due to aging of the concrete. The arrangement of the instrumentation during the structural acceptance test is shown in Drawing S-1051. The predicted containment deflections during the structural acceptance test are given in Table 3.8-6.

Stress and strain measurements are not necessary, due to the fact that prior to testing this containment, similar containments at the Farley, Arkansas, Millstone, and Summer Stations have been tested, thereby making this containment a nonprototype.

3.8.1.7.2.3 Containment Pressure Test

Subsequent to restoration of the containment opening following steam generator replacement, a containment pressure test as required by ASME Section XI, 1992 Edition with 1992 Addenda, Subsection IWL-5000 was performed to a pressure equal to the accident pressure. Qualified inspectors performed a visual examination of the containment wall concrete surface in the area of the opening to ensure there was no evidence of conditions indicative of damage or degradation. A record of the examination is maintained to document the preservice condition of the concrete surface of the containment wall in the area of the opening.

3.8.1.7.3 Inservice Surveillance

3.8.1.7.3.1 Inservice Leak Rate Testing

A discussion of inservice leak rate testing is contained in Subsection 6.2.6.1.

3.8.1.7.3.2 Inservice Tendon Surveillance Program

The inservice tendon surveillance program complies with the requirements of the 2007 Edition with the 2008 Addenda at Byron and the 2013 Edition at Braidwood of the ASME Boiler and Pressure Vessel Code, Section XI, Division I, Subsection IWL, and the modifications included in 10 CFR 50.55a(b)(2)(viii), "Examination of concrete containments." Predicted lift-off forces are determined consistent with the recommendations of Regulatory Guide 1.35.1.

End Anchorage Concrete Surveillance

See the Pre-Stressed Concrete Containment Tendon Surveillance Program for a discussion of end anchorage concrete surveillance.

3.8.1.8 Containment Ultimate Capacity

An analysis of the ultimate capacity of the Byron/Braidwood containment structure has been performed. It is described below.

Introduction

The Byron/Braidwood containment structures are post-tensioned concrete shells made up of a cylinder with a shallow dome roof and a reinforced flat concrete slab. The entire structure is lined on the inside with a steel plate which serves as a leak-tight membrane. The containment wall is prestressed with vertical and hoop tendons. The dome is post-tensioned with three groups of tendons anchored 120° to each other. Containment penetrations are provided for process piping, access hatches and electrical and instrument lines. A detailed description and drawings of the containment and its penetrations are given in previous subsections.

Design Pressure

The design pressure of the containment is 50 psi as stated in Table 3.8-4.

Original Design Criteria

The containment analysis and design were described previously and are in accordance with the ASME Code Section III, Division 2, issued April 1973, and the exceptions and additions to this code listed in the Standard Review Plan.

Probable Failure Modes

As internal pressure builds up, failure of the pressure boundary can result from the following causes:

a. failure of reinforcing steel,

b. failure of concrete in secondary compression,

c. failure of post-tensioned tendons,

d. failure in flexural shear at discontinuities,

e. failure in peripheral shear around penetrations,

f. failure of steel pressure retaining components,

g. buckling of steel pressure retaining components,

h. separation of penetration assembly and the retaining ring bolt of that penetration, and

i. failure of the liner.

Criteria for Ultimate Capacity

For the purpose of the study, the ultimate capacity is defined as the attainment of any one of the following limits:

a. Tensile yielding of post-tensioning tendons in conjunction with yielding of the reinforcing resulting in a state of general yield of any section where the yield stress for the tendon corresponds to 1% strain;

b. Maximum compressive strain of 0.003 in./in. in concrete;

c. Flexural and peripheral shear capacity of containment wall as per ACI 318-71, Section 11.15, utilizing reinforcing stresses up to yield;

d. Yielding of electrical and mechanical penetration components as per ASME Code allowables for the faulted condition;

e. Yielding of equipment hatch and personnel air locks as per actual yield strength defined from certified material test reports; and

f. Overcoming of retaining ring bolt preload of the penetration and the penetration assembly.

Analysis Details

In order to determine the ultimate pressure capacity of the concrete shell, an axisymmetric thin shell "DYNAX" model (as shown in Figure 3.8-88) was utilized. UFSAR Appendix D includes a description of DYNAX. This computer program is capable of representing the cracking of concrete, yielding of the reinforcing steel and tendons, occurring over various pressures. The model consists of 87 nodes and 192 elements. Separate elements are used to represent the unbonded post-tensioning tendons. Figure 3.8-89 shows a schematic diagram of a laminated shell element used to represent the reinforcing and the concrete. Minimum specified yield strength for both the reinforcing and the prestressing steel were used. Figures 3.8-90 through 3.8-92 show plots of the assumed material stress-strain relationships. The foundation was represented by distributed springs attached to the base mat. The response of the concrete containment at 120 psi is shown in Figure 3.8-93.

Mechanical and electrical penetration analysis were performed considering internal pressure loading. The analysis was in accordance with the ASME Code Section III, Subsection NE-3200, design by analysis.

The personnel locks and equipment hatch analysis was performed using linear elastic methods in accordance with the ASME Code Section III, Subsection NE-3200, design by analysis.

Static Pressure Capacities

The ultimate containment internal pressure capacity of the containment equipment hatch and personnel locks is 148.9 psi. This corresponds to the yielding of the personnel lock bulkhead plate panel above and below the doors.

The ultimate containment internal pressure capacity of the concrete shell is 125 psi. This corresponds to the initiation of yielding of the (hoop) post-tensioning tendons in conjunction with yielding of reinforcing near the mid-height of the containment wall. The peak liner strain is well within the ASME Code allowables for the factored load condition.

The ultimate containment internal pressure capacity of the electrical penetration is 108 psi. At this pressure a retaining ring bolt preload in a penetration assembly is overcome, and separation may occur.

The ultimate containment internal pressure capacity of the mechanical penetration is 182 psi.

Dynamic Capacity

The time rise of the LOCA pressure in the containment is shown in Figures 6.2-1 through 6.2-6a. This loading is essentially static in nature. As discussed in Subsection 6.2.5, studies performed in support of the revision to 10 CFR 50.44 determined that hydrogen release during design basis accidents is not risk significant and would not lead to early containment failure. Therefore the design-basis loss-of-coolant accident hydrogen release was eliminated from 10 CFR 50.44. In addition, the containment atmosphere mixing function discussed in Subsection 6.2.5.2.3 prevents local accumulation of combustible or detonable gases that could threaten containment integrity or equipment operating in a local compartment following a loss-of-coolant accident. Therefore, due to the absence of postulated dynamic loads, a dynamic pressure analysis was not necessary.

3.8.2 Steel Containment and ASME Class MC Components

This section pertains to the ASME Class MC components that are a part of the primary containment vessel described in Subsection 3.8.1. The MC components include the equipment hatch with integral personnel lock, emergency personnel lock, and piping and electrical penetrations. The MC components are also discussed in Subsection 3.8.1.1.3.5.

3.8.2.1 Description of ASME Class MC Components

3.8.2.1.1 Personnel Lock with Equipment Hatch

An equipment access hatch and integral personnel airlock is provided for access to the interior of the containment (refer to Figure 3.8-38). The equipment hatch is provided for access to the containment during shutdown. The transfer of equipment and components through the containment wall is accomplished through this opening. The equipment hatch is a round barrel frame with dished head access hatch; the cylindrical personnel lock is built integrally into the dished head. The dished head and integral personnel airlock are fully removable with a lifting device located near center of gravity of the entire removable assembly. The integral personnel airlock consists of two airtight doors in series which are mechanically interlocked so that one door cannot be opened unless the second door is sealed. If needed, the mechanical interlock can be overridden by use of a special procedure provided.

Either door may be operated from inside the containment, inside the personnel lock, or outside the containment. Each door is equipped with a Pressure Equalizing Device which equalizes pressure on both sides of the door before the door can be operated. Pressure Equalizing Device controls are located next to the door controls. Pressure Equalizing Devices are interlocked so that only one Pressure Equalizing Device can be opened at a time and only when the opposite door is closed and sealed. Indicators are provided to indicate the position of doors and Pressure Equalizing Devices.

The type B test for the airlock door seals shall be performed at a pressure between 3 and 12 psig either as described in Section III.D.2.biii of 10 CFR 50, Appendix J or by installing a continuous pressurization source to the airlock door seals that will be monitored by a flowmeter and alarm. The doors are manually operated.

3.8.2.1.2 Emergency Personnel Airlock

The emergency personnel airlock (refer to Figure 3.8-39) consists of two gasketed doors in series which are mechanically interlocked such that one door cannot be opened unless the second door is sealed. If needed, the mechanical interlock can be overridden by use of special procedure provided. The doors are manually operated. Either door can be operated from inside the containment, inside the airlock, or outside the containment. Each door is equipped with a Pressure Equalizing Device for equalizing pressure on both sides of the door before the door can be operated. Pressure Equalizing Devices can be operated from the same location at which the associated door can be operated. Pressure Equalizing Devices are interlocked so that only one Pressure Equalizing Device can be operated at a time, and only when the opposite door is closed and sealed. Indicators are located on the outside of the airlock at each door to show whether the opposite door and its Pressure Equalizing Devices are open or closed. The airlock can be pressure tested at any time without interfering with the normal operation of the plant. Provision is made to leak test the door seals on both doors.

3.8.2.1.3 Penetrations

Penetrations are provided to extend process piping and electrical conduits through the containment wall. Process piping penetrations act as process pipe supports and are capable of withstanding the following design conditions;

a. peak transient temperatures,

b. forces caused by fluid impingement from largest pipe,

c. thermal and mechanical stresses during operation, and

d. design pressure and operating pressures and temperatures.

The arrangement of the containment penetrations is shown in Figure 3.8-14, and the sizes and locations are listed in Table 3.8-1.

3.8.2.1.3.1 Penetration Types

Penetrations are of two major types:

a. instrument and process pipe penetrations, and

b. electrical penetrations.

3.8.2.1.3.1.1 Instrument and Process Pipe Penetrations

Instrument and process pipe penetrations are of four types, as shown in Figures 3.8-40 through 3.8-42 and Figure 6.2-30. For all process lines penetrating the containment, the sleeve is embedded into the concrete. Air gaps are provided around all pipes. Insulation and cooling coils are provided around hot pipes to reduce thermal stress in the containment during normal operations.

In addition to their function as primary containment barrier, the penetrations serve as anchors to the pipes and are designed to carry the loads associated with a postulated pipe break. Thermal growth and movement is absorbed in the piping system. For all three penetration types, the penetration sleeve is anchored in the wall and extends just inside the containment wall liner. For Type I penetrations, the head fitting and a section of the process pipe is one forged piece. For Type II, the head fitting is forged and is welded to the process pipe by a full penetration weld. The head fitting for Type III penetrations is a flat plate attached to the process pipe by a full penetration weld.

At the time of design, the determination of the penetration type is made based on the magnitude of the applicable loads.

3.8.2.1.3.1.2 Electrical Penetrations

Electrical penetrations assemblies are used to extend electrical conductors through the pressure boundary of the containment structure. Electrical penetrations are functionally grouped into low voltage power, low voltage control cable penetration assemblies, medium voltage power cable penetration assemblies and shielded cable penetration assemblies. Figure 3.8-43 shows a typical electrical penetration assembly in place within the containment wall. Hermetic seals between each conductor and header plates are obtained by the use of high strength, high temperature epoxy. An assembly is sized to be inserted in schedule 80 penetration nozzles.

3.8.2.1.3.2 Component Classification

The penetration sleeve in its entire length, is designed as an MC component in accordance with Subsection NE of the ASME B&PV Code, Section III (including applicable code cases and addenda).

The portion of the containment penetration assembly that consists of the head fitting, which is directly exposed to process pipe pressure (Head Fitting Type I only, Figure 3.8-40) is considered to be a piping component having the same classification as the process pipe and, as such, it is designed in accordance with Subsection NB, NC, or ND of the ASME Code, Section III, as applicable. All other head fittings are classified as MC components.

3.8.2.2 Applicable Codes, Standards, and Specifications

The following applicable codes were used:

a. AISC Manual of Steel Construction.

b. ASME Boiler and Pressure Vessel Code, Sections II, III, IX, and XI (including Applicable Code Cases and Addenda).

3.8.2.3 Loads and Loading Combinations

The loads and loading combinations for the Class MC components, except for the instrument and process piping penetrations, are given in Table 3.8-7. These loads and their combinations conform to Article NE-3000 of the ASME Code, Section III.

The load combinations from Table 3.8-7 have been compared to those in SRP Section 3.8.2.

The load combinations for class MC components from UFSAR Table 3.8-7 and SRP 3.8.2 correlate as shown in the following:

Load Comb No.	SRP 3.8.2	Load Comb No.	UFSAR Table 3.8-7
(1)	$D + L + P_t + T_t$	(2)	$D + L + P' + T_o$
(2)	$D + L + T_o + R_o$	(3)	$D + L + R_o + T_o + P_o + P_e$
(3)	$D + L + T_o + R_o + E$	(4)	$D + L + R_o + T_o + P_o + P_e + E$
(4)	$D + L + T_a + P_a + R_a + E$	(8) or (10)	$D + L + P_e + E + T_a + P_a + R_a$ or $D + L + P_e + E + R_r + T_a + P_a + R_a$

Load Comb No.	SRP 3.8.2	Load Comb No.	UFSAR Table 3.8-7
(5)	$D + L + T_e + P_e + R_e + E$	(4)	$D + L + R_o + T_o + P_o + P_e + E$
(6)	$D + L + T_a + R_a + P_a + E'$	(11)	$D + L + P_e + R_r + T_a + P_a + R_a + E'$
		or (12)	$D + L + P_e + R_r + T_a + P_a + R_a + E'$
(7)	$D + L + T_e + P_e + R_e + E'$	(7)	$D + L + R_o + T_o + P_o + E'$
(8)	$D + L + T_a + R_a + Y_r + Y_j$ $Y_m + E' + P_a$	(11)	$D + L + P_e + R_r + T_a + P_a + R_a + E'$
		or (12)	$D + L + P_e + R_r + T_a + P_a + R_a + E'$
(9)	$D + L + E + F_1$		None

The allowable stresses for the design of the class MC components covered under Table 3.8-7 can be determined from either paragraphs NE-3131 (a), (b) and (d), or NE-3131 (c) of ASME, Section III, Division I Code. (The applicable edition of ASME code is 1971, with coverage through the Summer Addenda, 1973. This is referenced in UFSAR Subsection 3.8.2.5.1.) The allowable stresses for the corresponding load combination equations are shown in the following:

Table 3.8-7 Load Combination Equation No.	ASME, Section III, Division I NE-3131 Paragraph
1	NE-3131 (a), (b) and (d)
2	NE-3131 (a), (b) and (d)
3	NE-3131 (a), (b) and (d)
4	NE-3131 (a), (b) and (d)
5	NE-3131 (a), (b) and (d)
6	NE-3131 (a), (b) and (d)
7	NE-3131 (c)
8	NE-3131 (a), (b) and (d)
9	NE-3131 (a), (b) and (d)
10	NE-3131 (a), (b) and (d)

Table 3.8-7 Load Combination Equation No.	ASME, Section III, Division I NE-3131 Paragraph
11	NE-3131 (c)
12	NE-3131 (c)

The SRP combination including post-LOCA flooding does not govern the design of the MC components. LOCA pressure produces larger loads on the penetrations because the surface elevation for the flood is evaluated up to 6 feet 3 inches above the base mat.

There are no deviations from the SRP Section 3.8.2 in the design of the metal portions of the containment.

3.8.2.3.1 Loads for Instrument and Process Pipe Penetrations

The forces and moments imposed at the piping penetration assembly boundaries are due to the following:

a. internal and external operating and design pressures and temperatures

b. process pipe reactions due to (as applicable):

   1. weight,
   2. operating basis earthquake (OBE),
   3. safe shutdown earthquake (SSE),
   4. thermal expansion,
   5. relative seismic displacements,
   6. hydraulic transients,
   7. main steam SRV, and
   8. pipe break and jet impingement.

3.8.2.3.2 Loading Combinations for Instrument and Process Piping Penetrations

a. Design Conditions

   1. design pressures and temperatures plus
   2. Load Cases (1) + (2) + (6), from Subsection 3.8.2.3.1(b).

b. Normal and Upset Conditions

1. For Expansion Stress Evaluation

Load Cases (4) + (5) from Subsection 3.8.2.3.1(b).

2. For Primary-plus-Secondary Stress Evaluation

a) operating pressures and temperatures, plus

b) Load Cases (1) + (2) + (4) + (5) + (6) + (7) from Subsection 3.8.2.3.1(b), plus thermal gradients.

c. Emergency Conditions

1. Operating pressures and temperatures plus

2. Load Cases (1) + (6) + (7) from Subsection 3.8.2.3.1(b).

d. Faulted Conditions

1. Operating pressures and temperatures plus Load Case (8) from Subsection 3.8.2.3.1(b). SSE is a faulted condition load; however, see Table 3.8-12, Note 1.

e. For Fatigue Evaluation

Same as Subsection 3.8.2.3.2(b)(2)(b).

f. Testing Conditions

In accordance with NB-3226, NB-6222, and 6322 or (as applicable) NE-6222 and NE-6322 of the ASME Code, Section III.

3.8.2.4 Design and Analysis Procedures

3.8.2.4.1 Access Hatches and Electrical Penetrations

The personnel lock and equipment hatch, emergency personnel lock and the electrical penetrations are designed as pressure retaining components. The portions of the sleeves not backed by concrete are analyzed and designed according to the provisions of Subsection NE of Section III of the ASME B&PV Code.

3.8.2.4.2 Instrument and Process Piping Penetrations

The entire penetration assembly, including sleeve, head fitting, and attached portion of pipe, is designed for the loads described in Subsections 3.8.2.3.1 and 3.8.2.3.2 by the finite element computer program PENAN (see Appendix D). The boundary conditions for the finite element model are taken as fixed against all degrees of freedom at the outside face of the containment wall. PENAN also evaluates thermal gradient for axisymmetric configuration. The final stress analysis of the piping penetration assemblies, including metal fatigue evaluation, is performed by PENAN.

3.8.2.5 Acceptance Criteria

3.8.2.5.1 Access Hatches and Electrical Penetrations

The access hatches and electrical penetrations are designed as Class MC components according to Subsection NE of Section III of ASME B&PV Code (including applicable code cases and addenda).

These components are designed for the loads and load combinations given in Subsection 3.8.2.3 for the allowables given below. Load combinations 1, 2, 3, 4, 5, and 9 in Table 3.8-7 are designed according to the allowable stresses specified in Paragraphs NE-3131 (a), (b) or (d). Loading combinations 6, 10, 11, and 12 in Table 3.8-7 are designed according to the allowable stresses specified in paragraph NE-3131(c) 1 and 2.

3.8.2.5.2 Instrument and Process Piping Penetration Assemblies

The instrument and process piping penetrations, with the exception of the electrical penetration cooling coils, are designed according to 1971 version of the ASME B&PV Code, Section III, and are checked for compliance with the 1974 edition of the code. In this subsection, reference is made to paragraphs in the 1974 edition to indicate applicable acceptance criteria. Cooling coils meet the requirements of ASME Section VIII.

3.8.2.5.2.1 Loading Conditions

Containment piping penetration sleeves and head fittings meet all stress limits associated with the worst loading combinations for design, normal, upset, emergency, faulted, and testing component conditions, in accordance with the requirements and provisions of Division 1 of the ASME Code, Section III.

3.8.2.5.2.2 Loading Combinations and Stress Limits for Penetration Sleeves and Head Fittings

The load components, loading combinations, and stress limits corresponding to each of the loading conditions stated in Subsection 3.8.2.5.2.1 are defined in Subsections 3.8.2.5.2.2.1 through 3.8.2.5.2.2.5 and are summarized in Tables 3.8-12 and 3.8-13.

3.8.2.5.2.2.1 Design Conditions

The penetration sleeves and head fittings are evaluated for the worst combination of design pressures and temperatures plus loads due to: weight, operating basis earthquake (OBE), hydraulic transients, as applicable (see Table 3.8-12).

Under these loading combinations, the head fittings and penetration sleeves meet all applicable stress requirements set forth in Paragraphs NB-3221 and NE-3221 of the ASME Code, Section III, respectively (see Table 3.8-13).

3.8.2.5.2.2.2 Normal and Upset Conditions

The penetration sleeves and head fittings are evaluated for the worst combination of maximum operating pressures and temperatures, plus thermal transients plus loads due to: weight, operating basis earthquake (OBE), thermal expansion, relative seismic displacements, hydraulic transients, as applicable (see Table 3.8-12).

Under these loading combinations, the head fittings and penetration sleeves meet all applicable stress requirements set forth in Paragraphs NB-3222 and NB-3223 of Section III for the head fittings, and in Paragraph NE-3222 of Section III for the penetration sleeves (see Table 3.8-13).

3.8.2.5.2.2.3 Emergency Conditions

The penetration sleeves and head fittings are evaluated for the worst combination of maximum operating pressures and temperatures plus loads due to: weight and hydraulic transients, as applicable (see Table 3.8-12).

Under these loading combinations, the head fittings and penetration sleeves meet all applicable stress requirements set forth in Paragraph NB-3224 of Section III (see Table 3.8-13).

3.8.2.5.2.2.4 Faulted Conditions

The penetration sleeves and head fittings are evaluated for:

a. The maximum operating pressures and temperatures with the worst combination of the following loads, applied at the outer face of the head fitting:

    1. axial load equal to (2) (1.26)(P)(A), where P is the maximum operating pressure, and A is the process pipe flow area;

    2. shear load equal to the axial load, above;

    3. bending moment equal to the limit bending moment capacity of the process pipe; and

4. torsional moment equal to the elastic bending moment capacity of the process pipe.

b. the process pipe maximum operating pressure applied in the annulus between the process pipe and the penetration sleeve.

Under each of these loading cases, the head fittings and penetration sleeves meet all applicable stress requirements described in F-1324.1, F-1324.6, and Table F-1322 of Appendix F of the ASME Code, Section III, for system inelastic - component elastic analysis (as noted in Table 3.8-12, the SSE load is not required for this type of analysis). These stress requirements are summarized in Table 3.8-13.

3.8.2.5.2.2.5 Testing Conditions

Penetration head fittings are evaluated for testing conditions and satisfy the requirements specified in Paragraphs NB-3226, NB-6222, and NB-6322 of the ASME Code, Section III.

Penetration sleeves are evaluated for testing conditions and satisfy the requirements specified in Paragraphs NE-6222 and NE-6322 of the ASME Code, Section III.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

Material requirements for steel elements that serve pressure vessel functions are listed in Appendix B-6. The physical properties of these materials are listed in Table 3.8-8. The penetration components mentioned in Appendix B-6 fully comply with the materials specified in Article NE-2000 of the ASME Code Section III, Division 1, 1971, Summer 1973 addendum.

Fabrication and installation requirements of Article NE-4000 of the Code, as well as with the provisions of Article NE-5000, are in compliance with examination of components. Standard construction techniques are used in the fabrication and erection of MC components.

3.8.2.7 Testing and Inservice Surveillance Requirements

3.8.2.7.1 Structural Acceptance and Initial Leak Rate Tests

All MC components are tested for their structural acceptance and leak rate at the same time of the containment tests described in Subsection 3.8.1.7.2. Type B leak rate tests are performed on all hatches by pressurizing the plenum between the double gaskets.

In addition, the personnel airlock and emergency personnel airlock is shop tested according to the following procedure:

a. Initial soap bubble test: interior of airlock is pressurized to 5 psig and soap bubble leak test is performed on all welded joints, penetrations and nozzles, and all double-compression seals around doors.

b. Overpressure test: airlock interior is pressurized to 57.5 psig and held for 1 hour.

c. Second soap bubble test: pressure reduced to 50 psig and second soap bubble test is performed.

d. Initial leak rate test: the pressure is held at 50 psig and maximum leakage did not exceed 1.0% of the volume of airlock in 24 hours.

3.8.2.7.2 Inservice Surveillance

Periodic leak rate tests on the containment, including the MC components, are performed as described in Subsection 3.8.1.7.

3.8.3 Containment Internal Structures

3.8.3.1 Description of Containment Internal Structures

Internal structures of the containment support and shield major nuclear steam supply equipment, their associated pipings and auxiliary equipment. They also support various gallery floors, contain water for refueling, and support the polar crane.

The internal structures include the following:

a. reactor vessel support,

b. steam generator supports,

c. reactor coolant pump supports,

d. pressurizer support,

e. primary shield wall and reactor cavity,

f. secondary shield wall,

g. reactor refueling pool,

h. interior base mat, and

i. polar crane support.

Drawings M-7 through M-11 give an overall plan of the containment including the internal structures. Drawings M-15 and M-18 show sections of the containment structure.

Containment building sections in the east-west and north-west directions, primary shield wall, and NSSS component enclosure plans are shown in Figures 3.8-46 through 3.8-51. The NSSS component supports are shown in Figures 3.9-4 through 3.9-10.

3.8.3.1.1 Reactor Support

Reactor support is described in Subsection 3.9.3.4.1.1.

3.8.3.1.2 Steam Generator Support

Steam generator support is described in Subsection 3.9.3.4.1.3.

3.8.3.1.3 Pressurizer Support

Pressurizer support is described in Subsection 3.9.3.4.1.2.

3.8.3.1.4 Reactor Coolant Pump Support

Reactor coolant pump support is described in Subsection 3.9.3.4.1.4.

3.8.3.1.5 Primary Shield Wall and Reactor Cavity

The primary shield wall is a circular cylindrical reinforced concrete structure. It forms the reactor cavity and also supports and shields the reactor vessel. It has an outside diameter of 34 feet and an inside diameter of 25 feet for the portion above the reactor support, and 17 feet 1 inch for the portion below the reactor support.

The primary shield wall is supported by the interior base mat and anchored to the 12-foot thick containment base mat for uplift load. This anchorage is described in Subsection 3.8.1.1.2.3.

3.8.3.1.6 Secondary Shield Wall

The secondary shield wall shields and provides lateral support for steam generators, reactor coolant pumps and pressurizer. Below the operating floor, the secondary shield wall is an irregular 12 sided polygonal structure 4 feet 6 inches thick and approximately 49 feet high. Above the operating floor it separates into five enclosure compartments around steam generators and pressurizer varying in thickness from 5 feet 0 inch to 2 feet 0 inch.

The secondary shield wall is supported by the interior base mat and for the portion above the operating floor it is partially supported by the refueling pool walls. The secondary shield wall is anchored to a 12 foot thick containment base mat. See Subsection 3.8.1.1.2.3 for this anchorage.

3.8.3.1.7 Reactor Refueling Pool

The reactor refueling pool contains water during refueling. The pool walls also support miscellaneous gallery floors, the operating floor, and secondary shield wall above the operating floor.

The bottom of the reactor refueling pool is divided into two levels. The upper level is at elevation 399 feet 1-1/2 inches and serves as the upper internals laydown area, while the lower level is at elevation 390 feet 0 inch and serves the lower internals laydown area during refueling operations. The pool wall thickness varies from 3 feet 6 inches to 5 feet 0 inch. The interior face of the pool walls and floor is lined with 3/16 inch austenitic stainless steel plate. During refueling the pool will be flooded to approximately elevation 424 feet 6 inches.

For Byron, the impact of a fuel drop on low profile nozzle hatch covers has been analyzed with mathematical calculation. The nozzle hatch covers are determined to be able to withstand the impact of a dropped fuel assembly without damage or leakage of any nozzle hatch cover. Fuel damage would not exceed that of a fuel drop elsewhere in the fuel transfer area or reactor cavity.

3.8.3.1.8 Interior Base Mat

The interior base mat is a 3-foot thick reinforced concrete slab 140 feet in diameter. It serves as a foundation for all internal structures except the polar crane supports. The base mat is poured on the bottom containment liner and is not doweled into the containment base mat except along the bases of primary shield wall, secondary shield wall, and pool walls.

3.8.3.1.9 Polar Crane Supporting Systems

Polar crane rides on the crane rail which is anchored on top of 18 circular curved crane girders. Crane girders are supported by 36 crane brackets which are cantilevered from and embedded in the containment shell. Each crane girder is supported on three brackets. The ends of the girder are provided with a sliding connection, so that the axial loads of the girder are resisted entirely at the middle bracket. A closed section with two webs has been provided for the middle bracket, whereas end brackets consist of an open section with a single web.

3.8.3.1.10 Intermediate Floors and Galleries

The intermediate floors and galleries serve the dual function of providing access to electrical and mechanical components and to structural support for these items. They consist of structural framing supported by the primary shield wall, secondary shield wall, and structural steel columns. Steel gratings as well as concrete on decking span between framing beams. Slotted connections are provided on various beams to allow thermal movements of the beams, thus, no thermal loads are developed in the beams.

3.8.3.2 <u>Applicable Codes, Standards, and Specifications</u>

This section lists codes, standards, specifications, regulatory guides, and other accepted guidelines which are adopted to the extent applicable, in the design and construction of the containment internal structures. The codes, standards, and specifications are listed in Table 3.8-2, for Regulatory Guides see Appendix A.

3.8.3.2.1 <u>Reactor Support, Steam Generator Support, Pressurizer Support, and Reactor Coolant Pump Support</u>

See Subsection 3.9.3.4 for the applicable codes, standards, and specifications for these supports.

3.8.3.2.2 <u>Internal Structures Other Than Those Listed In Subsection 3.8.3.2.1</u>

Those applicable are 1 through 21, and 24.

3.8.3.3 <u>Loads and Loading Combinations</u>

Tables 3.8-9 and 3.8-10 list the loads and loading combinations used in the design of containment internal structures. A description of load categories and a definition of loads is given in Table 3.8-4.

These loads and loading combinations comply with those portions of ACI-349 which are based on ACI-318.

The load categories defined in this section include any condition encountered during construction and in the normal operation of a nuclear power plant, as well as the conditions resulting from a single failure of the reactor coolant system and other high-energy lines plus those extreme environmental conditions postulated during the life of the facility and certain combinations thereof. Structures are designed and analyzed to meet performance and strength requirements for the applicable load combinations given in Tables 3.8-9 and 3.8-10.

The load factor for live load (L) in Table 3.8-10 is equal to 1.7 as required. The load combinations in Table 3.8-10 in which R_o and T_o are present are more conservative than those given in ACI 349-1976 and SRP Section 3.8.3 (p. 3.8.3-14). When T_o and R_o are present, ACI 349-1976 indicates the following combinations:

(9) $0.75\ [1.4\ D + 1.7\ L + 1.4\ T_o + 1.7\ R_o]$

(10) $0.75\ [1.4\ D + 1.7\ L + 1.7\ E_o + 1.4\ T_o + 1.7\ R_o]$

The SRP indicates the following load combinations to be used:

(1b) $0.75 [1.4 D + 1.7 L + 1.7 T_o + 1.7 R_o]$

(2b) $0.75 [1.4 D + 1.7 L + 1.9 E + 1.7 T_o + 1.7 R_o]$

Table 3.8-10 provides the following combinations when considering R_o and T_o:

(3) $1.4 D + 1.7 L + 1.3 T_o + 1.3 R_o$

(6) $1.4 D + 1.7 L + 1.9 E + 1.3 T_o + 1.3 R_o$

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Reactor Support, Steam Generator Support, Reactor Coolant Pump Support, and Pressurizer Support

The design and analysis procedures of the reactor support, steam generator support, reactor coolant pump support, and pressurizer support are described in Subsection 3.9.3.4.

3.8.3.4.2 Other Internal Structures

Primary shield wall, secondary shield wall with refueling pool walls, interior base mat and polar crane girders were analyzed using computer programs which are available in the Sargent & Lundy program library. These programs have all been validated by comparing results for selected problems with their closed-form solutions or by comparing the solution of a given problem with the solution obtained from one or more previously validated programs. A more detailed description of the various programs named in these paragraphs can be found in Appendix D.

The intermediate floor framings are designed using conventional elastic design methods.

Containment concrete internal structures were designed for both transient and static load. Asymmetric LOCA loads resulting from the postulated breaks of the reactor coolant piping at various locations have been investigated. These loads have been applied as subcompartmental pressure between the secondary shield wall and the primary shield wall. The pressures were applied based on a time history approach on the 53 postulated subcompartments. These pressures are transient in nature and were considered as a static load factored with a dynamic load factor. The peak pressures were applied to the structure utilizing the appropriate UFSAR load combinations. Dynamic compartment pressurization loads due to primary coolant loop pipe breaks have been eliminated based on GDC 4 and the leak-before-break analyses performed by Westinghouse. Even though the primary coolant pipe break compartment dynamic pressurization loads are no longer the design basis, they are controlling with regard to other pressurization loads and therefore no changes have been made to eliminate these loads in the following UFSAR sections.

3.8.3.4.2.1 Primary Shield Wall

Primary shield wall was analyzed using the finite element program DYNAX as a thick shell of revolution, using quadrilateral finite elements.

The effect of large openings are considered by providing modified material properties for those elements at and between these openings.

Thermal analysis was performed in the same manner as the containment which is described in Subsection 3.8.1.4.2.

The design and analysis procedures for the primary shield wall are in full compliance with ACI-349.

3.8.3.4.2.2 Secondary Shield Wall, Reactor Refueling Pool Walls and Operating Floor

Secondary shield wall, reactor refueling pool walls and operating floor are all analyzed in one complete finite element model using SLSAP program.

Pressures due to a loss-of-coolant accident were directly applied on the model. Also, the hydrodynamic loads of the refueling water were applied. Thermal analysis was performed the same way as the containment which is described in Subsection 3.8.1.4.2. The design and analysis procedures for the secondary shield wall, reactor refueling pool walls and operating floor are in compliance with ACI-349.

3.8.3.4.2.3 Interior Base Mat

The interior base mat was analyzed using the SLSAP-I (nonlinear) program. Due to symmetry, one-quarter of the mat was modeled. Foundation springs are modeled as nonlinear elements to the extent that they have compressive stiffness only and have no effect in uplift regions of the base mat.

3.8.3.4.2.4 Polar Crane Supporting Systems

The polar crane girders were analyzed using the STRUDL-II program as a two-dimensional frame with supports at the crane brackets. The reactions from the crane girders were applied on the bracket and the brackets were designed for these reactions.

The design and analysis procedures for the polar crane girders and crane brackets are in full compliance with AISC specifications.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Reinforced Concrete

For the analysis of reinforced concrete structures, the strain in the reinforcing steel and concrete are assumed to be directly proportional to the distance from the neutral axis. The strain in the steel is limited to 90% of the yield strain and the stress is equal to the steel modulus, E_s, times the strain. The concrete compressive stress-strain relationship will be defined by a parabola between the origin and the point where the strain is 0.002 and the stress is $0.85f'_c$, followed by a descending curve to an ultimate strain of 0.003, where f'_c is the specified compressive strength of concrete.

The load combinations in Table 3.8-10 are used in conjunction with the yield limit criteria given in this paragraph. The yield limit strength of the structure is defined for this design as the upper limit of elastic behavior of the effective load carrying material. The allowable stresses for this limit are defined as follows:

a. Concrete Compression

1. membrane compression - 0.6 f'_c,

2. membrane plus flexural compression - 0.75 f'_c,and

3. local compression - 0.9 f'_c.

b. Concrete Radial Shear

The design for radial shear will be in accordance with ACI-318 for the determination of the permissible shear stress carried by the concrete.

c. Concrete Tangential Shear

Cracking of concrete due to tangential shear can be postulated along a plane. The reinforcement passing through this plane (normal to the crack) is subject to a tensile force due to the relative movement of two sides of the crack. The magnitude of this tension in the reinforcement can be computed on the basis of shear friction theory.

Utilizing the shear friction concept, the tangential shear force was converted into an equivalent tension using a coefficient of friction of 1.0. This tension was then directly combined with the membrane force. The concrete section was designed for the flexural forces and the combined membrane force.

d. Reinforcing Steel

1. tension - 0.9 f_y, and

2. compression (loading-carrying) - 0.9 f_y, where f_y is the specified yield strength of the reinforcing steel.

Allowable stresses and strains for reinforced concrete slabs on decking are based on ultimate strength provisions of ACI-318.

3.8.3.5.2 Structural Steel

The stresses and strains of structural steel are limited to those specified in the AISC specifications. The related margins of safety are as described in the commentary, Section 1.5 of the specifications. While designing for severe environmental loading combinations, no overstresses are allowed. However, for abnormal, extreme environmental, abnormal/severe environmental and abnormal/extreme environmental, the allowable loads are increased to 1.6 times the AISC allowables but not more than .95 times the steel yield strength which gives a factor of safety of 1.05 against yielding. The deformation of steel is limited since in both loading cases the stresses are held within elastic range.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

Materials and quality control requirements for containment internal structures are listed in Appendix B. The physical properties of these materials are listed in the appropriate section of Appendix B.

3.8.3.7 Testing and Inservice Surveillance Requirements

Testing and inservice surveillance requirements for the containment internal structures are outlined in Appendix B.

3.8.4 Other Seismic Category I Structures

3.8.4.1 Description of the Structures

The Seismic Category I structures, other than the containment and its internals, are as follows:

a. auxiliary building,

b. fuel handling building,

c. refueling water storage tank and tunnels,

d. main steam tunnel and auxiliary-feedwater tunnel,

e. electrical duct runs,

f. essential service cooling towers,

g. river screen house, and

h. deep well enclosures.

3.8.4 Other Seismic Category I Structures

3.8.4.1 Description of the Structures

The Seismic Category I structures, other than the containment and its internals, are as follows:

a. auxiliary building,

b. fuel handling building,

c. refueling water storage tank and tunnels,

d. main steam tunnel and auxiliary-feedwater tunnel,

e. electrical duct runs, (contain no Class 1E cables; not currently maintained as Category 1)

f. lake screen house substructure, and

g. essential service water discharge structure.

3.8.4.1.1 Auxiliary Building

The auxiliary building is located between the containment structure and the turbine building (Drawings M-7 through M-12).

The auxiliary building is a reinforced concrete shear wall structure supported on mat foundation. The lower levels of the auxiliary building are continuous two-way slab and beam construction. The levels above the grade consist of steel framing with concrete slab on metal deck. The exterior walls are concrete for radiation shielding and missile protection. The interior walls are either concrete or concrete block. (See Figures 3.8-52 through 3.8-57.)

The auxiliary building contains the control room, electrical equipment room, switchgear room, battery and computer rooms. It also houses the diesel generators, radwaste processing facilities, laboratories, HVAC and filter rooms.

3.8.4.1.2 Fuel Handling Building

The fuel handling building is located adjacent to the auxiliary building between the containment structures (Drawing M-13).

The fuel handling building is a reinforced concrete structure up to grade, except in the fuel pit area where reinforced concrete is continued up to the mezzanine level. The walls below grade bear on a reinforced concrete mat foundation. The portion of building above grade has a structural steel frame with concrete slab on metal deck. The exterior walls are concrete for radiation shielding and missile protection. The interior walls are of either concrete or concrete block construction. Fuel access is at grade level where a railroad track is provided to permit use of an overhead crane for handling the fuel. (See Figures 3.8-52, 3.8-53, and 3.8-58.)

3.8.4.1.3 Refueling Water Storage Tank and Tunnel

The refueling water storage tank is a reinforced concrete cylindrical structure supported on a mat foundation. The inside wall of the tank is lined with stainless steel liner. The tunnel which connects the refueling water storage tank with the auxiliary building is a reinforced concrete box section. (See Figures 3.8-83 and 3.8-84.)

The tank and the tunnel are shown in Drawing M-15.

3.8.4.1.4 Main Steam and Auxiliary-Feedwater Tunnel

The main steam and auxiliary-feedwater tunnel is a bilevel reinforced concrete box section. It connects the containment with the turbine building through the auxiliary building. The top of the tunnel is 1 foot 0 inch below the grade level and it is shown in Drawings M-10 and M-11.

The isolation valve room, known as the safety valve room, is a reinforced concrete structure which is an integral part of the main steam and auxiliary-feedwater tunnel at the containment building. It is designed using a two-way slab theory for all walls and slabs. (See Figures 3.8-80 through 3.8-82.)

3.8.4.1.5 Electrical Duct Runs

Electrical duct runs are buried reinforced concrete conduits which carry Class lE cables for safety-related equipment. (At Braidwood, Class 1E cables are not buried in concrete electrical duct runs.

3.8.4.1.6 Essential Service Water Cooling Tower

The essential service cooling tower consists of two four-cell concrete structures erected over one common reinforced concrete cold water basin. The mat foundation supporting structure rests on a grouted rock strata 9 feet 0 inch below grade level. The internal water distribution system and the fill are supported on concrete beam and column system with bracings to resist lateral loads. (See Drawings S-239, S-241, S-243, S-245, S-247, S-249, and S-250)

The fan equipment including gear box are surrounded by 14 feet 0 inch high concrete recovery stack.

3.8.4.1.7 River Screen House

The river screen house consists of reinforced concrete structure with main floor 3 feet 6 inches above grade, internal and external concrete walls, and concrete mat foundation. The roof and intermediate slab consists of steel framing with slab on metal deck. The superstructure consists of structural steel braced framework covered by insulated siding. Drawing M-20 shows the structural arrangement of the river screen house. (See Figures 3.8-59 and 3.8-64.)

3.8.4.1.8 Deep Well Enclosures

The deep well enclosures consist of reinforced concrete walls on spread footing with a removable slab at the top. These enclosures are required to protect the components of the well systems located above grade during tornado conditions. (See Figure 3.8-79.)

3.8.4.1.9 Lake Screen House Substructure

The lake screen house consists of reinforced concrete walls and mat foundation. The essential service water pipes are embedded in mat foundation. Drawing M-19 shows the structural arrangement of the lake screen house. (See Figure 3.8-74 through 3.8-78.)

3.8.4.1.10 Essential Service Water Discharge Structure

The essential service water discharge structure is a reinforced concrete structure. It provides anchorage for the discharge end of the essential service water pipes in the essential service cooling pond (see Figure 3.8-95).

3.8.4.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications applicable to the design, fabrication, construction, testing, and inservice inspection of safety-related structures outside the containment are referenced in Table 3.8-2, for Regulatory Guides see Appendix A. All of the items listed in Table 3.8-2 are applicable, with the exception of Items 17 and 18.

3.8.4.3 Loads and Loading Combinations

The loading definitions and loading combinations applicable to the design of Seismic Category I structures outside the containment are listed in Tables 3.8-9, 3.8-10, and 3.8-4, respectively.

In addition to their own dead loads (equipment, piping and cable pan loads, etc.), floors are designed for live loads to withstand removal of equipment. The roofs are designed for snow, negative pressure due to tornado suction and checked for effects of probable maximum precipitation. The ability of floors and roofs to transmit shear loads through diaphragm action is also checked.

For loading combination applicable to river screen house refer to Table 3.8-11. It may be seen from this table that the river screen house is designed for the following extreme loading conditions:

a. SSE + Maximum Flood of Record.

b. OBE + Combined Event Flood.

The combined event flood is selected on the basis of a very low probability of exceedance of 10^{-6} per year, as described in Subsection 2.4.3.7. The river screen house is not designed against the probable maximum flood and the design-basis tornado. The makeup water system for the ultimate heat sink for the Byron Station consists of a combination of the river screen house and deep wells. The deep wells are designed for probable maximum flood and design-basis tornado.

It may be seen from Table 3.8-11 that load combinations due to high energy pipe break accidents are not included. This is because there are no high energy pipe lines within the Byron river screen house.

Loading combination number 11 of Table 3.8-10 is applicable to the design of the Byron deep well enclosures.

3.8.4.3 Loads and Loading Combinations

The loading definitions and loading combinations applicable to the design of Seismic Category I structures outside the containment are listed in Tables 3.8-9, 3.8-10, and 3.8-4, respectively.

In addition to their own dead loads (equipment, piping and cable pan loads, etc.), floors are designed for live loads to withstand removal of equipment. The roofs are designed for snow, negative pressure due to tornado suction and checked for effects of probable maximum precipitation. The ability of floors and roofs to transmit shear loads through diaphragm action is also checked.

3.8.4.4 Design and Analysis Procedure

The design and analysis of all structural components are based upon conventional elastic methods. The buildings are analyzed as shear wall-diaphragm structure with the exception of the river screen house which utilizes a vertical bracing system to resist lateral loads above grade. Exterior walls are designed to resist a combination of vertical loads, bending moments, lateral shear and overturning moments caused by seismic forces and tornado loads. Longitudinal and lateral shears are transferred to the mat through shear friction principles.

A modified frame model is used to analyze beams and columns in the auxiliary building. In addition, boundary conditions are determined, where critical, by stiffness evaluation of the actual intersecting structural member. The computer program STRUDL-II (Appendix D) is used to analyze these frames and computer programs CBEAM and PCAUC (Appendix D) are used for the design of beams and columns respectively. The beams and columns in the superstructure of buildings are analyzed and designed by the STAND Program (Appendix D). The Byron/Braidwood Stations comply with portions of ACI-349 which are based on ACI-318.

The spent fuel pool within the fuel handling building, is analyzed separately from the base mat. The analysis takes into account the dynamic effect of water (see Reference 2). The SLSAP-IV computer program (Appendix D) is used to analyze the finite element model. The stress in the reinforced concrete section is checked using the TEMCO computer program (Appendix D).

The refueling water storage tanks are analyzed using an axisymmetric finite element model. (DYNAX computer program-Appendix D). Reference 2 was again used to determine the hydrodynamic effects. The TEMCO computer program was used to check the stress in reinforced concrete section.

Yield line theory was used to analyze the effects of a high energy pipe line break outside the containment.

The electrical duct runs were analyzed using the beam on elastic foundation analogy. Seismic analysis for buried tunnels and duct runs follow the procedure described for buried Seismic Category I piping systems and tunnels given in Subsection 3.7.3.12.

3.8.4.5 Structural Acceptance Criteria

3.8.4.5.1 Reinforced Concrete

The stresses and strains of various structural components are based on the ultimate strength design provisions in ACI-318. The margin of safety is contained in the capacity reduction factor (ϕ) specified in the code. The deflection and serviceability of various structural components are provided as required by the code.

Yield line theory methods are used for those elements subject to impactive and impulsive loads on beams, walls, and slabs. Ductility ratios less than 10 have been maintained for loads due to tornado missiles. A hinge rotation at yield hinges has been limited to 0.07 radians for loads due to high energy pipe whip. Refer to ACI-349 for additional criteria.

The Byron/Braidwood Stations comply with portions of ACI-349 which are based on ACI 318.

3.8.4.5.2 Structural Steel

The stresses and strains of structural steel are limited to those specified in the AISC Specification and its subsequent revisions. The related margins of safety are as described in the commentary, Section 1.5 of the Specifications. While designing for severe environmental loading combinations, no overstress factors are allowed. However, for abnormal, extreme environmental, abnormal/severe environmental load and abnormal/extreme environmental combinations, the allowable loads are increased to 1.6 times the AISC allowable but not more than .95 times the steel yield strength which gives a factor of safety of 1.05 against steel yielding. The deformation of steel is limited since in both loading cases stresses are held within elastic range. This provides an additional margin of safety against failure since no plastic deformations are allowed.

In addition, deflections are checked and kept within the limits prescribed in the AISC Specification.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

Construction materials conform to the standards set forth in Appendix B. The procedures for sampling and testing of materials for quality assurance are also described in Appendix B. The quality control program for the design and construction of the Seismic Category I structure outside the containment is described in detail in Chapter 17.0.

3.8.4.7 Testing and Inservice Surveillance Requirements

No preliminary structural integrity or performance tests were conducted. However, rigorous inspection techniques and the quality control procedures described in Appendix B were adopted throughout construction.

3.8.4.8 Special Topics

3.8.4.8.1 Masonry Walls

3.8.4.8.1.1 Comparison to SEB Interim Criteria

The following is an assessment of the differences between the SEB Interim Criteria, Revision 1, and criteria used for the design of masonry walls at Byron/Braidwood Stations.

a. General Requirements

The materials, testing, analysis, design, construction and inspection of safety-related concrete masonry walls for Byron/Braidwood Stations conform to NCMA-1974, which is generally in agreement with Uniform Building Code - 1979, with the exception of the allowable stresses for the unreinforced masonry. There are no significant deviations between NCMA-1974 and ACI 531-79.

b. Loads and Load Combinations

The loads and load combinations used for the safety-related concrete masonry walls at Byron/Braidwood Stations are in agreement with the loads and load combinations of SEB Interim Criteria Rev. 1.

c. Allowable Stresses

The allowable stresses for unreinforced solid or hollow concrete masonry walls for Byron/Braidwood Stations are in conformance with NCMA-1974. Table 3.8-15 is a comparison of the allowable stresses used in the design of Byron/Braidwood masonry walls with the SEB interim criteria allowable stresses.

1. No overstress factor has been used in the design of safety-related concrete masonry walls for Byron/Braidwood Station for loading combinations containing OBE seismic loads, which is in compliance with the SEB criteria.

2. The safety-related concrete masonry walls for Byron/Braidwood Stations have been designed using NCMA allowable stresses corresponding to the special inspection category. Quality assurance/quality control procedures applied for the construction of safety-related concrete masonry walls substantiate compliance with the inspection requirements of the SEB criteria.

3. All safety-related concrete masonry walls of the Byron/Braidwood Stations have been designed spanning horizontally, thus precluding the use of tension perpendicular to the bed joint. Tension perpendicular to the bed joint occurs only in very localized sections adjacent to openings or discontinuities in horizontally spanning walls. This local tension stress has been limited to 39 psi for the normal and OBE load combinations and 65 psi for the SSE load combinations for solid concrete masonry units.

4. A load factor of 1.67 has been used for load conditions which represent extreme environmental, abnormal, abnormal/ severe environmental, and abnormal/extreme environmental conditions. For comparison of the allowable stresses with the load factors under SEB Interim Criteria, Revision 1, and the Criteria used for Byron/Braidwood Stations, see Table 3.8-15.

d. Design and Analysis Considerations

1. The analysis of the safety-related concrete masonry walls for Byron/Braidwood Stations has followed established principles of engineering mechanics, and has taken into account sound engineering practices.

2. The assumptions and modeling techniques used in the assessment of the safety-related concrete masonry walls have considered proper boundary conditions, cracking of sections, if any, and the dynamic behavior of the masonry walls.

3. The damping values for the safety-related concrete masonry walls for Byron/Braidwood Stations are in conformance with Regulatory Guide 1.61.

4. The seismic analysis for the safety-related concrete masonry walls is in accordance with the requirements of the Byron/Braidwood Stations, UFSAR Section 3.7.

5. The analysis of the safety-related masonry walls has considered both in-plane and out-of-plane loads.

6. Interstory drift effects have been evaluated at each floor elevation. The maximum shear strain due to interstory drift for Byron/ Braidwood Stations is approximately 0.0004. Shear

deformation of this magnitude will not impair the structural integrity of the safety-related concrete masonry walls.

7. There are no concrete masonry shear walls at Byron/Braidwood Stations.

8. All multiwythe safety-related concrete masonry walls meet the requirements of NCMA, ACI 531 as well as UBC-1979.

9. Where applicable, safety-related concrete masonry walls have been evaluated for the effects of accident pipe reaction (Y_r), jet impingement (Y_j), and missile impact (Y_m).

3.8.4.8.1.2 Design Criteria

a. All concrete masonry walls have been designed for out-of-plane seismic loadings. Vertical seismic acceleration is less than 1.0 g for all of these walls, thus causing no net tension on the wall.

   For in-plane inertial loads, calculations enveloping the ratios of height to length of wall and considering the maximum and minimum acceleration values have been performed. It has been observed that in-plane shear stresses under SSE load combinations are low (less than 18 psi) compared to allowable value of 52 psi as allowed under SEB Interim Criteria and actual tensile stresses normal to be bed joints are within the allowable value of 32 psi as per SEB Interim Criteria.

b. Multiwythe concrete masonry walls have been bonded together with continuous solid or grouted masonry header courses. This mechanism is the most positive means to assure composite action. This method is recommended in ACI 531-79, NCMA-1974 and recognized by other concrete masonry building codes. Moreover, the walls have been constructed with 3/16 inch diameter truss-type joint reinforcement every second course. These two mechanisms are sufficient to assure composite action of the multiwythe walls.

c. Structural steel columns have been used to provide lateral support for the masonry walls for out-of-plane loads, thereby creating simply supported wall panels with average aspect ratio of 2 vertical to 1 horizontal. For these aspect ratios, the moments in the vertical direction occur solely due to the Poisson ratio effect. These moments do not govern the design and are significantly less than one-half of the horizontal moment.

Furthermore, the assumption of one way horizontal action results in the greatest number of support columns since credit is not taken for the low moment in the vertical direction in reducing the horizontal moment.

d. Masonry wall support columns are neither fully embedded with Type M mortar in the masonry wall or located outside of the wall. For a fully embedded column, the wall inertial load is transferred to the column in bearing against the column flange.

   For columns located outside of the wall, the wall has been anchored to the column flange either by means of through bolts, expansion anchors, or grouted anchors. The connection between the column and wall has been designed to transfer the applied forces.

   Masonry wall support columns have been attached to the floor above and below by means of welding, bolting or expansion anchors. The top connections have been provided with vertical slotted holes.

   The first course of blocks at the bottom of the wall is laid on mortar bedding. Shear transfer is accomplished through this joint by utilizing friction concept. It should be noted, however, that all walls span horizontally and that the only shear which must be transferred across this joint is that which occurs locally adjacent to an opening. Coefficient of friction is assumed equal to 0.8.

3.8.4.8.1.3 Allowable Tensile Stresses Parallel to Bed Joints

The project allowable stress parallel to the bed joints is in accordance with National Concrete Masonry Association's (NCMA), "Specification for the Design and Construction of Load Bearing Concrete Masonry."

Review of the background information regarding allowable tensile stresses parallel to the bed joints indicates that the allowable values have been established by doubling the allowable stresses perpendicular to the bed joints which are based on mortar tensile bond strength. The stresses parallel to the bed joints are more a function of masonry unit strength than mortar. As such, it is conservative to use double the vertical span values.

Table 3.8-16 gives the summary of test results used to arrive at the value of modulus of rupture for horizontally spanning walls and, hence, the tensile stress parallel to the bed joint. The safety factors in the table have been calculated by dividing the actual modulus of rupture values by allowable stress values, 32 psi for Type N and O mortar, and 46 psi for Type M mortar.

For 15 concrete masonry walls with joint reinforcement, as is the case for masonry walls at Byron/Braidwood Stations, the safety factors averaged 5.6 for normal and OBE load combinations, and 5.6/1.67 = 3.35 for SSE load combinations. In addition, the test results for 43 walls containing no joint reinforcement indicate an average factor of safety of 5.3 for normal and OBE load combinations, and 5.3/1.67 = 3.17 for SSE load combinations. These values are comparable but slightly less than those for walls with joint reinforcement.

3.8.4.8.1.4 Steel Flexibility

The masonry walls at Byron/Braidwood stations are not supported at the top and are provided with a 1-inch gap at the top. Steel columns have been used to provide lateral support for out-of- plane loads. As such, the walls have been designed for horizontally spanning beam strip moments. A parametric study was performed to estimate the magnitude of vertical moments resulting from the flexibility of the steel columns. The possible variations in design parameters, such as column size, number of steel columns used in a wall, wall thickness, and length of wall, were considered by using more generalized design items A* and I* which are described below.

A* = Ratio of total wall mass to the total mass of steel columns for a given wall

= $P_m wt/P_s A_s$

I* = Ratio of total wall rigidity to the rigidity provided by steel columns

= $E_m wt^3/E_s I_s$

h/s = Ratio of block wall column height to block wall column spacing

where:

h = height of wall or block wall column

w = width of wall

t = thickness of wall

s = spacing of block wall columns

E_m = masonry modulus of elasticity

E_s = steel modulus of elasticity

P_m = masonry mass density

P_s = steel mass density

A_s = total area of steel columns

I_s = total moment of inertia of steel columns.

For a given ratio of h/w and h/s, plate vertical and horizontal moment coefficients can be generated for various values of A* and I*.

One hundred fourteen of 236 unreinforced masonry walls at Byron Station Unit 1 have been reviewed for the variations in A*, I*, and column aspect ratio h/s. Due to the geometry and construction of three walls, use of A* and I* approach is not appropriate. These walls have been analyzed using the finite element method which considers the effect of column flexibility. All the remaining walls have not been provided with block wall columns, and thus are not subject to the effects of column flexibility.

The masonry walls at Byron Unit 2 and Braidwood Units 1 and 2 have similar geometry, have been designed using the same criteria, and have been constructed in accordance with the same specification. Therefore, this review is valid for unreinforced masonry walls at both Byron and Braidwood Stations.

Table 3.8-17 gives the approximate breakdown of 114 unreinforced masonry walls with block wall columns in different groups of A* and I* for various types of column aspect ratios (h/s).

All of the 114 masonry walls at Byron Station Unit 1 were studied to determine the effect of column flexibility on the design of masonry walls. For a given wall, plate vertical moments are calculated using finite element analysis, and the effect of column flexibility is evaluated by determining if the moments in the vertical direction are less than the cracking moments. If so, the masonry wall design, based on calculation of design moments using a horizontally spanning beam strip between steel columns, is valid.

Since the allowable stresses perpendicular to the bed joints are approximately one-half of the allowable stresses parallel to the bed joints, the permissible values of plate vertical moment are determined by considering plate vertical cracking moments equal to one-half of the plate horizontal cracking moments. The horizontal cracking moments for hollow walls are based on the modulus of rupture determined by testing for the Clinton Power Station. The modulus of rupture for solid masonry wall is obtained by adjusting the modulus of rupture of hollow block wall with a factor equal to the ratio of the SEB allowable stress for solid masonry to the SEB allowable stress for hollow masonry.

Review of the actual plate vertical moments for the masonry walls at Byron Station Unit 1 indicates that all the masonry walls have actual plate vertical moments less than the vertical cracking moments based on the elastic analysis. The vertical moments which are developed due to flexibility of the block wall columns do not affect the structural integrity of the walls. Hence, the design of unreinforced masonry walls at Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2, based on horizontally spanning beam strip moments, is acceptable.

3.8.4.8.1.5 Evaluation of Structural Cracks (Braidwood)

The survey and evaluation of structural cracks in the Braidwood Station masonry walls was completed and an evaluation report was transmitted to the NRC (Reference 4). The evaluation was accepted in NUREG-1002, Safety Evaluation Report, Supplement No. 2, dated October 1986.

3.8.4.8.2 Concrete Expansion Anchors

The effects of the base plate flexibility on forces in the expansion anchors have been accounted for in extensive finite element studies. The flexible plate was modeled by plate elements and the anchors were modeled by truss elements. A bilinear load displacement curve for the anchors idealizing the load displacement behavior observed in tests was used in the analysis. The supporting concrete was modeled by one way compression springs. Nonlinearity was introduced in the analysis by the nonlinear behavior of the concrete springs and the bilinear load displacement behavior of the expansion anchors. A constant stiffness method was used to solve this nonlinear problem. The details of this analysis procedure are discussed in Reference 3. The following discussion briefly summarizes how the effect of plate flexibility was considered in the design.

There are two possible effects of the base plate flexibility on the forces in the expansion anchors as follows:

a. prying action; and

b. unequal distribution of forces among anchors based on the geometric configuration of anchors with respect to the applied loads.

The analysis procedure includes the consideration of the effect of both these factors on the forces in expansion anchors.

As a result of this analysis, amplification factors for use in the design of expansion anchor plates were developed. These amplification factors correlate the anchor forces determined by the nonlinear flexible plate analysis to the forces determined by a conventional rigid plate analysis.

The amplification factors were computed as follows:

a. Pure tension and pure moment were applied on the anchor plate assembly so that at least one anchor was stressed to its ultimate load capacity, Pu. A nonlinear approach described above accounting for plate flexibility was used for the analysis.

b. For the same plate assembly and same load the anchor force was calculated using a rigid plate analysis.

c. The ratio of the anchor force obtained by the nonlinear finite element approach to that obtained by the rigid plate analysis approach is defined as the amplification factor.

The amplification factors were determined for the Byron/ Braidwood specific plates varying the expansion anchor configuration and the plate size. The loading conditions considered were the direct tension and pure moment in the critical direction. An enveloping value of the amplification factors for each plate size and anchor configuration thus obtained was used in the design. A separate study had confirmed that the amplification factors for a combination of direct tension and moment will fall within the enveloping value of amplification factors.

The flexible plate test program which was conducted by Wiss, Janney, Elstner & Associates (WJE) indicated an amplification factor of 1.15 to 1.20. These amplification factors were reported at ultimate load. The analytical assessment for the same assemblies predicted the amplification factor to be 1.0. The residual load of 15% to 20% which was observed during the WJE test is not attributed to base plate flexibility or prying action effects because the plate corner displacements which were monitored during the tests showed that the corners had lifted and were not in contact with concrete at ultimate load. Thus amplification of anchor force due to plate flexibility or prying action is not possible. This residual in anchor load measurement is attributed to the behavior of the testing equipment at the ultimate load levels. The tests did establish that prying action in base plate assemblies with expansion anchors is insignificant as compared to plate assemblies with rigid bolts. According to analytical studies reported in Reference 3, the amplification factor was expected to be 2.1 with rigid bolts. The test results have substantiated that the rigid bolt behavior is not the true behavior and that the prying effect is relieved due to the flexible load-displacement characteristics of expansion anchors.

Examples of three expansion anchor plate assemblies are provided shown in Figure 3.8-94. The square plates are with four and eight anchors and the rectangular plate is with six anchors. The maximum applied load on the assemblies and the corresponding anchor forces obtained by the rigid plate analysis and the flexible plate analysis are given in Table 3.8-18. The amplification factors as defined above are also tabulated. The larger of the computed amplification factors for each plate assembly is used as the design amplification factor.

3.8.5 Foundations

3.8.5.1 Description of Foundations

3.8.5.1.1 Containment Building Foundation

The description of the containment building foundation is given in Subsection 3.8.1.

3.8.5.1.2 Main Building Complex

The auxiliary and fuel handling buildings are all supported on a continuous multilevel reinforced concrete mat foundation. See Figures 3.8-44 and 3.8-45 for the plan views of mats for Byron and Braidwood Stations respectively, and Drawings M-15 through M-18 for the sections through the mats. The thickness of the mat varies generally from 3 feet 0 inch to 6 feet 0 inch.

The main building complex mat is also continuous with the mat foundation of the turbine and heater bay buildings which are non-Seismic Category I. Because of the continuity and interconnection of the mat foundations of various buildings, the entire main building complex and turbine building are modeled as a unit for the seismic analysis (Subsection 3.7.2). However, in view of a very large overall mat area and limitations of the computer capacities, the mat foundation was divided in the following areas for detailed structural analysis and design:

a. auxiliary building: between Column Lines 10 and 26;

b. auxiliary building: between Column lines 6 and 10 for Unit 1 and between Column lines 26 and 31 for Unit 2;

c. spent fuel pool: between Column lines W and Y;

d. fuel handling building and refueling water storage tanks; and

e. turbine and heater bay buildings.

The moments from the shear wall are transferred to the mat as vertical loads along the wall, assuming the mat is completely rigid. The shear resulting from shear wall action and lateral shears is transmitted to the mat through shear friction reinforcement. Keys are also provided between the walls and the mat as an added resistance to the lateral shears.

3.8.5.1.3 Essential Service Water Cooling Tower (Byron)

The two units of the essential service water cooling towers are separately supported on 3 feet 0 inch thick reinforced concrete mat foundations (45 feet 0 inch by 174 feet 0 inch) resting on grouted foundation bedrock at elevation 865 feet 0 inch.

A 2-foot 0 inch thick base slab for adjoining electrical rooms rests on controlled compacted granular fill.

3.8.5.1.4 River Screen House

The river screen house is supported on a 3 foot 0 inch thick mat foundation resting on the natural soil at elevation 666 feet 0 inch and 660 feet 6 inches, approximately 22 feet below the grade level (Drawing M-20).

3.8.5.2 Applicable Codes, Standards, and Specifications

The codes, specifications, standards of practice, general design criteria, and other accepted industry guidelines which are adopted to the extent applicable in the design, fabrication, testing, inservice inspection, and construction of the foundations for Seismic Category I structures are found in Table 3.8-2, for Regulatory Guide see Appendix A. Those listed, which are applicable to this section, are 1 through 14 and 19 through 23.

3.8.5.3 Loads and Loading Combinations

The loads and loading combinations listed and discussed in Subsection 3.8.4.3 are Applicable to the design of the foundation. Refer to Tables 3.8-9 through 3.8-11 for a list of the load conditions that are considered in the design. The definitions of these load conditions and loading categories are presented in Table 3.8-4.

The following load combinations for overturning, sliding, and flotation have been considered:

a. D+H+E;

b. D+H+W;

c. D+H+E';

d. D+H+W_t; and,

e. D+F'.

In the above combinations, D is the dead load, H is the lateral earth pressure, E is an operating basis earthquake, E' is a safe shutdown earthquake, W_t is a design basis tornado, and F' is a design basis flood.

A 2-foot 0 inch thick base slab for adjoining electrical rooms rests on controlled compacted granular fill.

3.8.5.1.4 Lake Screen House

The lake screen house is supported on a 196 foot by 117 foot 6 inch mat foundation (varying from 4 to 5 feet thick) resting on the natural ground strata of glacial till at an approximate elevation of 565 feet, approximately 35 feet below grade level (Figures 3.8-77 and 3.8-78).

3.8.5.2 Applicable Codes, Standards, and Specifications

The codes, specifications, standards of practice, general design criteria, and other accepted industry guidelines which are adopted to the extent applicable in the design, fabrication, testing, inservice inspection, and construction of the foundations for Seismic Category I structures are found in Table 3.8-2, for Regulatory Guide see Appendix A. Those listed, which are applicable to this section are 1 through 14 and 19 through 23.

3.8.5.3 Loads and Loading Combinations

The loads and loading combinations listed and discussed in Subsection 3.8.4.3 are applicable to the design of the foundation. Refer to Tables 3.8-9 and 3.8-10 for a list of the load conditions that are considered in the design. The definitions of these load conditions and loading categories are presented in Table 3.8-4.

The following load combinations for overturning, sliding, and flotation have been considered:

a. D+H+E;

b. D+H+W;

c. D+H+E';

d. D+H+W_t; and,

e. D+F'.

In the above combinations, D is the dead load, H is the lateral earth pressure, E is an operating basis earthquake, E' is a safe shutdown earthquake, W_t is a design basis tornado, and F' is a design basis flood.

3.8.5.4 Design and Analysis Procedures

3.8.5.4.1 General

The analysis of Seismic Category I foundations is done using finite element technique. The finite element models comprise plate elements for the base mat and beam elements for the walls. The models are analyzed using SLSAP-IV computer program (Appendix D) for all internal and external loads transferred to the mat through shear walls and columns and due to elastic deformation of the slab. Appropriate boundary conditions, load distribution and soil spring values are used for individual mats to ensure proper deflection compatibility along match lines of the adjacent mat areas. The design and analysis of the foundations comply with the requirements of ACI 318.

3.8.5.4.2 Main Building Complex

The foundation mat is divided into five distinct areas as discussed in Subsection 3.8.5.1.2. All these mats are multilevel with stiff walls between different levels. The mat slab is considered as a plate at one level, stiffened by the shear walls and the walls between two levels. Boundary elements, in the form of vertical soil springs, are incorporated at all nodes. In-plane rotation of the mat is ignored in the finite element model.

The effect of settlement on the forces and moments in the mat are taken into account by soil springs as discussed above. (See Subsection 2.5.4.10 for further discussion on settlements.)

Lateral loads are transmitted to the soil through friction between the concrete and soil. Overturning forces due to the moment on shear walls are resisted through the resulting nonuniform reactive pressures offered by the soil (Subsection 2.5.4.10).

3.8.5.4.3 Essential Service Water Cooling Tower (Byron)

Each cooling tower consists of four identical cells. Therefore, only 2 cell-area mat size is modeled because of symmetry. SLSAP IV finite element program is used for this design. Effect of soil and water pressure on the walls is separately added to the mat.

As in Subsection 3.8.5.4.2 the lateral loads are resisted by the friction between soil and concrete.

3.8.5.4.4 River Screen House (Byron)

Finite element model technique as discussed in Subsection 3.8.5.4.2 is used to analyze and design of the mat.

Lateral active (static and seismic) pressure loads from soil and water are resisted by the combined effect of the passive soil pressure and friction between base concrete and soil.

3.8.5.4.5 Lake Screen House (Braidwood)

Due to symmetry about the N-S centerline only one-half of the base mat is modelled for finite element analysis by SLSAP IV. Lateral soil pressures and hydraulic pressure, both static and seismic, are considered as discussed in Subsection 3.8.5.4.4.

Possible nonsymmetric loads due to certain empty compartments (due to maintenance, etc.) have also been considered.

3.8.5.5 Structural Acceptance Criteria

3.8.5.5.1 Structural Member Design

The acceptance criteria for the reactor containment base slab are as specified in Subsection 3.8.1.5.

The foundations for main building complex and other Category I structures are proportioned according to the criteria set forth in Subsection 3.8.4.5.

3.8.5.5.2 Stability

3.8.5.5.2.1 Main Building Complex

Seismic loads resulting from a SSE event govern the loading combinations described in Subsection 3.8.5.3 for overturning and sliding. The factor of safety for overturning is 1.84 and for sliding is 1.32. The factor of safety for flotation using the combination described in Subsection 3.8.5.3 is 1.33.

3.8.5.5.2.2 Essential Service Water Cooling Tower (Byron)

Using the load combinations for overturning, sliding, and flotation of Subsection 3.8.5.3, the factors of safety for the essential service water cooling tower are as follows:

a. 1.4 against sliding for a tornado;

b. 2.9 against overturning for a SSE event; and,

c. 4.2 against flotation.

3.8.5.5.2.3 River Screen House (Byron)

The stability of the river screen house is checked under SSE conditions and under the combination of OBE and combined event flood as per load combination (Subsection 3.8.4.3). The calculated factors of safety against flotation are indicated below:

Construction Condition:	2.10	
Normal Operating Condition	3.50	with normal water level
	1.70	with 25 years water level
	1.20	with OBE and combined event flood

The factors of safety against sliding and overturning are given below:

	Overturning	Sliding
OBE + 25 year flood	11.5	4.0
OBE + Combined event flood	7.0	2.5
SSE + 25 year flood	6.0	1.75

3.8.5.5.2.4 Lake Screen House (Braidwood)

The stability of the lake screen house is investigated under seismic conditions with highest water level in the lake. The factors of safety are given below.

	Overturning	Sliding	Flotation
Highest Water Level	--	--	1.5
Highest Water Level and SSE	3.0	1.1	--
Highest Water Level and OBE	3.0	1.1	--

3.8.5.5.2.5 Containment Building

The factors of safety against overturning, sliding and buoyancy for the containment structure are greater than the required values contained in SRP Section 3.8.5. Factors of safety for the containment against overturning and sliding are given below:

	Overturning	Sliding
OBE + Dead Weight + Lateral Earth Pressure	3.28	9.30
SSE + Dead Weight + Lateral Earth Pressure	1.81	3.57

Overturning resistance is provided by the dead weight of the containment structures and components.

Sliding resistance is provided both by friction between the basemat poured against rock (coefficient of friction = 1.0) and in direct bearing against rock of the reactor cavity wall and the tendon tunnel walls.

Resistance to uplift forces due to buoyancy from design basis flooding is considered to be provided by the dead weight of the containment structure. The factor of safety is 3.40.

3.8.5.5.2.6 Essential Service Water Discharge Structure (Braidwood)

The factors considered in the static stability check of the essential service water discharge structure included the pipe discharge force, wave forces, weight of the structure, water pressure, buoyant forces, and seismic forces. The loading combinations incorporated SSE, OBE, and static loads and used two lake level elevations (598 feet 2 inches and 587 feet 0 inch). Elevation 598 feet 2 inches is the flood condition, and elevation 587 feet 0 inch is the low water condition that will occur if the lake dikes are damaged. Refer to Figure 3.8-95 for structural details.

The discharge structure has been checked for sliding, overturning, and bearing on the soil. Sliding is counteracted by the passive soil pressure developed along the sides of the structure and friction along the bottom of the structure. The overturning moments from seismic, wave, and discharge forces are offset by a resisting moment due to the deadweight of the structure. The bearing forces on the soil have been compared to the bearing capacity of the glacial till beneath the structure. The factors of safety are in accordance with those required by SRP Section 3.8.5.

3.8.5.6 Material, Quality Control, and Special Construction Techniques

The materials, quality control, and special construction techniques for foundations conform to those set forth in Seismic Category I structures and are discussed in Subsection 3.8.4.6.

3.8.5.7 Testing and Inservice Inspection Requirements

Regular and rigorous inspection during construction in conjunction with testing of the structural materials was carried out as outlined in Appendix B. Structural integrity and/or performance tests are as specified in Subsection 3.8.1.7 for the containment base slab.

3.8.6 References

1. Sampling Procedures and Tables for Inspection by Variable for Percent Defective, MIL-STD-414, Superintendent of Documents, Government Printing Office, Washington D.C., June 11, 1957.

2. Dynamic Pressures on Fluid Containers, Nuclear Reactor and Earthquakes TID-7024, USAEC (August 1963) (Chapter 6).

3. "Evaluation of Analysis Procedures for the Design of Expansion Anchored Plates in Concrete," May 31, 1979. Part of Commonwealth Edison Company's response to IE Bulletin 79-02 transmitted by C. Reed to J. G. Keppler dated July 5, 1979.

4. Letter from A. D. Miosi (Commonwealth Edison Company) to H. R. Denton, (NRC) dated August 22, 1986.

TABLE 3.8-1

CONTAINMENT PENETRATIONS

PENETRATION NUMBER	SIZE	WALL THICKNESS	ELEVATION (ft-in.)	AZIMUTH (degree-min)	VERTICAL SKEW ANGLE	HORIZONTAL SKEW ANGLE	DESCRIPITION

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.8-1 (Cont'd)

PENETRATION NUMBER	SIZE	WALL THICKNESS	ELEVATION (ft-in.)		AZIMUTH (degree-min)		VERTICAL SKEW ANGLE	HORIZONTAL SKEW ANGLE	DESCRIPTION
P-30	10.0	0.365	395	0	62	30	0°	0°	Makeup demineralizer
P-31	16.0	0.844	395	0	57	30	0°	0°	H_2 monitoring system (BY), Spare (BW)
P-32	10.0	0.500	391	0	125	00	0°	0°	Fuel pool cooling and cleanup
P-33	14.0	0.375	391	0	120	00	0°	0°	Chemical and volume control
P-34	12.0	0.406	391	0	115	00	0°	0°	Fire protection
P-36	22.0	0.875	391	0	105	00	0°	0°	H_2 monitoring system
P-37	14.0	0.375	391	0	100	00	0°	0°	Chemical and volume control
P-39	8.0	0.322	391	0	75	00	0°	0°	Instrument air supply
P-41	12.0	0.375	391	0	65	00	0°	0°	Chemical and volume control
P-42	22.0	0.875	391	0	60	00	0°	0°	Spare
P-43	16.0	0.844	391	0	57	30	0°	0°	Spare
P-44	10.0	0.365	387	0	127	30	0°	0°	Reactor coolant
P-45	16.0	0.844	387	0	122	30	0°	0°	H_2 monitoring system
P-47	10.0	0.365	387	0	112	30	0°	0°	Waste disposal
P-48	10.0	0.365	387	0	107	30	0°	0°	Component cooling
P-49	22.0	0.875	387	0	102	30	0°	0°	Spare
P-50	24.0	0.688	387	0	97	30	0°	0°	Safety injection
P-51	24.0	0.688	387	0	72	30	0°	0°	Safety injection
P-52	16.0	0.844	387	0	67	30	0°	0°	Process radiation monitoring
P-53	14.0	0.375	387	0	62	30	0°	0°	Chemical and volume control
P-54	22.0	0.875	387	0	57	30	0°	0°	Spare
P-55	10.0	0.365	383	0	125	00	0°	0°	Safety injection
P-56	10.0	0.365	383	0	120	00	0°	0°	Service air
P-57	8.0	0.322	383	0	115	00	0°	0°	Fuel pool cooling and cleanup
P-59	16.0	0.375	383	0	105	00	0°	0°	Safety injection
P-60	14.0	0.375	383	0	100	00	0°	0°	Safety injection
P-61	16.0	0.843	383	0	75	00	0°	0°	Spare
P-63	16.0	0.844	383	0	65	00	0°	0°	Spare (blind flanged to accommodate use during outages)
P-64	16.0	0.844	383	0	60	00	0°	0°	Spare (blind flanged to accommodate use during outages)
P-65	10.0	0.365	379	0	127	30	0°	0°	Reactor bldg. and cont. drn. to rad. equip. drains

TABLE 3.8-1 (Cont'd)

PENETRATION NUMBER	SIZE	WALL THICKNESS	ELEVATION (ft-in.)		AZIMUTH (degree-min)		VERTICAL SKEW ANGLE	HORIZONTAL SKEW ANGLE	DESCRIPTION
P-66	24.0	0.688	379	0	122	30	0°	0°	Safety injection
P-68	24.0	0.688	379	0	112	30	0°	0°	Residual heat removal
P-69	16.0	0.844	379	0	107	30	0°	0°	Off-gas system
P-70	12.0	0.375	379	0	102	30	0°	0°	Process sampling
P-71	14.0	0.375	379	0	97	30	0°	0°	Chemical and volume control
P-72	16.0	0.375	379	0	72	30	0°	0°	Spare
P-73	16.0	0.844	379	0	67	30	0°	0°	Safety injection
P-74	16.0	0.844	379	0	62	30	0°	0°	Spare (blind flanged to accommodate use during outages)
P-75	24.0	0.688	379	0	57	30	0°	0°	Residual heat removal
P-76	34.0	1.000	390	0	Note 1		-9°-30 min	0°	Feedwater
P-77	54.0	1.375	386	6	Note 1		-4°-45 min	0°	Main steam
P-78	54.0	1.375	386	6	Note 1		-4°-45 min	0°	Main steam
P-79	34.0	1.000	390	0	Note 1		+8°-15 min	0°	Feedwater
P-80	12.0	0.375	388	0	Note 1		+9°-30 min	0°	Steam generator blowdown
P-81	12.0	0.375	386	6	Note 1		9°-30 min	0°	Steam generator blowdown
P-82	12.0	0.375	385	0	Note 1		9°-30 min	0°	Steam generator blowdown
P-83	12.0	0.375	383	6	Note 1		9°-30 min	0°	Steam generator blowdown
P-84	34.0	1.000	390	0	Note 2		-9°-30 min	0°	Feedwater
P-85	58.0	1.500	386	6	Note 2		-4°-45 min	0°	Main steam
P-86	58.0	1.500	386	6	Note 2		+4°-45 min	0°	Main steam
P-87	34.0	1.000	390	0	Note 2		+8°-15 min	0°	Feedwater
P-88	12.0	0.375	388	0	Note 2		+9°-30 min	0°	Steam generator blowdown
P-89	12.0	0.375	386	6	Note 2		+9°-30 min	0°	Steam generator blowdown
P-90	12.0	0.375	385	0	Note 2		+9°-30 min	0°	Steam generator blowdown
P-91	12.0	0.375	383	6	Note 2		+9°-30 min	0°	Steam generator blowdown
P-92	28.0	0.375	*		-		0°	0°	Safety injection, cont. spray
P-93	28.0	0.375	*		-		0°	0°	Safety injection, cont. spray
P-94	16.0	0.500	474	6	108	00	0°	0°	Mini-flow purge exhaust
P-95	60.0	1.000	462	0	123	00	0°	0°	Containment purge exhaust
P-96	14.0	0.375	462	4	132	45	0°	0°	Mini-flow purge supply
P-97	60.0	1.000	462	4	139	00	0°	0°	Containment purge supply

TABLE 3.8-1 (Cont'd)

PENETRATION NUMBER	SIZE	WALL THICKNESS	ELEVATION (ft-in.)		AZIMUTH (degree-min)		VERTICAL SKEW ANGLE	HORIZONTAL SKEW ANGLE	DESCRIPTION
P-98	24.0	0.688	392	6	Note 3		-4°-10 min	0°	Fuel transfer tube
P-99	16.0	0.844	390	9	Note 1		-3°-30 min	0°	SG Wet Layup(Unit 1), Feedwater(Unit 2)
P-100	16.0	0.844	390	9	Note 1		+3°-30 min	0°	SG Wet Layup(Unit 1), Feedwater(Unit 2)
P-101	16.0	0.844	390	9	Note 2		-3°-30 min	0°	SG Wet Layup(Unit 1), Feedwater(Unit 2)
P-102	16.0	0.844	390	9	Note 2		+3°-30 min	0°	SG Wet Layup(Unit 1), Feedwater(Unit 2)
E-1	24.0	0.500	435	0	120	15	0°	0°	Reactor coolant pump
E-2	18.0	0.438	439	3	123	45	0°	0°	CRD Fan
E-3	12.0	0.406	439	3	127	15	0°	0°	Misc. ESF instrumentation
E-4	12.0	0.406	439	3	130	45	0°	0°	Reactor cont. fan cooler
E-5	12.0	0.406	439	3	134	15	0°	0°	Misc. control
E-6	12.0	0.406	439	3	137	45	0°	0°	Misc. control
E-7	12.0	0.406	439	3	141	15	0°	0°	Process instr.
E-8	12.0	0.406	439	3	120	15	0°	0°	Misc. power
E-9	12.0	0.406	435	0	123	45	0°	0°	Pressurizer heater
E-10	12.0	0.406	435	0	127	15	0°	0°	Pressurizer heater
E-11	12.0	0.406	435	0	130	45	0°	0°	Misc. power
E-12	12.0	0.406	435	0	134	15	0°	0°	Misc. control
E-13	12.0	0.406	435	0	137	45	0°	0°	Misc. instrumentation
E-14	12.0	0.406	435	0	141	15	0°	0°	Neutron monitoring
E-15	12.0	0.406	421	9	120	15	0°	0°	Reactor cont. fan cooler
E-16	12.0	0.406	421	9	123	45	0°	0°	Pressurizer heater
E-17	12.0	0.406	421	9	127	15	0°	0°	Pressurizer heater
E-18	12.0	0.406	421	9	130	45	0°	0°	Misc. control
E-19	12.0	0.406	421	9	134	15	0°	0°	Misc. power
E-20	24.0	0.500	421	9	137	45	0°	0°	Reactor coolant pump
E-21	18.0	0.438	421	9	141	15	0°	0°	CRD fan, ltg. and weld recep., cav. fan
E-22	12.0	0.406	417	6	120	15	0°	0°	Neutron monitoring
E-23	12.0	0.406	417	6	123	45	0°	0°	Incore-flux mapping det. instrumentation
E-24	12.0	0.406	417	6	127	15	0°	0°	Process instrumentation
E-25	12.0	0.406	417	6	130	45	0°	0°	Misc. instrumentation
E-26	12.0	0.406	417	6	134	15	0°	0°	Spare

TABLE 3.8-1 (Cont'd)

PENETRATION NUMBER	SIZE	WALL THICKNESS	ELEVATION (ft-in.)		AZIMUTH (degree-min)		VERTICAL SKEW ANGLE	HORIZONTAL SKEW ANGLE	DESCRIPITION
E-27	12.0	0.406	417	6	137	45	0°	0°	Misc. ESF instrumentation
E-28	18.0	0.438	417	6	141	15	0°	0°	Crane feed
E-29	12.0	0.406	439	3	174	45	0°	0°	Misc. instrumentation
E-30	12.0	0.406	439	3	178	15	0°	0°	Spare
E-31	12.0	0.406	439	3	181	45	0°	0°	Control rod drive unit
E-32	12.0	0.406	439	3	185	15	0°	0°	Control rod drive unit
E-33	24.0	0.500	439	3	188	45	0°	0°	Reactor coolant pump
E-34	12.0	0.406	439	3	192	15	0°	0°	Spare
E-35	12.0	0.406	439	3	195	45	0°	0°	Process instrumentation
E-36	12.0	0.406	435	0	174	45	0°	0°	Spare
E-37	12.0	0.406	435	0	178	15	0°	0°	Control rod drive unit
E-38	12.0	0.406	435	0	181	45	0°	0°	Control rod drive unit
E-39	12.0	0.406	435	0	185	15	0°	0°	Control rod drive unit
E-40	12.0	0.406	435	0	188	45	0°	0°	Reactor cont. fan cooler
E-41	12.0	0.406	435	0	192	15	0°	0°	Misc. power
E-42	12.0	0.406	435	0	195	45	0°	0°	Neutron monitoring
E-43	12.0	0.406	421	9	174	45	0°	0°	Misc. control
E-44	12.0	0.406	421	9	178	15	0°	0°	Misc. control
E-45	12.0	0.406	421	9	181	45	0°	0°	Misc. power
E-46	12.0	0.406	421	9	185	15	0°	0°	Misc. instr.
E-47	24.0	0.500	421	9	188	45	0°	0°	Reactor coolant pump
E-48	12.0	0.406	421	9	192	15	0°	0°	Misc. power
E-49	12.0	0.406	421	9	195	45	0°	0°	Reactor cont. fan cooler
E-50	12.0	0.406	417	6	174	45	0°	0°	Neutron monitoring
E-51	12.0	0.406	417	6	178	15	0°	0°	Process instrumentation
E-52	12.0	0.406	417	6	181	45	0°	0°	Spare
E-53	12.0	0.406	417	6	185	15	0°	0°	Spare
E-54	12.0	0.406	417	6	188	45	0°	0°	Spare
E-55	12.0	0.406	417	6	192	15	0°	0°	Spare
E-56	12.0	0.406	417	6	195	45	0°	0°	Spare

TABLE 3.8-1 (Cont'd)

PENETRATION NUMBER	SIZE	WALL THICKNESS	ELEVATION (ft-in.)	AZIMUTH (degree-min)	VERTICAL SKEW ANGLE	HORIZONTAL SKEW ANGLE	DESCRIPITION

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TABLE 3.8-2

LIST OF SPECIFICATIONS, CODES, AND STANDARDS*

SPECIFICATION REFERENCE NUMBER	SPECIFICATION OR STANDARD DESIGNATION	TITLE
1	ACI 318-71,77,83	Building Code Requirements for Reinforced Concrete
2	ACI 301	Specifications for Structural Concrete for Buildings
3	ACI 347 ANSI A145.1	Recommended Practice for Concrete Formwork
4	ACI 305 ANSI A170.1	Recommended Practice for Hot Weather Concreting
5	ACI 211.1	Recommended Practice for Selecting Proportions for Normal Weight Concrete
6	ACI 304	Recommended Practice for Measuring, Mixing, Trans-porting, and placing concrete
7	ACI 315	Manual of Standard Practice for Detailing Reinforced Concrete Structures
8	ACI 306	Recommended Practice for Cold Weather Concreting
9	ACI 309	Recommended Practice for Consolidation of Concrete
10	ACI 308	Recommended Practice for Curing Concrete
11	ACI 214 ANSI A146.1	Recommended Practice for Evaluation of Compression Test Results of Field

* References to edition dates are shown for the codes used in the design of safety-related structures. All other specifications delineated in this table are recommended practices and material specifications that do not affect the design of safety-related structures.

TABLE 3.8-2 (Cont'd)

SPECIFICATION REFERENCE NUMBER	SPECIFICATION OR STANDARD DESIGNATION	TITLE
12	ACI 311	Recommended Practice for Concrete Inspection
13	ACI 304	Preplaced Aggregate Concrete for Structural and Mass Concrete
14	Report by ACI Committee 304	Placing Concrete by Pumping Method
15	AISC-69,78	Specification for the Design, Fabrication, and Erection of Structural Steel for Building
16	AWS D1.1**	Structural Welding Code
17	ASME	Boiler & Pressure Vessel Code, Section III
	ASME-1971, S73	Division 1, Subsection NE
	ASME-1974, S75	Division 1, Subsection NF
	ASME-1973	Division 2, Proposed Standard Code for Concrete Reactor Vessels and Containments Issued for Trial Use and Comments
	ASME-1980	Division 2, CC 6000
	ASME-1992	1992 Addenda, Division 1, Section XI, Subsection IWL, IWE
18	American Public Health Assoc. (APHA)	Test Methods Sulphides in Water, Standard Methods for the Examination of Water and Waste Water
19	ASTM	Annual Books of ASTM Standards
20	CRSI MSP-1	Manual of Standard Practice

** Clarifications to, and deviations from portions of AWS D1.1, "Structural Welding Code," are made based on engineering” evaluations. Visual weld inspection requirements are based on guidelines in a document prepared by the Nuclear Construction Issues Group, NCIG-01, Revision 2, "Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants."

TABLE 3.8-2 (Cont'd)

SPECIFICATION REFERENCE NUMBER	SPECIFICATION OR STANDARD DESIGNATION	TITLE
21	ANSI N45.2.5	Proposed Supplementary Q.A. Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During Construction Phase of Nuclear Power Plants
22	CRD	Chief of Research and Development Standards, Department of the Army, Handbook for Concrete and Cement Volume I and II, Corps of Engineers U.S. Army
23	ACI-349-76,85	Code Requirements for Nuclear Safety Related Concrete Structures
24	AISI	Specification for design of cold-formed steel structural members

EXPLANATION OF ABBREVIATIONS

ACI - American Concrete Institute

AISC - American Institute of Steel Construction

AISI - American Iron and Steel Institute

ANSI - American National Standards Institute

ASME - American Society of Mechanical Engineers

ASTM - American Society of Testing Materials

AWS - American Welding Society

CRD - Chief of Research and Development Standards

CRSI - Concrete Reinforcing Steel Institute

NOTE: For exceptions and additions to these codes and standards refer to Appendix B.

TABLE 3.8-3

LOAD COMBINATIONS AND LOAD FACTORS

CONTAINMENT-SERVICE LOAD AND FACTORED LOAD CONDITIONS

LOAD COMBINATION CATEGORY (e)			LOAD FACTORS (d) (c)																		
			D	L	F	P'	P_a	P_o	T_o	T_a	E	E'	W	W'	R_o	R_a	Y_r	Y_m	Y_j	H'	M
I	CONSTRUCTION	1	0.75	0.75	1.0				0.75				0.75[a]								
		2	1.0	1.0	1.0				1.0												
II	TEST	3	1.0	1.0	1.0	1.0			1.0												
III	NORMAL	4	1.0	1.0	1.0			1.0	1.0						1.0						
IV	SEVERE	5	1.0	1.0	1.0			1.0	1.0				1.0		1.0						
	ENVIRONMENTAL	6	1.0	1.0	1.0			1.0	1.0		1.0				1.0						
V	ABNORMAL	7	1.0	1.0	1.0		1.5			1.0[b]						1.0					1.0
		8	1.0	1.0	1.0				1.0						1.0						
VI	EXTREME ENVIRONMENTAL	9	1.0	1.0	1.0			1.0	1.0					1.0	1.0						
		10	1.0	1.0	1.0			1.0	1.0			1.0			1.0						
		11	1.0	1.0	1.0			1.0	1.0						1.0					1.0	
VII	ABNORMAL/ SEVERE	12	1.0	1.0	1.0		1.25			1.0[b]	1.25					1.0	1.0	1.0	1.0		
	ENVIRONMENTAL	13	1.0	1.0	1.0		1.25			1.0[b]			1.25			1.0	1.0	1.0	1.0		

TABLE 3.8-3 (Cont'd)

LOAD COMBINATIONS AND LOAD FACTORS

CONTAINMENT-SERVICE LOAD AND FACTORED LOAD CONDITIONS

LOAD COMBINATION CATEGORY (e)			LOAD FACTORS										(d) (c)								
			D	L	F	P'	P_a	P_o	T_o	T_a	E	E'	W	W'	R_o	R_a	Y_r	Y_m	Y_j	H'	M
VIII	ABNORMAL/ EXTREME ENVIRONMENTAL	14	1.0	1.0	1.0		1.0			1.0[(b)]		1.0				1.0	1.0	1.0	1.0		

NOTES:
(a) For the construction category, the wind load for a ten year recurrence interval is used.
(b) T_a is based on a temperature corresponding to the factored pressure.
(c) Loads not applicable to a particular system are deleted.
(d) For any load combination, loads other than D are deleted when necessary to produce the most severe combination of loads.
(e) Categories I through IV present information for severe load designs; Categories V through VIII present information for fractured loads.

TABLE 3.8-4

DEFINITIONS OF STRUCTURAL TERMINOLOGY

	DEFINITIONS
	NOTE: These definitions apply to the terms used in Tables 3.8-7 through 3.8-11.
I LOADING CATEGORIES	
CONSTRUCTION	All events and loads during structural construction including the various stages of prestressing, but excluding those during testing.
TESTING	All events and loads applied during structural integrity tests and preoperational tests such as hydrostatic testing of equipment and the pressure tests. Each testing event is considered to be mutually exclusive of other testing events.
NORMAL	All events and loads that could reasonably be expected during the operation, shutdown, and normal maintenance of the power plant.
SEVERE ENVIRONMENTAL	All loads due to infrequent site-related environmental events like operating-basis earthquake and design wind.
ABNORMAL	All loads due to postulated accident events. They include pressure, temperature, pipe whip, jet impingement, and pipe reactions due to each break postulated for the design-basis accidents. This loading condition also includes plant-related nonenvironmental missiles. The loads from each postulated accident event are considered to be mutually exclusive of other postulated accidents.

TABLE 3.8-4 (Cont'd)

EXTREME ENVIRONMENTAL	All loads due to site-related environmental events which are credible but highly improbable. These events include the safe shutdown earthquake, design-basis tornado, probable maximum flood, and the postulated site-related accidents not included in the abnormal loading category.
ABNORMAL/SEVERE ENVIRONMENTAL	Loads due to the highly improbable simultaneous occurrence of abnormal and severe environmental loading categories. Only the specified combinations of these categories are considered.
ABNORMAL/EXTREME ENVIRONMENTAL	Loads due to the extremely improbable simultaneous occurrence of the abnormal and extreme environmental loading conditions. Only the specified combinations of these conditions are considered.

II LOADS

DEAD	D - The dead load of structure plus any other permanent load (except prestressing forces), including vertical and lateral pressures of liquids, piping, cable pans, weight of permanent equipment and its normal contents under operating conditions and the weight of soil cover for buried structures.
LIVE	L - Conventional floor and roof live loads, movable equipment loads, crane loads and other loads which vary in intensity and occurrence such as lateral soil pressure. The dynamic effect of operating equipment is accounted for by the use of appropriate impact factors.
PRESTRESS	F - Loads resulting from the application of prestressing forces.
OPERATING PRESSURE	P_o - The normal operating pressure differentials.

TABLE 3.8-4 (Cont'd)

ACCIDENT PRESSURE	P_a - The design accident pressure load. This pressure is based upon peak calculated pressure with appropriate margin provided for uncertainties in this calculation: Containment internal pressure, + 50.0 psig. Containment external pressure, ± 3.0 psig.
TEST PRESSURE	P' - The containment's test pressure, + 57.5 psig.
OPERATING PIPE REACTION	R_o - Normal operating or shutdown pipe reactions at supports or anchor points, based on the most critical transient or steady-state condition.
ACCIDENT PIPE REACTION	R_a - Pipe reactions generated by the design-basis accident including R_o.
THERMAL	T_o - Thermal effects and loads during normal operating, shutdown, construction and test conditions, including average temperature and temperature gradients. The combination of internal and ambient temperatures which produces the most critical transient or steady-state thermal gradient is used. Climatic temperature ranges: Maximum outside temperature: 110° F Minimum outside temperature: -25.0° F.
ACCIDENT THERMAL	T_a - Thermal effects and loads generated by the design-basis accident including T_o.

TABLE 3.8-4 (Cont'd)

OPERATING BASIS EARTHQUAKE	E - Loads generated by the operating-basis earthquake (OBE), including dynamic lateral soil pressure and hydrodynamic groundwater pressure for a horizontal ground acceleration at foundation elevation of 0.09g.
SAFE SHUTDOWN EARTHQUAKE	E' - Loads generated by the safe shutdown earthquake (SSE), including dynamic lateral soil pressure and hydrodynamic groundwater pressure for a horizontal ground acceleration at foundation elevation of 0.20g.
WIND	W - Design Wind Load.
TORNADO	W_t - Tornado loads including the effects of wind pressure, differential pressure loads due to rapid atmospheric pressure change and missile impact.
MAXIMUM PROBABLE FLOOD	H' - Forces associated with the maximum probable flood, seiche, or precipitation.
MISSILE	M - Loads associated with missiles other than a tornado (see Section 3.5).
RESTRAINT	Y_r - Equivalent static load on the restraint generated by the reaction of broken high-energy pipe during the postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load.
JET IMPINGEMENT	Y_j - Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load.
PIPE WHIP MISSILE	Y_m - Missile impact equivalent static load on a structure generated by or during the postulated break, as from pipe whipping, and including an appropriate dynamic factor to account for the dynamic nature of the load.

TABLE 3.8-4 (Cont'd)

EXTERNAL PRESSURE	P_e - External pressure on containment, not considering P_a.
BREAK LOAD	R_r - Load associated with a high energy break of a piping system = $Y_m + Y_j + Y_r$.

TABLE 3.8-5

ALLOWABLE STRESSES AND STRAINS

CONTAINMENT LINER PLATE AND ANCHORAGES

LINER PLATE ALLOWABLES

STRESS/STRAIN ALLOWABLE*

CATEGORY	MEMBRANE	COMBINED MEMBRANE AND BENDING
Construction	$f_{st} = f_{sc} = 2/3F_y$	$f_{st} = f_{sc} = 2/3F_y$
Service	ξ_{sc} = 0.002 in./in.	ξ_{sc} = 0.004 in./in.
	ξ_{st} = 0.001 in./in.	ξ_{st} = 0.002 in./in.
Factored	ξ_{sc} = 0.005 in./in.	ξ_{sc} = 0.014 in./in.
	ξ_{st} = 0.003 in./in.	ξ_{st} = 0.010 in./in.

The allowables are defined as:

f_{st} = allowable liner plate tensile stress, psi;

f_{sc} = allowable liner plate compressive stress, psi;

ξ_{sc} = allowable liner plate compressive strain; and

ξ_{st} = allowable liner plate tensile strain.

*The types of strains limited by this table are strains induced by deformation or constraint.

TABLE 3.8-5 (Cont'd)

LINER ANCHOR ALLOWABLES

FORCE/DISPLACEMENT ALLOWABLES

CATEGORY	MECHANICAL LOADS	DISPLACEMENT LIMITED LOADS
Test	Lesser of $F_a = 0.67F_y$ $F_a = 0.33F_u$	$\delta_a = 0.25\ \delta_u$
Normal	Lesser of: $F_a = 0.67F_y$ $F_a = 0.33F_u$	$\delta_a = 0.25\ \delta_u$
Severe Environmental	Lesser of $F_a = 0.67F_y$ $F_a = 0.33F_u$	$\delta_a = 0.25\ \delta_u$
Extreme Environmental	Lesser of $F_a = 0.67F_y$ $F_a = 0.33F_u$	$\delta_a = 0.25\ \delta_u$
Abnormal	$F_a = 0.9F_y$ $F_a = 0.5F_u$	$\delta_a = 0.50\ \delta_u$
Abnormal/severe environmental	$F_a = 0.9F_y$ $F_a = 0.5F_u$	$\delta_a = 0.50\ \delta_u$
Abnormal/extreme environmental	$F_a = 0.9F_y$ $F_a = 0.5F_u$	$\delta_a = 0.50\ \delta_u$

TABLE 3.8-5 (Cont'd)

CATEGORY	LINER PLATE (stress/strains) MEMBRANE	COMBINED MEMBRANE AND BENDING	ANCHORAGES (force/displacement allowables) MECHANICAL LOADS	DISPLACEMENT LIMITED LOADS
Construction	$f_{st} = f_{sc} = 0.67F_y$	$f_{st} = f_{sc} = 0.67F_y$		
Test	$\epsilon_{sc} = 0.002$ in/in	$\epsilon_{sc} = 0.004$ in/in	Lesser of $F_a = 0.67F_y$	$\delta_a = 0.25\ \delta_u$
			$F_a = 0.33\ F_u$	
	$\epsilon_{st} = 0.001$ in/in	$\epsilon_{st} = 0.002$ in/in		
Normal	$\epsilon_{sc} = 0.002$ in/in	$\epsilon_{sc} = 0.004$ in/in		
	$\epsilon_{st} = 0.001$ in/in	$\epsilon_{st} = 0.002$ in/in		
Severe Environmental	$\epsilon_{sc} = 0.002$ in/in	$\epsilon_{sc} = 0.004$ in/in		
	$\epsilon_{st} = 0.001$ in/in	$\epsilon_{st} = 0.002$ in/in		
Abnormal	$\epsilon_{sc} = 0.005$ in/in	$\epsilon_{sc} = 0.014$ in/in	Lesser of $F_a = 0.90F_y$	$\delta_a = 0.5\ \delta_u$
			$F_a = 0.50\ F_u$	
	$\epsilon_{st} = 0.003$ in/in	$\epsilon_{st} = 0.010$ in/in		
Extreme Environmental	$\epsilon_{sc} = 0.002$ in/in	$\epsilon_{sc} = 0.004$ in/in		
	$\epsilon_{st} = 0.001$ in/in	$\epsilon_{st} = 0.002$ in/in		
Abnormal Severe	$\epsilon_{sc} = 0.005$ in/in	$\epsilon_{sc} = 0.014$ in/in		
	$\epsilon_{st} = 0.003$ in/in	$\epsilon_{st} = 0.010$ in/in		
Abnormal Extreme	$\epsilon_{sc} = 0.005$ in/in	$\epsilon_{sc} = 0.014$ in/in		
	$\epsilon_{st} = 0.003$ in/in	$\epsilon_{st} = 0.010$ in/in		

TABLE 3.8-6

PREDICTED CONTAINMENT DEFLECTIONS

DURING THE STRUCTURAL ACCEPTANCE TEST

I. RADIAL DEFLECTION OF THE CYLINDRICAL WALL

METER NUMBER	LOCATION	OUTWARD DEFLECTION

Security - Related Information Figure Withheld Under 10 CFR 2.390

II. VERTICAL GROWTH WITH REFERENCE TO THE FOOT OF THE CYLINDRICAL WALL

METER NUMBER	LOCATION	UPWARD VERTICAL GROWTH (inch)

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.8-6 (Cont'd)

METER NUMBER	LOCATION	UPWARD VERTICAL GROWTH (inch)
V-9	Dome apex	0.61
V-10	Dome at radius of 18'-3"	0.57
V-11	Dome at radius of 51'-3"	0.30

III. TANGENTIAL DEFLECTION OF EQUIPMENT HATCH OPENING

METER NUMBER	LOCATION	TANGENTIAL DEFLECTION (inch)
T1	Horizontal plane across opening	0.06
T2	Vertical plane across opening	0.03

TABLE 3.8-7

LOAD DEFINITIONS AND COMBINATIONS FOR CLASS MC CONTAINMENT COMPONENTS*

(For Definitions See Table 3.8-4)

LOADING CATEGORY	ITEM NUMBER	LOAD FACTORS													
									SEVERE ENVIRONMENTAL	ABNORMAL					EXTREME ENVIRONMENTAL
		D	L	R_o	T_o	P_o	P_e	P'	E	R_r	T_a	P_a	R_a	M	E'
Construction	1	1.0	1.0		1.0										
Test	2	1.0	1.0		1.0**			1.0							
Normal	3	1.0	1.0	1.0	1.0	1.0	1.0								
Severe Environmental	4	1.0	1.0	1.0	1.0	1.0	1.0		1.0						
Abnormal	5	1.0	1.0				1.0				1.0	1.0	1.0		
	6	1.0	1.0				1.0			1.0	1.0	1.0	1.0		
Extreme Environmental	7	1.0	1.0	1.0	1.0	1.0									1.0

*Does not include process piping penetrations.
**Temperature at time of test.

TABLE 3.8-7 (Cont'd)

LOAD FACTORS

LOADING CATEGORY	ITEM NUMBER								SEVERE ENVIRONMENTAL			ABNORMAL			EXTREME ENVIRONMENTAL
		D	L	R_o	T_o	P_o	P_e	P'	E	R_r	T_a	P_a	R_a	M	E'
Abnormal/Severe Environmental	8	1.0	1.0				1.0		1.0		1.0	1.0	1.0		
	9	1.0	1.0						1.0		1.0		1.0	1.0	
	10	1.0	1.0				1.0		1.0	1.0	1.0	1.0	1.0		
Abnormal/Extreme Environmental	11	1.0	1.0				1.0			1.0	1.0	1.0	1.0		1.0
	12	1.0	1.0				1.0			1.0	1.0	1.0	1.0		1.0

TABLE 3.8-8

PHYSICAL PROPERTIES FOR MATERIALS TO BE USED

FOR PRESSURE PARTS OR ATTACHMENT TO PRESSURE PARTS

MC COMPONENTS

MATERIAL SPECIFICATION	Su MINIMUM ULTIMATE TENSILE (ksi)	Sy MINIMUM YIELD AT THE AMBIENCE (ksi)	Sy MINIMUM YIELD AT 340°F (ksi)	Sm ASME CODE ALLOWABLE STRESS INTENSITY AT 340°f (ksi)	NOTES
Plate					
SA516 Gr 70	70	38	33.26	17.5	
SA516 Gr 60	60	32	27.94	15	
SA240 Tp 304	75	30	21.78	16.44	
Pipe					
SA106 Gr B	60	35	30.6	15	
SA333 Gr 6	60	35	30.6	15	
SA333 Gr 1	55	30	26.24	13.75	
SA312 Type 304	75	30	21.78	16.44	
SA376 Type 304	75	30	21.78	16.44	
Forgings and Fittings					
SA350 LF-1	60	30	26.24	15.0	
SA350 LF-2	70	36	31.96	17.5	
SA182 F304	70	30	21.78	16.44	
SA182 Gr-F316	75	30	18.22	18.28	
Bolting					
SA193 B7	115	95	83.98	23	Between 2.5 in. and 4 in. diameter
SA193 B7	125	105	93.06	25	Under 2.5 in. diameter

TABLE 3.8-8 (Cont'd)

MATERIAL SPECIFICATION	Su MINIMUM ULTIMATE TENSILE (ksi)	Sy MINIMUM YIELD AT THE AMBIENCE (ksi)	Sy MINIMUM YIELD AT 340°F (ksi)	Sm ASME CODE ALLOWABLE STRESS INTENSITY AT 340°f (ksi)	NOTES
SA194 Gr 7*	---	---	---	---	
SA320 L43	125	1-5	94.14	25	4 in. diameter and under

* No yield or tensile strength specified. Assume it is same as an equivalent grade in SA-193-B7.

TABLE 3.8-9

CATEGORY I LOAD COMBINATION TABLE STRUCTURAL STEEL ELASTIC DESIGN

LOADING CONDITIONS		LOAD FACTORS																
		D	L	S	R_o	R_a	E	E'	W	W_t	T_o	T_a	P_a	H'	M	Y_r	Y_j	Y_m
CONSTRUCTION	1	1.0	1.0						1.0		1.0							
TEST	2	1.0	1.0	1.0	1.0						1.0							
NORMAL	3	1.0	1.0	1.0	1.0						1.0							
SEVERE ENVIRONMENTAL	4	1.0	1.0		1.0				1.0		1.0							
	5	1.0	1.0		1.0		1.0				1.0							
ABNORMAL	6	1.0	1.0	1.0		1.0						1.0	1.0					
	7	1.0	1.0	1.0	1.0						1.0				1.0			
EXTREME ENVIRONMENTAL	8	1.0	1.0		1.0			1.0			1.0							
	$9_{(6)}$	1.0	1.0		1.0					1.0	1.0							
	$10_{(6)}$	1.0	1.0		1.0						1.0			1.0				

TABLE 3.8-9 (Cont'd)

LOAD FACTORS

LOADING CONDITIONS		D	L	S	R_o	R_a	E	E'	W	W_t	T_o	T_a	P_a	H'	M	Y_r	Y_j	Y_m
ABNORMAL/SEVERE ENVIRONMENTAL	11	1.0	1.0			1.0	1.0					1.0	1.0			1.0	1.0	1.0
ABNORMAL/EXTREME ENVIRONMENTAL	12	1.0	1.0			1.0		1.0				1.0	1.0			1.0	1.0	1.0

NOTES: (1) The construction loading combination is used to check the capacity of structural members for construction loading. The live load used is specified by the contractor to reflect shoring loads, etc. For the construction category, the wind load corresponding to the ten year recurrence interval is used.

(2) For any load combination, loads other than D are deleted when necessary to produce the most severe combination of loads.

(3) Loads not applicable to a particular system are deleted.

(4) For both E and E', the resultant effects (resultant stresses) for both horizontal and vertical earthquake forces are determined by combining the individual effects by the square root of the sum of the squares method.

(5) AISC allowables are calculated in accordance with Part I of AISC - 69.

(6) Loading conditions number 9 and 10 are not applicable to Byron River Screen House.

TABLE 3.8-10

CATEGORY I

LOAD COMBINATION TABLE

REINFORCED CONCRETE ULTIMATE STRENGTH DESIGN

LOAD CONDITIONS		LOAD FACTORS															
		D	L	R_o	R_a	E	E'	W	W_t	T_o	T_a	P_a	H'	M	Y_r	Y_j	Y_m
CONSTRUCTION	1	1.1	1.3					1.3		1.3							
TEST	2	1.1	1.3	1.3						1.3							
NORMAL	3	1.4	1.7	1.3						1.3							
SEVERE ENVIRONMENTAL	4	1.4	1.7	1.3				1.7		1.3							
	5	1.2		1.3				1.7		1.3							
	6	1.4	1.7	1.3		1.9				1.3							
	7																
ABNORMAL	8	1.0	1.0		1.0						1.0	1.5					
	9	1.0	1.0	1.0						1.0				1.0			
EXTREME ENVIRONMENTAL	10	1.0	1.0	1.0			1.0			1.0							
	11	1.0	1.0	1.0					1.0	1.0							
	12	1.0	1.0	1.0						1.0			1.0				
ABNORMAL/SEVERE ENVIRONMENTAL	13	1.0	1.0		1.0	1.25					1.0	1.25			1.0	1.0	1.0

TABLE 3.8-10 (Cont'd)

										LOAD FACTORS							
LOAD CONDITIONS		D	L	R_o	R_a	E	E'	W	W_t	T_o	T_a	P_a	H'	M	Y_r	Y_j	Y_m
ABNORMAL/EXTREME ENVIRONMENTAL	14	1.0	1.0		1.0		1.0				1.0	1.0			1.0	1.0	1.0

NOTES: (1) The construction loading combination is used to check the capacity of structural members for construction loading. The live load used is specified by the contractor to reflect shoring loads, etc. For construction category, the wind load for a ten year recurrence interval is used.

(2) For any load combination, loads other than D are deleted when necessary to produce the most severe combination of loads.

(3) Loads not applicable to a particular system are deleted.

(4) For both E and E', the resultant effects for both horizontal and vertical earthquake are determined by combining the individual effects by the square root of the sum of the squares.

TABLE 3.8-11

CATEGORY I

LOAD COMBINATION TABLE FOR RIVER SCREEN HOUSE (BYRON)

REINFORCED CONCRETE ULTIMATE STRENGTH DESIGN

LOAD CONDITION		LOAD FACTORS									
		D	L	R_o	E	E'	W	T_o	T_a	H"	H"'
CONSTRUCTION	1	1.1	1.3				1.3	1.3			
NORMAL	2	1.4	1.7	1.3				1.3			
SEVERE ENVIRONMENTAL	3	1.4	1.7	1.3			1.7	1.3			
	4	1.2		1.3			1.7	1.3			
	5	1.4	1.7	1.3	1.9			1.3		1.4	
	6	1.2		1.3	1.9			1.3		1.2	

TABLE 3-8.11 (Cont'd)

LOAD CONDITION		D	L	R_o	E	E'	W	T_o	T_a	H"	H"'
		LOAD FACTORS									
EXTREME ENVIRONMENTAL	7	1.0	1.0	1.0	1.0			1.0			1.0
	8	1.0	1.0	1.0		1.0		1.0			
	9	1.0	1.0	1.0		1.0		1.0		1.0	

NOTES:

(1) The construction loading combination is used to check the capacity of structural members for construction loading. The live load used is specified by the Contractor to reflect shoring loads, etc. For construction category, the wind load for a ten year recurrence interval is used.

(2) For any load combination, loads other than D are deleted when necessary to produce the most severe combination of loads.

(3) Loads not applicable to a particular system are deleted.

(4) For both E and E' the resultant effects for both horizontal and vertical earthquake are determined by combining the individual effects by the square root of the sum of the squares.

(5) H" refers to maximum flood of record.

(6) H"' refers to combined event flood.

TABLE 3.8-12

LOADING COMBINATIONS FOR

PENETRATION SLEEVES AND HEAD FITTINGS

LOADING COMPONENTS	LOADING CONDITIONS	DESIGN	EXPANSION	PRIMARY + SECONDARY	EMERGENCY	FAULTED	FATIGUE	TESTING
Internal and External Pressure	Maximum Operating			X	X	X		
	Design	X						
	Transient			X			X	
	Test							X
Temperatures	Normal Operating		X	X	X	X		X
	Design	X						
	Transient			X			X	
Weight		X		X	X		X	
Thermal Expansion			X	X			X	
Seismic	OBE	X		X			X	
	SSE					X[1]		
	Relative Displacement		X	X			X	
Fluid Dynamics	Hydraulic Transients	X		X	X		X	
	Main Steam SRV			X	X		X	
Pipe Break and Jet Impingement						X		

Note 1 - SSE is a faulted condition load. However, the type of analysis used (system inelastic and component elastic) does not require the SSE load as input.

TABLE 3.8-13

ALLOWABLE STRESSES FOR

PENETRATION SLEEVES AND HEAD FITTINGS

	ALLOWABLE STRESS VALUES FOR EACH LOADING CONDITION (Note 1)			
STRESS CATEGORY	NORMAL AND UPSET	DESIGN (Note 3)	EMERGENCY (Note 3)	FAULTED (Notes 3 and 4)
PRIMARY STRESSES				
GENERAL MEMBRANE (P_m)	(Note 2)	S_m	The larger of $1.2S_m$, or S_y	The larger of $0.7S_u$, or $S_y + \frac{S_u - S_y}{3}$
LOCAL MEMBRANE (P_L)	(Note 2)	$1.5S_m$	The larger of $1.8S_m$, or $1.5S_y$	The larger of $1.05S_u$, or $1.5S_Y + \frac{S_u - S_Y}{2}$
MEMBRANE + BENDING (P_L+P_B)	(Note 2)	$1.5S_m$	The larger of $1.8S_m$, or $1.5S_y$	The larger of $1.05S_u$, or $1.5S_Y + \frac{S_u - S_Y}{2}$

TABLE 3.8-13 (Cont'd)

STRESS CATEGORY	NORMAL AND UPSET	DESIGN (Note 3)	EMERGENCY (Note 3)	FAULTED (Notes 3 and 4)
SECONDARY STRESSES				
EXPANSION STRESSES (P_e)	$3S_m$			
PRIMARY + SECONDARY (P_L+P_B+P_e+Q)	$3S_m$			
PEAK STRESSES				
(F)	(Note 5)			

NOTES

1. Values for S_m, S_y, and S_u shall be temperature-dependent and taken from Section III Tables, as follows: S_y from Tables I-2.0; S_u from Tables I-3.0; S_m from Tables I-1.0 for non-MC components, and from Tables I-10.0 for MC components.

2. There are no specific limits established on the primary stresses that result from Operating Conditions.

3. Design, emergency and faulted conditions do not require Secondary and Peak stress evaluation.

4. The specified stress limits for faulted conditions are applicable for system inelastic and component elastic evaluation.

5. Used in combination with all primary and secondary stresses for calculating alternating stresses (for fatigue evaluation).

TABLE 3.8-14

COMPARISON OF THIN SHELL VS THICK SHELL ANALYTICAL RESULTS*

SECTION	RADIUS R(ft)	HEIGHT Z(ft)	MERIDIONAL FORCE K THIN	MERIDIONAL FORCE K THICK	MERIDIONAL FORCE K VAR. %	MERIDIONAL MOMENT K-Ft. THIN	MERIDIONAL MOMENT K-Ft. THICK	MERIDIONAL MOMENT K-Ft. VAR. %	HOOP FORCE K THIN	HOOP FORCE K THICK	HOOP FORCE K VAR. %	HOOP MOMENT K-(ft) THIN	HOOP MOMENT K-(ft) THICK	HOOP MOMENT K-(ft) VAR. %
1	71.8	20.0	259.0	244.0	6.2	92.60	78.40	18.1	428.0	400.0	7.0	15.70	8.2	9.1
2	71.8	100.0	259.0	246.0	5.3	0.17	0.00	--	504.0	503.0	0.2	0.03	6.2	--
3	71.8	182.0	260.0	247.0	5.3	120.00	77.30	5.5	147.0	149.2	1.5	21.00	15.2	38.2
4	71.8	187.5	259.0	248.0	4.4	324.00	261.00	24.1	67.6	52.2	29.5	56.10	49.0	14.5
5	70.4	200.0	242.0	254.0	4.7	1324.00	1208.00	9.6	34.0	68.5	50.4	106.00	148.9	28.8
6	65.5	202.0	245.0	269.0	8.9	594.0	609.00	2.5	15.0	32.3	53.6	60.10	74.3	19.1
7	59.5	206.0	309.0	301.0	2.7	290.00	270.00	7.4	77.0	37.2	107.0	14.90	13.7	8.9
8	35.0	220.0	387.0	430.0	10.0	29.30	30.00	2.3	312.0	390.0	20.0	24.60	27.0	8.9

*Variation is recorded as % of thick shell stress resultants.

Thin shell definition: Wall thickness < 0.1 (Radius of Curvature)

(Reference: W. Flugge, "Stresses in Shells"
Springer-Verlag, 1960)

TABLE 3.8-15

COMPARISON OF ALLOWABLE STRESSES IN PSI (INSPECTED WORKMANSHIP)

FOR UNREINFORCED CONCRETE MASONRY DESIGN

TYPE M MORTAR fm = 1350 psi; M_o = 2500 psi

NUMBER	STRESS	TYPE OF MASONRY UNIT*	NORMAL AND OBE LOAD COMBINATIONS: CRITERIA USED ON THE PROJECT [(a)]	NORMAL AND OBE LOAD COMBINATIONS: SEB INTERIM CRITERIA REV. 1 [(b)]	SSE LOAD COMBINATIONS: CRITERIA USED ON THE PROJECT [(a)]	SSE LOAD COMBINATIONS: SEB INTERIM CRITERIA REV. 1 [(b)]
1.	Tension Perpendicular to Bed Joints F_{t_1}	H	23	25	38	32
		S&G	39	40	65	52
2.	Tension Parallel to Bed Joints $F_{t_{11}}$	H	46	50	77	75
		S&G	78	75	130	112
3.	Shear	H	34	40	57	52
		S&G	34	40	57	52
4.	Flexure Compressive Stress F_m	All	405	445	676	1112
5.	Bearing: On Full Area	All	337	337	563	842
	On 1/3 rd area or less	All	506	506	845	1265
6.	Reinforcement Jt. Wire Fy = 65 ksi		0.5 Fy 30,000 psi	0.5 Fy 30,000 psi	0.9 Fy	0.9 Fy

(a) Criteria used on the project is compatible with NCMA-1974.
(b) SEB Interim Criteria Rev. 1 is compatible with ACI 531-79.

* H = Hollow Concrete Masonry
S = Solid Concrete Masonry
G = Grouted Concrete Masonry

TABLE 3.8-16

FLEXURAL STRENGTH HORIZONTAL SPAN,
NONREINFORCED CONCRETE MASONRY WALLS

CONSTRUCTION	MORTAR TYPE	LOADING TYPE	LOADING psf	MODULUS OF RUPTURE NET AREA (psi)	SAFETY FACTOR ACTUAL/ ALLOWABLE
8 inch Monowythe Hollow, 3-Core (Reference 1)	N	Uniform	127	102	4.13
	N	Uniform	136	141	4.41
	N	Uniform	127	132	4.13
	N	Uniform	169	176	5.50
	N	Uniform	173	180	5.63
	O	Uniform	123	128	4.00
	O	Uniform	158	164	5.13
8-inch Monowythe Hollow Joint Reinforced @ 16 inch center to center (Reference 1)	N	Uniform	149	155	4.84
	N	Uniform	160	166	5.19
	N	Uniform	193	201	6.28
	O	Uniform	150	156	4.88
	O	Uniform	186	193	6.03
8-inch Monowythe Hollow Joint Reinforced @ 8 inch center to center (Reference 1)	N	Uniform	203	211	6.59
	N	Uniform	196	204	6.38
	O	Uniform	202	210	6.56
	O	Uniform	195	203	6.34
8-inch Monowythe Hollow (Reference 2)	N	1/4 pt	56	58	1.81
	N	1/4 pt	38	39	1.22
	N	1/4 pt	61	63	1.97
	N	1/4 pt	60	62	1.94
	N	1/4 pt	69	71	2.22
	N	1/4 pt	93	96	3.00
8-inch Monowythe Hollow, 2-Core (Reference 3)	M	Center	199	217	4.72
	M	Center	176	192	4.17
	M	Center	151	165	3.59
4-2-4 Cavity Wall, Hollow Units (Reference 3)	M	Center	111	210	4.57
	M	Center	135	255	5.54
	M	Center	95	180	3.91
8-inch Monowythe Hollow 2-Core Joint Reinforced @ 8" center to center (Reference 3)	M	Center	159	173	3.76
	M	Center	159	173	3.76
	M	Center	191	208	4.52

TABLE 3.8-16 (Cont'd)

CONSTRUCTION	MORTAR TYPE	LOADING TYPE	LOADING psf	MODULUS OF RUPTURE NET AREA (psi)	SAFETY FACTOR ACTUAL/ ALLOWABLE
4-2-4 Cavity of	M	Center	159	300	6.52
Hollow Units Tied	M	Center	159	300	6.52
with Joint	M	Center	159	300	6.52
Reinforced @ 8"					
center to center					
(Reference 3)					
4-inch Hollow	N	Center	138	365	11.41
Monowythe	N	Center	157	415	12.97
(Reference 4)	N	Center	101	268	8.38
8-inch Hollow	M	Center	268	202	4.39
Monowythe	M	Center	314	237	5.15
(Reference 4)	M	Center	314	237	5.15
8-inch Hollow	N	Center	277	210	6.56
Monowythe	N	Center	314	237	7.41
(Reference 4)	N	Center	314	237	7.41
8-inch Hollow	O	Center	259	195	6.09
Monowythe	O	Center	277	210	6.56
(Reference 4)	O	Center	277	210	6.56
8-inch Hollow	M	Center	268	202	4.39
Monowythe	M	Center	297	224	4.87
(Reference 4)	M	Center	277	210	4.56
8-inch Hollow	N	Center	277	210	6.56
Monowythe	N	Center	259	195	6.09
(Reference 4)	N	Center	297	224	7.00
8-inch Hollow	O	Center	360	271	8.45
Monowythe	O	Center	297	224	7.00
(Reference 4)	O	Center	268	202	6.31
12-inch Hollow	N	Center	352	142	4.44
Monowythe	N	Center	314	127	3.97
(Reference 4)	N	Center	333	134	4.19

a) Total number of tests without joint reinforcement = 43
Average safety factor (SF) = 5.3

b) Total number of tests with joint reinforcement = 15 Average safety factor (SF) = 5.6

TABLE 3.8-16 (Cont'd)

REFERENCES:

1. Hedstrom, R. O., "Load Tests of Patterned Concrete Masonry Walls," Proceedings, American Concrete Institute, Vol. 57, p. 1265, 1961.

2. Fishburn, Cyrus C., "Effect of Mortar Properties on Strength of Masonry Monograph 36," National Bureau of Standards, 1961.

3. Cox, F. W., and Ennenga, J. L., Transverse Strength of Concrete Block Walls, "Proceedings," ACI, Vol. 54, p. 951, 1958.

4. Livingston, A. R., Mangotich, E., and Dikkers, R. "Flexural Strength of Hollow Unit Concrete Masonry Walls in the Horizontal Span," Technical Report No. 62, NCMA, 1958.

TABLE 3.8-17

SUMMARY OF MASONRY WALLS WITH STEEL COLUMNS
BASED ON DESIGN PARAMETERS FOR BYRON UNIT 1

TYPE OF ASPECT RATIO	h/s	APPROXIMATE NUMBER OF MASONRY WALLS WITH STEEL COLUMNS				
		GROUP 1 A*>75 I*>500	GROUP 2 A*=38-75 I*=100-500	GROUP 3 A*=25-37 I*=50-99	GROUP 4 A*<25 I*<50	ALL VALUES OF A* AND I*
1	≤2.0	47	19	10	12	88
2	>2.0 and ≤3.0	6	8	2	3	19
3	>3.0	3	-	-	4	7
All Types	All values	56	27	12	19	114

TABLE 3.8-18

AMPLIFICATION FACTORS FOR TYPICAL EXPANSION ANCHOR BASE PLATES WITH WEDGE TYPE ANCHORS

PLATE NO.	ANCHOR ASSEMBLY	LOAD	MAX. ANCHOR REACTION (FLEXIBLE PLATE ANALYSIS)	MAX. ANCHOR REACTION (RIGID PLATE ANALYSIS)	AMPLIFICATION FACTOR
1	9 x 9 x 1/2 in.	12.8 k (tension)	3.2	3.2	1.0
	4 anchors 1/2 in.	43.6 in-k (moment)	2.85	3.2	1.0
2	9 x 15 x 1/2 in.	33.7 k (tension)	7.0	5.62	1.25
	6 anchors 1/2 in.	199 in-k (moment)	7.0	6.62	1.06
3	21 x 21 x 7/8 in.	114 k (tension)	16.0	14.25	1.12
	8 anchors 3/4 in.	900 in-k (moment)	16.0	14.7	1.09

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

The following five operating conditions as defined in Section III of the ASME B&PV Code are considered in the design of the reactor coolant system, RCS component supports, and reactor internals.

a. Normal Conditions

   Any condition in the course of startup, operation in the design power range, hot standby and system shutdown, other than upset, emergency, faulted, or testing conditions.

b. Upset Conditions (Incidents of Moderate Frequency)

   Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition shall be included in the Design Specifications.

c. Emergency Conditions (Infrequent Incidents)

   Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not cause more than 25 stress cycles having an S value greater than that for 10^6 cycles from the applicable fatigue design curves of the ASME Code Section III.

d. Faulted Conditions (Limiting Faults)

Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that consideration of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

e. Testing Conditions

Testing conditions are those pressure overload tests including hydrostatic tests, pneumatic tests, and leak tests specified. Other types of tests shall be classified under normal, upset, emergency, or faulted conditions.

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgment and experience. The transients selected are representative of operating conditions which prudently should be considered to occur during plant operation and are sufficiently severe or frequent enough to be of possible significance to component cyclic behavior. The transients selected may be regarded as a conservative representation of transients which, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant.

The following design conditions are given in the equipment specifications for RCS components.

The design transients and the number of cycles of each that is normally used for fatigue evaluations are shown in Table 3.9-1. In accordance with ASME III, emergency and faulted conditions are not included in fatigue evaluations.

Normal Conditions

The following primary system transients are considered normal conditions:

a. heatup and cooldown at 100°F/hr,

b. unit loading and unloading at 5% of full power/min,

c. step load increase and decrease of 10% of full power,

d. large step load decrease with steam dump,

e. steady-state fluctuations,

f. feedwater cycling at hot shutdown,

g. loop out of service,

h. unit loading and unloading between 0% and 15% of full power,

i. boron concentration equalization,

j. refueling,

k. turbine roll test,

l. primary side leak test,

m. secondary side leak test, and

n. tube leakage test.

o. recovery of main feedwater flow after isolation (Unit 1 Steam Generators only)

<u>Heatup and Cooldown at 100°F/hr.</u>

The design heatup and cooldown cases are conservatively represented by continuous operations performed at a uniform temperature rate of 100°F/hr. (These operations can take place at a lower rate approaching the minimum of 0°F/hr.)

For these cases, the heatup occurs from ambient (assumed to be 120°F) to the no-load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F/hr may not be attained because of other limitations such as:

a. Material ductility considerations which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature, which are below the design rate of 100°F per hour.

b. Slower initial heatup rates when using pump energy only.

c. Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal sampling, water chemistry, and gas adjustments.

The number of such complete heatup and cooldown operations is specified as 200 each, which corresponds to five such occurrences per year for the 40-year plant design life.

Unit Loading and Unloading at 5% of Full Power per Minute

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5%/min between 15% load and full load. This load swing is the maximum possible consistent with operation under automatic reactor control. The reactor temperature will vary with load as prescribed by the reactor control system. The number of loading and unloading operations is defined as 13,200. One loading operation per day yields 14,600 such operations during the 40-year design life of the plant. By assuming a 90% availability factor, this number is reduced to 13,200 once the anticipated unloadings due to trips from transients are subtracted out. These unloadings are counted separately.

For the Unit 1 Steam Generators, the equipment designer assumed plant unloading to be 12,240 operations. This is 960 less operations than loading due to the designer not including 200 large step decreases from 100% to 5% power and 760 occurrences of main feedwater flow termination. This difference (960) is accounted for in other design transients.

Step Load Increase and Decrease of 10% of Full Power

The ±10% step change in load demand is a transient which is assumed to be a change in turbine control valve opening due to disturbances in the electrical network into which the plant output is tied. The reactor control system is designed to restore plant equilibrium without reactor trip following a ±10% step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15% and 100% full load, the power range for automatic reactor control. In effect, during load change conditions, the reactor control system attempts to match turbine and reactor outputs in such a manner that (T_{ave} - T_{ref}) error is minimized and reactor coolant temperature is restored to its programmed setpoint at a sufficiently slow rate to prevent excessive pressurizer pressure change.

Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the RCS average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With the load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature setpoint change is made as a function of turbine-generator load as determined by first-stage turbine pressure measurement. The pressurizer pressure will also ultimately decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. During the decreasing pressure transient, the saturated water in the pressurizer flashes, which reduces the rate of pressure decrease. Subsequently the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature initially decreases. The control system automatically withdraws the control rods to increase core power and $T_{ave.}$ The increasing pressure transient due to pressurizer insurge is reversed by actuation of the pressurizer sprays, and the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient.

The number of each operation is specified at 2000 times or 50 per year for the 40-year plant design life.

Large Step Load Decrease With Steam Dump

This transient applies to a step decrease in turbine load from full power, of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump that will prevent both reactor trip and lifting of steam generator safety valves. Thus, since the Byron/Braidwood plants are designed to accept a step decrease of 50% from full power, the steam dump system provides the heat sink to accept 40% of the turbine load. The remaining 10% of the total step change is assumed by the reactor control system (control rods). If the steam dump system were not available to cope with this transient, the mismatch between turbine demand and reactor power would cause a reactor trip and lifting of steam generator safety valves.

The number of occurrences of this transient is specified at 200 times or five per year for the 40-year plant design life.

Steady-State Fluctuations

It is assumed that the reactor coolant temperature and pressure at any point in the system vary around the nominal (steady-state) values. For design purposes two cases are considered:

a. Initial Fluctuations - These are due to control rod cycling during the first 20 full-power months of reactor operation. Temperature is assumed to vary ± 3° and pressure by ± 25 psi, once during each 2-minute period. The total number of occurrences is limited to 1.5×10^5. These fluctuations are assumed to occur consecutively, and not simultaneously with the random fluctuations.

b. Random Fluctuations - Temperature is assumed to vary by ± 0.5°F and pressure by ± 6 psi, once every 6 minutes. With a 6-minute period, the total number of occurrences during the plant design life does not exceed 3.0 x 10^6.

For the Unit 1 Steam Generators, the equipment designer assumed that local variations (of the nominal/steady state values) may occur over a range of frequencies, but for design purposes the temperature is assumed to vary by ± 0.5°F and pressure by ±6 psi, once every 6 minutes. With a 6 minute period, the total number of occurrences during the plant design life does not exceed 3.5x10^6.

Feedwater Cycling at Hot Shutdown

These transients can occur when the plant is at "no load" conditions, during which intermittent feeding of 32°F feedwater into the steam generators is assumed. Due to fluctuations arising from this mode of operation, the reactor coolant average temperature decreases to a lower value and then immediately begins to return to normal no-load temperature. This transient is assumed to occur 2000 times over the life of the plant.

Loop Out of Service

The plant design considerations include the possibility of operation at power with a single loop out of service. Although Technical Specifications require all reactor coolant loops to be in operation during startup and power operation, removing a loop from service would be accomplished by reducing power level and tripping a single reactor coolant pump.

It is assumed that this transient occurs twice per year or 80 times in the life of the plant. Conservatively, it is assumed that all 80 occurrences can occur in the same loop. (It must be assumed that the whole RCS is subjected to 80 transients, while each loop is also subjected to 80 inactive loop transients.)

To return an inactive loop to service, the power level would be reduced to approximately 10% and the pump started. It is assumed that an inactive loop is inadvertently started up at maximum allowable power level 10 times over the life of the plant. (This transient is covered under upset conditions.) Thus, the normal startup of an inactive loop is assumed to occur 70 times during the life of the plant.

<u>Unit Loading and Unloading Between 0% and 15% of Full Power</u>

The unit loading and unloading cases between 0% and 15% power are represented by continuous and uniform ramp power changes, requiring 30 minutes for loading and 5 minutes for unloading. During loading, reactor coolant temperatures are increased from the no-load value to the normal load program temperatures at the 15% power level. The reverse temperature change occurs during unloading.

Prior to loading, it is assumed that the plant is at hot shutdown conditions, with 32°F feedwater cycling. During the 2-hour period following the beginning of loading, the feedwater temperature increases from 32°F to 300°F due to steam dump and turbine startup heat input to the feedwater. Subsequent to unloading, feedwater heating is terminated, steam dump is reduced to residual heat removal requirements, and feedwater temperature decays from 300°F to 32°F.

The number of these loading and unloading transients is assumed to be 500 each during the 40-year plant design life. The value 500 is based upon the Westinghouse T_{HOT} Reduction Report (WCAP-11388) which revised NSSS design transients, as appropriate, to envelop operating conditions over the revised RCS temperature range.

For the Unit 1 Steam Generators, the equipment designer assumed the unit loading and unloading cases between zero and 15 percent power are represented by non-uniform ramp power changes. During loading, reactor coolant temperatures are increased from hot shutdown temperature to the normal load program temperatures of 15 percent power. The reverse temperature change occurs during unloading. Prior to loading, it is assumed that the plant is at hot shutdown conditions with cyclic flow of feedwater at 32°F. In the time following the beginning of loading, the feedwater temperature increases from 32°F to 300°F in stages due to steam dump and turbine startup heat inputs to the feedwater. Subsequent to unloading, feedwater heating is terminated, steam dump is reduced to residual heat removal requirements and feedwater temperature decreases from 300°F to 32°F. The interval of time between one unloading transient and the subsequent loading transient determines whether the loading transient is classified as a "Cold T-G (Turbine Generator) Unit Startup" or a "Hot T-G Unit Startup" as noted in the equipment design specification.

Boron Concentration Equalization

Following any large change in boron concentration in the RCS, spray is initiated in order to equalize concentration between the loops and the pressurizer. This can be done by manually operating the pressurizer backup heaters, thus causing a pressure increase, which will initiate spray. The proportional sprays maintain the pressure at approximately 2250 psia by matching the heat input from the backup heater. This operational mode is continued until the concentration is equalized. For design purposes, it is assumed that this operation is performed once after each load change in the design load follow cycle. With two load changes per day and a 90% plant availability factor over the 40-year design life, the total number of occurrences is 26,400.

Refueling

At the end of plant cooldown, the fluid in the RCS is at 140°F. At this time the vessel head is removed and the refueling canal is filled. This is done by pumping water from the refueling water storage tank, which is outside and conservatively assumed to be at 32°F, into the loops. It is conservatively assumed that the cold water flows directly into the vessel and that all the fluid in the RCS is replaced with the colder water within 10 minutes.

This operation is assumed to occur twice per year or 80 times over the life of the plant.

Turbine Roll Test

This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power will be used to heat the reactor coolant to operating temperature (no-load conditions) and the steam generated will be used to perform a turbine roll test. However, the plant cooldown during this test will exceed the 100°F/hr design rate.

The number of such test cycles is specified at 20 times, to be performed at the beginning of plant operating life prior to irradiation. Since this transient occurs before plant startup, the number of cycles is independent of other operating transients.

Primary Side Leakage Test

Subsequent to each time the primary system has been opened, a leakage test will be performed. During this test the primary system pressure is raised to nominal operating pressure with the system temperature above the minimum temperature imposed by reactor vessel material ductility requirements, while the system is checked for leaks.

In actual practice, the primary system will be pressurized to nominal operating pressure as measured at the pressurizer, to prevent the pressurizer safety valves from lifting during the leak test.

During this leakage test, the secondary side of the steam generator must be pressurized so that the pressure differential across the tube sheet does not exceed 1600 psi. This is accomplished in mode 3 using feedwater tempering flow and the turbine bypass valves for pressure control. For design purposes it is assumed that 200 cycles of this test will occur during the 40-year life of the plant.

Secondary Side Leakage Test

During the life of the plant it may be necessary to check the secondary side of the steam generator (particularly, the manway closure) for leakage. Secondary side leakage testing is performed in accordance with the applicable ASME code. In order not to exceed a secondary side to primary side pressure differential of 670 psi, the primary side must also be pressurized.

The primary system must be above the minimum temperature imposed by reactor vessel material ductility requirements. It is assumed that this test is performed 80 times during the 40-year life of the plant.

Tube Leakage Test

During the life of the plant it may be necessary to check the steam generator for tube leakage and tube to tube sheet leakage. This is done by visual inspection of the underside (channel head side) of the tube sheet for water leakage, with the secondary side pressurized. Tube leakage tests are performed during plant cold shutdowns.

For these tests the secondary side of the steam generator is pressurized with water, initially at a relatively low pressure, and the primary system remains depressurized. The underside of the tube sheet is examined visually for leaks. If any are observed, the secondary side is then depressurized and repairs made by tube plugging. The maximum (final) secondary side test pressure reached is 840 psig.

The total number of tube leakage test cycles is defined as 800 during the 40-year life of the plant. For the Unit 1 Steam Generators, the equipment designer assumed 720 test cycles. Following is a breakdown of the anticipated number of occurrences at each secondary side test pressure:

Test Pressure, psig	Number of Occurrences
200	400
400	200
600	120
840	(a) 80
	(b) 0 Unit 1 Steam Generators

For Unit 2, the primary and secondary sides of the steam generators will be at ambient temperatures during these tests. For Unit 1, the Steam Generator equipment designer assumed the test temperature to be between 120°F to 250°F with the primary side at 0 pressure.

Recovery of Main Feedwater Flow After Isolation (Unit 1 Steam Generators only)

Recovery of main feedwater flow occurs after actuation of auxiliary feedwater flow. The re-establishment of main feedwater flow results in a linear ramp of the nozzle temperature from the auxiliary feedwater temperature of 32°F to 447°F in approximately 1 hour. The temperature of the feedwater being supplied to the steam generator will then ramp linearly to ambient temperature of 32°F in approximately 2 hours. An average steady-state flow rate of approximately 100 gpm is assumed.

Upset Conditions

The following primary system transients are considered upset conditions:

a. loss of load (without immediate reactor trip),

b. loss of power,

c. partial loss of flow,

d. reactor trip from full power,

e. inadvertent reactor coolant system depressurization,

f. inadvertent startup of an inactive loop,

g. control rod drop,

h. inadvertent emergency core cooling system actuation,

i. excessive feedwater flow, and

j. operating basis earthquake.

k. thermal stratification

l. cold overpressurization

Loss of Load (Without Immediate Reactor Trip)

This transient applies to a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip and represents the most severe pressure transient on the RCS under upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the reactor protection system (RPS). Since redundant means of tripping the reactor are provided as a part of the RPS, transients of this nature are not expected, but are included to ensure a conservative design.

The number of occurrences of this transient is specified at 80 times or two times per year for the 40-year plant design life.

Loss of Power

This transient applies to a loss of nonemergency a-c power situation involving the loss of offsite electrical power to the station, assumed to be operating initially at 100% power, followed by reactor and turbine trips. Under these circumstances, the reactor coolant pumps are deenergized and, following coastdown of the reactor coolant pumps, natural circulation in the system decays to some equilibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater, assumed to be at 32°F, from the auxiliary feedwater system operating from diesel-generator power. Steam is removed for reactor cooldown through atmospheric relief valves provided for this purpose.

The number of occurrences of this transient is specified at 40 times or one per year for the 40-year plant design life.

Partial Loss of Flow

This transient applies to a partial loss of flow from full power, in which a reactor coolant pump is tripped out of service as the result of a loss of power to that pump. The consequences of such an accident are a reactor and turbine trip, on low reactor coolant flow, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The flow reversal causes reactor coolant at cold leg temperature to pass through the steam generator and be cooled still further. This cooler water then flows through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizable reduction in the hot leg coolant temperature of the affected loop.

The number of occurrences of this transient is specified at 80 times or two times per year for the 40-year plant design life.

Reactor Trip From Full Power

A reactor trip from full power may occur from a variety of causes resulting in temperature and pressure transients in the RCS and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the RPS causes the control rods to move into the core.

Various moderator cooldown transients associated with reactor trips can occur as a result of excessive feed or steam dump after trip or large load increase. For design purposes, reactor trip is assumed to occur a total of 400 times or 10 times per year over the life of the plant. The various types of trips and the number of occurrences for each are as follows:

a. Reactor trip with no inadvertent cooldown - 230 occurrences.

b. Reactor trip with cooldown but no safety injection - 160 occurrences.

c. Reactor trip with cooldown actuating safety injection - 10 occurrences.

Inadvertent Reactor Coolant System Depressurization

Several events can be postulated as occurring during normal plant operation which will cause rapid depressurization of the RCS. These include:

a. actuation of a single pressurizer safety valve;

b. inadvertent opening of one pressurizer power-operated relief valve due either to equipment malfunction or operator error;

c. malfunction of a single pressurizer pressure controller causing two pressurizer spray valves to open;

d. inadvertent opening of one pressurizer spray valve, due either to equipment malfunction or operator error; and

e. inadvertent auxiliary spray.

Of these events, the pressurizer safety valve actuation causes the most severe transients, and is used as an "umbrella" case to conservatively represent the reactor coolant pressure and temperature variations arising from any of them.

When a pressurizer safety valve opens and remains open, the system rapidly depressurizes, the reactor trips, and the emergency core cooling system (ECCS) is actuated. Also, the passive accumulators of the ECCS are actuated when pressure decreases by approximately 1600 psi, about 12 minutes after the depressurization begins. The depressurization and cooldown are eventually terminated. All of these effects are completed within approximately 18 minutes. It is conservatively assumed that none of the pressurizer heaters are energized.

With pressure constant and safety injection in operation, boil-off of hot leg liquid through the pressurizer and open safety valve will continue.

For design purposes this transient is assumed to occur 20 times during the 40-year design life of the plant.

Inadvertent Startup of an Inactive Loop

The plant design considerations include the possibility of operation at power with a loop out of service. Although Technical Specifications require all reactor coolant loops to be in operation during startup and power operation, inadvertent startup of an inactive loop could occur with a loop out of service. With the plant operating at the maximum allowable power level, the reactor coolant pump in the inactive loop is started as a result of operator error. Reactor trip occurs on high nuclear flux. This transient is assumed to occur 10 times during the life of the plant.

Control Rod Drop

This transient occurs if a bank of control rods drops into the fully inserted position due to a single component failure. The reactor is tripped on either low pressurizer pressure or manually. It is assumed that this transient occurs 80 times over the life of the plant.

Inadvertent Emergency Core Cooling System Actuation

A spurious safety injection signal results in an immediate reactor trip followed by actuation of the high head centrifugal charging pumps. These pumps deliver to the RCS cold legs. The initial portion of this transient is similar to the reactor trip from full power with no cooldown. Controlled steam dump and feedwater flow after trip removes core residual heat. Reactor coolant temperature and pressure decrease as the control rods move into the core.

Later in the transient, the injected water causes the RCS pressure to increase to the pressurizer power-operated relief valve setpoint and the primary and secondary temperatures to decrease gradually. The transient continues until the operator stops the charging pumps. It is assumed that the plant is then returned to no-load conditions, with pressure and temperature changes controlled within normal limits.

For design purposes this transient is assumed to occur 60 times over the life of the plant.

Excessive Feedwater Flow

An excessive feedwater flow transient is conservatively defined as an umbrella case to cover occurrence of several events of the same general nature. The postulated transient results from inadvertent opening of a feedwater control valve while the plant is at the hot standby or no-load condition, with the feedwater, condensate and heater drain systems in operation.

It is assumed that the stem of a feedwater control valve fails and the valve immediately reaches the full open position. In the steam generator directly affected by the malfunctioning valve ("failed loop"), the feedwater flow step increases from essentially zero flow to the value determined by the system resistance and the developed head of the feedwater. The feedwater entering the steam generator is conservatively assumed to be 32°F. Feedwater flow is isolated on an S/G high-2 level signal; a low pressurizer pressure signal actuates the Safety Injection System. Auxiliary feedwater flow, initiated by the safety injection signal, is assumed, to continue with both pumps discharging into the affected steam generator. It is also assumed, for conservatism in the secondary side analysis, that auxiliary feedwater flows to the steam generators not affected by the malfunctioned valve, in the "unfailed loops." Plant conditions stabilize at the values reached in 600 seconds, at which time auxiliary feedwater flow is terminated. The plant is then either taken to cold shutdown, or returned to the no-load condition at a normal heatup rate with the auxiliary feedwater system under manual control.

For design purposes this transient is assumed to occur 30 times during the life of the plant.

Operating Basis Earthquake

The mechanical stresses resulting from the operating basis earthquake are considered on a component basis. Fatigue analysis, where required by the codes, is performed by the supplier as part of the stress analysis report. The earthquake loads are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transients resulting from fluid pressure and temperature. They are, however, considered in the design analysis. The number of occurrences for fatigue evaluation is assumed to be 20 earthquakes at 20 cycles each (400 cycles total).

Thermal Stratification

The modifications made to the feedwater/auxiliary feedwater system for Unit 1 to support installation of Babcock Wilcox Steam Generators provides a flow path where thermal stratification should occur. The mode in which stratification has the potential to occur is during the injection of cold auxiliary feedwater into a hot main feedwater pipe following a reactor trip or another transient which results in a loss of feedwater to the steam generators. Stratification could also occur after AF flow has been terminated and feedwater tempering flow is used to maintain steam generator secondary side level.

During the injection of cold auxiliary feedwater into the hot stagnant feedwater line, the cold auxiliary feedwater is assumed to flow along the bottom of the 16" feedwater pipe due to buoyancy (cold water is more dense than the stagnant hot water). It is assumed that the height of the cold layer will grow as auxiliary feedwater is injected and the hot water will be purged from the horizontal piping up to the steam generator nozzle in a matter of minutes.

During the injection of hot feedwater tempering into the cold main feedwater pipe, the hot water will flow along the top of the 16" feedwater pipe, up the riser and into the steam generator. During this hot injection phase, it is assumed that the cold stagnant water will not be purged by the incoming hot flow.

The number of assumed occurrences is 120 which is based upon 4 plant trips per year requiring AF injection for a period of 30 years (approximate years remaining on existing plant operating license at the time). The Westinghouse Steam Generators were replaced with Babcock & Wilcox Steam Generators on Unit 1.

RCS Cold Overpressurization

RCS cold overpressurization occurs during startup and shutdown conditions at low temperature, with or without existence of a steam bubble in the pressurizer, and is especially severe when the reactor coolant system is in a water-solid configuration. The event is inadvertent, and usually generated by any one of a variety of malfunctions or operator errors.

All events which have occurred to date may be categorized as belonging to either of the two following transient mechanisms:

a. Events resulting in the addition of mass (mass input transient)

b. Events resulting in the addition of heat (heat input transient)

Both of these scenarios are represented by a composite, "umbrella" design transient. Ten RCS Cold Overpressure events as represented by this umbrella transient are to be assumed over the plant design lifetime.

Emergency Conditions

The following primary system transients are considered emergency conditions:

a. small loss-of-coolant accident,

b. small steam break, and

c. complete loss of flow.

Small Loss-of-Coolant Accident

For design transient purposes the small loss-of-coolant accident is defined as a break equivalent to the severance of a 1-inch inside diameter branch connection. (Breaks smaller than 0.375 inch inside diameter can be handled by the normal makeup system and produce no significant fluid systems transients.) Breaks which are much larger than 1 inch will cause accumulator injection soon after the accident and are regarded as faulted conditions. For design purposes it is assumed that this transient occurs five times during the life of the plant. It should be assumed that the ECCS is actuated immediately after the break occurs and subsequently delivers water to the RCS at a minimum temperature of 32°F.

Small Steam Break

For design transient purposes, a small steam break is defined as a break equivalent in effect to a steam safety valve opening and remaining open. This transient is assumed to occur five times during the life of the plant. The following conservative assumptions are used in defining the transients:

a. The reactor is initially in a hot, zero-power condition.

b. The small steam break results in immediate reactor trip and ECCS actuation.

c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

d. The ECCS operates at a design capacity and repressurizes the RCS within a relatively short time.

Complete Loss of Flow

This accident involves a complete loss of flow from full power resulting from simultaneous loss of power at all reactor coolant pumps. The consequences of this incident are a reactor trip and turbine trip on undervoltage followed by automatic opening of the steam dump system. For design purposes this transient is assumed to occur five times during the plant life.

Faulted Conditions

The following primary system transients are considered faulted conditions. Each of the following accidents should be evaluated for one occurrence:

a. reactor coolant pipe break (large loss-of-coolant accident),

b. large steamline break,

c. feedwater line break,

d. reactor coolant pump locked rotor,

e. control rod ejection,

f. steam generator tube rupture, and

g. safe shutdown earthquake.

Based on leak-before-break analyses performed by Westinghouse, the dynamic effects associated with a large break in the main reactor coolant loop piping need not be considered.

Reactor Coolant Pipe Break (Large Loss-of-Coolant Accident)

Following a reactor coolant pipe break resulting in a large loss of coolant, the primary system pressure decreases causing the primary system temperature to decrease. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal will still be at or near the operating temperature by the end of blowdown. It is conservatively assumed that the ECCS is actuated to introduce water at a minimum temperature of 32°F into the RCS. The safety injection signal will also result in reactor and turbine trips.

Large Steamline Break

This transient is based on the complete severance of the largest steamline. The following conservative assumptions were made:

a. The reactor is initially in a hot, zero-power condition.

b. The steamline break results in immediate reactor trip and ECCS actuation.

c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

d. The ECCS operates at design capacity and repressurizes the RCS within a relatively short time.

The above conditions result in the most severe temperature and pressure variations which the primary system will encounter during a steam break accident.

Feedwater Line Break

This accident involves a double-ended break of the main feedwater piping from full power, resulting in the rapid blowdown of one steam generator and the termination of main feedwater flow to the others. The blowdown is completed in approximately 27 seconds. Conditions were conservatively chosen to give the most severe primary side and secondary side transients. All auxiliary feedwater flow exists at the break. The incident is terminated when the operator manually realigns the auxiliary feedwater system to isolate the break and to deliver auxiliary feedwater to the intact steam generators.

Reactor Coolant Pump Locked Rotor

This accident is based on the instantaneous seizure of a reactor coolant pump with the plant operating at full power. The locked rotor can occur in any loop. Reactor trip occurs almost immediately, as the result of low coolant flow in the affected loop.

Control Rod Ejection

This accident is based on the single most reactive control rod being instantaneously ejected from the core. This reactivity insertion in a particular region of the core causes a severe pressure increase in the RCS such that the pressurizer safety valves will lift. It also causes a more severe temperature transient in the loop associated with the affected region than in the other loops. For conservatism, the analysis is based on the reactivity insertion and does not include the mitigating effects (on the pressure transient) of coolant blowdown through the hole in the vessel head vacated by the ejected rod.

Steam Generator Tube Rupture

This accident postulates the double-ended rupture of a steam generator tube resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to the resulting safety injection signal. In addition, safety injection actuation automatically isolates the feedwater lines by tripping all feedwater pumps and closing the feedwater isolation valves. When this accident occurs, some of the reactor coolant blows down into the affected steam generator causing the shell side level to rise. The primary system pressure is reduced below the secondary safety valve setting. Subsequent recovery procedures call for isolation of the steamline leading from the affected steam generator. This accident will result in a transient which is no more severe than that associated with a reactor trip from full power. It therefore requires no special treatment insofar as fatigue evaluation is concerned, and no specific number of occurrences are postulated.

The preservice and inservice inspection of steam generator tubing was conducted in accordance with Regulatory Guide 1.83, as described in Appendix A until such time that Nuclear Energy Institute (NEI) 97-06 "Steam Generator Program Guidelines," was approved for use by the NRC via License Amendments 150 and 179 for Byron and License Amendments 144 and 172 for Braidwood.

Safe Shutdown Earthquake

The mechanical dynamic or static equivalent loads due to the vibratory motion of the safe shutdown earthquake are considered on a component basis.

Test Conditions

The following primary system transients under test conditions are discussed:

a. primary side hydrostatic test, and

b. secondary side hydrostatic test.

Primary Side Hydrostatic Test

The pressure tests include both shop and field hydrostatic tests which occur as a result of component or system testing. This hydro test is performed at a water temperature which is compatible with reactor vessel material ductility requirements and a test pressure of 3110 psig (1.25 times design pressure). In this test, the reactor coolant system is pressurized to 3110 psig coincident with steam generator secondary side pressure of 0 psig. The reactor coolant system is designed for 10 hydrostatic test cycles. These are performed prior to plant startup. The number of cycles is independent of other operating transients.

Secondary Side Hydrostatic Test

The secondary side of the steam generator is pressurized at 1.25 design pressure with a minimum water temperature of 120°F coincident with the primary side at 0 psig.

For design purposes it is assumed that the steam generator will experience 10 cycles of this test. These tests may be performed either prior to plant startup or subsequently, following shutdown for major repairs, or both. The number of cycles is therefore independent of other operating transients.

3.9.1.2 Computer Programs Used in Analyses

The following computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of Seismic Category I components and equipment. These are described and verified in References 1, 10, 11, 15, 16, and 17.

a. WESTDYN-7 - static and dynamic analysis of redundant piping systems,

b. FIXFM - time-history response of three-dimensional structures,

c. WESDYN2 - piping system stress analysis from time-history displacement data,

d. STHRUST - hydraulic loads on loop components from blowdown information,

e. WESAN - reactor coolant loop equipment support structures analysis and evaluation,

f. WECAN - finite-element structural analysis,

g. DARI-WOSTAS - dynamic transient response analysis of reactor vessel and internals,

h. OPTPIPE - piping system stress analysis for static and dynamic loadings, and

i. GAPPIPE - piping system stress analysis for static and dynamic loadings.

j. ANSYS – finite element analysis program used in dynamic, static and fatigue analyses.

K. WESTEMS – program used in design fatigue analyses and thermal stress ratchet evaluations.

The following computer programs have been used by Sargent & Lundy for dynamic, static and fatigue analyses to determine mechanical loads, stresses, and deformations of Seismic Category I components and equipment. A detailed description and method of verification can be found in Appendix D.

a. ADINA - automatic dynamic incremental nonlinear analysis.

b. ANCHOR - analysis of intermediate anchors on piping systems.

c. AXTRAN - axial temperature transients in welds.

d. CPASYS - comprehensive system of interactive computer programs designed to automate and simplify piping design calculations.

e. DYNAX - finite element program used for static and dynamic analyses of axisymmetric structures.

f. HYTRAN - hydraulic transient forcing function analysis.

g. MOMENT.MOX - Calculates moment ranges between user supplied load set data.

h. MOMENT.TRAN - Calculates thermal transient stress quantities.

i. NOHEAT - finite element nonlinear heat transfer analysis.

j. NONLIN - nonlinear dynamic analysis of 2-D structure.

k. PIPSYS - analyzes piping systems for static and dynamic loadings, and computes the combined stresses.

l. PWRRA - pipe whip restraint reaction analysis.

m. PWUR - break analysis for unrestrained pipes.

n. RELAP 4/MOD 5.

o. RELVAD - analysis of safety/relief valves.

p. SIPDA - simplified piping dynamic analysis of small bore piping systems.

q. SRVA - safety relief valve blowdown analysis.

r. QUICKPIPE/AUTOHANG - piping analysis program (Braidwood only).

s. RELAP5/MOD3.3 – transient two-phase thermo-hydraulic analysis (Byron only).

3.9.1.3 Experimental Stress Analysis

No experimental stress analysis methods are used for Seismic Category I systems or components. However, Westinghouse makes extensive use of measured results from prototype plants and various scale model tests as discussed in Subsection 3.9.2.

3.9.1.4 Considerations for the Evaluation of the Faulted Conditions

3.9.1.4.1 Loading Conditions

The structural stress analyses performed on the reactor coolant system consider the loadings specified as shown in Table 3.9-2. These loads result from thermal expansion, pressure, weight, operating basis earthquake (OBE), safe shutdown earthquake (SSE), design-basis loss-of-coolant accident (including loop hydraulic forces, asymmetric subcompartment pressure forces, and reactor vessel motion), and plant operational thermal and pressure transients.

3.9.1.4.2 Analysis of the Reactor Coolant Loop and Supports

The loads used in the analysis of the reactor coolant loop piping are described in detail below.

Pressure

Pressure loading is identified as either membrane design pressure or general operating pressure, depending upon its application. The membrane design pressure is used in connection with the longitudinal pressure stress and minimum wall thickness calculations in accordance with the ASME Code.

The term operating pressure is used in connection with determination of the system deflections and support forces. The steady-state operating hydraulic forces based on the system initial pressure are applied as general operating pressure loads to the reactor coolant loop model at change in direction or flow area.

Weight

A dead weight analysis is performed to meet Code requirements by applying a 1.0 g load downward on the complete piping system. The piping is assigned a distributed mass or weight as a function of its properties. This method provides a distributed loading to the piping system as a function of the weight of the pipe and contained fluid during normal operating conditions.

Seismic

The forcing functions for the reactor coolant loop seismic piping analyses are derived from dynamic response analyses of the containment building subjected to seismic ground motion. Input is in the form of floor response spectrum curves at various elevations within the containment building.

For the OBE and SSE seismic analyses, 2% and 4% critical damping, respectively, or ASME Code Case N-411 damping are used in the reactor coolant loop/ supports system analysis.

In the response spectrum method of analysis, the total response loading obtained from the seismic analysis consists of two parts; the inertia response loading of the piping system and the differential anchor movements loading. Two sets of seismic moments are required to perform an ASME Code analysis. The first set includes only the moments resulting from inertia effects and these moments are used in the resultant moment (M_i) value for Equations 9 and 13 of NB-3650. The second set includes the moments resulting from seismic anchor motions and are used in Equations 10 and 11 of NB-3650. Differential anchor movement is discussed in Section 3.7.

A separate analysis for differential seismic anchor movement loading is not required if a coupled interior concrete structure/reactor coolant system model is used, since effects of differential movement are accounted for directly.

Loss-of-Coolant Accident

Based on leak-before-break analyses performed by Westinghouse, the dynamic effects associated with a large break in the main reactor coolant loop piping need not be considered.

In the initial evaluation of the reactor coolant loop piping, blowdown loads were developed in the broken and unbroken reactor coolant loops as a result of transient flow and pressure fluctuations following a postulated pipe break in or attached to one of the reactor coolant loops. Structural consideration of dynamic effects of postulated pipe breaks requires postulation of a finite number of break locations.

Broken loop time history dynamic analysis is performed for these postulated break cases. Hydraulic models are used to generate time-dependent hydraulic forcing functions used in the analysis of the reactor coolant loop for each break case. For a further description of the hydraulic forcing functions, refer to Section 3.6.

Transients

The Code requires satisfaction of certain requirements relative to operating transient conditions. Operating transients are tabulated in Subsection 3.9.1.1.

The vertical thermal growth of the reactor pressure vessel nozzle centerlines is considered in the thermal analysis to account for equipment nozzle displacement as an external movement.

The hot modulus of elasticity E, the coefficient of thermal expansion at the metal temperature α, the external movements transmitted to the piping due to vessel growth, and the temperature rise above the ambient temperature ΔT, define the required input data to perform the flexibility analysis for thermal expansion.

To provide the necessary high degree of integrity for the reactor coolant system, the transient conditions selected for fatigue evaluation are based on conservative estimates of the magnitude and anticipated frequency of occurrence of the temperature and pressure transients resulting from various plant operation conditions.

3.9.1.4.3 Reactor Coolant Loop Models and Methods

The analytical methods used in obtaining the solution consists of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the response spectrum method for seismic dynamic analysis, and the time-history integration method for loss-of-coolant accident dynamic analysis.

The integrated reactor coolant loop/supports system model is the basic system model used to compute loadings on components, component supports, and piping. The system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, the stiffness of supports, the stiffnesses of auxiliary line piping which affects the system, and the stiffness of piping restraints. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

Static

The reactor coolant loop/supports system model, constructed for the WESTDYN-7 or BWSPAN computer program, is represented by an ordered set of data which numerically describes the physical system. Figure 3.9-1 shows an isometric line schematic of this mathematical model. The steam generator and reactor coolant pump vertical and lateral support members are described in Subsection 5.4.14.

The spatial geometric description of the reactor coolant loop model is based upon the reactor coolant loop piping layout and equipment drawings. The node point coordinates and incremental lengths of the members are determined from these drawings. Geometrical properties of the piping and elbows along with the modulus of elasticity E, the coefficient of thermal expansion α, the average temperature change from ambient temperature ΔT, and the weight per unit length are specified for each element. The primary equipment supports are included in the model or represented by stiffness matrices which define restraint characteristics of the supports. Due to the symmetry of the static loadings, the reactor pressure vessel centerline may be represented by a fixed boundary of the system mathematical model. The vertical thermal growth of the reactor vessel nozzle centerline is considered in the construction of the model.

The model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained. These loads are incorporated into the overall load vector.

After all the sections have been defined in this matter, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system.

The static solutions for deadweight, thermal, and general pressure loading conditions are obtained by using the WESTDYN-7 or BWSPAN computer program. The derivation of the hydraulic loads for the loss-of-coolant accident analysis of the loop is covered in Subsection 3.6.2.

Seismic

The model used in the static analysis is modified for the dynamic analysis by including the mass characteristics of the piping and equipment. All of the piping loops are included in the system model. The effect of the equipment motion on the reactor coolant loop/supports system is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model.

The steam generator is typically represented by three discrete masses. The lower mass is located at the intersection of the centerlines of the inlet and outlet nozzles of the steam generator. The middle mass is located at the steam generator upper support elevation, and the third mass is located at the top of the steam generator.

The reactor coolant pump is typically represented by a two discrete mass model. The lower mass is located at the intersection of the centerlines of the pump suction and discharge nozzles. The upper mass is located near the center of gravity of the motor.

The reactor vessel and core internals are typically represented by approximately 10 discrete masses. The masses are lumped at various locations along the length of the vessel and along the length of the representation of the core internals.

The component upper and lower lateral supports are inactive during plant heatup, cooldown, and normal plant operating conditions. However, these restraints become active when the plant is at power and under the rapid motions of the reactor coolant loop components that occur from the dynamic loadings and are represented by stiffness matrices and/or individual tension or compression spring members in the dynamic model. The analyses are performed at the full power condition.

The response spectra method employs the lumped mass technique, linear elastic properties, and the principal of modal superposition. The floor response spectra are applied along both horizontal axes and the vertical axis simultaneously.

From the mathematical description of the system, the overall stiffness matrix [K] is developed from the individual element stiffness matrices using the transfer matrix method. After deleting the rows and columns representing rigid restraints, the stiffness matrix is revised to obtain a reduced stiffness matrix $[K_R]$ associated with mass degrees of freedom only. From the mass matrix and the reduced stiffness matrix, the natural frequencies and the normal modes are determined. The modal participation factor matrix is computed and combined with the appropriate response spectra value to give the modal amplitude for each mode. The total modal amplitude is obtained by taking the square root of the sum of the squares of the contributions for each direction.

The modal amplitudes are then converted to displacement in the global coordinate system and applied to the corresponding mass point. From these data the forces, moments, deflections, rotations, support reactions, and piping stresses are calculated for all significant modes.

The total seismic response is computed by combining the contributions of the significant modes using the method described in Section 3.7.

Loss-of-Coolant Accident

It should be noted that the dynamic effects associated with a large break in the main reactor coolant loop piping need not be considered, based on leak-before-break analyses performed by Westinghouse.

The mathematical model used in the static analyses is modified for the loss-of-coolant accident analyses to represent the severance of the reactor coolant loop piping or attached piping at the postulated break location. Modifications include addition of the mass characteristic of the piping and equipment. To obtain the proper dynamic solution for reactor coolant loop piping breaks, two masses, each containing six dynamic degrees of freedom and located on each side of the break, are included in the mathematical model. The natural frequencies and eigenvectors are determined from this dynamic model.

The time-history hydraulic forces at the node points are combined to obtain the forces and moments acting at the corresponding structural node points.

The dynamic structural solution for the full-power loss-of-coolant accident and steamline break may be obtained by using a modified-predictor-corrector-integration or a direct-integration technique and normal mode theory.

When elements of the system can be represented as single acting members (tension or compression members), they may be considered as nonlinear elements, which are represented mathematically by the combination of a gap, a spring, and a viscous damper. The force in this nonlinear element is treated as an externally applied force in the overall normal mode solution. Multiple nonlinear elements can be applied at the same node, if necessary.

The time-history solution is performed in subprogram FIXFM. The input to this subprogram consists of the natural frequencies, normal modes, applied forces, and nonlinear elements. The natural frequencies and normal modes for the modified reactor coolant loop dynamic model are determined with the WESTDYN-7 program. The program BWSPAN may also be used to determine the natural frequencies and normal modes and perform the time history solution. To properly simulate the release of the strain energy in the pipe, the internal forces in the system at the postulated break location due to the initial steady-state hydraulic forces, thermal forces, and weight forces are determined. The release of the strain energy is accounted for by applying the negative of these internal forces as a step function loading. The initial conditions are equal to zero because the solution is only for the transient problem (the dynamic response of the system for the static equilibrium position). The time-history displacement solution of all dynamic degrees of freedom is obtained using subprogram FIXFM and employing 4% critical damping or using BWSPAN and employing model weighted damping.

When using a reactor vessel isolated model, the loss-of-coolant accident displacements of the reactor vessel are applied in time-history form as input to the dynamic analysis of the reactor coolant loop. The loss-of-coolant accident analysis of the reactor vessel includes all the forces acting on the vessel including internals reactions, cavity pressure loads, and loop mechanical loads. The reactor vessel analysis is described in Subsection 3.9.1.4.6.

The resultant asymmetric external pressure loads on the RCP and steam generator resulting from postulated pipe breaks and pressure buildup in the loop compartments are applied to the same integrated RCL/supports system model used to compute loadings on the components, component supports, and RCL piping as previously discussed. The response of the entire system is obtained for the various external pressure loading cases considered. For each pipe break case considered, the equipment support loads and piping stresses resulting from the external pressure loading are added to the support loads and piping stresses calculated using the loop LOCA hydraulic forces and RPV motion.

The break locations considered for subcompartment pressurization are those postulated for the RCL LOCA analysis, as discussed in Section 3.6 and WCAP-8172 (Reference 1 of Section 3.6). For Unit 1, the asymmetric subcompartment pressure loads are provided by Framatome Technologies, Inc. to Sargent & Lundy. For Unit 2, asymmetric subcompartment pressure loads are provided to Westinghouse by Sargent & Lundy. The analysis to determine these loads is discussed in Section 6.2.

The time-history displacement response of the loop is used in computing support loads and in performing stress evaluation of the reactor coolant loop piping.

The support loads [F] are computed by multiplying the support stiffness matrix [K] and the displacement vector [$\boldsymbol{\delta}$] at the support point. The support loads are used in the evaluation of the supports.

The time-history displacements are used to determine the internal forces, deflections, and stresses at each of the piping elements. For this calculation the displacements are treated as imposed deflections on the reactor coolant loop masses. The results of this solution are used in the piping stress evaluation.

Transients

Operating transients in a nuclear power plant cause thermal and/or pressure fluctuations in the reactor coolant fluid. The thermal transients cause time-varying temperature distributions across the pipe wall. These temperature distributions resulting in pipe wall stresses may be further subdivided in accordance with the Code into three parts, a uniform, a linear, and nonlinear portion. The uniform portion results in general expansion loads. The linear portion causes a bending moment across the wall and the nonlinear portion causes a skin stress.

The transients as defined in Subsection 3.9.1.1 are used to define the fluctuations in plant parameters. A one-dimensional finite difference heat conduction program is used to solve the thermal transient problem. The pipe is represented by at least 50 elements through the thickness of the pipe. The convective heat transfer coefficient employed in this program represents the time-varying heat transfer due to free and forced convection. The outer surface is assumed to be adiabatic, while the inner surface boundary experiences the temperature of the coolant fluid. Fluctuations in the temperature of the coolant fluid produce a temperature distribution through the pipe wall thickness which varies with time. An arbitrary temperature distribution across the wall is shown in Figure 3.9-2.

The average through-wall temperature, T_A, is calculated by integrating the temperature distribution across the wall. This integration is performed for all steps so that T_A is determined as a function of time.

$$T_A(t) = \frac{1}{H} \int_0^H T(X,t)\,dX \qquad (3.9\text{-}1)$$

The range of temperature between the largest and smallest value of T_A is used in the flexibility analysis to generate the moment loadings caused by the associated temperature changes.

The thermal moment about the midthickness of the wall caused by the temperature distribution through the wall is equal to:

$$M = E\alpha \int_0^H \left(X - \frac{H}{2}\right) T(X,t)\,dX \qquad (3.9\text{-}2)$$

The equivalent thermal moment produced by the linear thermal gradient as shown in Figure 3.9-2 about the midwall thickness is equal to:

$$M_L = E\alpha \frac{\Delta T_1}{12} H^2 \qquad (3.9\text{-}3)$$

Equating M_L and M, the solution for ΔT_1 as a function of time is:

$$\Delta T_1(t) = \frac{12}{H^2} \int_0^H (X - \frac{H}{2}) T(X,t) dX \tag{3.9-4}$$

The maximum nonlinear thermal gradient, ΔT_2, will occur on the inside surface and can be determined as the difference between the actual metal temperature on this surface and half of the average linear thermal gradient plus the average temperature.

$$\Delta T_2(t) = T(0,t) - T_A(t) - \frac{|\Delta T_1(t)|}{2} \tag{3.9-5}$$

Load Set Generation

A load set is defined as a set of pressure loads, moment loads, and through-wall thermal effects as a given location and time in each transient. The method of load set generation is based on Reference 2. The through-wall thermal effects are functions of time and can be subdivided into four parts:

a. Average temperature (T_A) is the average temperature through-wall of the pipe which contributes to general expansion loads.

b. Radial linear thermal gradient which contributes to the through-wall bending moment (ΔT_1).

c. Radial nonlinear thermal gradient (ΔT_2) which contributes to a peak stress associated with shearing of the surface.

d. Discontinuity temperature (T_A - T_B) represents the difference in average temperature at the cross sections on each side of a discontinuity.

Each transient is described by at least two load sets representing the maximum and minimum stress during each transient. The construction of the load sets is accomplished by combining the following to yield the maximum (minimum) stress state during each transient:

a. ΔT_1,

b. ΔT_2,

c. $\alpha_A T_A - \alpha_B T_B$,

d. moment loads due to T_A, and

e. pressure loads.

This procedure produces at least twice as many load sets as transients for each point.

As a result of the normal mode spectral technique employed in the seismic analysis, the load components cannot be given signed values. Eight load sets are used to represent all possible sign permutations of the seismic moments at each point, thus ensuring that the most conservative combination of seismic loads is used in the stress evaluation.

For all possible load set combinations, the primary-plus-secondary and peak stress intensities, fatigue reduction factors (K_e) and cumulative usage factors, U, are calculated. The WESTDYN-7 program is used to perform this analysis in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB-3650. Alternatively, detailed finite element stress analyses may be used to determine primary-plus-secondary and peak stress intensities, for the load set combinations. Since it is impossible to predict the order of occurrence of the transients over a 40-year life, it is assumed that the transients can occur in any sequence. This is a very conservative assumption.

The combination of load sets yielding the highest alternating stress intensity range is used to calculate the incremental usage factor. The next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9.1.4.4 Primary Component Supports Models and Methods

Primary component supports are discussed in Subsection 3.9.3.4.

3.9.1.4.5 Analysis of Primary Components

Equipment which serves as part of the pressure boundary in the reactor coolant loop include the steam generators, the reactor coolant pumps, the pressurizer, and the reactor vessel. This equipment is Seismic Category I and the pressure boundary meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. This equipment is evaluated for the loading combinations outlined in Table 3.9-2. The equipment is analyzed for (1) the normal loads of deadweight, pressure, and thermal; (2) mechanical transients of OBE, SSE, and pipe breaks, including the effects of asymmetric subcompartment pressurization; and (3) pressure and temperature transients outlined in Subsection 3.9.1.1.

The results of the reactor coolant loop analysis are used to determine the loads acting on the equipment nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an "umbrella" load basis. This is, on the basis of previous plant analyses, a set of loads is determined which should be larger than those seen in any single plant analysis. The umbrella loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. Upon completion of the system analysis, conformation is demonstrated between the actual plant loads and the loads used in the analyses of the components. Any deviations where the actual load is larger than the umbrella load will be handled by individualized analysis.

Seismic analyses are performed individually for the reactor coolant pump, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectrum corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. Seismic analyses for the steam generator and pressurizer are performed using 2% damping for the OBE and 4% damping for the SSE. The analysis of the reactor coolant pump for determination of loads on the motor, main flange, and pump internals is performed using the damping for bolted steel structures, that is 4% for the OBE and 7% for the SSE (2% for OBE and 4% for SSE is used in the system analysis). This damping is applicable to the reactor coolant pump, since the main flange, motor stand, and motor are all bolted assemblies (see Section 5.4). The reactor pressure vessel is qualified by static stress analysis based on loads that have been derived from dynamic analysis.

Reactor coolant pressure boundary components are further qualified to ensure against unstable crack growth under faulted conditions by performing detailed fracture analyses of the critical areas of this boundary. Actuation of the emergency core cooling system produces relatively high thermal stresses in the system. Regions of the pressure boundary which come into contact with emergency core cooling system water are given primary consideration. These regions include the reactor vessel beltline region, the reactor vessel inlet nozzles, and the safety injection nozzles in the piping system.

Two methods of analysis are used to evaluate thermal effects in the regions of interest. The first method is linear elastic fracture mechanics (LEFM). The LEFM approach to the design against failure is basically a stress intensity consideration in which criteria are established for fracture instability in the presence of a crack. Consequently, a basic assumption employed in LEFM is that a crack or crack-like defect exists in the structure. The essence of the approach is to relate the stress field developed in the vicinity of the crack tip to the applied stress on the structure, the material properties, and the size of defect necessary to cause failure.

The elastic stress field at the crack-tip in any cracked body can be described by a single parameter designated as the stress intensity factor, K. The magnitude of the stress intensity factor K is a function of the geometry of the body containing the crack, the size and location of the crack, and the magnitude and distribution of the stress.

The criterion for failure in the presence of a crack is that failure will occur whenever the stress intensity factor exceeds some critical value. For the opening mode of loading (stresses perpendicular to the major plane of the crack) the stress intensity factor is designated as K_I, and the critical stress intensity factor is designated K_{IC}. Commonly called the fracture toughness, K_{IC} is an inherent material property which is a function of temperature and strain rate. Any combination of applied load, structural configuration, crack geometry, and size which yields a stress intensity factor K_{IC} for the material will result in crack instability.

The criterion of the applicability of LEFM is based on plasticity considerations at the postulated crack tip. Strict applicability (as defined by ASTM) of LEFM to large structures where plane strain conditions prevail requires that the plastic zone developed at the tip of the crack does not exceed 2.25% of the crack depth. In the present analysis, the plastic zone at the tip of the postulated crack can reach 20% of the crack depth. However, LEFM has been successfully used quite often to provide conservative brittle fracture prevention evaluations, even in cases where strict applicability of the theory is not permitted due to excessive plasticity. Recently, experimental results from Heavy Section Steel Technology (HSST) Program intermediate pressure vessel tests, have shown that LEFM can be applied conservatively as long as the pressure component of the stress does not exceed the yield strength of the material. The addition of the thermal stresses, calculated elastically, which result in total stresses in excess of the yield strength does not affect the conservatism of the results, provided that these thermal stresses are included in the evaluation of the stress intensity factors. Therefore, for faulted condition analyses, LEFM is considered applicable for the evaluation of the vessel inlet nozzle and beltline region.

In addition, it has been well established that the crack propagation of existing flaws in a structure subjected to cyclic loading can be defined in terms of fracture mechanics parameters. Thus, the principles of LEFM are also applicable to fatigue growth of a postulated flaw at the vessel inlet nozzle and beltline region.

For the safety injection and charging line nozzles, which are fabricated from 304 stainless steel, LEFM is not applicable because of extreme ductility of the material. For these nozzles, the thermal effects are evaluated using the principles of Miner's hypothesis of linear cumulative damage in conjunction with fatigue data from constant stress or strain fatigue tests. The cumulative usage fatigue defined as the sum of the ratios of the number of cycles of each transient, n, to the allowable number of cycles for the stress range associated with the transient, N, must not exceed 1.0.

An example of a faulted condition evaluation carried out according to the procedure discussed previously is given in Reference 3. This report discusses the evaluation procedure in detail as applied to a severe faulted condition (a postulated loss-of-coolant accident).

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of ASME III. These valves are identified in Subsection 3.9.3.2.

Valves in sample lines connected to the RCS are not considered to be Seismic Category I nor ASME Class 1. This is because the nozzles where the line connects to the primary system piping are orificed to a 3/8-inch hole. This hole restricts the flow such that loss through a severance of one of these lines can be made up by normal charging flow.

3.9.1.4.6 Dynamic Analysis of Reactor Pressure Vessel for Postulated Loss-of-Coolant Accident

The structural analysis of the reactor vessel and internals considers simultaneous application of the time-history loads resulting from the reactor coolant loop mechanical loads, internal hydraulic pressure transients, and reactor cavity pressurization. The vessel is restrained by four reactor vessel supports under every other reactor vessel nozzle and the reactor coolant loops with the primary supports of the steam generators and the reactor coolant pumps.

Based on leak-before-break analyses performed by Westinghouse, the dynamic effects associated with pipe breaks of the main reactor coolant loop piping need not be considered. The next largest breaks to consider are the largest auxiliary branch line nozzles penetrating the main reactor coolant loop piping. Guillotine nozzle breaks of the auxiliary branch lines closest to the vessel inlet and outlet nozzles would give the highest reactor vessel support loads and the highest vessel displacements. By considering these breaks, the most severe reactor vessel support loads are determined.

3.9.1.4.6.1 Loading Conditions

Following a postulated pipe break at the locations closest to the vessel nozzles, the reactor vessel is excited by time-history forces. As previously mentioned, these forces are the combined effect of three phenomena: (1) reactor coolant loop mechanical loads, (2) reactor cavity pressurization forces, and (3) reactor internal hydraulic forces.

The reactor coolant loop mechanical forces are derived from the elastic analysis of the loop piping for the postulated break. The reactions on the nozzles of all the unbroken piping legs are applied to the vessel in the reactor pressure vessel blowdown analysis.

Reactor cavity pressurization forces arise for the pipe breaks at the vessel nozzles from the steam and water which is released into the reactor cavity through the annulus around the broken pipe. The reactor cavity is pressurized asymmetrically with higher pressure on the side of the broken pipe resulting in horizontal forces applied to the reactor vessel. Small vertical forces arising from pressure on the bottom of the vessel and the vessel flanges are also applied to the reactor vessel. The cavity pressure analysis is described in Section 6.2.

The internals reaction forces develop from asymmetric pressure distributions inside the reactor vessel. For a vessel inlet nozzle break, the depressurization wave path is through the broken loop inlet nozzle and into the region between the core barrel and reactor vessel. This region is called the downcomer annulus. The initial waves propagate up, down, and around the downcomer annulus and up through the fuel. In the case of a reactor pressure vessel outlet nozzle break the wave passes through the reactor pressure vessel outlet nozzle and directly into the upper internals region, depressurizes the core, and enters the downcomer annulus from the bottom of the vessel. Thus, for an outlet nozzle break, the downcomer annulus is depressurized with much smaller differences in pressure horizontally across the core barrel than for the inlet break. For both the inlet and outlet nozzle breaks, the depressurization waves continue their propagation by reflection and translation through the reactor vessel fluid but the initial depressurization wave has the greatest effect on the loads.

The reactor internals hydraulic pressure transients were calculated including the assumption that the structural motion is coupled with the pressure transients. This phenomena has been referred to as hydroelastic coupling or fluid-structure interaction. The hydraulic analysis considers the fluid-structure interaction of the core barrel by accounting for the deflections of constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized regions. The analysis also contains the option to consider the core barrel as rigid, which is a more conservative modeling approach. The analytical methods used to develop the reactor internals hydraulics are described in WCAP-8708. (6)

3.9.1.4.6.2 <u>Reactor Vessel and Internals Modeling</u>

The reactor vessel model consists of two nonlinear elastic models connected at a common node. One model represents the dynamic vertical characteristics of the vessel and its internals, and the other model represents the translational and rotational characteristics of the structure. These two models are combined in the DARI-WOSTAS code (Reference 1) to represent motion of the reactor vessel and its internals in the plane of the vessel centerline and the broken pipe centerline.

The model for horizontal motion is shown in Figure 3.9-23. Each node has one translational and one rotational degree of freedom in the vertical plane containing the centerline of the nozzle attached to the broken pipe and the centerline of the vessel. A combination of beam elements and concentrated masses are used to represent the components including the vessel, core barrel, neutron panels, fuel assemblies, and upper support columns. Connections between the various components are either pin-pin rigid links, translational impact springs with damping, or rotational springs.

The model for vertical motion is shown in Figure 3.9-24. Each mass node has one translational degree of freedom. The structure is represented by concentrated masses, springs, dampers, gaps, and frictional elements. The model includes the core barrel, lower support columns, bottom nozzles, fuel rods, top nozzles, upper support structure, and reactor vessel.

The horizontal and vertical models are coupled at the elevation of the primary nozzle centerlines. Node 1 of the horizontal model is coupled with node 2 of the vertical model at the reactor vessel nozzle elevation. This coupled node has external restraints characterized by linear horizontal springs which describe the tangential resistance of the supports and by individual nonlinear vertical stiffness elements which provide downward restraint only. The supports as represented in the horizontal and vertical models are not indicative of the complexity of the support system used in the analysis. The individual supports are located at the actual support pad locations and accurately represent the independent nonlinear behavior of each support.

3.9.1.4.6.3 Analytical Methods

The time-history effects of the cavity pressurization loads, internals loads, and loop mechanical loads are combined and applied simultaneously to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis includes the displacements of the reactor vessel and the loads in the reactor vessel supports which are combined with other applicable faulted condition loads and subsequently used to calculate the stresses in the supports. Also, the reactor vessel displacements are applied as a time-history input to the dynamic reactor coolant loop blowdown analysis. The resulting loads and stresses in the piping components and supports include both loop blowdown loads and reactor vessel displacements. Thus, the effect of the vessel displacements upon loop response and the effect of loop blowdown upon vessel displacements are both evaluated. In addition, using the results of the RCL analysis, the actual break opening area is verified to be less than the estimated area used in the analysis and assures that the analysis is conservative.

3.9.1.4.7 Stress Criteria for Class 1 Components

All Class 1 components are designed and analyzed for the design, normal, upset, and emergency conditions to the rules and requirements of the ASME Code, Section III. The design analysis or test methods and associated stress or load allowable limits that will be used in evaluation of faulted conditions are those that are defined in Appendix F of the ASME Code with supplementary options outlined below.

The test load method given in F-1370(d) is an acceptable method of qualifying components in lieu of satisfying the stress/load limits established for the component analysis.

The reactor vessel support pads are qualified using the test option. The reactor pressure vessel support pads are designed to restrain unidirectional horizontal motion in addition to supporting the vessel. The design of the supports allows radial growth of the vessel but restrains the vessel from horizontal displacements since tangential displacement of the vessel is prevented at each vessel nozzle.

To duplicate the loads that act on the pads during faulted conditions, the tests, which utilized a one-eighth linear scale model, were performed by applying a unidirectional static load to the nozzle pad. The load on the nozzle pad was reacted by a support shoe which was mounted to the test fixture.

The above modeling and application of load thus allows the maximum load capacity of the support pads to be accurately established. The test load, L_T, was then determined by multiplying the maximum collapse load by 64 (ratio of prototype area to model area) and including temperature effects in accordance with the rules of the ASME Code, Section III.

The loads on the reactor vessel support pads, as calculated in the system analysis for faulted conditions are limited to the value of 0.80 L_T. The tests performed and the limits established for the test load method ensure that the experimentally obtained value for L_T is accurate and that the support pad design is adequate for its intended function.

Loading combinations and allowable stresses for ASME Class 1 components are given in Tables 3.9-2 and 3.9-3, respectively.

Tables 3.9-2 and 3.9-3 are applicable to both NSSS and BOP Class 1 components.

The methods of load combination for each operating condition are as follows:

Design

Loads are combined by algebraic sum.

Normal, Upset

These loads are used in the fatigue evaluation in accordance with the methods prescribed in the ASME code. Load sets are defined for each transient including the OBE and are combined such that the maximum stress ranges are obtained without regard to the order in which the transients occur. (This is discussed in more detail in Subsection 3.9.1.4.3.)

Emergency

Loads are combined by algebraic sum.

Faulted

LOCA and SSE loads are combined using the square root of the sum of the squares (SRSS) method on a load component basis (i.e., the LOCA F_x is combined with the SSE F_x by SRSS, the LOCA F_y is combined with the SSE F_y by SRSS, and likewise for F_z, M_x, M_y, and M_z). The sustained loads, such as weight effects, are combined with the SRSS results by algebraic sum.

3.9.1.4.8 Analytical Methods for RCS Class 1 Branch Lines

The analytical methods used to obtain the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the response spectrum method for seismic dynamic analysis, and static or dynamic structural analysis for the effect of a reactor coolant loop pipe break. It should be noted that the dynamic effects associated with a large break in the main reactor coolant loop piping need not be considered, based on leak-before-break analyses performed by Westinghouse.

The integrated Class 1 piping and supports system model is the basic system model used to compute loadings on components, component and piping supports, and piping. The system models include the stiffness and mass characteristics of the Class 1 piping components, the reactor coolant loop, and the stiffness of supports which affect the system response. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

Static

The Class 1 piping system models are constructed for the GAPPIPE, PIPSYS, OPTPIPE or WESTDYN computer programs, which numerically describe the physical system. A network model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the characteristic stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained.

After all the sections have been defined in this manner, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points are determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system. The support loads are also computed by multiplying the stiffness matrix by the displacement vector at the support point.

Seismic

The models used in the static analyses are modified for use in the dynamic analyses by including the mass characteristics of the piping and equipment.

The lumping of the distributed mass of the piping systems is accomplished by locating the total mass at points in the system which will appropriately represent the response of the distributed system. Effects of the primary equipment motion, that is, reactor vessel, steam generator, reactor coolant pump, and pressurizer, on the Class 1 piping system are obtained by modeling the mass and the stiffness characteristics of the primary equipment and loop piping in the overall system model.

The supports are represented by stiffness matrices in the system model for the dynamic analysis. Shock suppressors which resist rapid motions are also included in the analysis. The solution for the seismic disturbance employs the response spectra method. This method employs the lumped mass technique, linear elastic properties, and the principle of model superposition.

The total response obtained from the seismic analysis consists of two parts: the inertia response of the piping system and the response form differential anchor motions. The stresses resulting from the anchor motions are considered to be secondary and, therefore, are included in the fatigue evaluation.

Loss-of-Coolant Accident

The mathematical models used in the seismic analyses of the Class 1 lines are also used for RCL pipe break effect analysis. To obtain the dynamic solution for lines six inches and larger and certain small-bore lines required for ECCS considerations, the time-history deflections from the analysis of the reactor coolant loop are applied at branch nozzle connections. For other small bore lines which must maintain structural integrity, the motion of the RCL is applied statically.

Fatigue

A thermal transient heat transfer analysis is performed for each different piping component on all the Class 1 branch lines. The normal, upset, and test condition transients identified in Subsection 3.9.1.1 are considered in the fatigue evaluation.

The thermal quantities ΔT_1, ΔT_2 and ($\alpha_a T_a$, $-\alpha_b T_b$) are calculated on a time-history basis, using a one-dimensional finite difference heat transfer computer program. Stresses due to these quantities were calculated for each time increment using the methods of NB-3650 of ASME III.

For each thermal transient, two load sets are defined, representing the maximum and minimum stress states for that transient.

As a result of the normal mode spectral technique employed in the seismic analysis, the load components cannot be given signed values. Eight load sets are used to represent all possible sign permutations of the seismic moments at each point, thus insuring the most conservative combinations of seismic loads are used in the stress evaluation.

The GAPPIPE, PIPSYS, OPTPIPE or WESTDYN computer programs are used to calculate the primary-plus-secondary and peak stress intensity ranges, fatigue reduction factors and cumulative usage factors for all possible load set combinations. It is conservatively assumed that the transients can occur in any sequence, thus resulting in the most conservative and restrictive combinations of transients.

The combination of load sets yielding the highest alternating stress intensity range is determined and the incremental usage factor calculated. Likewise, the next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9.1.4.9 Evaluation of Control Rod Drive Mechanisms and Supports

The control rod drive mechanisms (CRDMs) and CRDM support structure are evaluated for the loading combinations outlined in Table 3.9-3.

A detailed finite element model of the CRDMs and CRDM supports is constructed using the WECAN computer program with beam, pipe, and spring elements. For the LOCA analysis, nonlinearities in the structure are represented. The time-history motion of the reactor vessel head, obtained from the RPV analysis described in 3.9.1.4.6 is input to the dynamic model. Maximum forces and moments in the CRDMs and support structure are then determined. For the seismic analysis, the structural model is linearized and the floor response spectra corresponding to the CRDM tie rod elevation is applied to determine the maximum forces and moments in the structure.

The bending moments calculated for the CRDMs for the various loading conditions are compared with maximum allowable moments determined from a detailed finite element stress evaluation of the CRDMs. Adequacy of the CRDM support structure is verified by comparing the calculated stresses to the criteria given in ASME III, Subsection NF.

3.9.2 Dynamic Testing and Analysis

3.9.2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

During preoperational testing, normal operating modes were observed for vibration. Engineers familiar with the subject piping visually inspected the lines to determine the acceptability of the steady-state vibrations. Normal vibration was noted. If piping system vibration was judged excessive, one of the following corrective actions was taken:

a. The piping was monitored by instrumentation at locations which appeared to be excessive to demonstrate that the measured pipe deflections when converted to stress did not exceed the following allowable stress amplitude, S_a, used for steady-state piping vibration:

   S_a = 7,690 psi for carbon steel with UTS < 80 ksi

   S_a = 12,000 psi for stainless steel.

   These stress amplitudes represent values, based on 80% of the alternating stress intensity of 10^6 cycles for carbon steels and 60% of the alternating stress intensity at 10^6 cycles for stainless steels, divided by a factor of safety of 1.3. The values of alternating stress intensity are taken from Figure I-9.1 Appendix I of ASME Code Section III.

b. The cause of the excessive vibration was reduced to allowable levels.

c. The support system was modified to reduce the vibration to acceptable limits.

The systems to be monitored shall be selected from the essential systems listed in Table 3.9-19 based on the system design and safety function. Essential systems are those systems that may be required to bring the plant to a safe shutdown from normal or accident conditions.

The systems were heated to test temperatures and checked at various temperatures to verify proper expansion as described in Subsection 3.9.2.8. In addition, systems were operated and performance of the pumps, valves, and auxiliary equipment checked. Testing included the following transient testing:

A. Reactor Coolant Loops

The reactor coolant pumps are tested up to full flow and full operating pressure and temperature prior to installation. These tests include pump starts, pump trips, and normal transients.

It should be noted that the layout, size, etc., of the reactor coolant pump and surge line piping, used in the Byron/Braidwood units, are very similar to that employed in Westinghouse plants now in operation. The operating experience that has been obtained from these plants indicates that the reactor coolant loop and surge line pipe are adequately designed and supported to minimize vibration. In addition, vibration levels of the reactor coolant pump, which is the only mechanical component that could cause vibration of the reactor coolant loops and surge line piping, are measured and held to the limits given in Subsection 5.4.1. Thus, there should be no excessive vibration of the reactor coolant loop and surge line piping.

B. Turbine Stop Valve Closure

The effect of turbine stop valve closure has been evaluated analytically by means of a dynamic analysis of piping systems. Forcing functions are applied at point of fluid momentum change, such as elbows. The forcing functions are described by fluid momentum equations and the shock wave velocity.

C. Relief Valve Operation

The effects of the relief valve operation on system piping are evaluated analytically by means of dynamic analysis of the valve station and discharge piping. Forcing functions are applied at points of momentum change in the system. Forcing functions are described by fluid momentum equations and a shock wave velocity.

The equivalent static analysis of open discharge systems is performed in accordance with the provisions of Code Case 1569 and a dynamic load factor (DLF) which is based on safety and relief valve opening time and system dynamic characteristics as determined per the rules of ANSI B31.1-1977, Appendix II.

During performance testing, piping components and other major equipment will be observed for indications of excessive vibrations, overheating, and noise.

Recent operating reactor experience indicates that vibratory loads associated with the operation of positive displacement pumps have contributed to high cycle fatigue pipe failure. Such failures are known to occur on both the suction and discharge sides of positive displacement pumps in PWR charging systems.

To absorb these vibratory loads pulsation dampeners are installed directly on the suction and discharge of the positive displacement charging pumps in the chemical and volume control system. The discharge dampeners limit downstream pressure fluctuations to a maximum of ± 2% of discharge pressure. Past testing at approximately 2260 psig has shown 70 psig and 1.6 psig peak-to-peak fluctuations in the discharge and suction piping respectively.

The dampeners, manufactured by Greer Hydraulics, are of the bladder type and are sized at 5 gallons each.

3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

3.9.2.2.1 NSSS

The operability of Seismic Category I mechanical equipment must be demonstrated if the equipment is determined to be active, i.e., if mechanical operation is relied on to perform a safety function. The operability of active Class 2 and 3 pumps, active Class 1, 2, or 3 valves, and their respective drives, operators and vital auxiliary equipment is shown by satisfying the criteria given in Subsection 3.9.3.2. Other active mechanical equipment is shown operable by testing, analysis, or a combination of testing and analysis. The operability programs implemented on this other active equipment are similar to the program described in Subsection 3.9.3.2 for pumps and valves. Testing procedures similar to the procedures outlined in Section 3.10 for electrical equipment are used to demonstrate operability if the component is mechanically or structurally complex such that its response cannot be adequately predicted analytically. Analysis may be used if the equipment is amenable to modeling and dynamic analysis.

Inactive Seismic Category I equipment is shown to have structural integrity during all plant conditions by one of the following manner: 1) by analysis satisfying the stress criteria applicable to the particular piece of equipment, or 2) by test showing that the equipment retains its structural integrity under the simulated test environment.

A list of Seismic Category I equipment and the method of qualification used is provided in Table 3.2-1.

3.9.2.2.2 Balance of Plant

The following dynamic testing procedures are used for Seismic Category I mechanical equipment and equipment supports.

3.9.2.2.2.1 Seismic Testing and Analysis

The ability of equipment to perform its Seismic Category I functions during and after an earthquake is demonstrated by tests and/or analysis. The selection of testing and/or analysis for a particular piece of equipment is based on practical considerations. When practical, the Seismic Category I operations are activated and tested during the vibratory testing. When this is not practical, these operations are simulated by a combination of tests and analysis.

3.9.2.2.2.2 Seismic Analysis

Equipment which is large, simple, and/or consumes large amounts of power is usually qualified by an analysis to show that the loads, stresses, and deflections are less than the values which give assurance of proper operation. Analysis is also used to show that there are no natural frequencies below the frequency range of a test facility.

3.9.2.2.2.3 Basis for Test Input Motion

When equipment is qualified by test, the response spectrum or the time history at the point of attachment to the supporting structure is the basis for determining the test input motion.

3.9.2.2.2.4 Random Vibration Input

When random vibration input is used, the actual input motion envelopes the appropriate floor input motion at the individual modes. However, single frequency input, such as sine beats, can be used provided one of the following conditions are met:

a. The characteristics of the required input motion are dominated by one frequency.

b. The anticipated response of the equipment is adequately represented by one mode.

c. The input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra will envelope the corresponding response spectra of the individual modes.

3.9.2.2.2.5 Input Motion

The input motion is applied to vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa, then the input motion is applied to one direction at a time. In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions is such that a purely rectilinear resultant input is avoided.

3.9.2.2.2.6 Fixture Design

The fixture design simulates the actual service mounting and causes no dynamic coupling to the equipment.

3.9.2.2.2.7 Equipment Testing

Equipment testing is based on prototype basis.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The vibration characteristics and behavior due to flow-induced excitation are very complex and not readily ascertained by analytical means alone. Reactor components are excited by the flowing coolant which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area should provide the forcing functions to be used in the dynamic analysis of the structures. In view of the complexity of the geometries and the random character of the pressure oscillations, a closed form solution of the vibratory problem by integration of the differential equation of motion is not always practical and realistic. The determination of the forcing functions as a direct correlation of pressure oscillations cannot be practically performed independently of the dynamic characteristics of the structure. The main objective is to establish the characteristics of the forcing functions that essentially determine the response of the structures. By studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing function can be deduced. These studies indicate that the most important forcing functions are flow turbulence and pump related excitation. The relevance of such excitations depends on many factors such as type and location of component and flow conditions. The effects of these forcing functions have been studied from test runs on models, prototype plants and in component tests (References 4 and 5).

The Indian Point Unit 2 plant has been established as the prototype for a four-loop plant internals verification program and was fully instrumented and tested during hot functional testing. In addition, the Trojan plant instrumentation program provides prototype data applicable to Byron/Braidwood (References 4 and 5).

The Byron/Braidwood plants are similar to Indian Point Unit 2; the only significant differences are the modifications resulting from the use of 17 x 17 fuel and the replacement of the annular thermal shield with neutron shielding pads. These differences are addressed below.

a. 17 x 17 Fuel

   The only structural changes in the internals resulting from the design change from the 15 x 15 to the 17 x 17 fuel assembly are the guide tube and control rod drive line. The new 17 x 17 guide tubes are stronger and more rigid, hence they are less susceptible to flow induced vibration. The fuel assembly itself is relatively unchanged in mass and spring rate, thus no significant deviation is expected from the vibration of plants having 15 x 15 fuel assemblies.

b. Neutron Shielding Pads Lower Internals

   The primary cause of core barrel excitation is flow turbulence generated in the downcomer annulus. The vibration levels due to core barrel excitation for Trojan and Byron/Braidwood, both having neutron shielding pads, are expected to be similar. Since Byron/Braidwood has slightly greater velocities than Trojan, vibration levels due to the core barrel excitation are expected to be slightly greater than those for Trojan (proportional to flow velocity raised to a small power) (Reference 4). However, scale model test results (Reference 4) and results from Trojan (Reference 5) show that core barrel vibration of plants with neutron shielding pads is significantly less than that of plants with thermal shields. This information and the fact that low core barrel stresses and large safety margins were obtained from data recorded at Indian Point Unit 2 (thermal shield configuration) lead to the conclusion that stresses approximately equal to those of Indian Point Unit 2 will result on the Byron/Braidwood internals with the attendant large safety margins.

The original test and analysis of the four-loop configuration is augmented by References 4 and 5 to cover the effects of successive hardware modifications.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Because the Byron/Braidwood reactor internals design configuration is well characterized, as was discussed in Subsection 3.9.2.3, it is not considered necessary to conduct instrumented tests of the Byron/Braidwood plant hardware. The recommendations of a comprehensive vibration assessment program are satisfied by conducting the confirmatory pre- and post-hot functional examinations for internals integrity. This examination included more than 30 features (illustrated in Figure 3.9-3), with special emphasis on the following areas:

a. All major load-bearing elements of the reactor internals relied upon to retain the core structure in place.

b. The lateral, vertical, and torsional restraints provided within the vessel.

c. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals.

d. Those other locations on the reactor internal components which are similar to those which were examined on the prototype designs.

e. The inside of the vessel is inspected before and after the hot functional test, with all the internals removed, to verify that no loose parts or foreign material are in evidence.

A particularly close inspection was made on the following items or areas using a 5X or 10X magnifying glass or an appropriate inspection:

a. Lower Internals

   1. Upper barrel to flange girth weld.

   2. Upper barrel to lower barrel girth weld.

   3. Upper core plate aligning pin. Examine bearing surfaces for shadow marks, burnishing, buffing, or scoring. Inspect welds for integrity.

4. Irradiation specimen guide screw locking devices and dowel pins. Check for lockweld integrity.

5. Baffle assembly locking devices. Check for lockweld integrity.

6. Lower barrel to core support girth weld.

7. Neutron shielding pads screw locking devices and dowel pin lock welds. Examine the interface surfaces for evidence of tightness. Check for lockweld integrity.

8. Radial support key welds.

9. Insert screw locking devices. Examine soundness of lockwelds.

10. Core support columns and instrumentation guide tubes. Check the joints for tightness and soundness of the locking devices.

11. Secondary core support assembly screw locking devices for lockweld integrity.

12. Lower radial support keys and inserts. Examine bearing surfaces for shadow marks, burnishing, buffing, or scoring. Check the integrity of the lockwelds. These members supply the radial and torsional constraint of the internals at the bottom relative to the reactor vessel while permitting axial and radial growth between the two. One would expect to see, on the bearing surfaces of the key and keyway, burnishing, buffing, or shadow marks which would indicate pressure loading and relative motion between these parts. Minor scoring of engaging surfaces is also possible and acceptable.

13. Gaps and baffle joints. Check gaps between baffle to baffle joints.

b. Upper Internals

1. Thermocouple conduits, clamps, and couplings.

2. Guide tube, support column, orifice plate, cover plate, and thermocouple assembly locking devices.

3. Support column and thermocouple conduit assembly clamp welds.

4. Upper core plate alignment inserts. Examine bearing surfaces for shadow marks, burnishing, buffing, or scoring. Check the locking devices for integrity of lockwelds.

5. Thermocouple conduit fitting locking tab and clamp welds.

6. Rigidness and condition of accessible thermocouple tips.

7. Guide tube enclosure and card welds.

Acceptance standards are the same as required in the shop by the original design drawings and specifications.

During the hot functional test, the internals were subjected to a total operating time at greater than normal full-flow conditions (four pumps operating) of at least 240 hours. This provides a cyclic loading of approximately 10^7 cycles on the main structural elements of the internals. In addition there was some operating time with only one, two, and three pumps operating.

Pre- and post-hot functional inspection results confirmed that the internals were well behaved. No signs of abnormal wear and harmful vibrations are detected and no apparent structural changes take place. The four-loop core support structures were considered to be structurally adequate and sound for operation.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

The following events are considered in the faulted conditions category:

a. loss-of-coolant accident (LOCA), both cold-leg and hot-leg breaks are considered; and

b. safe shutdown earthquake (SSE).

It should be noted that the dynamic effects associated with a large break in the main reactor coolant loop piping need not be considered, based on leak-before-break anlayses performed by Westinghouse.

Maximum stresses for SSE and LOCA are obtained and combined.

Maximum stress intensities are compared to allowable stresses for each of the above conditions. Elastic analysis is used to obtain the response of the structure, and the stress analysis on each component is performed according to ASME Code approved techniques.

For faulted conditions, stresses are above yield in a few locations. For these cases only, some inelastic stress limits are applied. The criteria for acceptability in regard to mechanical integrity analyses are that adequate core cooling and core shutdown must be assured. This implies that the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established for the internals are concerned with the deflections and stability of the parts in addition to stress criteria to assure integrity of the components.

For the critical internal structures, maximum allowable deflections, based on functional performance criteria, are listed in Table 3.9-4. The basic operational or functional criterion to be met for the reactor internals is that the plant shall be shut down and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits following the design-basis accident.

Reactor Internals Analysis

Analysis of the reactor internals for blowdown loads resulting from a loss-of-coolant accident is based on the time-history response of the internals to simultaneously applied blowdown forcing functions. The forcing functions are defined at points in the system where changes in cross section or direction of flow occur such that differential loads are generated during the blowdown transient. The dynamic mechanical analysis employs the displacement method, lumped parameters, and stiffness matrix formulations; it assumes that all components behave in a linearly elastic manner.

In addition, because of the complexity of the system and the components, it is necessary to use finite element stress analysis codes to provide more detailed information at various points.

A blowdown digital computer program (Reference 6) which was developed for the purpose of calculating local fluid pressure, flow, and density transients that occur in pressurized water reactor coolant systems during a loss-of-coolant accident is applied to the subcooled, transition, and saturated two-phase blowdown regimes. This blowdown code is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy, are solved numerically using a fixed mesh in both space and time.

Although spatially one-dimensional conservation laws are employed, the code can be applied to describe three-dimensional system geometries by use of the equivalent piping networks. Such piping networks may contain any number of pipes of channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps, and free surfaces (such as in the pressurizer). System losses such as friction, contraction, and expansion, as well as some effects of the fluid-structure (hydroelastic) interaction, are considered.

The blowdown code evaluates the pressure and velocity transients for a maximum of 2400 locations throughout the system. Each reactor component for which calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

a. the pressure differential across the element;

b. flow stagnation on, and unrecovered orifice losses across the element; and

c. friction losses along the element.

Input to the code, in addition to the blowdown pressure and velocity transients, includes the effective area of each element on which the force acts due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

The mechanical analysis has been performed using the following assumptions:

a. The analysis considers some effect of the fluid-structure (hydroelastic) interaction.

b. The reactor internals are represented by a multimass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses while elastic elements are represented by springs.

c. The model described is considered to have a sufficient number of degrees of freedom to represent the most important modes of vibration in the vertical direction. This model is conservative in the sense that further mass-spring resolution of the system would lead to further attenuation of the shock effects obtained with the present model.

The pressure waves generated within the reactor are highly dependent on the location and nature of the postulated pipe failure. In general, the more rapid the severance of the pipe, the more severe the imposed loadings on the components. A 1-millisecond severance time is taken as the limiting case.

In the case of the hot leg break, the vertical hydraulic forces produce an initial upward lift of the core. A rarefaction wave propagates through the reactor hot leg nozzle into the interior of the upper core barrel. Since the wave has not reached the flow annulus on the outside of the barrel, the upper barrel is subjected to an impulsive compressive wave. Thus, dynamic instability (buckling) or large deflections of the upper core barrel, or both, are possible responses of the barrel during hot leg break, resulting in transverse loading on the upper core components as the fluid exits the hot leg nozzle.

In the case of the cold leg break, a rarefaction wave propagates along a reactor inlet pipe, arriving first at the core barrel at the inlet nozzle of the broken loop. The upper barrel is then subjected to a nonaxisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel. After the cold leg break, the initial steady-state hydraulic lift forces (upward) decrease rapidly (within a few milliseconds) and then increase in the downward direction. These cause the reactor core and lower support structure to move initially downward.

If a simultaneous seismic event with the intensity of the SSE is postulated with the loss-of-coolant accident, the imposed loading on the internals component may be additive in certain cases and therefore the combined loading must be considered. In general, however, the loading imposed by the earthquake is small compared to the blowdown loading.

A summary of the mechanical analysis is presented in the following paragraphs. Reference 7 provides further details of the method used in the reactor internals blowdown analysis.

<u>Vertical Excitation Model for Blowdown</u>

For the vertical excitation, the reactor internals are represented by a multimass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components. Also incorporated in the multimass system is a representation of the motion of the fuel elements relative to the fuel assembly grids. The fuel elements in the fuel assemblies are kept in position by friction forces originating from the preloaded fuel assembly grid fingers. Coulomb type friction is assumed in the event that sliding between the rods and the grid fingers occurs. In order to obtain an accurate simulation of the reactor internals response, the effects of internal damping, clearances between various internals, snubbing action caused by solid impact, Coulomb friction induced by fuel rod motion relative to the grids, and preloads in hold down springs have been incorporated in the analytical model. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses, while elastic elements are represented by springs.

The appropriate dynamic differential equations for the multimass model describing the aforementioned phenomena are formulated and the results obtained using a digital computer program (Reference 8) which computes the response of the multimass model when excited by a set of time dependent forcing functions. The appropriate forcing functions are applied simultaneously and independently to each of the masses in the system. The results from the program give the forces, displacements, and deflections as functions of time for all the reactor internals components (lumped masses). Reactor internals response to both hot and cold leg pipe breaks were analyzed.

Transverse Excitation Model for Blowdown

Various reactor internal components are subjected to transverse excitation during blowdown and are analyzed to determine their response to this excitation. The core barrel, guide tubes, and upper support columns analyses are discussed in the following paragraphs.

Core Barrel

For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse.

The barrel is then analyzed for dynamic buckling using the following conservative assumptions:

a. The effect of the fluid environment is neglected.

b. The shell is treated as simply supported.

During cold leg blowdown, the upper barrel is subjected to a nonaxisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel.

The analysis of transverse barrel response to cold-leg blowdown is performed as follows:

a. The core barrel is analyzed as a shell with two variable sections to model the support flange and core barrel.

b. The barrel with the core and thermal shielding pads is analyzed as a beam elastically supported at the top and at the lower radial support, and the dynamic response is obtained.

Guide Tubes

The dynamic loads on rod cluster control guide tubes are more severe for a loss-of-coolant accident caused by a hot leg break than for an accident by a cold leg break, since the cold leg break leads to much smaller changes in the transverse coolant flow over the rod cluster control guide guides. The guide tubes in closest proximity to the outlet nozzle break are the most severely loaded. The transverse guide tube forces during a blowdown decrease with increasing distance from the nozzle break location.

A detailed structural analysis of the rod cluster control guide tubes was performed to establish the equivalent cross section properties and elastic end support conditions. An analytical model is verified by subjecting the control rod cluster guide tube to a concentrated force applied at the midpoint of the lower guide tube. In addition, the analytical model has been previously verified through numerous dynamic and static tests performed on the 17 x 17 guide tube design.

The response of the guide tubes to the transient loading from blowdown is found by representing the guide tube as an equivalent single degree of freedom system and assuming the slope of the time dependent load to be a step function with constant slope front end.

Upper Support Columns

Upper support columns located close to the broken nozzle during hot-leg break will be subjected to transverse loads due to cross flow. The loads applied to the columns are computed with a method similar to the one used for the guide tubes, i.e., by taking into consideration the increase in flow across the column during the accident. The columns are studied as beams with variable section and the resulting stresses are obtained using the reduced section modulus and appropriate stress risers for the various sections.

Results of Reactor Internals Analysis

Maximum stresses due to the safe shutdown earthquake (vertical and horizontal components) and loss-of-coolant accident (hot-leg or cold-leg break) were obtained and combined. All core support structure components were found to be within acceptable stress and deflection limits for a loss-of-coolant accident occurring simultaneously with the safe shutdown earthquake; the stresses and deflections which could result following a faulted condition are less than those which would adversely affect the integrity of the core support structures. For the transverse excitation, it is shown that the barrel does not buckle during a hot-leg break and that it meets the allowable stress limits during all specified transients.

Also, the natural and applied frequencies are such that resonance problems will not occur.

The results obtained from linear analyses indicate that the relative displacement between the components will close the gaps and consequently the structures will impinge on each other. Linear analysis will not provide information about the impact forces generated when components impinge on each other; however, in some instances, linear approximations can, and are applied prior to and after gap closure. The effects of the gaps that could exist between vessel and barrel, between fuel assemblies, between fuel assemblies and baffle plates, and between the control rods and their guide paths were considered in the analysis using both linear approximations and nonlinear techniques. Both static and dynamic stress intensities are within acceptable limits.

Even through control rod insertion is not required for plant shutdown, this analysis shows that most of the guide tubes will deform within the limits established experimentally to assure control rod insertion. For the guide tubes deflected above the no loss of function limit, it must be assumed that the rods will not drop. However, the core will still shut down due to the negative reactivity insertion in the form of core voiding. Shutdown will be aided by the great majority of rods that do drop. Seismic deflections of the guide tubes are generally negligible by comparison with the no loss of function limit.

3.9.2.6 Correlations of Reactor Internals Vibration Tests With the Analytical Results

As stated in Subsection 3.9.2.3, it is not considered necessary to conduct instrumented tests of the Byron/Braidwood reactor vessel internals. Adequacy of these internals is verified by use of the Indian Point and Trojan results.

3.9.2.7 Loose Parts Monitoring System

A loose parts monitoring system is provided. This system uses an array of accelerometers externally mounted to the major components of the reactor system, signal conditioning equipment, recording and alarm equipment, and diagnostic equipment and software. The purpose of this system is to collect information which may be of use in the detection, location, and identification of loose parts within the reactor coolant system (including the reactor core) and associated systems.

The system utilizes high-temperature accelerometers with high-temperature cable assemblies and radiation hardened preamplifiers located on reactor coolant system equipment. Active accelerometers supply monitoring channels through a selector panel.

Sensors can be selected to provide a concentrated detector and analysis capability in selected areas of the reactor coolant system. Accelerometer locations are as follows: two on the reactor vessel head studs; two on the reactor vessel bottom; two in the area of the reactor coolant inlet nozzle to each steam generator; one on each reactor coolant pump cooling line; one in the area of the upper tap for the narrow-range level indication system for each steam generator; and one in the area of the inlet feedwater pipe around the lower tap of the narrow-range level indication system for each steam generator.

3.9.2.8 Preoperational Hot Functional Test

The snubbers on pipelines whose operating temperature exceeds 250°F, and which are predicted to experience movements greater than or equal to 1/2 inch, were included in the test program.

During hot functional testing, the following items were verified for essential systems whose operating temperature exceeds 250°F. Inspectors performing the following examinations were qualified to the requirements of Section XI:

a. To verify by snubber movement that components and piping can expand without restriction of movement on system heatup and return to their approximate baseline positions on system cooldown.

b. For systems whose maximum normal operating temperature was not attained during testing, the expected amount of movement was calculated and evaluated to assure that snubbers would remain within their stroke capabilities.

If vibration levels are noted beyond the acceptance level, an analysis of the vibration effects on the piping was performed, which have necessitated the addition of corrective restraints to limit stress and fatigue levels to within design limits or the source of the vibration was reduced to the extent that the acceptance levels are met. If no travel is observed for a snubber, its ability to move was verified.

Components were checked for correct installation according to specifications and drawings. Piping supports were checked for correct location and settings based on calculations. The cold locations for the reactor system components, such as steam generators and reactor coolant pumps, were recorded.

3.9.3 ASME Code Class 1, 2, and 3 Piping, Components and Component Support Structures

General

During the initial plant construction, ASME Code Class 1, 2, and 3 Piping and Components were designed and constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code and Code Case(s). ASME Code Class 1, 2, and 3 Piping and Components are repaired and replaced in accordance with ASME XI design and construction requirements. For component support structure for Piping, the jurisdictional boundaries have been established in accordance with the ASME Section III, 1974 Edition through Summer 1975 Addenda.

In accordance with ASME, a specification has been provided for piping supports which defines the jurisdictional boundary for the NF portion of the piping support. The auxiliary steel for the piping support is considered an extension of the building structure and has been designed to the AISC Code. The piping support auxiliary steel is identified in the design specification and the support drawings as not being within the jurisdiction of Subsection NF of the ASME Code.

For equipment component supports, such as those for pumps and vessels, the supports have generally been furnished by the manufacturer along with the equipment. The supports have been designed and classified by the vendors and meet either ASME Subsection NF, the rules for the class of the component being furnished, or AISC, as appropriate.

Unit 2 reactor coolant loop piping and associated components and component supports were designed and analyzed by Westinghouse and Sargent & Lundy. Unit 1 reactor coolant loop piping and associated components and component supports were designed and analyzed by FTI.

Loading conditions, stress limits, design transients and methods of analysis for ASME Code Class 1 reactor coolant loop piping and associated components and component supports are discussed in Subsection 3.9.1.

Loading conditions, limits and deformation criteria, and methods of analysis and testing for reactor internals (including core supports structures) are discussed in Subsections 3.9.2 and 3.9.5. Loading combinations for reactor internals are given in Table 3.9-21.

3.9.3.1 Loading Combinations and Stress Limits

ASME 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 and Code Case N-275. Hydrostatic testing is performed per Section III and Code Case N-240. The allowable buckling loads are as per F-1370 of Appendix F of 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.

Load combinations are listed in Tables 3.9-2 and 3.9-5. The stress limits for component supports are provided in Table 3.9-20.

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

Significant discontinuities were considered such as nozzles, flanges, etc. In addition to the design calculation required by the ASME 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 Component and Component Supports Purchased In Accordance with NSSS Specification

ASME Code Class 1

See Subsection 3.9.1 for a discussion of ASME Code Class 1 components.

ASME Code Class 2 and 3

The design loading combinations for ASME Code Class 2 and 3 components and supports are given in Table 3.9-5. The design loading combinations are categorized with respect to normal, upset, emergency, and faulted conditions. Design of primary equipment supports is discussed in Subsection 3.9.3.4.

The design stress limits established for Class 2 and 3 components are sufficiently low to assure that violation of the pressure-retaining boundary will not occur. Stress limits for each of the loading combinations are presented in Tables 3.9-6. 3.9-7, 3.9-8 and 3.9-9 for tanks, inactive* pumps, active pumps, and valves, respectively. Active** pumps and valves are discussed in Subsection 3.9.3.2.

The criteria for Class 2 and 3 component supports are as follows:

a. Supports for Vessels Procured After July 1, 1974:

   Class 2 and 3 vessel supports are designed and analyzed to the rules and requirements of ASME III, Subsection NF.

   For linear supports designed by analysis, the increased design limit for stress identified in NF-3231.1 (a) shall be limited to the smaller of 2.0 S_y or S_u, unless otherwise justified by shakedown analysis. The methods for analysis and associated allowable limits that are used in the

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* Inactive components are those whose operability is not relied upon to perform a safety function during the transients or events considered in the respective operating condition category.

** Active components are those whose operability is relied upon to perform a safety function (as well as to accomplish and maintain a safe reactor shutdown) during and following the transients and events considered in the respective operating condition categories.

evaluation of linear supports from faulted conditions are those defined in ASME III Appendix F.

Plate and shell supports shall satisfy the following stress criteria for faulted conditions:
$\sigma_1 \leq 2.0$ S, $\sigma_1 + \sigma_2 \leq 2.4$ S. (σ_1 and σ_2 are defined in NF-3221.1 of ASME III.)

b. Supports for Vessels Procured Prior to July 1, 1974:

1. Linear

a) Normal - The allowable stresses of AISC-69, Part 1 are employed for normal condition allowables.

b) Upset - Stress limits for upset conditions are 33% higher than those specified for normal conditions. This is consistent with paragraph 1.5.6 of AISC-69, Part 1 which permits one-third increase in allowable stresses for wind or seismic loads.

c) Emergency - Not applicable.

d) Faulted - Stress limits for faulted condition are the same as for the upset condition.

2. Plate and Shell

a) Normal - Normal condition limits are those specified in ASME Section VIII, Division 1 or AISC-69, Part 1.

b) Upset - Stress limits for upset conditions are 33% higher than those specified for normal conditions. This is consistent with paragraph 1.5.6 of AISC, Part 1 which permits one-third increase in allowable stresses for wind or seismic loads.

c) Emergency - Not applicable.

d) Faulted - Stress limits for faulted condition are the same as for the upset condition.

c. Supports for Pumps

The stress limits used Class 2 and 3 pumps supports are identical to those used for the supported component, as indicated in Tables 3.9-7 and 3.9-8.

3.9.3.1.2 Balance of Plant Components and Component Supports

ASME Code Class 1

See Subsection 3.9.1 for a discussion of ASME Code Class 1 components.

ASME Code Class 2 and 3

For safety-related ASME Code Class 2 and 3 components and component supports the combinations of design loadings are categorized with respect to plant conditions identified as normal, upset, emergency, or faulted as shown in Table 3.9-5.

The design stress limits for each of the loading combinations are presented in Tables 3.9-6 through 3.9-9. Inelastic methods as permitted by ASME Section III for Class I components were not used for these components.

3.9.3.1.3 Piping and Piping Supports

ASME Code Class 1

Piping

For ASME Code Class 1 piping, the combinations of design loadings are categorized with respect to plant conditions identified as normal, upset, emergency, or faulted as shown in Tables 3.9-11 and 3.9-11a. The design stress limits for each of the loading combinations are presented in Table 3.9-12.

Piping Supports

For pipe supports, the design loading combinations are presented in Tables 3.9-11 and 3.9-11a. The design stress limits for all loading conditions shall be consistent with ASME Section III, Subsection NF.

ASME Code Class 2 and 3

Piping

For ASME Code Class 2 and 3 piping the combinations of design loadings are categorized with respect to plant conditions identified as normal, upset, emergency or faulted as shown in Tables 3.9-13 and 3.9-13b. The design stress limits for each of the loading combinations are presented in Table 3.9-14.

Piping Supports

For pipe supports, the design loading combinations are presented in Tables 3.9-13 and 3.9-13b. The design stress limits for all loading conditions shall be consistent with ASME Section III, Subsection NF.

Functional Capability

To address the functional capability of Class 2 and 3 piping, the criteria outlined in Texas Utilities letter TXX 3423 is used. These criteria have been reviewed and accepted by the Mechanical Engineering Branch of the NRC.

3.9.3.1.4 Field Run Piping (Balance of Plant)

No Seismic Category I field run piping system exists. Category II piping, 2-inch nominal pipe size and smaller, and 200°F and colder, are field run. Criteria are provided to the contractor to ensure proper routing and design interface with Seismic Category I systems and equipment, or interfaces, are appropriately controlled by guides.

3.9.3.2 Pump and Valve Operability Assurance

Balance of Plant

Design methods are a combination of analysis, past testing, and operating experience.

Active mechanical equipment classified as Seismic Category I has been shown capable of performing its function during the life of the plant under postulated plant conditions.

Equipment with operating condition functional requirements includes "active" (active equipment must perform a mechanical motion during the course of accomplishing a safety function) pumps and valves in fluid systems such as the residual heat removal system, safety fluid injection systems, and the essential service water system.

Operability will be ensured by satisfying the requirements of the following programs. Continued operability is ensured by periodic testing.

NSSS

Mechanical equipment classified as safety-related must be capable of performing its function under postulated plant conditions. Equipment with faulted condition functional requirements includes active pumps and valves in fluid systems such as the residual heat removal system, safety injection system, and the containment.

All of the Westinghouse active pump applications have gathered extensive operating time. These pumps are seismically qualified by a combination of analysis and test which includes structural and operability analysis. Each pump is tested in the vendor's shop to verify hydraulic and mechanical performance. Performance is again checked at the plant site during preoperational system checks and periodically per ASME Inservice Testing Criteria. Pump design is specified, with strong consideration given to shaft critical speed, bearing, and seal design. Thermal transient and 100-hour endurance tests have been completed on the centrifugal charging and the safety injection pumps. Additional rotor dynamics tests have been performed on the centrifugal charging pumps which are the highest speed applications. A thermal transient analysis has been performed on the RHR pump; this analysis is supported by the vendor's test on a similar design.

Endurance and leak determination testing has been completed on the mechanical seals by the seal supplier or long-term seal reliability has been demonstrated by previous industry operating experience and by technical evaluation. Seal testing included various temperature, pressure, radiation, and boric acid concentration levels. These test conditions were substantially elevated over those expected during normal or post-accident conditions, or test differences were technically evaluated on a case-by-case basis to justify and document the long-term reliability and operability of the seals.

Subsection 6.3.2.5, "System Reliability" discusses the reliability of pumps used for long-term core cooling. The reliability program extends to the procurement of the ECCS components so that only designs which have been proven by past use in similar applications are acceptable for use. For example, the equipment specification for the ECCS pumps (safety injection, centrifugal charging, and residual heat removal pumps) require them to be capable of performing their long-term cooling function for one year. The same type of pumps have been used extensively in other operating plants. Their function during recurrent normal power and cooldown operations in such plants as Zion, D. C. Cook, Trojan, and Farley has successfully demonstrated their performance capability. Reliability tests and inspections (see Subsection 6.3.4.2) further confirm their long-term operability. Nevertheless, design provisions are included that would allow maintenance on ECCS pumps, if necessary, during long-term operation.

The operability of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

All ECCS equipment has been designed to perform its system operating function for at least 1 year without any periodic maintenance. The specific accident scenario and the associated emergency operating procedures determine the continuous period of time, from the onset of the accident, that each subsystem of ECCS pumps (CV, SI, RH) is required to operate in support of the long-term core cooling function of the ECCS. The two independent ECCS subsystems or trains allow maintenance to be performed on any pump, if it is necessary, during long-term operation.

The NRC has revised its guidance for determining susceptibility of PWR recirculation sump screens to the effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS or Containment Spray System (CSS). The revised guidance was developed as part of the efforts to resolve Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance".

For the evaluation of PWR recirculation performance in the context of GSI-191, the NRC has specified the extended period of time for long term core cooling is considered to be 30 days. Therefore, the CSS and ECCS system components have been evaluated and have been found acceptable for 30 days of operation under debris laden fluid conditions. The resolution to GSI-191 is covered in more details in UFSAR Section A1.82.

3.9.3.2.1 Pumps

Balance of Plant

All active pumps as listed in Table 3.9-15 are qualified for operability by first being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shop tests include (1) hydrostatic tests of pressure-retaining parts; and (2) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump/motor parameters. After the pump is installed in the plant, it undergoes the cold hydro tests, functional tests, and the required periodic inservice inspection and operation. These tests demonstrate reliability of the pump for the design life of the plant.

NSSS

All active pumps, listed in Table 3.9-15 are qualified for operability by first being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shop tests include (1) hydrostatic tests of pressure-retaining parts to 150% of the design pressure, and (2) performance tests to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump parameters. Also monitored during these operating tests are bearing temperatures and vibration levels. Bearing temperature limits are determined by the manufacturer based on the bearing material, clearances, oil type, and rotational speed. These limits are approved by Westinghouse. After the pump is installed in the plant, it undergoes the cold hydro tests, hot functional tests, and the required periodic inservice inspection and operation. These tests demonstrate that the pump will function as required during all normal operating conditions for the design life of the plant.

In addition to these tests, the safety-related active pumps are qualified for operability by assuring that the pump will start up, continue operating, and not be damaged during the faulted condition.

The pump manufacturer is required to show by analysis correlated by tests, prototype tests, or existing documented data that the pump will perform its safety function when subjected to loads imposed by the maximum seismic accelerations and the maximum faulted nozzle loads. It is required that test or dynamic analysis be used to show that the lowest natural frequency of the pump is greater than 33 hertz. The pump, when having a natural frequency above 33 hertz, is considered essentially rigid. This frequency is sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed with the conservative safe shutdown earthquake (SSE) accelerations of 2.1g in two orthogonal horizontal directions and 2.1g vertical acting simultaneously. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The nature of seismic disturbances dictates that the maximum contact (if it occurs) will be of short duration. If rubbing or impact is predicted, it is required that it be shown by prototype tests or existing documented data that the pump will not be damaged or cease to perform its design function. The effect of impacting on the operation of the pump is evaluated by comparison of the impacting surfaces of the pump to similar surfaces of pumps which have been or will be tested.

In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the limits indicated in Table 3.9-8. In addition, the pump casing stresses caused by the maximum faulted nozzle loads are limited to the stresses outlined in Table 3.9-8. The changes in operating rotor clearances caused by casing distortions due to these nozzle loads will be considered. The maximum seismic nozzle loads combined with the loads imposed by the seismic accelerations are also considered in an analysis of the pump supports. Furthermore, the calculated misalignment is shown to be less than that misalignment which could cause pump misoperation. The stresses in the supports are below those in Table 3.9-8, thus the support distortion is elastic and of short duration (equal to the duration of the seismic event).

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9-8 as allowables assures that critical parts of the pump will not be damaged during the short duration of the faulted condition and that, therefore, the reliability of the pump for post-faulted condition operation will not be impaired by the seismic event.

If the natural frequency is found to be below 33 hertz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations are determined using the same conservatisms contained in the 2.1g horizontal and 2.1g vertical accelerations used for "rigid" structures. The static analysis is performed using the adjusted accelerations. The stress limits stated in Table 3.9-8 must still be satisfied.

To complete the seismic qualification procedures, the pump motor will be qualified for operation during the maximum seismic event. Any auxiliary equipment, which is identified to be vital to the operation of the pump or pump motor and which is not proven adequate for operation by the pump or motor qualifications, is also separately qualified by meeting the requirements of IEEE 344-1975 with the additional requirements and justifications outlined in Subsection 3.9.3.2.3.

The program above gives the required assurance that the safety-related pump and motor assemblies will not be damaged, will continue operating under SSE loadings and therefore will perform their intended functions. These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

Since the pump is not damaged during the faulted condition, the functional ability of active pumps after the faulted condition is assured since only normal operating loads and steady-state nozzle loads exist. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be identical to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2.1.1 Seismic Analysis of Pumps

Balance of Plant

In addition to these required tests, the pumps are designed and supplied in accordance with the following specified criteria:

a. In order to ensure that the active pump will not be damaged during the seismic event; the pump manufacturer is required to demonstrate by test or analysis that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, will be considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. The natural frequency of the support is determined and used in conjunction with the applicable seismic response spectra.

   In case the natural frequency is found to be below 33 Hz, a dynamic or pseudo dynamic analysis is performed to determine the amplified input accelerations necessary to perform the stress analysis.

b. Additional loads considered in the stress analysis of the pumps and their supports are the nozzle loads for the applicable plant condition from interconnecting piping systems.

c. In addition to the stress analysis, a static shaft deflection analysis of the rotor is performed. The deflection determined from the static shaft analysis is compared to the allowable rotor clearances.

d. To complete the seismic qualification procedures, the pump motor and all appurtenances vital to the operation of the pump are independently qualified for operation during the maximum seismic event in accordance with IEEE Standard 344-1975 (see Section 3.10). In the analysis interaction between the pump and motor is considered.

e. Alternatively, the entire pump assembly with appurtenances may be qualified by testing in accordance with IEEE Standard 344-1975. In performing the seismic testing the nozzle loads for the applicable plant condition must be applied.

From this, it is concluded that the safety-related pump and motor assemblies will not be damaged, will continue operating under SSE loadings and will perform their intended functions. These requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

3.9.3.2.2 Valves

All valves in the reactor coolant pressure boundary and in other Seismic Category I systems whose operation is relied upon either to assure safe plant shutdown or to mitigate the consequences of a transient or accident are tabulated in Table 3.9-16, which lists the type and size of valve and the actuator types. The environmental conditions to which the valves are qualified are tabulated in Table 3.11-2.

Balance of Plant

Safety-related active valves, listed in Table 3.9-16, must perform their mechanical motion in times of an accident. Qualification tests and analyses have been conducted for all active valves to assure valve operability under seismic and environmental conditions.

The valves were subjected to testing prior to service (in-shop and preoperational field) and in situ (during plant life) as required by specific service and functional requirements.

In-shop tests include the following: a) ASME Code required hydrostatic tests to assure pressure boundary integrity; b) specified conformance to Manufacturers' Standard Practice code requirements regarding hydrostatic tests and main seat leakage; c) specified timed operational tests (valve stroking) when additional verification of design requirements is necessary.

Cold hydro qualification tests, hot functional qualification tests, and periodic inservice operation are performed in situ to verify and ensure the functional ability of the valve. These tests and appropriate maintenance ensure operability of the valve for the design life of the plant. The valves are designed using either the standard or the alternate design rules of ASME III.

On all active valves, an analysis of the extended structure is also performed for static equivalent seismic loads applied at the center of gravity of the extended structure. The maximum stresses and deflection allowed in these analyses demonstrate operability and structural integrity.

Valves which are safety-related but can be classified as not having an overhanging structure, such as check valves and safety/relief valves, are considered separately.

Due to the particular simple characteristics of the check valves, they will be qualified by a combination of the following tests and analysis:

a. stress analysis including the seismic loads where applicable,

b. in-shop hydrostatic tests,

c. in-shop seat leakage tests, and

d. periodic in situ valve exercising and inspection to ensure the functional capability of the valve.

The safety/relief valves are qualified by the following procedures. These valves are also subjected to tests and analysis similar to check valves; stress analyses including the seismic loads, in-shop hydrostatic seat leakage and performance tests. In addition to these tests, periodic in situ valve inspection, as applicable, and periodic valve removal, refurbishment, performance testing, and reinstallation are performed to ensure the functional capability of the valve.

Using the methods described, all the safety-related active valves in the systems are qualified for operability during a seismic event. These methods, proposed conservatively, simulate the seismic event and ensure that the active valves will perform their safety-related function when necessary.

NSSS

Safety-related active valves, listed in Table 3.9-16, must perform their mechanical motion in times of an accident. Assurance is supplied that these valves will operate during a seismic event. Test and analyses are conducted to qualify active valves.

The safety-related valves were subjected to a series of stringent tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test to ASME Section III requirements, backseat and main seat leakage tests, disc hydrostatic tests, and operational test to verify that the valve will open and close. For the qualification of motor operators for environmental conditions refer to Section 3.11. Cold hydro tests, hot functional qualifications tests, periodic inservice inspections, and periodic inservice operations are performed in situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant. The valves are constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. On active valves, an analysis of the extended structure is performed for static equivalent seismic SSE loads applied at the center of gravity of the extended structure. The maximum stress limits used for active Class 2 and 3 valves are shown in Table 3.9-9.

All valves in safety-related applications are subjected to seat leakage testing per MSS (Manufacturer's Standardization Society) SP-61, which requires testing in the closed position with a pressure differential of no less than 1.10 times the 100°F rating across the disc. Even though the criteria for pressure boundary items allow the system design pressure to peak as high as 1.5 times the system design pressure (under the faulted condition), in actuality, none of the safety-related balance of plant system pressure transients peak higher than 1.06 times the design pressure. This occurs in the main steam system and is caused by operation of the main steam relief valves. The design pressure in this case is 1185 psig, with the peak transient pressure reaching 1250 psig. Therefore, production seat leakage per SP-61 is proof that no valve disc will fail while in the closed position.

In addition to these tests and analyses, representative valves of each design type are tested for verification of operability during a simulated plant faulted condition event by demonstrating operational capabilities within the specified limits. A representative value of a specific design type is identified for this testing by the specification (e.g., globe valve, motor-operated valve, etc.) type for that particular type of valve. A stratification of design type is further made based upon the valve size, pressure rating, type of operator, and previous operability testing to evaluate the need for additional testing of a particular design type. The testing procedures are described below.

The valve is mounted in a manner which conservatively represents typical valve installations. The valve includes the operator pilot solenoid valves and limit switches when such are normally attached to the valve in service. The faulted condition nozzle loads are considered in the test in either of two ways: (1) loads equivalent to the faulted condition nozzle loads are simultaneously applied to the valve through its mounting during the below described test, or (2) by analysis, the nozzle loads are shown not to affect the operability of the valve. Interface requirements are specified to limit nozzle loads such that deflection or deformation of the valve materials will not affect the operability of the valve. The operability of a rigid valve (natural frequency equal to or greater than 33 Hertz) is demonstrated by satisfying the following criteria:

a. The actuator and yoke of the valve system are statically deflected by an amount equal to the deflection caused by the faulted condition accelerations applied at the center of gravity of the operator alone in the direction of the weakest axis of the yoke. The design pressure of the valve will be simultaneously applied to the valve during the static deflection tests.

b. The valve is cycled while in the deflected position. The time required to open or close the valve in the deflection position will be compared to similar data taken in the undeflected condition to evaluate the significance of any change.

c. Motor operators, external limit switches, and pilot solenoid valves necessary for operation are qualified by IEEE 344-1975 with the additional requirements and justifications as supplied in Subsection 3.9.3.2.2.

The accelerations which are used for the static valve qualification shall be equivalent, as justified by analysis, to 2.1g in two orthogonal horizontal directions and 2.1g vertical. The piping designer must maintain the operator accelerations to these levels.

If the natural frequency of the valve is less than 33 hertz, a dynamic analysis of the valve will be performed to determine the equivalent acceleration which will be applied during the static test. The analysis will provide the amplification of the input acceleration considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted accelerations will be determined using the same conservatisms contained in the 2.1g horizontal and 2.1g vertical accelerations used for "rigid" valves. The adjusted acceleration will then be used in the static analysis and the valve operability will be assured by the methods outlined in steps b and d above, using the modified acceleration input.

The above testing program applies to valves with extended structures. The testing is conducted on a representative number of valves. Valves from each of the primary safety-related design types are tested. Valve sizes which cover the range of sizes in service are qualified by the tests, and the results are used to qualify all valves within the intermediate range of sizes.

Valves which are safety-related but can be classified as not having an extended structure, such as check valves and safety valves, are considered separately.

Check valves are characteristically simple in design, and their operation will not be affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact and there are no extended structures or masses whose motion could cause distortions which could restrict operation of the valve. The nozzle loads due to maximum seismic excitation will not affect the functional ability of the valve, since the valve disc is typically designed to be isolated from the body wall. The clearance supplied by the design around the disc will prevent the disc from becoming bound or restricted due to any body distortions caused by nozzle loads. Therefore, the design of these valves is such that once the structural integrity of the valves are assured using standard methods, the ability of the valve to operate is assured by the design features. The valves have also undergone the following: 1) in-shop hydrostatic test, 2) in-shop seat leakage test, and 3) periodic in situ valve exercising and inspection to assure the functional ability of the valves.

The pressurizer safety valves are qualified by the following procedures (these valves are also subjected to tests and analysis similar to check valves): stress and deformation analyses of critical items which may affect operability for faulted condition loads, and in-shop hydrostatic and seat leakage tests. In addition to these tests, a static load equivalent to that applied by the faulted condition is applied at the top of the bonnet and the pressure is increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve assures its overpressurization safety capabilities during a seismic event. The valves also undergo periodic in situ valve inspection and periodic in situ or in-shop setpoint verification testing to ensure their functional capability.

Using these methods, all the safety-related valves in the system are qualified for operability during a faulted event.

These methods outlined above conservatively simulate the seismic event and assure that the active valves will perform their safety-related function when necessary.

3.9.3.2.3 Pump Motor and Valve Operator Qualification

NSSS

Active pump motors (and vital pump appurtenances) and active valve motor operators (and limit switches and solenoid valves) were seismically qualified in accordance with IEEE 344-1975. If the testing option was chosen, sine-beat testing was used. Justification of sine-beat testing was provided by satisfying one or more of the following requirements to demonstrate that multifrequency response is negligible or the sine-beat input is of sufficient magnitude to conservatively account for this effect.

a. The equipment response is basically due to one mode.

b. The sine-beat response spectra envelopes the floor response spectra in the region of significant response.

c. The floor response spectra consists of one dominant mode and has a peak at this frequency.

If the degree of coupling in the equipment was small, then single-axis testing was justified. Multi-axis testing was required if there was considerable cross coupling. However, if the degree of coupling was determined, then single-axis testing was used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

Seismic qualification by analysis alone or by a combination of analysis and testing may be used when justified. The analysis program can be justified by: 1) demonstrating that equipment being qualified is amenable to analysis, and 2) demonstrate that the analysis can be correlated with test or be performed using standard analysis techniques.

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 for the input forcing functions. Therefore, a dynamic amplification factor is inherently accounted for in the analysis.

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.

In addition to steam relief conditions including water slug effects, water relief resulting from protection against the cold overpressure condition during cooldown is also considered. The water relief rates used in the loading analysis are calculated for each transient case using the valve discharge coefficient which is obtained directly from the valve drawings provided by the vendor. The water hammer condition is not applicable to support analysis.

3.9.3.3.1 Pressurizer Safety and Relief System

The pressurizer safety and relief valve discharge piping systems provide overpressure protection for the RCS. The three spring-loaded safety valves, located on top of the pressurizer, are designed to prevent system pressure from exceeding design pressure by more than 10%. The two power-operated relief valves, also located on top of the pressurizer, are designed to prevent system pressure from exceeding the normal operating pressure by more than 100 psi. A water seal is maintained upstream of each valve to minimize leakage. Condensate accumulation on the inlet side of each valve prevents any leakage of hydrogen gas or steam through the valves. The valve outlet side is sloped to prevent the formulation of additional water pockets.

The pressurizer safety valves, manufactured by Crosby, are self-actuated spring-loaded valves with backpressure compensation. The power-operated relief valves, manufactured by Copes-Vulcan, are air-operated globe valves, capable of automatic operation via high pressure signal or remote manual operation. The safety valves and relief valves are located in the pressurizer cubicle and are supported by the attached piping which, in turn, is supported by a system of beams, struts, and snubbers.

If the pressure exceeds the setpoint and the valves open, the water slug from the loop seal discharges. The water slug, driven by high system pressure, generates transient thrust forces at each location where a change in flow direction occurs. The valve discharge conditions considered in the analysis of the Pressurizer Safety and Relief Valve (PSARV) piping systems are as follows: 1) the three safety valves are assumed to open simultaneously while the relief valves remain closed, and 2) the two relief valves open simultaneously while the safety valves are closed. In addition to these two cases, which consider water seal discharge (water slug) followed by steam, solid water from the pressurizer (cold overpressure) is also investigated.

For each pressurizer safety and relief piping system, an analytical hydraulic model is developed. The piping from the pressurizer nozzle to the relief tank nozzle is modeled as a series of legs. The pressurizer is modeled as a reservoir which contains steam at constant pressure (approximately 2500 psia for safety system and approximately 2350 psia for relief system) and at constant temperature of approximately 680°F. The pressurizer relief tank is modeled as a sink which contains steam and water mixture.

Fluid acceleration inside the pipe generates reaction forces on all segments of the line which are bounded at either end by an elbow or bend. Reaction forces resulting from fluid pressure and momentum variations are calculated. These forces are defined in terms of the fluid properties for the transient hydraulic analysis.

Unbalanced forces are calculated for each straight segment of pipe from the pressurizer to the relief tank. The time histories of these forces are used for the subsequent structural analysis of the pressurizer safety and relief lines.

The structural model used in the seismic analysis of the safety and relief lines is modified for the valve thrust analysis to represent the safety and relief valve discharge. The time-history hydraulic forces are applied to the piping system lump mass points. The dynamic solution for the valve thrust is obtained by using a modified predictor-corrector-integration technique and normal mode theory.

The time-history solution is performed in subprogram FIXFM3. The input to this subprogram consists of the natural frequencies and normal modes, applied forces, and nonlinear elements. The natural frequencies and normal modes for the modified pressurizer safety and relief line dynamic model are determined with the WESTDYN program. The support loads are computed by multiplying the support stiffness matrix and the displacement vector at each support point. The time-history displacements of the FIXFM3 subprogram are used as input to the WESDYN2 subprogram to determine the internal forces, deflections, and stresses at each end of the piping elements.

The loading combinations considered in the analysis of the PSARV piping are given in Tables 3.9-11a and 3.9-13b. These load combinations are consistent with the final recommendations of the piping subcommittee of the EPRI PWR PSARV performance test program.

3.9.3.4 NSSS Component Supports

3.9.3.4.1 Description of the NSSS Component Supports

The general arrangement of the NSSS component support systems are provided in Figure 3.9-4 through 3.9-10.

3.9.3.4.1.1 Reactor Vessel Supports

The reactor vessel is supported at four of the eight nozzles by four individual weldments set in the reactor primary shield wall as shown in Figure 3.9-4. Each of the four nozzle pads bears on a shoe, supported by a frame which wraps around the shoe. The frame is cooled to avoid harmful heating of concrete at the support frame. The reactor support system allows the reactor to expand radially over the supports but resists translational and rotational movement by the combined tangential restraining action of the nozzle support.

3.9.3.4.1.2 Pressurizer Support

The pressurizer is supported at the bottom by a ring girder which in turn is supported vertically by four columns and horizontally by an embedment. The vessel is also restrained horizontally by a second support approximately 26 feet above the lower support.

The upper restraint consists of four individual weldments embedded in concrete which allow the pressurizer to expand radially but resist rotational and translational movements by providing lateral support through the restraint lugs on the pressurizer. The pressurizer lateral supports are shown in Figure 3.9-5.

3.9.3.4.1.3 Steam Generator Support

Each steam generator is supported by a structural system consisting of four vertical support columns and an upper and a lower lateral restraint. The vertical columns, as shown in Figure 3.9-6, have a universal pinned connection at each end to accommodate both the radial growth of the steam generator itself and the radial movement of the vessel from the reactor center.

The lower lateral support shown in Figure 3.9-7 consists of an inner frame, keyed and shimmed to the four steam generator support feet to accommodate radial growth. The inner frame slides in an outer frame which is embedded in the adjacent structural concrete. The transfer of horizontal forces between the inner and outer frame is accomplished by means of a series of shimmed points which act as both guides and limit-stops to allow for expansion along a line directed toward the center of the reactor. The lower lateral support restrains both rotational and translational movements of the steam generator.

The function of the upper lateral support shown in Figure 3.9-8 is to allow vertical thermal displacements of the steam generator at that elevation and to restrain the steam generator laterally. To allow thermal displacements, but still provide seismic and LOCA restraint, hydraulic snubbers are provided aligned with the direction of the movement. These snubbers are designed such that their stiffness, stroke, lockup velocity, bleed rate after lockup, and load fall within the following conservatively specified analytical parameters:

Steam Generator/ Snubber Data	Byron Unit 1	Braidwood Unit 1	Byron/ Braidwood Unit 2
Maximum Load per Snubber			
Normal (kips)	5	25	25
Faulted (SSE + Pipe Break)(kips)	651	899	1570
Snubber Design			
Maximum Bleed Rate after Lockup (in./min.)	0.30	0.30	0.30
Lockup velocity (in./min.)	6-10	5-7	5-7
Total Stroke (in.)	4.5	4.25	4.25
Total Stiffness per Snubber Assembly (kips/in.)	6040±604	9000 ± 1800	9000 ± 1800

Bearings at either end of the snubber can accommodate a minimum of 2 inches of vertical growth, and 1° of rotation in the horizontal direction.

The snubbers were subjected to static tests and an analytic stress report was generated, showing compliance with the foregoing performance parameters. Proper cold position installation has been specified, and each snubber will be checked regularly for proper thermal movement. The snubbers are accessible for inspection, testing, and repair/replacement.

3.9.3.4.1.4 Reactor Coolant Pump Support

The reactor coolant pump is supported vertically by three universally pin-ended columns which rest on the base slab. This structural column system resists vertical movement and in conjunction with the horizontal framing system resists overturning while allowing for expansion from the center of reactor. The reactor coolant pump supports are shown in Figures 3.9-9 and 3.9-10.

The pump is restrained against horizontal translation and against rotation about the vertical axis by a structural framing system supported by the secondary shield wall as shown in Figure 3.9-10.

3.9.3.4.2 Applicable Codes, Standards, and Specifications

a. ASME Boiler and Pressure Vessel Code.

   1. Section III - Division 1, Subsection NA, Appendices I, XVII, and F.

   2. Section III - Division 1, Subsection NF

   3. Code Case 1644.

b. ACI 318, Building Code Requirements for Reinforced Concrete

c. AISC, Specification for the Design, Fabrication and Erection of Structural Steel for Buildings.

d. Regulatory Guides are addressed in Appendix A.

3.9.3.4.3 Loads and Loading Combinations

The loads, load combinations, and load factors considered in the design of the supports are shown in Table 3.9-2.

3.9.3.4.4 Design and Analysis Procedures

3.9.3.4.4.1 General

Design of the component supports has been achieved by means of a joint effort between Sargent & Lundy, Westinghouse Corporation, and Framatome Technologies, Inc. (Unit 1 only). The selection of support geometry and sizing of members has been carried out by Sargent & Lundy. For Unit 1, the analysis of the main loop including the supports, has been performed by Framatome Technologies, Inc. However, the pressurizer was not included in the main loop analysis by Framatome Technologies, Inc. for Unit 1. For Unit 2, the analysis of the main loop including the supports has been performed by Westinghouse. Analysis and design of the pressurizer supports for both Units 1 and 2 are based on the main loop analysis by Westinghouse.

The NSSS component supports have been assessed for faulted condition loads which include the effects of subcompartment pressurization and have been found to be within the allowables described in Subsection 3.9.3.4 and Regulatory Guides 1.124 and 1.130, which include the 2/3 critical buckling stress limitations.

The design of the pressurizer supports for Unit 1 and NSSS component supports for Unit 2 was originally based on conservative procedures for calculating and combining forces. The forces due to earthquake were calculated on the basis of bounding SSE spectrum which is correct for the steam generator upper lateral supports but conservative for the remainder of the system. Peak values of the LOCA forces were considered to act simultaneously even though they occur at different times in the time history of the LOCA. The earthquake induced forces were then absolutely summed with the forces due to LOCA, etc., where the direction of the earthquake force was chosen to give the worst effect possible on the support.

The reassessment of the pressurizer supports for Unit 1 and NSSS supports for Unit 2 for asymmetric pressurization and a limitation of stresses to 2/3 critical buckling stress utilized the following loads to obtain a more accurate estimate of these loads:

a. A time history analysis of the NSSS components coupled to the inner structure was used to generate the support earthquake forces.

b. The actual values of the force components (F_x, F_y, F_z) due to LOCA at the time steps which control the design were utilized in the analysis.

c. The effect of the earthquake was combined with the effect of LOCA by the SRSS method per NUREG-0484.

In addition to these refinements, the steam generator lower lateral support was modified by the addition of a brace to reduce weak axis bending effects.

The NSSS component supports are within design allowables.

For Unit 1, design of the NSSS supports, with the exception of the pressurizer supports, has been performed using support earthquake forces from response spectrum analysis of the main loop. The effect of the earthquake is combined with the effect of a LOCA using either absolute sum or the SRSS method per NUREG-0484. The direction of resulting support forces due to a LOCA and an earthquake is chosen to produce the worst effect possible on the supports.

The analysis of the supports was performed using the methods of analysis, computer codes and models described below and in Figures 3.9-11 through 3.9-15.

The critical buckling stresses and the allowable stresses used were obtained from the ASME code and Regulatory Guides 1.124 and 1.130.

For design of the original NSSS supports, a preliminary design was done by estimating loads and sizing members to withstand the corresponding forces. Analytical models of the supports were generated in order to evaluate their stiffnesses, which were incorporated in the Westinghouse loop model. The Westinghouse loop model was then employed to predict the forces acting on the supports by means of spectral analyses for the OBE and SSE earthquakes and time-history analyses for the postulated breaks for LOCA. The results of the loop analysis were used to assess the state of stress in the supports and the support design was revised when the code limits were exceeded. The analytical models were modified to reflect the revisions, new stiffnesses developed and the loop model employed again to assess the stress states in the component supports and the supports again modified. The cycle was repeated and a final analysis was performed to demonstrate that the supports are in compliance with code provisions.

The effects of jet impingement consequent to the postulated breaks were taken into account in the design of the supports.

Modifications have been made in the Unit 1 steam generator upper lateral supports. NSSS support stiffnesses used in the Westinghouse main loop analysis have been modified by Framatome Technologies, Inc. in the main loop analysis of Unit 1.

The effects of normal operating temperatures on the strength of component supports were taken into account in the design of supports. In the design by analysis, the effect of temperature is accounted for by a reduction in yield stress (S_y), ultimate tensile stress (S_u), and modulus of elasticity (E), as specified in Subsection NF-3229, Appendix XVII, Article 1121, and Appendix F, Section 1370(a) of the ASME Code, Section III, Division 1, Summer 1975. The reduction in yield stress, ultimate stress and modulus of elasticity is in accordance with ASME Section III, Appendix I or Code Case 1644. Table 3.9-17 shows the operating temperatures for the component supports.

The development and use of the analytical models employed to determine stiffnesses and states of stress are described below.

3.9.3.4.4.2 Assumptions and Limitations

The design is based on an elastic analysis of the system and supports with stress limits in accordance with ASME Section III, Subsection NF. Consequently the supports have been treated by means of small deflection linearly elastic beam and plate theory for purposes of the analysis.

Shimming between supports and components is effective in transfer of compression but ineffective in transfer of tension across the component-support interface. This fact has been recognized in the development of the support stiffness matrices and the system analysis by developing and employing stiffness matrices for appropriate combinations of effective shims.

Local yielding due to a combination of residual stresses, imperfections, and second-order effects is expected under faulted conditions. This fact is neglected in keeping with the conventional elastic analysis method.

It was assumed that the massive base slab and secondary shield walls are sufficiently rigid, compared to the supports, that the embedments therein may be treated as rigid fixed ends for purposes of the analysis.

3.9.3.4.4.3 Analytical Models

Finite element analytical models of each of the component supports were prepared in order to assess support stiffnesses for use in the Westinghouse and Framatome Technologies, Inc. loop models and to determine stresses in the supports. The nature of the models varied according to the mode of construction of the supports as described below.

a. Reactor Pressure Vessel Supports

   The finite element model of an RPV support is shown in Figure 3.9-11. The 3-inch-thick central plate was represented by means of plane stress elements. The stiffeners were represented by means of beam elements. The bars provided beneath the 5-inch plate, which are expected to behave essentially as shims, were treated as simple two-force members effective only in compression. The shims between the RPV support and the shoe affixed to the nozzle were treated in a similar manner.

   The concrete in the immediate vicinity of the RPV was represented by means of plane stress elements in the model in order to take into account the effects of local deformations.

   The stress and stiffness analyses were accomplished by means of the program SLSAP IV. The stiffness values were employed in the Westinghouse and Framatome Technologies, Inc. loop analyses. The forces from the loop analysis were in turn introduced as loads on the model, stresses determined, and compared with limiting values from Subsection NF and Appendix F, and found to be within the limits in the final design cycle.

b. Steam Generator Upper Lateral Support

   The finite-element model employed to assess stiffness and state of stress for the steam generator upper lateral support is shown in Figure 3.9-12. The model includes a portion of the steam generator shell in order to take into account the effects of shell deformation. The segment of the shell is taken from diaphragm to diaphragm and assumed fixed. The shims placed between the support ring and the steam generator and between the bracket and the wall were represented by means of two-force members. The analysis was accomplished by performing several analyses in succession with the shim elements found to be in tension in the preceding cycle removed. The bolts in the splice were treated as two-force members when the snubbers acted in tension.

The ring, the plates in the bracket, and the stiffeners on the bracket were represented by means of plate bending-plane stress elements in order to check local bending as well as membrane stresses.

For purposes of analysis the shell segment representing the steam generator was held fixed and loads applied at the location of the snubber bolts and/or the points of contact between the support embedments and the brackets.

c. Steam Generator Lower Lateral Support

The finite element model employed to assess the stiffness and state of stress for the steam generator lower lateral support is shown in Figure 3.9-13. The outer frame was represented as a linear system composed entirely of beam column elements. Completely fixed restraints are provided where the outer frame members are embedded into the concrete. Another restraint for out-of-plane deflection and rotation is provided on the outer frame member above the steam generator outlet nozzle. The inner frame model for Byron Unit 1 consists of an assemblage of plate-bending plane-stress elements employed to represent the 3-inch and 4-inch plates and the beam elements employed to represent the stiffeners. The inner frame model for Byron Unit 2 and Braidwood Units 1 and 2 consists of a geometrically similar assemblage of plate-bending plane-stress elements representing the 6-1/2-inch plate employed there.

The shims between the inner and outer frame and those between the inner frame and the steam generator were represented by two-force members which were assumed to be effective only when they were found to be in compression. The brackets which transmit vertical forces between the inner and outer frame were represented by beam elements.

The effects of shear deformation in the beam elements of the frame were taken into account in the analysis. Loads normal to the horizontal middle surface of the support due to support weight and second order effects were taken into account in the design.

d. Steam Generator Vertical Support Columns

Since the steam generator columns are provided with spherical joints at their extremities, they have been treated analytically as simple two-force members with different tension and compression stiffness for purposes of the loop analysis. The radii of the spherical surfaces were determined to minimize bending at the ends of the column stubs.

The plate detail pieces were treated analytically as plates in order to ensure satisfactory performance under tensile loads and to determine the overall stiffnesses for treatment as two-force members. Different stiffnesses were employed for tension and compression.

The 5-1/2-inch diameter bolts employed as pins were investigated analytically to confirm their adequacy under shear in the slack state.

e. Reactor Coolant Pump Lateral Support

The reactor coolant pump lateral support was treated as a linear frame and represented analytically by means of the computer program SLSTRUDL-II in the form shown in Figure 3.9-14. The shims which transfer forces between the pump lug opposite the cold leg and the frame members are inclined with respect to the axes of the members abutting the lug in order to permit unrestricted thermal motion. This inclination induces a force normal to the axis of length of the member in the absence of friction. A friction force of comparable or lesser magnitude is expected in this instance. Therefore loads normal to the axes of the members which were expected for the friction-free case were employed in the design calculations. Friction forces normal to the axis of length of the members were considered to act on the ends of the members abutting the lugs adjacent to the cold legs.

Construction tolerances were specified on the design drawings and were considered in the support design as well as the as-built conditions.

f. Reactor Coolant (RC) Pump Vertical Support Columns

The RC pump vertical support columns were treated analytically in the same manner as the steam generator vertical support columns with allowance made for the difference in end details at the tops.

g. Pressurizer Upper Lateral Support

The pressurizer upper lateral support was treated as a plate structure and represented analytically by means of the finite-element model in order to evaluate stresses and stiffness. The stiffeners were represented by means of beam elements, and the plate and an effective area of the concrete slab were represented by means of plane stress elements. The support was designed to resist both radial and tangential loads, but is expected to carry only tangential loads with the shim detail employed.

h. Pressurizer Lower Lateral Support and Vertical Support Column

The pressurizer lower lateral support serves to transfer horizontal and vertical force due to LOCA and earthquake to the secondary shield wall and vertical forces to the columns. The SLSAP IV finite-element model employed to represent the pressurizer lower lateral support for purposes of analysis is shown in Figure 3.9-15. The boxed ring girder employed to transfer vertical forces to the columns and the beam stubs extending from the ring girder to the wall were represented by means of beam elements. The plate extending from the box girder to the wall between the beam stubs was represented by means of plate/shell elements. The columns were represented by the appropriate boundary condition at the junction of the column and ring girder.

3.9.3.4.5 Structural Acceptance Criteria

The NSSS component supports have been designed to provide adequate strength and sufficient stiffness to limit displacements as required. Supports were designed to provide stiffnesses in ranges requested by Westinghouse for the several component supports. The displacements were checked as part of the Westinghouse loop analysis.

The stress limits of Subsection NF of Section III, Division I of the ASME Boiler and Pressure Vessel Code, that are appropriate for use with elastic analysis of the system and supports were employed in conjunction with the major portion of the provisions of the nonmandatory Appendix F of Subsection NA as the strength criteria with the exception of shear stress limits for high strength support bolts, which are determined by the requirements of Regulatory Guide 1.124.

3.9.3.4.6 Materials, Quality Control, and Special Construction Techniques

The materials, quality control, and special construction provisions are discussed in Appendix B.

3.9.3.4.7 Testing and Inservice Surveillance Program

Testing and inservice surveillance comply with the requirements of Subsection NF Section III, Division I and Section XI of the ASME Code.

3.9.4 Control Rod Drive Systems

3.9.4.1 Descriptive Information of CRDS

Full Length Control Rod Drive Mechanism

Control rod drive mechanisms are located on the dome of the reactor vessel. They are coupled to rod control clusters which have absorber material over the entire length of the control rods and derive their name from this feature. The full length control rod drive mechanism is shown in Figure 3.9-16 and schematically in Figure 3.9-17.

The primary function of the full length control rod drive mechanism is to insert or withdraw rod cluster control assemblies within the core to control average core temperature and to shut down the reactor.

The full length control rod drive mechanism is a magnetically operated jack. A magnetic jack is an arrangement of three electromagnets which are energized in a controlled sequence by a power cycler to insert or withdraw rod cluster control assemblies in the reactor core in discrete steps. Rapid insertion of the rod cluster control assemblies occurs when electrical power is interrupted.

The control rod drive mechanism consists of four separate sub-assemblies. They are the pressure vessel, coil stack assembly, latch assembly, and the drive rod assembly.

a. The pressure vessel includes a latch housing and a rod travel housing which are connected by a threaded, seal welded maintenance joint which facilitates replacement of the latch assembly. The closure at the top of the rod travel housing is a threaded plug with a canopy seal weld for pressure integrity.

A reduced diameter seismic sleeve fits over the top of the rod travel housing and passes through the missile shield. This sleeve provides seismic support for the control rod drive mechanism. The closure plug at the top of the rod travel housing has an opening for venting the control rod drive mechanism. A threaded vent plug is installed to provide a pressure boundary seal. This vent plug may be accessed through an opening in the top of the seismic sleeve using a special tool.

A second threaded plug and seal weld may be installed in the vent opening if leakage past the original vent plug occurs. This second plug and seal weld must be removed in order to access the original vent plug.

The latch housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel and provides space for the drive rod during its upward movement as the control rods are withdrawn from the core.

b. The coil stack assembly includes the coil housings, an electrical conduit and connector, and three operating coils: (1) the stationary gripper coil, (2) the movable gripper coil, and (3) the lift coil.

   The coil stack assembly is a separate unit which is installed on the drive mechanism by sliding it over the outside of the latch housing. It rests on the base of the latch housing without mechanical attachment.

   Energizing the operating coils causes movement of the pole pieces and latches in the latch assembly.

c. The latch assembly includes the guide tube, stationary pole pieces, movable pole pieces, and two sets of latches; (1) the movable gripper latches and (2) the stationary gripper latches.

   The latches engage grooves in the drive rod assembly. The movable gripper latches are moved up or down in 5/8-inch steps by the lift pole to raise or lower the drive rod. The stationary gripper latches hold the drive rod assembly while the movable gripper latches are repositioned for the next 5/8-inch step.

d. The drive rod assembly includes a flexible coupling, a drive rod, a disconnect button, a disconnect rod, and a locking button.

   The drive rod has 5/8-inch grooves which receive the latches during holding or moving of the drive rod. The flexible coupling is attached to the drive rod and provides the means for coupling to the rod cluster control assembly.

   The disconnect button, disconnect rod, and locking button provide positive locking of the coupling to the rod cluster control assembly and permit remote disconnection of the drive rod.

The control rod drive mechanism is a trip design. Tripping can occur during any part of the power cycler sequencing if electrical power to the coils is interrupted.

The control rod drive mechanism is threaded and seal welded on an adaptor on top of the reactor vessel and is coupled to the rod cluster control assembly directly below.

The mechanism is capable of raising or lowering a 360-pound load (which includes the drive rod weight) at a rate of 45 in./min. Withdrawal of the rod cluster control assembly is accomplished by magnetic forces, while insertion is by gravity.

The mechanism internals are designed to operate in 650°F reactor coolant. The pressure vessel is designed to contain reactor coolant at 650°F and 2500 psia. The three operating coils are designed to operate at 392°F with forced air cooling required to maintain that temperature.

The full length control rod drive mechanism shown schematically in Figure 3.9-17 withdraws and inserts a rod cluster control assembly as shaped electrical pulses are received by the operating coils. An ON or OFF sequence, repeated by silicon controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 42 discrete coils mounted on the position indicator assembly surrounding the rod travel housing. Each coil magnetically senses the entry and presence of the top of the ferromagnetic drive rod assembly as it moves through the coil center line.

During plant operation, the stationary gripper coil of the drive mechanism holds the rod cluster control assembly in a static position until a stepping sequence is initiated at which time the movable gripper coil and lift coil is energized sequentially.

<u>Rod Cluster Control Assembly Withdrawal</u>

The rod cluster control assembly is withdrawn by repetition of the following sequence of events (refer to Figure 3.9-17):

a. Movable Gripper Coil (B) - ON

   The latch locking plunger raises and swings the movable gripper latches into the drive rod assembly groove. A 0.047-inch axial clearance exists between the latch teeth and the drive rod.

b. Stationary Gripper Coil (A) - OFF

   The force of gravity, acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move downward 0.047 inch until the load of the drive rod assembly and attached control rod is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

c. Lift Coil (C) - ON

The 5/8-inch gap between the movable gripper pole and the lift pole closes and the drive rod assembly raises one step length (5/8 inch).

d. Stationary Gripper Coil (A) - ON

The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing and the stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 0.047 inch. The 0.047-inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

e. Movable Gripper Coil (B) - OFF

The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

f. Lift Coil (C) - OFF

The gap between the movable gripper pole and lift pole opens. The movable gripper latches drop 5/8 inch to a position adjacent to a drive rod assembly groove.

g. Repeat Step 1

The sequence described above (Items a through f) is termed as one step or one cycle. The rod cluster control assembly moves 5/8 inch for each step or cycle. The sequence is repeated at a rate of up to 72 steps per minute and the drive rod assembly (which has a 5/8-inch groove pitch) is raised up to 72 grooves per minute. The rod cluster control assembly is thus withdrawn at a rate up to 45 inches per minute.

Rod Cluster Control Assembly Insertion

The sequence for rod cluster control assembly insertion is similar to that for control rod withdrawal, except the timing of lift coil (C) ON and OFF is changed to permit lowering the control assembly.

a. Lift Coil (C) - ON

   The 5/8-inch gap between the movable gripper and lift pole closes. The movable gripper latches are raised to a position adjacent to a drive rod assembly groove.

b. Movable Gripper Coil (B) - ON

   The latch locking plunger raises and swings the movable gripper latches into a drive rod assembly groove. A 0.047-inch axial clearance exists between the latch teeth and the drive rod assembly.

c. Stationary Gripper Coil (A) - OFF

   The force of gravity, acting upon the drive rod assembly and attached rod cluster control assembly, causes the stationary gripper latches and plunger to move downward 0.047 inch until the load of the drive rod assembly and attached rod cluster control assembly is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

d. Lift Coil (C) - OFF

   The force of gravity and spring force separates the movable gripper pole from the lift pole, and the drive rod assembly and attached rod cluster control drop down 5/8 inch.

e. Stationary Gripper (A) - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the three stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 0.047 inch. The 0.047-inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

f. Movable Gripper Coil (B) - OFF

The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

g. Repeat Step 1

The sequence is repeated, as for rod cluster control assembly withdrawal, up to 72 times per minute, which gives an insertion rate of 45 in./min.

Holding and Tripping of the Control Rods

During most of the plant operating time, the control rod drive mechanisms hold the rod cluster control assemblies withdrawn from the core in a static position. In the holding mode, only one coil, the stationary gripper coil (A), is energized on each mechanism. The drive rod assembly and attached rod cluster control assemblies hang suspended from the three latches.

If power to the stationary gripper coil is cut off, the combined weight of the drive rod assembly and the rod cluster control assembly and the stationary gripper return spring is sufficient to move latches out of the drive rod assembly groove. The control rod falls by gravity into the core. The trip occurs as the magnetic field, holding the stationary gripper plunger half against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by the stationary gripper return spring and weight acting upon the latches. After the rod cluster control assembly is released by the mechanism, it falls freely until the control rods enter the dashpot section of the thimble tubes in the fuel assembly.

3.9.4.2 Applicable CRDS Design Specifications

For those components in the control rod drive system comprising portions of the reactor coolant pressure boundary, conformance with General Design Criteria 15, 30, 31, 32 and 10 CFR 50: Section 50.55a is discussed in Section 5.2. Conformance with Regulatory Guides pertaining to materials suitability is described in Section 4.5 and Subsection 5.2.3.

Design Bases

Bases for temperature, stress on structural members, and material compatibility are imposed on the design of the reactivity control components.

Design Stresses

The control rod drive system is designed to withstand stresses originating from various operating conditions as summarized in Table 3.9-1. Loading combinations for the Class 1 components of the control rod drive system are given in Table 3.9-2.

a. Allowable Stresses

For normal operating conditions Section III of the ASME Boiler and Pressure Vessel Code is used. All pressure boundary components are analyzed as Class I components under Article NB-3000.

b. Dynamic Analysis

The cyclic stresses due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the control rod drive system.

Control Rod Drive Mechanisms

The control rod drive mechanism (CRDM) pressure housings are Class 1 components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Boiler and Pressure Vessel Code. Both static and alternating stress intensities are considered. The stresses originating from the required design transients are included in the analysis.

A dynamic seismic analysis is required on the CRDMs when a seismic disturbance has been postulated to confirm the ability of the pressure housing to meet ASME Code, Section III allowable stresses and to confirm its ability to trip when subjected to the seismic disturbance.

Full Length Control Rod Drive Mechanism Operational Requirements

The basic operational requirements for the full length CRDMs are:

a. 5/8-inch step;

b. 144-inch travel;

c. 360-pound maximum load;

d. nominally step in or out in manual at 48 steps/minute for control banks and 64 steps/minute for shutdown banks and vary from 8 to 72 steps/minute in automatic;

e. electrical power interruption shall initiate release of drive rod assembly;

f. trip delay time of less than 150 milliseconds - free fall of drive rod assembly shall begin less than 150 milliseconds after power interruption no matter what holding or stepping action is being executed with any load and coolant temperature of 100°F to 550°F, and

g. 40-year design life with normal refurbishment.

Pressurized Components in the CDRM

The loading combinations for each plant conditions are:

Plant Conditions	Loading Combinations
Design	Deadweight, Design Pressure, Design Temperature, OBE
Upset	Deadweight, Upset Condition Transients, OBE
Faulted	Deadweight, Faulted Condition Transients, SSE or SSE and LOCA

(OBE = Operating Basis Earthquake, SSE = Safe Shutdown Earthquake, and LOCA = Loss-of-Coolant Accident)

The stress intensity limits are identified in Section III of the ASME Boiler and Pressure Vessel Code.

Nonpressurized Components in the CRDM

The CRDM nonpressure retaining components consist of the latch assembly, the drive rod assembly and the coil stack assembly. The design of these components does not come under the jurisdiction of the ASME Boiler and Pressure Vessel Code.

The latch assembly and the drive rod assembly are designed to withstand the dynamic loads associated with the stepping sequence. The coil stack assembly is designed to function outside the CRDM pressure boundary.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

3.9.4.3.1 Pressure Vessel

The pressure retaining components are analyzed for loads corresponding to normal, upset, emergency, and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

a. control rod trip (equivalent static load),

b. differential pressure,

c. spring preloads,

d. coolant flow forces (static),

e. temperature gradients,

f. differences in thermal expansion,

    1. due to temperature differences, and

    2. due to expansion of different materials,

g. interference between components,

h. vibration (mechanically or hydraulically induced),

i. all operational transients listed in Table 3.9-1,

j. pump overspeed,

k. seismic loads (operating basis earthquake and design-basis earthquake), and

l. blowdown forces (due to cold and hot leg break).

The main objective of the analysis is to satisfy allowable stress limits, given in NB-3200 and NA Appendix F, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established to assure that peak stresses will not reach unacceptable values, and to limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard methods of strength of materials are used to establish the stresses and deflections of these components. The dynamic behavior of the reactivity control components has been studied using experimental test data and experience from operating reactors.

3.9.4.3.2 Drive Rod Assembly

All postulated failures of the full length drive rod assemblies either by fracture or uncoupling lead to a reduction in reactivity. If the drive rod assembly fractures at any elevation, that portion remaining coupled falls with, and is guided by, the rod cluster control assembly. This always results in reactivity decrease for full length control rods.

3.9.4.3.3 Latch Assembly and Coil Stack Assembly

With respect to the control rod drive mechanism system as a whole, critical clearances are present in the following areas:

a. latch assembly (diametral clearances),

b. latch arm-drive rod clearances,

c. coil stack assembly-thermal clearances, and

d. coil fit in coil housing.

The following defines clearances that are designed to provide reliable operation in the control rod drive mechanism in these four critical areas. These clearances have been proven by life tests and actual field performance at operating plants.

Latch Assembly - Thermal Clearances

The magnetic jack has several clearances where parts made of Type 410 stainless steel fit over parts made from Type 304 stainless steel. Differential thermal expansion is therefore important. Minimum clearance of these parts at 68°F is 0.011 inch. At the maximum design temperature of 650°F, minimum clearance is 0.0045 inch; at the maximum expected operating temperatures of 550°F it is 0.0057 inch.

Latch Arm - Drive Rod Clearances

The control rod drive mechanism incorporates a load transfer action. The movable or stationary gripper latch is not under load during engagement, as previously explained, due to load transfer action.

Figure 3.9-18 shows latch clearance variation with the drive rod as a result of minimum and maximum temperatures. Figure 3.9-19 shows clearance variations over the design temperature range.

Coil Stack Assembly - Thermal Clearances

The assembly clearance of the coil stack assembly over the latch housing was selected so that the assembly could be removed under all anticipated conditions of thermal expansion.

At 70°F the inside diameter of the coil stack is 7.308/7.298 inches. The outside diameter of the latch housing is 7.260/7.270 inches.

Thermal expansion of the mechanism due to operating temperature of the control rod drive mechanism results in a minimum inside diameter of the coil stack of 7.310 inches at 222°F and of the maximum latch housing diameter of 7.302 inches at 532°F.

Under the extreme tolerance conditions listed above it is necessary to allow time for a 70°F coil housing to heat during a replacement operation.

Four coil stack assemblies were removed from four hot control rod drive mechanisms mounted on 11.035-inch centers on a 550°F test loop, allowed to cool, and then placed without incident as a test to prove the preceding.

Coil Fit in Coil Housing

Control rod drive mechanism and coil housing clearances are selected so that coil heat up results in a close or tight fit. This is done to facilitate thermal transfer and coil cooling in a hot control rod drive mechanism.

3.9.4.3.4 CRDS Performance Assurance Program

Evaluation of Adequacy of Materials

The ability of the pressure housing components to perform throughout the design lifetime as defined in the equipment specification is confirmed by the stress analysis report required by the ASME Boiler and Pressure Vessel Code, Section III.

Internal components subjected to wear will withstand a minimum of 3,000,000 steps without refurbishment as confirmed by life tests (Reference 8).

To confirm the mechanical adequacy of the fuel assembly, the control rod drive mechanism, and full length rod cluster control assembly, functional test programs have been conducted on a full scale 12-foot control rod. The 12-foot prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated about 3,000,000 steps and 600 trips. At the end of the test the control rod drive mechanism was still operating satisfactorily. A correlation was developed to predict the amplitude of flow-excited vibration of individual fuel rods and fuel assemblies. Inspection of the drive line components did not reveal significant fretting.

These tests include verification that the trip time achieved by the full length control rod drive mechanisms meets the design requirement of 2.7 seconds or less from beginning of decay of stationary gripper coil voltage to dashpot entry. This trip time requirement was confirmed for each control rod drive mechanism prior to initial reactor operation and is confirmed at periodic intervals as required by the Technical Specifications.

There are no significant differences between the prototype control rod drive mechanisms and the production units. Design materials, tolerances and fabrication techniques are the same.

These tests have been reported in Reference 8.

In addition, dynamic testing programs have been conducted by Westinghouse and Westinghouse licensees to demonstrate that control rod scram time is not adversely affected by postulated seismic events. Acceptable scram performance is assured by also including the effects of the allowable displacements of the driveline components in the evaluation of the test results.

It is expected that all control rod drive mechanisms will meet specified operating requirements for the duration of plant life with normal refurbishment.

If a rod cluster control assembly cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one rod cluster control assembly can be tolerated. More than one inoperable rod cluster control assembly could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable rod cluster control assemblies has been limited as discussed in the Technical Specification B3.1.4.

In order to demonstrate proper operation of the control rod drive mechanism and to ensure acceptable core power distributions during operation, rod cluster control assembly partial movement checks are performed on the rod cluster control assemblies (refer to Technical Specification 3.1.4). In addition, periodic drop tests of the full length rod cluster control assemblies are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements, to ensure core subcriticality after reactor trip, and to limit potential reactivity insertions from a hypothetical rod cluster control assembly ejection. During these tests the acceptable drop time of each assembly is not greater than 2.7 seconds, at full flow and operating temperature, from the beginning of decay of stationary gripper coil voltage to dashpot entry.

Actual experience in operating many Westinghouse plants indicates excellent performance of control rod drive mechanisms.

All units are production tested prior to shipment to confirm ability of the control rod drive mechanism to meet design specification-operational requirements.

Each production full length control rod drive mechanism undergoes a production test as listed below:

Test	Acceptable Criteria
Cold (ambient) hydrostatic	ASME Section III
Confirm step length and load transfer (stationary gripper to movable gripper or movable gripper to stationary gripper)	Step Length 5/8 ± 0.015 inch axial movement Load Transfer 0.047 inch nominal axial movement
Cold (ambient) performance test at design load - 5 full travel excursions	Operating Speed 45 in./min Trip Delay Free fall of drive rod to begin within 150 milliseconds

3.9.5 Reactor Vessel Internals

3.9.5.1 Design Arrangements

The reactor vessel internals are described as follows:

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and neutron shield pad assembly), the upper core support structure and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and guides for the incore instrumentation. The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the vessel wall and then into a plenum at the bottom of the vessel. It then reverses and flows up through the core support and through the lower core plate. The lower core plate is sized to provide the desired inlet flow distribution to the core. After passing through the core, the coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles. A small portion of the coolant flows between the baffle plates and the core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel outlet nozzles.

Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 3.9-20. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support which is welded to the core barrel. All the major material for this structure is Type 304 stainless steel. The lower core support structure is supported at its upper flange from a ledge in the reactor vessel head flange. Its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are an axial baffle and a lower core plate, both of which are attached to the core barrel wall and form the enclosure periphery of the assembly core. The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a member through which the necessary flow distribution holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the core support of the core barrel in order to provide stiffness and to transmit the core load to the core support. Adequate coolant distribution is obtained through the use of the lower core plate and core support.

The neutron shield pad assembly consists of four pads that are bolted and pinned to the outside of the core barrel. These pads are constructed of Type 304 stainless steel and are approximately 48 inches wide by 148 inches long by 2.8 inches thick. The pads are located azimuthally to provide the required degree of vessel protection. Specimen guides in which material surveillance samples can be inserted and irradiated during reactor operation are attached to the pads. The samples are held in the guide by a preloaded spring device at the top and bottom to prevent sample movement. Additional details of the neutron shield pads and irradiation specimen holders are given in Reference 9.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, hydraulic loads, and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the core support and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell and distributed between the lower radial support to the vessel wall, and to the vessel flange. Transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by upper core plate alignment pins which are welded into the core barrel.

The main radial support system of the lower end of the core barrel is accomplished by "key" and "keyway" joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel clevis block is welded to the vessel inner diameter. Another Inconel insert block is bolted to each of these blocks and has a "keyway" geometry. Opposite each of these is a "key" which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam supported at the top and bottom.

Radial and axial expansions of the core barrel are accommodated, but transverse movement of the core barrel is restricted by this design. With this system, cyclic stresses in the internal structures are within the ASME Section III limits. In the event of an abnormal downward vertical displacement of the internals following a hypothetical failure, energy-absorbing devices limit the displacement after contacting the vessel bottom head. The load is then transferred through the energy-absorbing devices of the internals to the vessel.

The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy-absorbing devices.

Upper Core Support Assembly

The upper core support assembly, shown in Figures 3.9-21 and 3.9-22, consists of the top support plate assembly and the upper core plate between which are contained support columns and guide tube assemblies. The support columns establish the spacing between the top support plate assembly and the upper core plate and are fastened at top and bottom to these plates. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies sheath and guide the control rod drive shafts and control rods. They are fastened to the top support plate and are restrained by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the upper guide tube which is attached to the upper support plate and guide tube.

The upper core support assembly is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 90° from each other. Four slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies and control rods is thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads and fuel assembly preload are transmitted through the upper core plate via the support columns to the top support plate assembly and then the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

At Byron Unit 1, one of the two fuel assembly locating pins from two core locations (P-9 and R-9) have been removed. These pins were damaged during the installation of the reactor vessel upper internals during B1R05 because of a misalignment of fuel assemblies. At Byron Unit 2, one of the two fuel assembly locating pins has been removed from each of six core locations (B-8, B-9, D-7, D-9, D-10, E-9). These pins were damaged during movement of the upper internals package during B2R02.

Fuel assemblies in these locations continue to meet the transverse and vertical loading criteria with a single locating pin providing support (References 12, 13, and 14).

Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom (Figure 7.7-9 shows the Basic Flux-Mapping System).

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to inline columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper incore instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144 inches and the trailing ends of the thimbles (at the seal line) are extracted approximately 15 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal line is cleared for the retraction operation.

The incore instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence. These are the only conditions which affect the incore instrumentation support structure. Reactor vessel surveillance specimen capsules are covered in Subsection 5.3.1.6.

3.9.5.2 Design Loading Conditions

The Byron/Braidwood reactor design precedes the specific applicability of Subsection NG of Section III of the ASME Code. The intent of the code is applied with load combinations and allowable stresses. A summary of the maximum total stress, deformation and usage factors is available for audit from Westinghouse.

The design loading conditions that provide the basis for the design of the reactor internals are:

a. Normal and Upset

The normal and upset loading conditions that provide the basis for the design of the reactor internals are:

1. fuel and reactor internals weight,

2. fuel and core component spring forces including spring preloading forces,

3. differential pressure and coolant flow forces,

4. temperature gradients,

5. vibratory loads including OBE seismic,

6. the normal and upset operational thermal transients listed in Table 3.9-1,

7. control rod trip (equivalent static load),

8. loads due to loop(s) out of service, and

9. loss of load pump overspeed.

b. Emergency Conditions

The emergency loading conditions that provide the basis for the design of the reactor internals are:

1. small loss-of-coolant accident,

2. small steam break, and

3. complete loss of flow.

c. Faulted Conditions

The faulted loading conditions that provide the basis for the design of the reactor internals are:

1. the large loss-of-coolant accident, and

2. the safe shutdown earthquake.

It should be noted that the dynamic effects associated with a large break in the reactor coolant loop piping need not be considered, based on leak-before-break analyses performed by Westinghouse.

The main objectives of the design analysis are to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Both low and high cycle fatigue stresses are considered when the allowable amplitude of oscillation is established. Dynamic analyses on the reactor internals are provided in Subsection 3.9.2.

As part of the evaluation of design loading conditions, extensive testing and inspections are performed from the initial selection of raw materials up to and including component installation and plant operation. Among these tests and inspections are those performed during component fabrication, plant construction, startup and checkout, and during plant operation.

3.9.5.3 Design Loading Categories

The combination of design loadings fits into either the normal, upset, emergency or faulted conditions as defined in the ASME Code, Section III, and as indicated by Figures NG-3221.1, NG-3224.1 and by Appendix F, Rules for Evaluating Faulted Conditions.

Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions as summarized in Table 3.9-1.

The scope of the stress analysis problem is very large requiring many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation under consideration.

Allowable Deflections

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For the loss-of-coolant accident plus the safe shutdown earthquake condition, the deflection criteria of critical internal structures are the limiting values given in Table 3.9-4. The corresponding no loss of function limits are included in Table 3.9-4 for comparison purposes with the allowed criteria.

The criteria for the core drop accident is based upon analyses which have to determine the total downward displacement of the internal structures following a hypothesized core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately 1/2 inch. An additional displacement of approximately 3/4 inch would occur due to strain of the energy-absorbing devices of the secondary core support; thus the total drop distance is about 1-1/4 inches, which is insufficient to permit the tips of the rod cluster control assembly to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident of the core support (core barrel, barrel flange, etc.). There are four supports in each reactor. This device limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated assuming a complete and instantaneous failure of the primary core support and is absorbed during the plastic deformation of the controlled volume of stainless steel, loaded in tension. The maximum deformation of this austenitic stainless piece is limited to approximately 15%, after which a positive stop is provided to ensure support.

For additional information on design loading categories see Subsection 3.9.2.

3.9.5.4 Design Bases

The design bases for the mechanical design of the reactor vessel internals components are as follows:

a. The reactor internals in conjunction with the fuel assemblies shall direct reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements are met for all modes of operation. In addition, required cooling for the pressure vessel head shall be provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

b. In addition to neutron shielding provided by the reactor coolant, the reactor internals are designed to limit the exposure of the pressure vessel in order to maintain the required ductility of the material for all modes of operation.

c. Provisions shall be made for installing incore instrumentation useful for the plant operation and vessel material test specimens required for a pressure vessel irradiation surveillance program.

d. The core internals are designed to withstand mechanical loads arising from the operating basis earthquake, the safe shutdown earthquake, and pipe breaks and to meet the requirements of Item "e" below.

e. The reactor shall have mechanical provisions which are sufficient to adequately support the core and internals and to assure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.

f. Following the design-basis accident, the plant shall be capable of being shut down and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

The functional limitations for the core structures during the design basis accident are shown in Table 3.9-4. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited so as not to exceed the value shown in Table 3.9-4.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Section 3.9.2.

The basis for the design stress and deflection criteria is identified below:

Allowable Stresses

For normal operating conditions, the intent of Section III of the ASME Nuclear Power Plant Components Code is used as a basis for evaluating acceptability of calculated stresses. Both static and alternating stress intensities are considered. It should be noted that the allowable stresses in Section III of the ASME Code are based on unirradiated material properties. In view of the fact that irradiation increases the strength of the Type 304 stainless steel used for the internals, although decreasing its elongation, it is considered that use of the allowable stresses in Section III is appropriate and conservative for irradiated internal structures.

The allowable stress limits during the design-basis accident used for the core support structures are based on the intent of the draft ASME Code for Core Support Structures, Subsection NG, and the Criteria for Faulted Conditions.

The stress criteria for the reactor internals that Westinghouse applied before the existence of Subsection NG of the ASME code are composed of two parts, and depend upon the nature of the stress state membrane or bending. A direct or membrane stress has a uniform stress distribution over the cross section. The allowable (maximum) membrane or direct stress is taken to be equal to the stress corresponding to 20% of the uniform material strain or the yield strength whichever is higher. For unirradiated Type 304 stainless steel at operating temperature, the stress corresponding to 20% of the uniform strain is 39,500 psi.

For a bending state of stress, the strain is linearly distributed over a cross section. The average strain value is, therefore, one-half of the outer fiber strain where the stress is maximum. Thus, by requiring the average bending stress to satisfy the allowable criteria for the direct state of stress, the average absolute strain may be 20% of the uniform strain. Consequently, the outer fiber strain may be 40% of the uniform strain. The maximum allowable outer fiber bending stress is then taken to be equal to the stress corresponding to 40% of the uniform strain or the yield strength, whichever is higher. For unirradiated Type 304 stainless steel at operating temperature, the stress is 50,000 psi.

3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of pumps and valves is done in accordance with a plan approved per 10 CFR 50.55a(f).

Byron and Braidwood have been approved to implement 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants." This regulation provides an alternative approach for establishing requirements for treatment of structures, systems, and components (SSCs) using a risk-informed method of categorizing SSCs according to their safety significance. Specifically, for SSCs categorized as low safety-significance, alternate treatment requirements may be implemented rather than treatments chosen by the inservice testing of pumps and valves program. Refer to Section 3.2.3 for further information.

3.9.6.1 Inservice Testing of Pumps

All ASME Code Class 1, 2, and 3 pumps requiring inservice testing are listed in the Inservice Testing (IST) Program Plan. Surveillance requirements for inservice testing of ASME Code Class 1, 2, and 3 pumps are included in the Technical Specifications and the station IST Program Plan.

3.9.6.2 Inservice Testing of Valves

ASME Code Class 1, 2, and 3 valves requiring inservice testing are listed in the IST Program Plan. Surveillance requirements for inservice testing of ASME Code 1, 2, and 3 valves are included in the Technical Specifications and the station IST Program Plan.

3.9.7 Motor-Operated Valve (MOV) Testing

A motor-operated valve program has been established to satisfy the NRC recommendations in NRC Generic Letter (GL) 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” and NRC GL 96-05, “Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves.” This program includes a comprehensive testing program to verify valve operability and ensure that correct switch settings are established, maintained and monitored throughout the life of the plant to ensure high reliability of safety-related MOVs. MOVs in safety-related systems are static tested with diagnostics and full differential pressure (dp) testing with diagnostics is performed, when practicable. The NRC concluded that an acceptable program was established to verify periodically the design-basis capability of the safety-related MOVs based on implementing all three phases of the Joint Owner’s Group Program on MOV Periodic Verification (References 18 and 19).

3.9.8 References

1. WCAP-8252, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," Revision 1, April 1977.

2. "Sample Analysis of a Class 1 Nuclear Piping System," prepared by ASME Working Group on Piping, ASME Publication, 1972.

3. W. H. Bamford and C. B. Buchalet, "Methods for Fracture Mechanics Analysis of Nuclear Reactor Vessels Under Severe Thermal Transients," WCAP-8510, June 1976.

4. WCAP-8317-A, "Predication of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests," July 1975.

5. C. N. Bloyd, W. Ciarametaro, and N. R. Singleton, "Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant," WCAP-8780, May 1976.

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9. S. Kraus, "Neutron Shielding Pads," WCAP-7870, May 1972.

10. "Benchmark Problem Solutions Employed for Verification of WECAN Computer Program," WCAP-8929, June 1977.

11. "OPTPIPE Piping Analysis Computer Program Verification," NUTECH Engineers, Inc. Report OPT-87-001, Revision 0, dated February 15, 1987.

12. Rarig, B. E., "Fuel Assembly Alignment Pin Removal," Westinghouse SECL 90-561, Revision 1, November 7, 1990.

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14. Humphries, B. S., "Fuel Assembly Alignment Pin Damage - Final Configuration," Westinghouse CAE-93-149, March 30, 1993.

15. Robert L. Cloud & Associates, Inc., "A Topical Report on the Methodology, Verification and Applications of Computer Program GAPPIPE," RLCA/P94/04-94/009, June 1, 1994.

16. Westinghouse Letter LTR-SST-13-100, “Software Release Letter for ANSYS Mechanical APDL 14.5.7 and ANSYS Workbench 14.5.7 on the Windows 7 and SLES11_SP2 System States,” November 4, 2013.

17. Westinghouse Letter LTR-PAFM-15-25. “Software Release Letter for WESTEMS Version 4.5.7,” April 2015.

18. Letter from U.S. NRC to O.D. Kingsley (Commonwealth Edison Company), "Generic Letter 96-05, 'Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves,' Braidwood Station, Units 1 and 2 (TAC Nos. M97018 and M97019)," dated July 30, 1999.19. Letter from U.S. NRC to O.D. Kingsley (Commonwealth Edison Company), "Generic Letter 96-05, 'Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves,' Byron Station, Units 1 and 2 (TAC Nos. M97025 and M97026)," dated December 10, 1999.

19. Letter from U.S. NRC to O.D. Kingsley (Commonwealth Edison Company), “Generic Letter 96-05, ‘Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves,’ Byron Station, Units 1 and 2 (TAC Nos. M97025 and M97026),” dated December 10, 1999.

TABLE 3.9-1

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

NORMAL CONDITIONS		OCCURRENCES
1.	Heatup and cooldown at 100°F/hr (pressurizer cooldown 200°F/hr)	200 (each)
2.	Unit loading @ 5% of full power/min.	13,200 (each)
	Unit unloading @ 5% of full power/min.	13,200 (each) 12,240 Unit 1 Steam Generators
3.	Step load increase and decrease of 10% of full power	2,000 (each)
4.	Large step load decrease with steam dump	200
5.	Steady-state fluctuations	
	a. Initial fluctuations	1.5×10^5
	b. Random fluctuations	3.0×10^6
	c. Unit 1 Steam Generators	3.5×10^6
6.	Feedwater cycling at hot shutdown	2,000
7.	Loop out of service	
	a. Normal loop shutdown	80
	b. Normal loop startup	70
8.	Unit loading and unloading between 0% and 15% of full power	500 (each)
	Loading (Unit 1 Steam Generators)	330 Cold Turbine Generator 1,130 Hot Turbine Generator 1,460 Total
9.	Boron concentration equalization	26,400
10.	Refueling	80
11.	Turbine roll test	20
12.	Primary side leak test	200

TABLE 3.9-1 (Cont'd)

13. Secondary side leak test	80
14. Tube leakage test	800 Unit 2 Steam Generators
	720 Unit 1 Steam Generators
15. Recovery of Main Feedwater Flow After Isolation (Unit 1 Steam Generators)	760
UPSET CONDITIONS	
1. Loss of load, without immediate reactor trip	80

TABLE 3.9-1 (Cont'd)

UPSET CONDITIONS	OCCURRENCES
2. Loss of power (loss of nonemergency a-c power with natural circulation in the reactor coolant system)	40
3. Partial loss of flow (loss of one pump)	80
4. Reactor trip from full power	
a. Without cooldown	230
b. With cooldown, without safety injection	160
c. With cooldown and safety injection	10
5. Inadvertent reactor coolant depressurization	20
6. Inadvertent startup of an inactive loop	10
7. Control rod drop	80
8. Inadvertent emergency core cooling system actuation	60
9. Excessive feedwater flow	30
10. Operating basis earthquake (20 earthquakes of 20 cycles each)	400 cycles
11. Thermal Stratification (Unit 1 FW/AF only)	120
12. Cold Overpressurization	10
EMERGENCY CONDITIONS*	
1. Small loss-of-coolant accident	5
2. Small steam break	5
3. Complete loss of flow	5
FAULTED CONDITIONS*	
1. Main reactor coolant pipe break** (large loss-of-coolant accident)	1
2. Large steam break	1
3. Feedwater line break	1
4. Reactor coolant pump locked rotor	1
5. Control rod ejection	1

TABLE 3.9-1 (Cont'd)

FAULTED CONDITIONS*		OCCURRENCES
6.	Steam generator tube rupture	(included under upset conditions, reactor trip from full power with safety injection)
7.	Safe shutdown earthquake	1
TEST CONDITIONS		
1.	Primary side hydrostatic test	10
2.	Secondary side hydrostatic test	10

*In accordance with the ASME Nuclear Power Plant Components Code, emergency and faulted conditions are not included in fatigue evaluation.

**Based on leak-before-break analyses performed by Westinghouse, the dynamic effects associated with a large break in the main reactor coolant loop piping need not be considered.

TABLE 3.9-2

LOADING COMBINATIONS FOR ASME CLASS 1 COMPONENTS AND COMPONENT SUPPORTS

CONDITION CLASSIFICATION	LOADING COMBINATION
Design	Design Pressure, Design Temperature, Deadweight, Operating Basis Earthquake
Normal	Normal Condition Transients, Deadweight
Upset	Upset Condition Transients, Deadweight, Operating Basis Earthquake
Emergency	Emergency Condition Transients, Normal Operating Temperature Transients, Deadweight
Faulted	Faulted Condition Transients, Deadweight, Normal Operating Temperature Transients, Safe Shutdown Earthquake, or Safe Shutdown Earthquake and Pipe Break Loads

TABLE 3.9-3

ALLOWABLE STRESSES FOR ASME SECTION III CLASS 1 COMPONENTS

OPERATING CONDITION CLASSIFICATION	VESSELS/ TANKS	PIPING	PUMPS	VALVES
Design	NB-3221 (Design)	NB-3652 (Design)	NB-3221 (Design)	NB-3521 (Design)
Normal	NB-3222 (Level A)	NB-3653 (Level A)	NB-3222 (Level A)	NB-3525 (Level A)
Upset	NB-3223 (Level B)	NB-3654 (Level B)	NB-3223 (Level B)	NB-3525 (Level B)
Emergency	NB-3224 (Level C)	NB-3655 (Level C)	NB-3224 (Level C)	NB-3526 (Level C)
Faulted	NB-3225 (Level D)	NB-3656 (Level D)	NB-3225 (Level D)	See Table 3.9-3a

Notes:

Limits identified refer to subsections of the ASME Code, Section III.

TABLE 3.9-3a

CLASS 1 VALVE FAULTED CONDITION CRITERIA[1]

ACTIVE	INACTIVE
a) Calculate Pm from para. NB3545.1 with Internal Pressure Ps = 1.25Ps $Pm \leq 1.5Sm$	a) Calculate Pm from para. NB3545.1 with Internal Pressure Ps = 1.50Ps $Pm \leq 2.4Sm$ or $0.7Su$
b) Calculate Sn from para. NB3545.2 with Cp = 1.5 Ps = 1.25Ps Qt2 = 0 Ped = 1.3X value of Ped from equations of 3545.2(b) (1) $Sn \leq 3Sm$	b) Calculate Sn from para. NB3545.2 with Cp = 1.5 Ps = 1.50Ps Qt2 = 0 Ped = 1.3X value of Ped from equations of NB3545.2(b) (1) $Sn \leq 3Sm$

1. The parameters shown are representative of class 1 valve criteria developed by Westinghouse. Refer to the applicable valve specification for the specific values.

TABLE 3.9-4

MAXIMUM DEFLECTIONS ALLOWED FOR REACTOR INTERNAL SUPPORT STRUCTURES

COMPONENT	ALLOWABLE DEFLECTIONS (in.)	NO LOSS OF FUNCTION DEFLECTIONS (in.)
Upper Barrel		
Radial inward	4.1	8.2
Radial outward	1.0	1.0
Upper Package	0.10	0.15
Rod Cluster Guide Tubes	1.00	1.75

TABLE 3.9-5

DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 2 AND CLASS 3 COMPONENTS AND SUPPORTS

CONDITION CLASSIFICATION	LOADING COMBINATION
Design and Normal	Design pressure, Design temperature*, Dead weight, Thermal***, Nozzle loads**
Upset	Upset condition pressure, Upset condition metal temperature*, Deadweight, Thermal†, OBE, Nozzle loads**
Emergency	Emergency condition pressure, Emergency condition metal temperature*, Deadweight, SSE (BOP only), Thermal***, Nozzle loads**
Faulted	Faulted condition pressure, Faulted condition metal temperature*, Deadweight, Thermal***, SSE (NSSS only), Nozzle loads**

Note: Seismic loads are combined in accordance with Regulatory Guide 1.92. The loads are then combined algebraically using ± seismic. (See Table 3.9-13 for piping and piping supports.)

* Temperature is used to determine allowable stress only.

** Nozzle loads are those loads associated with the particular plant operating conditions for the component under consideration. Both the most positive and most negative nozzle loads are considered. The nozzle loads are combined as described in the note above.

*** Thermal loads are for the normal operating Thermal Mode.

† Thermal loads are the larger of the normal operating thermal loads and the thermal loads associated with the upset condition event.

TABLE 3.9-6

STRESS CRITERIA FOR SAFETY-RELATED ASME CODE CLASS 2 AND CLASS 3 VESSELS

CONDITION	STRESS LIMITS[1]
Design and Normal	The vessel shall conform to the requirements of ASME Section III, NC-3300 (or ND-3300)
Upset	$\sigma_m \leq 1.1\ S$
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65\ S$
Emergency	$\sigma_m \leq 1.5\ S$
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80\ S$
Faulted	$\sigma_m \leq 2.0\ S$
	$(\sigma_m \text{ or } \sigma_L) + \sigma_m \leq 2.4\ S$

[1]Stress limits are taken from ASME III, Subsections NC and ND, or, for vessels procured prior to the incorporation of these limits into ASME III, from Code Case 1607.

TABLE 3.9-7

STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3 INACTIVE PUMPS AND PUMP SUPPORTS

CONDITION	STRESS LIMITS*	P_{max}**
Design and Normal	The pump shall conform to the requirements of ASME Section III, NC-3400 (or ND-3400)	
Upset	$\sigma_m \le 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \le 1.65$ S	1.1
Emergency	$\sigma_m \le 1.5$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \le 1.80$ S	1.2
Faulted	$\sigma_m \le 2.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \le 2.4$ S	1.5

*Stress limits are taken from ASME III, Subsections NC and ND, or, for pumps procured prior to the incorporation of these limits into ASME III, from Code Case 1636.

**The maximum pressure shall not exceed the tabulated factors listed under P_{max} times the design pressure.

TABLE 3.9-8

DESIGN CRITERIA FOR ACTIVE PUMPS AND PUMP SUPPORTS

CONDITION	DESIGN CRITERIA[*]
Design and Normal	ASME Section III Subsection NC-3400 and ND-3400
Upset	$\sigma_m \leq 1.0$ S $\sigma_m + \sigma_b \leq 1.5$ S
Emergency	$\sigma_m \leq 1.2$ S $\sigma_m + \sigma_b \leq 1.65$ S
Faulted	$\sigma_m \leq 1.2$ S $\sigma_m + \sigma_b \leq 1.8$ S

[*]The stress limits specified for active pumps are more restrictive than the ASME III limits. For the Faulted Condition (membrane plus bending), stresses may exceed 1.8 S but must remain below the material yield stress. In such cases, a deflection analysis is performed to assure that the maximum displacements are within the deflection limits which will not impair the operability of the equipment.

TABLE 3.9-9

STRESS CRITERIA FOR SAFETY-RELATED ASME CODE CLASS 2 AND CLASS 3 NSSS AND BOP VALVES (ACTIVE AND INACTIVE)

CONDITION	STRESS LIMITS (NOTES 1-4 & 6-7)	P_{max}(Note 5)
Design and Normal	Valve bodies shall conform to ASME Section III.	
Upset	$\sigma_m < 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S	1.1
Emergency	$\sigma_m < 1.5$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.80$ S	1.2
Faulted	$\sigma_m \leq 2.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 2.4$ S	1.5

Notes:

1. Valve nozzle (piping load) stress analysis is not required when both of the following conditions are satisfied: (1) the section modulus and area of every plane, normal to the flow, through the region defined as the valve body crotch are at least 110% of those for the piping connected (or joined) to the valve body inlet and outlet nozzles; and, (2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated above shall be multiplied by the ratio of S_{pipe}/S_{valve}. If unable to comply with this requirement, the design by analysis procedure of NB3545.2 is an acceptable alternate method.

2. Casting quality factor in accordance with ASME Section III shall be used.

TABLE 3.9-9 (Cont'd)

3. These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

4. Design requirements listed in this table are not applicable to valve stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet. See Note 8 for criteria used to ensure that the valve disc will not fail should the valve be subjected to “Pmax” while in the closed position.

5. The maximum pressure resulting from upset, emergency, or faulted conditions shall not exceed the tabulated factors listed under P_{max} times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in this table are considered to be satisfied.

6. Stress limits are taken from ASME III, Subsections NC and ND, or, for valves procured prior to the incorporation of these limits into ASME III, from Code Case 1635.

7. Consideration shall be given to the need for qualification testing of complex active devices such as valve operators and gate or disk assemblies where analytical methods may not provide sufficient assurance of operability.

8. All valves in safety-related applications are subject to seat leakage testing. Testing may be accomplished per MSS (Manufacturer’s Standardization Society) SP-61, which requires testing in the closed position with a pressure differential of no less than 1.10 times the 100 F rating across the disc, or per the requirements of specification L/F-2884 which requires testing in the closed position with a differential pressure of 275 psig. Even though the criteria for pressure boundary items allow the system design pressure to peak as high as 1.5 times the system design pressure (under the faulted condition), in actuality, none of the safety-related balance of plant system pressure transients peak higher than 1.06 times the design pressure. This occurs in the main steam system and is caused by operation of the main steam relief valves. The design pressure in this case is 1185 psig, with the peak transient pressure reaching 1250 psig. Therefore, production seat leakage per SP-61 or L/F-2884 is proof that no valve disc will fail when in the closed position.

TABLE 3.9-10

This Table was intentionally deleted.

TABLE 3.9-11

LOADING COMBINATIONS FOR ASME SECTION III
CLASS 1 PIPING AND SUPPORTS

SERVICE LEVEL	OPERATING CONDITION CLASSIFICATION	LOADING COMBINATION
A	Design	Design Pressure, Design Temperature, Deadweight
A	Normal	Normal Condition Transients, Deadweight
B	Upset	Upset Condition Transients, Deadweight, Operating Basis Earthquake
C	Emergency	Emergency Condition Transients, Normal Operating Temperature Transients[3], Deadweight
D	Faulted	Faulted Condition Transients, Normal Operating Temperature Transients3, Deadweight, Safe Shutdown Earthquake, or Safe Shutdown Earthquake and Pipe Break Loads

[3] Thermal loads apply to support combinations only.

TABLE 3.9-11a

ADDITIONAL LOAD COMBINATIONS AND ACCEPTANCE CRITERIA
FOR PRESSURIZER
SAFETY AND RELIEF VALVE PIPING AND SUPPORTS+

ASME CLASS 1 PORTION

CONDITION CLASSIFICATION	LOAD COMBINATION**	SERVICE LIMIT
Design	Design Pressure, Weight	Design
Normal	Sustained loads during normal plant operation.	Level A
Upset	Sustained loads during normal plant operation, OBE*, Relief valve discharge transient***	Level B
Emergency	Sustained loads during normal plant operation, Safety valve discharge transient***	Level C
Faulted	Sustained loads during normal plant operation, SSE*, Maximum relief valve/safety valve discharge transient or transition flow***	Level D

* The OBE and SSE loadings include the effects of seismic anchor motions.

** Dynamic loads are combined by SRSS.

*** Valve thrust loads (VT) are loads resulting from the rapid acceleration or deceleration of a water mass, noncondensible gases, or both.

+For supports, the load due to pipe thermal expansion shall be considered in the load combination.

TABLE 3.9-12

ALLOWABLE STRESS FOR ASME SECTION III CLASS 1 PIPING

OPERATING CONDITION CLASSIFICATION	STRESS LIMITS
Normal	ASME Section III NB-3600
Upset	ASME Section III NB-3600
Emergency	ASME Section III NB-3600
Faulted	ASME Section III NB-3600

TABLE 3.9-13a

LOADING COMBINATIONS FOR ASME SECTION III CLASS 2 AND CLASS 3 PIPING AND SUPPORTS

SERVICE LEVEL	OPERATING CONDITION CLASSIFICATION	LOADING COMBINATION
A	Design and Normal	Design Pressure, Normal Operating Thermal, Deadweight
B	Upset	Upset Condition Pressure, Deadweight, Thermal[4], Operating Basis Earthquake
C	Emergency	Emergency Condition Pressure Deadweight, Thermal**,
D	Faulted	Faulted Condition Pressure, Thermal**, Deadweight, Safe Shutdown Earthquake, or Safe Shutdown Earthquake and Pipe Break Loads

[4] Temperature effects are based on worst case conditions, comparing the temperature effects associated with normal operating conditions and the temperature effects associated with the upset event.

** Thermal loads are required for support load combinations only. The loads are based on normal operating conditions.

TABLE 3.9-13b

ADDITIONAL LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER SAFETY AND RELIEF VALVE PIPING AND SUPPORTS+

B31.1 SEISMICALLY DESIGNED DOWNSTREAM PORTION

CONDITION CLASSIFICATION	LOAD COMBINATION**	SERVICE LIMIT
Design	Design Pressure, Weight	Design
Normal	Sustained loads during normal plant operation	Level A
Upset	Sustained loads during normal plant operation, OBE*, Relief valve discharge transient***	Level B
Emergency	Sustained loads during normal plant operation, Safety valve discharge transient***	Level C
Faulted	Sustained loads during normal plant operation, SSE*, Maximum relief valve/safety valve discharge transient or transient flow***	Level D

* The OBE and SSE loadings include the effects of seismic anchor motions.

** Dynamic loads are combined by SRSS.

*** Valve thrust loads (VT) are loads resulting from the rapid acceleration or deceleration of a water mass, noncondensible gases, or both.

+For supports, the load due to pipe thermal expansion shall be considered in the load combination.

TABLE 3.9-14

ALLOWABLE STRESSES FOR ASME SECTION III CLASS 2 AND CLASS 3 PIPING

OPERATING CONDITION CLASSIFICATION	STRESS LIMITS
Normal	ASME Section III NC/ND-3600
Upset	ASME Section III NC/ND-3600
Emergency	ASME Section III NC/ND-3600
Faulted	ASME Section III NC/ND-3600

TABLE 3.9-15

ACTIVE PUMPS FOR BYRON UNITS 1 & 2

EQUIPMENT NUMBER	NAME	QUALITY GROUP	P&ID
0AB03P	Boric Acid Transfer Pump	C	M-65-5A
1AB03P	Boric Acid Transfer Pump	C	M-65-5A
2AB03P	Boric Acid Transfer Pump	C	M-65-5A
1AF01PA	Auxiliary Feedwater Pump	C	M-37
1AF01PB	Auxiliary Feedwater Pump	C	M-37
2AF01PA	Auxiliary Feedwater Pump	C	M-122
2AF01PB	Auxiliary Feedwater Pump	C	M-122
0CC01P	Component Cooling Pump	C	M-66-3A
1CC01PA	Component Cooling Pump	C	M-66-3A
1CC01PB	Component Cooling Pump	C	M-66-3A
2CC01PA	Component Cooling Pump	C	M-66-3A
2CC01PB	Component Cooling Pump	C	M-66-3A
1CS01PA	Containment Spray Pump	B	M-46-1A
1CS01PB	Containment Spray Pump	B	M-46-1A
2CS01PA	Containment Spray Pump	B	M-129-1A
2CS01PB	Containment Spray Pump	B	M-129-1A
1CV01PA	Centrifugal Charging Pump	B	M-64-3A
1CV01PB	Centrifugal Charging Pump	B	M-64-3A
2CV01PA	Centrifugal Charging Pump	B	M-138-3A
2CV01PB	Centrifugal Charging Pump	B	M-138-3A
1DO01PA	Diesel Oil Transfer Pump	G	M-50-1B
1DO01PB	Diesel Oil Transfer Pump	G	M-50-1A
1DO01PC	Diesel Oil Transfer Pump	G	M-50-1B
1DO01PD	Diesel Oil Transfer Pump	G	M-50-1A
2DO01PA	Diesel Oil Transfer Pump	G	M-130-1A
2DO01PB	Diesel Oil Transfer Pump	G	M-130-1B
2DO01PC	Diesel Oil Transfer Pump	G	M-130-1A
2DO01PD	Diesel Oil Transfer Pump	G	M-130-1B
1RH01PA	Residual Heat Removal Pump	B	M-62

TABLE 3.9-15 (Cont'd)

EQUIPMENT NUMBER	NAME	QUALITY GROUP	P&ID
1RH01PB	Residual Heat Removal Pump	B	M-62
2RH01PA	Residual Heat Removal Pump	B	M-137
2RH01PB	Residual Heat Removal Pump	B	M-137
1SI01PA	Safety Injection Pump	B	M-61-1A
1SI01PB	Safety Injection Pump	B	M-61-1A
2SI01PA	Safety Injection Pump	B	M-136-1
2SI01PB	Safety Injection Pump	B	M-136-1
0SX02PA	Essential Service Water Makeup Pump	C	M-42-6
0SX02PB	Essential Service Water Makeup Pump	C	M-42-6
1SX01PA	Essential Service Water Pump	C	M-42-1B
1SX01PB	Essential Service Water Pump	C	M-42-1A
1SX04P	Engine Driven Cooling Water Pump	C	M-42-3
2SX01PA	Essential Service Water Pump	C	M-42-1B
2SX01PB	Essential Service Water Pump	C	M-42-1A
2SX04P	Engine Driven Cooling Water Pump	C	M-126-1
0WO01PA	Control Room Chilled Water Pump	C	M-118-1
0WO01PB	Control Room Chilled Water Pump	C	M-118-1

Note: Miscellaneous active skid mounted lube oil and cooling water pumps associated with the above major system active pumps are not listed.

TABLE 3.9-15

ACTIVE PUMPS FOR BRAIDWOOD UNITS 1 & 2

EQUIPMENT NUMBER	NAME	QUALITY GROUP	P&ID
0AB03P	Boric Acid Transfer Pump	C	M-65-5A
1AB03P	Boric Acid Transfer Pump	C	M-65-5A
2AB03P	Boric Acid Transfer Pump	C	M-65-5A
1AF01PA	Auxiliary Feedwater Pump	C	M-37
1AF01PB	Auxiliary Feedwater Pump	C	M-37
2AF01PA	Auxiliary Feedwater Pump	C	M-122
2AF01PB	Auxiliary Feedwater Pump	C	M-122
0CC01P	Component Cooling Pump	C	M-66-3A
1CC01PA	Component Cooling Pump	C	M-66-3A
1CC01PB	Component Cooling Pump	C	M-66-3A
2CC01PA	Component Cooling Pump	C	M-66-3A
2CC01PB	Component Cooling Pump	C	M-66-3A
1CS01PA	Containment Spray Pump	B	M-46-1A
1CS01PB	Containment Spray Pump	B	M-46-1A
2CS01PA	Containment Spray Pump	B	M-129-1A
2CS01PB	Containment Spray Pump	B	M-129-1A
1CV01PA	Centrifugal Charging Pump	B	M-64-3A
1CV01PB	Centrifugal Charging Pump	B	M-64-3A
2CV01PA	Centrifugal Charging Pump	B	M-138-3A
2CV01PB	Centrifugal Charging Pump	B	M-138-3A
1DO01PA	Diesel Oil Transfer Pump	G	M-50-1B
1DO01PB	Diesel Oil Transfer Pump	G	M-50-1A
1DO01PC	Diesel Oil Transfer Pump	G	M-50-1B
1DO01PD	Diesel Oil Transfer Pump	G	M-50-1A
2DO01PA	Diesel Oil Transfer Pump	G	M-130-1A
2DO01PB	Diesel Oil Transfer Pump	G	M-130-1B
2DO01PC	Diesel Oil Transfer Pump	G	M-130-1A
2DO01PD	Diesel Oil Transfer Pump	G	M-130-1B
1RH01PA	Residual Heat Removal Pump	B	M-62

TABLE 3.9-15 (Cont'd)

EQUIPMENT NUMBER	NAME	QUALITY GROUP	P&ID
1RH01PB	Residual Heat Removal Pump	B	M-62
2RH01PA	Residual Heat Removal Pump	B	M-137
2RH01PB	Residual Heat Removal Pump	B	M-137
1SI01PA	Safety Injection Pump	B	M-61-1A
1SI01PB	Safety Injection Pump	B	M-61-1A
2SI01PA	Safety Injection Pump	B	M-136-1
2SI01PB	Safety Injection Pump	B	M-136-1
1SX01PA	Essential Service Water Pump	C	M-42-1B
1SX01PB	Essential Service Water Pump	C	M-42-1A
1SX04P	Engine Driven Cooling Water Pump	C	M-42-3
2SX01PA	Essential Service Water Pump	C	M-42-1B
2SX01PB	Essential Service Water Pump	C	M-42-1A
2SX04P	Engine Driven Cooling Water Pump	C	M-126-1
0WO01PA	Control Room Chilled Water Pump	C	M-118-1
0WO01PB	Control Room Chilled Water Pump	C	M-118-1

Note: Miscellaneous active skid mounted lube oil and cooling water pumps associated with the above major system active pumps are not listed.

TABLE 3.9-16

ACTIVE VALVES FOR BYRON - UNITS 1 & 2

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
AF001A,B	---	6	Check	C	M-37/M-122
AF003A,B	---	6	Check	C	M-37-1/M-122-1
AF005A-H	Air	3	Globe	C	M-37/M-122
AF006A,B	Motor	6	Gate	C	M-37-1/M-122-1
AF013A-H	Motor	4	Globe	B	M-37/M-122
AF014A-H	---	4	Check	B	M-37-1/M-122-1
AF017A,B	Motor	6	Gate	C	M-37-1/M-122-1
AF029A,B	---	6	Check	C	M-37-1/M-122-1
AF053A,B	---	1.5x2.5	Relief	C	M-55-7E/M-55-9
AF058A,B	---	1	Check	C	M-55-7E/M-55-9
AF059A,B	---	1	Check	C	M-55-7E/M-55-9
0CC9464	---	12	Check	C	M-66-3B
CC201A,B	Motor	2.5	Globe	C	M-66-3A
CC202A,B	Motor	2.5	Globe	C	M-66-3A
CC091B	---	2	Check	C	M-66-3A
CC092B	---	2	Check	C	M-66-3A
CC685	Motor	3	Gate	B	M-66-1A/M-139-1
CC9412A,B	Motor	12	Gate	C	M-66-2/M-139-2
CC9413A,B	Motor	6	Gate	B	M-66-1A/M-139-1
CC9414	Motor	6	Gate	B	M-66-1A/M-139-1
CC9415	Motor	16	Gate	C	M-66-4D
CC9416	Motor	6	Gate	B	M-66-1A/M-139-1
CC9437A,B	Air	3	Globe	B	M-66-1A/M-139-1
CC9438	Motor	4	Gate	B	M-66-1A/M-139-1
CC9458	Manual	16	Gate	C	M-66-3B
CC9459A,B	Manual	16	Gate	C	M-66-3A
CC9463A,B	---	12	Check	C	M-66-3B
CC9467A,B,C	Manual	16	Gate	C	M-66-3B,4D
CC9473A,B	Motor	16	Gate	C	M-66-3B
CC9486	---	6	Check	B	M-66-1A/M-139-1
CC9495A,B,C,D	---	2	Check	C	M-66-1B/M-139-1
CC9503	Manual	16	Butterfly	C	M-66-3A
CC9507A,B	Manual	12	Butterfly	C	M-66-2/M-139-2
CC9512B	Manual	6	Gate	C	M-66-4C
CC9513B	Manual	2	Globe	C	M-66-4C

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
CC9516	Manual	4	Butterfly	C	M-66-4C
CC9518	---	0.75	Check	B	M-66-1A/M-139-1
CC9520A,B	---	3	Check	C	M-66-4A,4B
CC9534	---	0.75	Check	B	M-66-1A/M-139-1
CS001A,B	Motor	14	Gate	B	M-61-4/M-136-4
CS003A,B	---	10	Check	B	M-46-1A/M-129-1A
CS007A,B	Motor	10	Gate	B	M-46-1C/M-129-1C
CS008A,B	---	10	Check	B	M-46-1C/M-129-1C
CS08MA,MB	---	2x2	Relief	B	M-46-1B/M-129-1B
CS009A,B	Motor	16	Gate	B	M-61-4/M-136-4
CS011A,B	---	6	Check	B	M-46-1A/M-129-1A
CS019A,B	Motor	3	Gate	B	M-46-1B/M-129-1B
CS020A,B	---	3	Check	B	M-46-1B/M-129-1A
CV112B,C	Motor	4	Gate	B	M-64-4/M-138-4
CV112D,E	Motor	8	Gate	B	M-64-4/M-138-4
CV459	Air	3	Globe	A	M-64-5/M-138-5B
CV460	Air	3	Globe	A	M-64-5/M-138-5B
CV8100	Motor	2	Globe	B	M-64-2/M-138-2
CV8104	Motor	2	Globe	B	M-64-4/M-138-4
CV8105	Motor	3	Gate	B	M-64-3B/M-138-3B
CV8106	Motor	3	Gate	B	M-64-3B/M-138-3B
CV8110	Motor	2	Globe	B	M-64-3A/M-138-3A
CV8111	Motor	2	Globe	B	M-64-3A/M-138-3A
CV8112	Motor	2	Globe	B	M-64-2/M-138-2
CV8113	---	0.75	Check	B	M-64-2/M-138-2
CV8114	Solenoid	2	Globe	B	M-64-3A/M-138-3A
CV8116	Solenoid	2	Globe	B	M-64-3A/M-138-3A
CV8124	---	0.75x1	Relief	B	M-64-4B/M-138-4
CV8152	Air	3	Globe	B	M-64-5/M-138-5A
CV8160	Air	3	Globe	B	M-64-5/M-138-5A
CV8355A-D	Motor	2	Globe	B	M-64-1,2/M-138-1,2
CV8368A-D	---	2	Check	B	M-64-1,2/M-138-1,2
CV8440	---	4	Check	B	M-64-4B/M-138-4
CV8442	---	2	Check	B	M-64-4/M-138-4
CV8480A,B	---	2	Check	B	M-64-3A/M-138-3A
CV8481A,B	---	4	Check	B	M-64-3A/M-138-3A

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
CV8546	---	8	Check	B	M-64-4/M-138-4
CV8804A	Motor	8	Gate	B	M-64-4/M-138-4
DG5043A,B	Air	6	Globe	G	M-152-14
DG5182A,B	Air	2.5	Globe	G	M-152-20
DG5183A,B	Air	2.5	Globe	G	M-152-20
DG5184A,B	---	2.5	Check	G	M-152-20
DG5185A,B	---	2.5	Check	G	M-152-20
DO003A-D	---	1.5	Check	C	M-50-1A,1B/M-130-1A,1B
FP010	Air	4	Globe	B	M-52-1
FP360	---	0.75 x 1	Relief	C	M-52-1
FW009A-D	Hydraulic	16	Gate	B	M-36-1A,1D/M-121-1A,1D
FW035A-D	Air	3	Globe	B	M-36-1A,1D/M-121-1A,1D
FW039A-D	Air	6	Gate	B	M-36-1A,1D/M-121-1A,1D
FW043A-D(Unit 2 only)	Air	3	Globe	B	M-121-1A,1D
FW079A-D	---	16	Check	B	M-36-1A,1D/M-121-1A,1D
IA065	Air	3	Globe	B	M-55-4/M-55-5
IA066	Air	3	Globe	B	M-55-4/M-55-5
IA091	---	0.75	Check	B	M-55-4/M-55-2
MS001A,D	Hydraulic	30.25	Gate	B	M-35-2,1/M-120-2,1
MS001B,C	Hydraulic	32.75	Gate	B	M-35-1,2/M-120-1,2
MS013A-D	---	6x10	Relief	B	M-35-1,2/M-120-1,2
MS014A-D	---	6x10	Relief	B	M-35-1,2/M-120-1,2
MS015A-D	---	6x10	Relief	B	M-35-1,2/M-120-1,2
MS016A-D	---	6x10	Relief	B	M-35-1,2/M-120-1,2
MS017A-D	---	6x10	Relief	B	M-35-1,2/M-120-1,2
MS018A-D	Hydraulic	6x6	Relief	B	M-35-1,2/M-120-1,2
MS019A-D	Manual	8	Gate	B	M-35-1,2/M-120-1,2
MS101A-D	Air	4	Globe	B	M-35-1,2/M-120-1,2
0OG061	Motor	3	Butterfly	B	M-47-2
0OG062	Motor	3	Butterfly	B	M-47-2
0OG063	Motor	3	Butterfly	B	M-47-2
0OG064	Motor	3	Butterfly	B	M-47-2
OG057A	Motor	3	Butterfly	B	M-47-2/M-150-2
OG079	Motor	3	Butterfly	B	M-47-2/M-150-2
OG080	Motor	3	Butterfly	B	M-47-2/M-150-2

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
OG081	Motor	3	Butterfly	B	M-47-2/M-150-2
OG082	Motor	3	Butterfly	B	M-47-2/M-150-2
OG083	Motor	3	Butterfly	B	M-47-2/M-150-2
OG084	Motor	3	Butterfly	B	M-47-2/M-150-2
OG085	Motor	3	Butterfly	B	M-47-2/M-150-2
PR001A,B	Air	1	Globe	B	M-78-10/M-151-1
PR032	---	1	Check	B	M-78-10/M-151-1
PR066	Air	1	Globe	B	M-78-10/M-151-1
PS228A,B	Solenoid	0.5	Globe	B	M-68-7/M-140-6
PS229A,B	Solenoid	0.5	Globe	B	M-68-7/M-140-6
PS230A,B	Solenoid	1	Globe	B	M-68-7/M-140-6
PS231A,B	---	0.75	Check	B	M-68-7/M-140-6
PS9354A,B	Air	1	Globe	B	M-68-1/M-140-1
PS9355A,B	Air	1	Globe	B	M-68-1/M-140-1
PS9356A,B	Air	1	Globe	B	M-68-1/M-140-1
PS9357A,B	Air	1	Globe	B	M-68-1/M-140-1
RC014A-D	Solenoid	1	Globe	A	M-60-1B/M-135-1B
RE1003	Air	3	Diaphragm	B	M-70-1/M-141-1
RE9157	Air	1	Diaphragm	B	M-70-1/M-141-1
RE9159A,B	Air	0.75	Diaphragm	B	M-70-1/M-141-1
RE9160A,B	Air	1	Diaphragm	B	M-70-1/M-141-1
RE9170	Air	3	Diaphragm	B	M-70-1/M-141-1
RE022	---	0.75 x 1	Relief	B	M-70-1/M-141-1
RF026	Air	2	Plug	B	M-48-6B
RF027	Air	2	Plug	B	M-48-6A
RF055	---	0.75 x 1	Relief	B	M-48-6B
RH610	Motor	3	Gate	B	M-62-1/M-137-1
RH611	Motor	3	Gate	B	M-62-1/M-137-1
RH8701A,B	Motor	12	Gate	A	M-62-1/M-137-1
RH8702A,B	Motor	12	Gate	A	M-62-1/M-137-1
RH8716A,B	Motor	8	Gate	B	M-62-1/M-137-1
RH8730A,B	---	8	Check	B	M-62-1/M-137-1
RY030A,B	---	0.75x1	Relief	C	M-60-8/M-135-8
RY455A	Air	3	PORV	A	M-60-5/M-135-5
RY456	Air	3	PORV	A	M-60-5/M-135-5
RY8000A,B	Motor	3	Gate	A	M-60-5/M-135-5

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
RY8010A,B,C	---	6	Safety	A	M-60-5/M-135-5
RY8025	Air	0.375	Globe	B	M-60-6/M-135-6
RY8026	Air	0.375	Globe	B	M-60-6/M-135-6
RY8028	Air	3	Diaphragm	B	M-60-6/M-135-6
RY8033	Air	0.75	Diaphragm	B	M-60-6/M-135-6
RY8046	---	3	Check	B	M-60-6/M-135-6
RY8047	---	0.75	Check	B	M-60-6/M-135-6
SA032	Air	1.5	Globe	B	M-54-2
SA033	Air	1.5	Globe	B	M-54-2
SA181A,B,C,D	---	2	Check	C	M-54-4A,4B
SD002A-H	Air	2	Globe	B	M-48-5A/M-48-5B
SD005A-D	Air	0.375	Globe	B	M-48-5A/M-48-5B
1SD054A-H	Air	2	Globe	B	M-48-5A
2SD054B,D,F,H	Air	2	Globe	B	M-48-5B
SI101A/B	Manual	4	Gate	B	M-61-2/M-136-2
SI121A,B	---	0.75 x 1	Relief	B	M-64-1/M-136-4
SI8801A,B	Motor	4	Gate	B	M-61-2/M-136-2
SI8802A,B	Motor	4	Gate	B	M-61-3/M-136-3
SI8804B	Motor	8	Gate	B	M-61-1A/M-136-1
SI8806	Motor	8	Gate	B	M-61-1A/M-136-1
SI8807A,B	Motor	6	Gate	B	M-61-1A/M-136-1
SI8809A,B	Motor	8	Gate	B	M-61-4/M-136-4
SI8811A,B	Motor	24	Gate	B	M-61-4/M-136-4
SI8812A,B	Motor	12	Gate	B	M-61-4/M-136-4
SI8813	Motor	2	Globe	B	M-61-1B/M-136-1
SI8814	Motor	1.5	Globe	B	M-61-1A/M-136-1
SI8815	---	3	Check	A	M-61-2/M-136-2
SI8818A-D	---	6	Check	A	M-61-4/M-136-4
SI8819A-D	---	2	Check	A	M-61-3/M-136-3
SI8821A,B	Motor	4	Gate	B	M-61-3/M-136-3
SI8835	Motor	4	Gate	B	M-61-3/M-136-3
SI8840	Motor	12	Gate	B	M-61-3/M-136-3
SI8841A,B	---	8	Check	A	M-61-3/M-136-3
SI8871	Air	0.75	Globe	B	M-61-6/M-136-6
SI8880	Air	1	Globe	B	M-61-6/M-136-6
SI8888	Air	0.75	Globe	B	M-61-3/M-136-3

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
SI8900A-D	---	1.5	Check	A	M-61-2/M-136-2
SI8905A-D	---	2	Check	A	M-61-3/M-136-3
SI8919A,B	---	1.5	Check	B	M-61-1A/M-136-1
SI8920	Motor	1.5	Globe	B	M-61-1A/M-136-1
SI8922A,B	---	4	Check	B	M-61-1A/M-136-1
SI8923A,B	Motor	6	Gate	B	M-61-1A/M-136-1
SI8924	Motor	6	Gate	B	M-61-1A/M-136-1
SI8926	---	8	Check	B	M-61-1A/M-136-1
SI8948A-D	---	10	Check	A	M-61-5,6/M-136-5,6
SI8949A-D	---	6	Check	A	M-61-3/M-136-3
SI8956A-D	---	10	Check	A	M-61-5,6/M-136-5,6
SI8958A,B	---	12	Check	B	M-61-4/M-136-4
SI8964	Air	0.75	Globe	B	M-61-6/M-136-6
SI8968	---	1	Check	B	M-61-6/M-136-6
0SX007	Motor	24	Butterfly	C	M-42-2A
0SX028A,B	---	8	Check	C	M-42-6
0SX146	Motor	30.00	Butterfly	C	M-42-2A
0SX147	Motor	30.00	Butterfly	C	M-42-2A
0SX161A,B	Manual	6	Gate	C	M-42-7
0SX163A-H	Motor	24	Butterfly	C	M-42-7
SX005	Motor	30.00	Butterfly	C	M-42-1A/M-42-1A
0SX143A,B	---	8	Check	C	M-42-6
0SX162A-D	Motor	24	Butterfly	C	M-42-7
0SX284A,B	---	12	Check	C	M-42-6/M-42-6
SX002A,B	---	36	Check	C	M-42-1/M-126-3
SX016A,B	Motor	16	Butterfly	B	M-42-5A, 5B/M-126-3
SX027A,B	Motor	16	Butterfly	B	M-42-5A, 5B/M-126-3
SX112A,B	Air	12	Butterfly	C	M-42-3/M-126-1
SX114A,B	Air	12	Butterfly	C	M-42-3/M-126-1
SX147A,B	Air	16	Butterfly	C	M-42-3/M-126-1
SX168	Air	3	Globe	C	M-42-3/M-126-1
SX169A,B	Air	10	Butterfly	C	M-42-3/M-126-1
SX174	---	6	Check	C	M-42-3/M-126-1

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
VQ001A,B	Hydraulic	48	Butterfly	B	M-105-1/M-106-1
VQ002A,B	Hydraulic	48	Butterfly	B	M-105-1/M-106-1
VQ003	Air	8	Butterfly	B	M-105-1/M-106-1
VQ004A,B	Air	8	Butterfly	B	M-105-1/M-106-1
VQ005A,B,C	Air	8	Butterfly	B	M-105-1/M-106-1
WM191	---	2	Check	B	M-49-1
0WO002A,B	---	6	Check	C	M-118-1
0WO028A,B	---	1.5 x 2.5	Relief	C	M-118-1
WO006A,B	Motor	10	Gate	B	M-118-5/M-118-7
WO007A,B	---	10	Check	B	M-118-5/M-118-7
WO020A,B	Motor	10	Gate	B	M-118-5/M-118-7
WO056A,B	Motor	10	Gate	B	M-118-5/M-118-7
WO079A,B	---	0.75 x 1	Relief	B	M-118-5/M-118-7

TABLE 3.9-16

ACTIVE VALVES FOR BRAIDWOOD - UNITS 1 & 2

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
AF001A,B	---	6	Check	C	M-37/M-122
AF003A,B	---	6	Check	C	M-37-1/M-122-1
AF005A-H	Air	3	Globe	C	M-37/M-122
AF006A,B	Motor	6	Gate	C	M-37-1/M-122-1
AF013A-H	Motor	4	Globe	B	M-37/M-122
AF014A-H	---	4	Check	B	M-37-1/M-122-1
AF017A,B	Motor	6	Gate	C	M-37-1/M-122-1
AF029A,B	---	6	Check	C	M-37-1/M-122-1
AF053A,B	---	1.5x2.5	Relief	C	M-55-8
AF058A,B	---	1	Check	C	M-55-8
AF059A,B	---	1	Check	C	M-55-8
0CC9464	---	12	Check	C	M-66-3B
CC070A,B	---	3	Check	C	M-66-4A,4B
CC201A,B	Motor	2.5	Globe	C	M-66-3A
CC202A,B	Motor	2.5	Globe	C	M-66-3A
CC685	Motor	3	Gate	B	M-66-1A/M-139-1
CC9412A,B	Motor	12	Gate	C	M-66-2/M-139-2
CC9413A,B	Motor	6	Gate	B	M-66-1A/M-139-1
CC9414	Motor	6	Gate	B	M-66-1A/M-139-1
CC9415	Motor	16	Gate	C	M-66-4D
CC9416	Motor	6	Gate	B	M-66-1A/M-139-1
CC9437A,B	Air	3	Globe	B	M-66-1A/M-139-1
CC9438	Motor	4	Gate	B	M-66-1A/M-139-1
CC9458	Manual	16	Gate	C	M-66-3B
CC9459A,B	Manual	16	Gate	C	M-66-3A
CC9463A,B	---	12	Check	C	M-66-3B
CC9467A,B,C	Manual	16	Gate	C	M-66-3B,4D
CC9473A,B	Motor	16	Gate	C	M-66-3B
CC9486	---	6	Check	B	M-66-1A/M-139-1
CC9495A,B,C,D	---	2	Check	C	M-66-1B/M-139-1
CC9507A,B	Manual	12	Butterfly	C	M-66-2/M-139-2
CC9518	---	0.75	Check	B	M-66-1A/M-139-1
CC9520A,B	---	3	Check	C	M-66-4A
CC9534	---	0.75	Check	B	M-66-1A/M-139-1

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
CS001A,B	Motor	14	Gate	B	M-61-4/M-136-4
CS003A,B	---	10	Check	B	M-46-1A/M-129-1A
CS007A,B	Motor	10	Gate	B	M-46-1C/M-129-1C
CS008A,B	---	10	Check	B	M-46-1C/M-129-1C
CS009A,B	Motor	16	Gate	B	M-61-4/M-136-4
CS011A,B	---	6	Check	B	M-46-1A/M-129-1A
CS019A,B	Motor	3	Gate	B	M-46-1B/M-129-1B
CS020A,B	---	3	Check	B	M-46-1B/M-129-1A
CS08MA,MB	---	2	Relief	B	M-46-1B/M-129-1B
CV112B,C	Motor	4	Gate	B	M-64-4/M-138-4B
CV112D,E	Motor	8	Gate	B	M-64-4/M-138-4A
CV459	Air	3	Globe	A	M-64-5/M-138-5C
CV460	Air	3	Globe	A	M-64-5/M-138-5C
CV8100	Motor	2	Globe	B	M-64-2/M-138-2
CV8105	Motor	3	Gate	B	M-64-3B/M-138-3B
CV8106	Motor	3	Gate	B	M-64-3B/M-138-3B
CV8110	Motor	2	Globe	B	M-64-3A/M-138-3A
CV8111	Motor	2	Globe	B	M-64-3A/M-138-3A
CV8112	Motor	2	Globe	B	M-64-2/M-138-2
CV8113	---	0.75	Check	B	M-64-2/M-138-2
CV8114	Solenoid	2	Globe	B	M-64-3A/M-138-3A
CV8116	Solenoid	2	Globe	B	M-64-3A/M-138-3A
CV8117	---	2x3	Relief	B	M-64-5/M-138-5C
CV8124	---	0.75x1	Relief	B	M-64-4B/M-138-4A
CV8152	Air	3	Globe	B	M-64-5/M-138-5
CV8160	Air	3	Globe	B	M-64-5/M-138-5
CV8355A-D	Motor	2	Globe	B	M-64-1,2/M-138-1,2
CV8368A-D	---	2	Check	B	M-64-1,2/M-138-1,2
CV8440	---	4	Check	B	M-64-4B/M-138-4B
CV8480A,B	---	2	Check	B	M-64-3A/M-138-3A
CV8481A,B	---	4	Check	B	M-64-3A/M-138-3A
CV8546	---	8	Check	B	M-64-4/M-138-4A
CV8804A	Motor	8	Gate	B	M-64-4/M-138-4A
DG5182A,B	Air	0.375	Globe	G	M-152-20
DG5183A,B	Air	0.375	Globe	G	M-152-20
DG5184A,B	---	0.375	Check	G	M-152-20
DG5185A,B	---	0.375	Check	G	M-152-20
DG5205A,B	---	0.375	Check	G	M-152-20

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
DG5206A,B	---	0.375	Check	G	M-152-20
DG5207A,B	Air	0.375	Gate	G	M-152-20
DG5208A,B	Air	0.375	Gate	G	M-152-20
DG5209A,B	Air	0.375	Gate	G	M-152-20
DG5210A,B	Air	0.375	Gate	G	M-152-20
DO003A-D	---	1.5	Check	C	M-50-1A,1B/M-130-1A,1B
FP450	---	0.75 x 1	Relief	C	M-52-1
FP010	Air	4	Globe	B	M-52-1
FW009A-D	Hydraulic	16	Gate	B	M-36-1A,1D/M-121-1A,1D
FW035A-D	Air	3	Globe	B	M-36-1A,1D/M-121-1A,1D
FW039A-D	Air	6	Gate	B	M-36-1A,1D/M-121-1A,1D
FW043A-D(Unit 2 only)	Air	3	Globe	B	M-121-1A,1D
FW079A-D	---	16	Check	B	M-36-1A,1D/M-121-1A,1D
IA065	Air	3	Globe	B	M-55-5/M-55-10
IA066	Air	3	Globe	B	M-55-5/M-55-10
IA091	---	0.75	Check	B	M-55-5/M-55-10
MS001A,D	Hydraulic	30.25	Gate	B	M-35-2,1/M-120-2,1
MS001B,C	Hydraulic	32.75	Gate	B	M-35-1,2/M-120-1,2
MS013A-D	---	6x10	Relief	B	M-35-1,2/M-120-1,2
MS014A-D	---	6x10	Relief	B	M-35-1,2/M-120-1,2
MS015A-D	---	6x10	Relief	B	M-35-1,2/M-120-1,2
MS016A-D	---	6x10	Relief	B	M-35-1,2/M-120-1,2
MS017A-D	---	6x10	Relief	B	M-35-1,2/M-120-1,2
MS018A-D	Hydraulic	6x6	Relief	B	M-35-1,2/M-120-1,2
MS019A-D	Manual	8	Gate	B	M-35-1,2/M-120-1,2
MS101A-D	Air	4	Globe	B	M-35-1,2/M-120-1,2
OG057A	Motor	3	Butterfly	B	M-47-2/M-150-2
OG079	Motor	3	Butterfly	B	M-47-2/M-150-2
OG080	Motor	3	Butterfly	B	M-47-2/M-150-2
OG081	Motor	3	Butterfly	B	M-47-2/M-150-2
OG082	Motor	3	Butterfly	B	M-47-2/M-150-2
OG083	Motor	3	Butterfly	B	M-47-2/M-150-2
OG084	Motor	3	Butterfly	B	M-47-2/M-150-2
OG085	Motor	3	Butterfly	B	M-47-2/M-150-2
PR001A,B	Air	1	Globe	B	M-78-10/M-151-1
PR032	---	1	Check	B	M-78-10/M-151-1
PR066	Air	1	Globe	B	M-78-10/M-151-1

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
PS228A,B	Solenoid	0.5	Gate	B	M-68-7/M-140-6
PS229A,B	Solenoid	0.5	Gate	B	M-68-7/M-140-6
PS230A,B	Solenoid	0.5	Gate	B	M-68-7/M-140-6
PS231A,B	---	0.75	Check	B	M-68-7/M-140-6
PS9354A,B	Air	0.375	Globe	B	M-68-1/M-140-1
PS9355A,B	Air	0.375	Globe	B	M-68-1/M-140-1
PS9356A,B	Air	0.375	Globe	B	M-68-1/M-140-1
PS9357A,B	Air	0.375	Globe	B	M-68-1/M-140-1
RC014A-D	Solenoid	1	Globe	A	M-60-1B/M-135-1B
RE1003	Air	3	Diaphragm	B	M-70-1/M-141-1
RE9157	Air	1	Diaphragm	B	M-70-1/M-141-1
RE9159A,B	Air	0.75	Diaphragm	B	M-70-1/M-141-1
RE9160A,B	Air	1	Diaphragm	B	M-70-1/M-141-1
RE9170	Air	3	Diaphragm	B	M-70-1/M-141-1
RE040	---	0.75 x 1	Relief	B	M-70-1/M-141-1
RF026	Air	2	Plug	B	M-48-6B
RF027	Air	2	Plug	B	M-48-6A
RF060	---	0.75 x 1	Relief	B	M-48-6B
RH610	Motor	3	Gate	B	M-62-1/M-137-1
RH611	Motor	3	Gate	B	M-62-1/M-137-1
RH8701A,B	Motor	12	Gate	A	M-62-1/M-137-1
RH8702A,B	Motor	12	Gate	A	M-62-1/M-137-1
RH8716A,B	Motor	8	Gate	B	M-62-1/M-137-1
RH8730A,B	---	8	Check	B	M-62-1/M-137-1
RY030A,B	---	0.75x1	Relief	C	M-60-8/M-135-8
RY085A,B	---	2	Check	C	M-60-8/M-135-8
RY086A,B	---	2	Check	C	M-60-8/M-135-8
RY455A	Air	3	PORV	A	M-60-5/M-135-5
RY456	Air	3	PORV	A	M-60-5/M-135-5
RY8000A,B	Motor	3	Gate	A	M-60-5/M-135-5
RY8010A,B,C	---	6	Safety	A	M-60-5/M-135-5
RY8025	Air	0.375	Globe	B	M-60-6/M-135-6
RY8026	Air	0.375	Globe	B	M-60-6/M-135-6
RY8028	Air	3	Diaphragm	B	M-60-6/M-135-6
RY8033	Air	0.75	Diaphragm	B	M-60-6/M-135-6
RY8046	---	3	Check	B	M-60-6/M-135-6
RY8047	---	0.75	Check	B	M-60-6/M-135-6
SA032	Air	1.5	Globe	B	M-54-2
SA033	Air	1.5	Globe	B	M-54-2

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
SA181A,B,C,D	---	2	Check	C	M-54-4A,4B
SD002A-H	Air	2	Globe	B	M-48-5A/M-48-5B
SD005A-D	Air	0.375	Globe	B	M-48-5A/M-48-5B
1SD054A-H	Air	2	Globe	B	M-48-5A
2SD054B,D,F,H	Air	2	Globe	B	M-48-5B
SI101A/B	Manual	4	Gate	B	M-61-2/M-136-2
SI121A,B	---	0.75 x 1	Relief	B	M-64-4/M-136-4
SI8801A,B	Motor	4	Gate	B	M-61-2/M-136-2
SI8802A,B	Motor	4	Gate	B	M-61-3/M-136-3
SI8804B	Motor	8	Gate	B	M-61-1A/M-136-1
SI8806	Motor	8	Gate	B	M-61-1A/M-136-1
SI8807A,B	Motor	6	Gate	B	M-61-1A/M-136-1
SI8809A,B	Motor	8	Gate	B	M-61-4/M-136-4
SI8811A,B	Motor	24	Gate	B	M-61-4/M-136-4
SI8812A,B	Motor	12	Gate	B	M-61-4/M-136-4
SI8813	Motor	2	Globe	B	M-61-1B/M-136-1
SI8814	Motor	1.5	Globe	B	M-61-1A/M-136-1
SI8815	---	3	Check	A	M-61-2/M-136-2
SI8818A-D	---	6	Check	A	M-61-4/M-136-4
SI8819A-D	---	2	Check	A	M-61-3/M-136-3
SI8821A,B	Motor	4	Gate	B	M-61-3/M-136-3
SI8835	Motor	4	Gate	B	M-61-3/M-136-3
SI8840	Motor	12	Gate	B	M-61-3/M-136-3
SI8841A,B	---	8	Check	A	M-61-3/M-136-3
SI8871	Air	0.75	Globe	B	M-61-6/M-136-6
SI8880	Air	1	Globe	B	M-61-6/M-136-6
SI8888	Air	0.75	Globe	B	M-61-3/M-136-3
SI8900A-D	---	1.5	Check	A	M-61-2/M-136-2
SI8905A-D	---	2	Check	A	M-61-3/M-136-3
SI8919A,B	---	1.5	Check	B	M-61-1A/M-136-1
SI8920	Motor	1.5	Globe	B	M-61-1A/M-136-1
SI8922A,B	---	4	Check	B	M-61-1A/M-136-1
SI8924	Motor	6	Gate	B	M-61-1A/M-136-1
SI8926	---	8	Check	B	M-61-1A/M-136-1
SI8948A-D	---	10	Check	A	M-61-5,6/M-136-5,6
SI8949A-D	---	6	Check	A	M-61-3/M-136-3

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	P&ID
SI8956A-D	---	10	Check	A	M-61-5,6/M-136-5,6
SI8958A,B	---	12	Check	B	M-61-4/M-136-4
SI8964	Air	0.75	Globe	B	M-61-6/M-136-6
SI8968	---	1	Check	B	M-61-6/M-136-6
0SX007	Motor	24	Butterfly	C	M-42-2A
0SX063A,B	Motor	8	Gate	C	M-42-4
0SX146	Motor	30.00	Butterfly	C	M-42-2A
0SX147	Motor	30.00	Butterfly	C	M-42-2A
SX002A,B	---	36	Check	C	M-42-1A/1B
SX005	Motor	30.00	Butterfly	C	M-42-1A/M-42-1A
SX007	Motor	24	Butterfly	C	M-42-2B
SX016A,B	Motor	16	Butterfly	B	M-42-5A, 5B/M-126-3
SX027A,B	Motor	16	Butterfly	B	M-42-5A, 5B/M-126-3
SX112A,B	Air	12	Butterfly	C	M-42-3/M-126-1
SX114A,B	Air	12	Butterfly	C	M-42-3/M-126-1
SX147A,B	Air	16	Butterfly	C	M-42-3/M-126-1
SX150A,B	Motor	8	Butterfly	C	M-42-1A, 1B
SX168	Air	3	Globe	C	M-42-3/M-126-1
SX169A,B	Air	10	Butterfly	C	M-42-3/M-126-1
SX174	---	6	Check	C	M-42-3/M-126-1
SX178	Air	6	Gate	C	M-42-3/M-126-1
VQ003	Air	8	Butterfly	B	M-105-1/M-106-1
VQ004A,B	Air	8	Butterfly	B	M-105-1/M-106-1
VQ005A,B,C	Air	8	Butterfly	B	M-105-1/M-106-1
WM191	---	2	Check	B	M-49-1
0WO002A,B	---	6	Check	C	M-118-1
0WO028A,B	---	1.5 x 2.5	Relief	C	M-118-1
0WO205A,B	---	1	Check	C	M-118-1
WO006A,B	Motor	10	Gate	B	M-118-5/M-118-7
WO007A,B	---	10	Check	B	M-118-5/M-118-7
WO020A,B	Motor	10	Gate	B	M-118-5/M-118-7
WO056A,B	Motor	10	Gate	B	M-118-5/M-118-7
WO091A,B	---	0.75 x 1	Relief	B	M-118-5/M-118-7

TABLE 3.9-17

OPERATING TEMPERATURES FOR SUPPORT ELEMENTS

COMPONENT	SUPPORT	ELEMENT	OPERATING TEMPERATURE (°F)
Steam Generator	upper lateral	band	525 at S.G. 127 at attachment edges
		snubbers	ambient (120)
	lower lateral	inner frame	500 at support leg 125 at outer edges
		outer frame	ambient (120)
	vertical support columns	at support hinge	500
		at upper hinge and below	ambient (120)
R.C. Pump	lateral	all	300
	columns	at support lug	500
		at upper hinge	300
		below upper hinge	ambient (120)
R.P.V.	--	at Westinghouse	500
		shoe at primary shield wall concrete	ambient (120)
Pressurizer	upper	all	500
	lower	all	ambient below pressurizer skirt (120)

Table 3.9-18 been Deleted intentionally.

TABLE 3.9-19

ESSENTIAL SYSTEMS

AF	-	Auxiliary Feedwater
CC	-	Component Cooling
CS	-	Containment Spray (Except ring header and riser)
CV	-	Chemical and Volume Control
FW	-	Main Feedwater (Safety-related portion)
MS	-	Main Steam (Safety-related portion)
OG	-	Off-Gas (H_2 recombiner)
RH	-	Residual Heat Removal Pumps
SI	-	Safety Injection System
SX	-	Essential Service Water System
DG	-	Diesel Generator
RC	-	Reactor Coolant
WØ	-	Chilled Water
FC	-	Fuel Pool Cooling and Clean-up
RY	-	Reactor Coolant Pressurizer System

TABLE 3.9-20

STRESS LIMITS (AND REFERENCES) FOR PLATE AND SHELL TYPE COMPONENT SUPPORTS*
(ELASTIC ANALYSIS)

CLASSIFICATION LOADING CONDITION	CLASS 1 COMPONENT SUPPORT (CS-1)	CLASS 2 COMPONENT SUPPORT (CS-2)	CLASS 3 COMPONENT SUPPORT (CS-3)	CLASS MC COMPONENT SUPPORT (CS-MC)
DESIGN CONDITION	$P_m \leq Sm$ $P_m + P_b \leq 1.5\ S_m$ Sm FROM I-1.0 NF-3221, NF-3229, FIG. NF-3221-1 REG. GUIDE 1.130	$\sigma_1 \leq S$, $\sigma_1 + \sigma_2 \leq 1.5\ S$ $\sigma_3 \leq 0.5\ S$ (see FIG. NF-3321.1(c)-1 NF-3321.1, TABLE NF-2121(a)-1 I-7.0 OR I-8.0	SAME AS FOR CS-2 WITH S FROM TABLE I-8.0 OR I-7.0 NF-3400	SAME AS FOR CS-2 WITH S FROM TABLE I-10.0 TABLE NF-2121(a)-1
SERVICE LEVEL A (NORMAL)	$Pe \leq 3\ S_m$ $P_m + P_b + P_e + Q \leq 3\ S_m$ S_m FROM I-1.0 NF-3222, NF-3229, FIG. NF-3221-1 REG. GUIDE 1.130	SAME AS DESIGN CONDITIONS I-7.0 OR I-8.0 NF-3321.2(a)	SAME AS DESIGN CONDITIONS NF-3400	SAME AS DESIGN CONDITIONS
SERVICE LEVEL B (UPSET)	SAME AS LEVEL A LIMITS ABOVE	SAME AS DESIGN CONDITIONS I-7.0 OR I-8.0 NF-3321.2(b)	SAME AS DESIGN CONDITIONS NF-3400	SAME AS DESIGN CONDITIONS
SERVICE LEVEL C (EMERGENCY)	$P_m \leq 1.2\ S_m$ $P_m + P_b \leq 1.8\ S_m$ S_m FROM I-1.0 NF-3224, NF-3229, FIG. NF-3221-1 REG. GUIDE 1.130	σ_1 1.2 S $\sigma_1 + \sigma_2 \leq 1.8\ S$ $\sigma_3 \leq 0.5\ S$ I-7.0 OR I-8.0 NF-3321.2 (c)	SAME AS CS-2 FOR LEVEL C WITH S FROM TABLE I-8.0 OR TABLE I-7.0 NF-3400	SAME AS CS-2 WITH S FROM TABLE I-10.0

TABLE 3.9-20 (Cont'd)

CLASSIFICATION LOADING CONDITION	CLASS 1 COMPONENT SUPPORT (CS-1)	CLASS 2 COMPONENT SUPPORT (CS-2)	CLASS 3 COMPONENT SUPPORT (CS-3)	CLASS MC COMPONENT SUPPORT (CS-MC)
SERVICE LEVEL D	SAME AS FOR LEVEL D LIMITS FOR CLASS 1 COMPONENTS	$\sigma_1 \leq$ LESSER OF 1.5 S or 0.4 Su $\sigma_1 + \sigma_2 \leq$ LESSER OF 2.25 S, or 0.6 S_u.	SAME AS CS-2 LEVEL D	SAME AS CS-2 WITH S FROM TABLE I-10.0
(FAULTED)	NF-3225, TABLE F-1322.2-1 Sy FROM TABLE I-2.0 Su FROM TABLE I-3.0	I-7.0 or I-8.0, I-3.0 NF-3321.2(d)	S FROM I.80-0 OR I-7.0 NF-3400	
FATIGUE LIMITS	NB-3222.4(e) OF SECTION III			

* For linear type supports: as per NF-3230 of Section III.
For component standard supports: As per NF-3230 or NF-3320 of Section III, as applicable.

TABLE 3.9-21

LOADING COMBINATIONS FOR REACTOR INTERNALS

Normal Operation

1. Steady state - power operation
2. Steady state - shutdown operation
3. Heatup and cooldown
4. Plant loading and unloading
5. Variation in coolant flow
6. Control element drop

Abnormal Transients and Accident (Upset)

1. Operating-basis earthquake
2. Loss of power
3. Loss of flow
4. Pump rotor locking
5. Loss of load
6. Reactor overpower
7. Pump overspeed

Accident Conditions

1. Core drop
2. Steamline break
3. Reactor coolant pipe break[5]
4. Design-basis earthquake

Shipping, Handling and Refueling

The stress limits adopted for internal designs are shown in Section III, Figure N-414 of the 1968 ASME Section III Code.

[5] Based on leak-before-break analyses performed by Westinghouse, the dynamic effects associated with a large break in the main reactor coolant loop piping need not be considered.

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.1 Seismic Qualification Criteria

Byron and Braidwood have been approved to implement 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants." This regulation provides an alternative approach for establishing requirements for treatment of structures, systems, and components (SSCs) using a risk-informed method of categorizing SSCs according to their safety significance. Specifically, for SSCs categorized as low safety significant, alternate treatment requirements may be implemented rather than treatments chosen by the seismic qualification program. Refer to Section 3.2.3 for further information.

3.10.1.1 NSSS Equipment

Seismic design criteria for engineered safety features electrical equipment require that this equipment perform its safety function for both the operating-basis earthquake and the safe shutdown earthquake. Likewise, environmental criteria presented in Section 3.11 require engineered safety features equipment to perform its safety function in the environment which would be characteristic of postulated accidents. The implementation of these criteria provides a high degree of assurance that the equipment will perform its required function in different sequences of conditions. The following list identifies the instrumentation and electrical equipment which require seismic qualification:

a. pressure transmitters and differential pressure transmitters,

b. process control equipment cabinets,

c. solid-state protection system cabinets,

d. nuclear instrumentation system cabinets,

e. safeguards test racks,

f. resistance temperature detectors,

g. instrument supply inverters,

h. reactor trip switchgear,

i. power range neutron detectors,

j. incore thermocouple system,

k. main control board,

l. Class 1E equipment (BOP),

m. supporting structures - panels (BOP),

n. electrical equipment supports (BOP), and

o. cable tray supports (BOP).

The seismic qualification testing program which will be implemented for NSSS equipment supplied by Westinghouse is specified in Reference 1. According to Regulatory Guide 1.89 (Reference 10), equipment for plants in the stage of construction permit application and having an issue date for the Safety Evaluation Report after July 1, 1974, such as the Byron/Braidwood Stations, will take into account aging effects prior to seismic qualification as specified in IEEE 323-1974. Subsection 3.11.2 presents the commitment to meet IEEE 323-1974. The seismic tests conform to the procedures specified in IEEE 344-1975 which account for multiaxis and multifrequency effects of seismic excitation as well as fatigue effects caused by a number of OBE events. This commitment was satisfied by implementation of the final Staff-approved version of Reference 1.

Westinghouse has previously type tested and qualified items "a" through "h" including instrumentation included in the cabinets, to IEEE 344-1971. Item "i" is discussed in Subsection 3.10.2. Reference 2 presents the Westinghouse testing procedures used to qualify equipment by type testing. Seismic qualification testing of this equipment to IEEE 344-1971 is documented in References 3 through 9.

The main control board (item k), including the main control room control panels and remote shutdown panels, are qualified by a combination of testing and analysis. This qualification satisfies the practices recommended by IEEE 344-1975 and is documented in References 11 and 12. Items "1" through "o" are not supplied by Westinghouse.

3.10.1.2 <u>Balance-of-Plant Equipment</u>

For purposes of the following discussion, Seismic Category I electrical power equipment is synonymous with Class 1E equipment as identified in IEEE 308-1971. The loads acting on Class 1E equipment are listed in Table 8.3-1. Class 1E power components are also identified in this table.

For purposes of the following discussion, Seismic Category I electrical and electromechanical instrumentation is synonymous with Class 1E instrumentation as defined in IEEE 308-1974. This instrumentation is listed in Subsection 7.1.1.

The seismic design criteria for Seismic Category I electrical equipment and instrumentation are as follows:

a. They are designed to withstand, without the loss of nuclear safety function, safe shutdown earthquake forces and any other applicable loads transferred to the floor on which they are located.

b. Equipment possessing stationary (passive) safety functions (e.g., cable supports, instrument supports, and other components which do not perform a mechanical motion as part of their safety function) is designed to ensure the operability of safety-related equipment.

c. Equipment possessing nonstationary (active) safety functions (e.g., switches, motor-operators, and other equipment which perform a mechanical motion as part of their safety function) is designed such that the operability of the equipment is demonstrated during and after the simulated seismic event by analysis, testing, or a combination of both.

3.10.1.2.1 Cable Tray and Bus Duct Supports Criteria

The following criteria are used in the design of cable trays and bus duct supports:

a. Regardless of cable tray or bus duct function, all supports are designed to meet the requirements of Seismic Category I structures by dynamic analysis using the appropriate seismic response spectra.

b. The analytical maximum values are obtained by taking the square root of the sum of the squares of the stresses and reactions of all significant modes.

c. Cable tray loading of 45 lb/ft^2 is used throughout the design regardless of tray height or weight.

3.10.2 Seismic Analysis Testing Procedure and Restraint Measures

3.10.2.1 NSSS Compliance

Seismic Category I Westinghouse-supplied instrumentation and electrical equipment were seismically tested using sine beat inputs to each of three perpendicular axes independently applied according to the procedures of IEEE 344-1971, Section 3.2. At the time of this testing, which is reported in References 3 through 9, implementation of the IEEE 344-1971 testing method fulfilled all seismic qualification requirements. The results show that there were no electrical irregularities that would leave the plant in an unsafe condition.

In the reported tests, the equipment operated properly during and after testing with equivalent ground accelerations at zero period ranging up to 0.4g and higher. The sine beat inputs were applied not only at the equipment natural frequencies but also at many frequencies (spaced at about 1/2 octave) below 33 hertz to ensure that the equipment would function normally regardless of uncertainties of building or equipment natural frequencies. The sine beat test is severe because it excites the resonant response of the equipment thereby producing the most damaging effect to the components. This test not only excites the component to motion greater than the input but also produces fatigue damage well above that produced by seismic disturbances. This method assumes that building natural frequency coincides with that of the equipment and is as conservative as the one proposed by the NRC staff. Any possible coupling effect loses importance when compared to the excitation of components at sensitive frequencies as is done by the sine beat test. This test therefore provides more positive proof of equipment capability than the simultaneous random input test which, because of phase relationships, could result in less severe application of the seismic input.

The nuclear instrumentation system power range neutron detector has been tested in both the horizontal and vertical directions. Current, resistance, and capacitance checks were made before and after the tests. No significant changes were observed and no mechanical damage was noted. Additional tests of the power range neutron detectors were conducted using multi-frequency, multiaxial excitation as specified in Reference 1.

Equipment for a particular plant is procured on a similar basis to that which is qualified. Any equipment design changes are evaluated to determine if the changes were of a nature that could affect the results of the seismic tests. If it is determined that the changes may affect the seismic characteristics of the equipment, then the equipment is requalified for seismic integrity.

The criteria and verification procedure employed to account for the possible amplified design loads (frequency and amplitude) for Westinghouse-supplied safety-related instrumentation and electrical equipment is presented in the references of Section 3.10 (specifically Reference 2 and Appendix B of Reference 3).

3.10.2.2 Balance of Plant Compliance With IEEE 344-1971 and 344-1975

The balance of plant Class 1E equipment meets the requirement that the seismic qualification should demonstrate the capability to perform the required function during and after the safe shutdown earthquake (SSE). Both analysis and testing were used, but most equipment was qualified by testing. Analysis was used to determine the adequacy of mechanical strength (mounting bolts, etc.) after operating electrical capability was established by testing.

a. Analysis

   The balance of plant Class 1E equipment with primary mechanical safety functions (pressure boundary devices, etc.) was analyzed since the passive nature of its critical safety role usually made testing impractical. Analytical methods sanctioned by IEEE 344-1975 were utilized in such cases. Refer to Table 3.10-1 for indication of which items were qualified by analysis (for complete listing of components, see the equipment/components identified as safety related in the Engineering controlled equipment/component database(s).

b. Testing

The balance of plant Class 1E equipment having a primary active electrical safety function was tested in compliance with IEEE 344-1971 or -1975, depending on the time the test was performed. The majority of the equipment has been tested as per the 1975 issue of IEEE-344.

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

3.10.3.1 NSSS Equipment

The adequacy of Westinghouse-supplied electrical equipment supports is verified by testing conducted according to the procedures outlined in Subsection 3.10.2.

3.10.3.2 Balance of Plant Equipment

3.10.3.2.1 Design of Cable Trays

a. Seismic Qualification Criteria

1. Applicable codes, standards, and specifications

   a) AISI "Specification for Design of Cold-Formed Steel Structural Members," 1968 Edition.

2. Loads

   a) dead load D: 45 psf including the weights of cables and cable trays;

   b) live load L: 200 pounds at any location between the two supports, during construction;

   c) severe environmental load E: peak acceleration of 4% critical damping floor spectra generated by the operating-basis earthquake; and

   d) extreme environmental load E': peak acceleration of 7% critical damping floor spectra generated by safe shutdown earthquake.

3. Load combinations

    a) D + L,

    b) D + E, and

    c) D + E'.

b. Method of Analysis and Design

The equivalent static load corresponding to the peak acceleration of the floor spectra is used for analysis and design.

c. Procedures of Analysis and Design

1. Calculate the allowable span of the cable tray subjected to dead load and live load; use 1.0 times the allowable stresses.

2. Calculate the allowable span of the cable tray subjected to dead load, vertical seismic load, and horizontal seismic load; use 1.6 times the allowable stresses, with minimum factor of safety equal to 1.05.

3. The stresses due to vertical and horizontal seismic load are combined by using SRSS (square root of the sum of the squares) method.

4. The stresses due to dead load and seismic load are combined linearly.

5. Computer program utilized: SEISHANG - Seismic Analysis of Hangers (see Appendix D).

3.10.3.2.2 <u>Design of Cable Tray, Nonsegregated Bus Duct, and Conduit Supports</u>

a. Seismic Qualification Criteria

1. Applicable codes, standards, and specifications

    a) AISI "Specification for Design of Cold-Formed Steel Structural Members," and

    b) AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," 1969 Edition.

    c) AWS Dl.l "Structural Welding Code."

    Clarifications to, and deviations from portions of AWS Dl.l are made based on engineering evaluations. Visual weld inspection requirements are based on guidelines in a document prepared by the Nuclear Construction Issues Group, NCIG-01, Revision 2, "Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants."

2. Loads

   a) Dead load D: loads including the weights of the cables, cable trays, conduits, nonsegregated bus ducts, and supports.

   b) Severe environmental load E: operating-basis earthquake response spectra of 4% critical damping at the floor where the support is erected.

   c) Extreme environmental load E': safe shutdown earthquake response spectra of 7% critical damping at the floor where the support is erected.

3. Load combinations

   a) D

   b) D + E

   c) D + E'

b. Method of Analysis

   1. Response Spectrum Method

c. Procedures of Analysis and Design

   1. The masses are lumped at nodes.

   2. For cable tray and bus duct supports, the following applies:

      a) For standard supports, three structural modes, i.e., one from each direction, are used for dynamic analysis.

      b) For special supports, all the structural modes up to a frequency of 33 hertz are considered.

3. For conduit supports, the following applies:

   a) For trapeze type supports, item 2 applies.

   b) For other types of supports such as cantilevers, direct-mounted types, etc., the peak acceleration of the floor spectra is used in the manual analysis.

4. Seismic excitation from vertical and two horizontal directions is considered. The stresses due to seismic loads from different directions are combined by the SRSS (square root of the sum of the squares) method.

5. The stresses due to dead loads and seismic loads are combined linearly.

6. The allowable stresses for dead load only are 1.0 times the allowable. The allowable stress for dead load plus seismic load is 1.6 times the allowable stresses, with minimum factor of safety equal to 1.05.

7. The allowable slenderness ratios for compression members are given in Appendix D, Table D-45.

8. Computer programs utilized:

   a) SEISHANG, Seismic Analysis of Hangers (see Appendix D) for supports, and

   b) PIPSYS, Integrated Piping Analysis System (see Appendix D) for supports.

3.10.3.3 NSSS - Testing/Analysis

The adequacy of Westinghouse-supplied electrical equipment supports is verified by testing conducted according to the procedures outlined in Subsection 3.10.2.

3.10.4 Operating License Review

The results of the tests and analyses to demonstrate adequate seismic qualification and implementation of proper criteria are presented in Subsection 3.10.2.

3.10.5 References

1. "Environmental Qualification of Westinghouse Class 1E Equipment," WCAP-8587, Rev. 1, September 1977.

2. A. Morrone, "Seismic Vibration Testing with Sine Beats," WCAP-7558, October 1971.

3. E. L. Vogeding, "Seismic Testing of Electrical and Control Equipment," WCAP-7817 (Non-Proprietary), December 1971.

4. E. L. Vogeding, "Seismic Testing of Electrical and Control Equipment (WCID Process Control of Equipment)" WCAP-7817, Supplement 1, December 1971.

5. L. M. Potochnik, "Seismic Testing of Electrical and Control Equipment (Low Seismic Plants)," WCAP-7817, Supplement 2, December 1971.

6. E. L. Vogeding, "Seismic Testing of Electrical and Control Equipment (Westinghouse Solid State Protection System) (Low Seismic Plants)," WCAP-7817, Supplement 3, December 1971.

7. J. B. Reid, "Seismic Testing of Electrical and Control Equipment (WCID NUCANA 7300 Series) (Low Seismic Plants)," WCAP-7817, Supplement 4, November 1972.

8. E. L. Vogeding, "Seismic Testing of Electrical and Control Equipment (Instrument Bus Distribution Panel)," WCAP-7817, Supplement 5, March 1974.

9. E. K. Figenbaum and E. L. Vogeding, "Seismic Testing of Electrical and Control Equipment (Type DB Reactor Trip Switchgear)," WCAP-7817, Supplement 6, August 1974.

10. U.S. Nuclear Regulatory Commission, "Qualification of Class 1E Equipment for Nuclear Power Plants," Regulatory Guide 1.89.

11. B. F. Maurer, "Seismic Qualification of the Byron/Braidwood Main Control Board," WCAP-10393 (Proprietary Class 2), August 1983.

12. B. F. Maurer, "Seismic Qualification of the Byron/Braidwood Main Control Room Control Panel and Remote Shutdown Panels," WCAP-10412 (Proprietary Class 2), October 1983.

TABLE 3.10-1

STATUS OF QUALIFICATION (IEEE-344) DOCUMENTS FOR ELECTRICAL COMPONENTS

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION LEVEL	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICATION LEVEL	SEISMIC QUALIFICATION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICATION LEVEL	SEISMIC QUALIFICATION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICATION LEVEL	SEISMIC QUALIFICATION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICATION LEVEL	SEISMIC QUALIFICATION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA- TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICATION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICATION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICA-TION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICATION LEVEL	SEISMIC QUALIFICATION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICATION LEVEL	SEISMIC QUALIFICATION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICATION LEVEL	SEISMIC QUALIFICATION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

TABLE 3.10-1 (Cont'd)

SPECIFICATION	EQUIPMENT[8]	INCLUDES CLASS 1E ELEC. EQUIP. (YES/NO)	VENDOR	LOCATION	SEISMIC QUALIFICATION LEVEL	SEISMIC QUALIFICATION METHOD	STATUS[4][5]

Security - Related Information Figure Withheld Under 10 CFR 2.390

NOTES

1. Where "varies" is stated for location, equipment is located in several areas in plant. In these cases, maximum levels are given to envelope all such areas.

2. Mounted in piping - not mounted rigidly to structure; qualification levels are based on the maximum response of the pipe or duct.

3. The equipment under this group is non-safety-related. (See the equipment/component classifications in the Engineering controlled equipment/component database(s).

4. The seismic qualification can be verified through calculations that are related to equipment/components in the Engineering controlled equipment/component database(s).

5. Completed as of May 16, 1990.

6. Equipment is Seismic Category 1 but it does not include Class 1E electrical equipment power. The seismic qualification can be verified through calculations that are related to equipment/components in the Engineering controlled equipment/component database(s).

7. Power and control cables are not required to be seismically tested for IEEE Standard - 383.

8. Electrical portion only.

9. The cubicle cooler units contain both non-Class 1E cooling coils and also Class 1E fan-motor assemblies. The cubicle cooler units were seismically qualified, excluding fan-motor assemblies. The fan-motor assemblies were seismically qualified independently and are listed separately in Table 3.10-1.

10. Containment charcoal filter units at Braidwood Station have been abandoned in place.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

The mechanical, instrumentation, and electrical portions of the engineered safety features and the reactor protection system are designed to ensure acceptable performance in all environments anticipated under normal, test, and design-basis accident conditions. This section presents information on the design-basis and qualification verifications for mechanical, instrumentation, and electrical equipment in the engineered safety features and the reactor protection system. Mechanical and electrical components have been identified and classified relative to their safety classification in Section 3.2. Section 3.7 presents the seismic design requirements and Section 3.10 presents the seismic qualification of electrical equipment. Mechanical equipment design basis considerations are described in Chapters 5.0, 6.0, 9.0, and 10.0.

Byron and Braidwood have been approved to implement 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants." This regulation provides an alternative approach for establishing requirements for treatment of structures, systems, and components (SSCs) using a risk-informed method of categorizing SSCs according to their safety significance. Specifically, for SSCs categorized as low safety significant, alternate treatment requirements may be implemented rather than treatments chosen by the environmental qualification program. Refer to Section 3.2.3 for further information.

3.11.1 Equipment Identification and Environmental Conditions

The safety-related equipment which can operate during all relevant plant conditions is included in the program for Environmental Qualification.

The normal and accident environmental conditions for various zones throughout the plant are tabulated in Table 3.11-2. Figure 3.11-1 shows these zones on plant general arrangements.

The definition of the terms, normal, abnormal, and accident environmental conditions which are used in Table 3.11-2 are as follows:

a. Normal - Those conditions that are expected to occur regularly and for which plant equipment is expected to perform its function, as required, on a continuous steady-state basis. NOTE: Abnormal conditions have been made a subset of normal conditions. This is the operating range that is expected in certain areas for short-term transients and is defined as follows:

   1. Only at Braidwood Station, for areas served by the control room ventilation system (VC) (Zone areas A1, A2, A2a, and A5), the abnormal condition is expected to occur when there is a chlorine release accident outside the station.

2. For areas served by the auxiliary building ventilation system (VA), the abnormal condition is expected to occur during a two-hour delay in powering the VA supply and exhaust fans upon Loss of Offsite Power (LOOP) coincident with LOCA.

3. For the ESF switchgear rooms, the diesel generator rooms, and the miscellaneous electrical equipment and battery rooms (zone areas A3 and A6), the abnormal condition is expected to occur when a high energy line break in the turbine building causes the HELB dampers to close and briefly interrupts HVAC operation.

b. Accident - Those conditions which may occur during plant operation and which constitute a harsh environment that results from component failure or external event such as LOCA, MSLB, and HELB.

All Class lE equipment supplied by Westinghouse was environmentally qualified according to the environmental qualification program described in Reference 1.

Systems essential to the safe shutdown of the plant are identified in Tables 3.6-1 and 3.6-3. All equipment and materials used in these systems are compatible with both their normal and abnormal environment to the extent that their essential function is not impaired. The environmental conditions and design basis for the environmental zones are shown on Table 3.11-2.

A harsh environment is defined as any area which will experience a significant change in one or more of the environmental parameters as a result of an accident. The parameters that are considered are temperature, pressure, humidity, caustic spray, radiation, and submergence.

Harsh environments also include areas which are exposed to an abnormally high temperature, pressure, humidity, and/or total integrated radiation dose (TID) of greater than 10^4 rads during normal plant operation.

10 CFR 50.49 defines a mild environment as an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences. The 10 CFR 50.49 requirements for environmental qualification of electrical equipment important to safety do not apply to equipment located in a mild environment.

The EQ Binders present a discussion of all equipment located in an environmental zone with a TID greater than 10^4 rads. None of the EQ equipment in harsh environmental zones has been exempted from radiation qualification.

Qualification of all Class 1E equipment with solid state components was reviewed according to the program described in Reference 4.

3.11.2 Qualification Tests and Analyses

This subsection outlines the Exelon Generation Company's commitment to meet the requirements of IEEE 323-1974 for both NSSS and non-NSSS equipment. Further clarification is provided in Commonwealth Edison Company's response (March 3, 1978) to S. A. Varga's letter to R. L. Bolger, dated October 4, 1977 (NRC Docket Numbers 50-454 and 50-455).

For equipment that is required to operate in a harsh environment, the preferred test sequence is that recommended by IEEE 323-1974. Any deviation from the IEEE 323-1974 recommended sequence is justified in the EQ binders. Test sequences are described and justified in the EQ Binders.

NSSS

For Westinghouse NSSS harsh Class lE equipment, Westinghouse meets IEEE 323-1974, including the Nuclear Power Engineering Committee (NPEC) Position Statement of July 24, 1975, (IEEE 323a-1975), by an appropriate combination of any or all of the following type testing, operating experience, and qualification by analysis. This commitment was satisfied by implementation of the final Staff approved version of Reference 1.

In the overall Class 1E Westinghouse equipment qualification program, generic enveloping environmental conditions, e.g., (when applicable) temperature, pressure, humidity, chemistry, radiation were established for the various pieces of Westinghouse supplied Class 1E equipment. These conditions vary according to location of the equipment. The environmental conditions for which the equipment is qualified are reported in the equipment qualification data package (Reference 2).

Westinghouse NSSS motors outside containment comply with the qualification control requirements of Criterion III to Appendix B of 10 CFR 50. These requirements are satisfied by qualification in accordance with IEEE 323-1974 as described in WCAP 8587 and its supplement, which contains appropriate EQDPs (Equipment Qualification Data Packages) for Westinghouse supplied continuous duty motors.

How the requirements of the General Design Criteria (GDC) 1, 4, 23, and 50 are met is addressed in Section 3.1. Specific information concerning how GDC 1 and 4 are met is reported in Appendix A of Reference 1. Specific information concerning how GDC 23 is met can be found in Chapter 7.0. Information concerning how Appendix B of 10 CFR 50 is met is located in Quality Assurance Topical Report NO-AA-10. Applicable Regulatory Guides are addressed in Appendix A.

Balance of Plant

The environmental qualification parameters shown on Table 3.11-2 indicate the worst environmental zone to which the balance of plant equipment was qualified. The seismic qualification level that the equipment was qualified to meet is shown on Table 3.10-1.

Generally, the qualification of harsh Class lE equipment is in accordance with the requirements of IEEE 323-1974. Where specific equipment qualification standards are available, they have also been complied with.

Qualification programs, when necessary, were developed and the tests were performed in accordance with these programs. A description of the qualification tests and analyses that have been performed are contained in the Byron/ Braidwood EQ Binders.

The Binders summarize the environmental conditions being used in the qualification of equipment and the status of that qualification work on each component which could be subjected to a harsh environment.

3.11.3 Qualification Test Results

The results of qualification tests have been documented in the Environmental Qualification Binders.

3.11.4 Loss of Ventilation

3.11.4.1 Control Room and Auxiliary Electric Equipment Rooms

The control room and the auxiliary electric equipment rooms are served by the control room HVAC system which is described in detail in Section 6.4 and Subsection 9.4.1. The control room HVAC system is an engineered safety feature system and is designed to perform its intended function under all normal and abnormal plant operating conditions.

The control room HVAC system consists of two 100% capacity equipment trains each consisting of supply air fan, cooling coils, heating coils, return air fan, refrigeration unit, chilled water pump and associated piping, valves, ductwork, and filters. A failure of any equipment component of the control room HVAC system is audibly and visibly annunciated on the main control board. The operator can remote manually start the standby system from the main control board.

No single failure results in the loss of control room air conditioning. Operability of the safety-related control and electrical equipment located in this area will not be impaired.

All harsh Class lE equipment associated with the control room HVAC system is environmentally qualified. Refer to applicable EQ Binders.

3.11.4.2 ESF Switchgear, Miscellaneous Electrical Equipment, and Batteries

The redundant ESF switchgear, Class 1E batteries, and associated electrical equipment are located in separate rooms in the auxiliary building. Each room is served by a separate engineered safety feature ventilation system as described in Subsection 9.4.5.

The Seismic Category I (Class 1E) equipment provided for each ventilation system satisfies IEEE 279-1971 and IEEE 308-1971 design criteria. All harsh Class lE equipment associated with the ventilation systems for these areas is environmentally qualified as shown in applicable EQ Binders.

Each division of ESF switchgear is provided with supply fans and filters to maintain an inside ambient temperature of less than or equal to 108°F, compatible with equipment requirements.

Backfdraft dampers and fire dampers are installed in the ventilation openings penetrating the fire wall between the ESF switchgear rooms and the Turbine Building. During the unlikely event of a high energy line break on Turbine Building elevation 426 feet, the pressures will cause the spring loaded open backdraft dampers to close. Because of the proximity of the rooms, the backdraft dampers for both ESF rooms may close. The ventilation fans will trip due to high differential pressure causing a temporary loss of ventilation in the ESF rooms. After a time delay the fans will restart and establish ventilation flow to the rooms. A mild room environment will be maintained and the equipment required for safe shutdown located in these rooms will still operate and perform its safety related function.

Similarly, backdraft dampers and fire dampers are installed in the ventilation opening penetrating the fire wall between the miscellaneous electrical equipment rooms and the Turbine Building. During the unlikely event of a high energy line break on Turbine Building elevation 451 feet, the pressures will cause the spring loaded open backdraft dampers to close. Because of the proximity of the rooms, the backdraft dampers for both ESF rooms may close. The ventilation fans will trip due to differential pressure causing a loss of ventilation in the ESF rooms. After a time delay the fans will restart and establish ventilation flow to the rooms. A mild room environment will be maintained and the equipment required for safe shutdown located in these rooms will still operate and perform its safety related function.

3.11.4.3 Diesel Generator

The redundant diesel generators and associated equipment are located in separate rooms in the auxiliary building. Each room is served by a separate independent engineered safety feature ventilation system as described in Subsection 9.4.5.

The Seismic Category I (Class 1E) electrical equipment provided for each system satisfies IEEE 279-1971 and IEEE 308-1971 design criteria. All harsh Class 1E equipment associated with the ventilation systems for these areas is environmentally qualified as shown in applicable EQ Binders.

In the event of a single failure to a diesel-generator ventilation system, the safety-related function is performed by the redundant ventilation system and diesel generator, and safe shutdown of the reactor is not affected.

The diesel generator room supply fans exhaust through backdraft dampers and fire dampers provided in the ventilation opening between the diesel generator rooms and the Turbine Building. During the unlikely event of a high energy line break on turbine building elevation 401 feet, the pressures will cause the spring loaded open backdraft dampers to close. Because of the proximity of the rooms, the backdraft dampers for both diesel generator room supply fans may close. The ventilation fans will trip due to high differential pressure causing a temporary loss of ventilation in the diesel generator rooms. After a time delay the fans will restart and establish ventilation flow to the rooms. A mild room environment will be maintained and the equipment required for safe shutdown located in these rooms will still operate and perform its safety related function.

3.11.4.4 Auxiliary Building

The redundant ESF pumps which are required for safe shutdown of the reactor are located in separate rooms or areas of the auxiliary building. This equipment is served by the auxiliary building HVAC system and is further described in Subsection 9.4.5.

The following redundant equipment is served by a cubicle cooler consisting of a cooling coil and fans:

a. essential service water pumps,

b. residual heat removal pumps,

c. safety injection pumps,

d. containment spray pumps,

e. centrifugal charging pump, and

f. auxiliary feedwater pumps (diesel-driven).

The component cooling pumps and motor-driven auxiliary feedwater pumps are located in general areas of the auxiliary building and are served by the auxiliary building HVAC system.

The auxiliary building HVAC system consists of four supply fans and four exhaust fans; two supply and two exhaust fans are operating during normal plant operation. During the unlikely event of having a LOCA coincident with a loss of offsite power (LOOP) in one unit, the supply and exhaust fans powered from that unit are tripped and charcoal booster fans started. The supply fan and exhaust fan associated with the second unit continue to operate. Two hours after the above event, the supply and exhaust fans associated with the LOCA/LOOP unit may be manually restarted at the operator's discretion.

All harsh Class 1E equipment associated with this equipment is environmentally qualified as shown in applicable EQ Binders. All Class 1E electrical equipment associated with the cubicle coolers and auxiliary building HVAC system satisfies IEEE 279-1971 and IEEE 308-1971 design criteria.

A failure of any equipment component is audibly and visibly annunciated in the main control room. The operator can manually start and stop the standby system from the main control room.

In the event of a single failure of one of the cubicle coolers, the safety-related function is performed by the other ESF division pump and cooler, and safe shutdown of the reactor is not affected.

In the event of a single failure of an auxiliary building HVAC system supply or exhaust fan, the safety-related function is performed by one of the other ESF division supply or exhaust fans and safe shutdown of the reactor is not affected.

3.11.4.5 Containment

The equipment located within the containment which is required to safely shut down the reactor is served by the reactor containment fan cooler (RCFC) system which is described in detail in Subsections 6.2.2 and 9.4.8.

The 100% redundant RCFC units are designed to remove the heat generated by a loss-of-coolant accident. The electrical equipment associated with the RCFC units are Class 1E and are environmentally qualified to the postaccident environment as shown in applicable EQ Binders.

A single failure of the RCFC units will not impair safe shutdown of the reactor since 100% redundancy has been provided.

3.11.5 Estimated Chemical and Radiation Environment

The environmental qualification program follows the methodology established by the NSSS supplier (Reference 5). The methodology has been reviewed and approved by the NRC (Reference 6). The methodology is based on the IEEE-323-1974 Standard (Reference 7), which describes testing conditions for qualifying equipment under postaccident conditions that result in the equipment being exposed to a water spray. The environmental qualification program for Class 1E equipment located within the containment considers the water chemistry environment resulting from a LOCA. This condition is considered to be the most extreme as compared to a HELB (secondary system feedwater line or a steamline break) within containment. Class 1E equipment is qualified to conditions that expose the equipment to a water spray containing a boron concentration of 2,500 ppm and a solution pH of 10.5 resulting from the addition of sodium hydroxide (NaOH) to the solution.

During the period when containment spray is operating and NaOH is being educted, the spray pH may exceed the upper EQ limit of 10.5. An evaluation has been performed to verify that Class 1E equipment will not be degraded under these conditions. During operation with the containment spray pump suction from the recirculation sump without NaOH addition, the spray pH will be the same as the recirculation sump pH (8.0 - 10.5).

The hydrogen generation as a result of core radiolysis, radiolysis of sump water, and the control of hydrogen buildup within the containment are discussed in Subsection 6.2.5.

The organic materials inside the containment and the material released due to the radiation exposure are discussed in Subsection 6.1.2.

The design radiation environment for each area of the plant is listed in Table 3.11-2, for both normal operation and design-basis accident conditions.

The values listed represent gamma plus beta integrated doses. Safety-related equipment (Class 1E) which can withstand the specified radiation levels is chosen based on its location in the plant.

Specifications for the equipment purchased state that a gamma irradiation test to the specified radiation level is acceptable for meeting the specified radiation level beta plus gamma requirement.

The calculation of the design-basis accident is presented in Subsection 15.6.5. All source data and assumptions are presented there.

Source terms and chemical environments for which the NSSS scope equipment was originally qualified to are described in Appendix A to Reference 1.

3.11.6 Operability Requirements

Specific postaccident operability requirements for each device are developed from the guidelines which are given in Table 3.11-4.

The operability requirement for each piece of Class 1E equipment is the length of time the equipment is required to remain functional during accident mitigation. A margin of at least 1 hour of equipment operating time has been included in the qualification program for each piece of applicable Class 1E equipment.

Some equipment, e.g. transmitters, was not specified to maintain trip function accuracy requirements for longer than 5 minutes after an accident. However, peak HELB temperatures will be reached within the specified operability time. This operability time was conservatively established based on the reactor trip engineered safeguards function performed by each equipment item, considering the consequences failure of the device would have on the operator and on the mitigation of the event. Margins for trip function requirements are contained in the HELB envelopes which encompass a full spectrum of break sizes and are also justified by the fact that the signal generated by the sensor is "locked-in" by the protection system and will not reset should the sensor fail after the designated trip time requirement. Most equipment was also specified and qualified for much longer postaccident monitoring function times to slightly reduce accuracy requirements.

3.11.7 Detection of Age-Related Degradation in Equipment

The effects of aging on the Class 1E equipment were addressed for all pieces of equipment identified in the EEQR. Aging was addressed either by performing accelerated aging on the equipment or by developing an aging analysis program to evaluate the stresses imposed on the equipment which degrade performance. The objective of an aging analysis is to determine the qualified life of the equipment. An examination is performed to determine which of the materials are susceptible to aging by heat (thermal), radiation, or both heat and radiation, and then determine the qualified life for the most susceptible material. Arrhenius techniques were utilized in the determination of qualified life. The qualified life for the most susceptible material/component is used to establish a periodic replacement schedule if the qualified life is less than 40 years.

For NSSS Class 1E equipment the aging evaluation program is described in Appendix B to WCAP-8587. Accelerated thermal aging parameters are described in Appendix D to WCAP-8587.

3.11.8 Detection of Age-Related Degradation in Equipment in Harsh Environments

Equipment located in harsh environments is also qualified to address potential age-related degradation. Accelerated aging techniques (Arrhenius principle) are used to simulate age. Based on this data, components with limited life are then maintained or replaced through an Equipment Qualification Maintenance and Surveillance Program implemented at Byron/Braidwood stations. Data for this program is derived and evaluated from EQ Binders and manufacturer's recommendations. Known low dose rate effects and synergisms are included in the environmental qualification program. Additional existing programs supplement this program. These programs are:

a. Technical specification requirements which verify through performance tests that equipment is functional;

b. Vibration monitoring is used to do comparative testing against established baselines on rotating equipment;

c. Lubrication Program;

d. Instrument Calibration/Surveillance Program;

e. Inservice testing program on pumps and valves and inservice inspection program per Section XI of ASME Boiler and Pressure Vessel Code;

f. A history/trending program is applied to identify equipment concerns.

3.11.9 Description of Qualified Cable

Safety-related cables installed during construction and the first few years of operation for the balance-of-plant systems in the harsh zones are as follows:

a. Power and Control Cable:

Type of Cable	Cat. No.	Manufacturer
EPR/HYP	Okolon	Okonite

b. Instrumentation Cable:

Type of Cable	Cat. No.	Manufacturer
EPDM/HYP	-	Samuel Moore

Future safety-related cable purchases may be from different cable manufacturers and may be constructed from various jacket and insulation types.

Hypalon-chlorosulfonated polyethylene is used to construct only the jacket of the Okonite and Samuel Moore cables. The cable as supplied by Okonite and Samuel Moore has been qualified by each of the respective manufacturers to the following standards:

a. IEEE 383-1974 and

b. IEEE 323-1974.

Future safety-related cable purchased for the balance-of-plant systems are qualified to the above standards.

3.11.10 High Energy Line Break (HELB)

3.11.10.1 Auxiliary Building

High energy line breaks in the auxiliary building have been identified and analyzed in accordance with Section 3.6. High energy lines are defined as pipes in which the fluid temperature exceeds 200°F or the pressure exceeds 275 psig during normal plant operation. Breaks are postulated in these lines. The resulting temperature, pressure, and humidity conditions are included in the environmental qualification program. The potential for pipe whip and jet impingement effects has been investigated and additional protective features incorporated where required.

Section 3.6 of the UFSAR describes the approach used to evaluate high energy line break effects. The results of the subcompartment analyses are included in Attachment A3.6 of the UFSAR. The high temperatures and pressures predicted in the subcompartments are not an equipment qualification concern because the object of compartmentalization of safety equipment was to ensure that a high energy line failure will not result in additional failures which would violate the plant design basis. The plant is designed such that capability to safely shut down is maintained following an initiating event and the resulting failures, plus an independent single active failure. To verify that this design approach has been successful, the high energy line break conditions are included as accident conditions for the applicable environmental zones in Table 3.11-2 and equipment in these zones are qualified to the accident conditions if they are required to function in the accident scenario. For environmental zones in which the conditions are not affected by high energy line breaks, the accident conditions are shown as “NA” (not applicable) in Table 3.11-2.

The only areas identified as experiencing elevated temperatures and/or pressures beyond normal or abnormal conditions following a high energy line break are, with one exception, subcompartments. The subcompartments have been designed such that failure of a high energy line in the subcompartment will not result in failures beyond the single train of a safety system which is in the subcompartment. The resulting harsh environment affects only equipment in the failed train. All safety equipment used to mitigate the break is unaffected by the harsh environment. As a result, the plant design basis is valid in spite of harsh environments caused by high energy line breaks and no equipment must be qualified for the harsh environments which result from high energy line breaks in auxiliary building subcompartments.

The only area other than the subcompartments which could experience an elevated temperature is the upper area of Zone A13c, the containment piping penetration area. A break in a 3-inch letdown line in the chemical and volume control system could release steam into this area which has no natural ventilation. The temperature would then increase above the environment specified for this area. The only safety-related items in this area are isolation valves on the safety injection and essential service water systems which are required to function in this accident and are redundant, and an isolation valve on the failed line which fails as is. The break flow is limited to 120 gpm by orifices. Immediate indications of the break will be supplied by two main control board alarms (high flow and high letdown heat exchanger outlet temperature). The plant can be safely shut down without the equipment which would be affected by the increased temperature.

3.11.10.2 MS Tunnel and Safety Valve Enclosures

As stated earlier, equipment in zones affected by the analyzed HELBs is qualified to the applicable accident conditions if required to function in the accident scenario. Qualification of equipment in Zone T3 involves the time dependence of the accident conditions, as well as operator action. Zone T3 includes the main steam pipe tunnel and the safety valve enclosures.

Evaluations for MSLB outside containment consider operator action (i.e., to terminate auxiliary feedwater flow to the faulted SG) at 20 minutes after reactor trip. The evaluated maximum safe shutdown temperature of 413.5 °F at the time of main steam isolation is well within the qualification envelope for safe shutdown equipment in Zone T3. Furthermore, equipment necessary for postaccident monitoring, with the exception of the main steam line radiation monitors and penetration area radiation monitors, has been qualified to the worst-case break temperature of 518.4 °F.

The main steamline radiation monitors and the penetration area radiation monitors serve no safe shutdown function for this accident. Their function is to detect a steam generator tube rupture event, which does not cause a harsh environment with respect to temperature excursion in Environmental Zone T3. Also, these radiation monitors do not provide a postaccident monitoring function for a HELB in Zone T3. Emergency operating procedures direct the operators to verify that radiation levels are below the alert alarm setpoint values for nonisolated steam generators in order to verify isolation of the faulted steam generator. This information is available by sampling.

3.11.10.3 Turbine Building

The Turbine Building contains no safety-related components or other components required for safe shutdown of the Unit. However, there are adjacent rooms in the Auxiliary Building that contain such equipment and that communicate with the Turbine Building through ventilation openings. Therefore, the equipment in those adjacent rooms must be protected from or shown to be able to withstand the effects of HELB in the Turbine Building. The HELB mitigation strategy for the Auxiliary Building rooms involves (1) keeping the Turbine Building environment out of the Auxiliary Building rooms by means of HELB backdraft dampers; (2) configuring the fire dampers to close only in the event of a fire (thereby keeping them open during the HELB to allow the room ventilation exhaust path to remain open); and (3) automatically restoring room cooling (by installing auto-restart capability for the room ventilation fans).

Section 3.6 of the UFSAR describes the approach used to evaluate the effects of high energy line breaks, including Turbine Building HELB. The resultant environmental conditions in the adjacent Auxiliary Building rooms have been determined per Reference 9. Due to the limited magnitude and short duration of the transient, the environmental parameters within these zones would not be significantly more severe than the environment that would occur during normal plant operation.

3.11.11 References

1. Westinghouse Staff, "Environmental Qualification of Westinghouse Class 1E Equipment," WCAP-8587, Rev. 2, March 5, 1979.

2. Westinghouse Staff, "Equipment Qualification Data Packages," Supplement 1, Rev. 1, to WCAP-8587, November 15, 1978.

3. Reference deleted.

4. Reference deleted.

5. Butterworth, G., and Miller, R. B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," WCAP-8587, Revision 6-A, Westinghouse Electric Corporation, March 1983.

6. Letter to E. P. Rahe, Jr., Manager, Nuclear Safety Department, Westinghouse Electric Corporation, from C. O. Thomas, Chief Standardization & Special Projects Branch, Division of Licensing, NRC, "Acceptance for Referencing of Licensing Topical Reports WCAP-8587, Revision 6 (NP), 'Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment,' and WCAP-9714 (P), 9750 (NP), 'Methodology for the Seismic Qualification of Westinghouse WRD Supplied Equipment,'" November 10, 1983.

7. IEEE Standard 323-1974 "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," February 28, 1974.

8. Exelon Design Analysis No. BRW-01-0153-E/BYR01-068, "Environmental Parameters of EQ Zones."

9. Exelon Design Analysis No. BRW-12-0084-M/BYR12-070, "Auxiliary Building Environment Following a High Energy Line Break in the Turbine Building."

TABLE 3.11-1

STATUS OF QUALIFICATION (IEEE-323) DOCUMENTS

FOR ELECTRICAL COMPONENTS

Refer to the Controlled Documents module of Passport.

TABLE 3.11-2

PLANT ENVIRONMENTAL CONDITIONS

ENVIRON-MENTAL ZONE	AREA	ENVIRONMENTAL CONDITIONS	TEMPERATURE (°F) (Note 1)	RELATIVE HUMIDITY (%)	PRESSURE IN WATER GAUGE (unless other-wise noted)	MAXIMUM INTEGRATED EXPOSURE (Note 2) (rad-carbon)
	AUXILIARY BUILDING					
A1	Control Room, Aux. Elec. Equip. Rooms, Record and Storage Secondary Sec. Office, Kitch/Locker Room, Tech. Support Center	Normal (Note 4) Accident	75±2 NA	20-60 NA	0.125 NA	$1.1x10^{2}$ 10^{3}
A2	Control Room HVAC Equipment Rooms	Normal (Note 4) Accident	Max 104 NA	20-60 NA	0.125 NA	$1.1x10^{2}$ 10^{3}
A2a	Control Room Makeup Filter	Normal (Note 4) Accident	80±10 NA	20-60 NA	0.125 NA	$1.1x10^{3}$ 10^{4}
A3	Misc. Elec. Equip. and Battery Room, ESF Switch-gear Rooms, Div. 12 and 22 Cable Spreading Areas	Normal(Note 5) Accident	Max 108 NA	8-70 NA	-0.5 to 0.5 NA	$1.4x10^{3}$ 10^{4}
A4	Non-Essential Switchgear Rooms	Normal Accident	Max 109 NA	8-70 NA	0.00 to 0.25 NA	$2.1x10^{3}$ 10^{4}
A4a	Reactor Containment Chiller Room, Radwaste HVAC Equipment Rooms	Normal Accident	Max 108 NA	8-70 NA	-0.25 to 0.0 NA	$2.1x10^{3}$ 10^{4}

TABLE 3.11-2 (Cont'd)

ENVIRON-MENTAL ZONE	AREA	ENVIRONMENTAL CONDITIONS	TEMPERATURE (°F) (Note 1)	RELATIVE HUMIDITY (%)	PRESSURE IN WATER GAUGE (unless other-wise noted)	MAXIMUM INTEGRATED EXPOSURE (Note 2) (rad-carbon)
A5	Upper Cable Spreading Rooms	Normal (Note 4) Accident	Max 90 NA	8-70 NA	0.125 NA	$2.1x10^3$ 10^4
A5a	Lower Cable Spread Rooms	Normal Accident	Max 108 NA	8-70 NA	0.0 NA	$2.1x10^3$ 10^4
A6	Diesel-Generator Rooms, Diesel Oil Storage Rooms	Normal (Note 6) Accident	Max 132 NA	8-70 NA	-0.5 to 0.5 NA	$2.1x10^3$ 10^4
A7	Laboratory, Counting Room, Rad. Offices, Radwaste and Remote Shutdown Control Room, Computer Rooms, Sample Room	Normal Accident	75±2 NA	40±5 NA	-0.25 to 0.25 NA	$2.1x10^3$ 10^4
A7a	Laundry Room and Laboratory HVAC Equipment Room	Normal Accident	Max 90 NA	40±5 NA	-0.25 to 0.25 NA	$2.1x10^3$ 10^4
A8	Auxiliary Bldg. General (Accessible) Areas (except as noted)	Normal (Note 14) Accident	Max 122 NA	8-70 NA	-0.25 to 0.0 NA	$2.1x10^3$ 10^4
A8	Fuel Handling Building	Normal (Note 14) Accident	Max 122 NA	8-95 NA	-0.25 to 0.0 NA	$2.1x10^3$ 10^4

TABLE 3.11-2 (Cont'd)

A8a	Radwaste Evaporator Monitor Tank Cubicle	Normal	Max 122	8-70	-0.25 to 0.0	$2.1x10^3$
		Accident	NA	NA	NA	10^4
A8b	Permeate Storage Tank Room, Laundry Drain Tank Room, Decontamination Change Area, Decontamination Pad and Machine Room	Normal (Note 14)	Max 125.1	8-70	-0.25 to 0.0	$2.1x10^3$
		Accident	NA	NA	NA	10^4

TABLE 3.11-2 (Cont'd)

ENVIRON-MENTAL ZONE	AREA	ENVIRONMENTAL CONDITIONS	TEMPERATURE (°F) (Note 1)	RELATIVE HUMIDITY (%)	PRESSURE IN WATER GAUGE (unless other-wise noted)	MAXIMUM INTEGRATED EXPOSURE (Note 2) (rad-carbon)
A8c	Drum Fill Relay Cabinet and Drum Fill Accessible Equipment Area El. 383 ft 0 in.	Normal (Note 14)	Max 108.6	8-70	-0.25 to 0.0	2.1×10^3
		Accident	NA	NA	NA	10^4
A9	Essential Service Water Pump Area	Normal (Note 14)	Max 122	8-70	-0.25 to 0.0	2.1×10^3
		Accident	NA	NA	NA	10^4
A10	Recycle Holdup Tank Cubicle, Gas Decay Pipe Tunnel, Floor Drain Tank Cubicle, Gas Decay Tank Cubicle, Recycle Holdup Pipe Tunnel, Off-Gas Compressor Room, Drain Sump	Normal	Max 122	8-70	-0.25 to 0.0	1.2×10^7
		Accident	NA	NA	NA	1.2×10^7
A10a	Recycle Evap. Cubicle El. 346 ft 0 in.	Normal	Max 92.5	8-70	-0.25 to 0.0	1.2×10^7
		Accident	Max 295	50-100	1.2 psid	1.2×10^7
A10b	Recycle Evap. Pipe Tunnel, Gas Decay Valve Aisle	Normal	Max 121.2	8-70	-0.25 to 0.0	1.2×10^7
		Accident	NA	NA	NA	1.2×10^7
A10c	Blowdown Condenser Room El. 364 ft 0 in.	Normal	Max 112	8-70	-0.25 to 0.0	7.4×10^4
		Accident	Max 365	10-100	2.4 psid	10^5

TABLE 3.11-2 (Cont'd)

ENVIRON-MENTAL ZONE	AREA	ENVIRONMENTAL CONDITIONS	TEMPERATURE (°F) (Note 1)	RELATIVE HUMIDITY (%)	PRESSURE IN WATER GAUGE (unless other-wise noted)	MAXIMUM INTEGRATED EXPOSURE (Note 2) (rad-carbon)
A10d	Auxiliary Equipment Drain Tank Room	Normal (Note 14)	Max 110	8-70	-0.25 to 0.0	$1.2x10^{7}$
		Accident	NA	NA	NA	$1.2x10^{7}$
A11	Penetration Areas, El. 426, and 414 ft. Diesel and Motor Driven Aux. Feedwater Pump Area,	Normal (Note 14)	Max 133	8-70	-0.25 to 0.0	$2.1x10^{3}$
		Accident	NA	NA	NA	10^{4}
A12	Demineralizer Cubicles, Drumming Area, Spent Fuel Pit Pump Room, Process Filter Cubicles, Spent Resin Tank and Pump Rooms, Concentrate Holding Tank Room	Normal	Max 122	8-70	-0.25 to 0.0	$9.9x10^{6}$(Note 7)
		Accident	NA	NA	NA	10^{7}(Note 7)
A12a	Boric Acid Tank Cubicle	Normal (Note 14)	Max 128.6	8-70	-0.25 to 0.0	$9.9x10^{6}$
		Accident	NA	NA	NA	10^{7}
A12b	Surface Condenser Room El. 401 ft 0 in. Radwaste Evaporator	Normal	Max 118	8-70	-0.25 to 0.0	$9.9x10^{6}$
		Accident (Note 13)	Max 300	40-100	1.25 psid	10^{7}
A12c	Filter Pipe Tunnel Area El. 383 ft 0 in., and El. 375 ft 6 in. Filter Valve Aisle Pipe Tunnel	Normal	Max 136.5	8-70	-0.25 to 0.0	$7.5x10^{6}$(Note 12)
		Accident	NA	NA	NA	$7.5x10^{6}$(Note 12)

TABLE 3.11-2 (Cont'd)

ENVIRON- MENTAL ZONE	AREA	ENVIRONMENTAL CONDITIONS	TEMPERATURE (°F) (Note 1)	RELATIVE HUMIDITY (%)	PRESSURE IN WATER GAUGE (unless other- wise noted)	MAXIMUM INTEGRATED EXPOSURE (Note 2) (rad-carbon)
A13	Seal Water Hx Room	Normal	Max 100.6	8-70	-0.25 to 0.0	$1.3x10^6$
		Accident	NA	NA	NA	$1.3x10^6$
A13a	Letdown Heat Exchanger and Valve Aisle El. 383 ft 0 in.	Normal	Max 102	8-70	-0.25 to 0.0	$1.3x10^6$
		Accident	Max 212	0-100	0.7 psid	$1.3x10^6$
A13b	Letdown Reheat Heat Exchanger Room El. 346 ft 0 in.	Normal	Max 91.9	8-70	-0.25 to 0.0	$1.3x10^6$
		Accident	Max 212	10-100	0.7 psid	$1.3x10^6$
A13c	RHR Heat Exchanger Rooms, Containment Spray Pump Rooms, RHR Pump Rooms, Sample Cooler and Sample Drain Tank Rooms, Safety Injection Pump Rooms, Penetration Areas, El. 346, 364, 383, and 401 ft 0 in.	Normal (Note 14)	Max 130	8-70	-0.25 to 0.0	$5x10^6$
		Accident	NA	NA	NA	$1.1x10^7$
A13d	Containment Purge Rooms, Auxiliary Building Exhaust Filter Cubicle	Normal	Max 105.1	8-70	-0.25 to 0.0	$1.1x10^3$
		Accident	NA	NA	NA	10^6
A13e-1	Personnel/Equipment Hatch	Normal	Max 103.8	8-70	-0.25 to 0.0	$2.1x10^3$
		Accident	NA	NA	NA	10^5
A13e-2	H_2 Recombiner, TSC Intake Filters, Containment Air Sample Area	Normal	Max 103.8	8-70	-0.25 to 0.0	$1.1x 10^3$
		Accident	NA	NA	NA	10^4

TABLE 3.11-2 (Cont'd)

ENVIRON-MENTAL ZONE	AREA	ENVIRONMENTAL CONDITIONS	TEMPERATURE (°F) (Note 1)	RELATIVE HUMIDITY (%)	PRESSURE IN WATER GAUGE (unless other-wise noted)	MAXIMUM INTEGRATED EXPOSURE (Note 2) (rad-carbon)
A13f	Positive Displacement Pump Rooms	Normal	Max 122	8-70	-0.25 to 0.0	$1.3x10^{6}$
		Accident	NA	NA	NA	$1.3x10^{6}$
A13g	Centrifugal Charging Pump Rooms	Normal	Max 122	8-70	-0.25 to 0.0	$7.2x10^{6}$
		Accident	NA	NA	NA	$1.3x10^{7}$
A13h	Volume Control Tank Room	Normal (Note 14)	Max 107.6	8-70	-0.25 to 0.0	$8.9x10^{7}$
		Accident	NA	NA	NA	$8.9x10^{7}$
A13j	Refueling Water Pipe Tunnels	Normal	Max 122	20-100	-0.25 to 0.0	$1.4x10^{3}$
		Accident	NA	NA	NA	$1.1x10^{5}$
A13k	Refueling Water Pipe Tunnels	Normal	Max 75	20-100	-0.25 to 0.0	$1.4x10^{3}$
		Accident	NA	NA	NA	$1.1x10^{4}$
	<u>CONTAINMENT BUILDING</u>					
C1	Reactor Cavity, Pressure Vessel Annulus, Hot and Cold Leg Nozzles, Neutron Detector Cavity (Note 8)	Normal	Max 124.7	20-50	-0.1 to 1.0 psig	$1.1x10^{11}$
		Accident	Max 333 (Note 9)	100	50 psig/0-20 min. Saturated ambient/ 20 min.-1 yr.	$1.1x10^{11}$
C2	Reactor Coolant Pump Area (Note 8)	Normal	Max 120	20-50	-0.1 to 1.0 psig	$1.2x10^{7}$
		Accident	Same as C1	100	Same as C1	$2.02x10^{8}$
C3	CRD Shroud (Note 8)	Normal	Max 165	20-50	-0.1 to 1.0 psig	$4.1x10^{6}$
		Accident	Same as C1	100	Same as C1	$2x10^{8}$

TABLE 3.11-2 (Cont'd)

ENVIRON-MENTAL ZONE	AREA	ENVIRONMENTAL CONDITIONS	TEMPERATURE (°F) (Note 1)	RELATIVE HUMIDITY (%)	PRESSURE IN WATER GAUGE (unless other-wise noted)	MAXIMUM INTEGRATED EXPOSURE (Note 2) (rad-carbon)
C5	Pressurizer Enclosure (Note 8)	Normal Accident	Max 151 Same as C1	20-70 100	-0.1 to 1.0 psig Same as C1	$4.1x10^{6}$ $2x10^{8}$
C6	All Other Areas (Note 8)	Normal Accident	Max 120 Same as C1	20-50 100	-0.1 to 1.0 psig Same as C1	$4.1x10^{6}$ $2x10^{8}$
	OUTSIDE AREAS					
01	Outside Areas	Normal Accident	Max 95 NA	0-100 NA	0.0 NA	$5.3x10^{2}$ 10^{4}
	PUMP HOUSES					
P1	Circulating Water (Byron) River Screen (Byron) Lake Screen (Braidwood) River Screen (Braidwood)	Normal Accident	Max 105 NA	8-70 NA	0.0 to 0.25 NA	$5.3x10^{2}$ 10^{4}
P2	Diesel-Driven Fire Pump Rooms, Diesel Oil Tank Rooms	Normal Accident	Max 122 NA	8-70 NA	0 to 0.25 NA	$5.3x10^{2}$ 10^{4}

TABLE 3.11-2 (Cont'd)

ENVIRON-MENTAL ZONE	AREA	ENVIRONMENTAL CONDITIONS	TEMPERATURE (°F) (Note 1)	RELATIVE HUMIDITY (%)	PRESSURE IN WATER GAUGE (unless other-wise noted)	MAXIMUM INTEGRATED EXPOSURE (Note 2) (rad-carbon)
	<u>SERVICE BUILDING</u>					
S1	Machine Shop and Storerooms El. 401, 417, and 433 ft	Normal Accident	Max 104 NA	8-70 NA	-0.1 to 0.1 NA	$5.3x10^2$ 10^4
S2	Service Building, El. 401, 417, 433, and 451 ft, and Radwaste Control Room	Normal Accident	75±2 NA	45±5 NA	0.0 to 0.1 NA	$5.3x10^2$ 10^4
S3	Radwaste Drum Storage, Volume Reduction System	Normal Accident	Max 133 (Note 10) NA	8-70 NA	-0.25 to 0.0 NA	$1.5x10^7$ $1.5x10^7$
S5	Service/Radwaste Building Dry Waste Storage	Normal Accident	Max 104 NA	8-70 NA	-0.25 to 0.0 NA	$1.1x10^3$ 10^4
S6	Smear Test and Labeling Area, Transfer Pits	Normal Accident	Max 125 NA	8-70 NA	-0.25 to 0.0 NA	$1.0x10^6$ 10^6
	<u>TURBINE BUILDING</u>					
T1	Operating Floor and Above	Normal Accident	Max 104 NA	8-70 NA	0.0 NA	$5.3x10^2$ 10^4

TABLE 3.11-2 (Cont'd)

ENVIRON-MENTAL ZONE	AREA	ENVIRONMENTAL CONDITIONS	TEMPERATURE (°F) (Note 1)	RELATIVE HUMIDITY (%)	PRESSURE IN WATER GAUGE (unless other-wise noted)	MAXIMUM INTEGRATED EXPOSURE (Note 2) (rad-carbon)
T2	Below Operating Floor, Heating Boiler Rooms, Water Purifying Building, Demineralizer Building	Normal Accident	Max 104 NA	8-70 NA	0.0 NA	$5.3x10^2$ 10^4
T3	Steam Pipe Tunnels and Safety Valve Enclosures	Normal Accident	Max 123 (Note 3) Max (Note 11) 518.4‡‡‡	8-70 100	-0.25 to 0.0 Max 28.6 psig	$5.3x10^2$ 10^4
T3a	Auxiliary Feedwater Piping Tunnel, El. 362 ft. 6 in.	Normal Accident	Max 100 NA	8-70 NA	-0.25 to 0.0 NA	$5.3x10^2$ 10^4
T4	Auxiliary Steam Piping Tunnel El. 394 ft. 0 in.	Normal Accident (Note 13)	Max 122 Max 300	8-70 50-100	-0.25 to 0.0 1.85 psid	$5.5x10^6$ (Note 12) $5.5x10^6$ (Note 12)
T5	Secondary Sample Room, Control Room Offices, Turbine Room Future Offices, 250-V Battery Room	Normal Accident	75±2 NA	40±5 NA	-0.1 to 0.1 NA	$5.3x10^2$ 10^4

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1. Temperature values given in this table are the maximum temperatures for the given zones. These "maximum" temperatures are bulk average temperatures for the area at large; they may not bound local temperatures in the vicinity of hot pipes or heat-generating equipment. Where appropriate, EQ evaluations must include consideration of temperature rise due to the effects of local heat sources.
2. Entries in this table under Accident include 60 years of normal exposure (continuous operation) plus postaccident exposure and at least 10% margin for postaccident exposure; they represent the maximum integrated dose, in rads carbon, external to the equipment only. For equipment in contact with radioactive liquids, the qualification value might increase considerably.

TABLE 3.11-2 (Cont'd)

3. Normal temperature of 123°F is the weighted average, which is shown to be a conservative value for the determination of component service life. Normal temperature may exceed 123°F for some duration. However, this does not impact the qualification of EQ-related components in these areas.

4. The abnormal condition is expected to occur for Braidwood only, when there is a chlorine release accident outside the station. Environmental parameters within these zones (A1, A2, A2a and A5) are not expected to change.

5. Reported value is the maximum temperature for routine plant operation. Under conditions of HELB in the Turbine Building, rooms will have a brief temperature and pressure excursion while humidity remains within the normal range. Postulated peak temperatures are 132°F for the ESF switchgear rooms (due to HELB on Turbine Building Elev.426’) and 114°F for the miscellaneous electrical equipment (worst-case due to HELB on Turbine Building Elevs. 426’ or 451’) [Reference 9]. The battery rooms are conservatively assigned the same peak temperature (114°F) as the miscellaneous equipment rooms. Even though some plots in Reference 9, show increasing trend in relative humidity at the end of the 1500-second computer run, relative humidity at this point is the maximum value since this coincides with termination of mass and energy releases. The temperature excursions are very brief (on the order of minutes) and are too short to significantly affect equipment temperature, which remains below the peak temperature due to thermal lag effects.

6. Reported value is the maximum temperature for routine plant operation. Under conditions of a HELB in the Turbine Building, rooms will have a brief temperature and pressure excursion while humidity remains within the normal range. The postulated peak temperature is 150°F for the diesel generator rooms due to HELB on Turbine Building Elev. 401’ [Reference 9]. The temperature excursion is very brief duration (on the order of minutes) and is too short to significantly affect equipment temperature, which remains below the peak temperature due to thermal lag effects.

7. Equipment inside cubicles containing spent resin sources (spent resin tank, pump and mixed bed demineralizer) may be exposed to normal operating doses of up to $2x10^8$ rads carbon.

8. Equipment in Environmental Zones C1, C2, C3, C5, and C6 can be subject to a containment spray solution with a pH exceeding 10.5 until completion of NaOH injection from the CSAT. Under accident conditions, spray chemistry will vary depending upon the operational mode of containment spray. See UFSAR Subsection 3.11.5.

TABLE 3.11-2 (Cont'd)

9. A peak temperature of less than 334°F inside containment has been calculated for the MSLB design basis accident.

For EQ purposes, the bounding time/temperature profile is as follows:

Temperature	Time
120°F to 255°F	0 to 1.0 sec.
255°F to 285°F	1.0 to 3.0 sec.
285°F to 325°F	3 to 20 sec.
325°F to 333°F	20 to 53 sec.
333°F to 334°F	53 to 61 sec.
334°F	61 to 70 sec.
334°F to 320°F	70 to 90 sec.
320°F	90 to 180 sec
320°F to 270°F	3 to 5 min
270°F	5 to 20 min
270°F to 265°F	20 to 30 min
265°F to 185°F	30 min to 1 day
185°F to 155°F	1 to 20 days
155°F	20 days to 1 year

10. Within zone S3, the Fluid Bed Dryer Area has a maximum temperature of 133°F. The HEPA Filter Area and the Gas Solid Separator Area have a maximum temperature of 129°F. The remaining areas of the zone have a maximum temperature of 122°F.

11. Maximum accident conditions are defined by main steamline break (MSLB) outside containment. For safe shutdown functions, the maximum temperature is 413.5 °F. For post-accident monitoring considerations, the maximum temperature in Zone T3 is 518.4 °F. See Subsection 3.11.10 for more detail.

12. This value was calculated using a usage factor corresponding to a system operating time of 100 hours per year.

13. Accident conditions in the surface condenser rooms, radwaste evaporator rooms, and auxiliary steam piping tunnel do not apply at Byron. A blank plate has been installed in the auxiliary steam supply line prior to that line entering the tunnel.

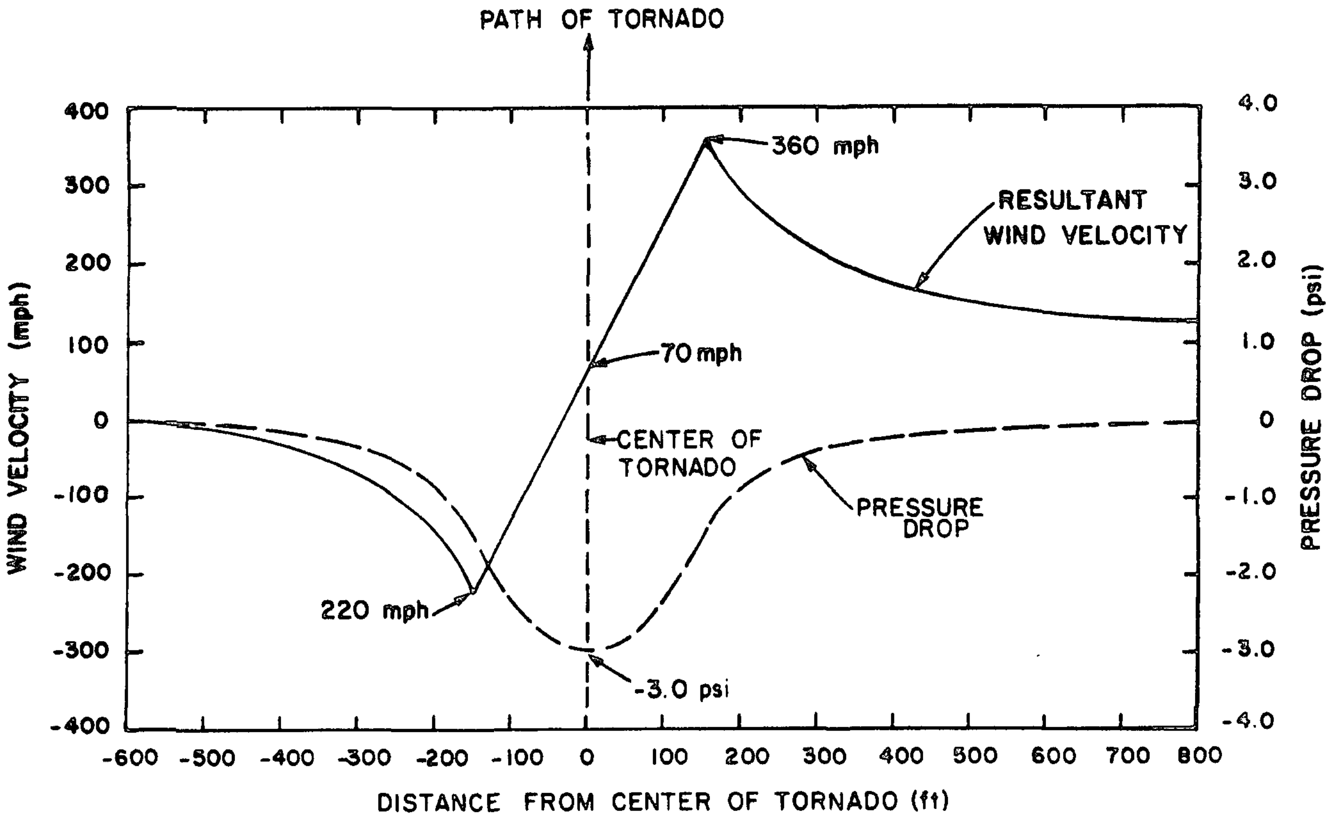
14. Reported value is the maximum temperature for routine plant operation. Under conditions of LOCA and LOOP, maximum temperature will increase during 2-hour period due to HVAC systems operation using available electrical power sources and with consideration of equipment heat loads for event mitigation. Maximum temperatures during this condition are: 143°F in Zone A13h; 140°F in Zones A8, A8b, A8c, and A11; 130°F in Zones A10d and A13c; 129°F in Zone A12a; and 122°F in Zone A9. These maximum temperatures also apply during the 4-hour coping period following station blackout. [Reference 8].

Table 3.11-3 has been deleted intentionally.

TABLE 3.11-4

SPECIFIC POSTACCIDENT OPERABILITY REQUIREMENT

	EQUIPMENT	REQUIRED POSTACCIDENT OPERABILITY
1.	Equipment necessary to perform trip functions	5 minutes
2.	Equipment that is located outside containment, is accessible, and can be repaired, replaced, or recalibrated	2 weeks
3.	Equipment located inside containment and required for postaccident monitoring	4 months (This number is based on an acceptable amount of time to allow the instrument to be repaired, replaced, or recalibrated or an equivalent indication to be obtained)
4.	Equipment that is located inside containment, is inaccessible, or cannot be repaired, replaced or recalibrated	1 year
5.	Equipment located in a mild environment following an accident	Continuous



TORNADO PARAMETERS REGION I	V_c = 290 mph, V_t (max) = 70 mph, V_t (min) = 5 mph R_c = 150 ft, P_{max} = 3.0 psi, max (dP/dt) = 2.0 psi/sec

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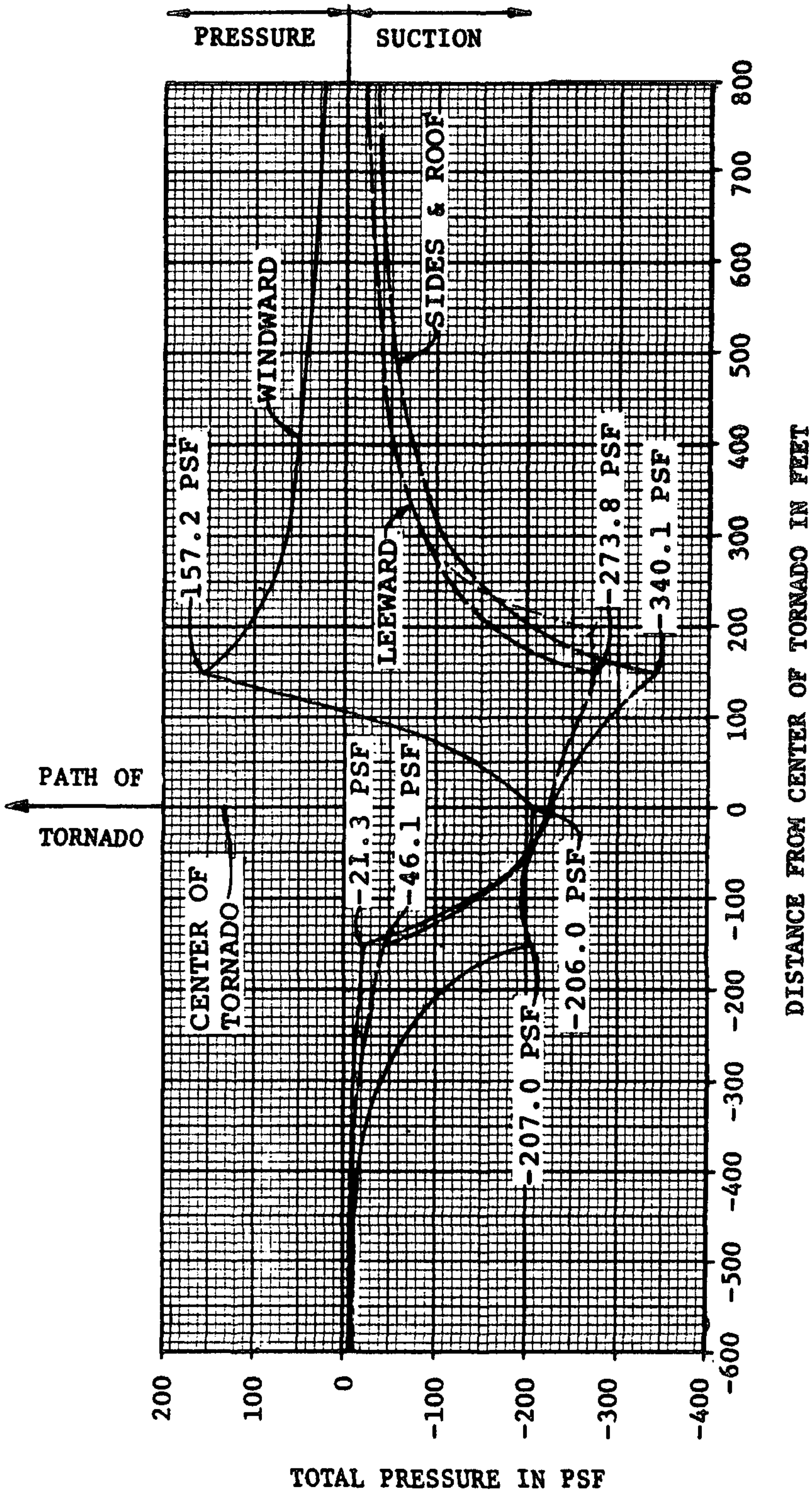
FIGURE 3.3-1

TORNADO WIND VELOCITY
AND PRESSURE DROP



Tornado Parameters	V_c = 290 mph, $V_t(max)$ = 70 mph, $V_t(min)$ = 5 mph
Region I	R_c = 150 ft, P_{max} = 3.0 psi, max (dP/dt) = 2.0 psi/sec.

TOTAL TORNADO PRESSURE DUE TO TORNADO WIND AND PRESSURE DROP (W_w + 0.5 W_p)

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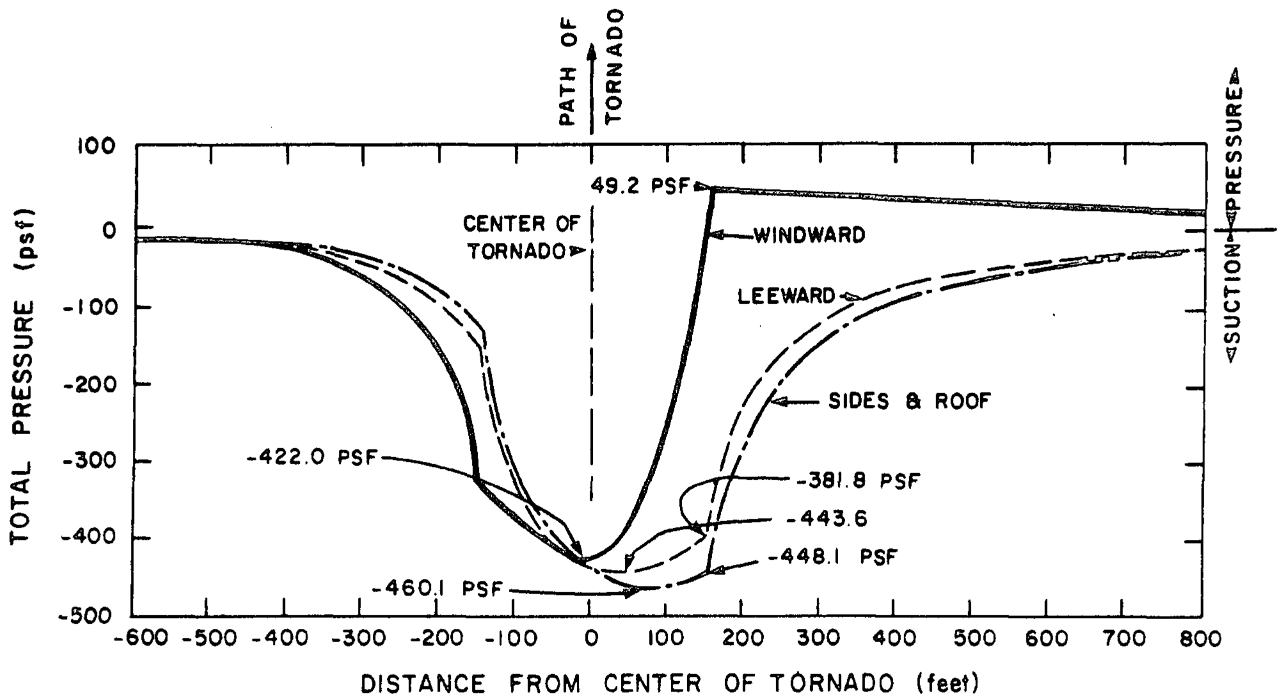
FIGURE 3.3-2

TORNADO WIND PRESSURE



TORNADO PARAMETERS REGION I	V_c = 290 mph, $V_t(max)$ = 70 mph, $V_t(min)$ = 5 mph R_c = 150 ft, P_{max} = 3.0 psi, max(dP/dt) = 2.0 psi/sec

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FIGURE 3.3-3

TOTAL TORNADO PRESSURE DUE TO
TORNADO WIND AND PRESSURE DROP

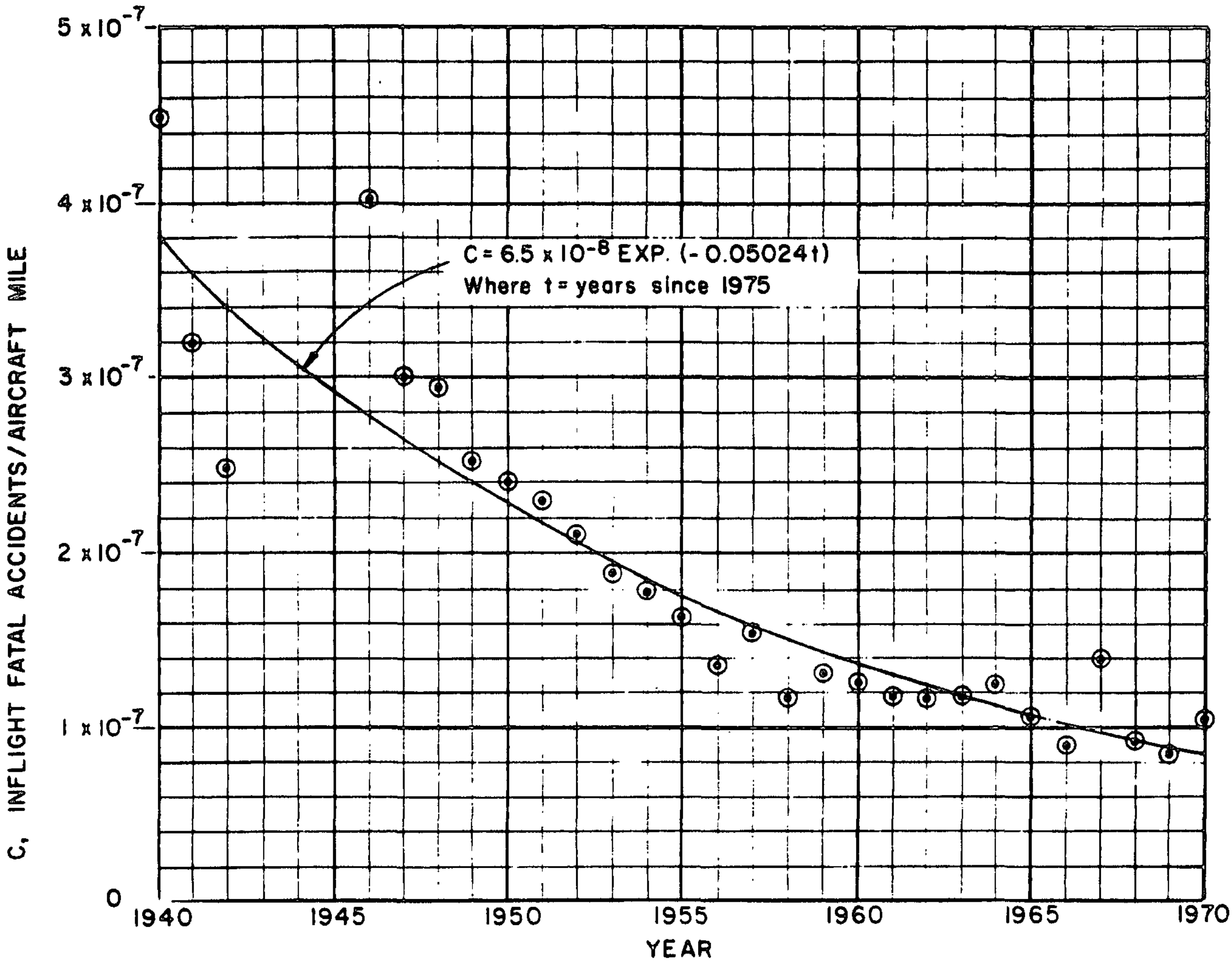

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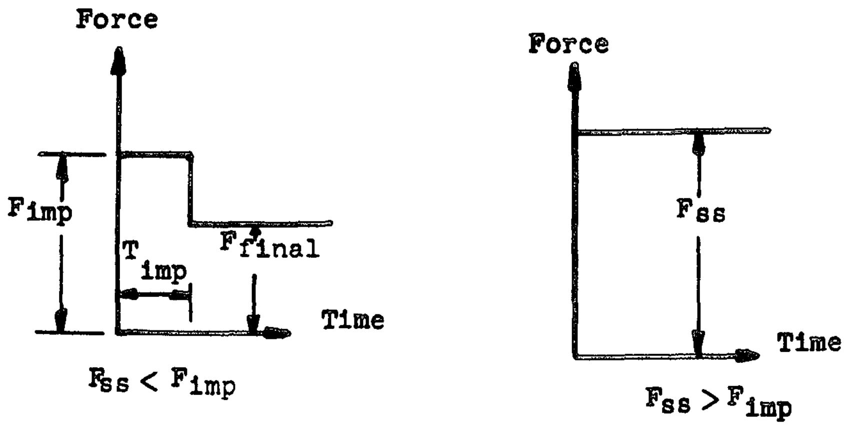
FIGURE 3.5-1

PLOT OF ACCIDENT RATES FOR
GENERAL AVIATION AIRCRAFT

Figure 3.5-2 has been deleted intentionally.

Figures 3.6-1 through 3.6-12 have been deleted intentionally.



PIPE THRUST REACTION RESULTING FROM A CIRCUMFERENTIAL BREAK

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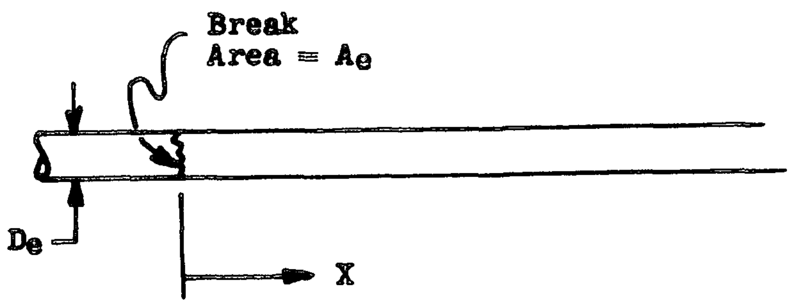
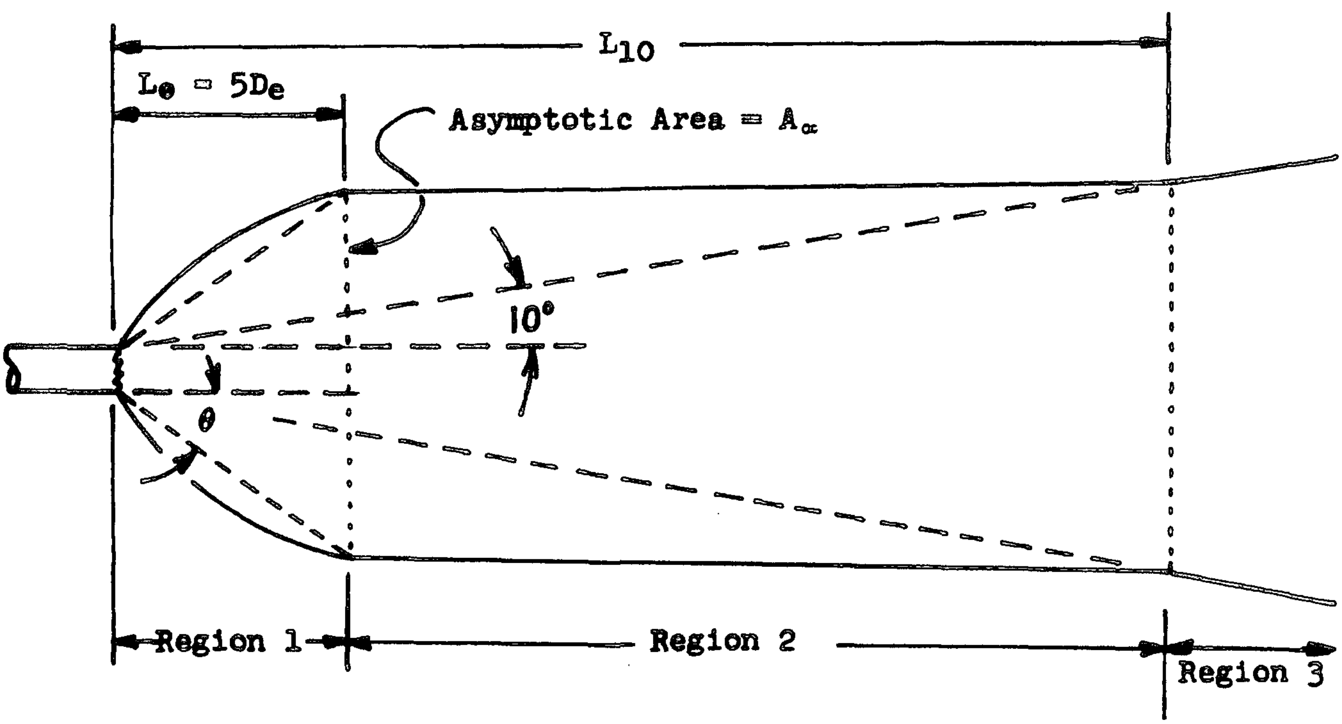
FIGURE 3.6-13

BLOWDOWN FORCE TIME HISTORIES FOR
STEAM AND FLASHING WATER



NON-FLASHING/NON-EXPANDING FLUID



FLASHING/EXPANDING FLUID

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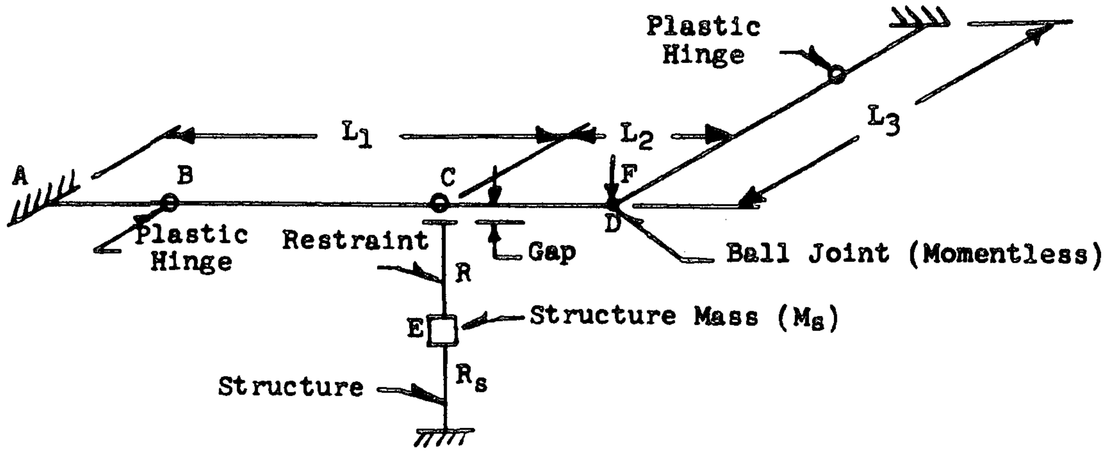
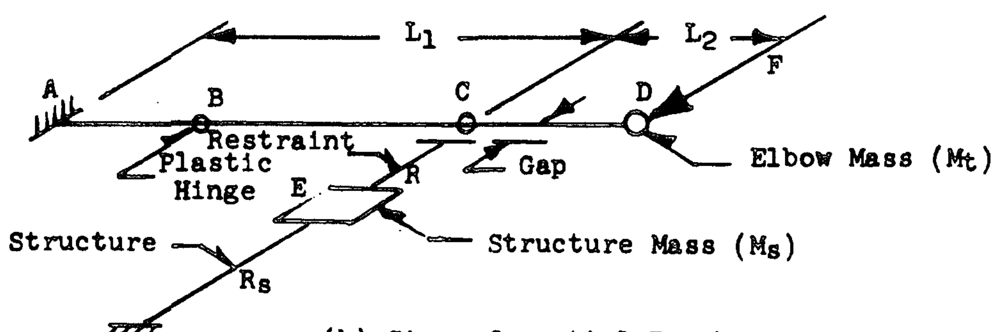
FIGURE 3.6-14

IMPINGEMENT AREA



(a) Longitudinal Break at Elbow



(b) Circumferential Break at Elbow

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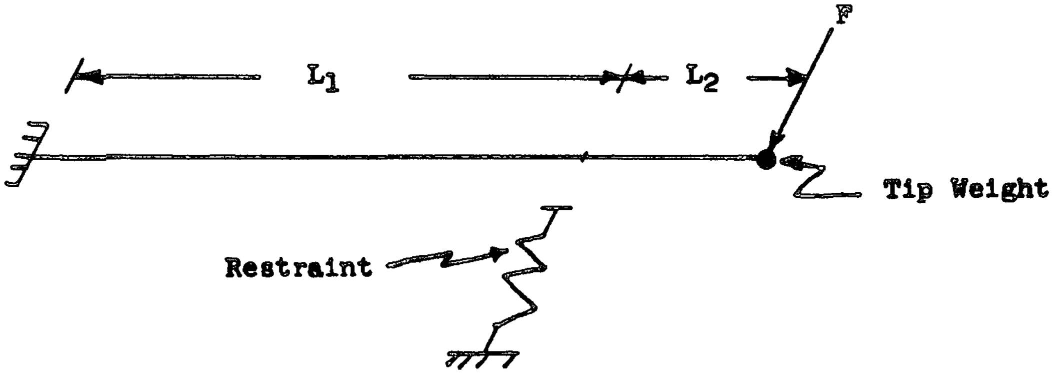
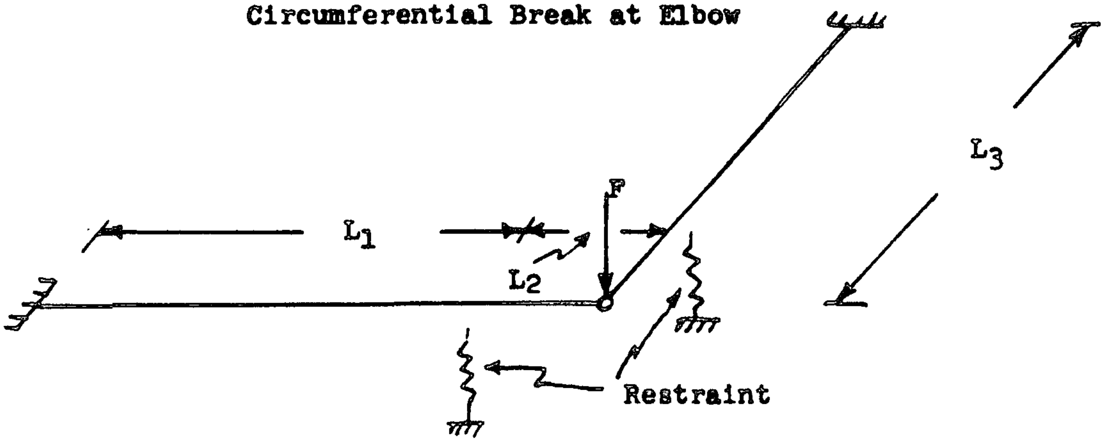
FIGURE 3.6-15

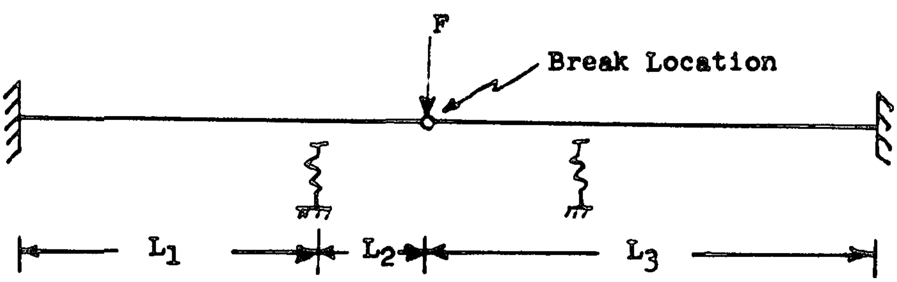
MODELS FOR DYNAMIC ANALYSIS OF
PIPE WHIP-ENERGY BALANCE METHOD



Circumferential Break at Elbow



Longitudinal Break at an Elbow

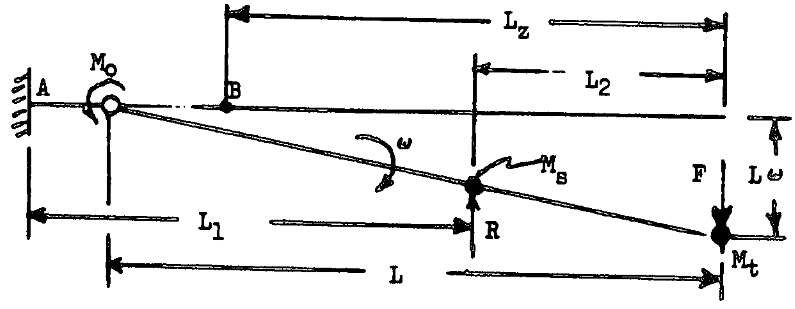


Longitudinal Break at an Interior Point

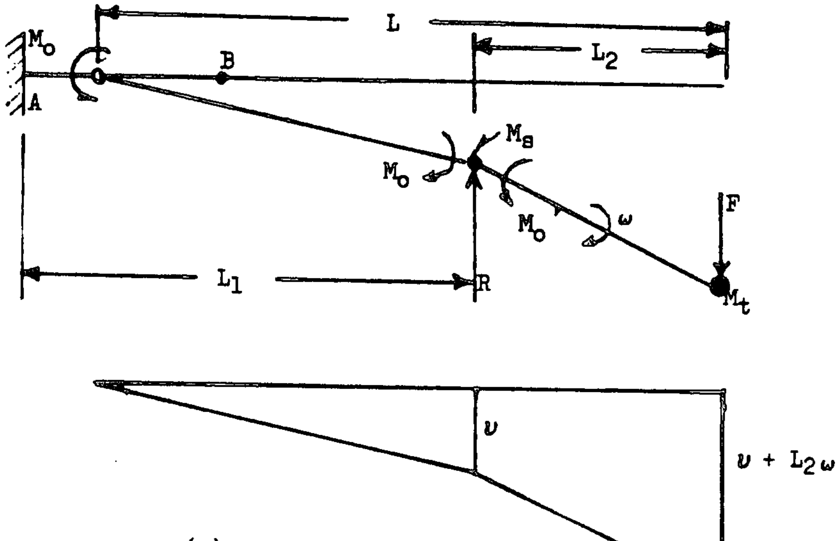
BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-16

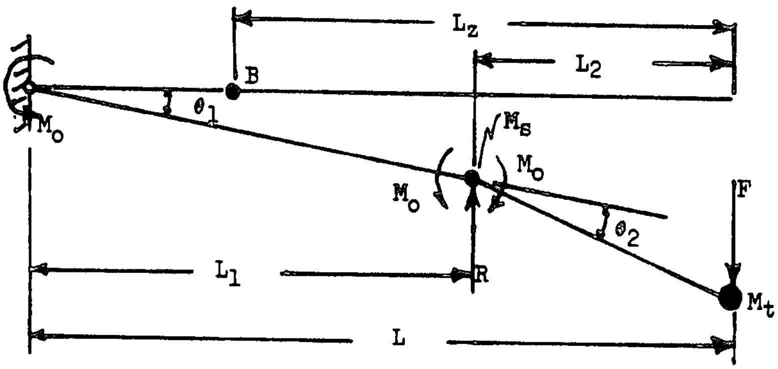
PIPE WHIP MODELS FINITE DIFFERENCE METHOD



(a) No Hinge at Restraint
Second Stage



(b) Hinge at Restraint
Second Stage

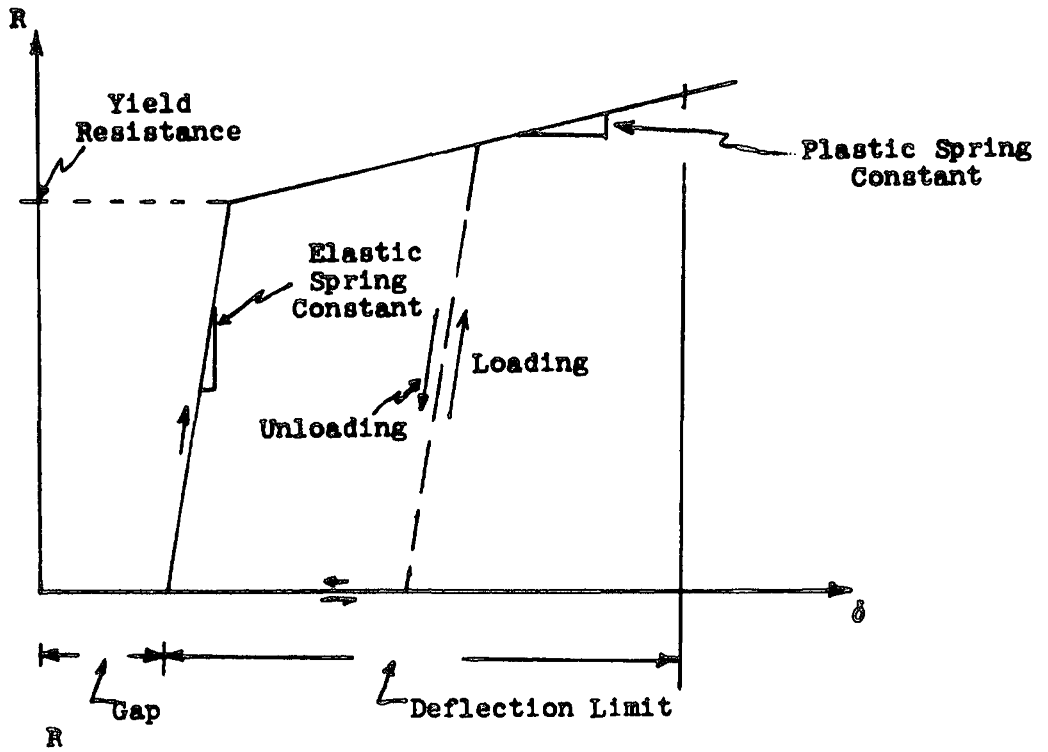


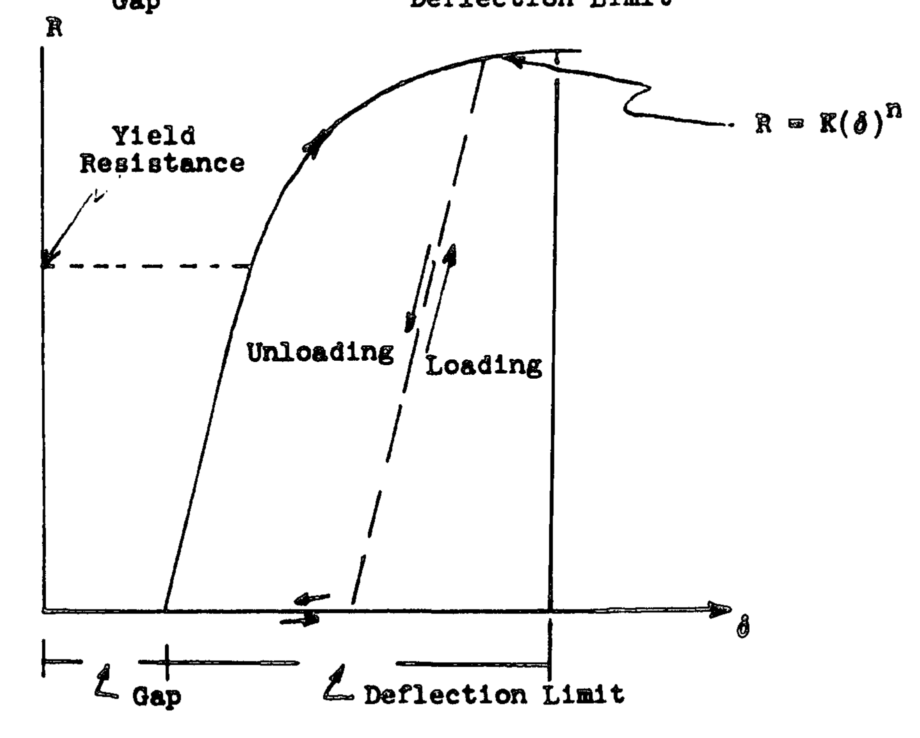
(c) Hinge at Support
Third Stage

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-17

SECOND AND THIRD STAGES OF
MOTION-ENERGY BALANCE METHOD



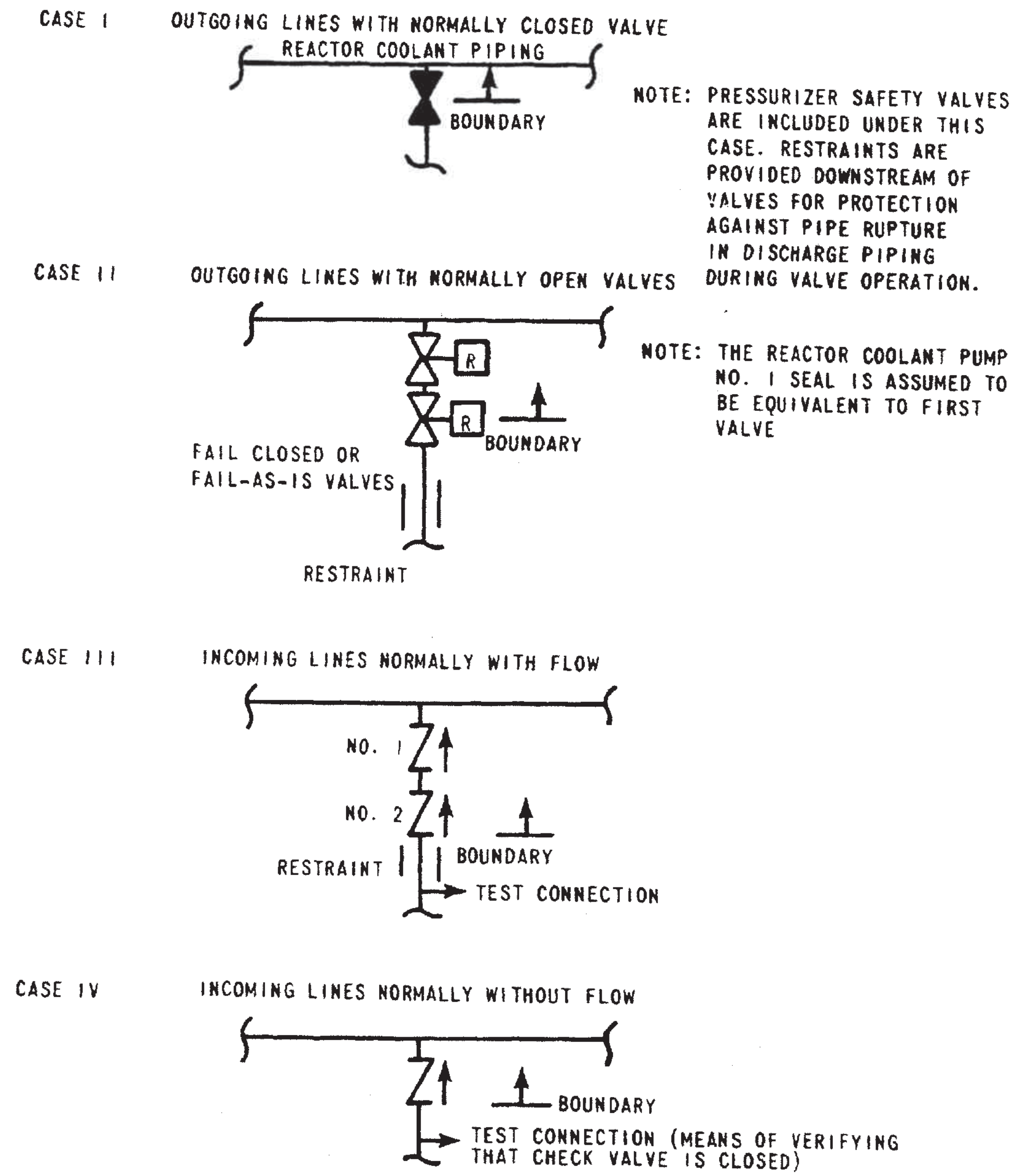


BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-18

RESTRAINT PROPERTIES

Figures 3.6-19 through 3.6-22 have been deleted intentionally.



CASE V ALL INSTRUMENTATION TUBING AND INSTRUMENTS CONNECTED DIRECTLY TO THE REACTOR COOLANT SYSTEM IS CONSIDERED AS A BOUNDARY. HOWEVER. A BREAK WITHIN THIS BOUNDARY RESULTS IN A RELATIVELY SMALL FLOW WHICH CAN NORMALLY BE MADE UP WITH THE CHARGING SYSTEM.

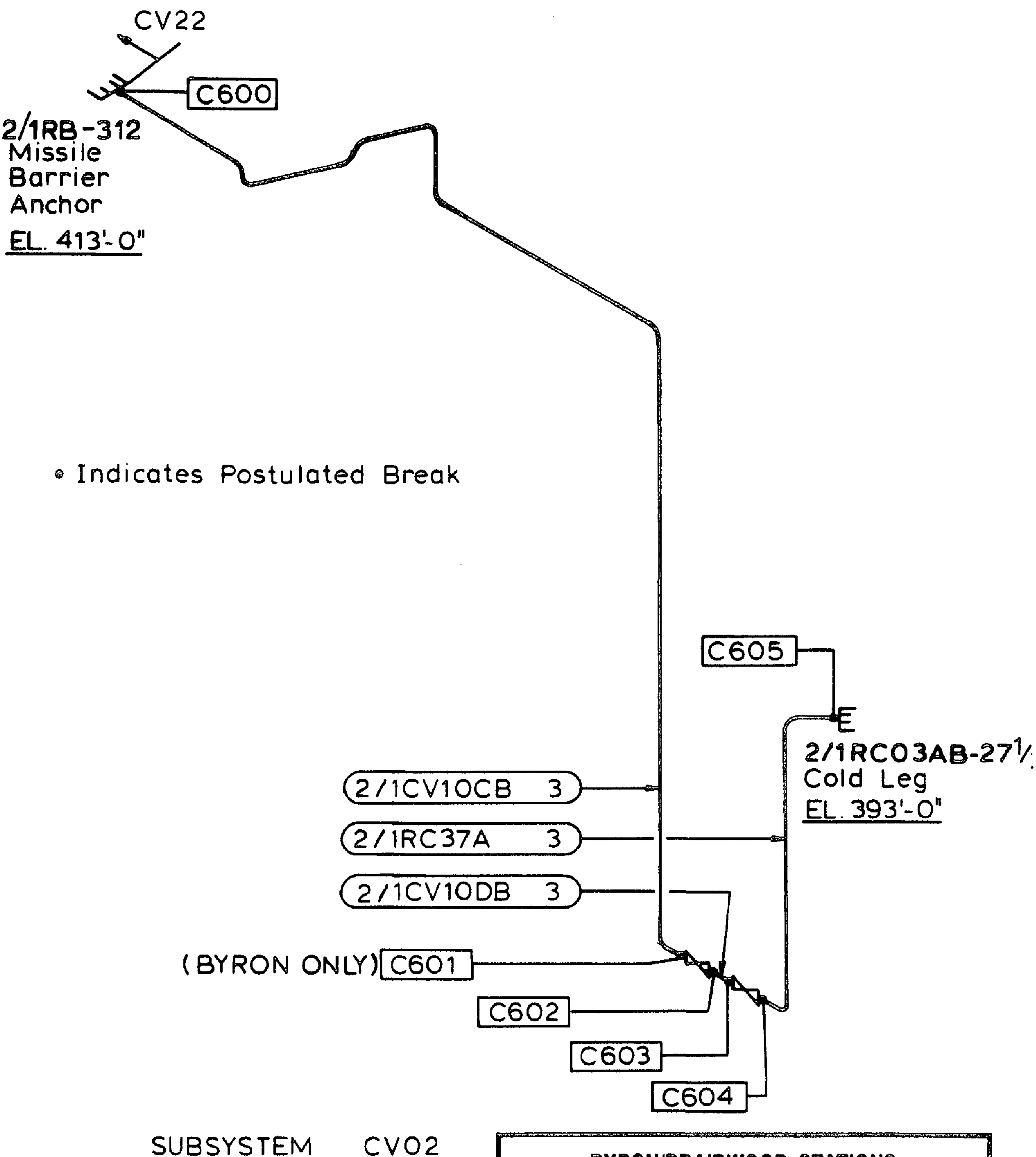
NOTE: THE TERM "BOUNDARY" USED IN THIS FIGURE REFERS TO THE OUTERMOST LOCATION THAT A BREAK IN THE LINE WOULD RESULT IN A LOSS OF REACTOR COOLANT. BEYOND THIS LOCATION, IT IS ASSUMED THAT THE BREAK WOULD BE ISOLATED. THE SYSTEMS ARE DESIGNED SUCH THAT COMPONENTS REQUIRED TO PROVIDE THE BOUNDARY FUNCTION ARE PROTECTED FROM THE EFFECTS OF A LINE BREAK DOWNSTREAM OF THE BOUNDARY LOCATION.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-23

LOSS OF REACTOR COOLANT ACCIDENT BOUNDARY LIMITS

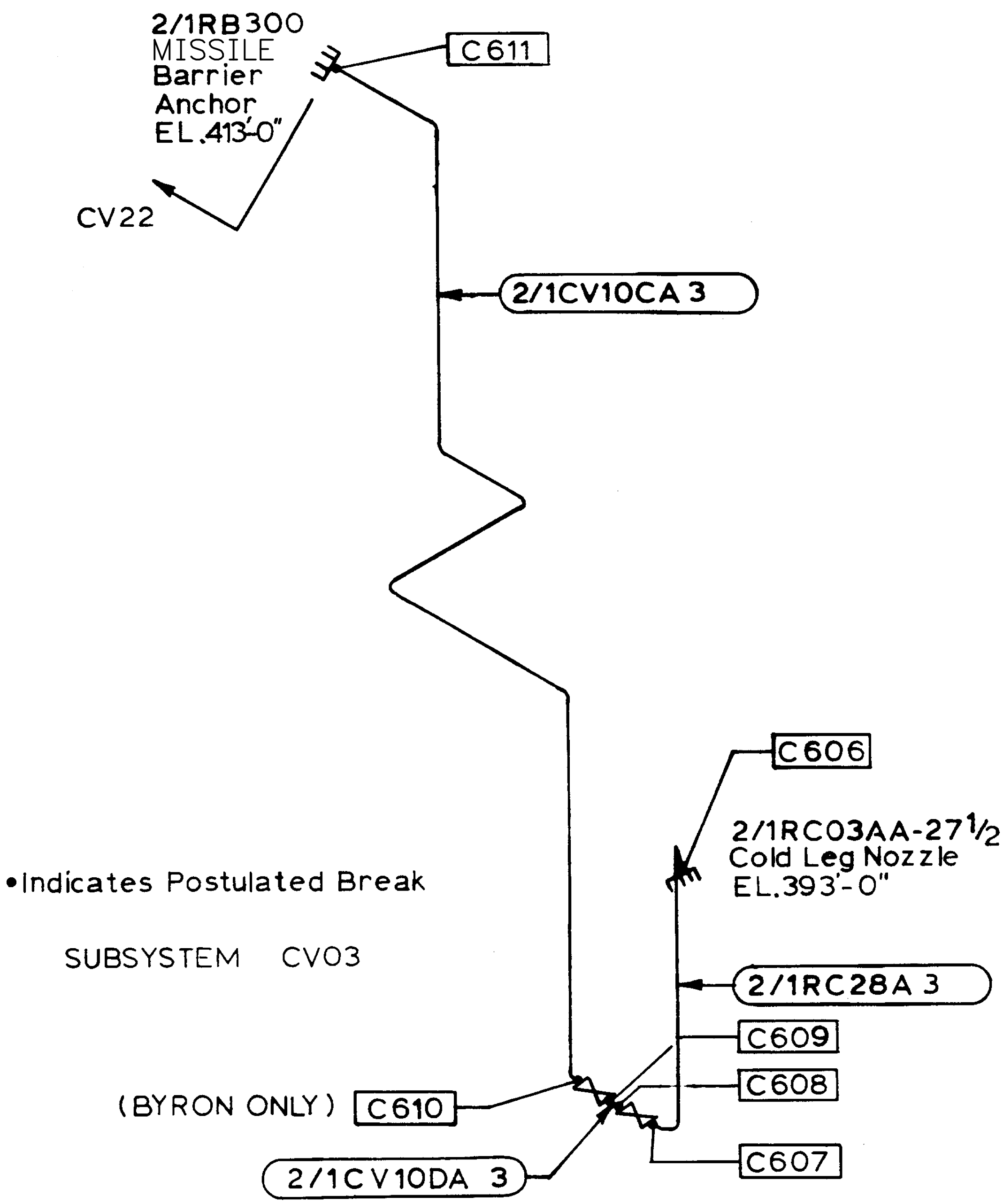
Figure 3.6-24 has been deleted intentionally.



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-25

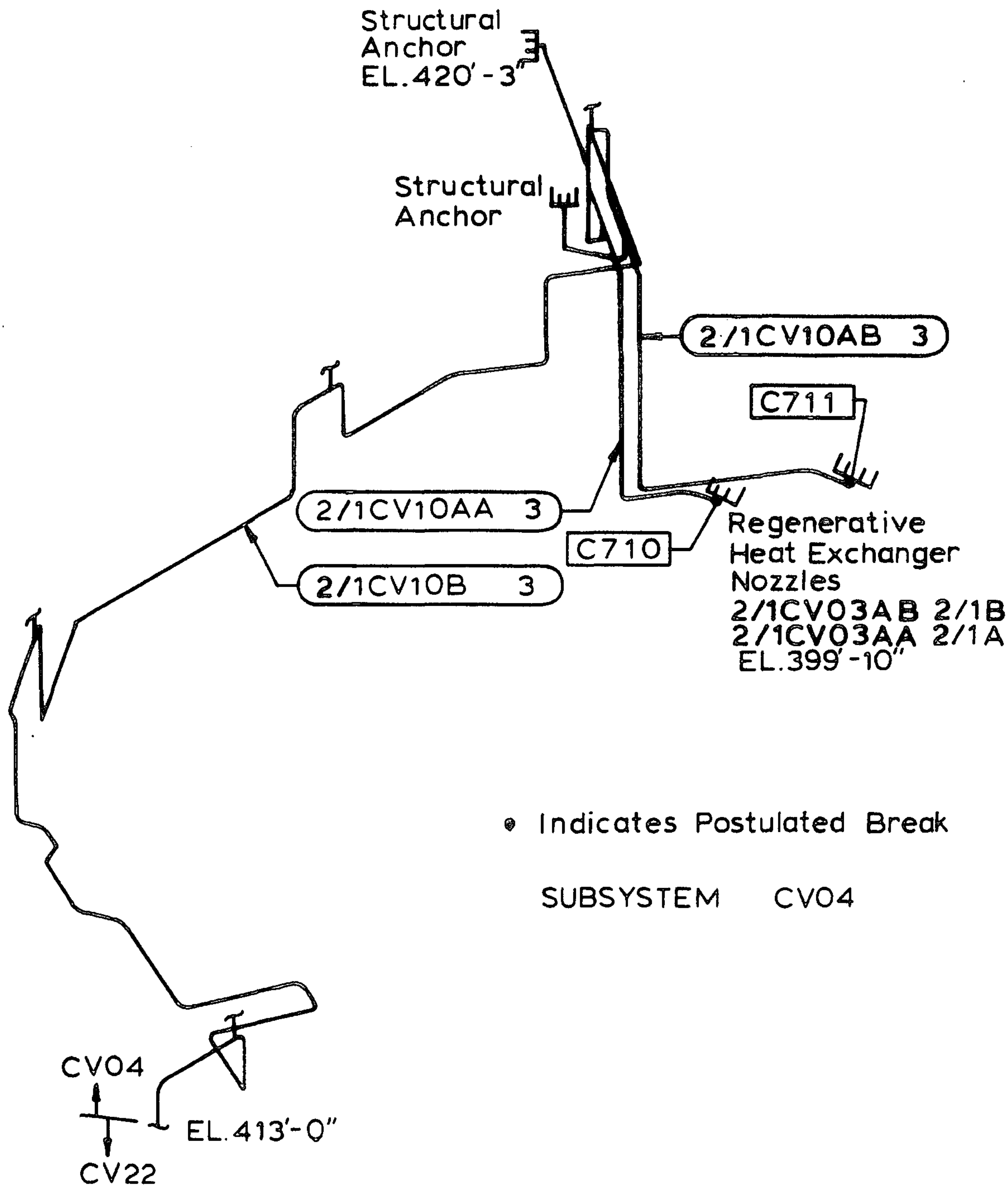
CHARGING LINE LOOP 2 SUBSYSTEM CV02
POSTULATED BREAK LOCATIONS



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-26

CHARGING LINE LOOP 1 SUBSYSTEM CV03
POSTULATED BREAK LOCATIONS



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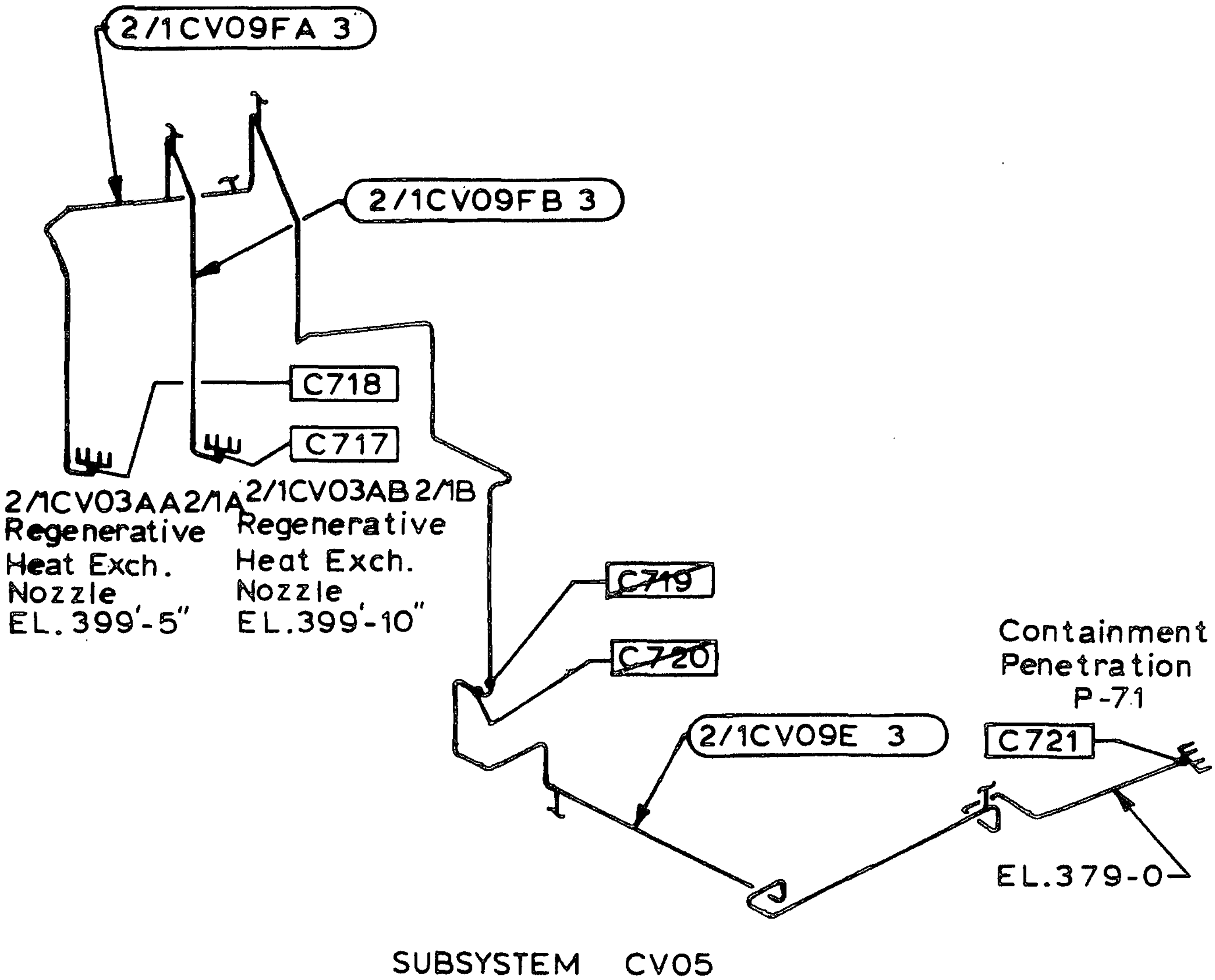
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-27

CVCS SUBSYSTEM CV04
POSTULATED BREAK LOCATIONS

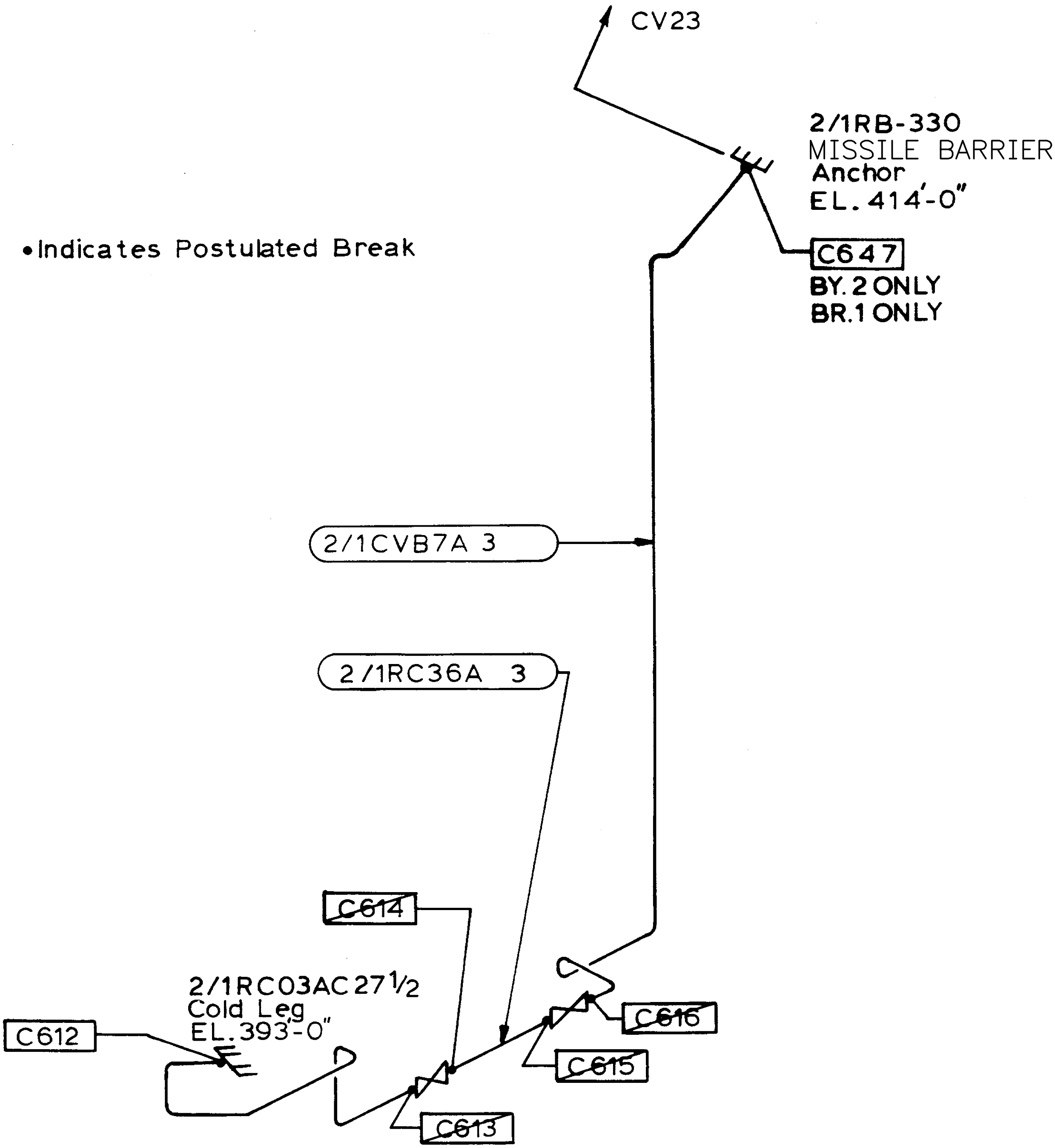


Note:

▭ Arbitrary Intermediate Break.

● Indicates Postulated Break

BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-28 CVCS SUBSYSTEM CV05 POSTULATED BREAK LOCATIONS



SUBSYSTEM CV06

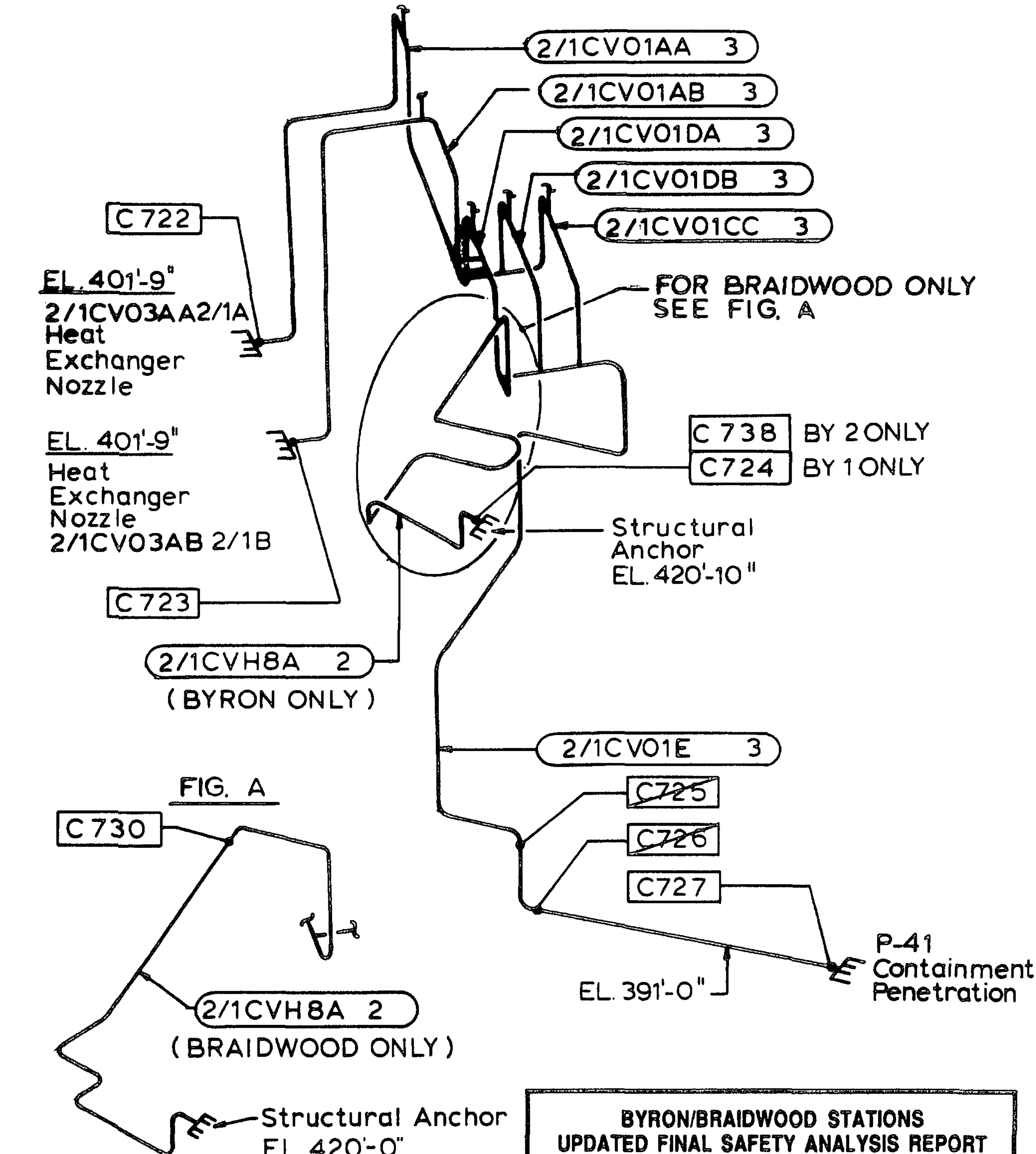
NOTE:

Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-29

NORMAL LETDOWN LOOP 3 CVCS SUBSYSTEM CV06
POSTULATED BREAK LOCATIONS



NOTE:

Arbitrary Intermediate Break

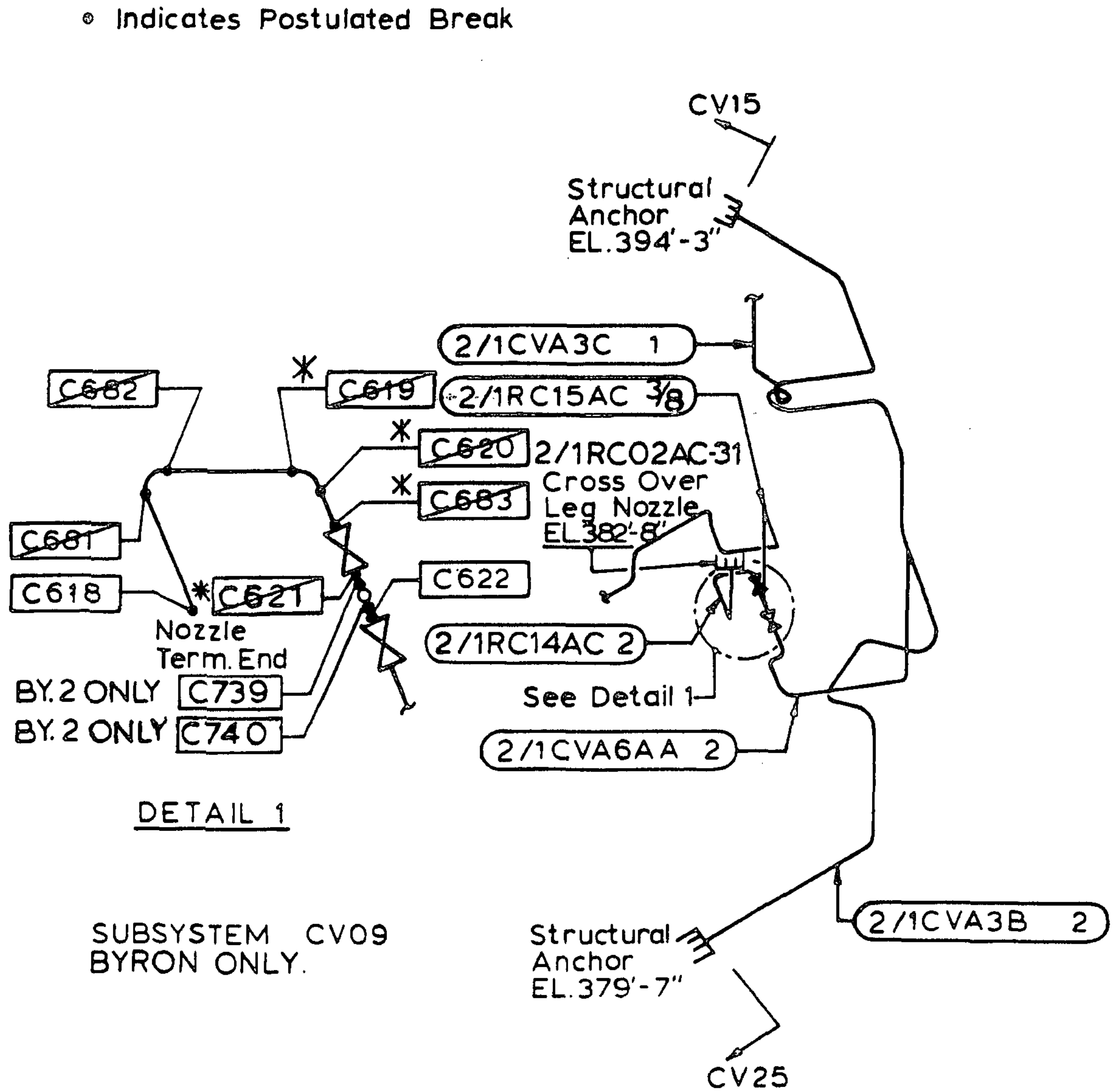
◉ Indicates Postulated Break

SUBSYSTEM CV07

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-30

CVCS SUBSYSTEM CV07
POSTULATED BREAK LOCATIONS



Note:

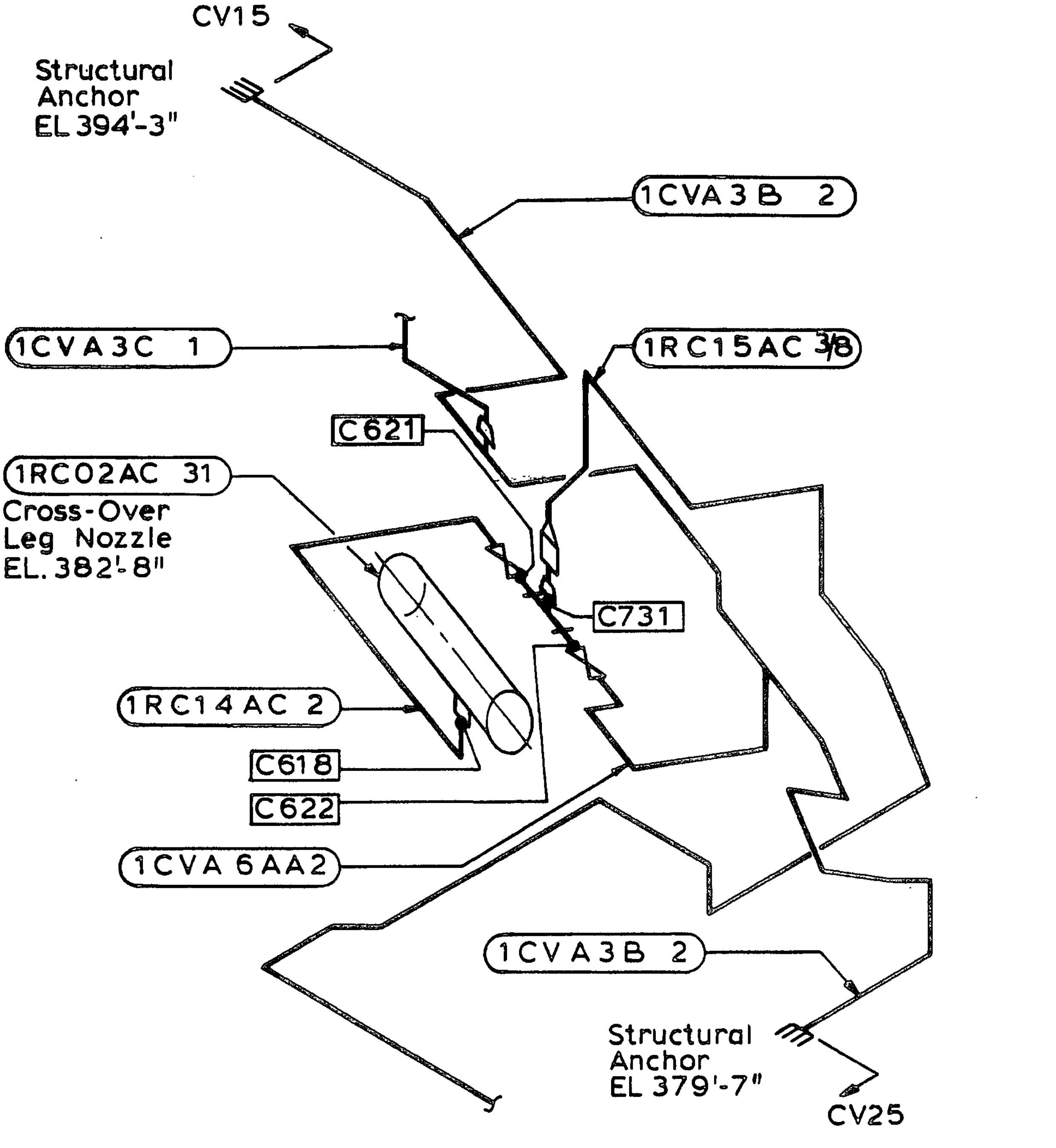
Arbitrary Intermediate Break.

* Break Restored For Byron 2

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-31

CVCS SUBSYSTEM CV09
POSTULATED BREAK LOCATIONS
(BYRON)



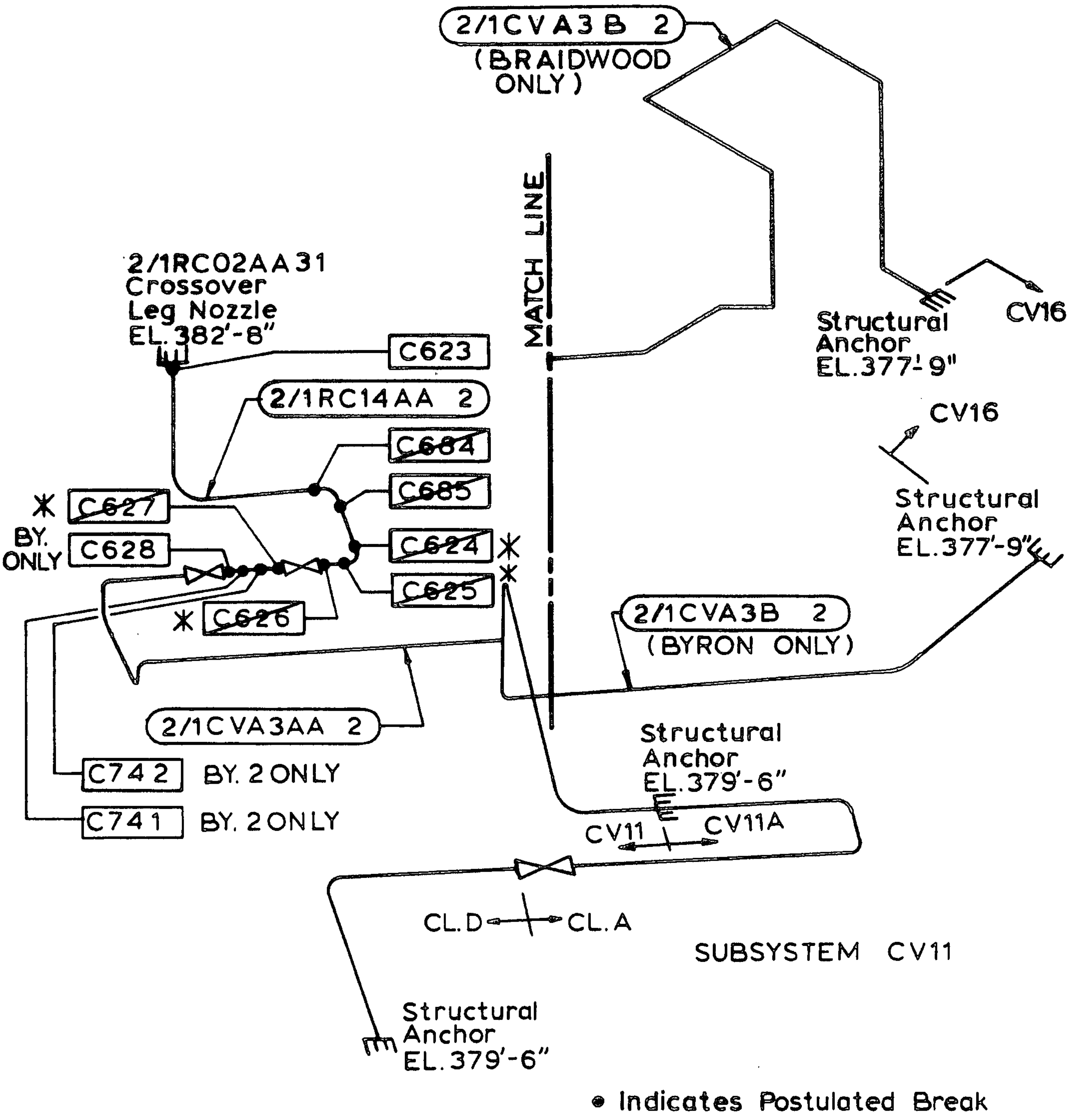
● Indicates Postulated Break

SUBSYSTEM CV09
BRAIDWOOD ONLY

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-31a

CVCS SUBSYSTEM CV09
POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



NOTE:

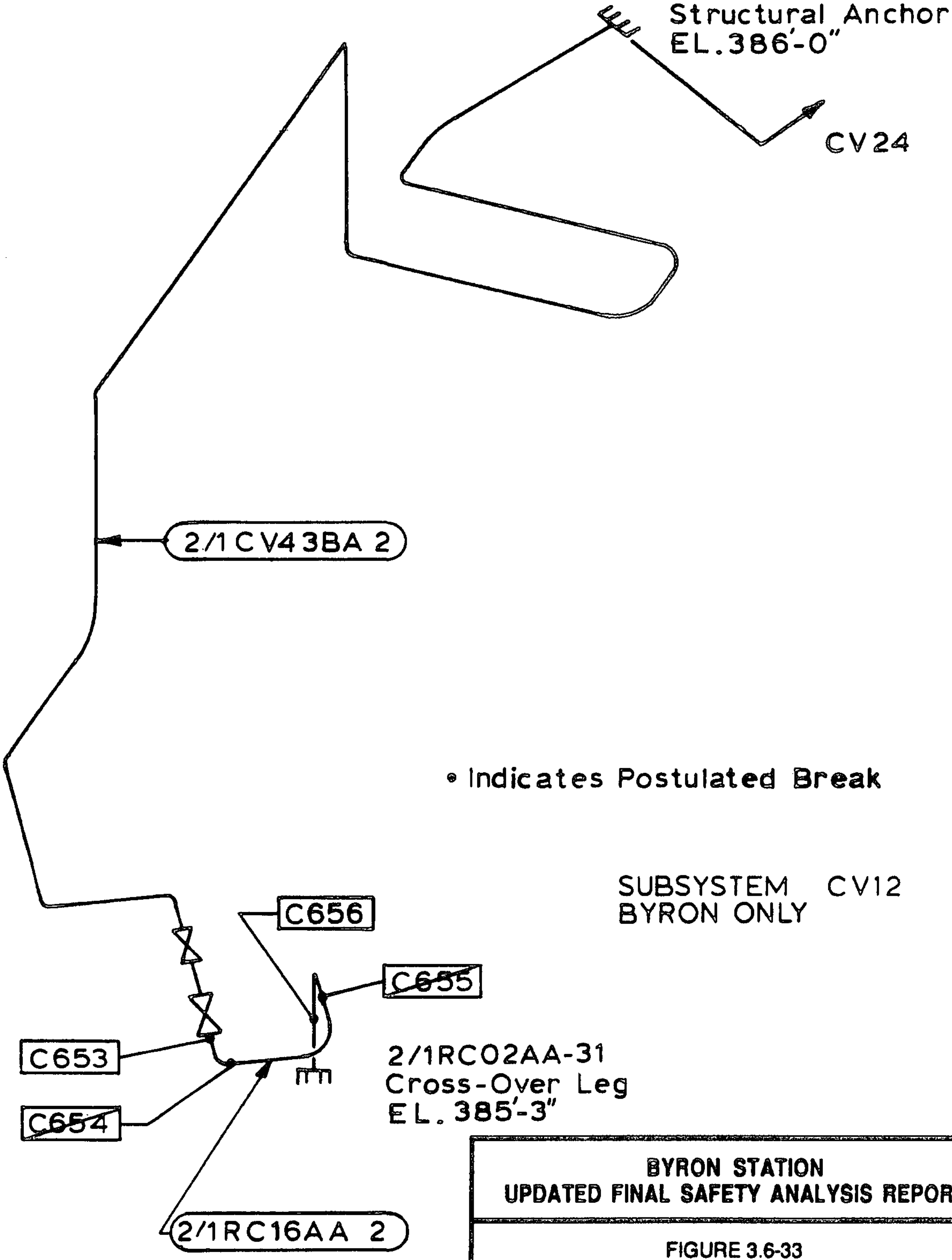
Arbitrary Intermediate Break

✱ Break Restored For Byron 2

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-32

EXCESS LETDOWN LOOP 1 SUBSYSTEM CV11
POSTULATED BREAK LOCATIONS

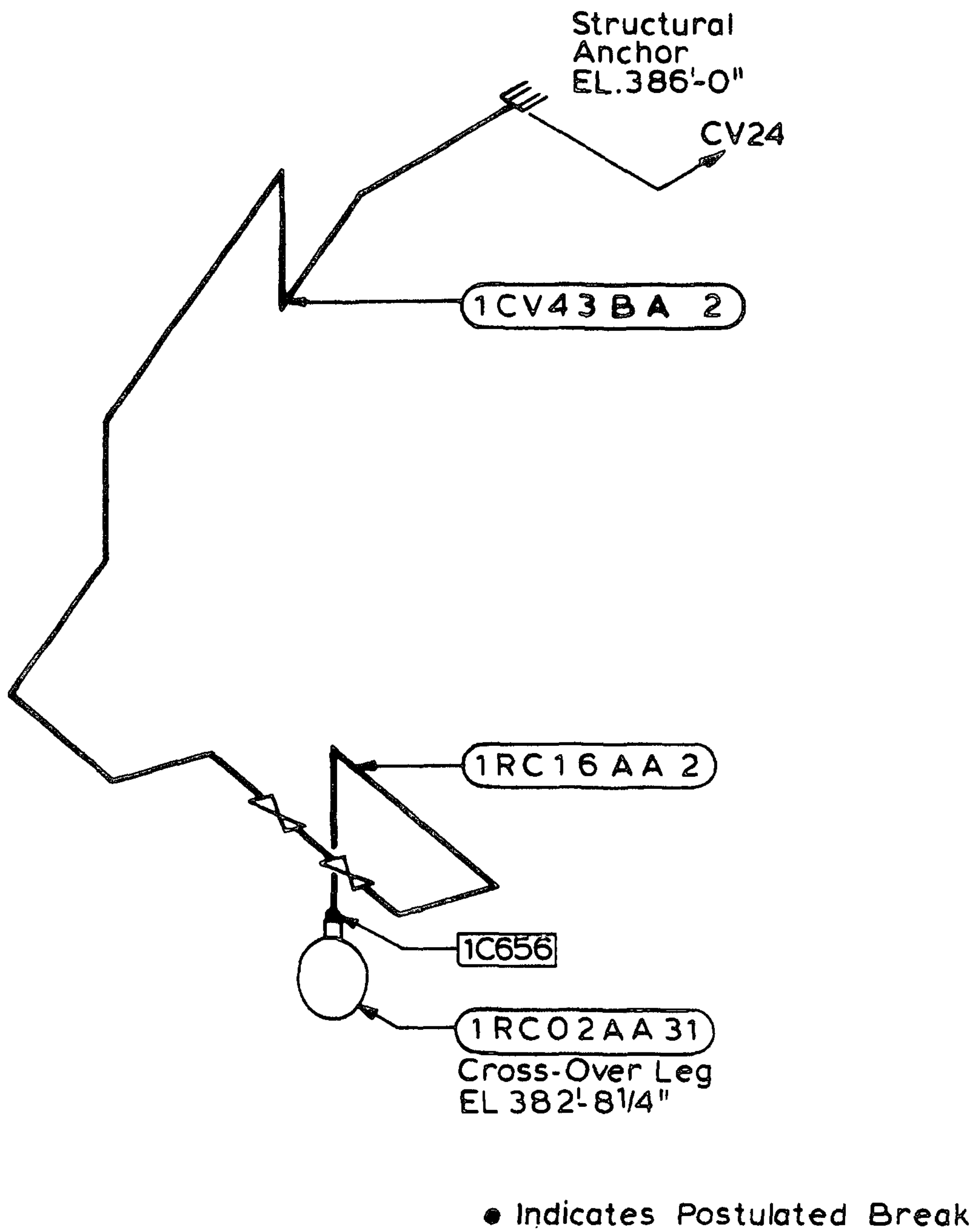


NOTE:
▱ ARBITRARY INTERMEDIATE BREAK

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

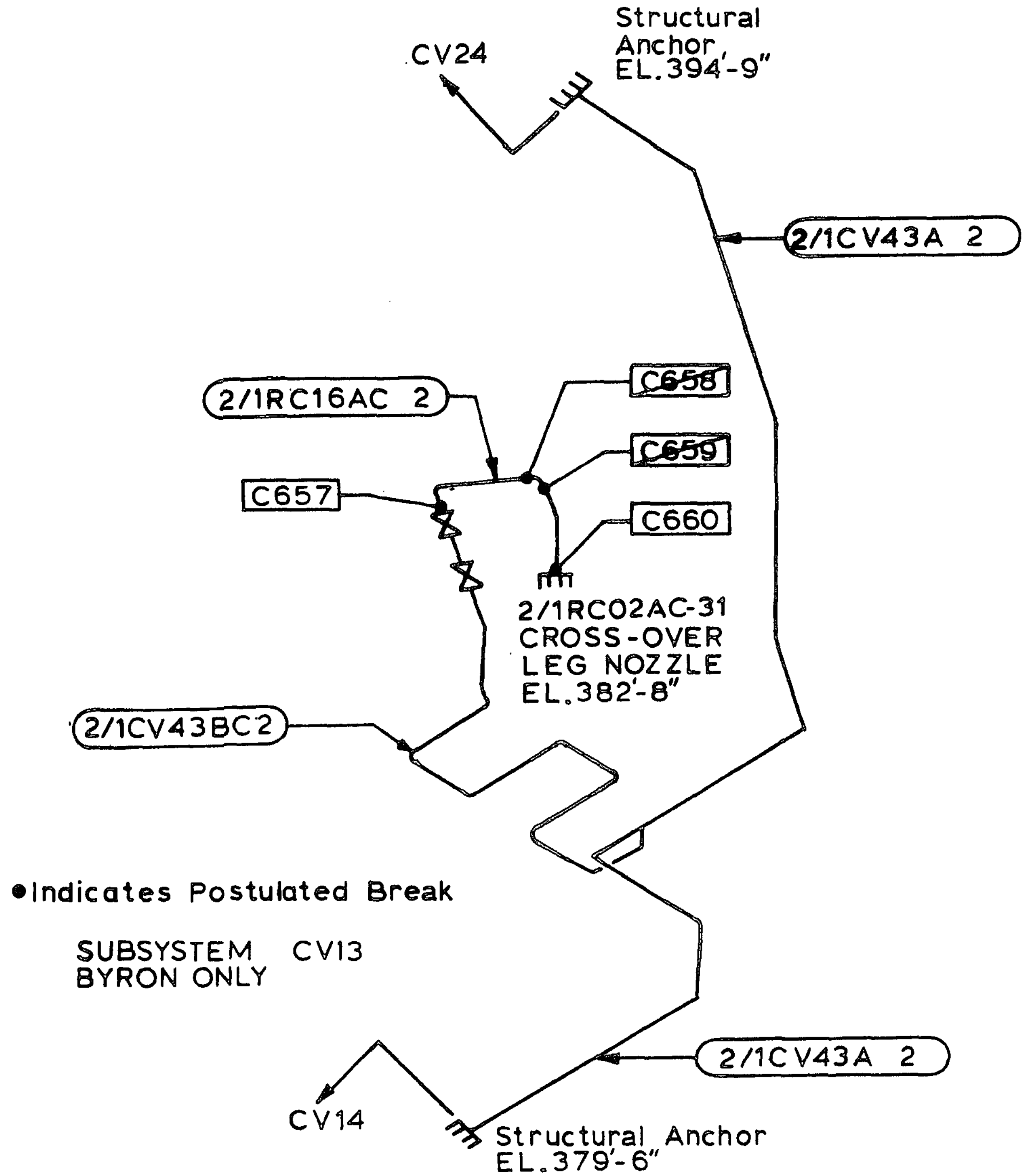
FIGURE 3.6-33

LOOP FILL LINE LOOP 1 SUBSYSTEM CV12
POSTULATED BREAK LOCATIONS
(BYRON)



SUBSYSTEM CV12
BRAIDWOOD ONLY

BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-33a LOOP FILL LINE LOOP 1 SUBSYSTEM CV12 POSTULATED BREAK LOCATIONS (BRAIDWOOD)

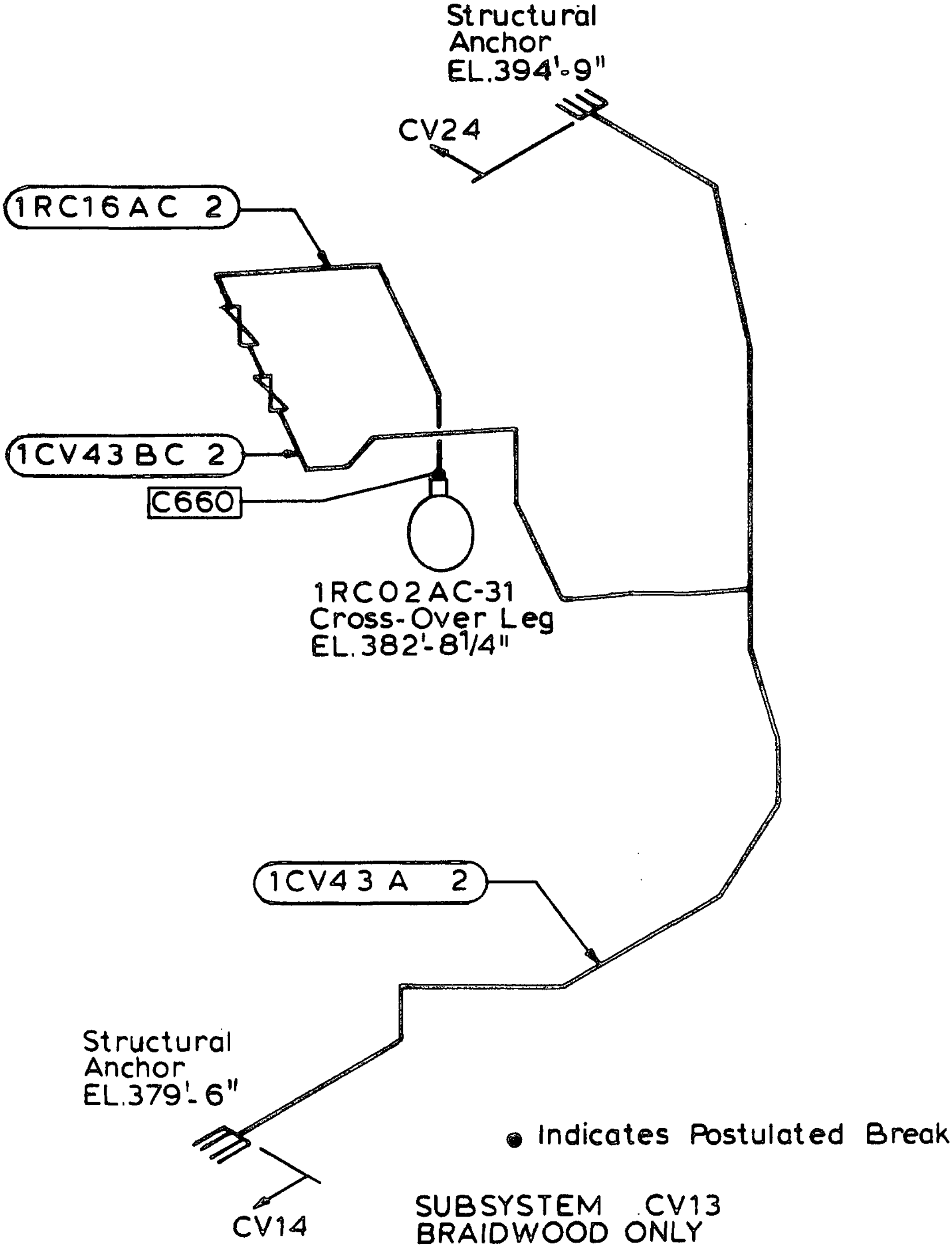


NOTE:
▱ Arbitrary Intermediate Break

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-34

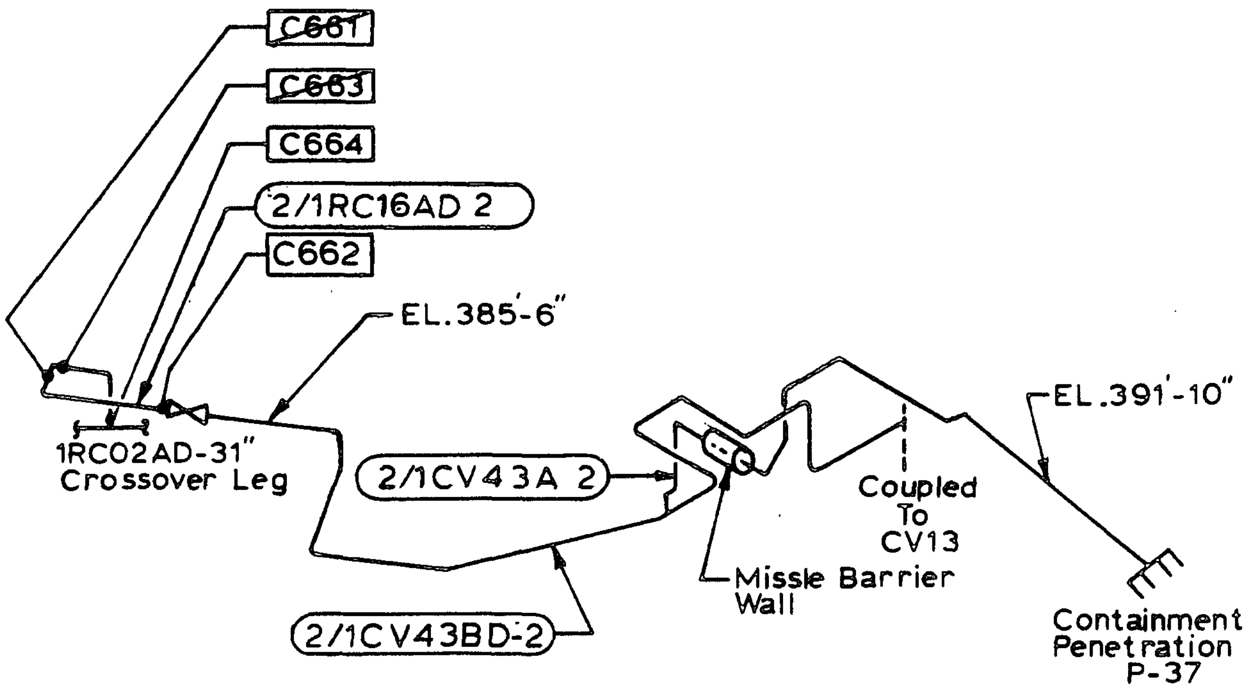
LOOP FILL LINE LOOP 3 SUBSYSTEM CV13
POSTULATED BREAK LOCATIONS
(BYRON)



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-34a

LOOP FILL LINE LOOP 3 SUBSYSTEM CV13
POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



◦ Indicates Postulated Break

SUBSYSTEM CV14
BYRON ONLY

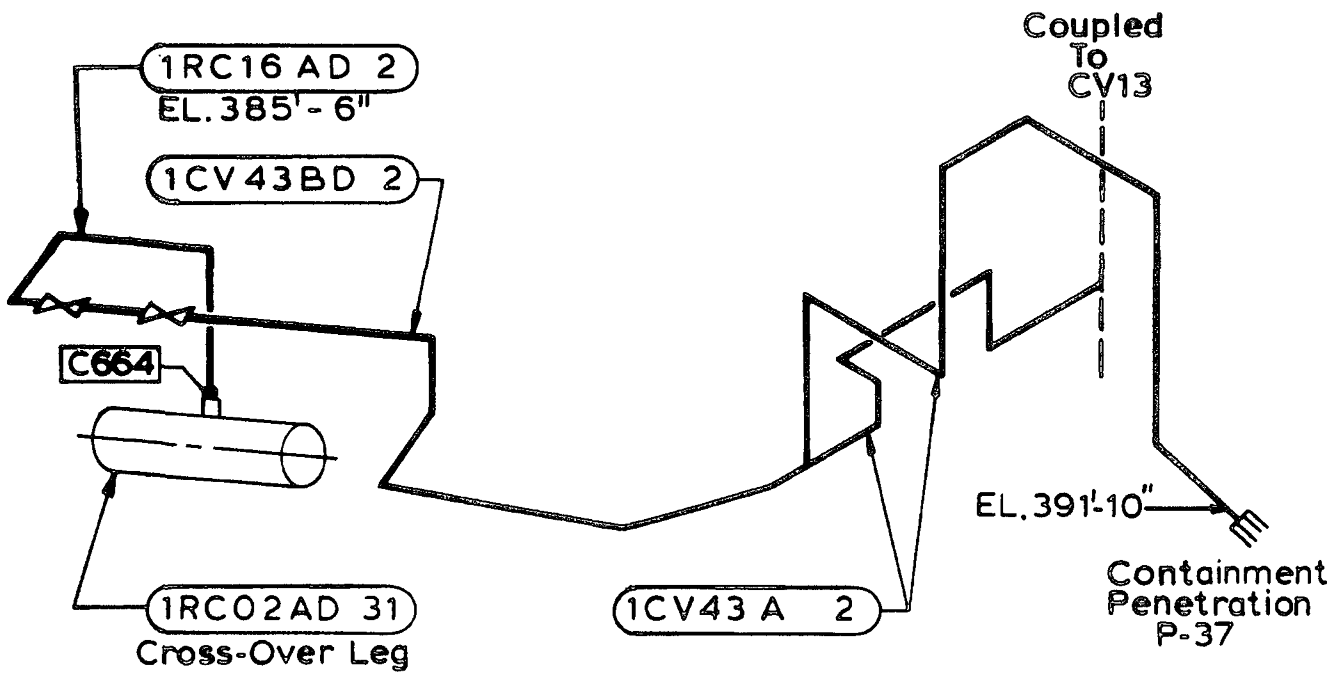
Note:

Arbitrary Intermediate Break

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-35

LOOP FILL LINE LOOP 4 SUBSYSTEM CV14
POSTULATED BREAK LOCATIONS
(BYRON)

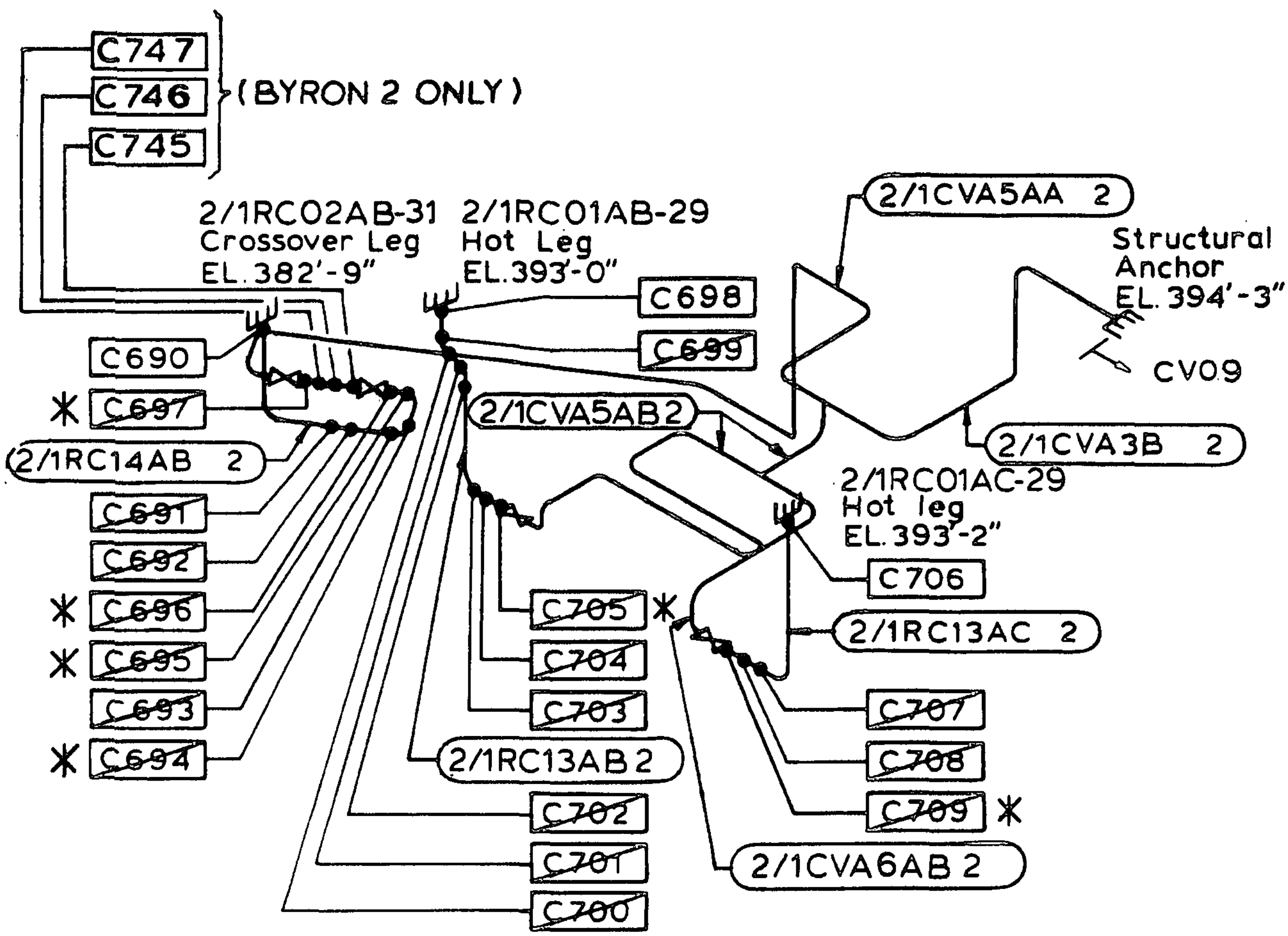


● Indicates Postulated Break

SUBSYSTEM CV14
BRAIDWOOD ONLY

BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-35a LOOP FILL LINE LOOP 4 SUBSYSTEM CV14 POSTULATED BREAK LOCATIONS (BRAIDWOOD)

● Indicates Postulated Break



SUBSYSTEM CV15
BYRON ONLY

Note:

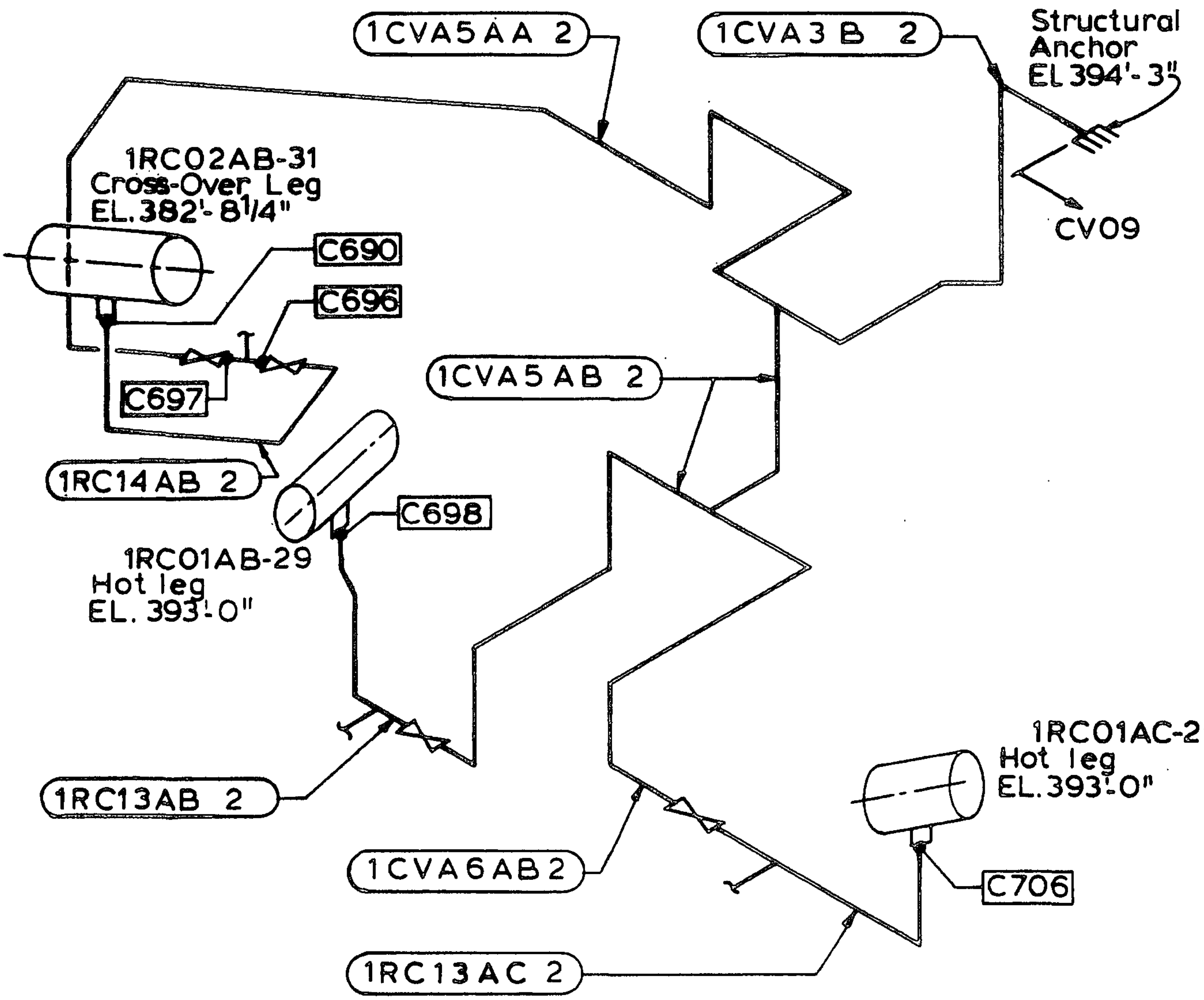
Arbitrary Intermediate Break

✳ Break Restored For Byron 2

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-36

EXCESS LETDOWN LOOPS 2 AND 3 SUBSYSTEM
CV15 POSTULATED BREAK LOCATIONS
(BYRON)



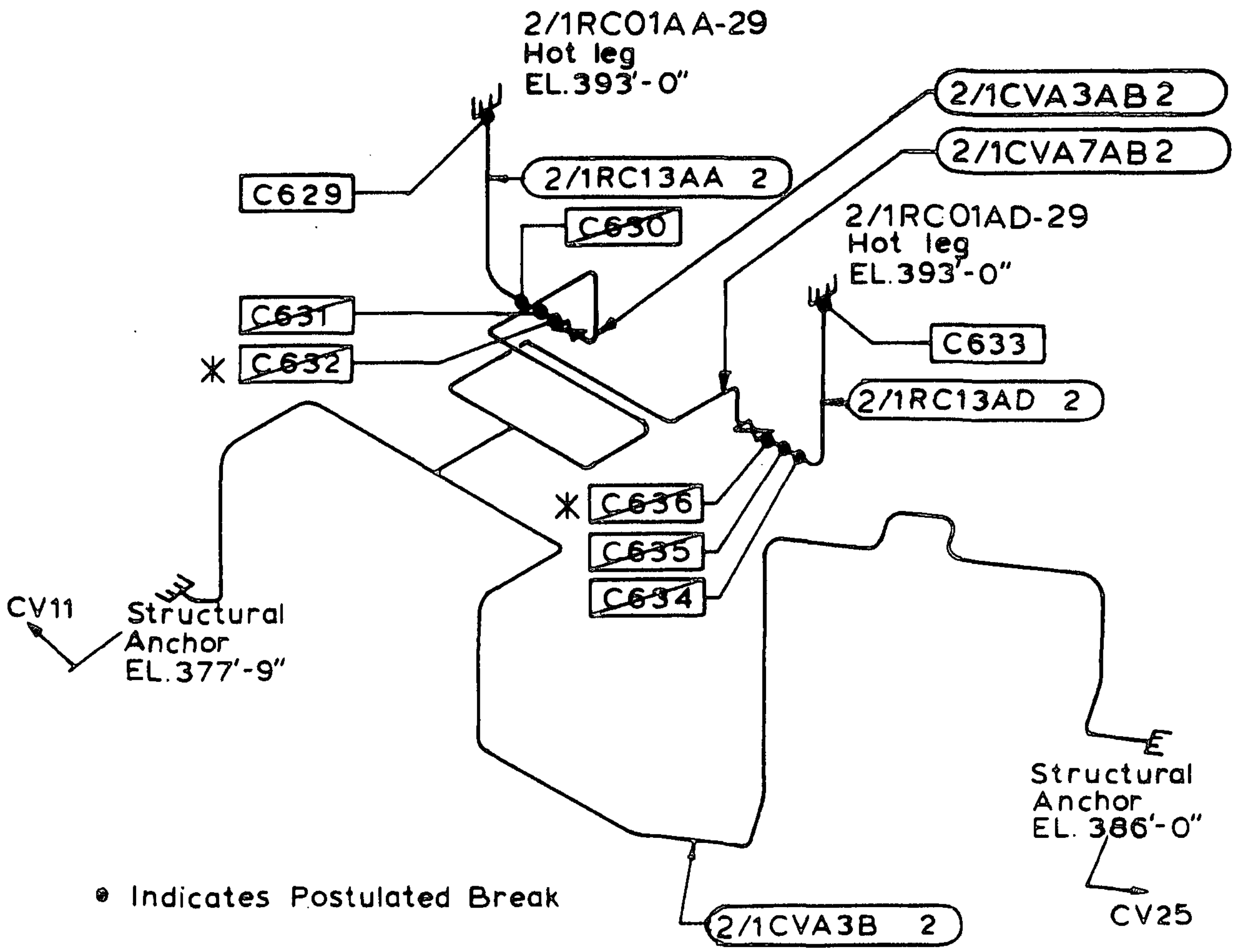
● Indicates Postulated Break

SUBSYSTEM CV15
BRAIDWOOD ONLY

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-36a

EXCESS LETDOWN LOOPS 2 AND 3 SUBSYSTEM CV15 POSTULATED BREAK LOCATIONS (BRAIDWOOD)



SUBSYSTEM CV16
BYRON ONLY

Note:

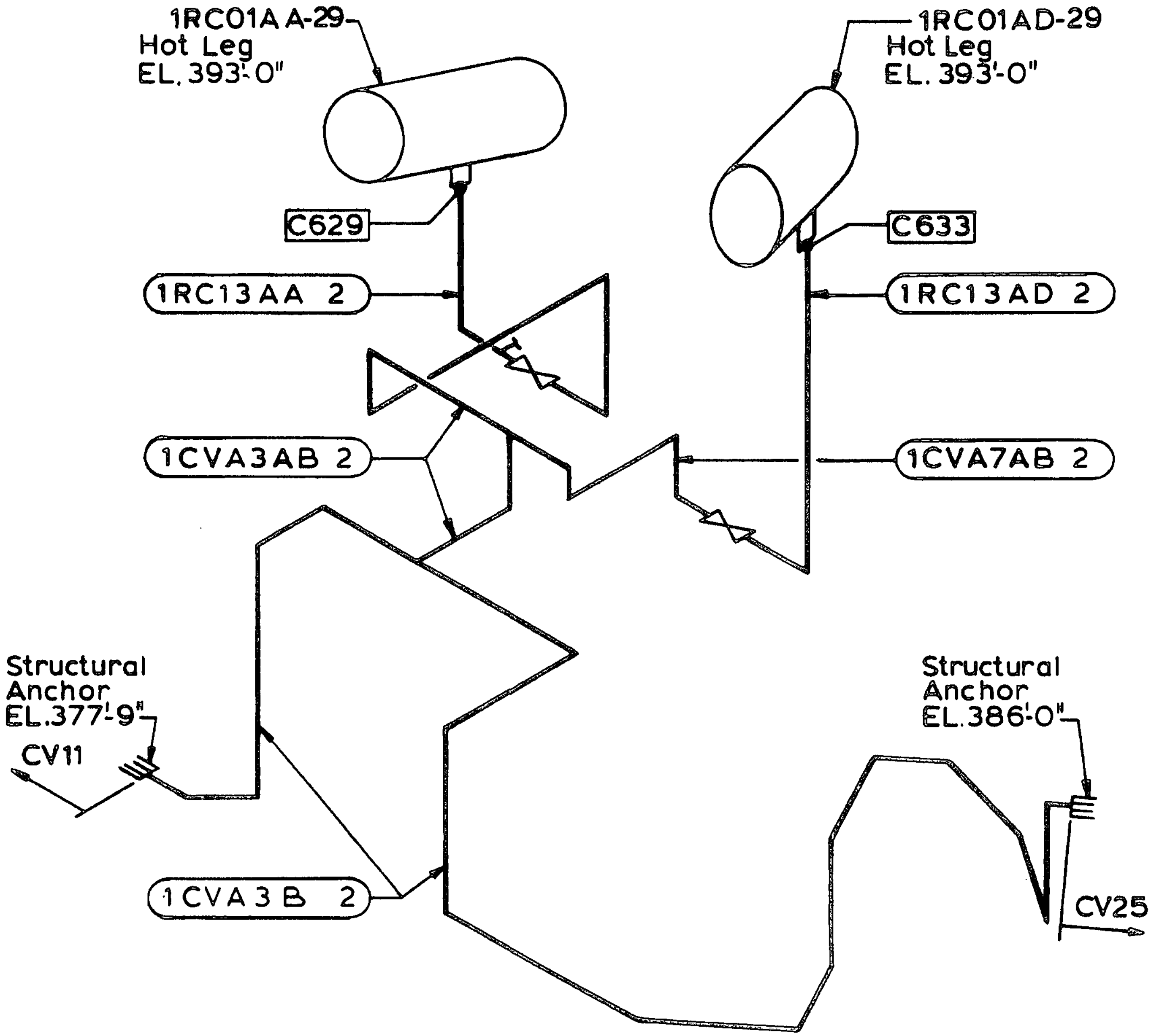
Arbitrary Intermediate Break

Break Restored For Byron 2

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-37

EXCESS LETDOWN LOOPS 1 AND 4 SUBSYSTEM CV16 POSTULATED BREAK LOCATIONS (BYRON)



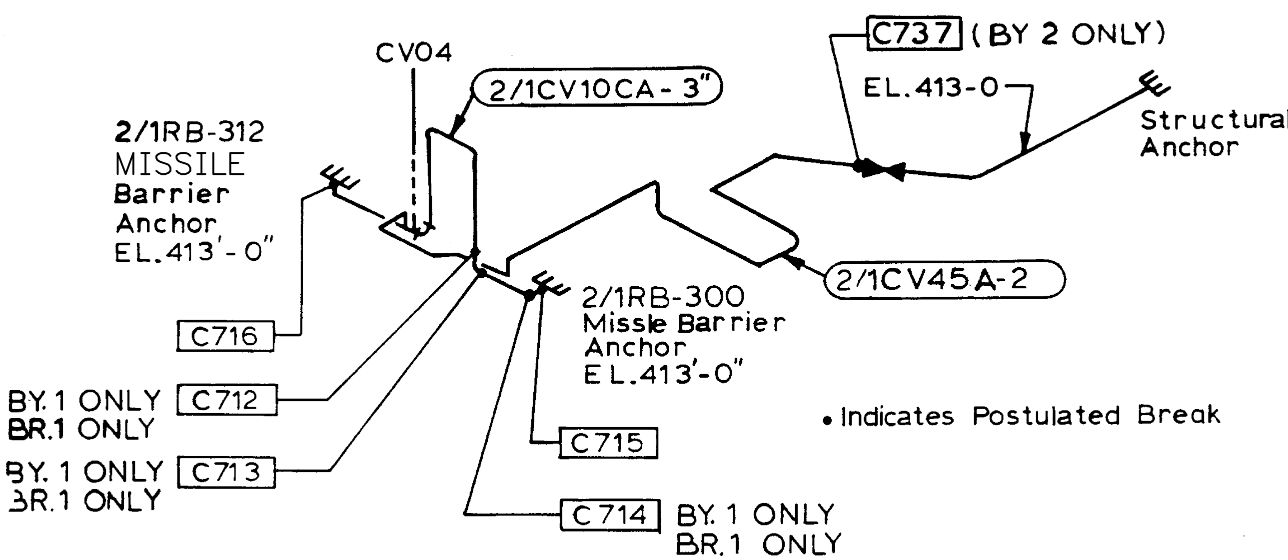
● Indicates Postulated Break

SUBSYSTEM CV16
BRAIDWOOD ONLY

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-37a

EXCESS LETDOWN LOOPS 1 AND 4 SUBSYSTEM
CV16 POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



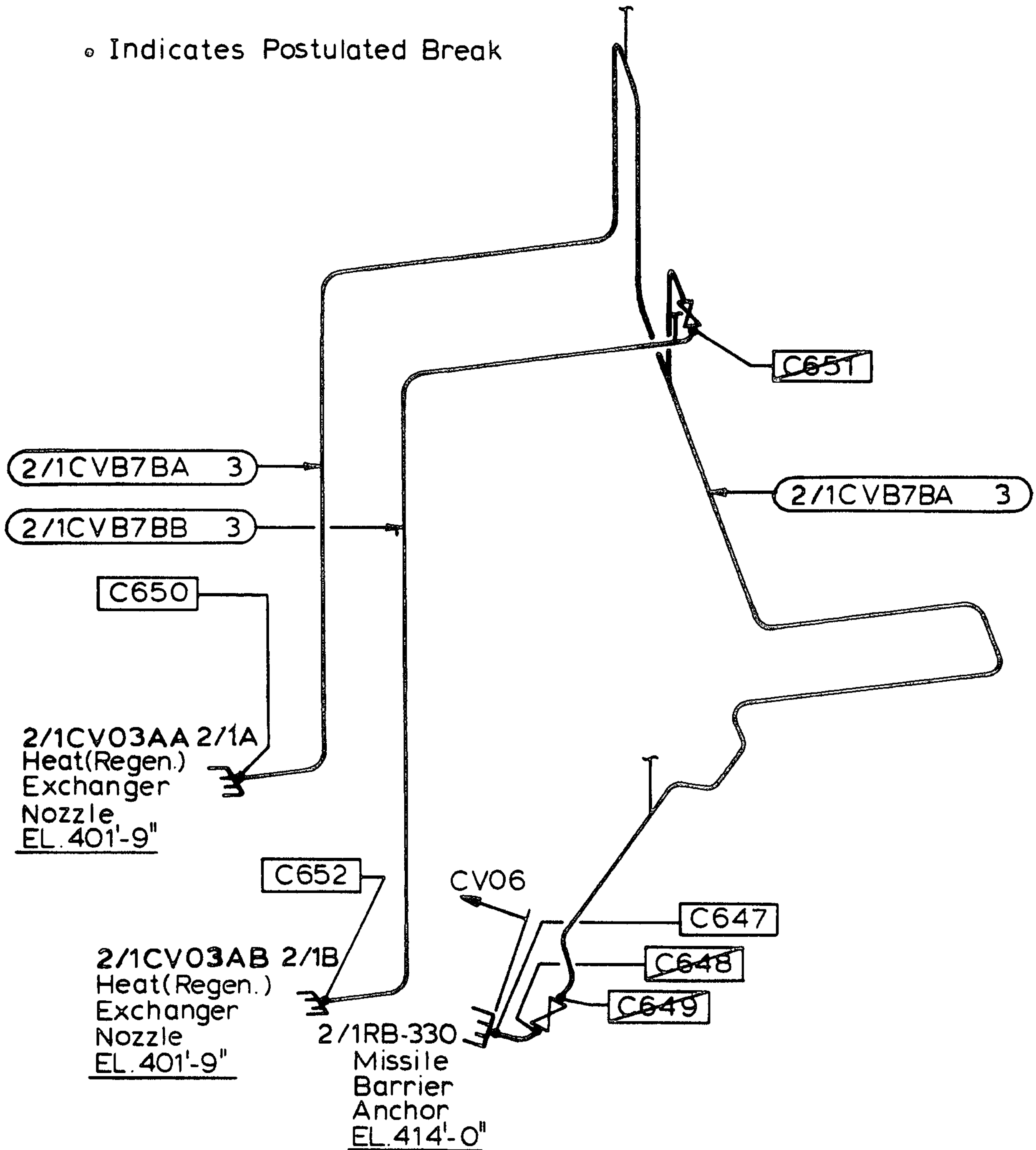
SUBSYSTEM CV22

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-38

CVCS SUBSYSTEM CV22
POSTULATED BREAK LOCATIONS

◦ Indicates Postulated Break



SUBSYSTEM CV23

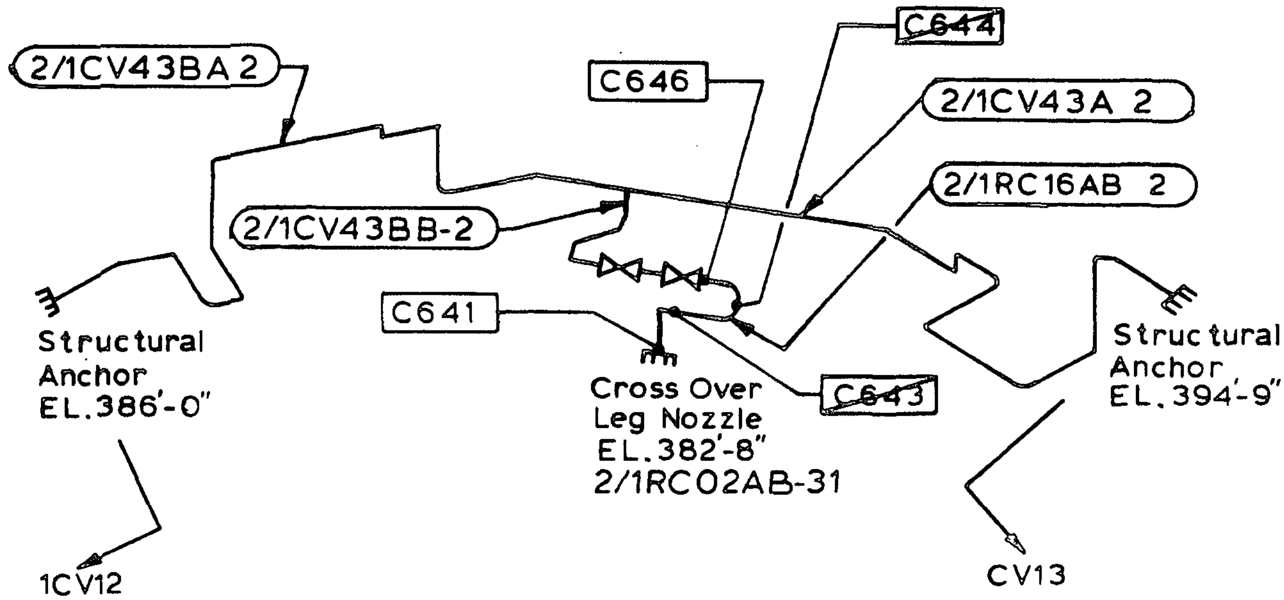
NOTE:

ARBITRARY INTERMEDIATE BREAK

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-39

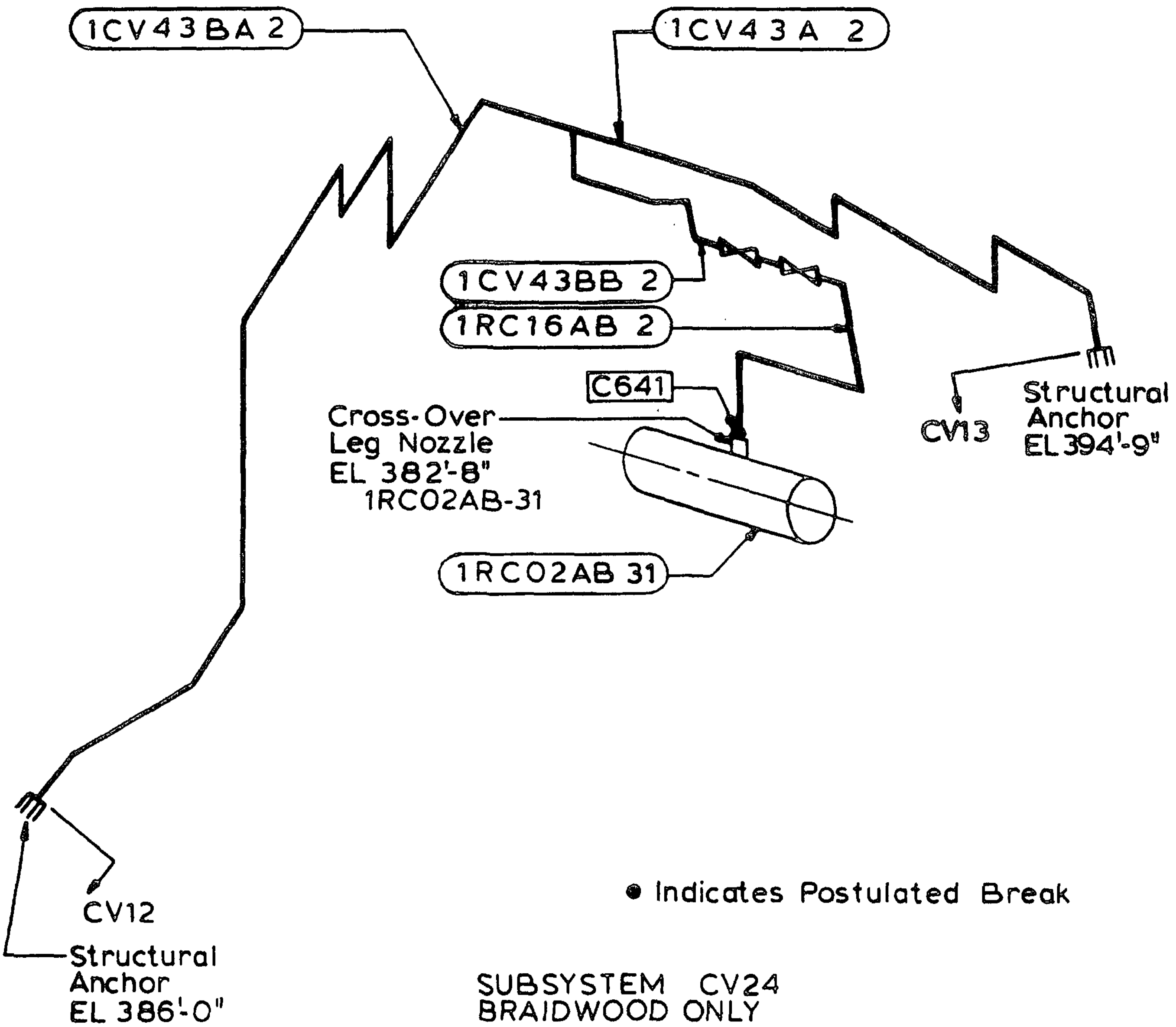
CVCS SUBSYSTEM CV23
POSTULATED BREAK LOCATIONS



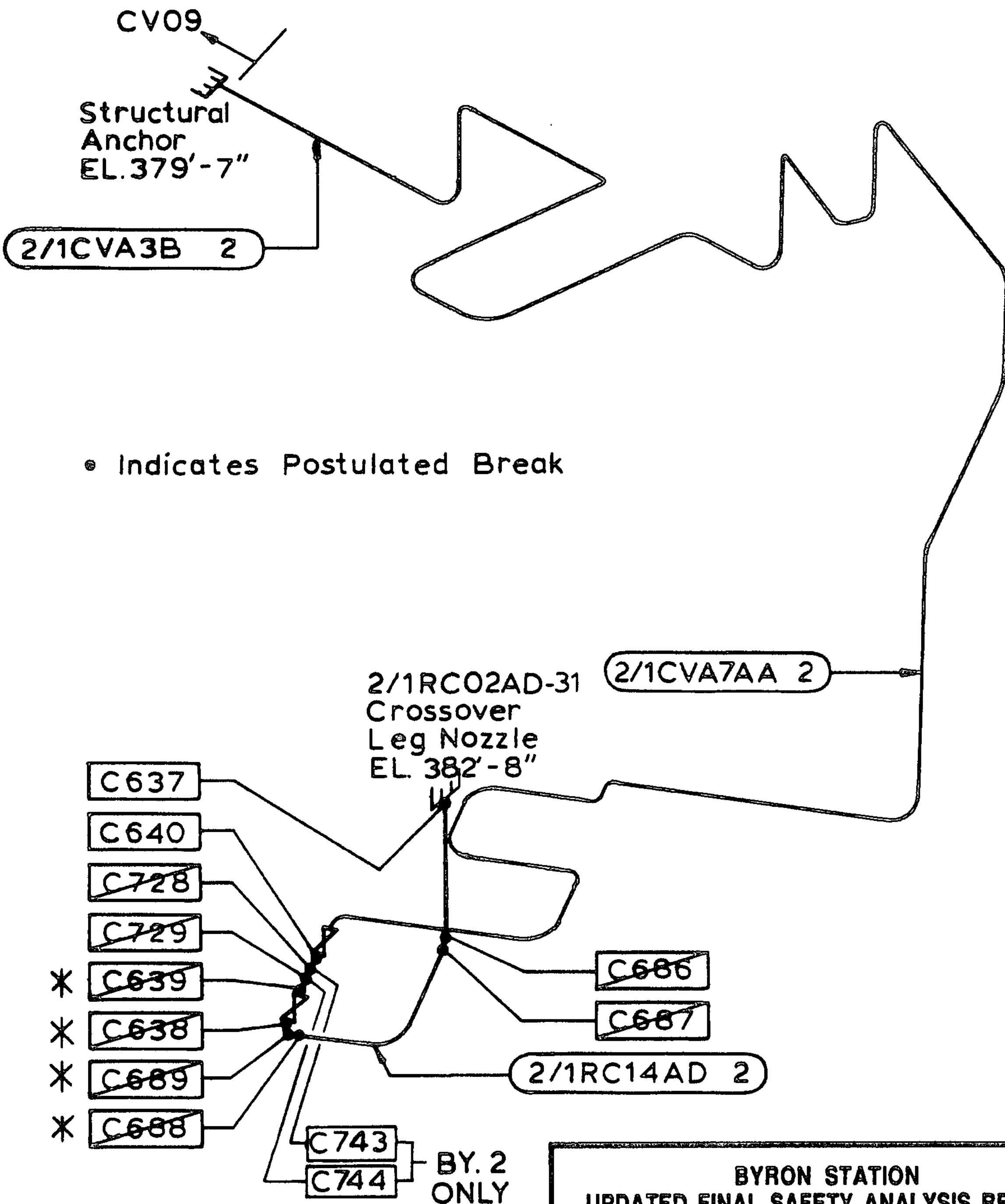
SUBSYSTEM CV24
BYRON ONLY

Note
Arbitrary Intermediate Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-40 LOOP FILL LINE LOOP 2 SUBSYSTEM CV24 POSTULATED BREAK LOCATIONS (BYRON)



BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-40a LOOP FILL LINE LOOP 2 SUBSYSTEM CV24 POSTULATED BREAK LOCATIONS (BRAIDWOOD)



SUBSYSTEM CV25
BYRON ONLY

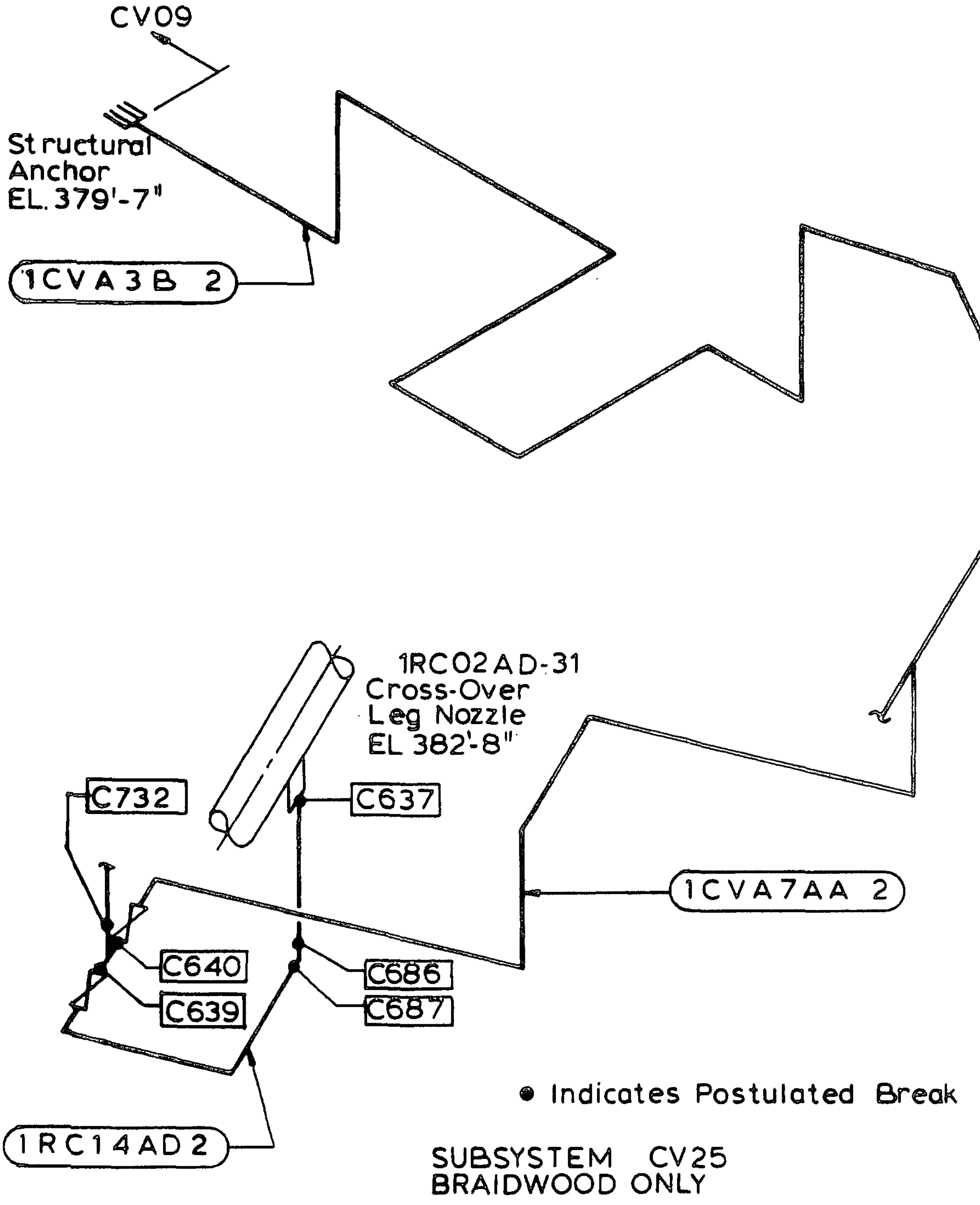
NOTE:
ARBITRARY INTERMEDIATE BREAK

* Break Restored For Byron 2

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-41

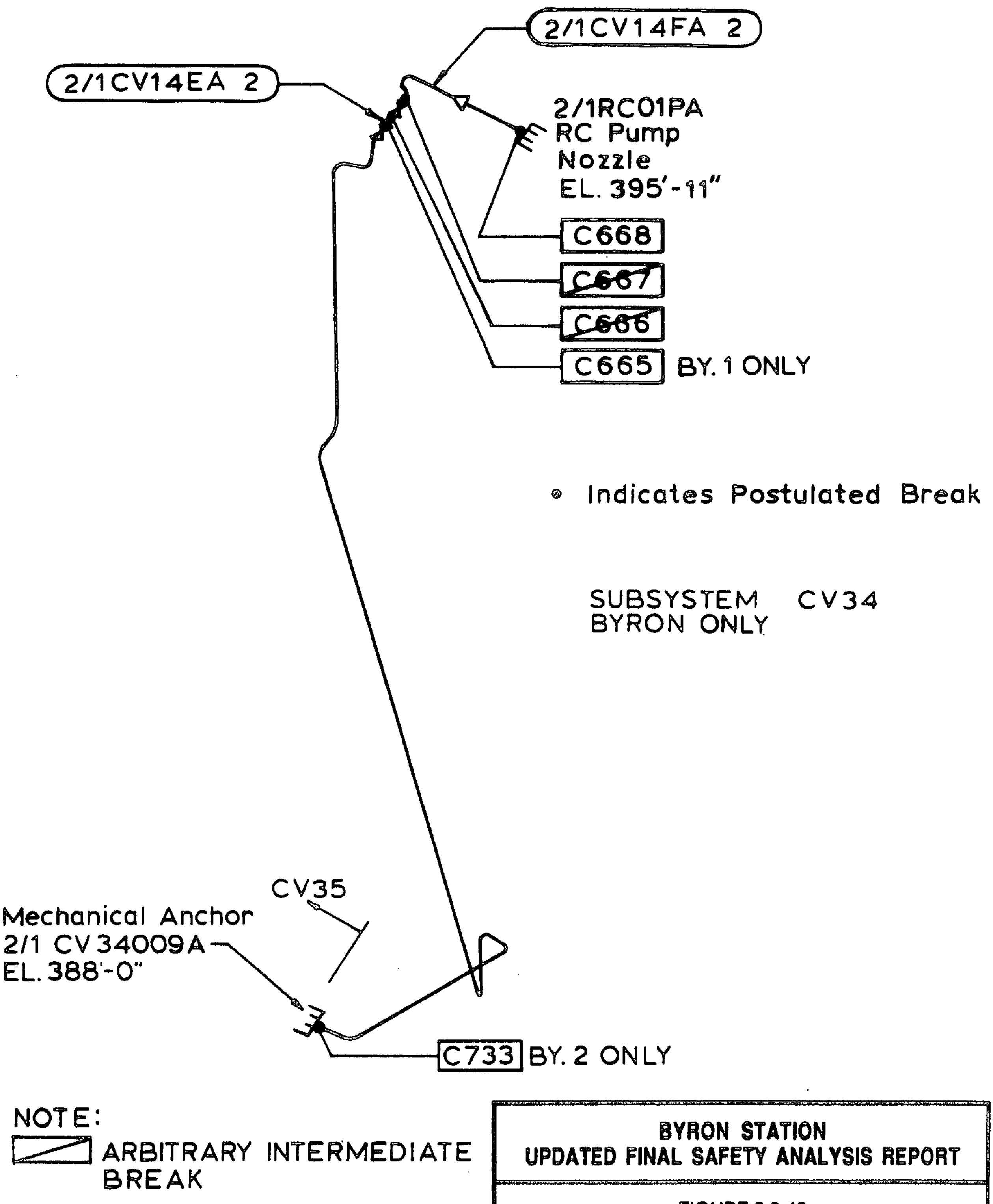
EXCESS LETDOWN LOOP 4 SUBSYSTEM CV25
POSTULATED BREAK LOCATIONS
(BYRON)



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-41a

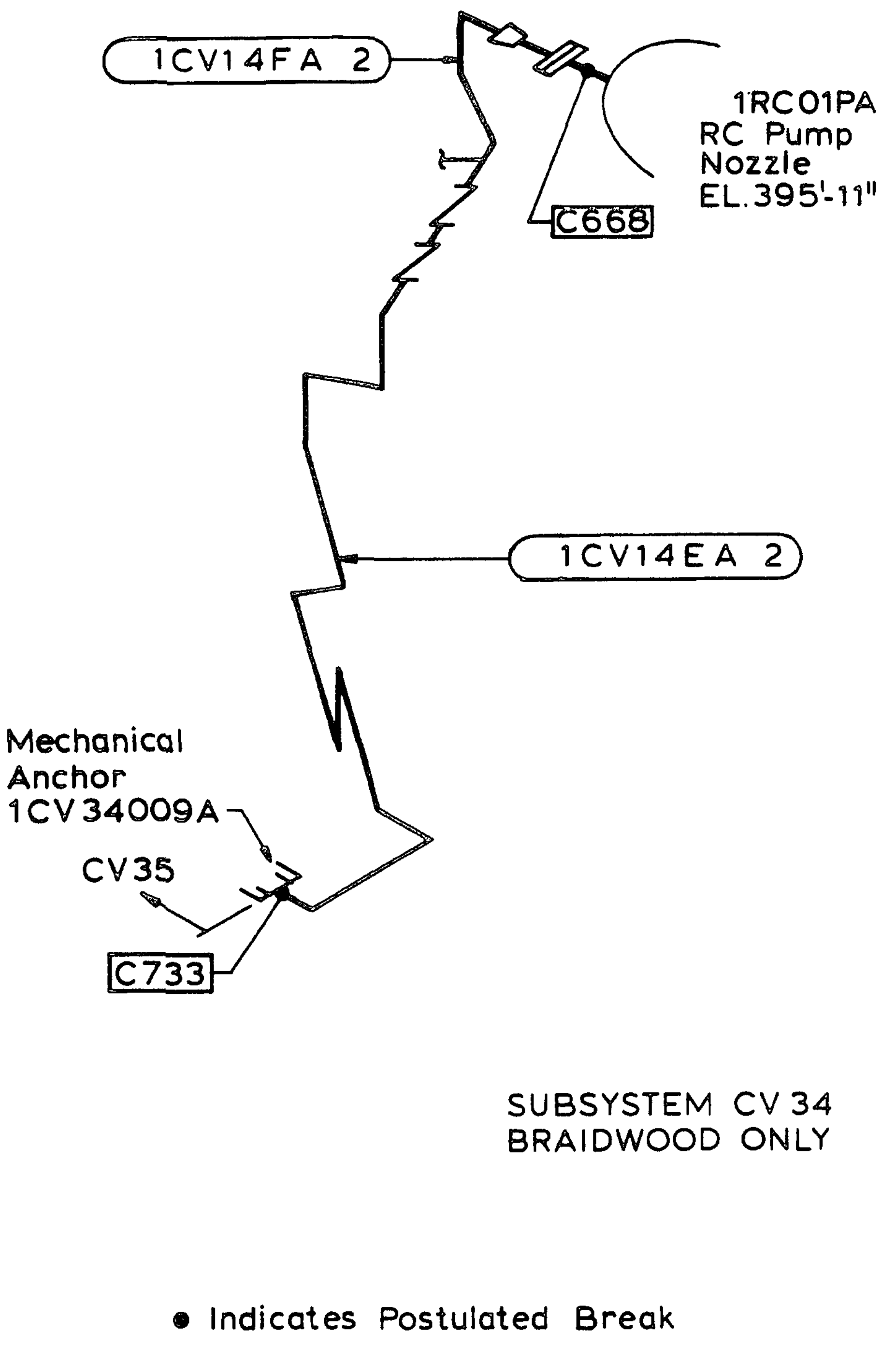
EXCESS LETDOWN LOOP 4 SUBSYSTEM CV25
POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-42

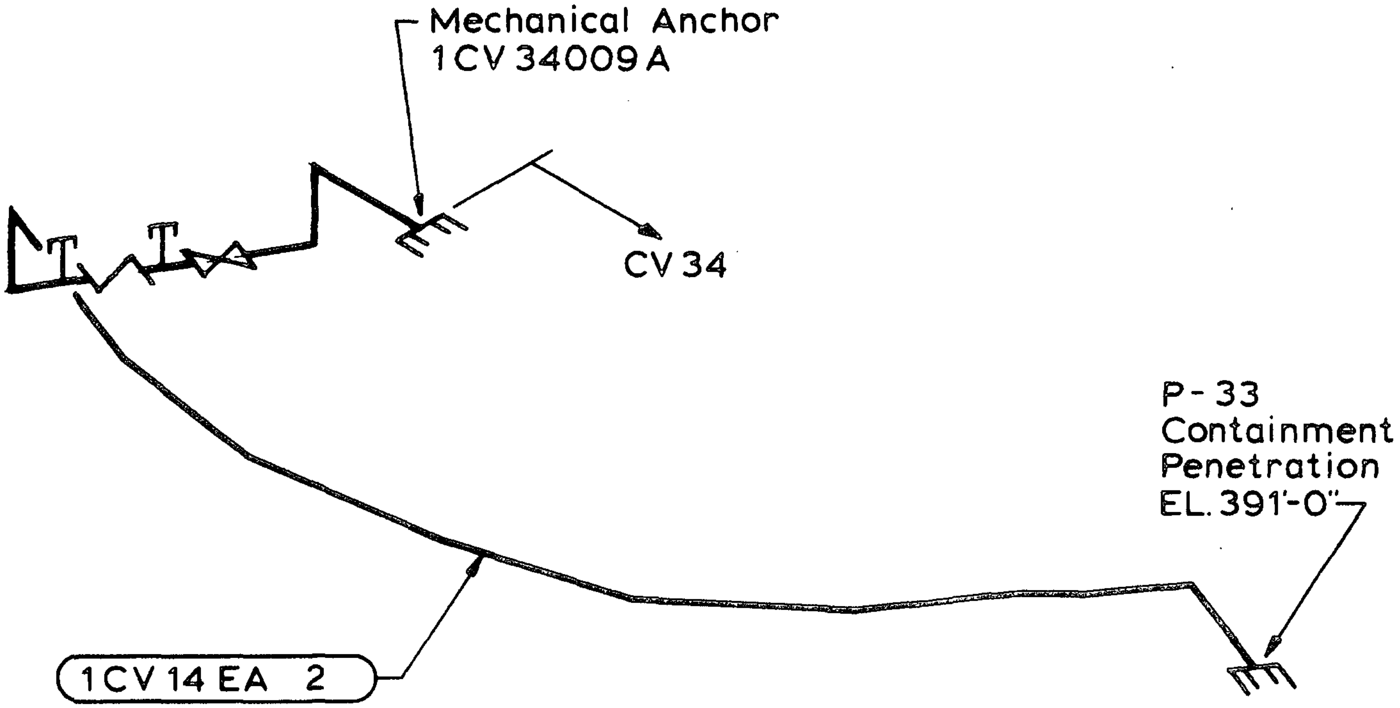
RCP SEAL WATER INJECTION LOOP 1 SUBSYSTEM
CV34 POSTULATED BREAK LOCATIONS
(BYRON)



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-42a

RCP SEAL WATER INJECTION LOOP 1 SUBSYSTEM
CV34 POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



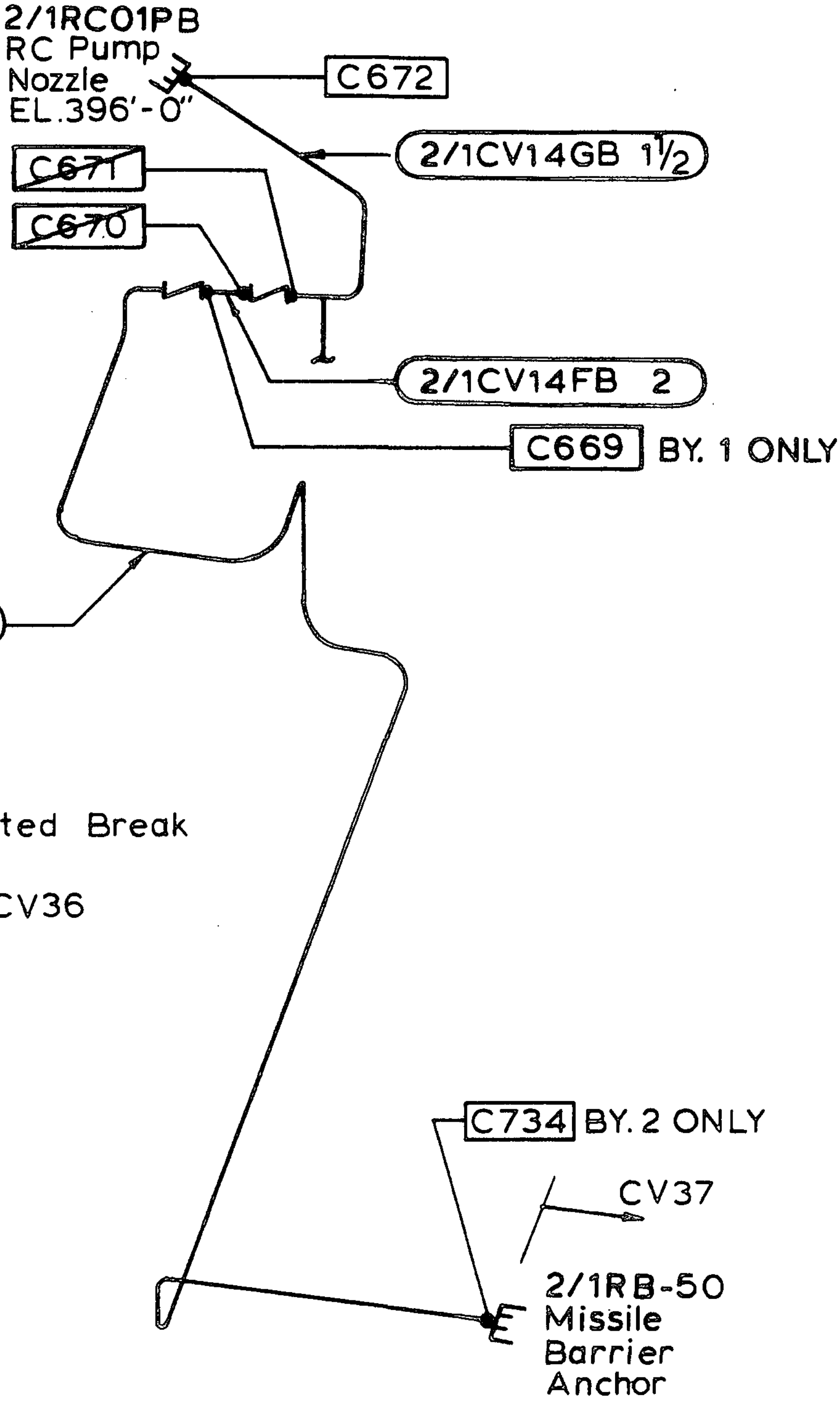
SUBSYSTEM CV 35
BRAIDWOOD ONLY

NOTE:
Breaks Are Postulated At All Fittings Between Anchor 1CV34009A & Penetration P-33.

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-42b

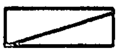
RCP SEAL WATER INJECTION LOOP 1 SUBSYSTEM CV35 POSTULATED BREAK LOCATIONS (BRAIDWOOD)



● Indicates Postulated Break

SUBSYSTEM CV36
BYRON ONLY

Note

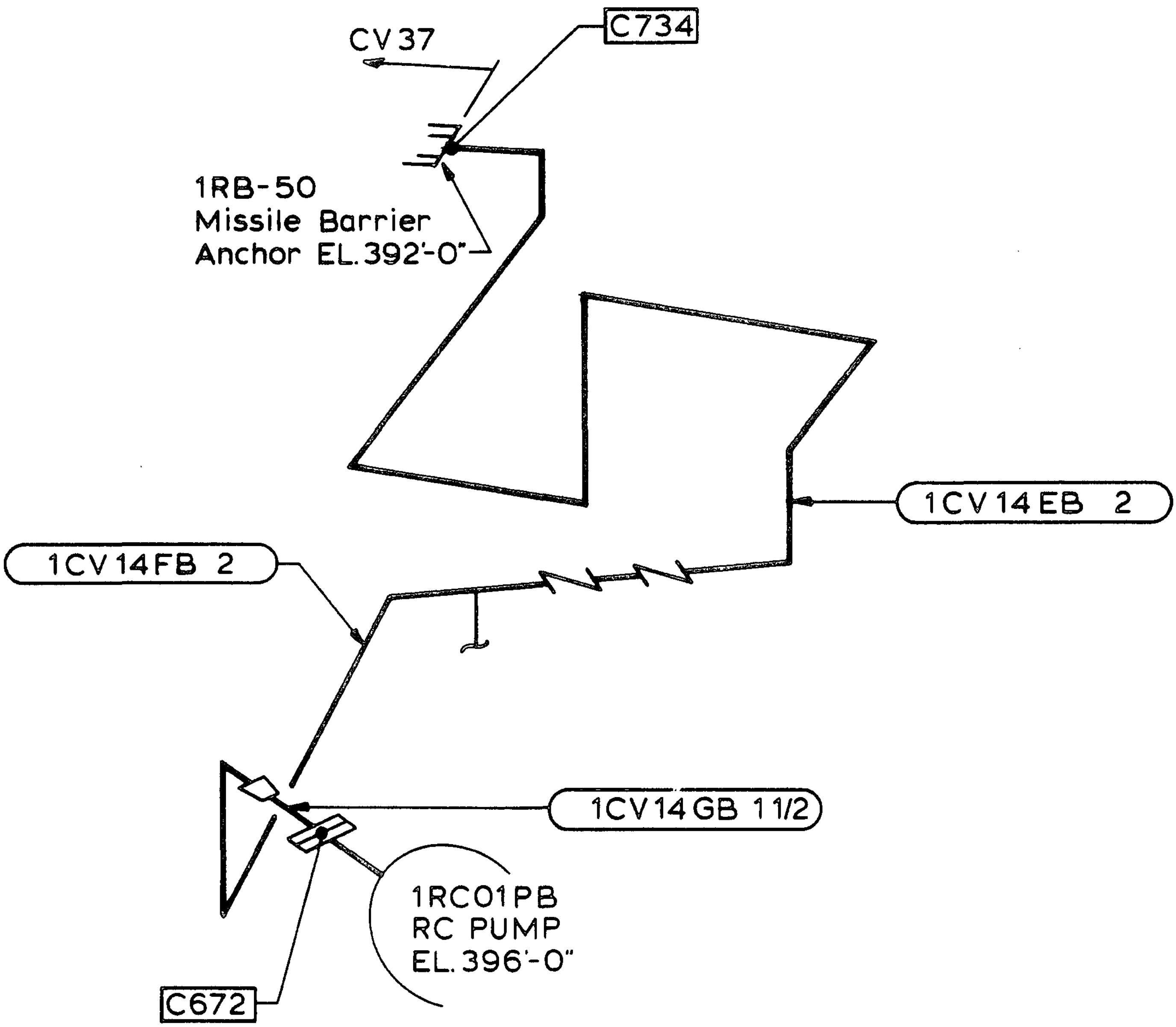


Arbitrary Intermediate Break.

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-43

RCP SEAL WATER INJECTION LOOP 2 SUBSYSTEM
CV36 POSTULATED BREAK LOCATIONS
(BYRON)



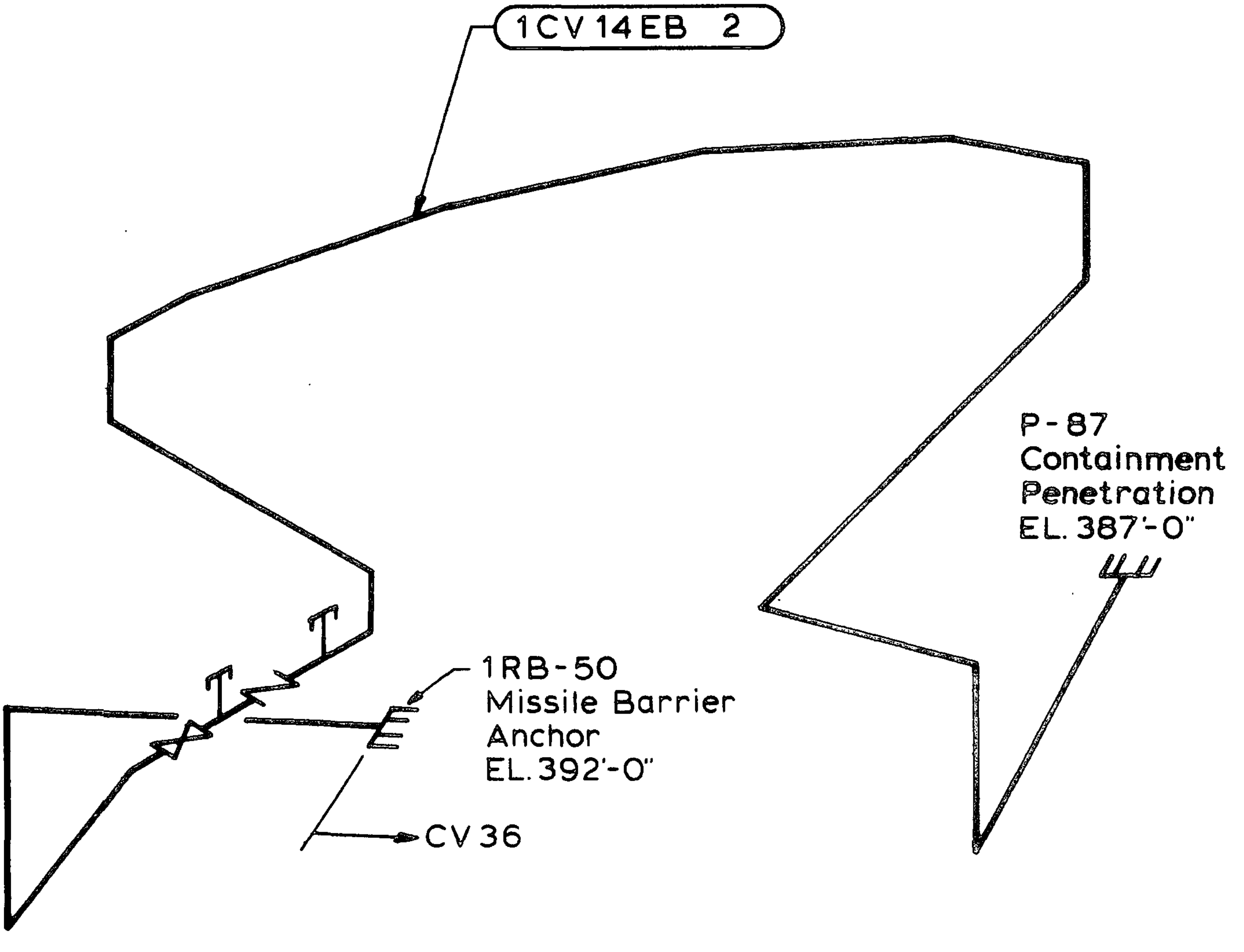
SUBSYSTEM CV36
BRAIDWOOD ONLY

● Indicates Postulated Break

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-43a

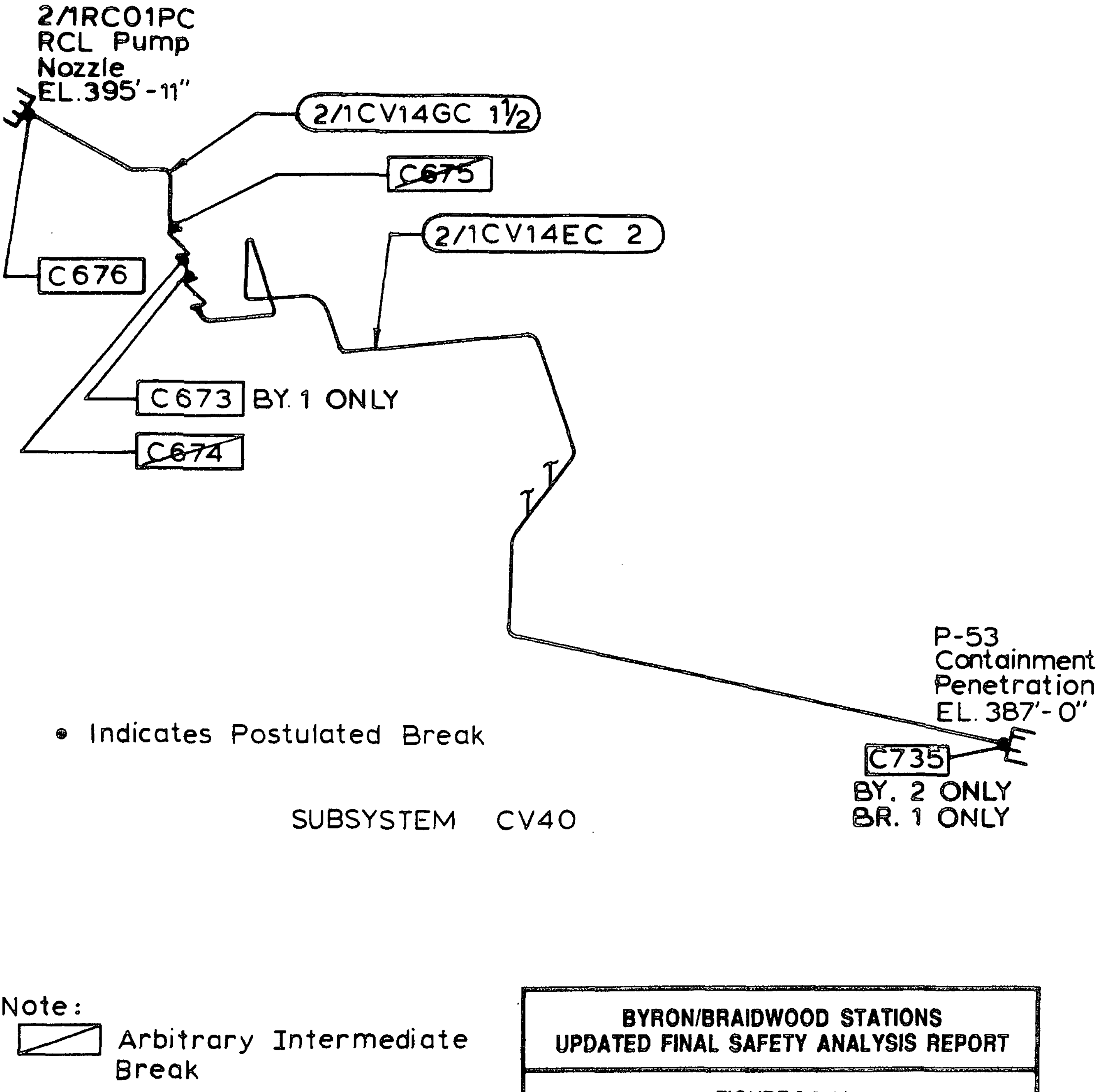
RCP SEAL WATER INJECTION LOOP 2 SUBSYSTEM
CV36 POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



SUBSYSTEM CV 37
BRAIDWOOD ONLY

NOTE:
Breaks Are Postulated At All Fittings Between Missile Barrier Anchor 1RB-50 & Penetration P-87.

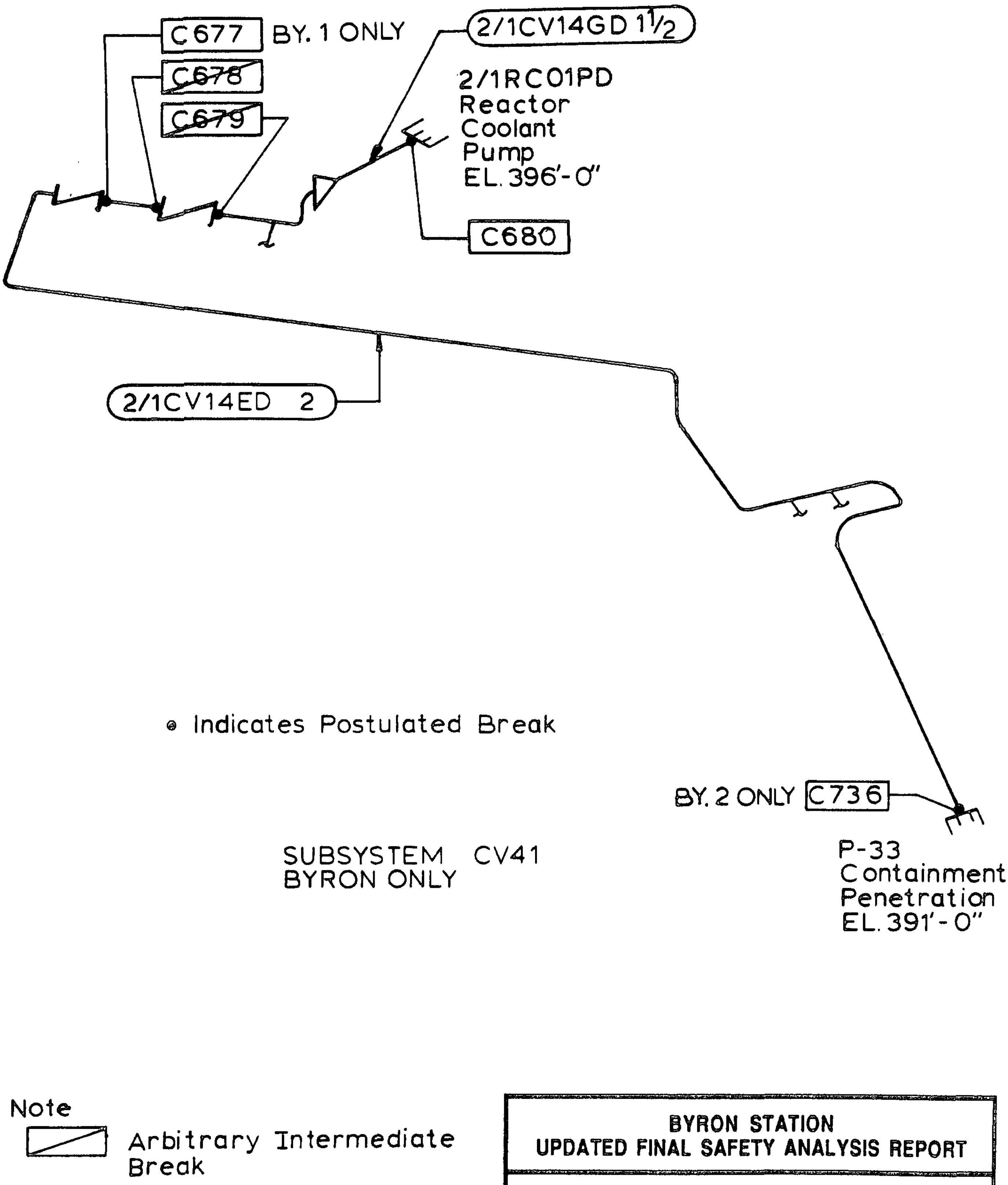
BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-43b RCP SEAL WATER INJECTION LOOP 2 SUBSYSTEM CV37 POSTULATED BREAK LOCATIONS (BRAIDWOOD)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-44

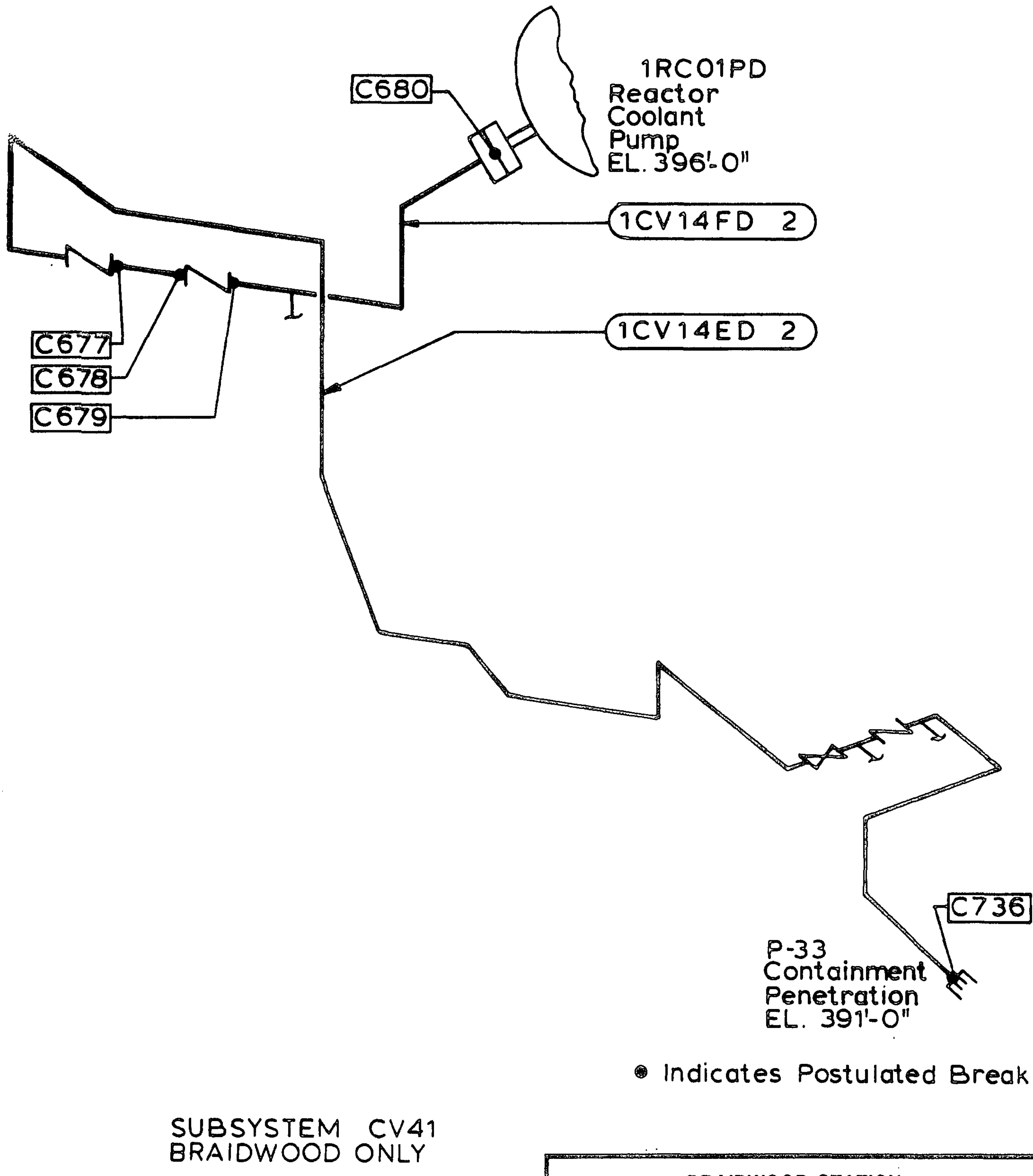
RCP SEAL WATER INJECTION LOOP 3 SUBSYSTEM CV40 POSTULATED BREAK LOCATIONS



BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-45

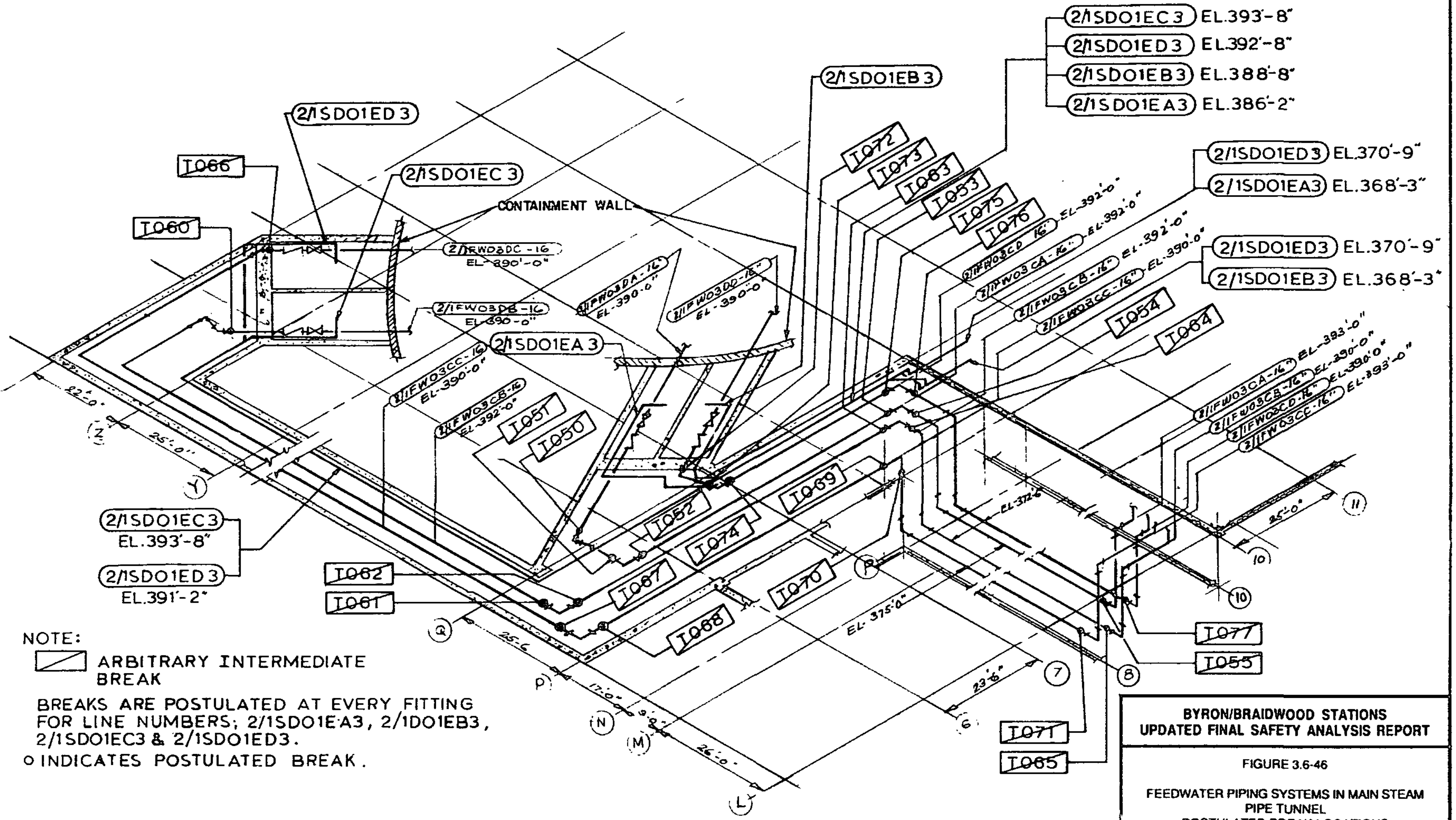
RCP SEAL WATER INJECTION LOOP 4 SUBSYSTEM CV41 POSTULATED BREAK LOCATIONS (BYRON)



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-45a

RCP SEAL WATER INJECTION LOOP 4 SUBSYSTEM CV41 POSTULATED BREAK LOCATIONS (BRAIDWOOD)



NOTE:

ARBITRARY INTERMEDIATE BREAK

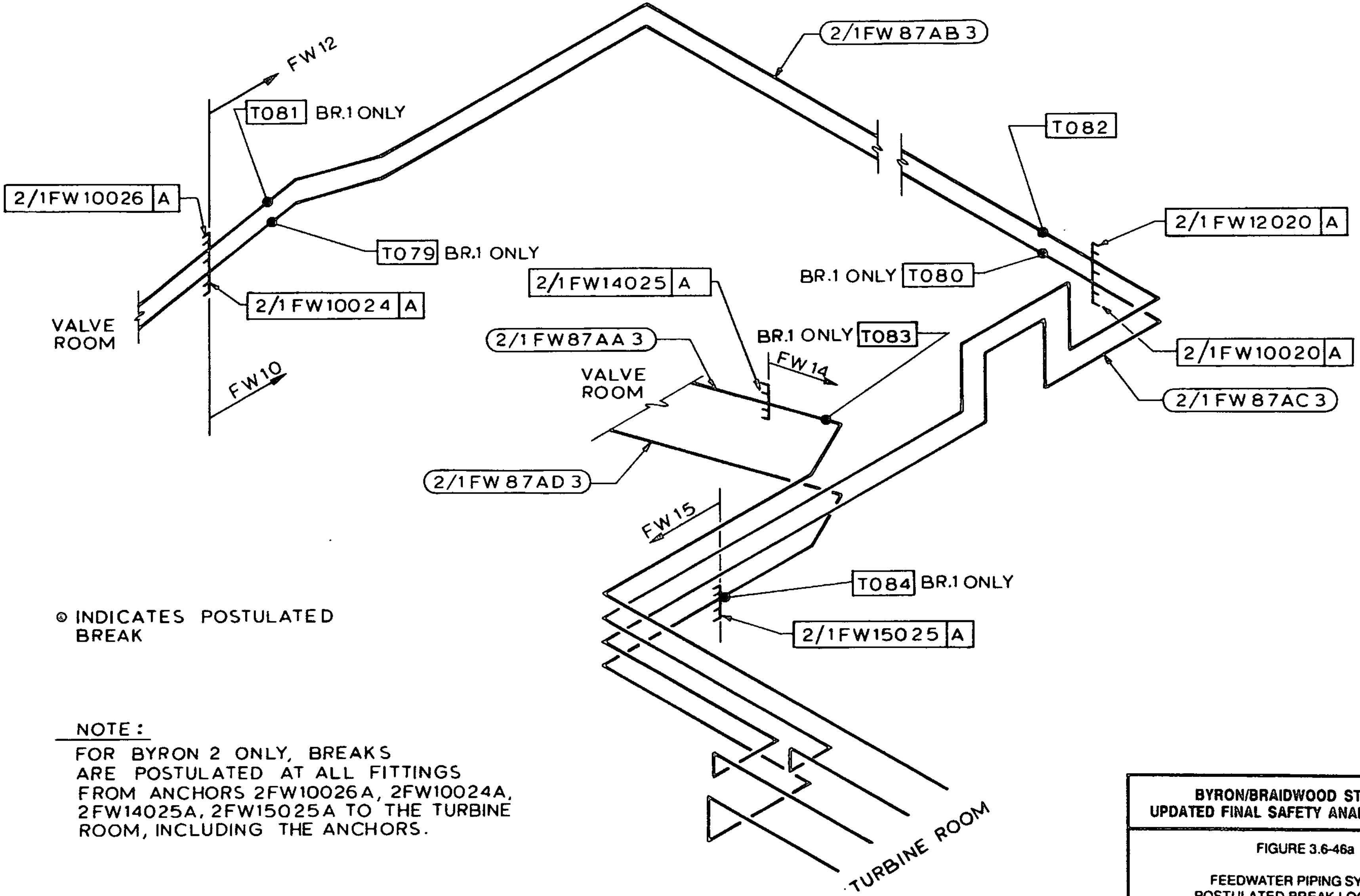
BREAKS ARE POSTULATED AT EVERY FITTING FOR LINE NUMBERS; 2/1SDO1EA3, 2/1DO1EB3, 2/1SDO1EC3 & 2/1SDO1ED3.

○ INDICATES POSTULATED BREAK.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-46

FEEDWATER PIPING SYSTEMS IN MAIN STEAM PIPE TUNNEL
POSTULATED BREAK LOCATIONS



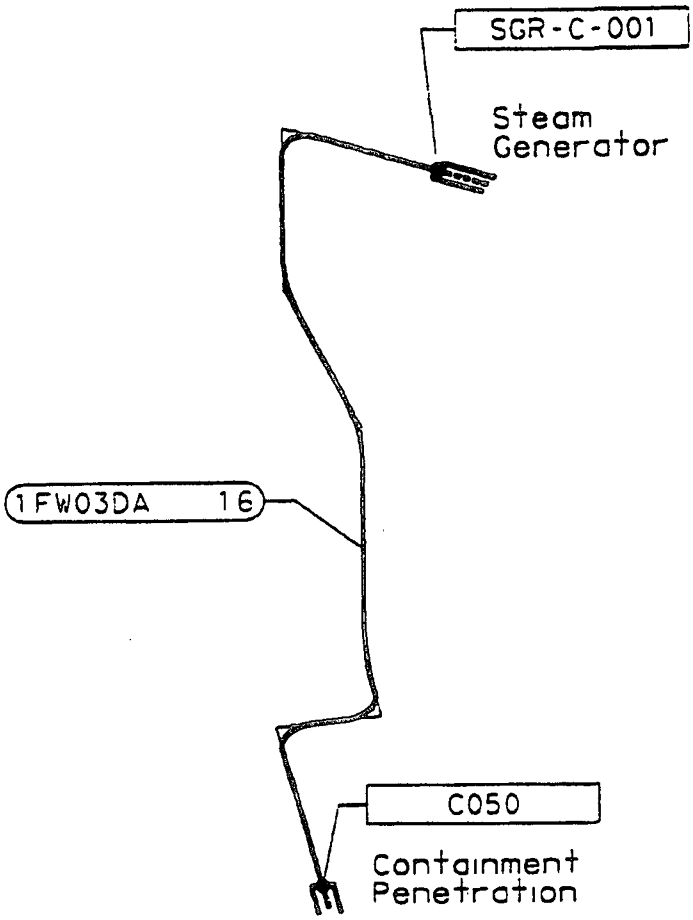
⊗ INDICATES POSTULATED BREAK

NOTE:

FOR BYRON 2 ONLY, BREAKS ARE POSTULATED AT ALL FITTINGS FROM ANCHORS 2FW10026A, 2FW10024A, 2FW14025A, 2FW15025A TO THE TURBINE ROOM, INCLUDING THE ANCHORS.

BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-46a FEEDWATER PIPING SYSTEM POSTULATED BREAK LOCATIONS

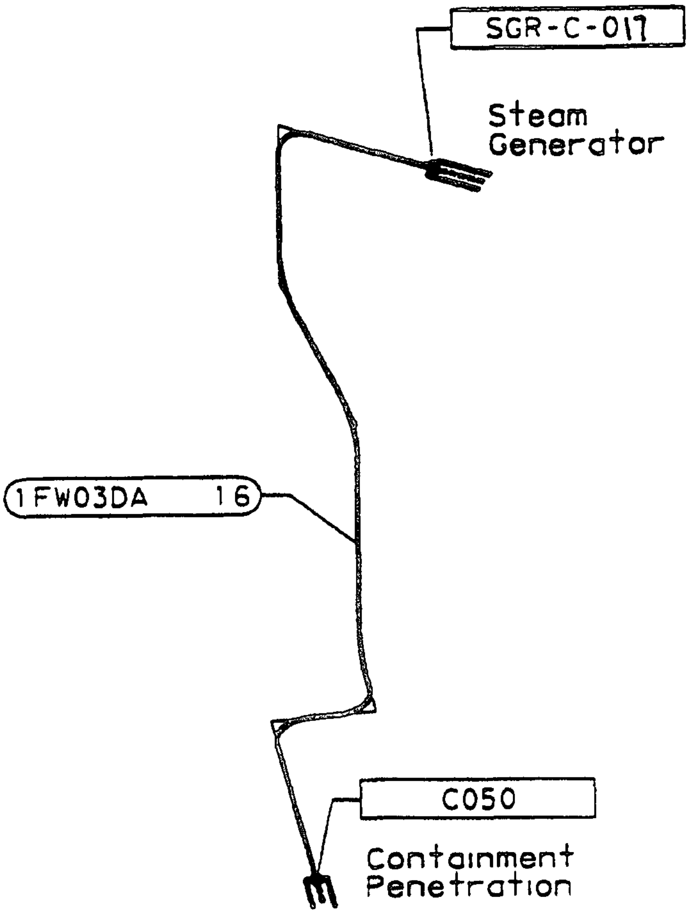
REVISION 7
DECEMBER 1998



SUBSYSTEM 1FW02

● Indicates Postulated Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-47 FEEDWATER LOOP 1 SUBSYSTEM 1FW02 POSTULATED BREAK LOCATIONS-UNIT 1

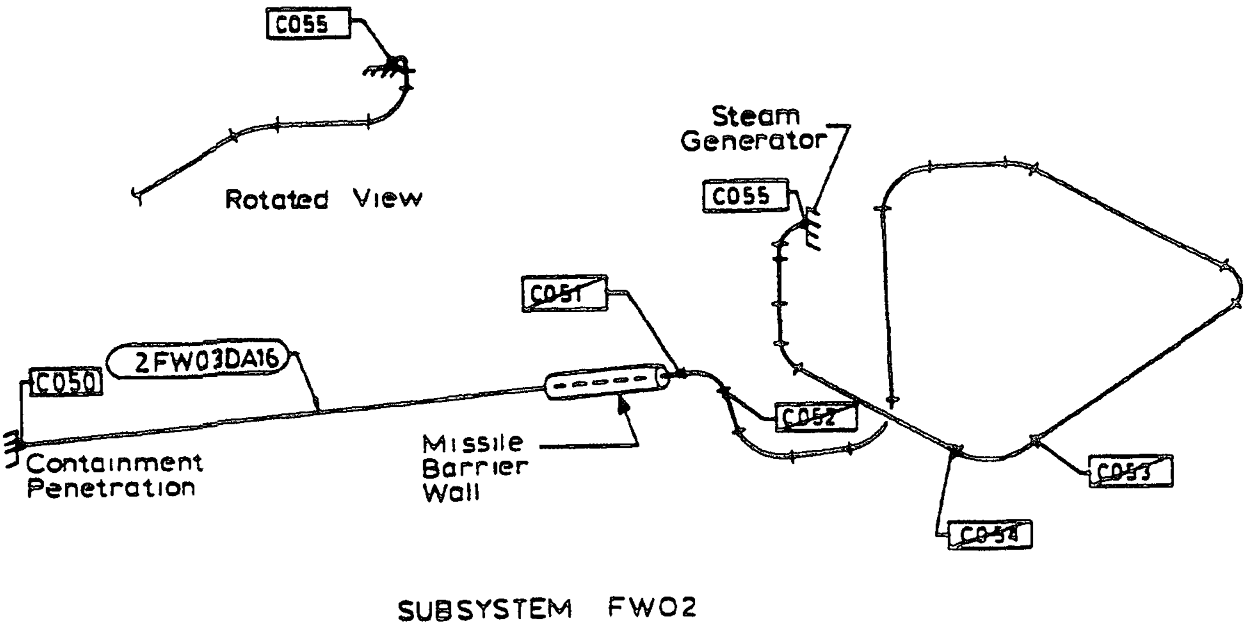


SUBSYSTEM 1FW02

● Indicates Postulated Break

BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-47a FEEDWATER LOOP 1 SUBSYSTEM 1FW02 POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998

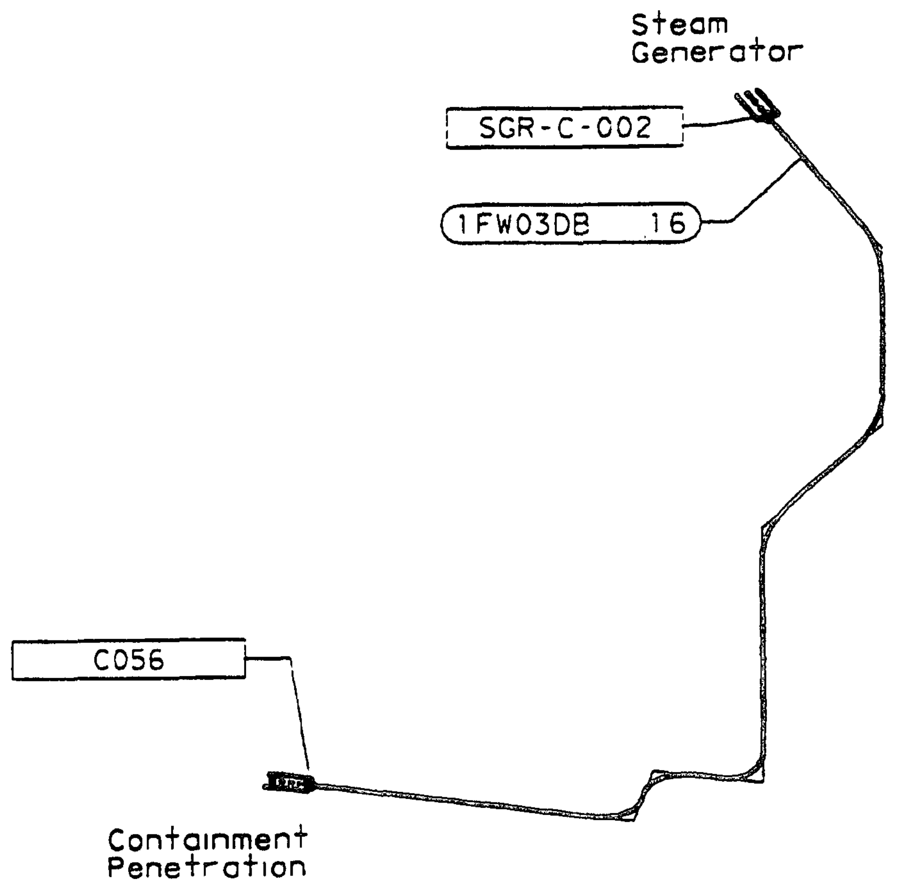


⊙ Indicates Postulated Break

Note
Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-47b FEEDWATER LOOP 1 POSTULATED BREAK LOCATIONS - UNIT 2

REVISION 7
DECEMBER 1998

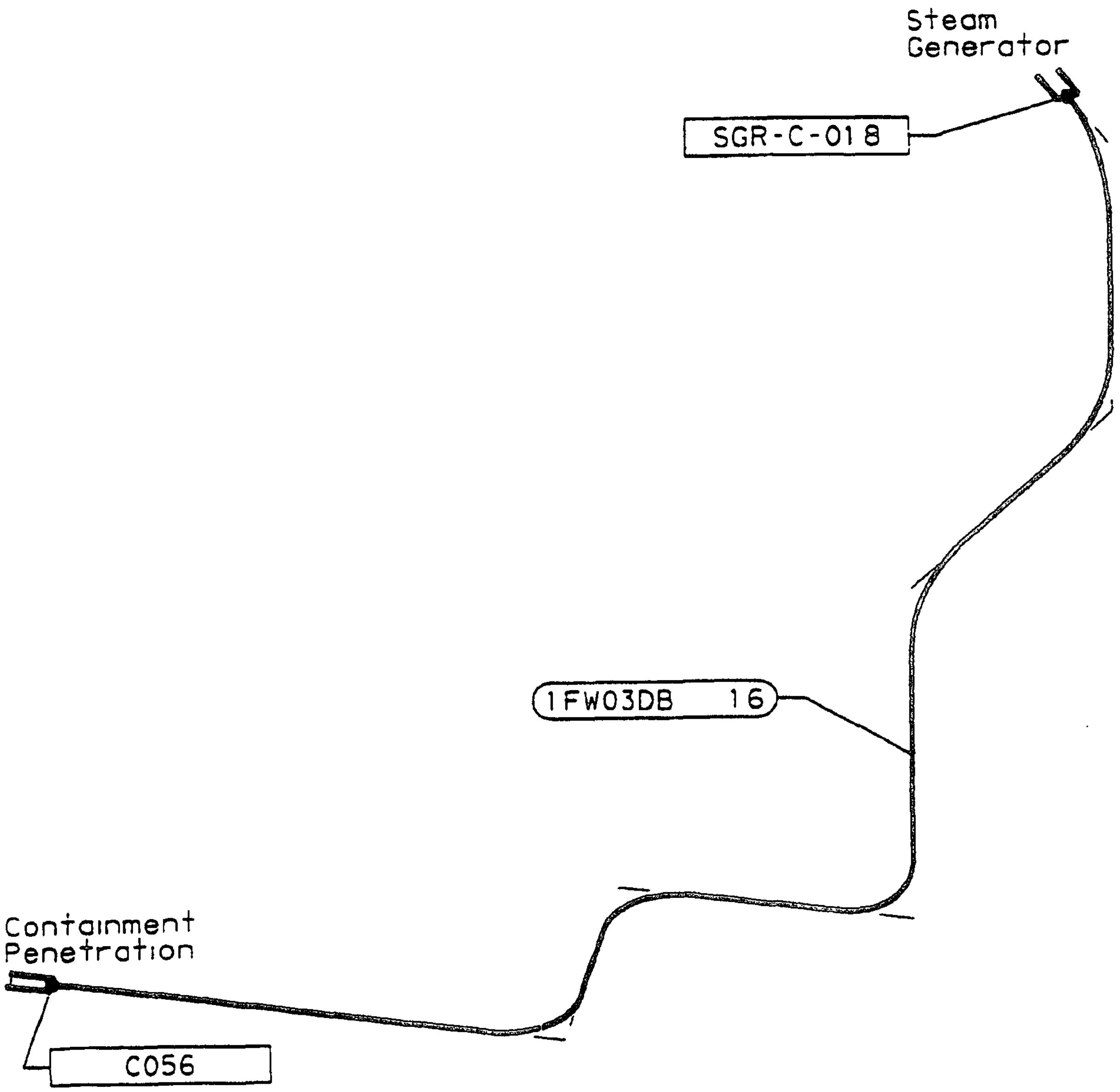


SUBSYSTEM 1FW03

● Indicates Postulated Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-48 FEEDWATER LOOP 2 SUBSYSTEM 1FW03 POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998

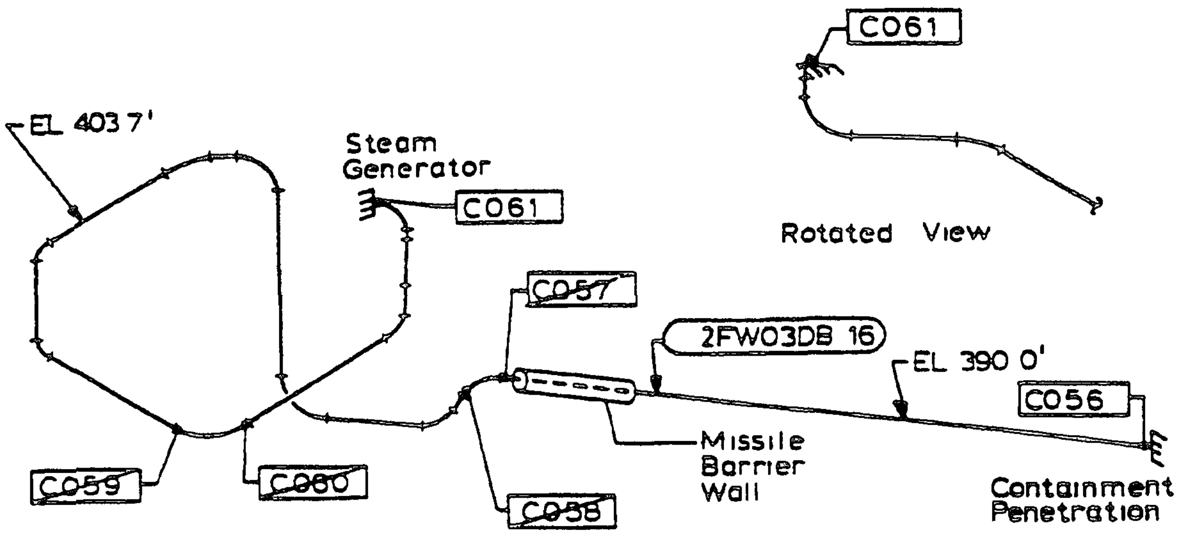


SUBSYSTEM 1FW03

● Indicates Postulated Break

BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-48c FEEDWATER LOOP 2 SUBSYSTEM 1FW03 POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998



SUBSYSTEM FW03

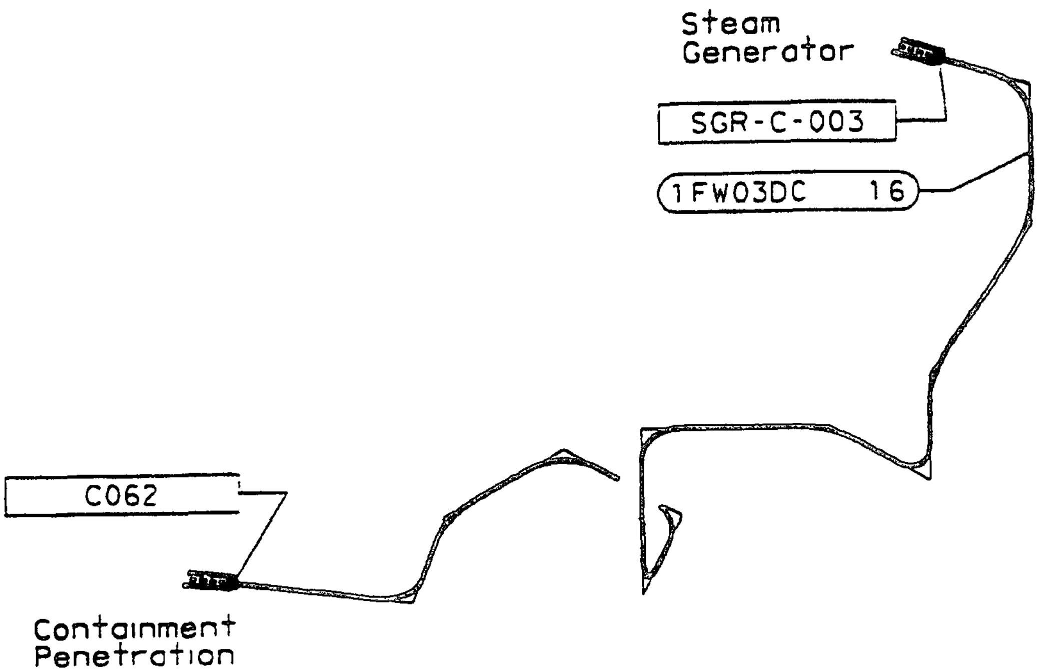
⊙ Indicates Postulated Break

Note

Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-48b FEEDWATER LOOP 2 POSTULATED BREAK LOCATIONS - UNIT 2

REVISION 7
DECEMBER 1998

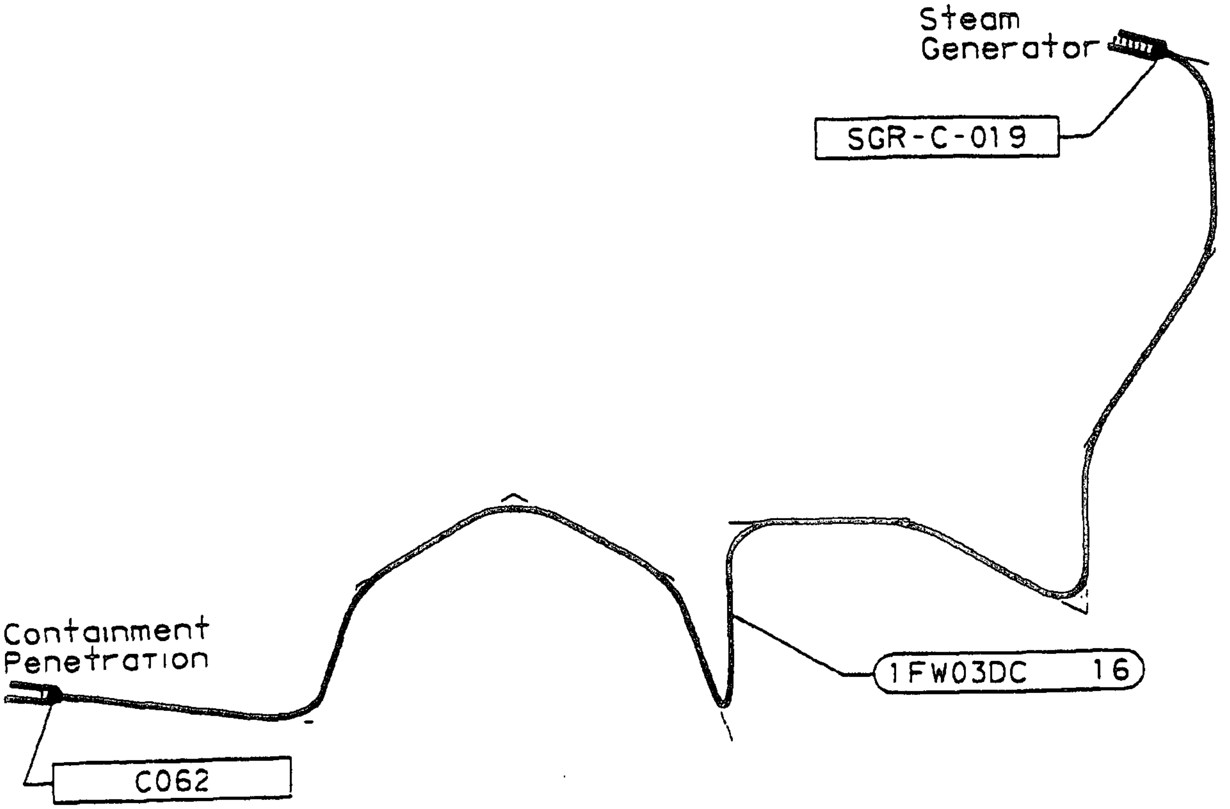


SUBSYSTEM 1FW04

⊗ Indicates Postulated Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-49 FEEDWATER LOOP 3 SUBSYSTEM 1FW04 POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998



SUBSYSTEM 1FW04

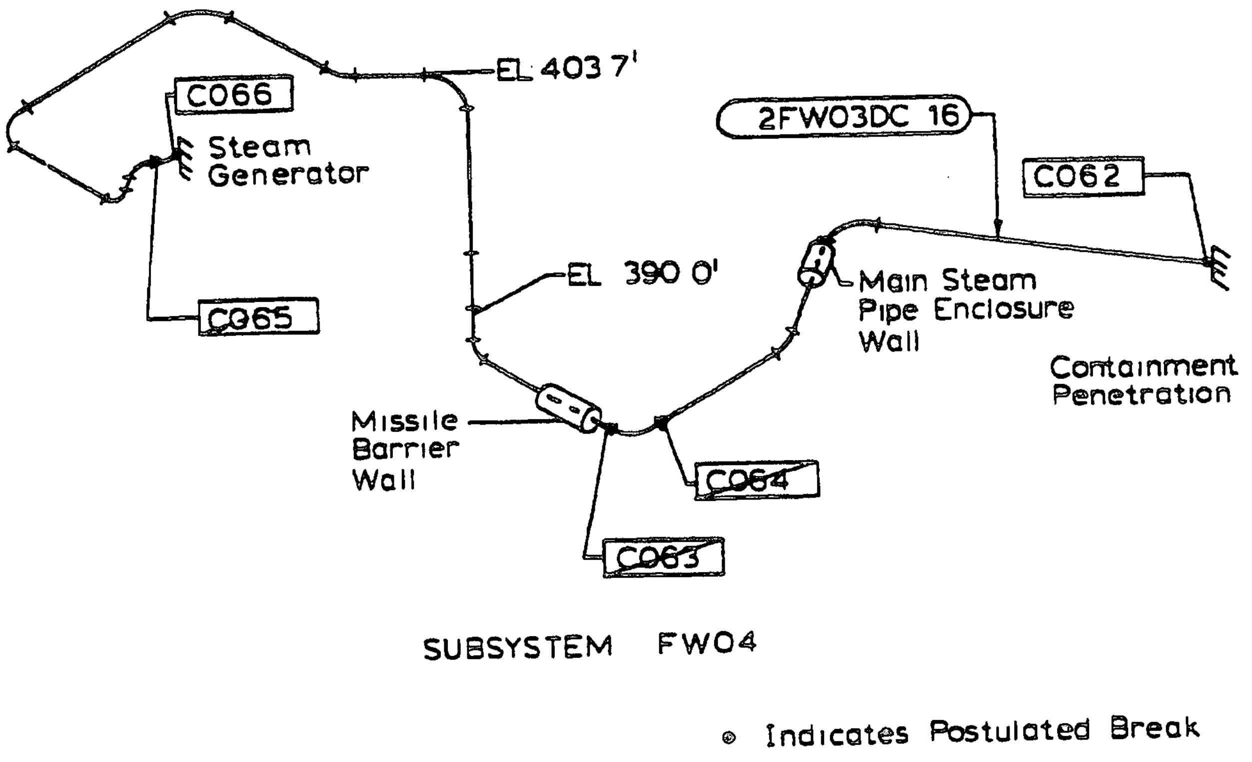
● Indicates Postulated Break

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-49a

FEEDWATER LOOP 3
SUBSYSTEM 1FW04
POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998



Note

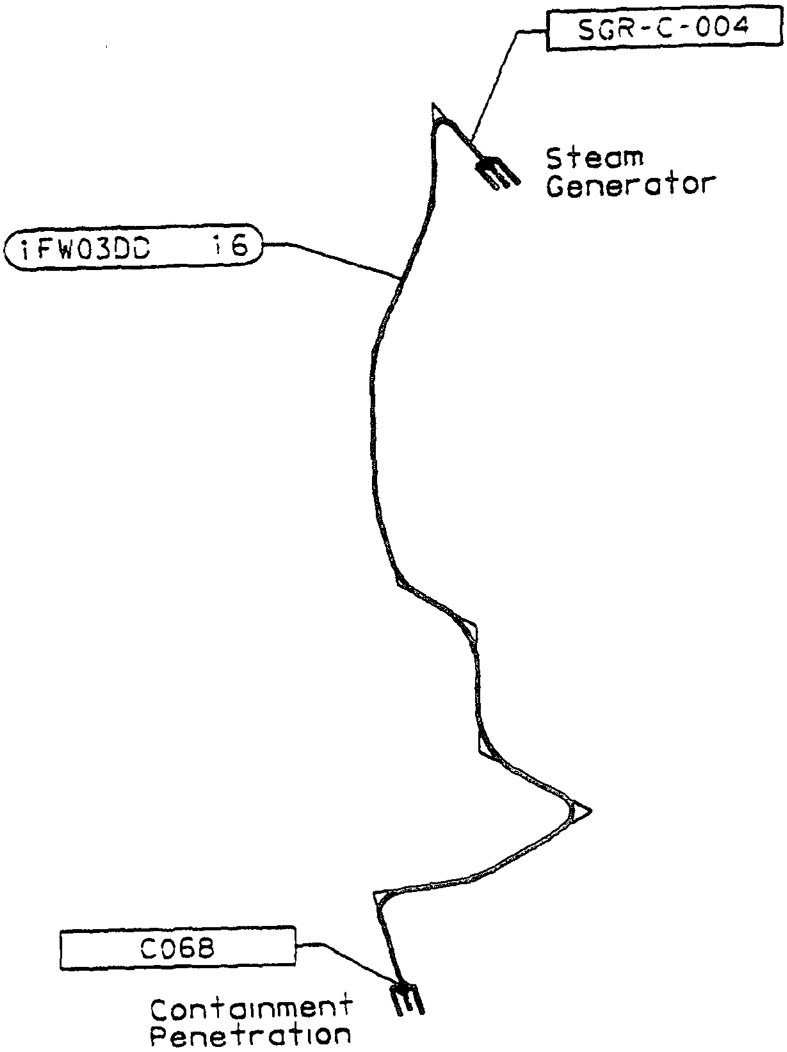
Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-49b

FEEDWATER LOOP 3
POSTULATED BREAK LOCATIONS - UNIT 2

REVISION 7
DECEMBER 1998

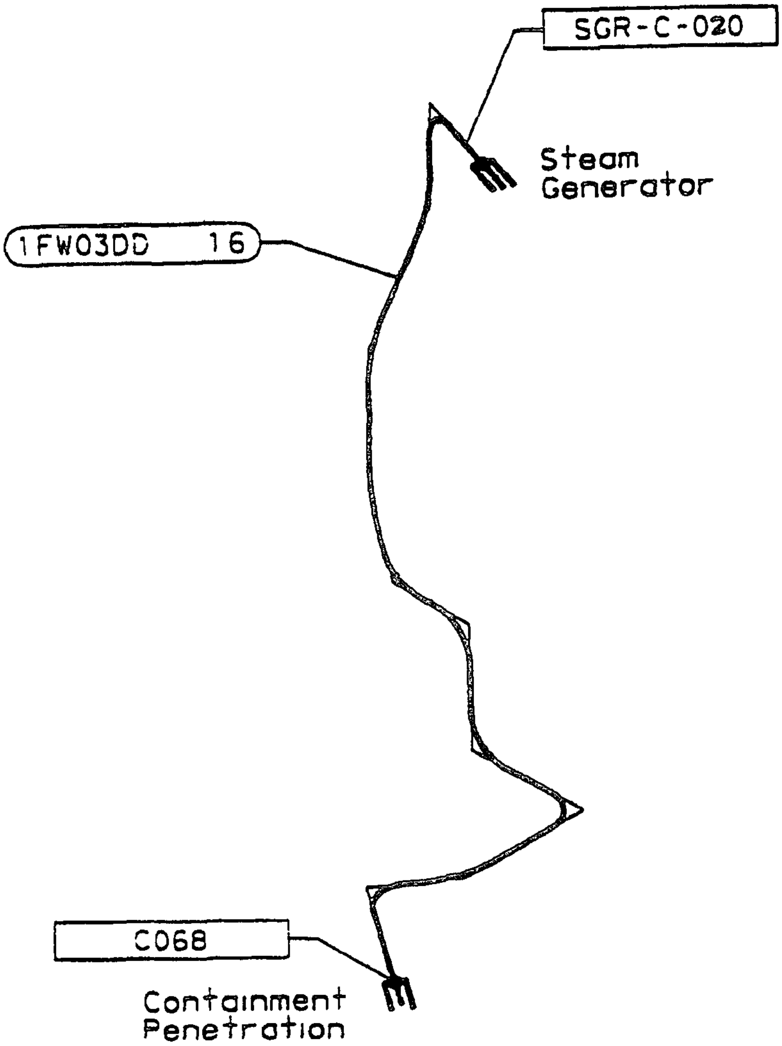


SUBSYSTEM 1FW05

⊛ Indicates Postulated Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-50 FEEDWATER LOOP 4 SUBSYSTEM 1FW05 POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998

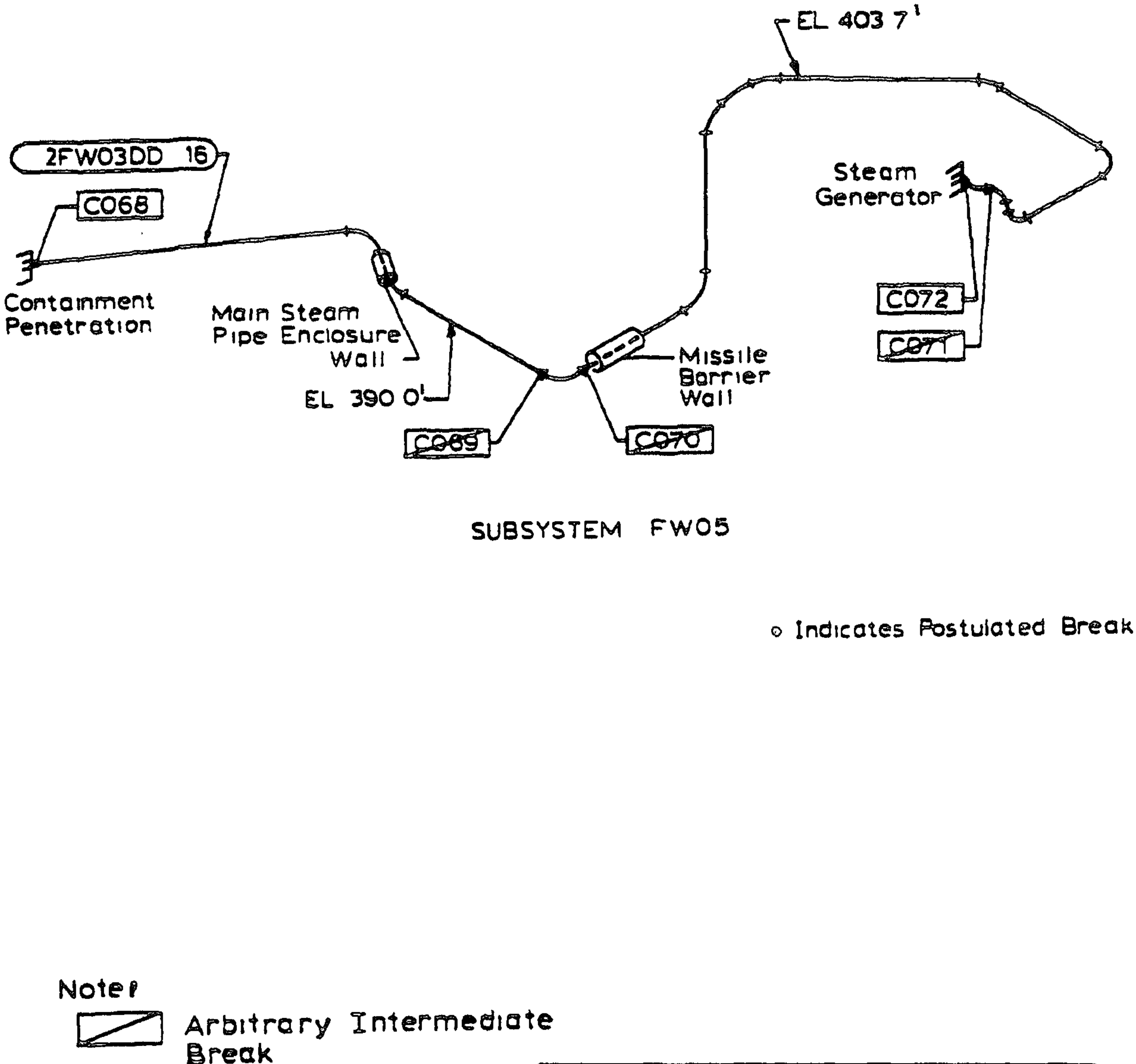


SUBSYSTEM 1FW05

● Indicates Postulated Break

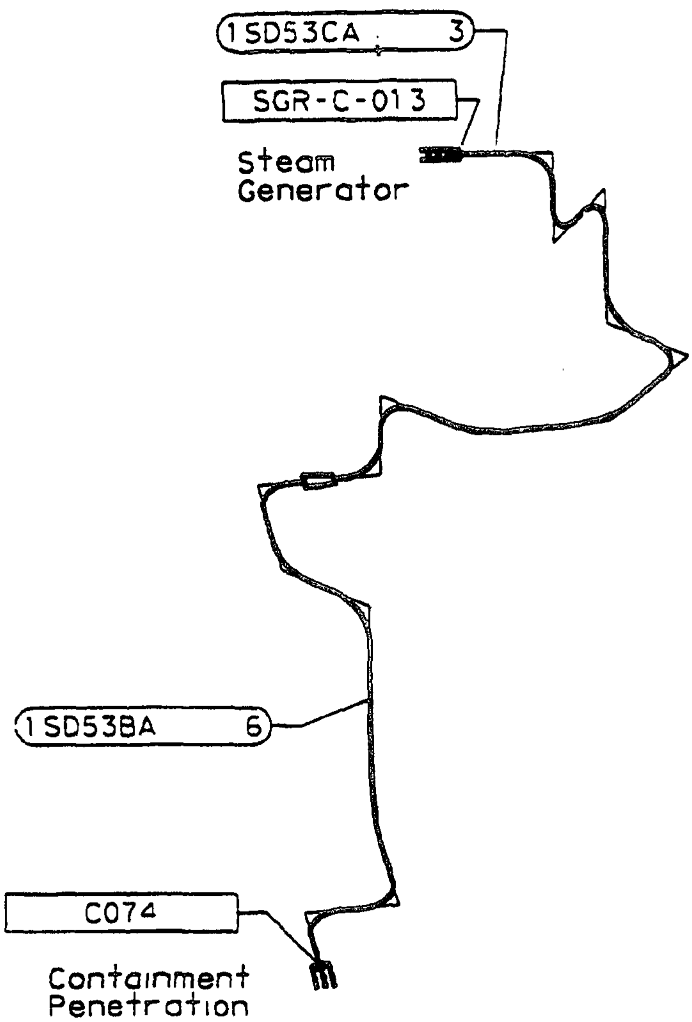
BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-50c FEEDWATER LOOP 4 SUBSYSTEM 1FW25 POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998



BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-50b FEEDWATER LOOP 4 POSTULATED BREAK LOCATIONS - UNIT 2

REVISION 7
DECEMBER 1998

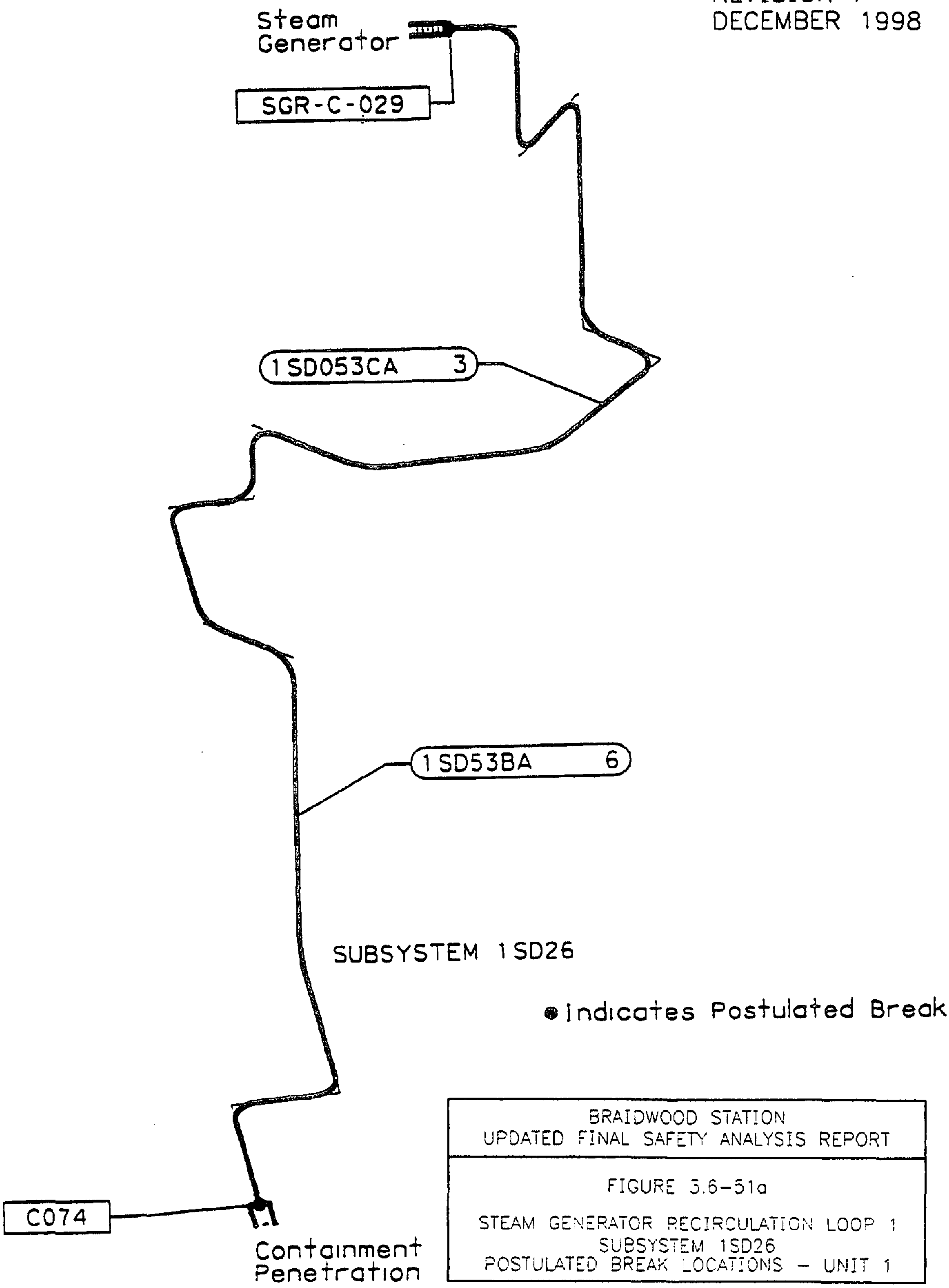


SUBSYSTEM 1SD26

● Indicates Postulated Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-51 STEAM GENERATOR RECIRCULATION LOOP 1 SUBSYSTEM 1SD26 POSTULATED BREAK LOCATIONS - UNIT 1

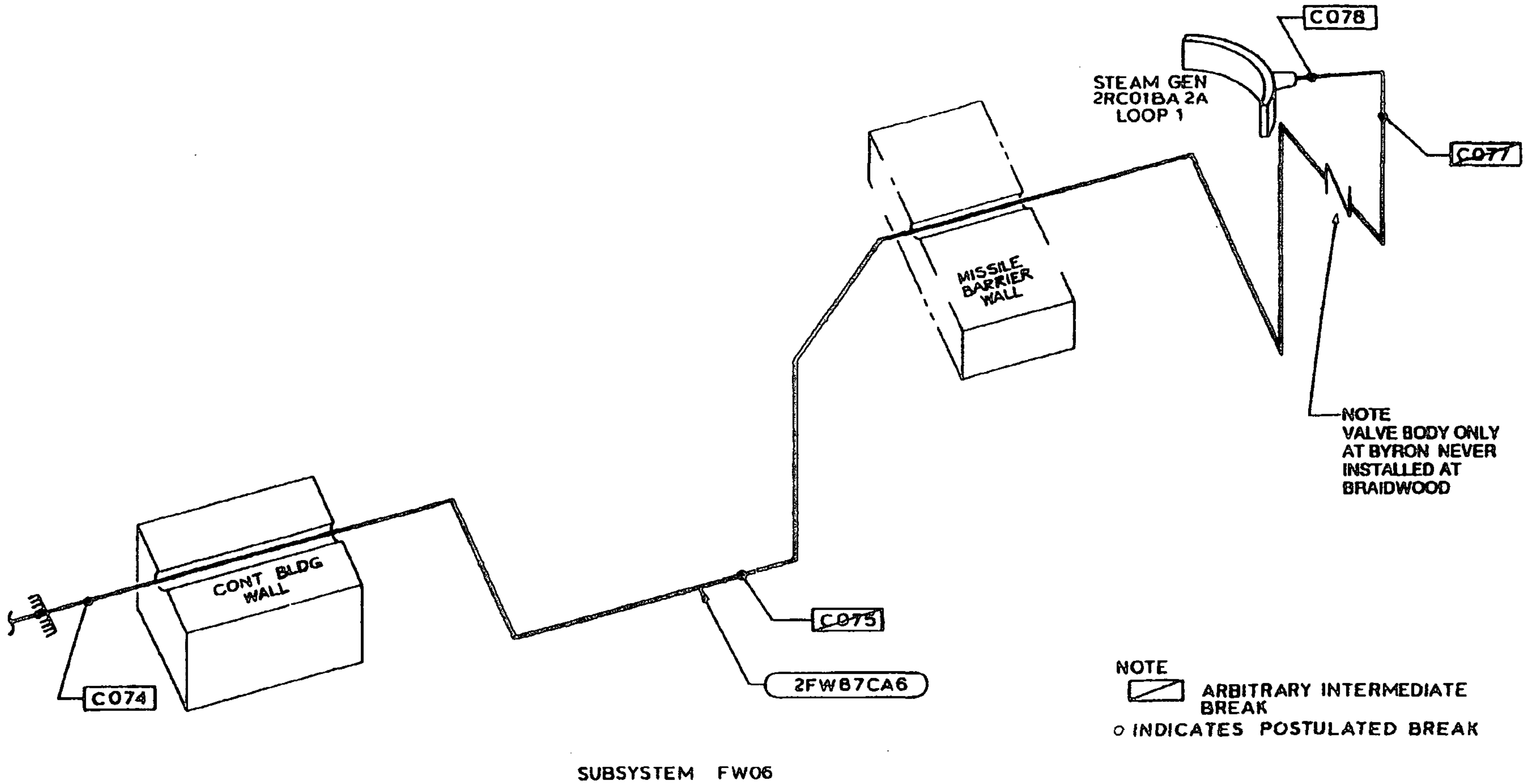
REVISION 7
DECEMBER 1998



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-51a

STEAM GENERATOR RECIRCULATION LOOP 1
SUBSYSTEM 1SD26
POSTULATED BREAK LOCATIONS – UNIT 1

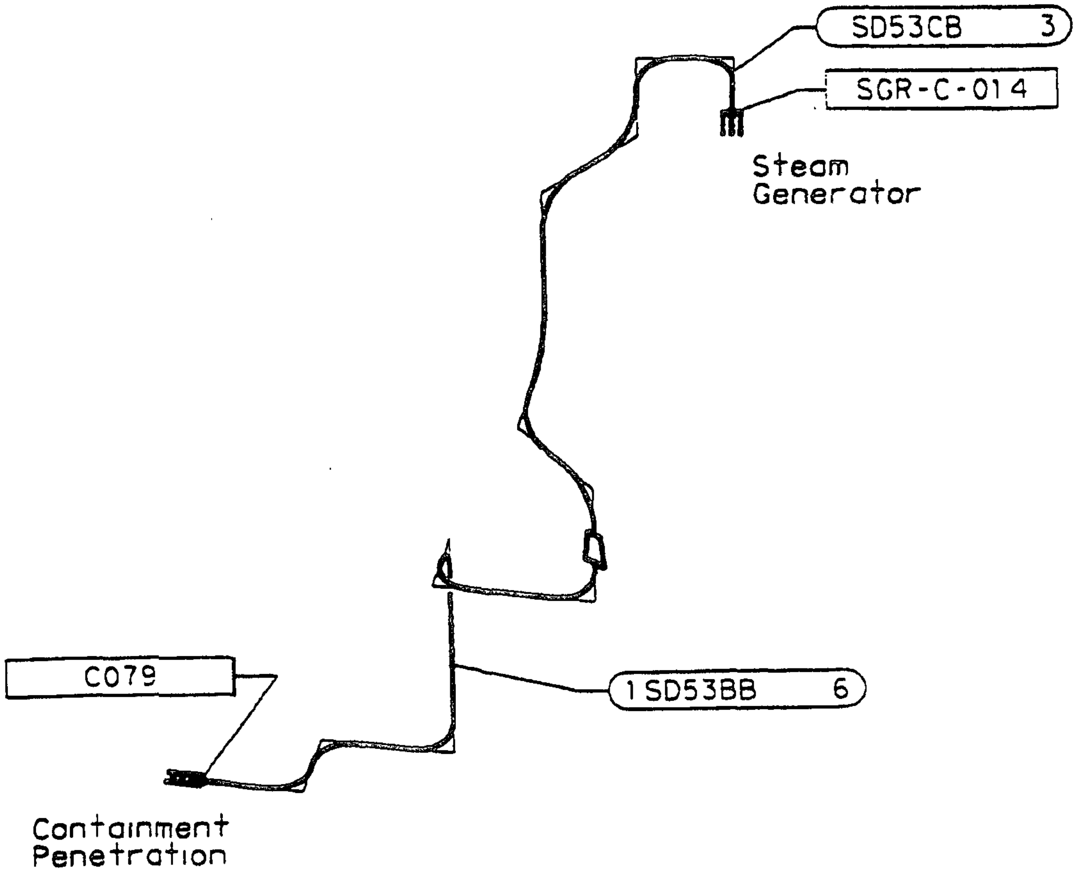


BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-51b

FEEDWATER-AUXILIARY FEEDWATER LOOP 1
POSTULATED BREAK LOCATIONS - UNIT 2

REVISION 7
DECEMBER 1998

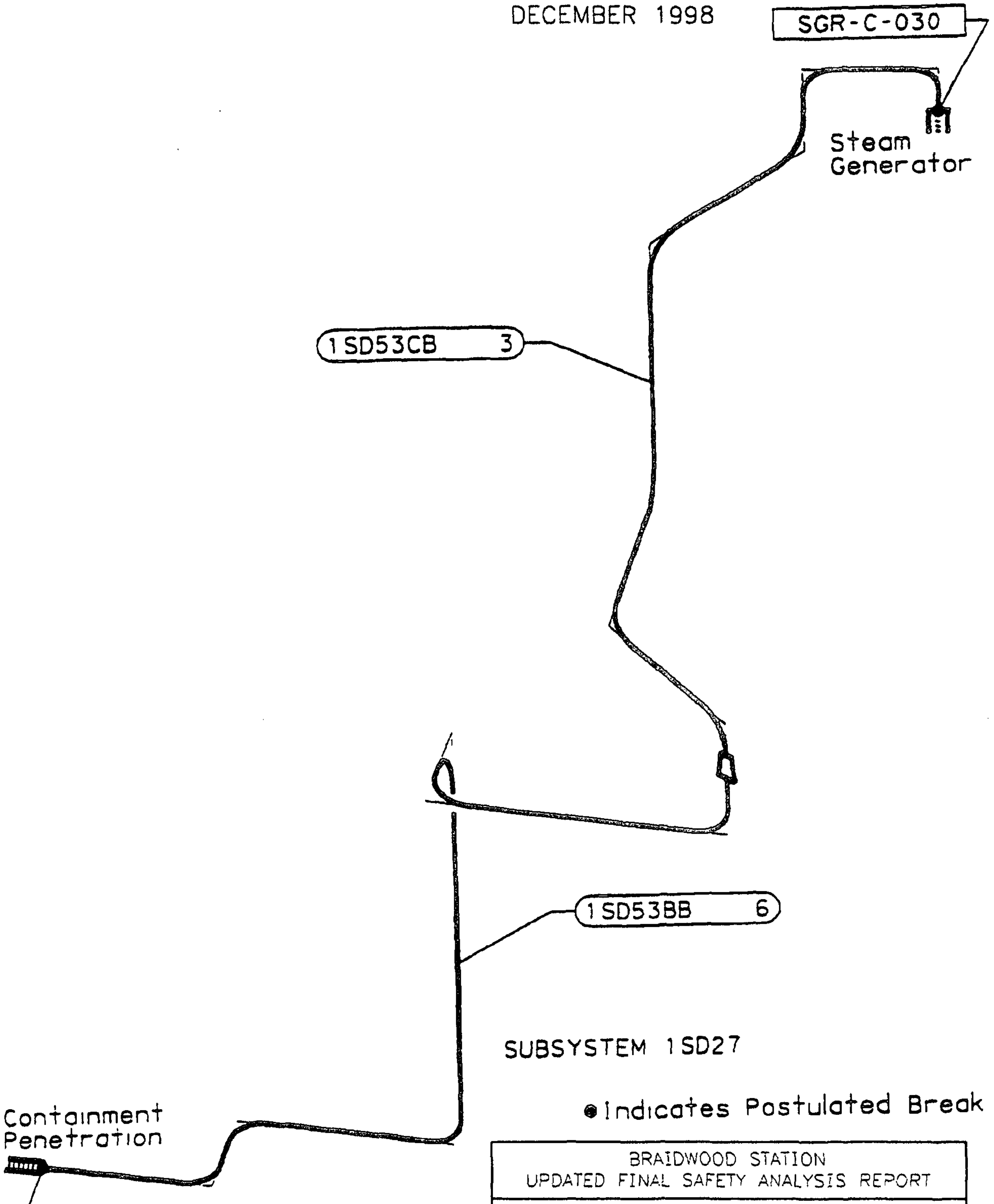


SUBSYSTEM 1SD27

● Indicates Postulated Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-52 STEAM GENERATOR RECIRCULATION LOOP 2 SUBSYSTEM 1SD27 POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998



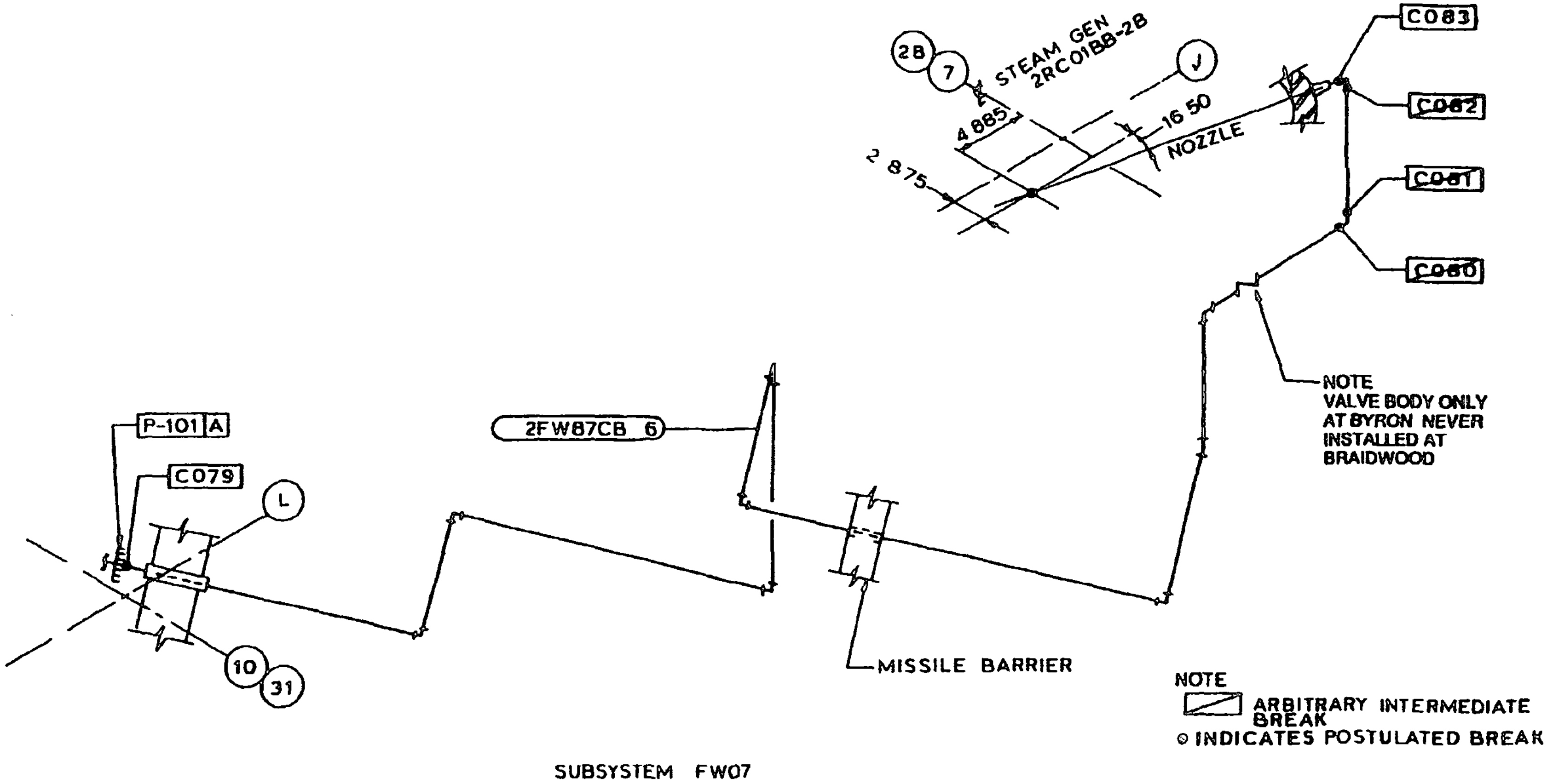
SUBSYSTEM 1SD27

● Indicates Postulated Break

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-52a

STEAM GENERATOR RECIRCULATION LOOP 2
SUBSYSTEM 1SD27
POSTULATED BREAK LOCATIONS - UNIT 1

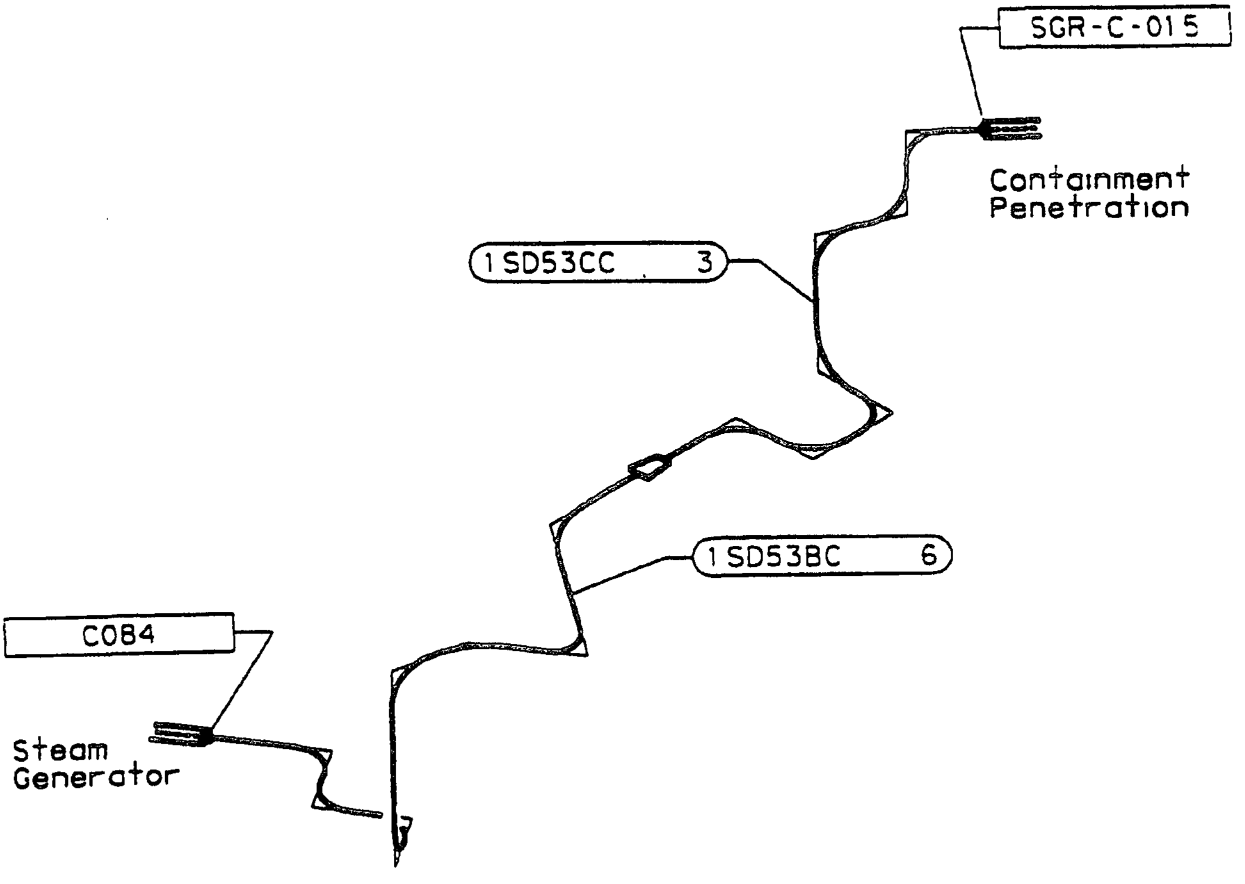


BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-52b

FEEDWATER-AUXILIARY FEEDWATER LOOP 2
POSTULATED BREAK LOCATIONS - UNIT 2

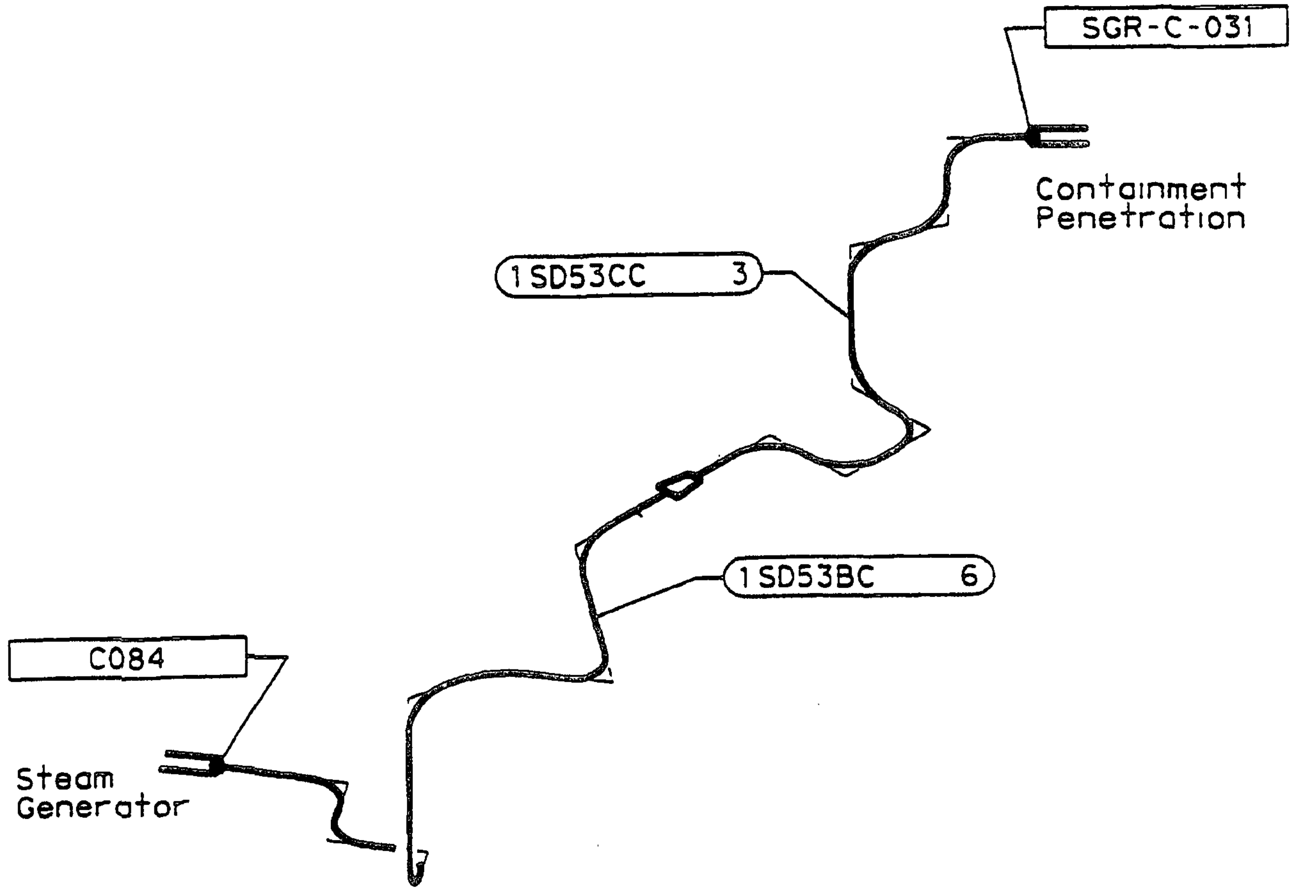
REVISION 7
DECEMBER 1998



SUBSYSTEM 1SD28

Indicates Postulated Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-53 STEAM GENERATOR RECIRCULATION LOOP 3 SUBSYSTEM 1SD28 POSTULATED BREAK LOCATIONS - UNIT 1

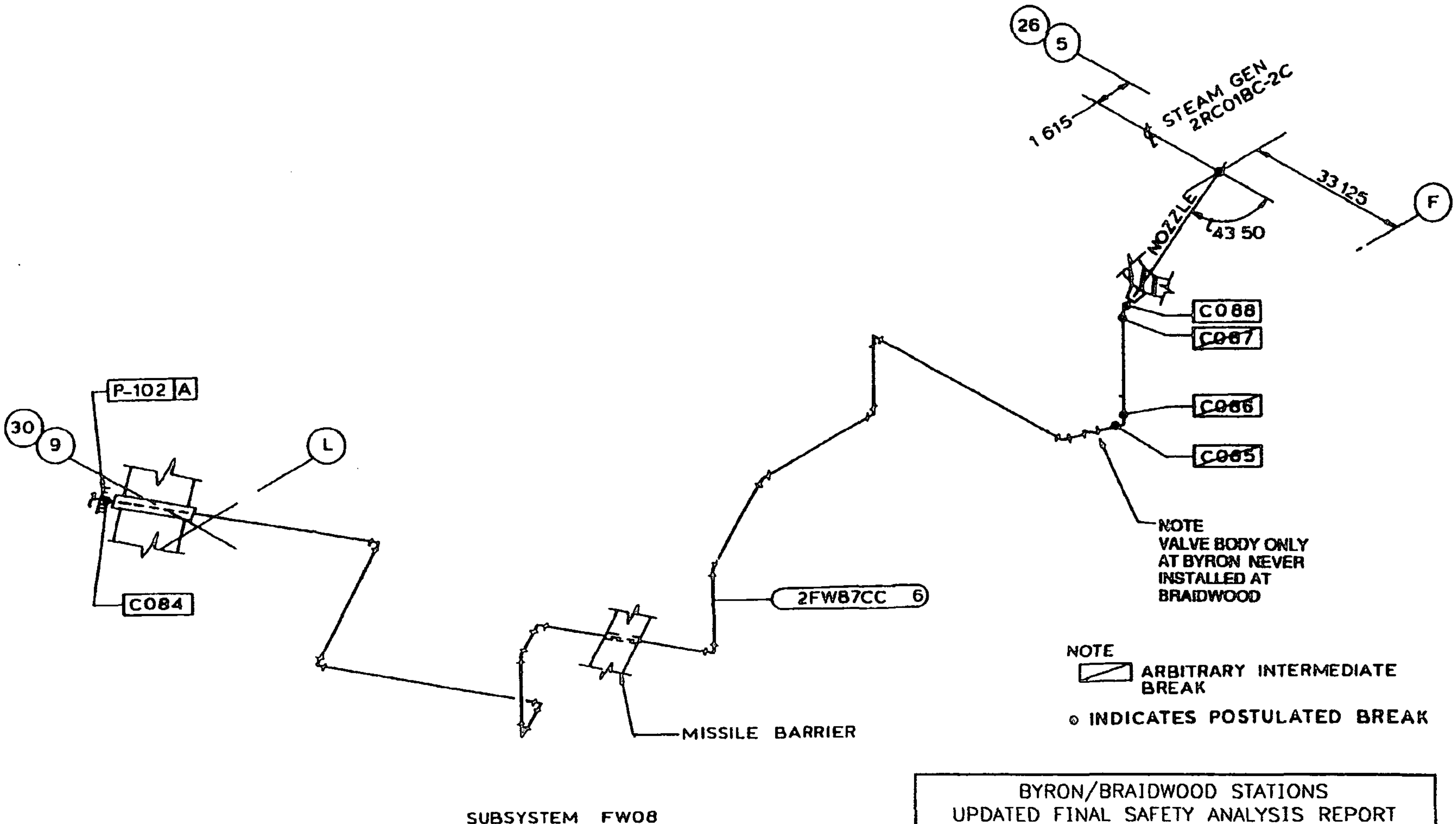


SUBSYSTEM 1SD28

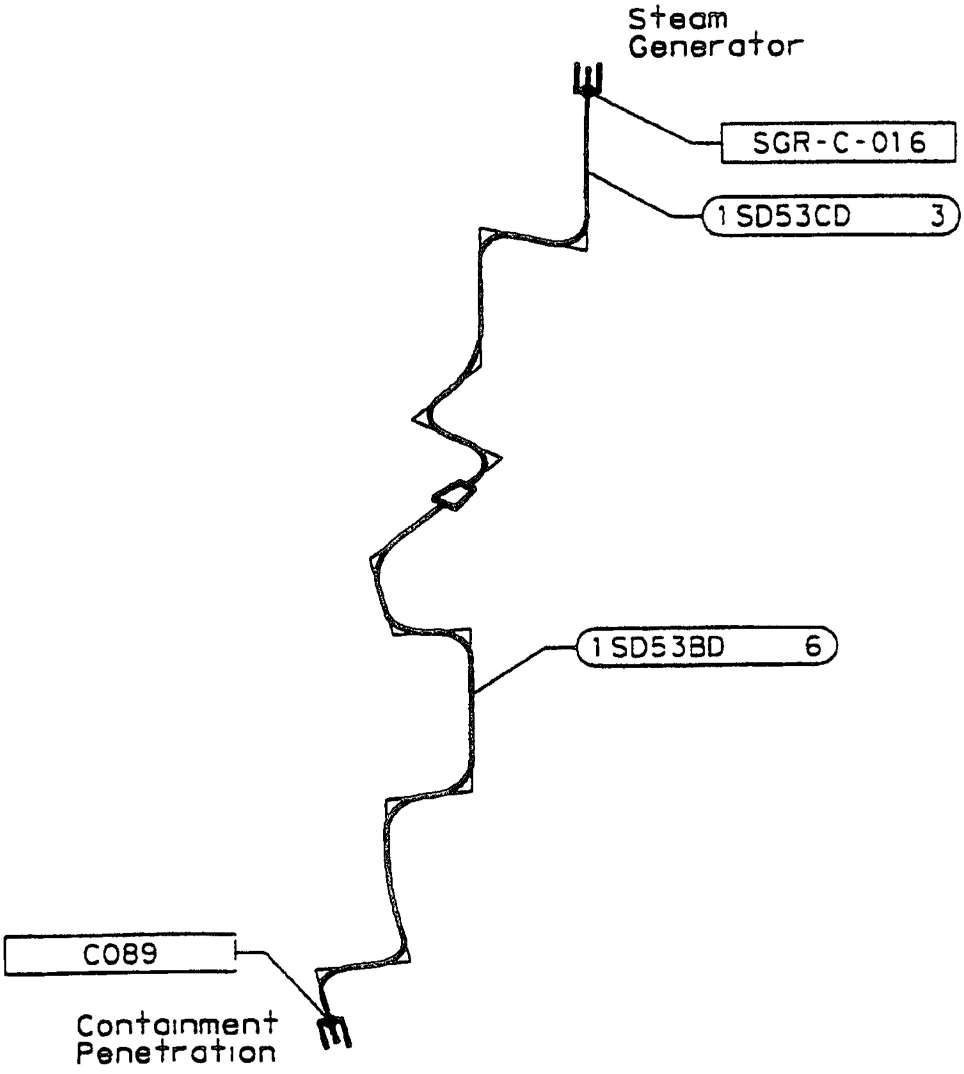
● Indicates Postulated Break

BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-53a STEAM GENERATOR RECIRCULATION LOOP 3 SUBSYSTEM 1SD28 POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998



BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-53b FEEDWATER-AUXILIARY FEEDWATER LOOP 3 POSTULATED BREAK LOCATIONS - UNIT 2



SUBSYSTEM 1SD29

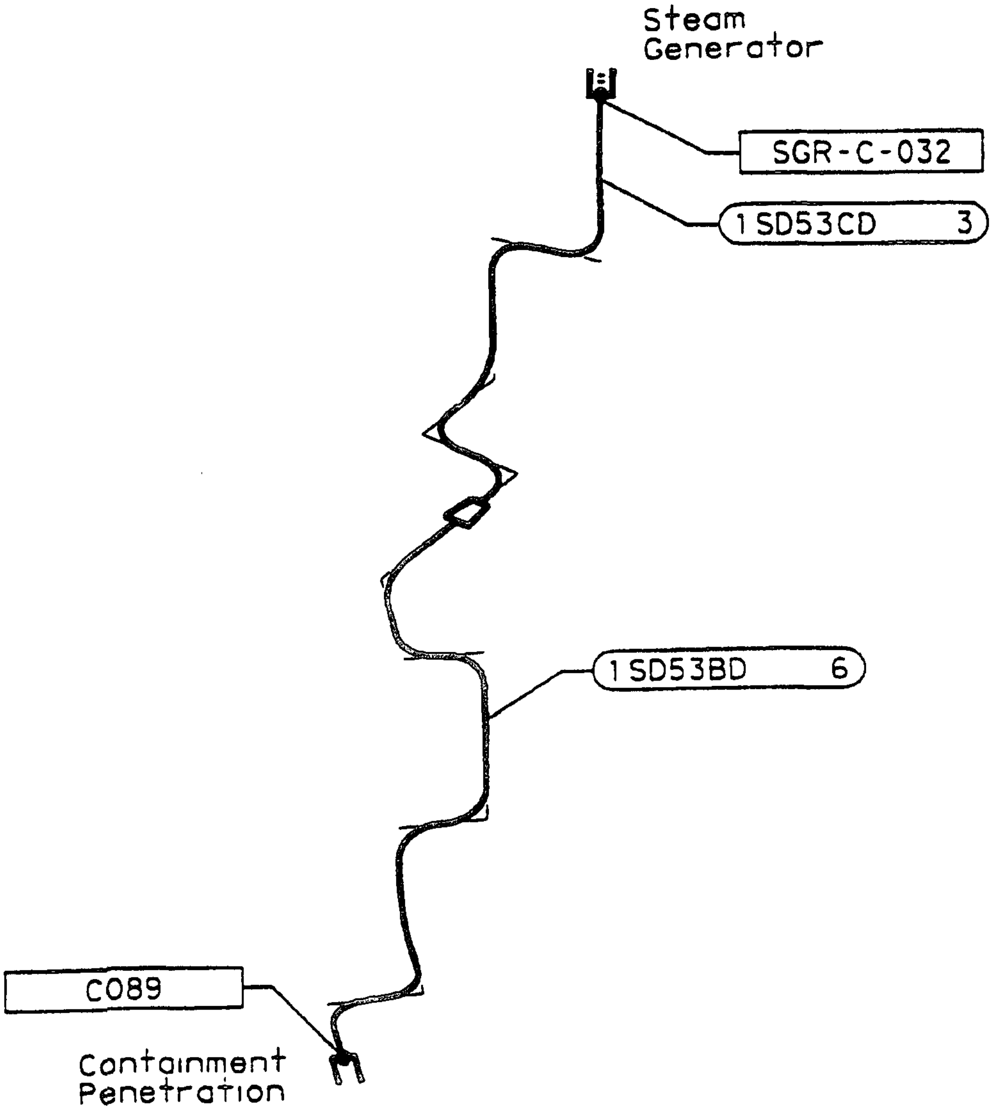
● Indicates Postulated Break

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-54

STEAM GENERATOR RECIRCULATION LOOP 4
SUBSYSTEM 1SD29
POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998



SUBSYSTEM 1SD29

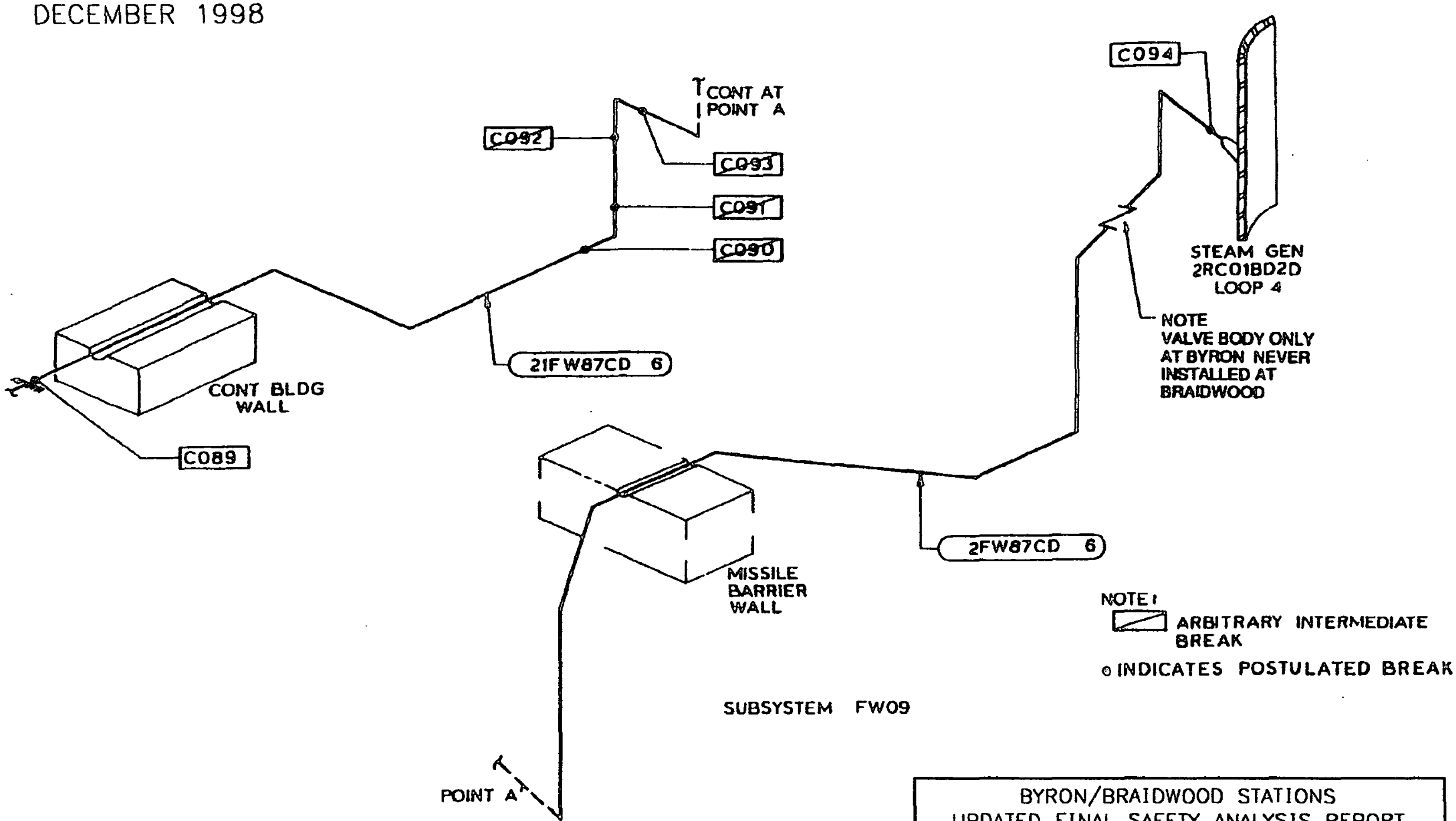
● Indicates Postulated Break

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

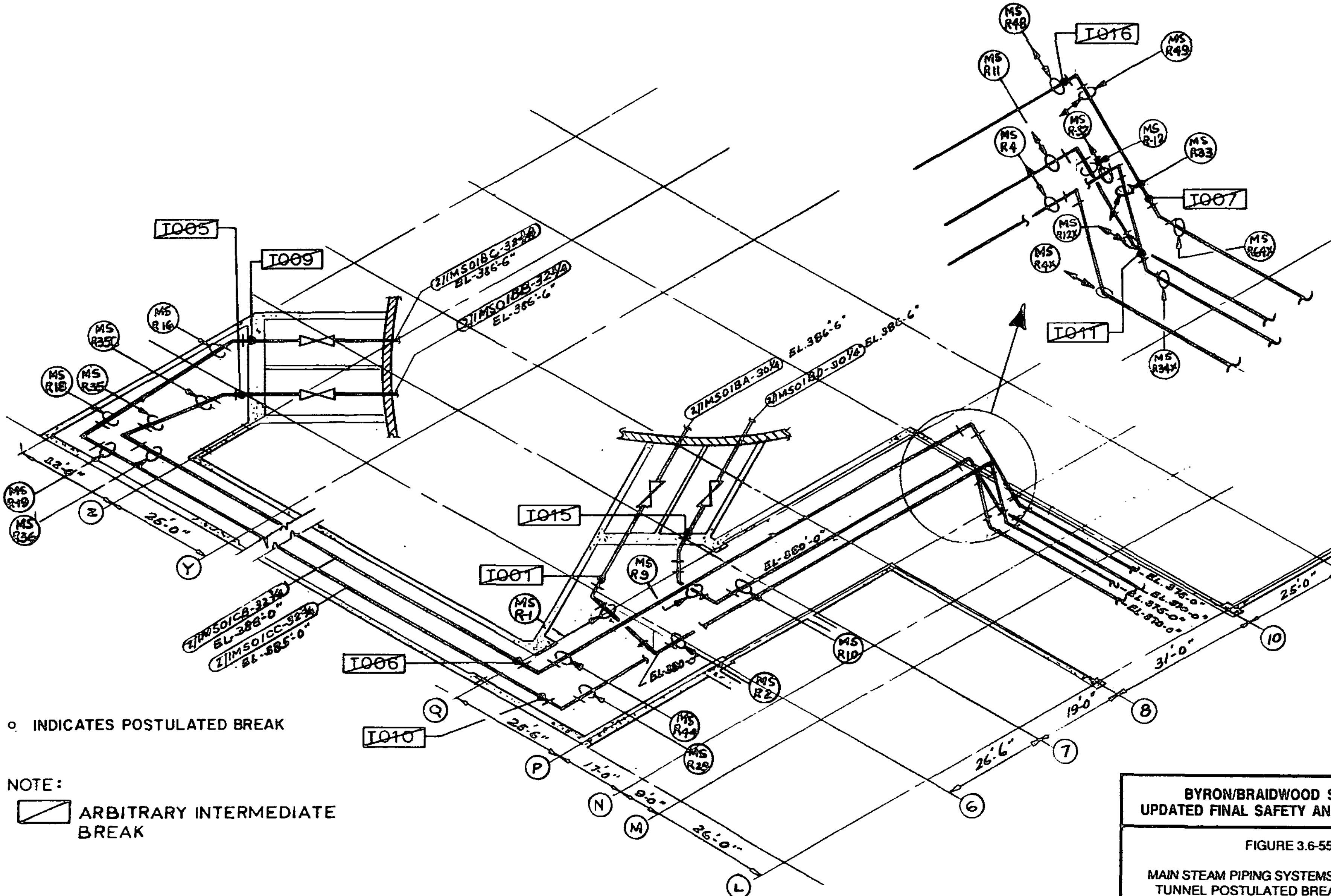
FIGURE 3.6-54a

STEAM GENERATOR RECIRCULATION LOOP 4
SUBSYSTEM 1SD29
POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998



BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-54b FEEDWATER-AUXILIARY FEEDWATER LOOP 4 POSTULATED BREAK LOCATIONS - UNIT 2



o INDICATES POSTULATED BREAK

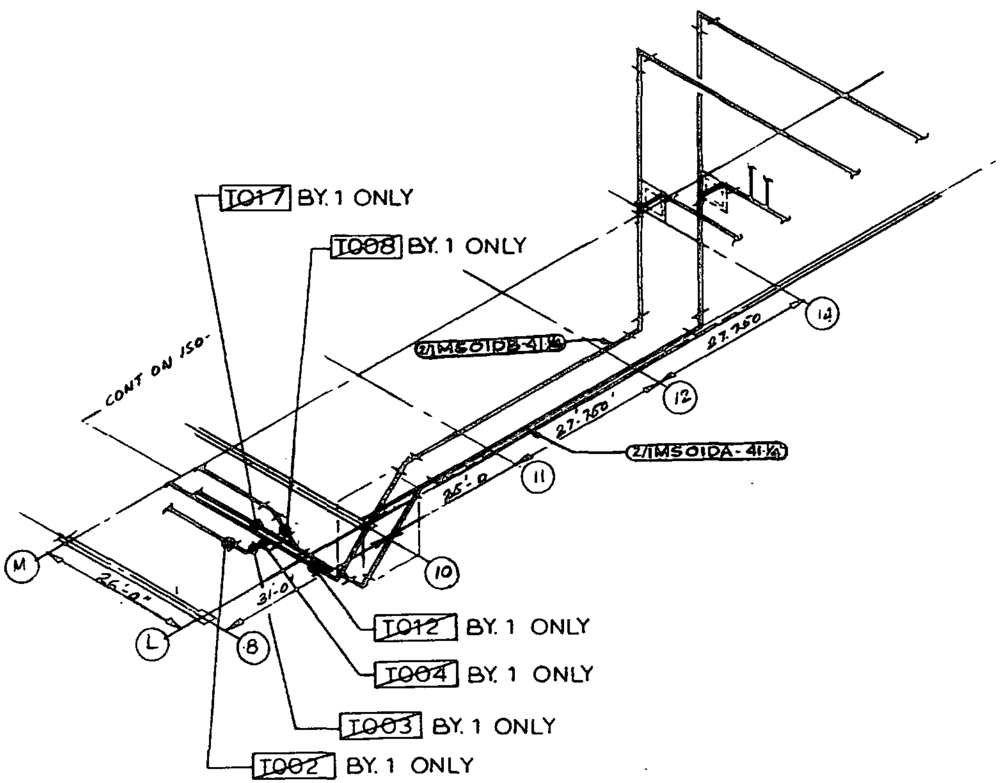
NOTE:

ARBITRARY INTERMEDIATE BREAK

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-55

MAIN STEAM PIPING SYSTEMS IN MAIN STEAM TUNNEL POSTULATED BREAK LOCATIONS
(SHEET 1 OF 2)



NOTE:

1. FEEDWATER LINES ARE PROTECTED AGAINST THE FULL EFFECTS OF POSTULATED MAIN STEAM PIPE RUPTURES.
2. WHERE THE PIPING IS UNRESTRAINED, ADJACENT STRUCTURES ARE DESIGNED TO PROVIDE PROTECTION AGAINST THE FULL EFFECTS OF THE POSTULATED PIPE RUPTURES.

o INDICATES POSTULATED BREAK

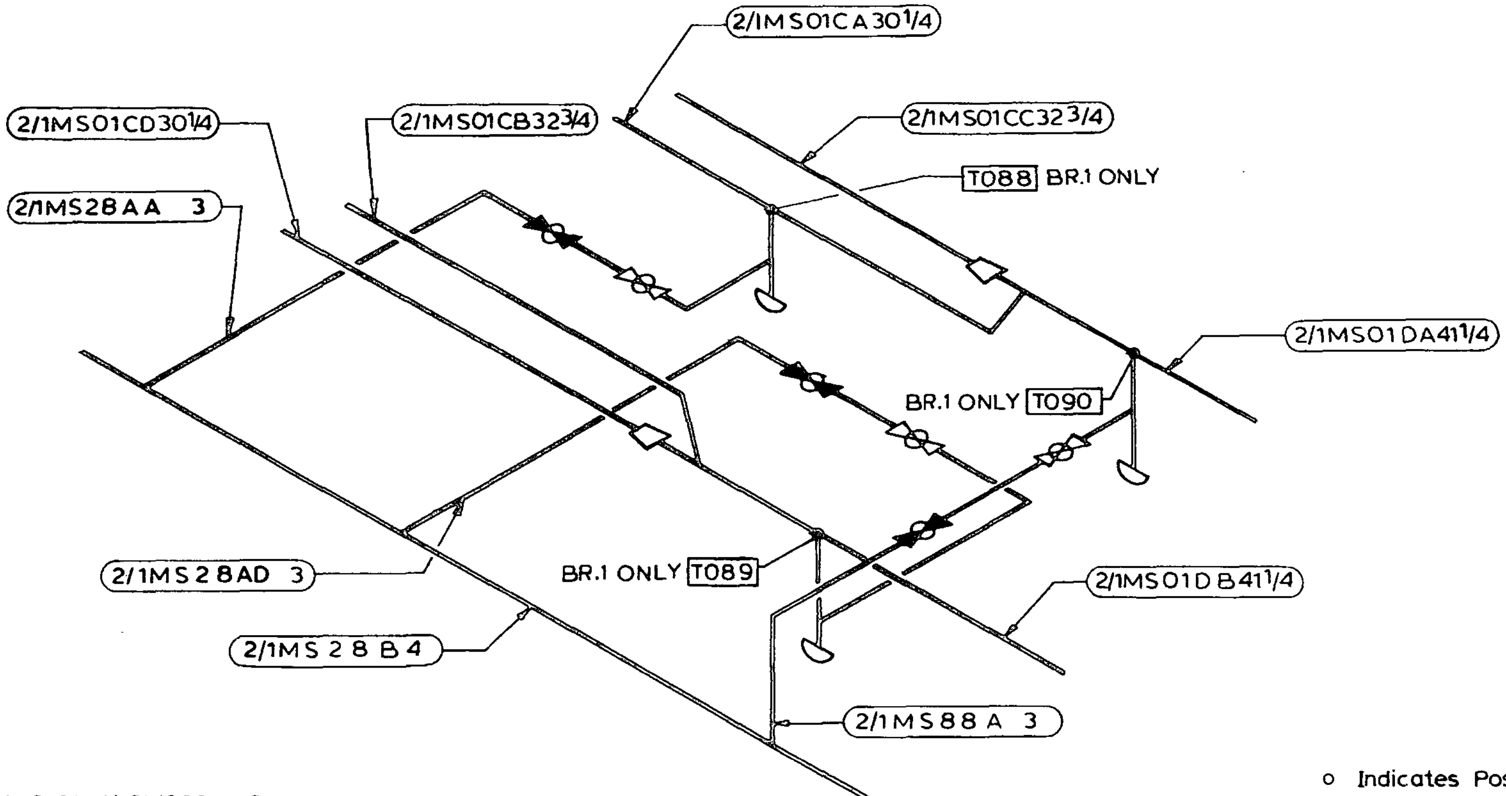
NOTE:

▭ ARBITRARY INTERMEDIATE BREAK

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-55

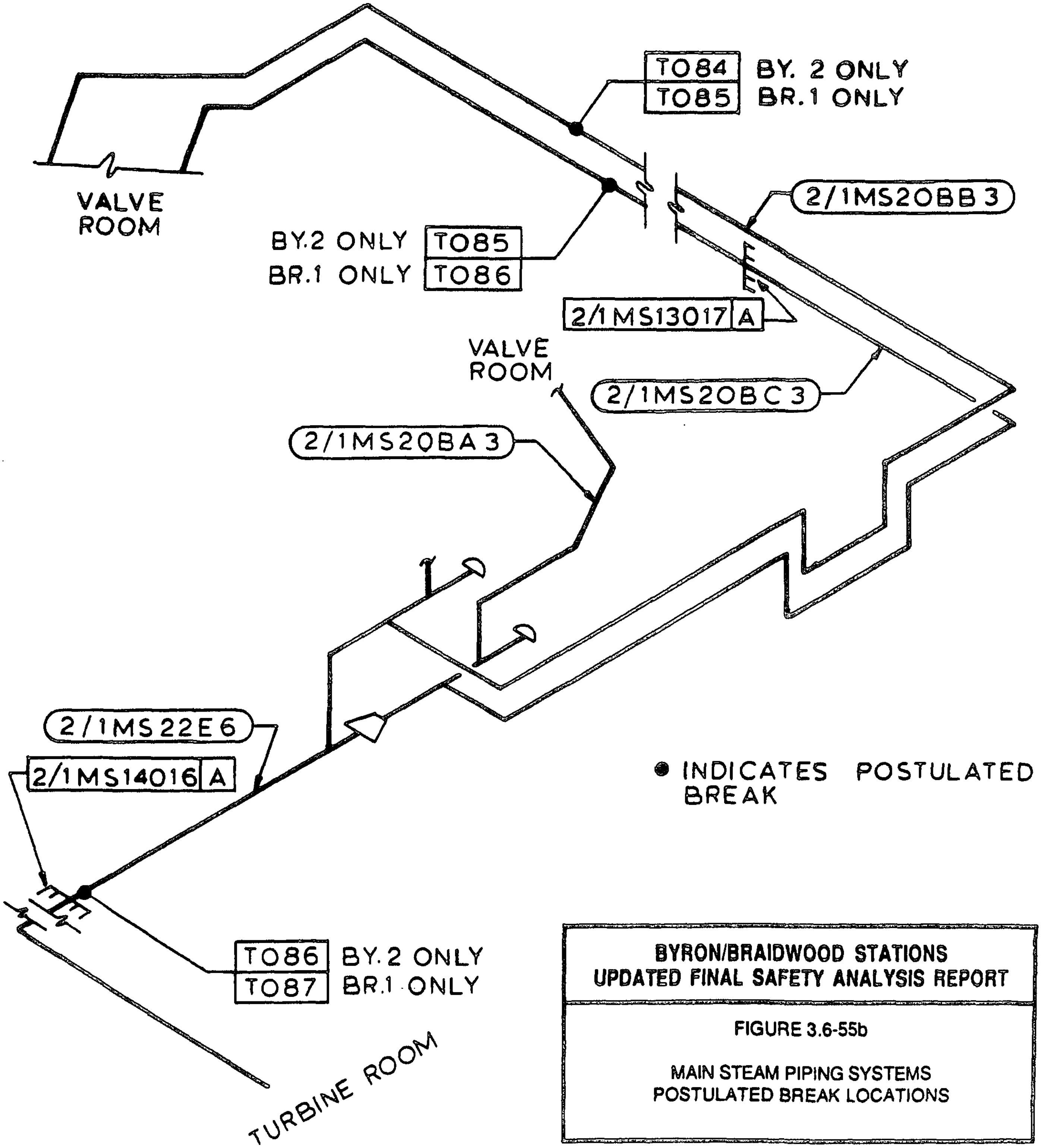
MAIN STEAM PIPING SYSTEMS IN MAIN STEAM TUNNEL POSTULATED BREAK LOCATIONS (SHEET 2 OF 2)



NOTE:
FOR BYRON 2 ONLY, 2MS28AA3, 2MS28AD3 & 2MS88A3 BREAKS ARE POSTULATED TO OCCUR ANYWHERE ALONG THE SPAN OF PIPE EXTENDING FROM THE HEADER UP TO THE SECOND CLOSED VALVE.

o Indicates Postulated Break

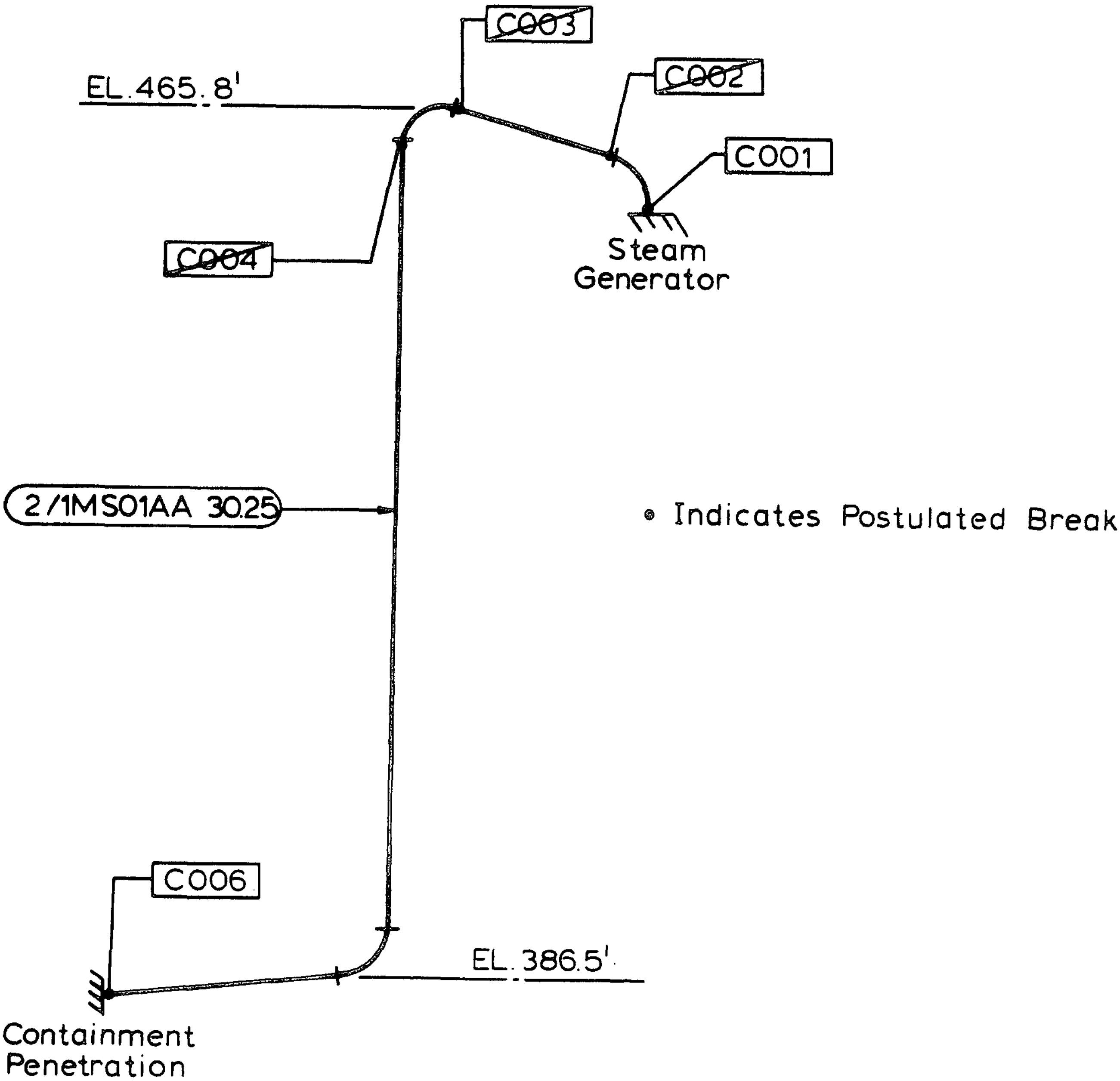
BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-55a MAIN STEAM PIPING SYSTEMS POSTULATED BREAK LOCATIONS



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-55b

MAIN STEAM PIPING SYSTEMS
POSTULATED BREAK LOCATIONS



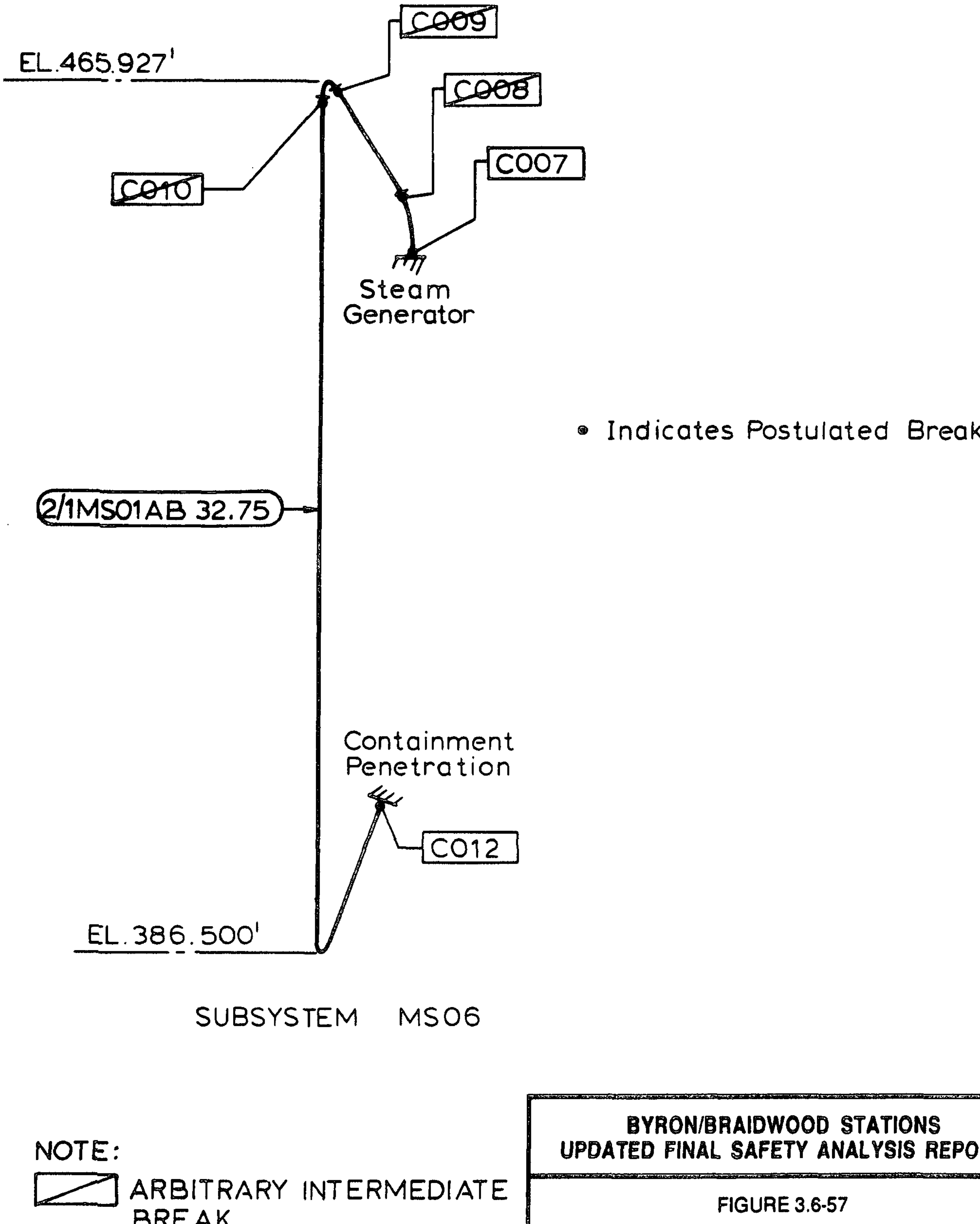
SUBSYSTEM MS05

NOTE:
ARBITRARY INTERMEDIATE BREAK

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-56

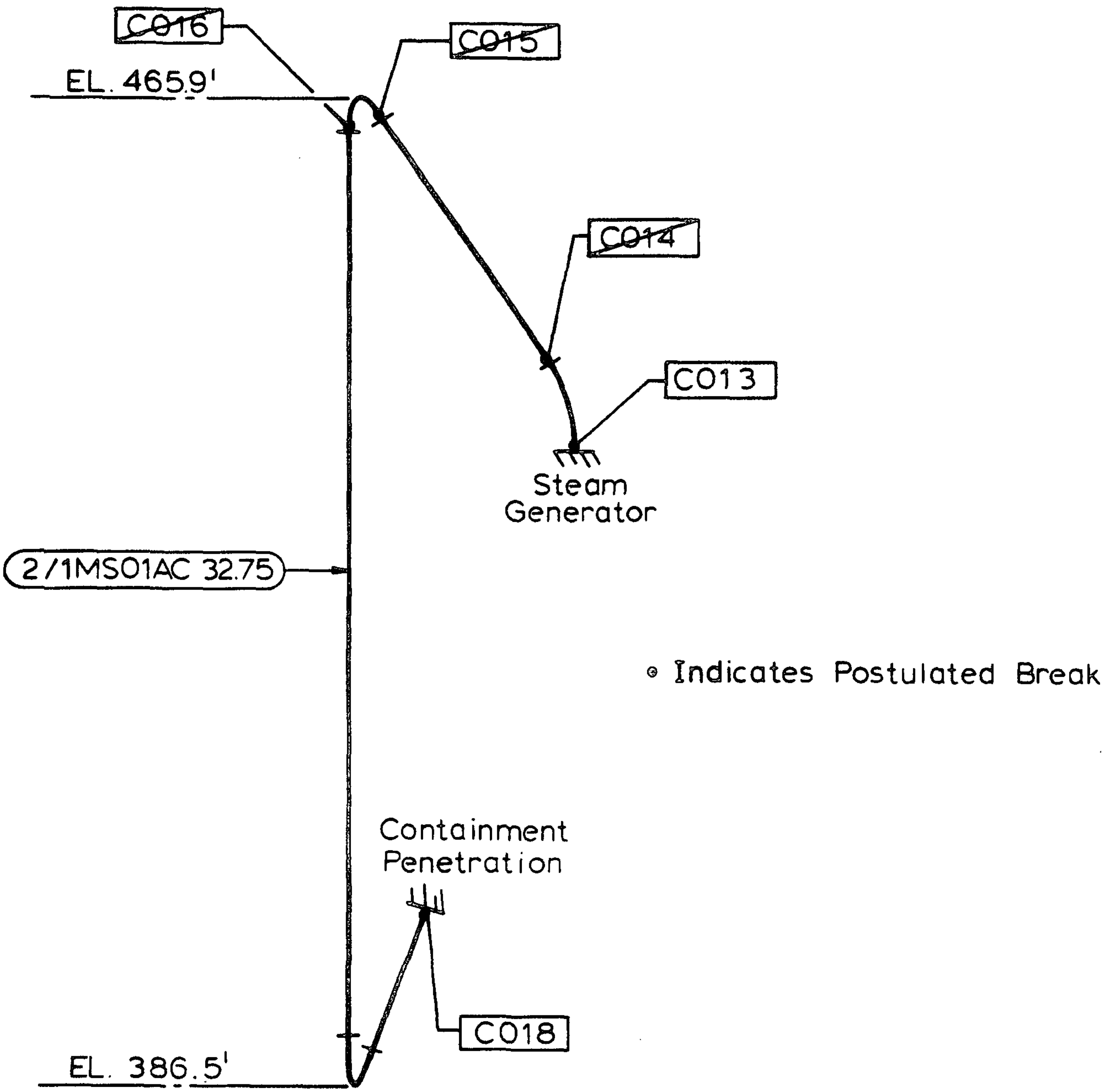
MAIN STEAM LOOP 1
POSTULATED BREAK LOCATIONS



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-57

MAIN STEAM LOOP 2
POSTULATED BREAK LOCATIONS



SUBSYSTEM MS07

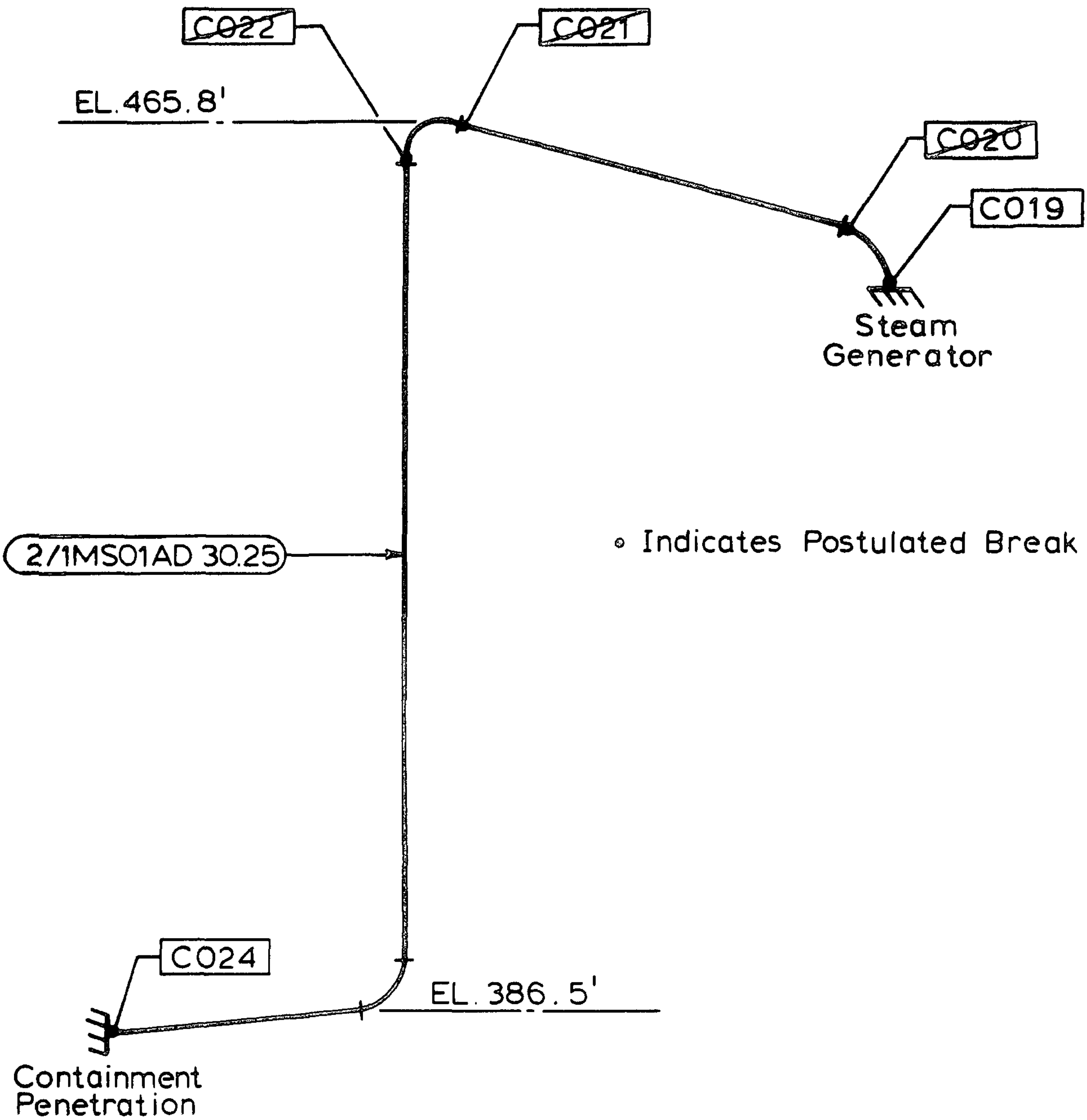
NOTE:


Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-58

MAIN STEAM LOOP 3
POSTULATED BREAK LOCATIONS



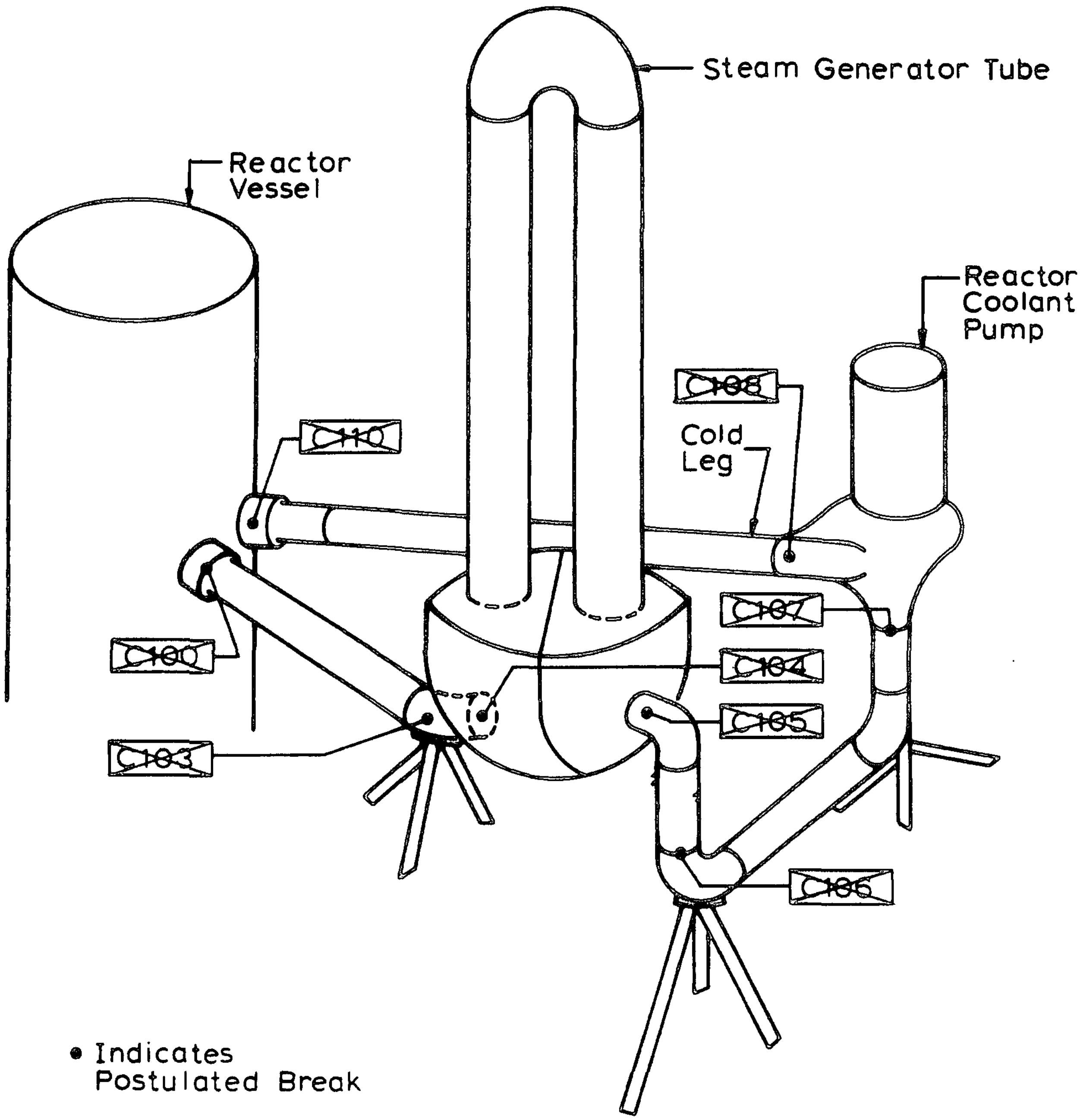
SUBSYSTEM MS08

NOTE:
Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-59

MAIN STEAM LOOP 4
POSTULATED BREAK LOCATIONS



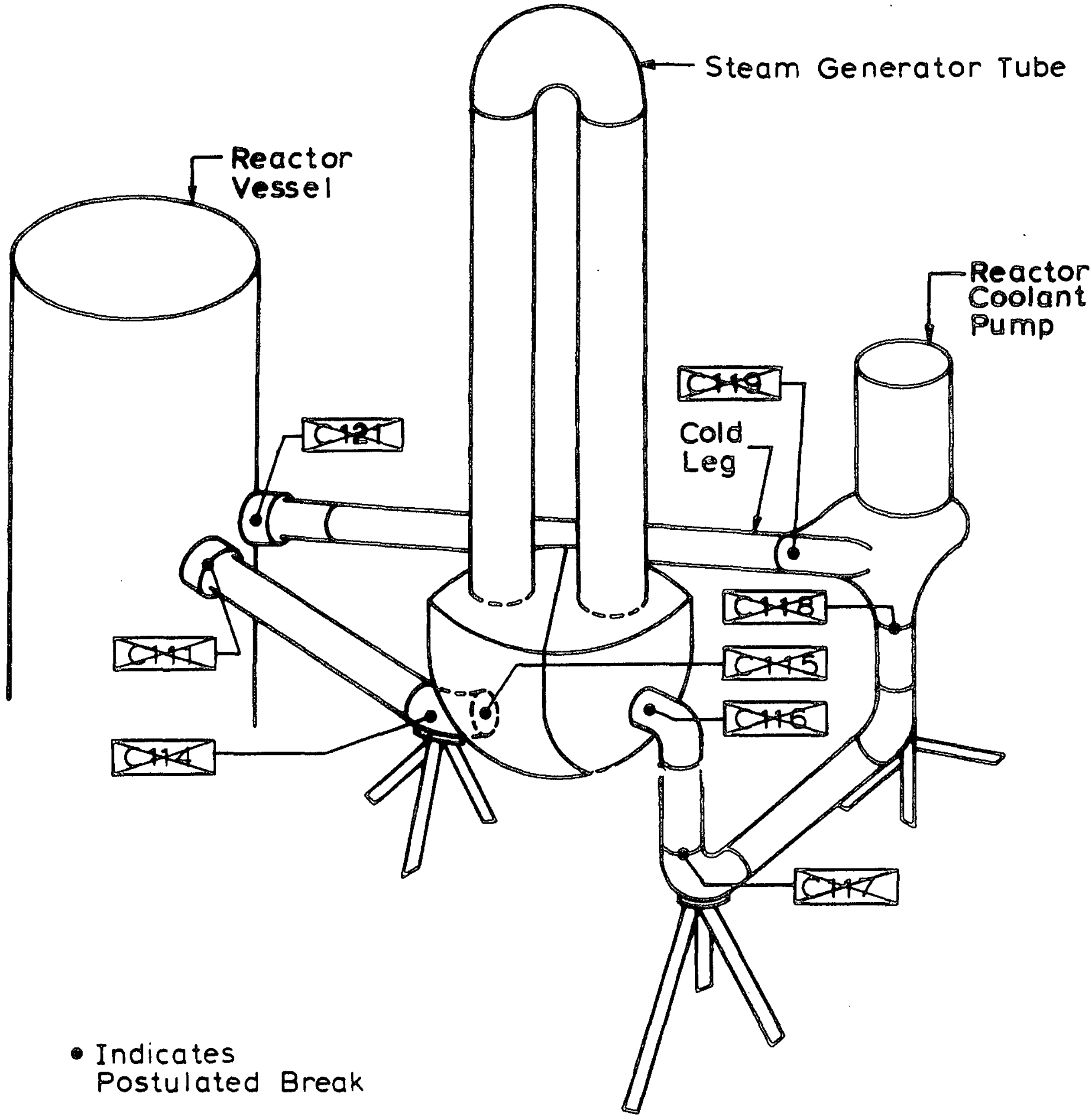
LOOP 1

NOTE:
The dynamic effects associated with these breaks are no longer considered due to application of the leak-before-break concept.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-60

REACTOR COOLANT LOOP 1
POSTULATED BREAK LOCATIONS

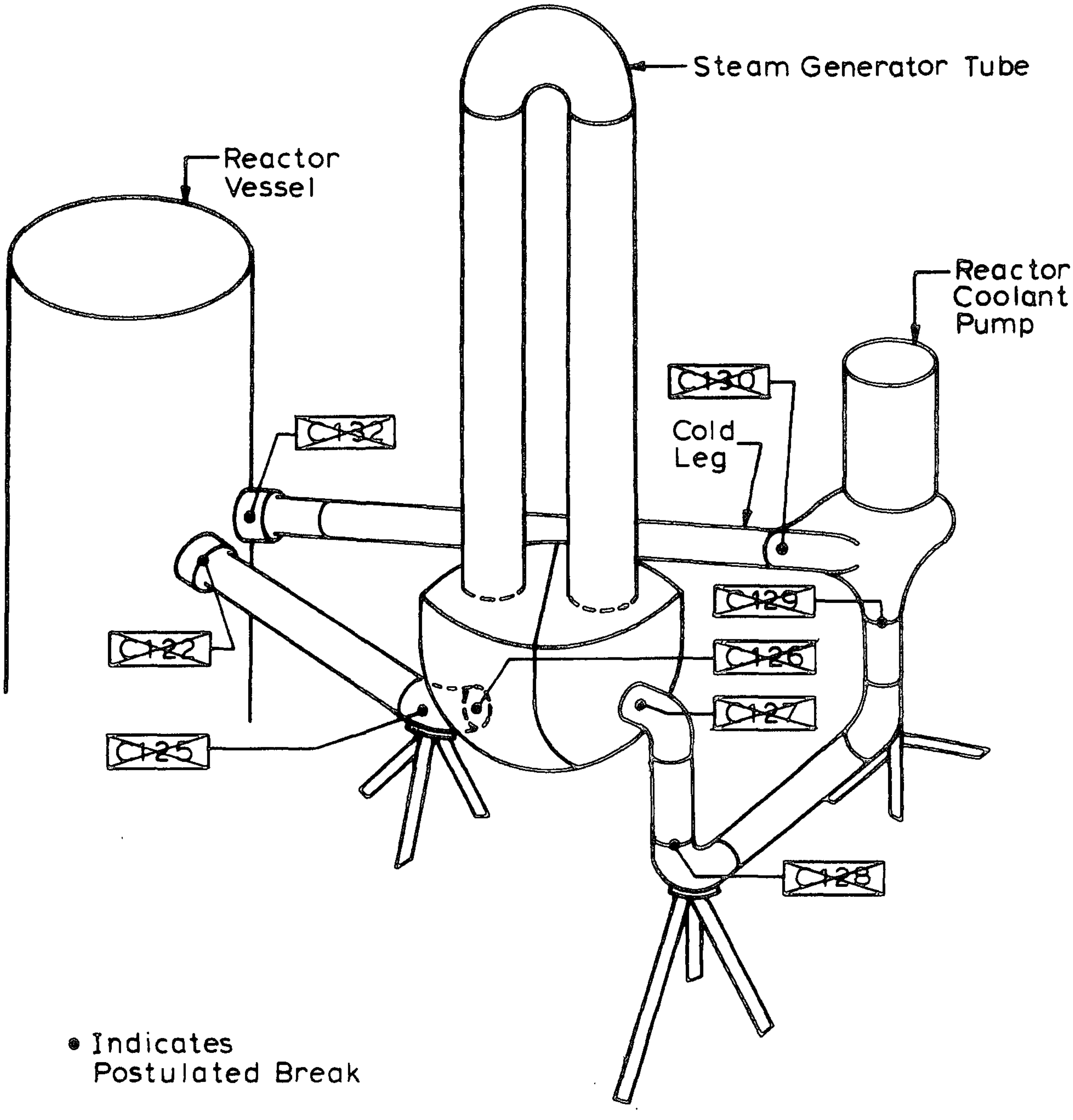


• Indicates Postulated Break

LOOP 2

NOTE:
The dynamic effects associated with these breaks are no longer considered due to application of the leak-before-break concept.

BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-61 REACTOR COOLANT LOOP 2 POSTULATED BREAK LOCATIONS

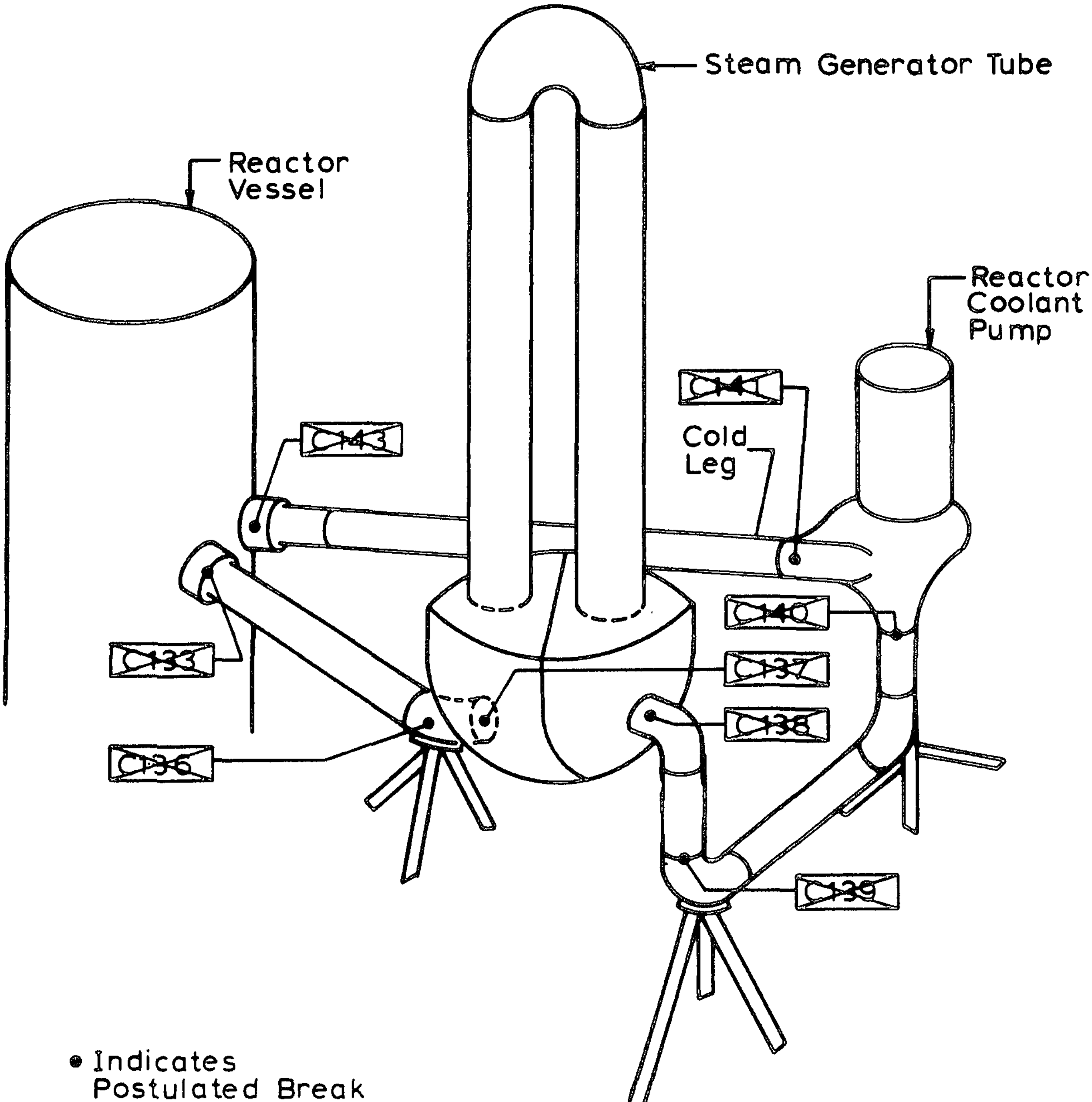


LOOP 3

NOTE:

☒ The dynamic effects associated with these breaks are no longer considered due to application of the leak-before-break concept.

BYRON/BRAIDWOOD STATIONS **UPDATED FINAL SAFETY ANALYSIS REPORT**
FIGURE 3.6-62 REACTOR COOLANT LOOP 3 POSTULATED BREAK LOCATIONS



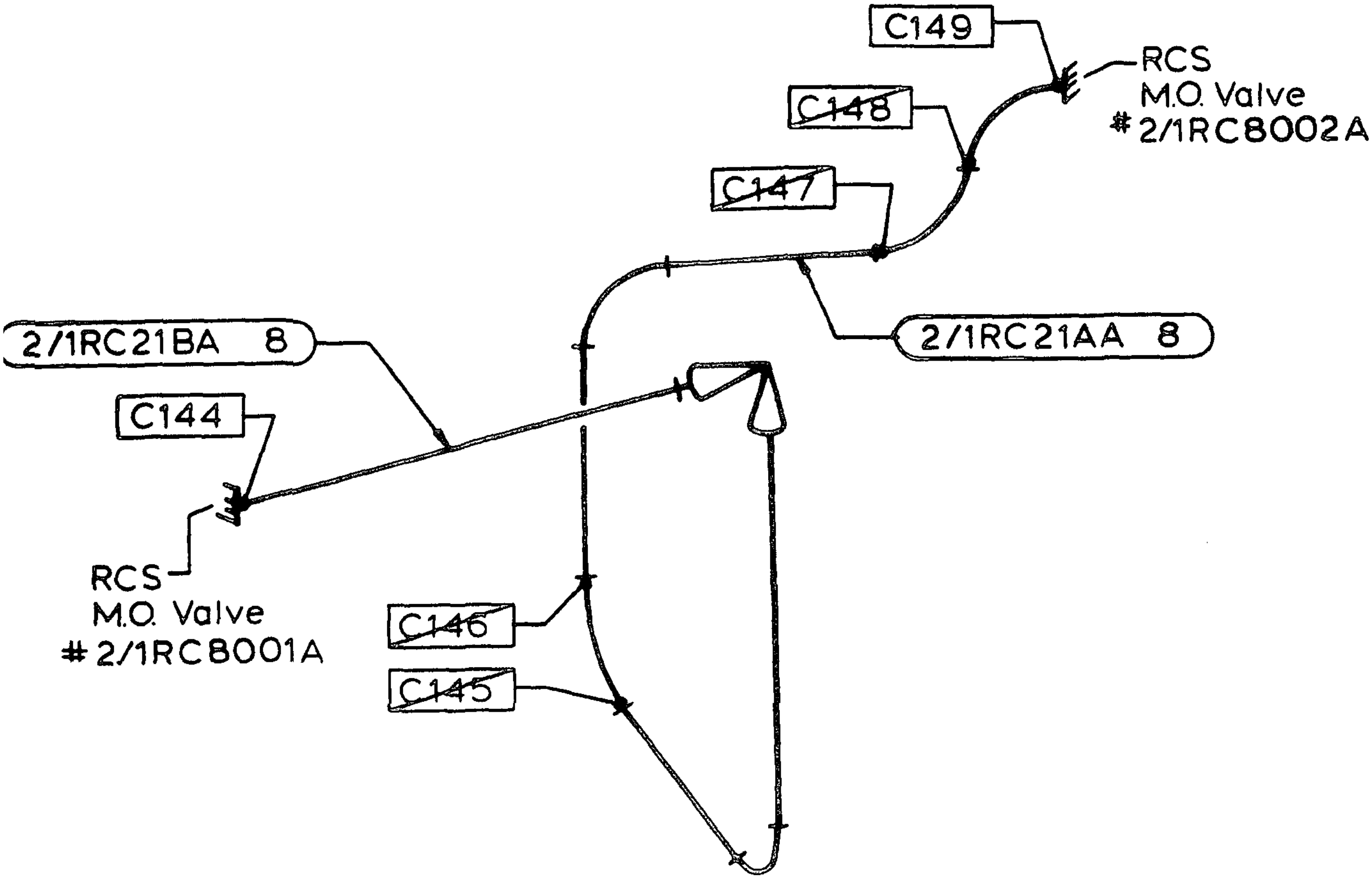
LOOP 4

NOTE:
The dynamic effects associated with these breaks are no longer considered due to application of the leak-before-break concept.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-63

REACTOR COOLANT LOOP 4
POSTULATED BREAK LOCATIONS



SUBSYSTEM RC01

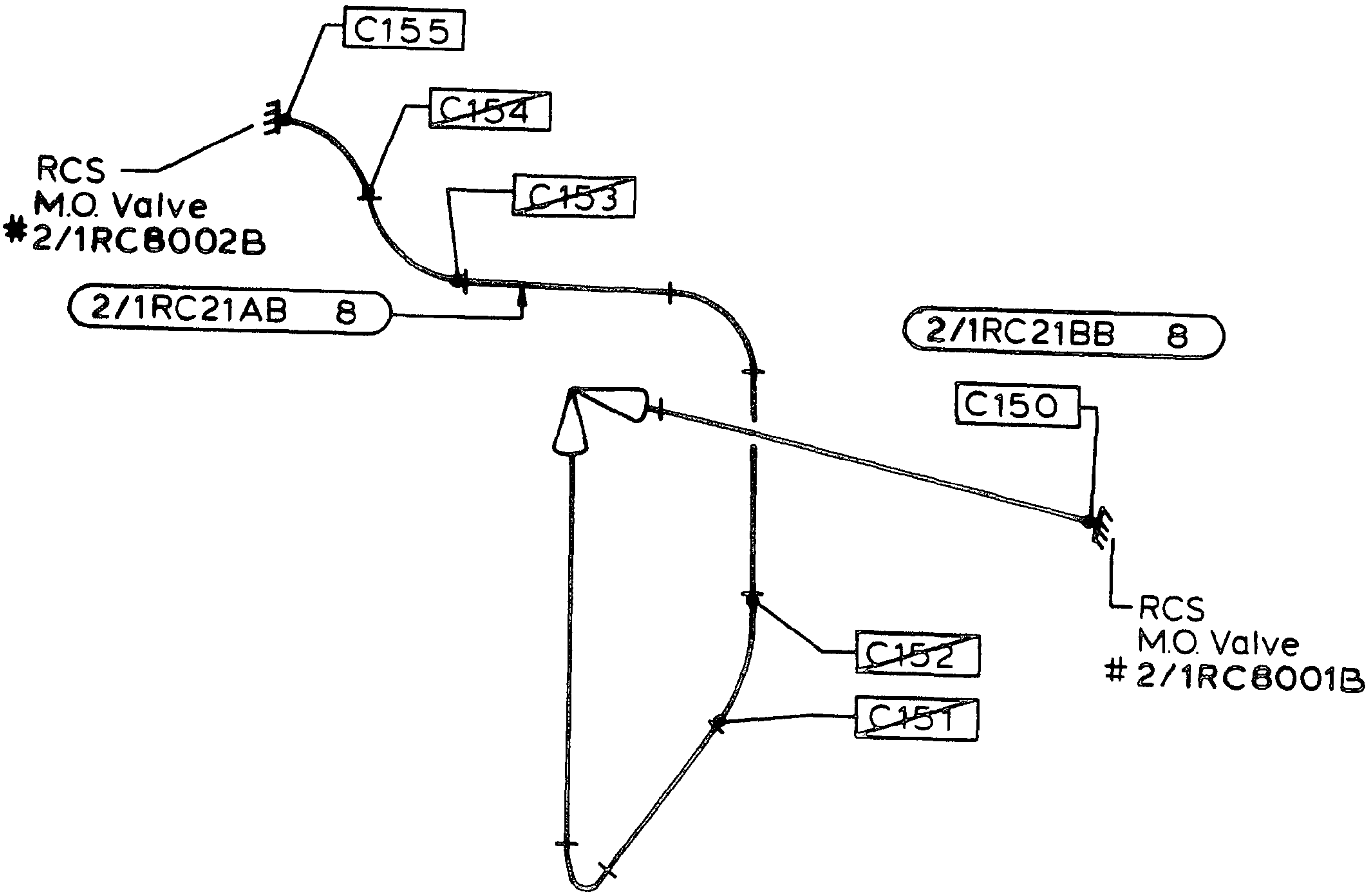
• Indicates Postulated Break.

NOTE:
▭ Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-64

REACTOR COOLANT BYPASS LOOP 1
POSTULATED BREAK LOCATIONS



SUBSYSTEM RC02

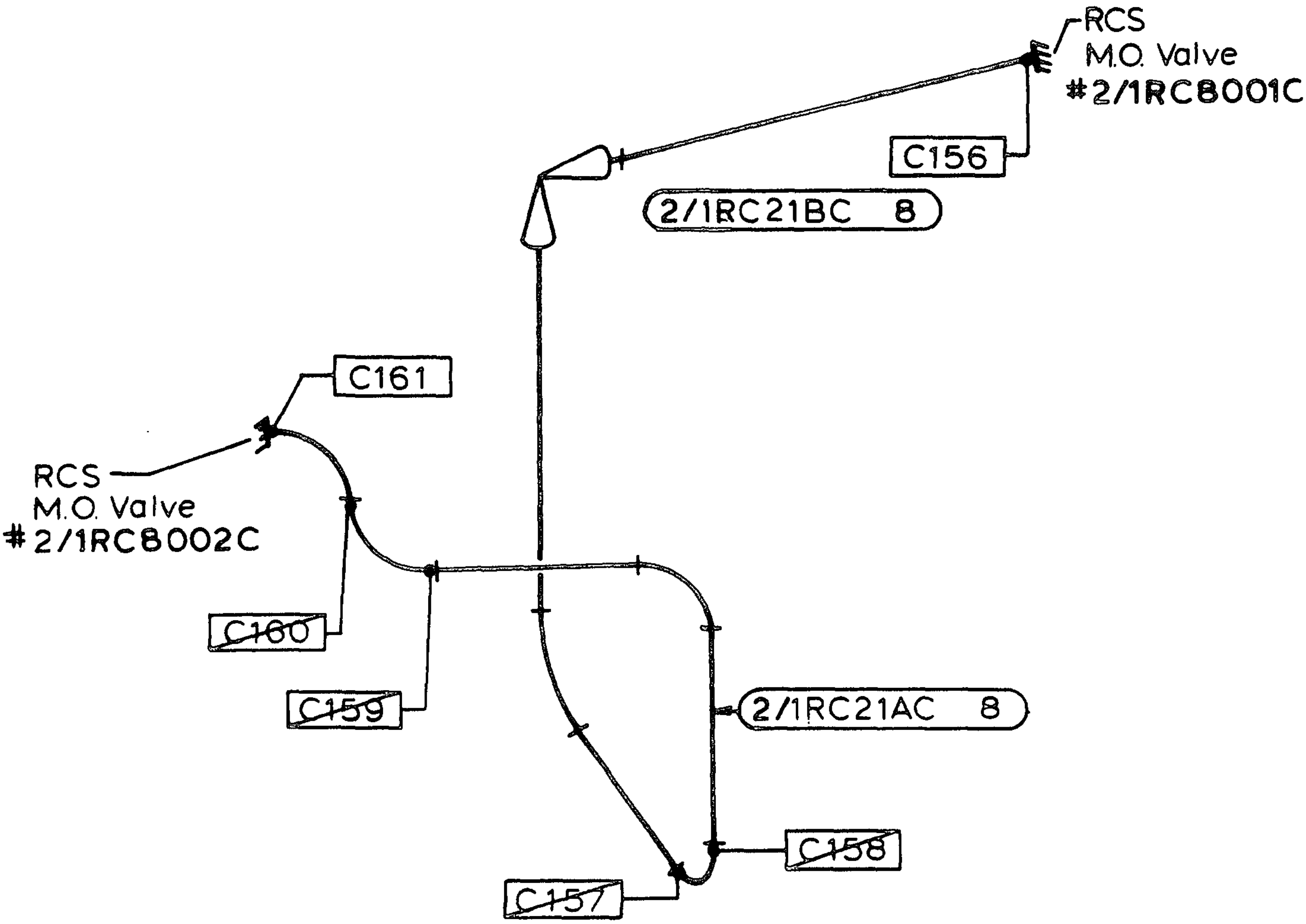
• Indicates Postulated Break

NOTE:
Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-65

REACTOR COOLANT BYPASS LOOP 2
POSTULATED BREAK LOCATIONS



SUBSYSTEM RC03

⊛ Indicate Postulated Break

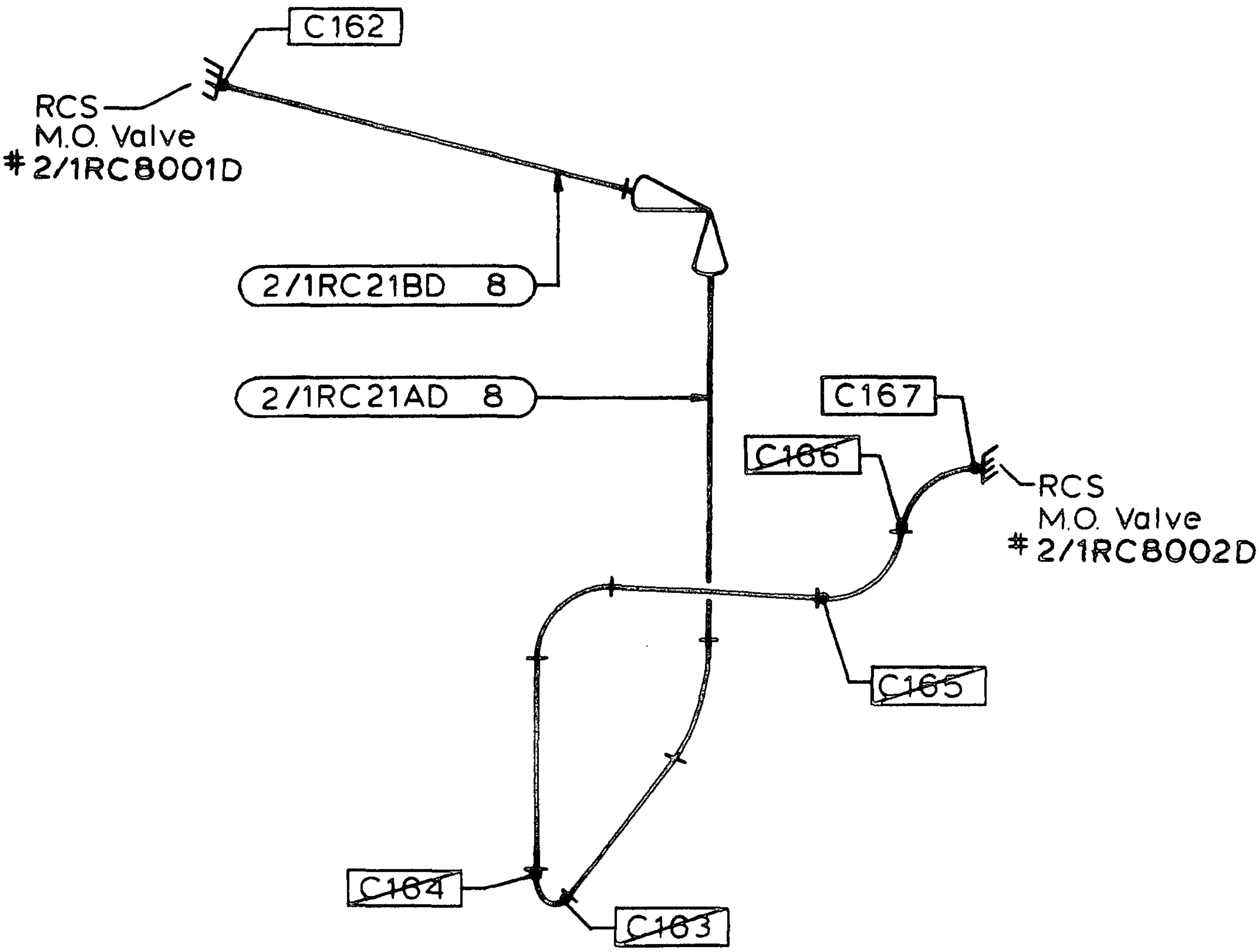
NOTE:

▱ Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-66

REACTOR COOLANT BYPASS LOOP 3
POSTULATED BREAK LOCATIONS



SUBSYSTEM RC04

• Indicates Postulated Break

NOTE:

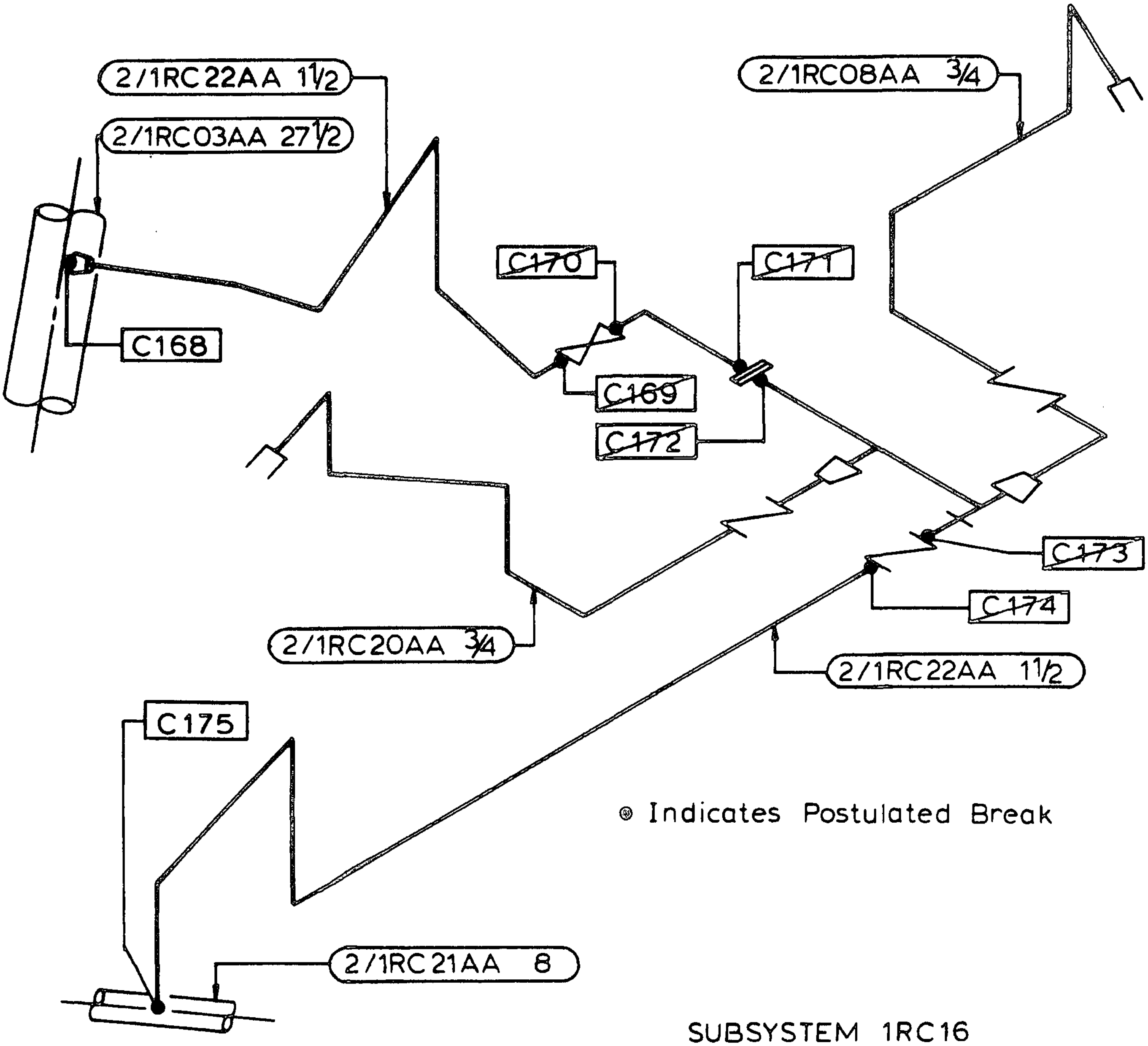
▭ Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-67

REACTOR COOLANT BYPASS LOOP 4
POSTULATED BREAK LOCATIONS

Figures 3.6-68 through 3.6-71 have been deleted intentionally.



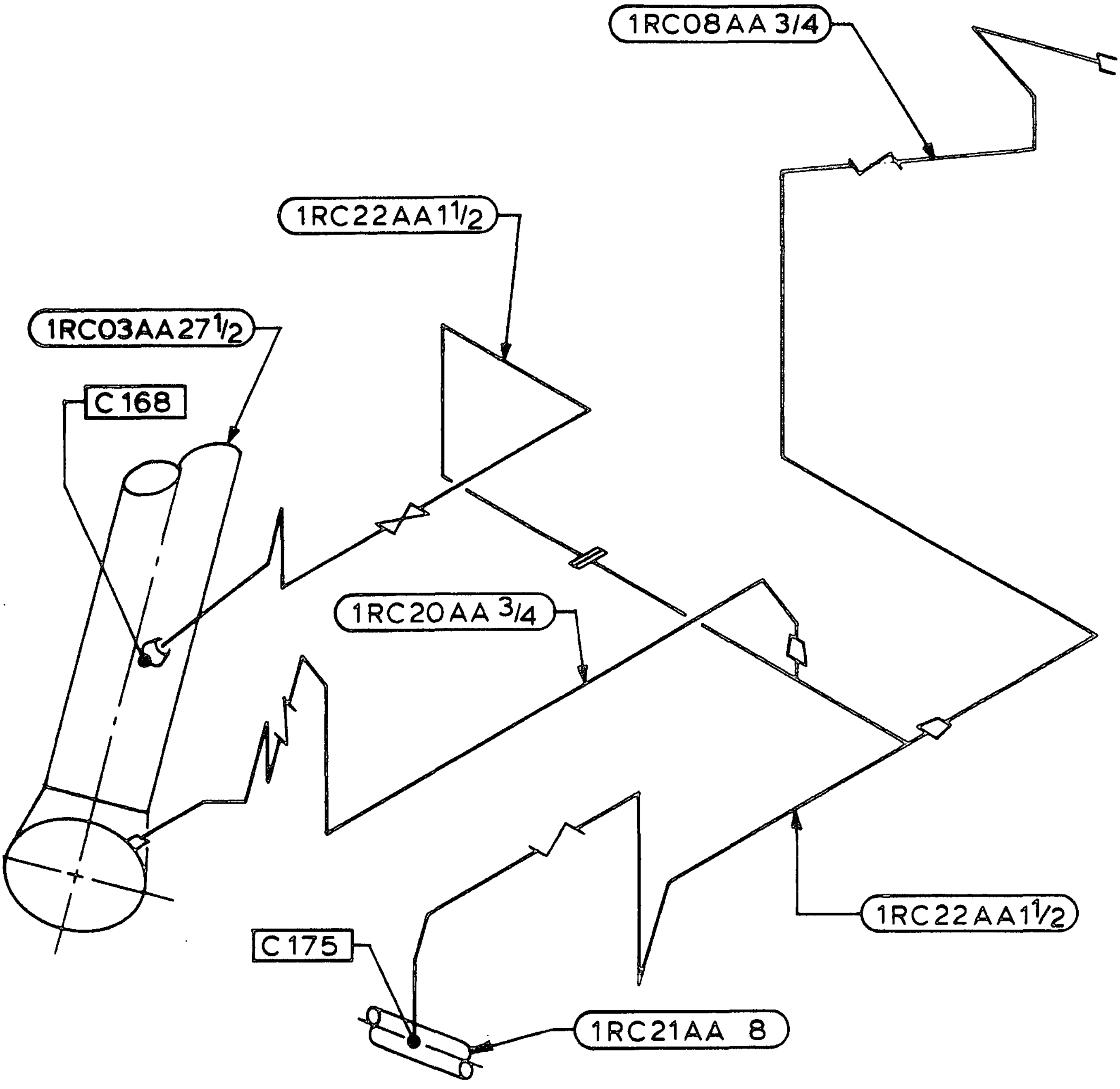
SUBSYSTEM 1RC16
BYRON ONLY

Note
Arbitrary Intermediate Break

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-72

REACTOR COOLANT SUBSYSTEM RC16
POSTULATED BREAK LOCATIONS
(BYRON)



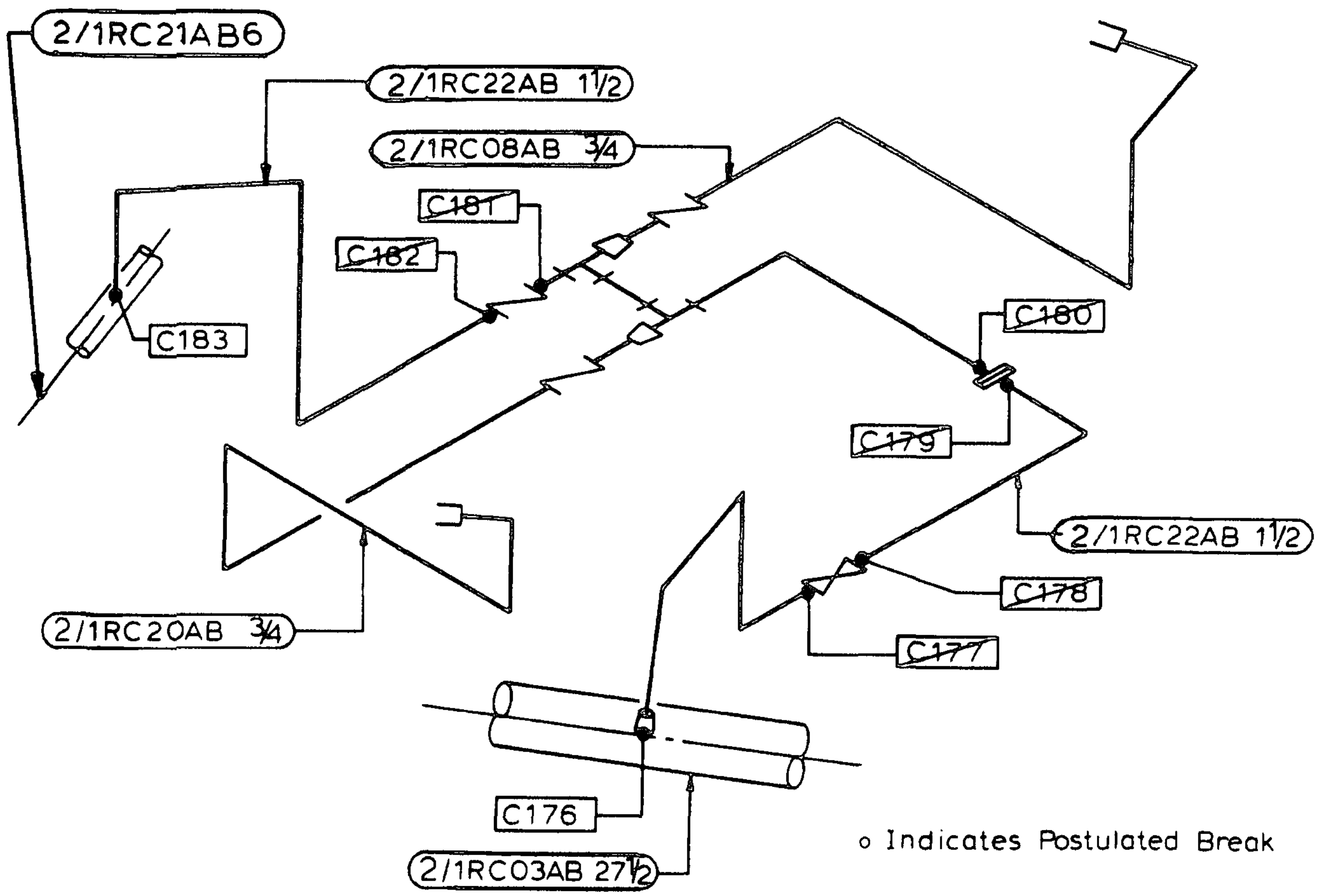
⊛ Indicates Postulated Break

SUBSYSTEM RC 16
BRAIDWOOD ONLY

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-72a

REACTOR COOLANT SUBSYSTEM RC16
POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



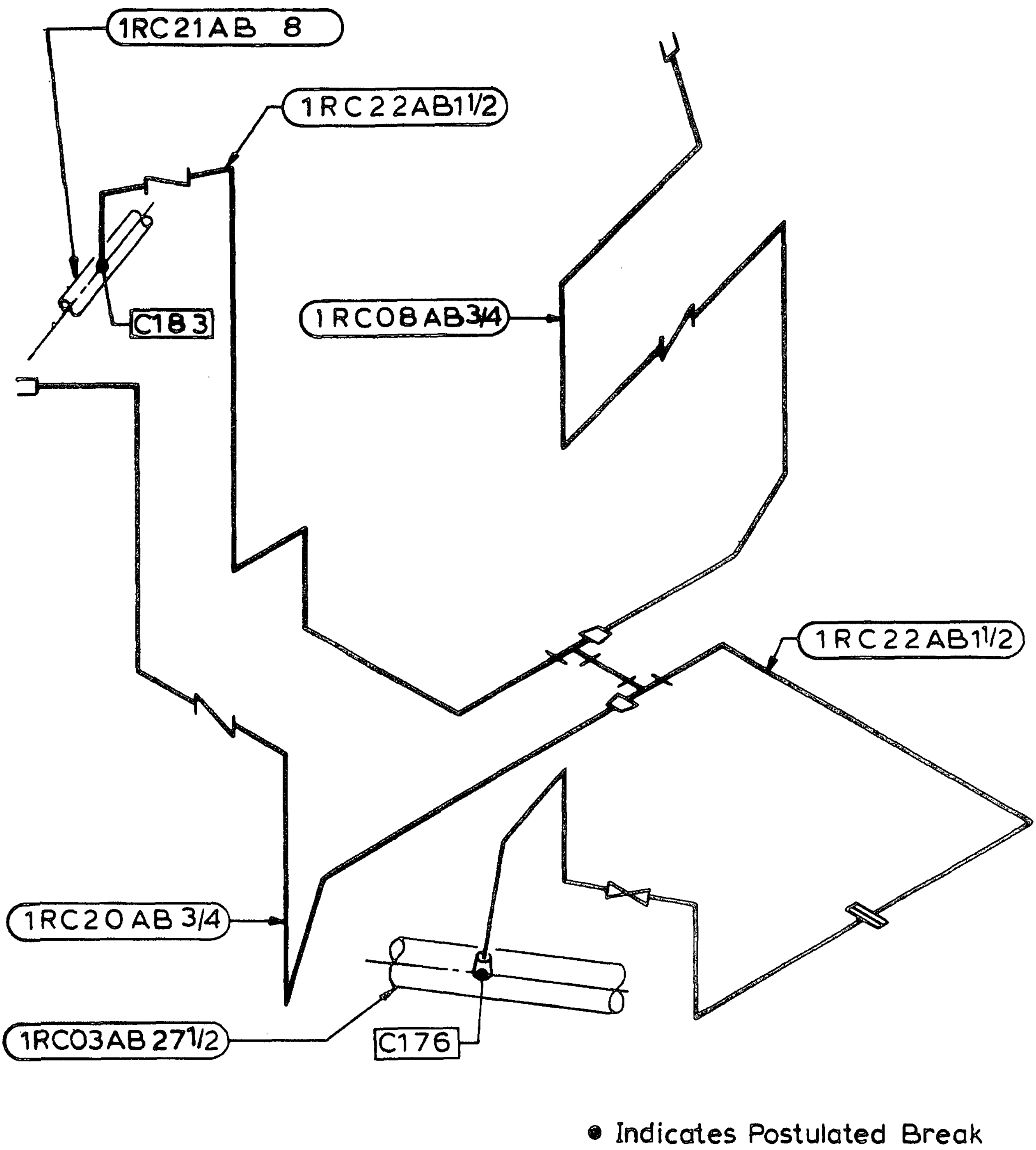
SUBSYSTEM RC17
BYRON ONLY

Note
~~   ~~ Arbitrary Intermediate Break

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-73

REACTOR COOLANT SUBSYSTEM RC17
POSTULATED BREAK LOCATIONS
(BYRON)

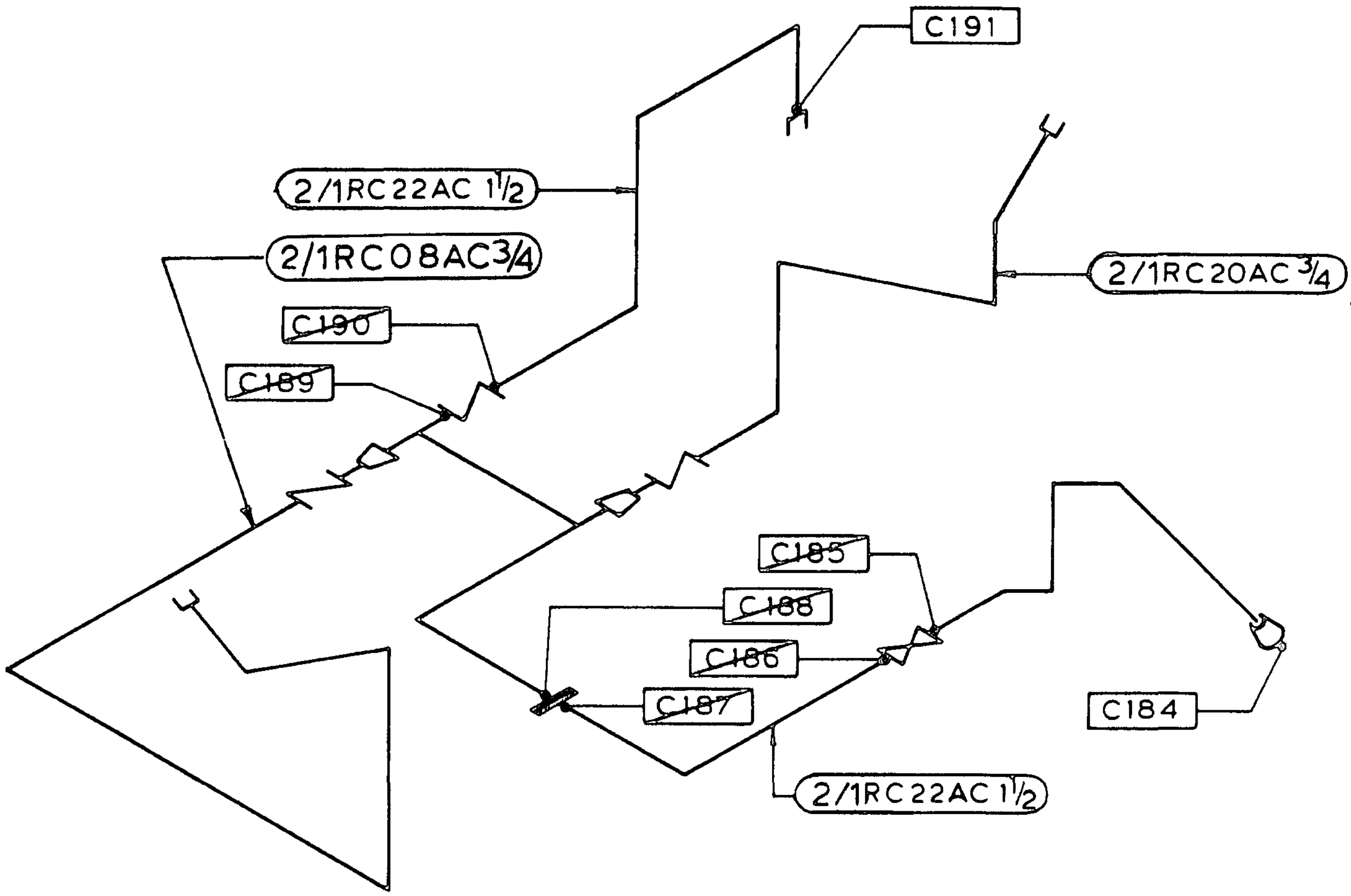


SUBSYSTEM RC17
BRAIDWOOD ONLY

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-73a

REACTOR COOLANT SUBSYSTEM RC17
POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



• Indicates Postulated Break

SUBSYSTEM RC18
BYRON ONLY

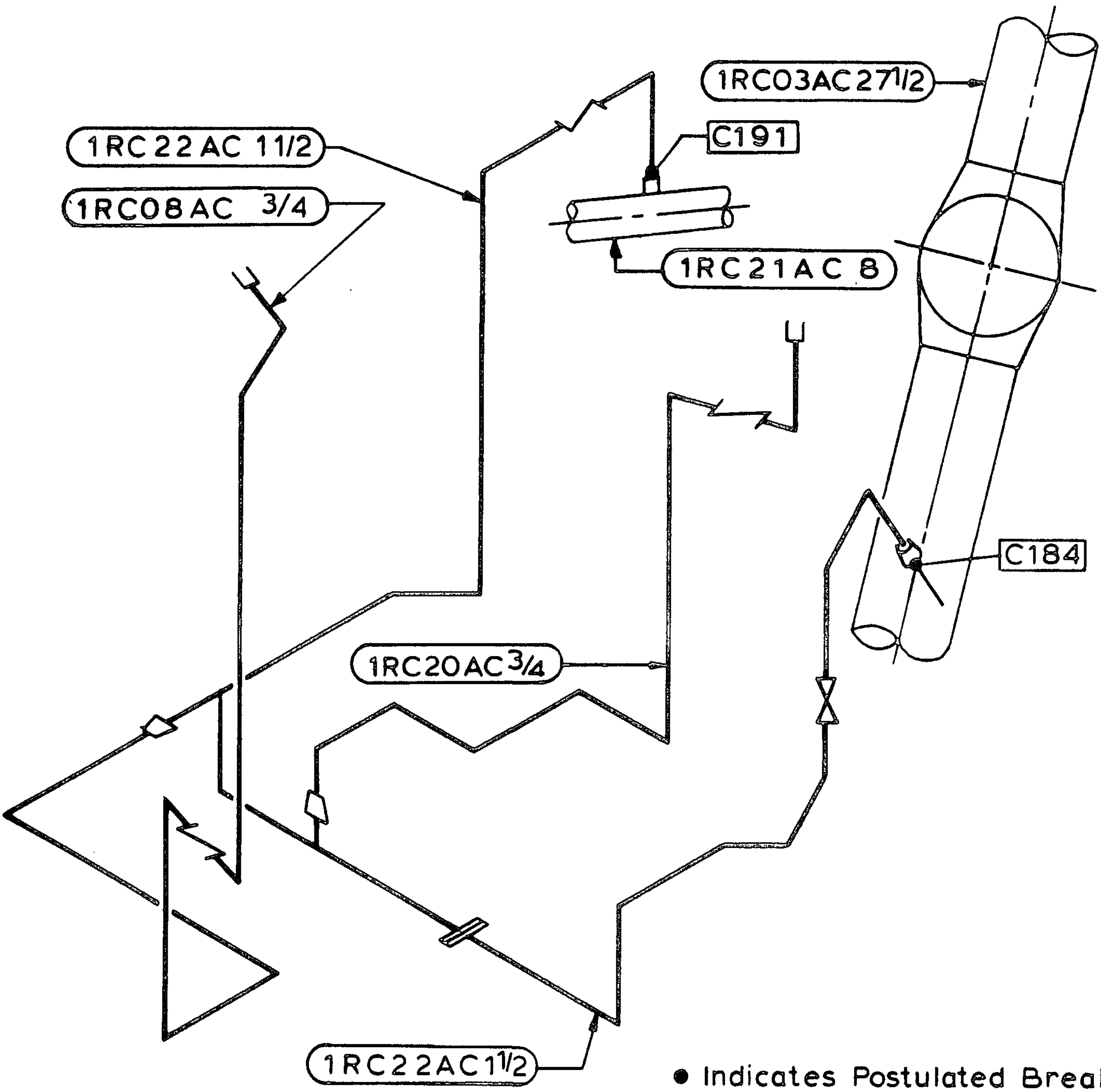
Note:

Arbitrary Intermediate Break

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-74

REACTOR COOLANT SUBSYSTEM RC18
POSTULATED BREAK LOCATIONS
(BYRON)



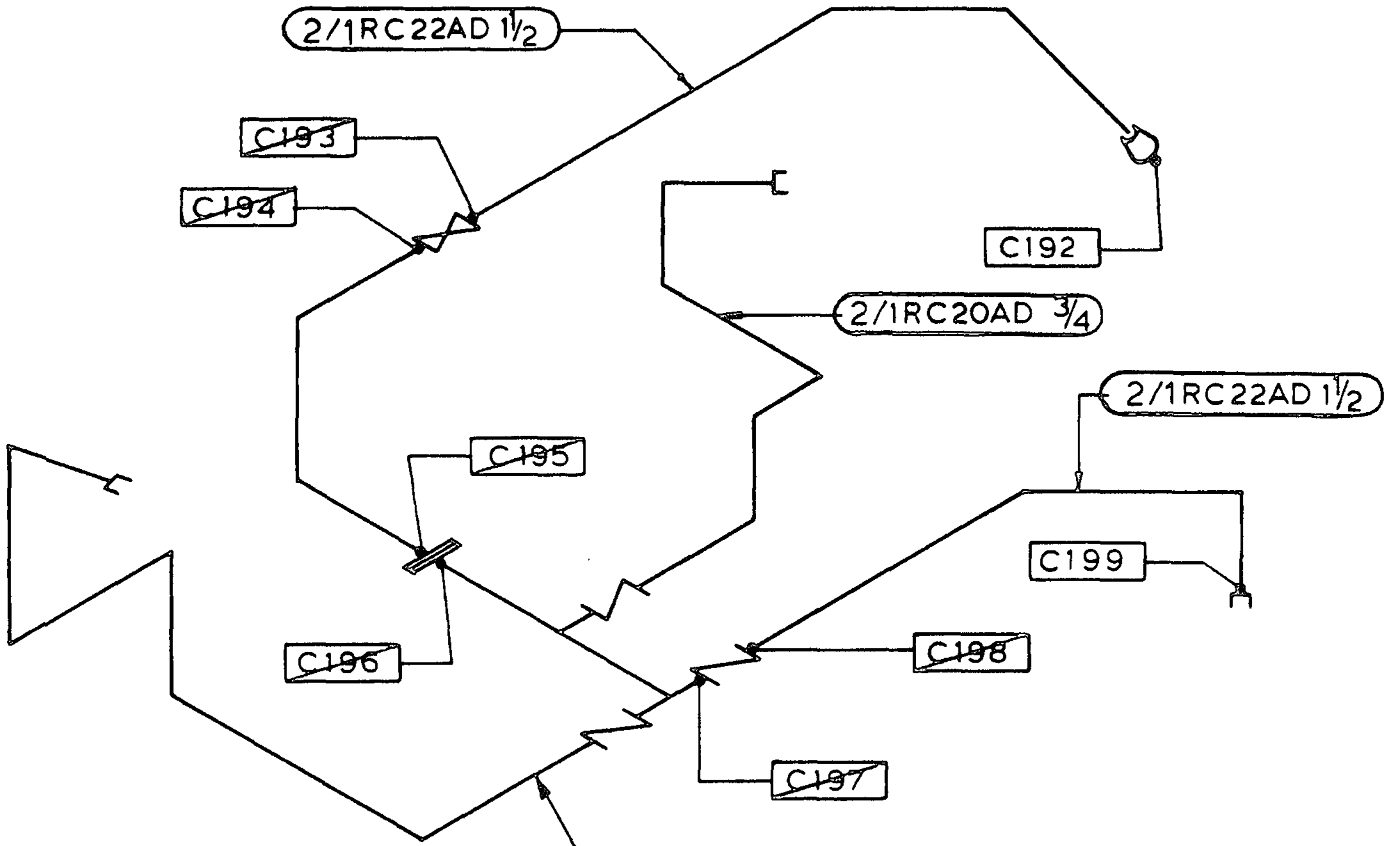
● Indicates Postulated Break

SUBSYSTEM RC18
BRAIDWOOD ONLY

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-74a

REACTOR COOLANT SUBSYSTEM RC18
POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



› Indicates Postulated Break

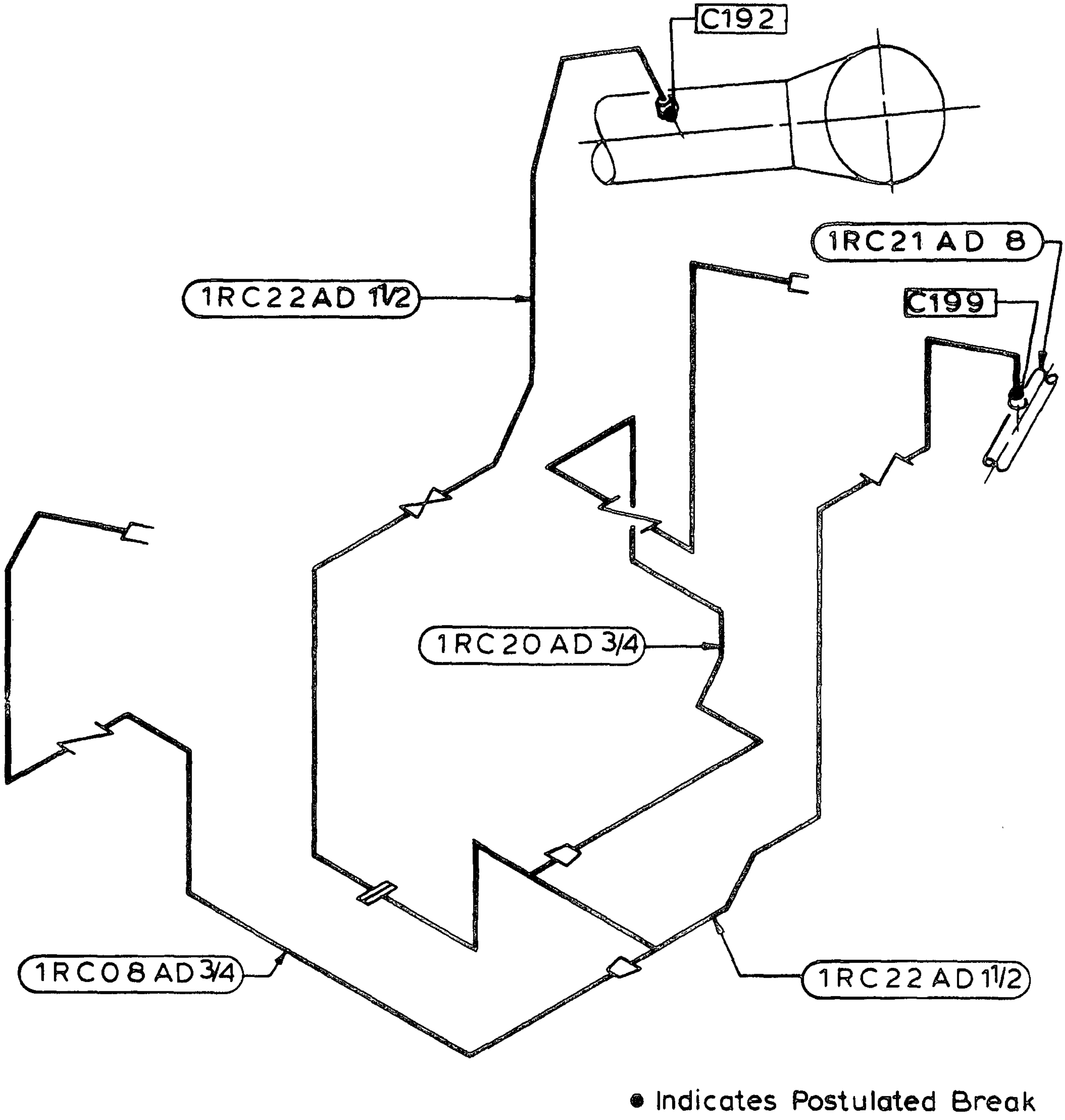
SUBSYSTEM RC19
BYRON ONLY

Note:
Arbitrary Intermediate Break

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-75

REACTOR COOLANT SUBSYSTEM RC19
POSTULATED BREAK LOCATIONS
(BYRON)

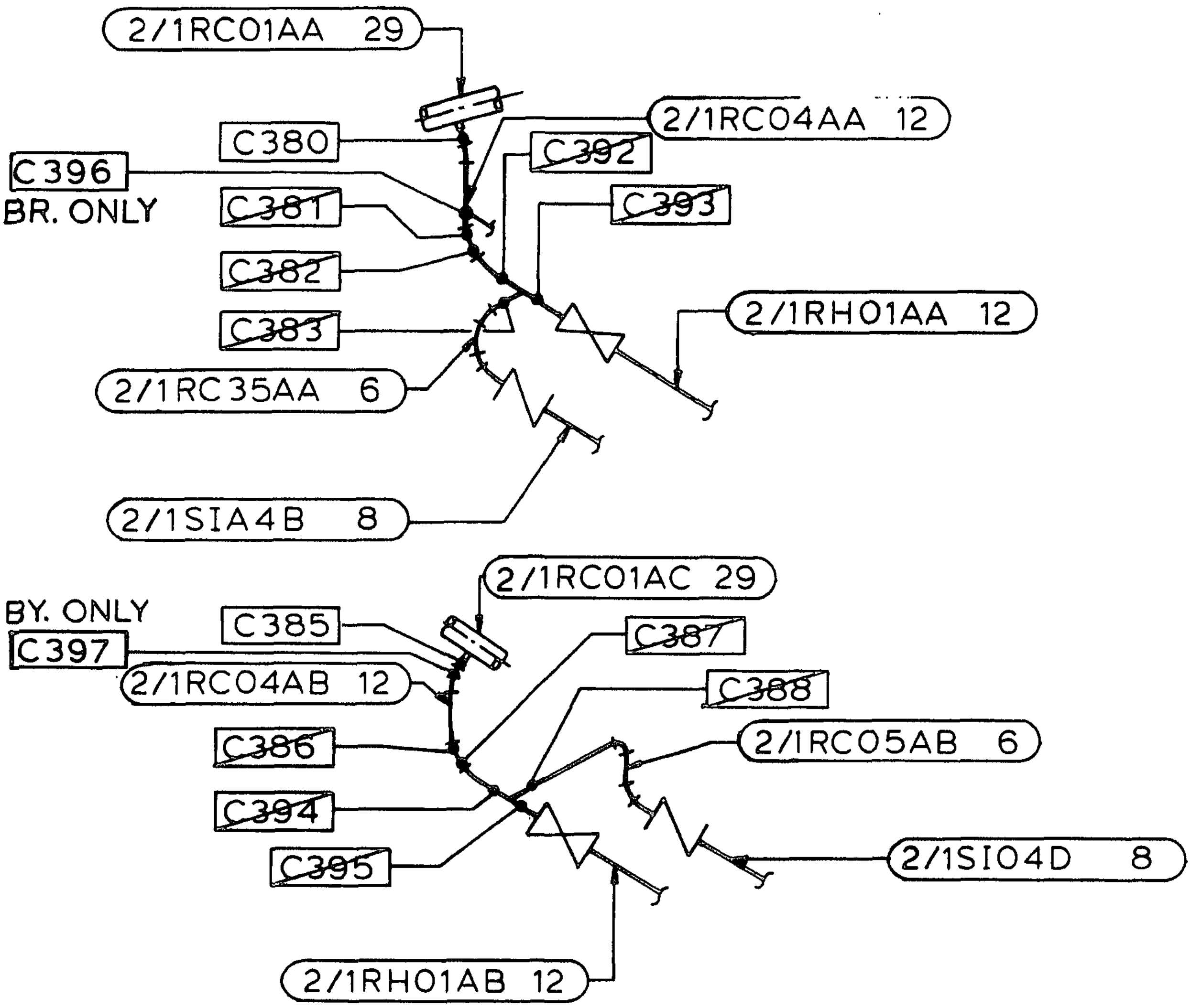


SUBSYSTEM RC19
BRAIDWOOD ONLY

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-75a

REACTOR COOLANT SUBSYSTEM RC19
POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



SUBSYSTEM RH02

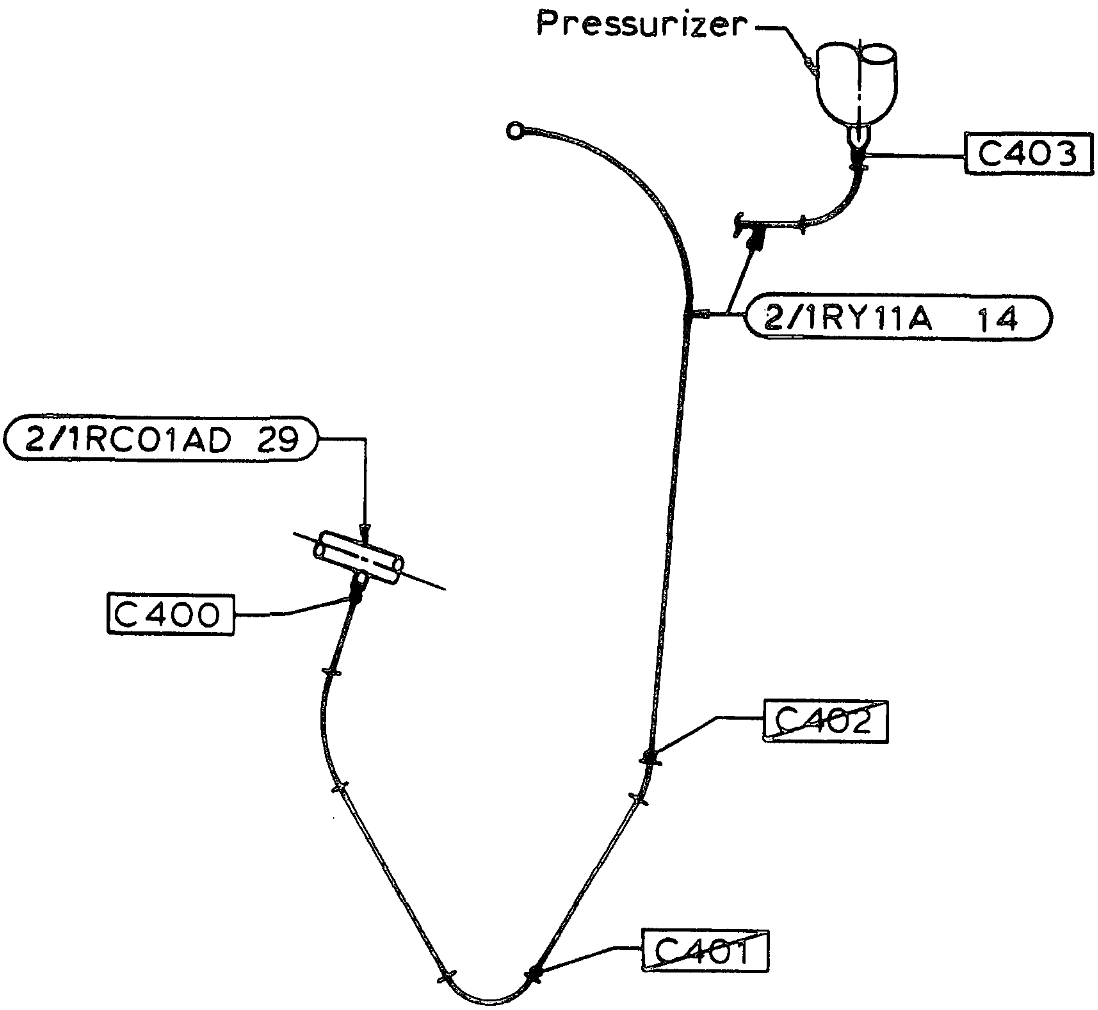
◦ Indicates Postulated Break

NOTE:
Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-76

RESIDUAL HEAT REMOVAL LOOPS 1 AND 3
POSTULATED BREAK LOCATIONS



SUBSYSTEM RY05

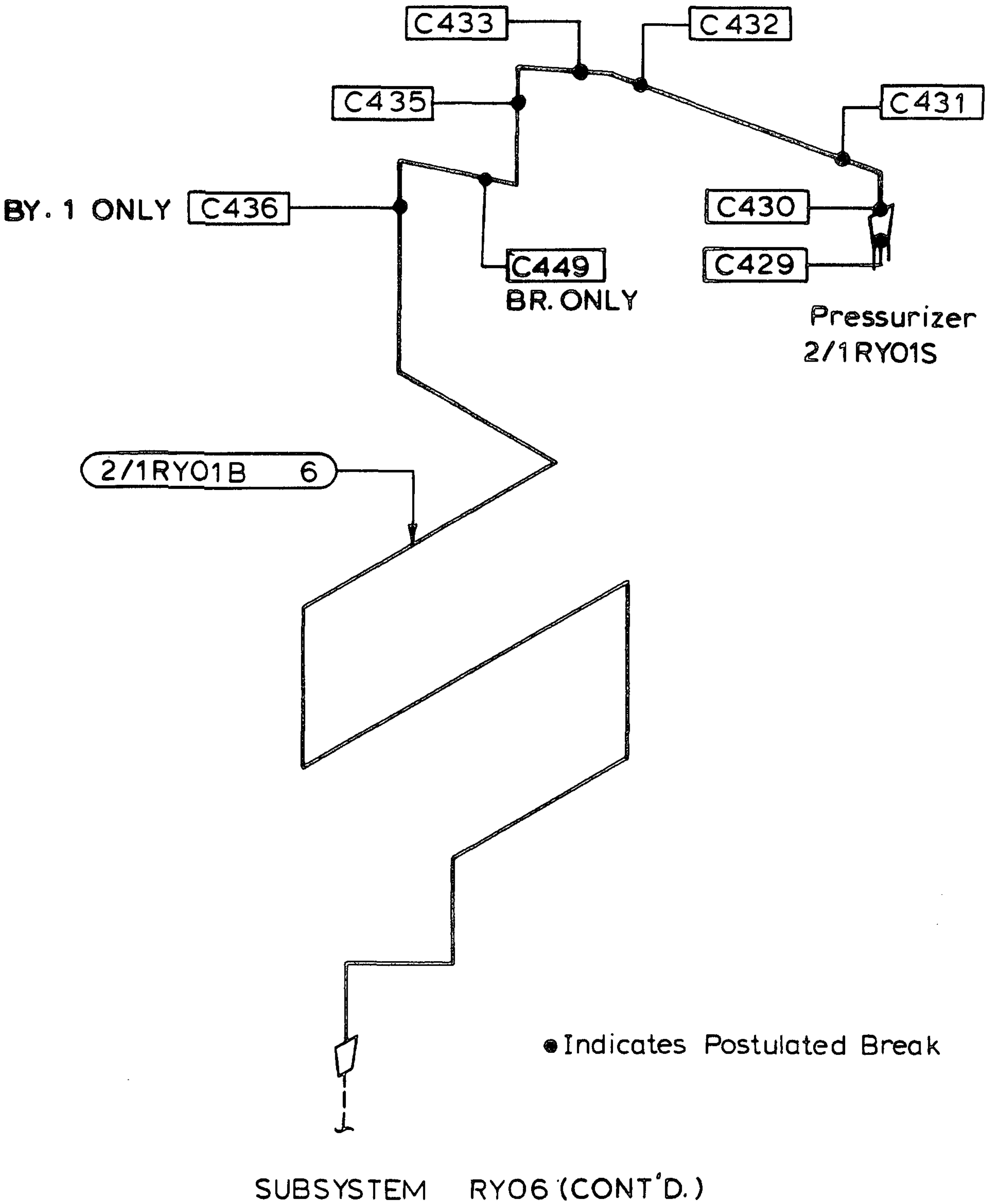
◎ Indicates Postulated Break

NOTE:
Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-77

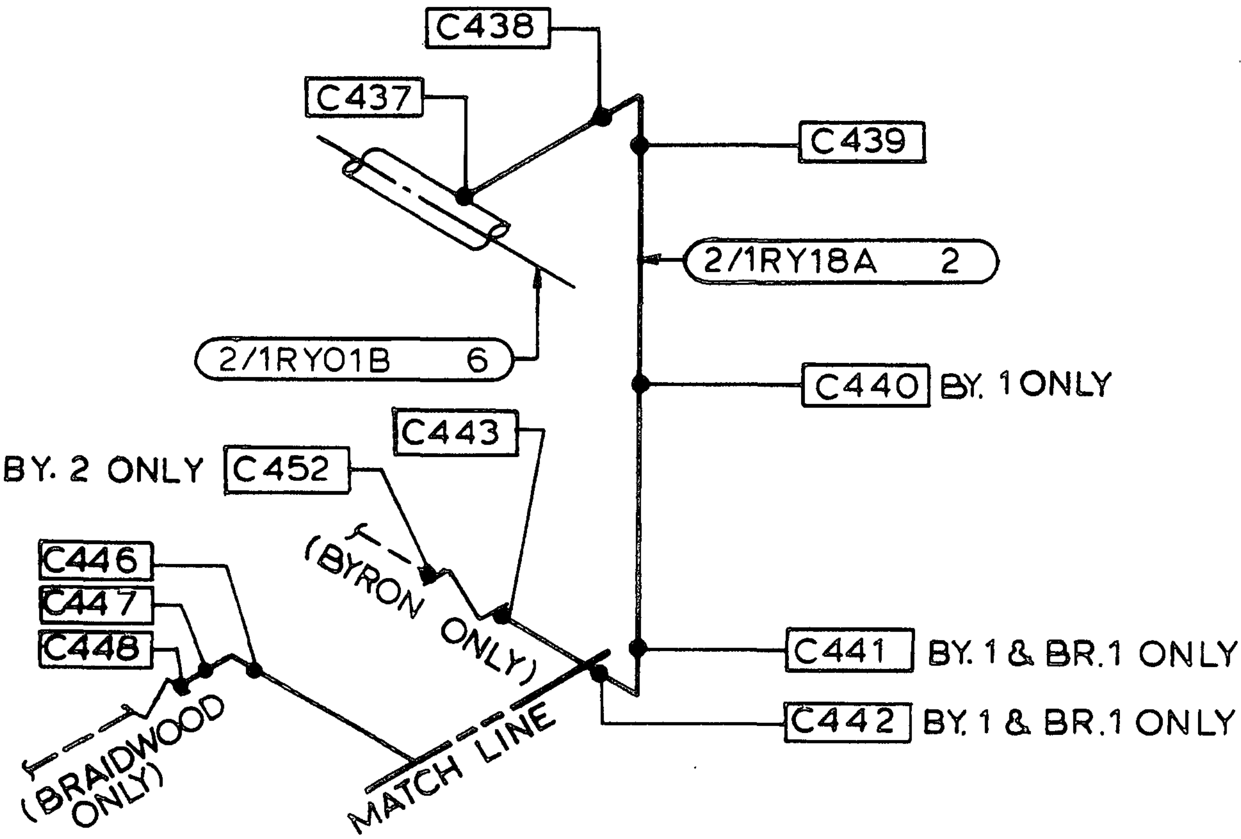
PRESSURIZER SURGE LINE
POSTULATED BREAK LOCATIONS



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-78

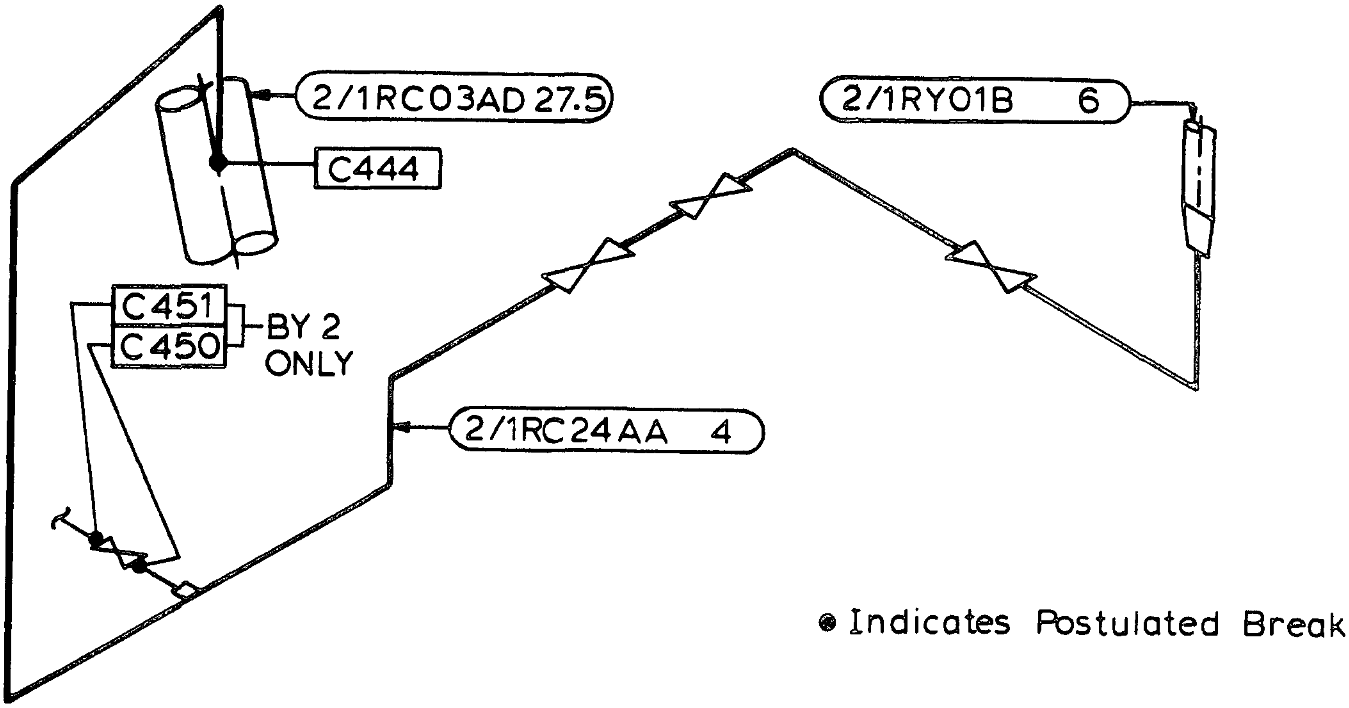
PRESSURIZER SPRAY LINE
POSTULATED BREAK LOCATIONS
(SHEET 1 OF 4)



SUBSYSTEM RY06 (Cont'd.)

● Indicates Postulated Break

BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-78 PRESSURIZER SPRAY LINE POSTULATED BREAK LOCATIONS (SHEET 2 OF 4)

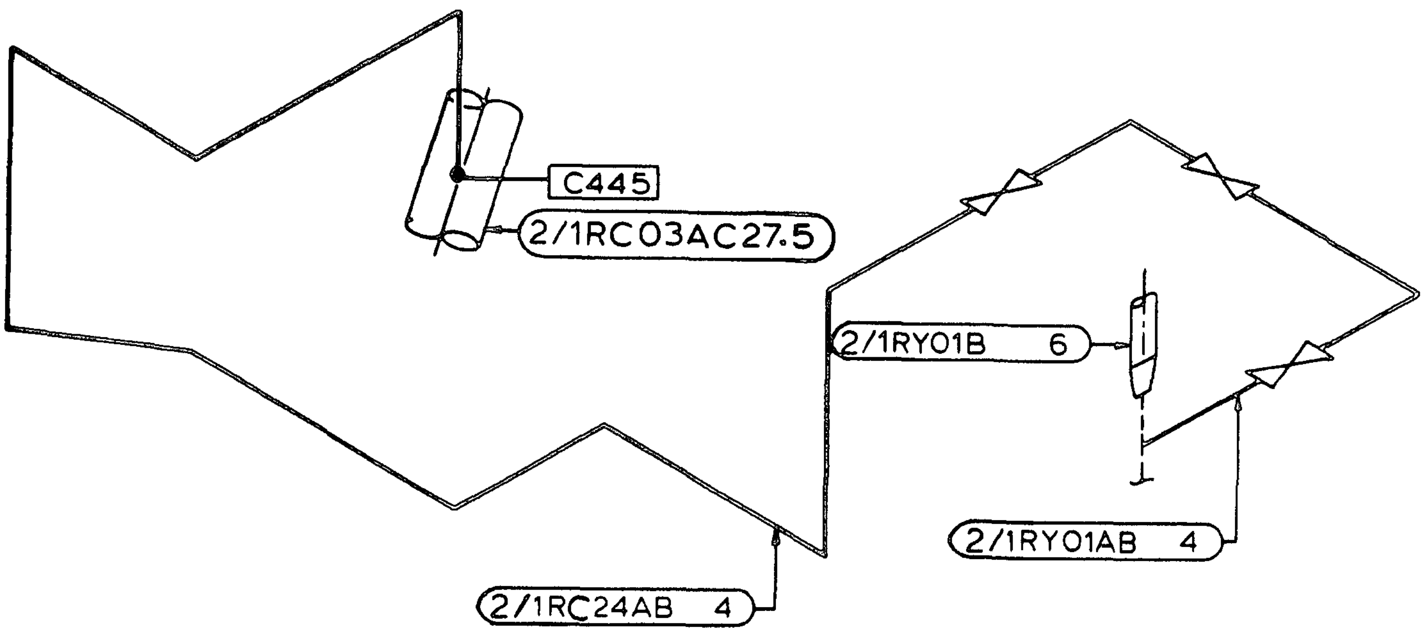


SUBSYSTEM RY06 (Cont'd.)

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-78

PRESSURIZER SPRAY LINE
POSTULATED BREAK LOCATIONS
(SHEET 3 OF 4)



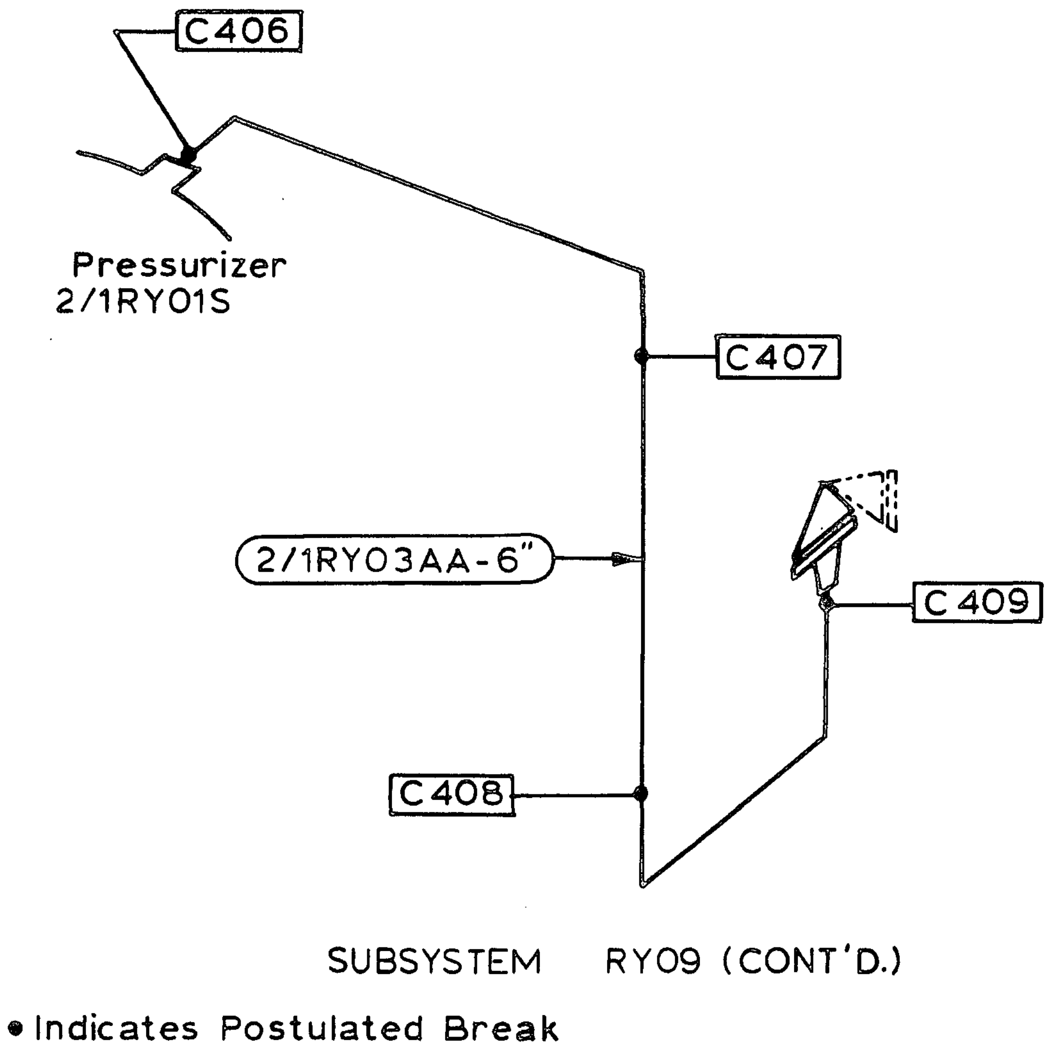
SUBSYSTEM RY06 (Cont'd.)

● Indicates Postulated Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-78

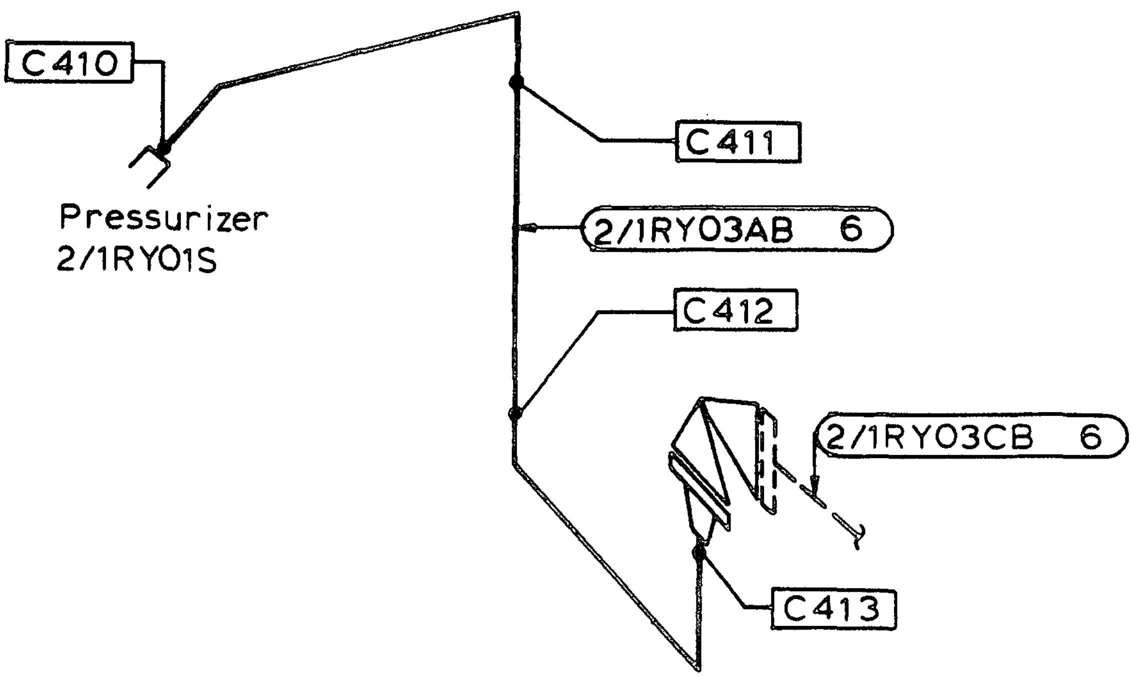
PRESSURIZER SPRAY LINE
POSTULATED BREAK LOCATIONS
(SHEET 4 OF 4)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-79

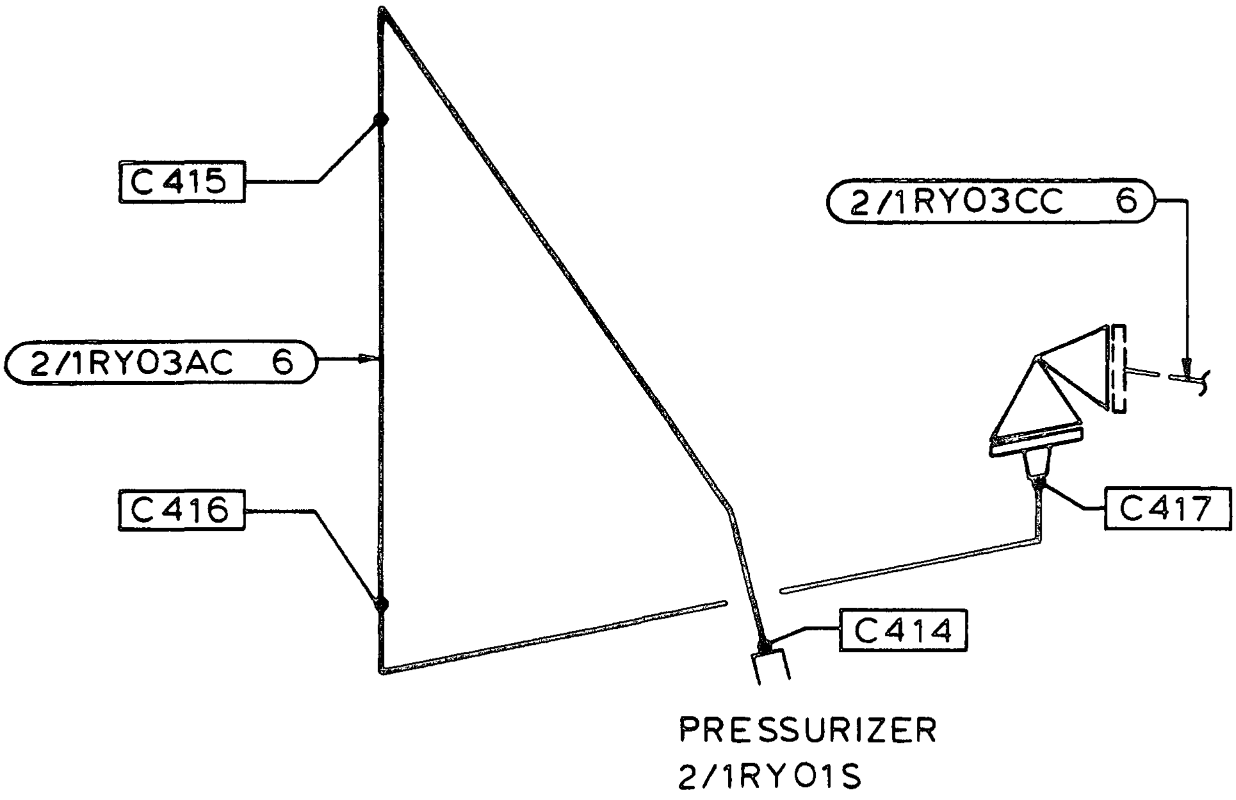
PRESSURIZER SAFETY/RELIEF VALVE LINES
POSTULATED BREAK LOCATIONS
(SHEET 1 OF 4)



• Indicates Postulated Break

SUBSYSTEM RY09 (CONT'D)

BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-79 PRESSURIZER SAFETY/RELIEF VALVE LINES POSTULATED BREAK LOCATIONS (SHEET 2 OF 4)



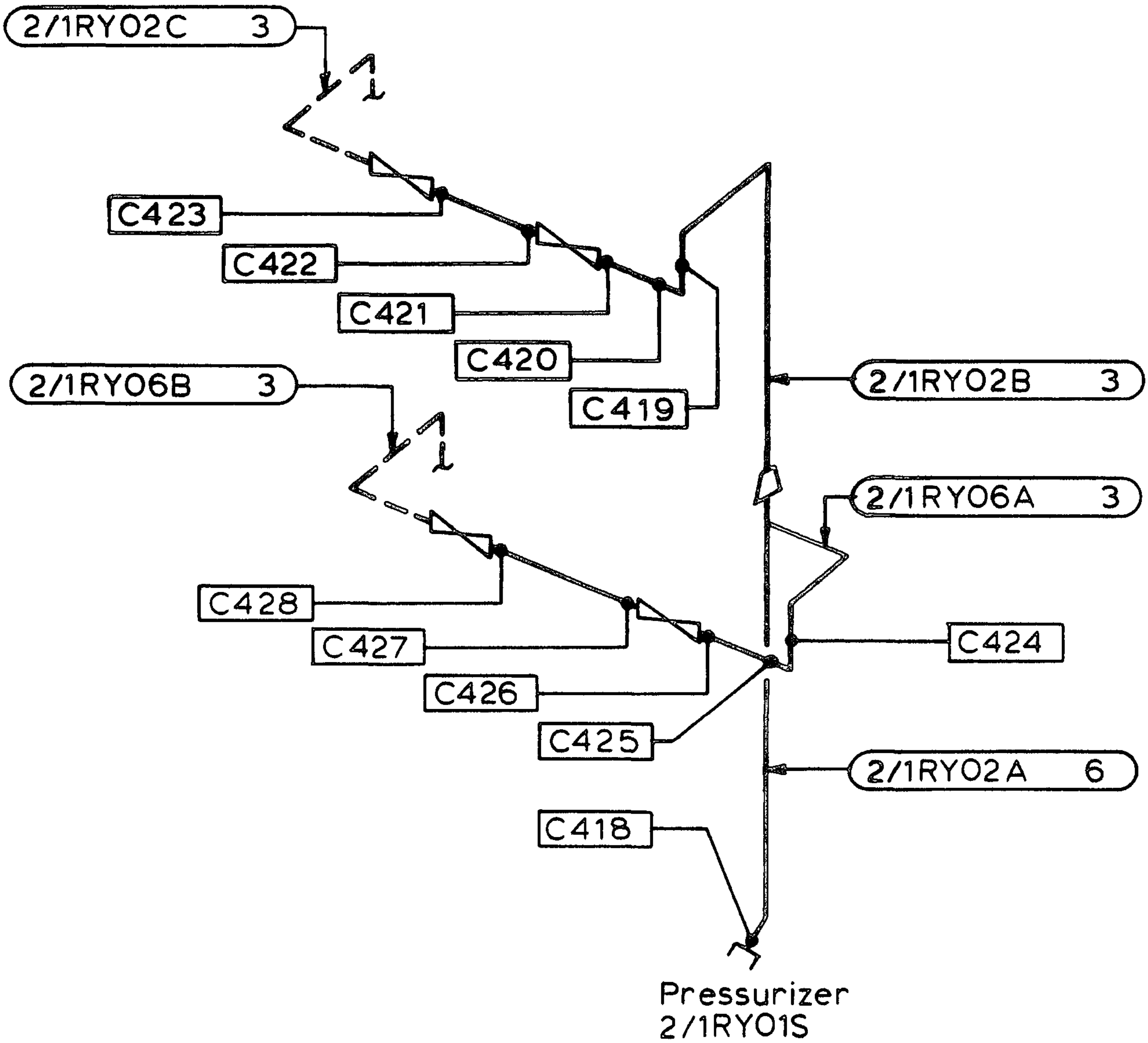
SUBSYSTEM RY09 (Cont'd)

● Indicates Postulated Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-79

PRESSURIZER SAFETY/RELIEF VALVE LINES
POSTULATED BREAK LOCATIONS
(SHEET 3 OF 4)



• Indicates Postulated Break

SUBSYSTEM RY09 (Cont'd)

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-79

PRESSURIZER SAFETY/RELIEF VALVE LINES
POSTULATED BREAK LOCATIONS
(SHEET 4 OF 4)

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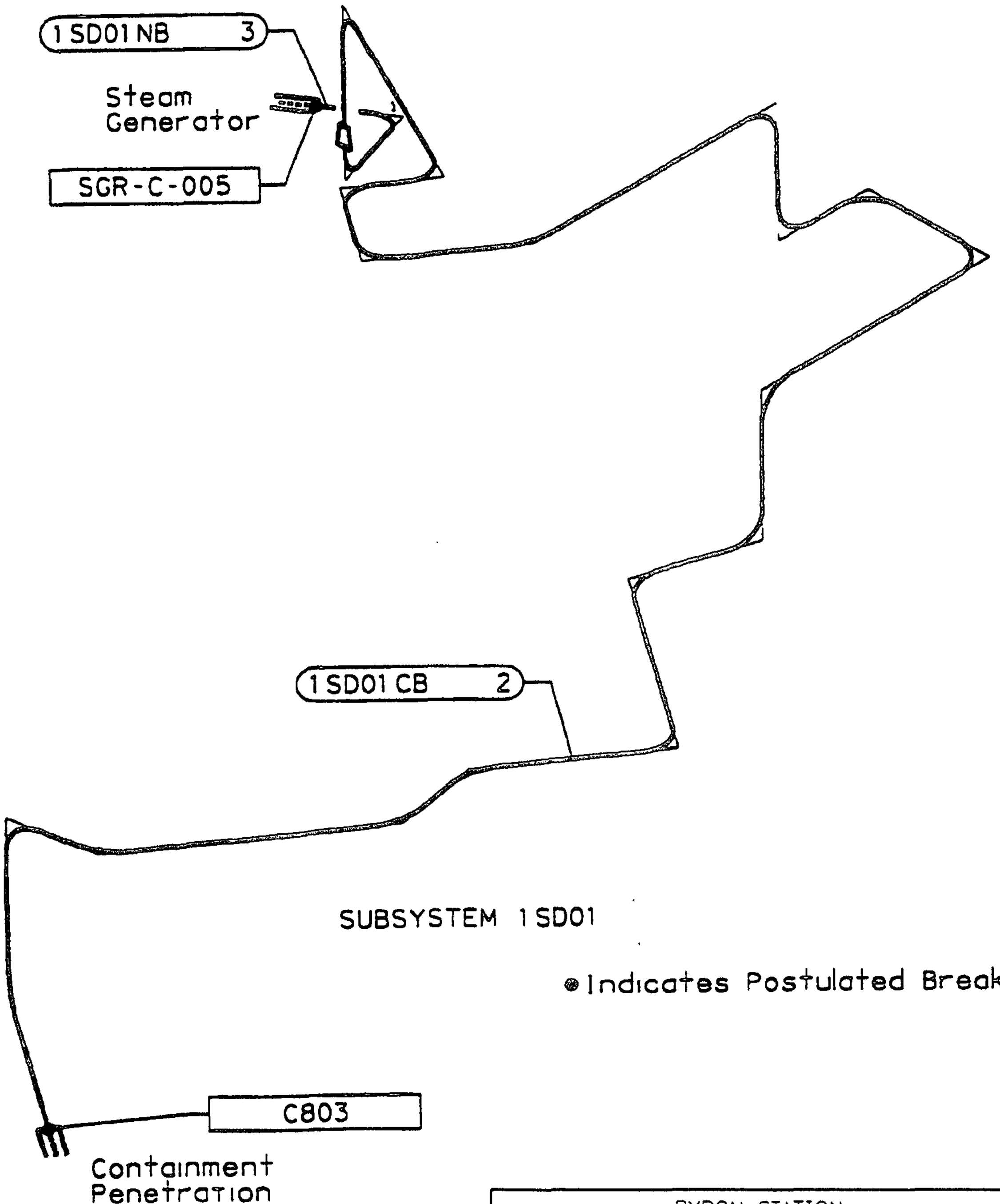
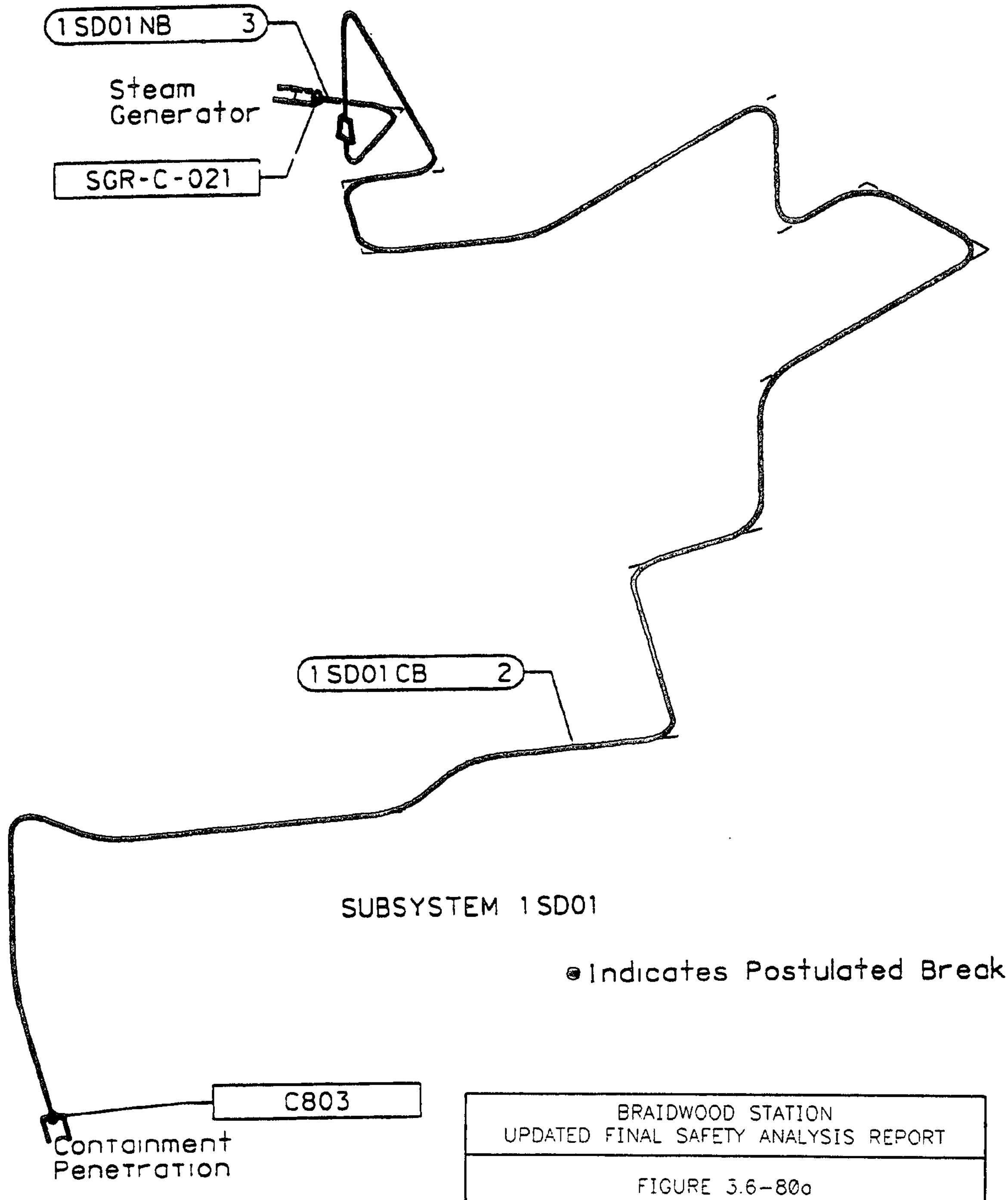


BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-80

STEAM GENERATOR BLOWDOWN LOOP 1
SUBSYSTEM 1SD01
POSTULATED BREAK LOCATIONS – UNIT 1



SUBSYSTEM 1SD01

● Indicates Postulated Break

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-80a

STEAM GENERATOR BLOWDOWN LOOP 1
SUBSYSTEM 1SD01
POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
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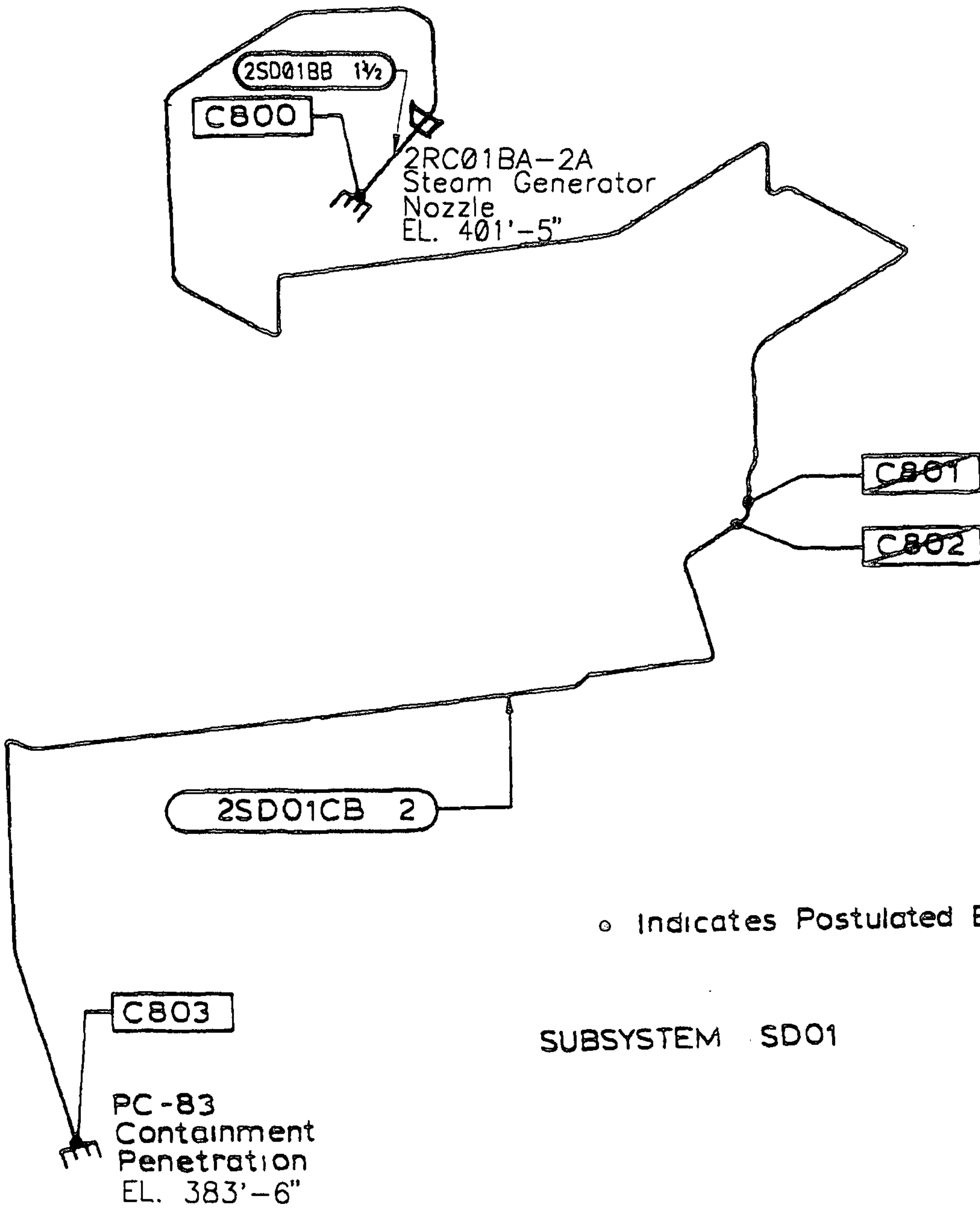


⊙ Indicates Postulated Break

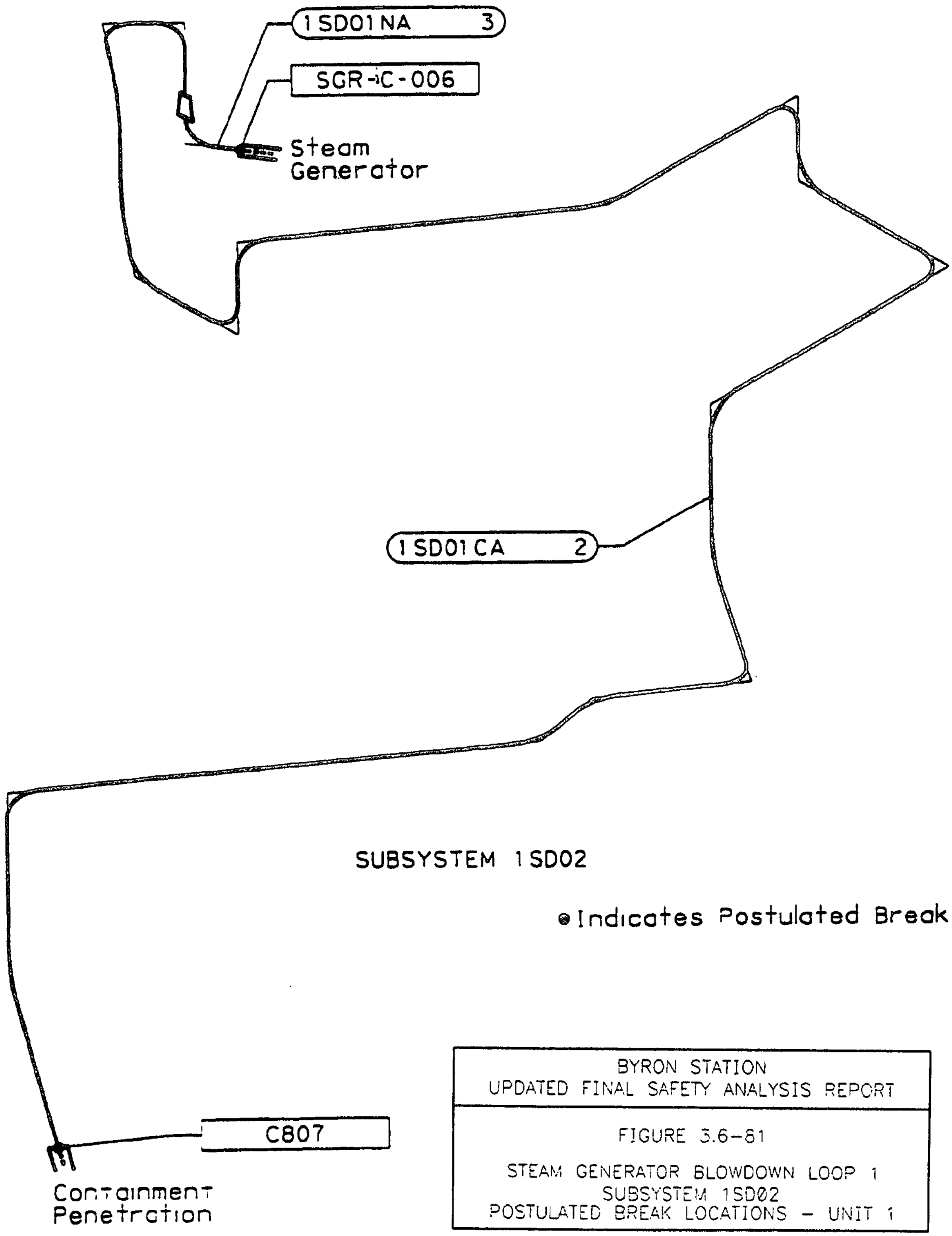
SUBSYSTEM SD01

Note

Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-80b STEAM GENERATOR BLOWDOWN LOOP 1 SUBSYSTEM SD01 POSTULATED BREAK LOCATIONS - UNIT 2

REVISION 7
DECEMBER 1998

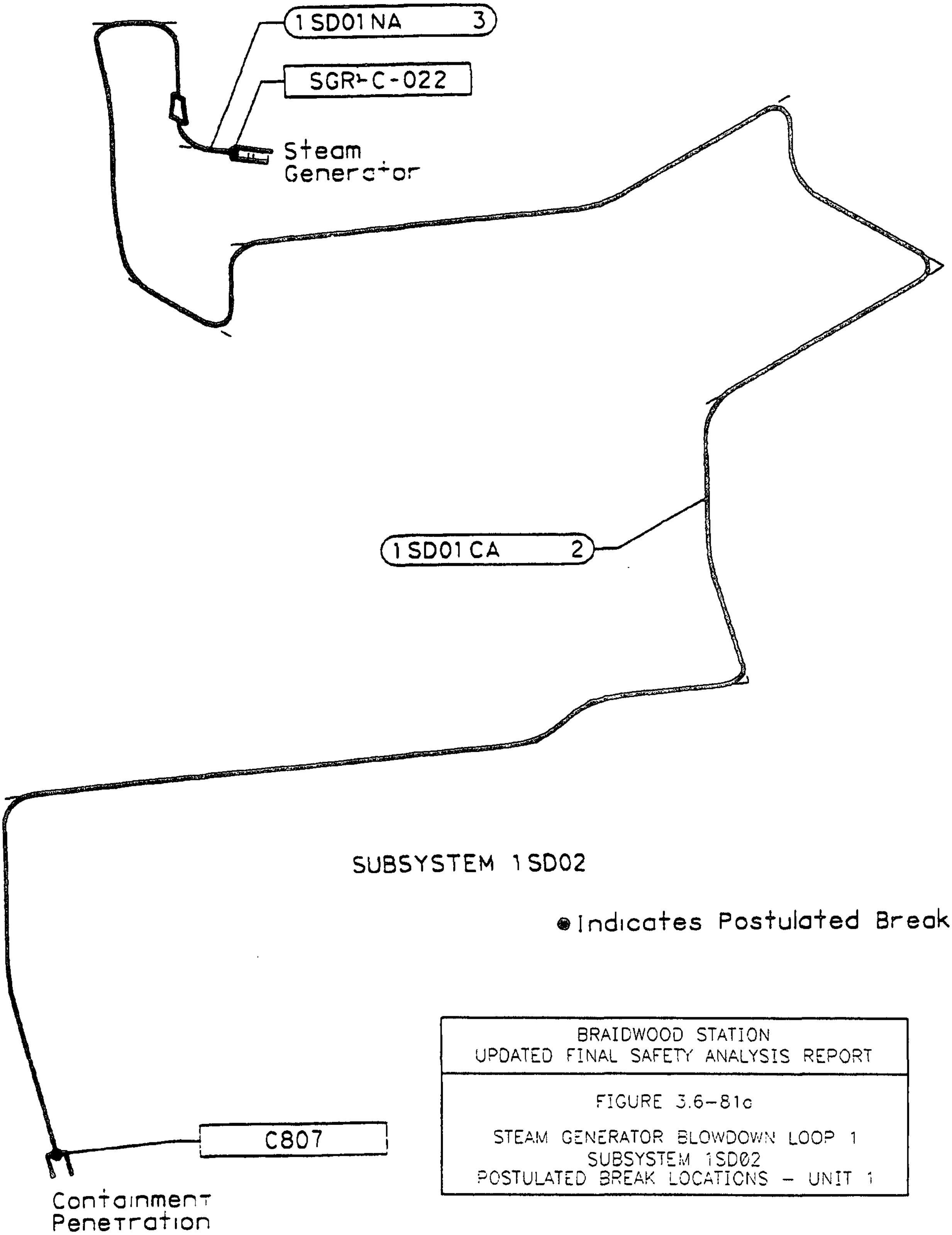


BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-81

STEAM GENERATOR BLOWDOWN LOOP 1
SUBSYSTEM 1SD02
POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-81c

STEAM GENERATOR BLOWDOWN LOOP 1
SUBSYSTEM 1SD02
POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
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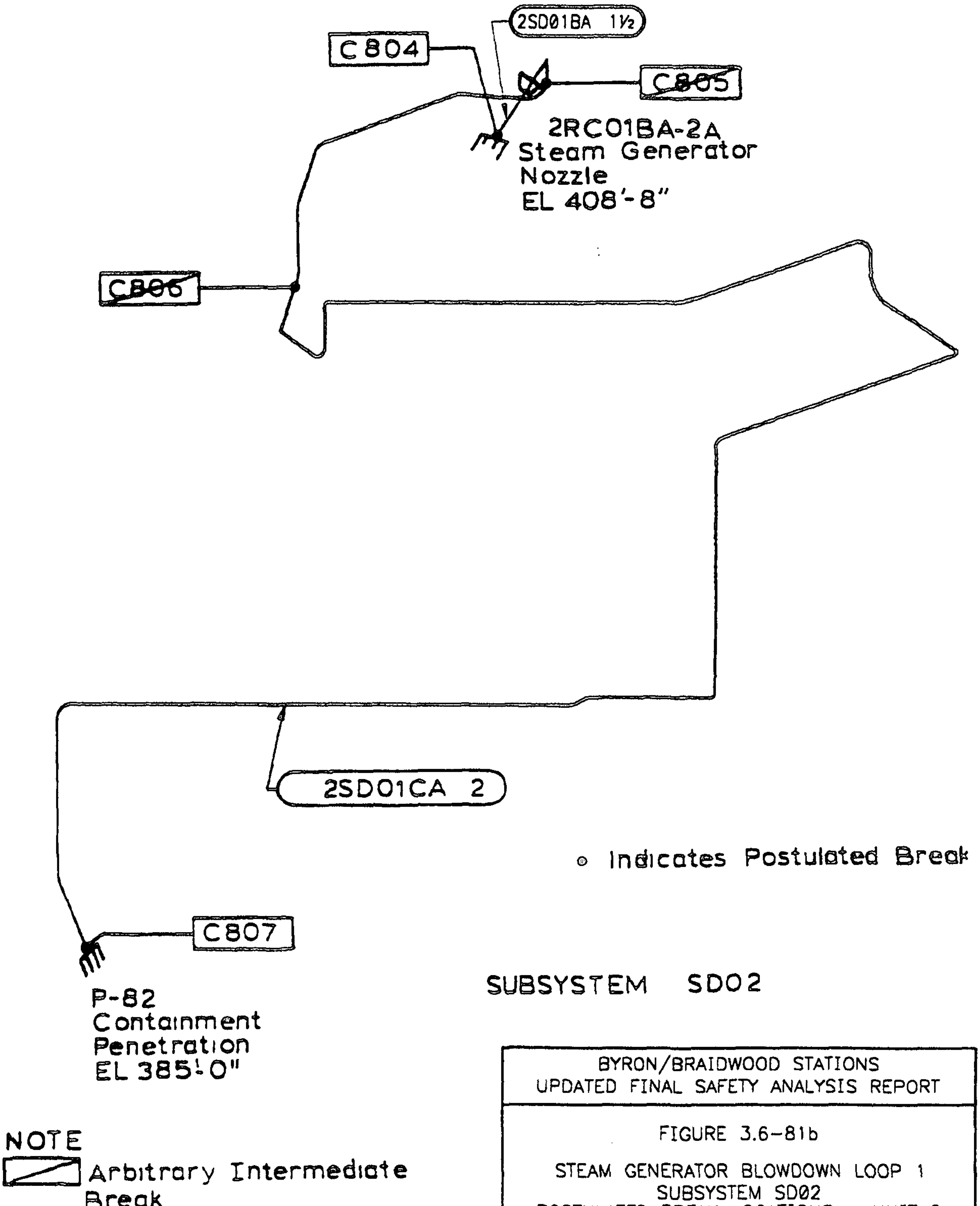

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-81b

STEAM GENERATOR BLOWDOWN LOOP 1
SUBSYSTEM SD02
POSTULATED BREAK LOCATIONS - UNIT 2

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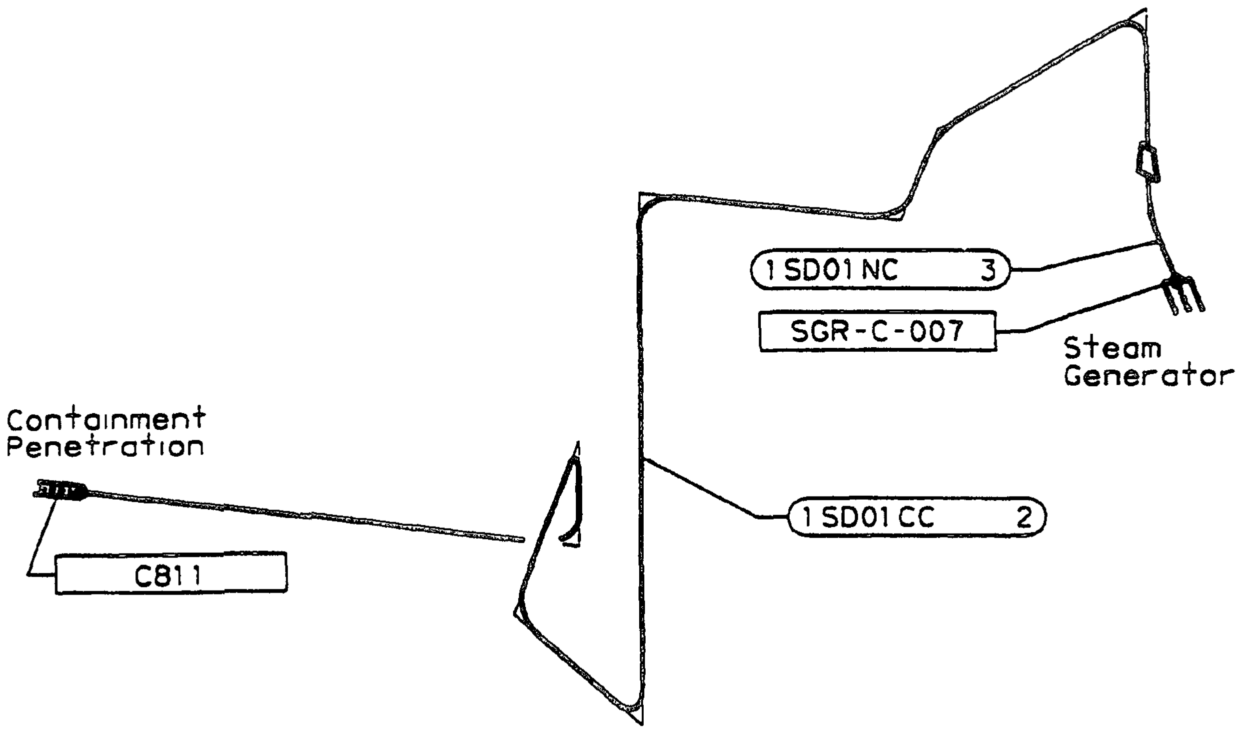

SUBSYSTEM 1SD03

● Indicates Postulated Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-82 STEAM GENERATOR BLOWDOWN LOOP 2 SUBSYSTEM 1SD03 POSTULATED BREAK LOCATIONS - UNIT 1

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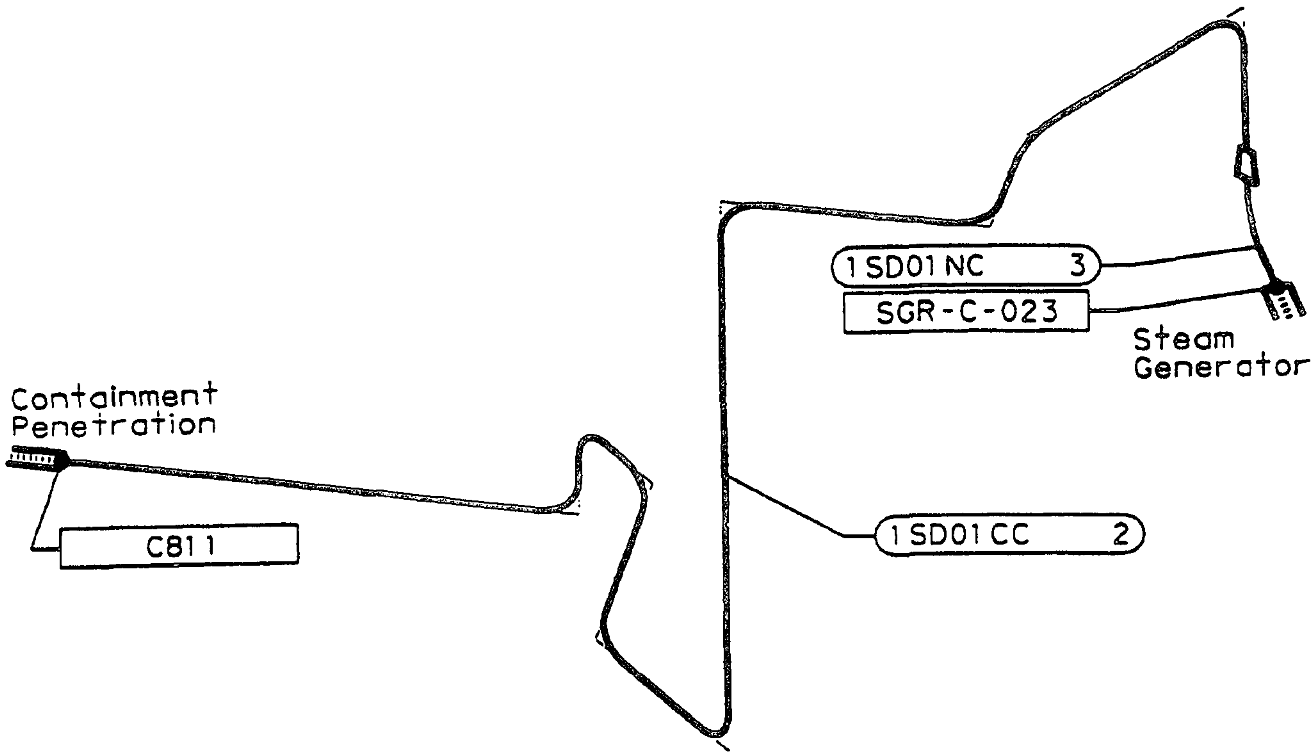

SUBSYSTEM 1SD03

● Indicates Postulated Break

BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-82a STEAM GENERATOR BLOWDOWN LOOP 2 SUBSYSTEM 1SD03 POSTULATED BREAK LOCATIONS - UNIT 1

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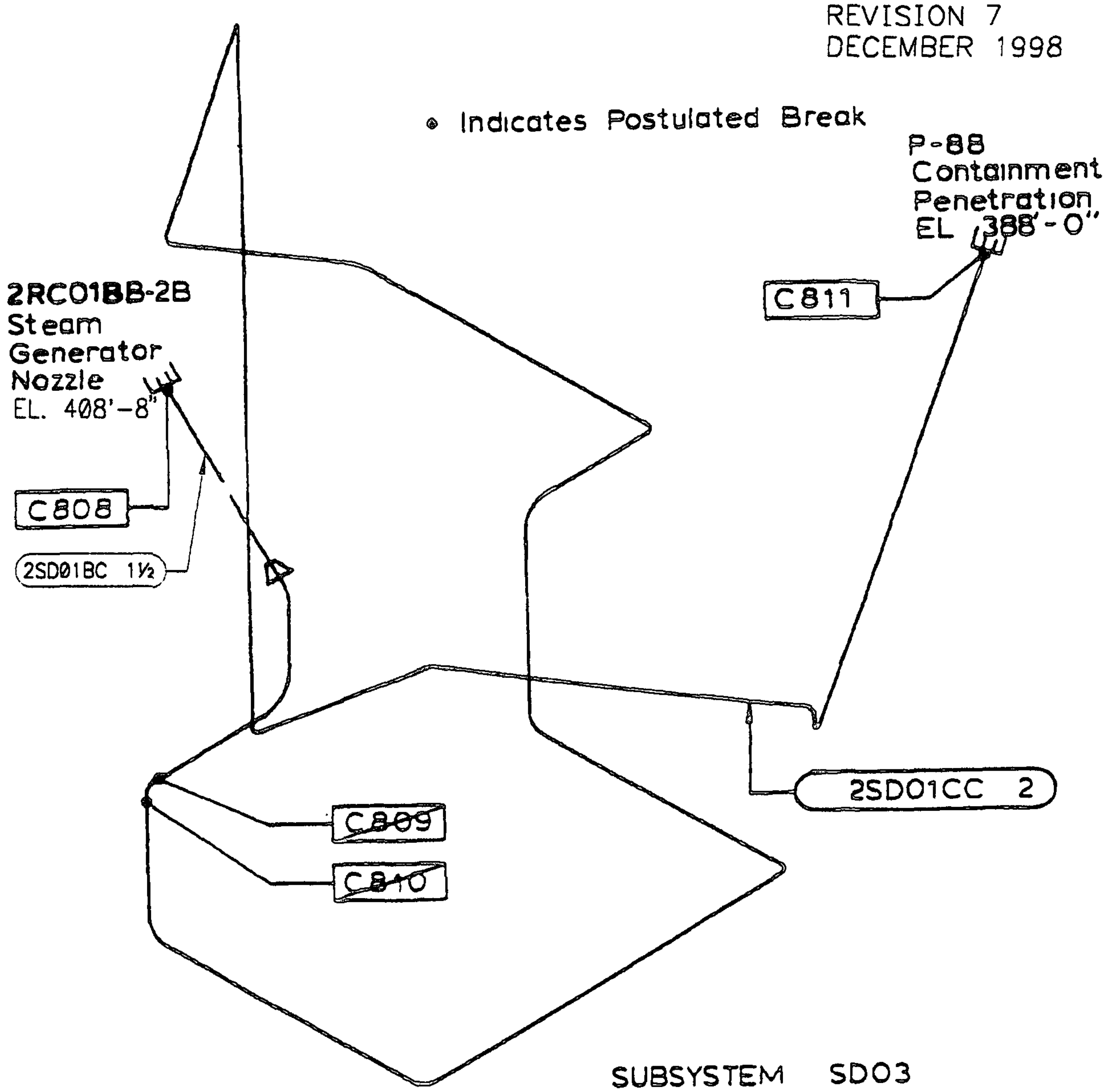

SUBSYSTEM SD03

NOTE:
Arbitrary Intermediate Break

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-82b

STEAM GENERATOR BLOWDOWN LOOP 2
SUBSYSTEM SD03
POSTULATED BREAK LOCATIONS (BYRON)-UNIT 2

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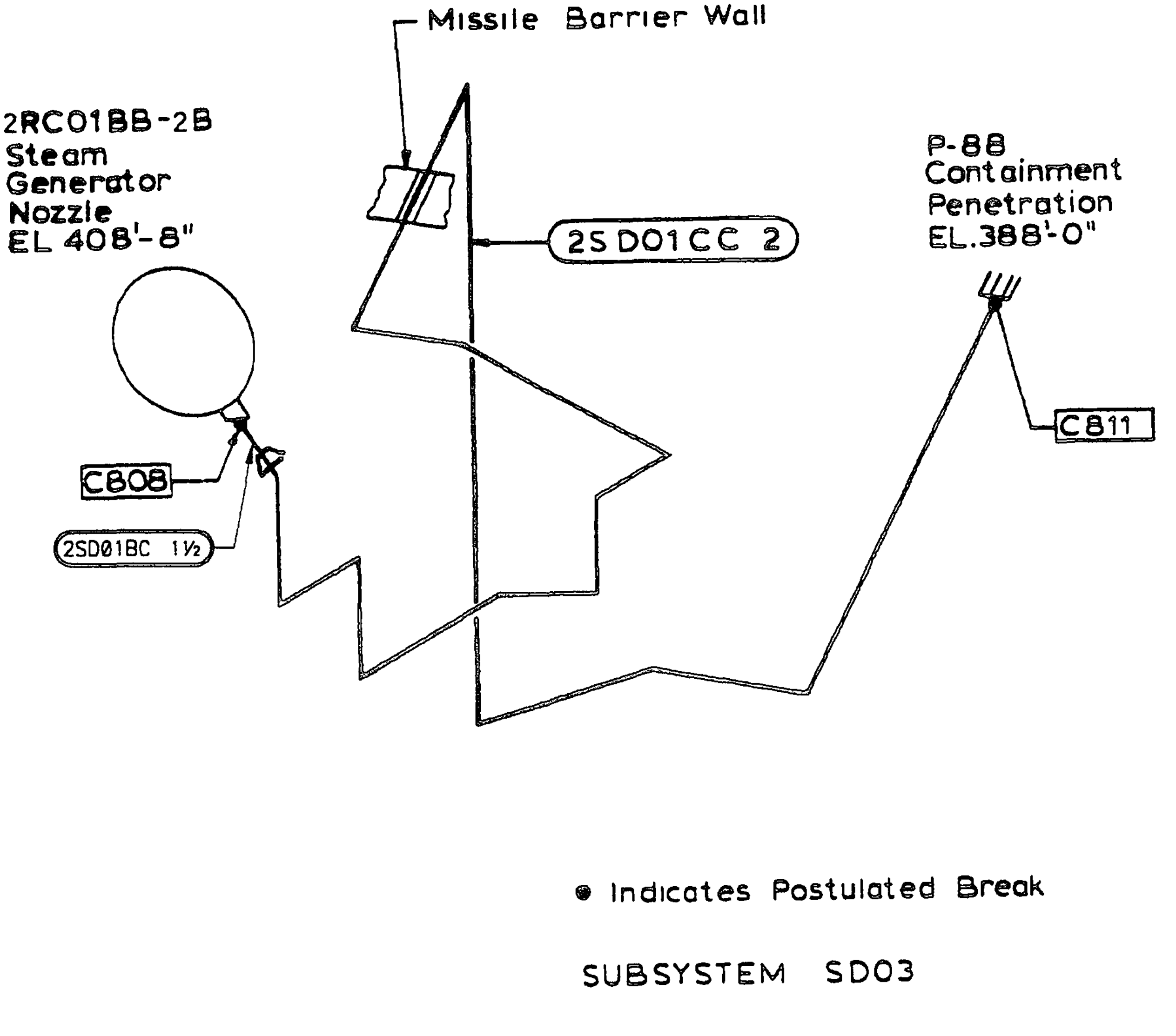

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-82c
STEAM GENERATOR BLOWDOWN LOOP 2
SUBSYSTEM SD03
POSTULATED BREAK LOCATIONS (BRAIDWOOD)
UNIT 2

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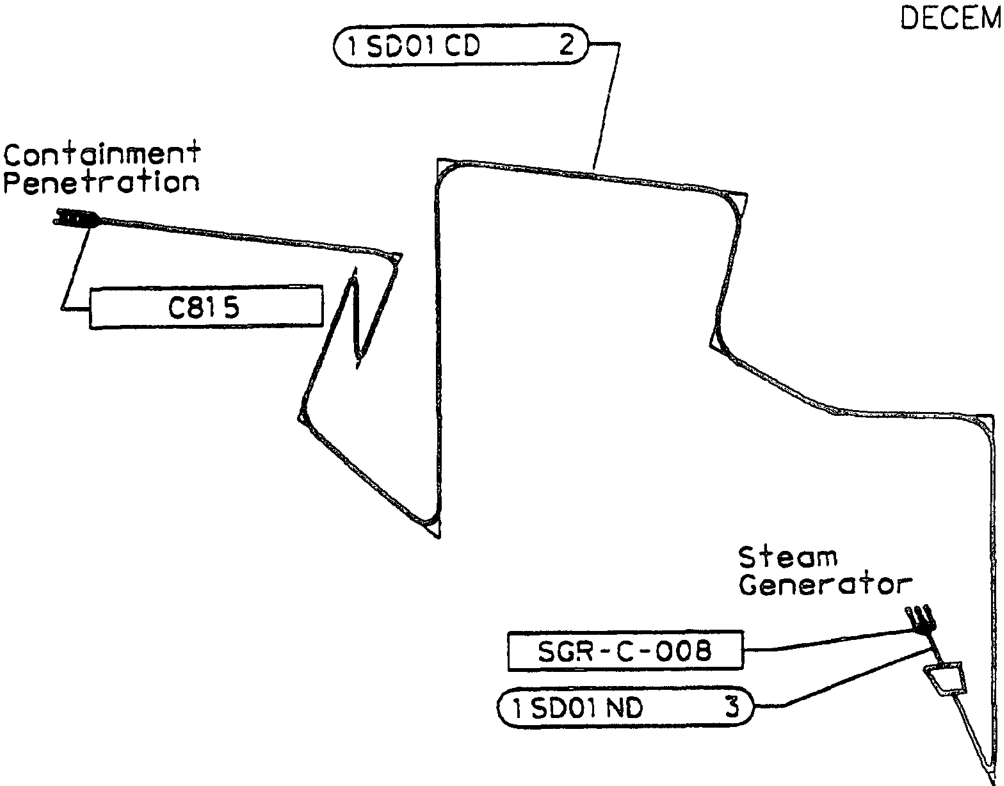
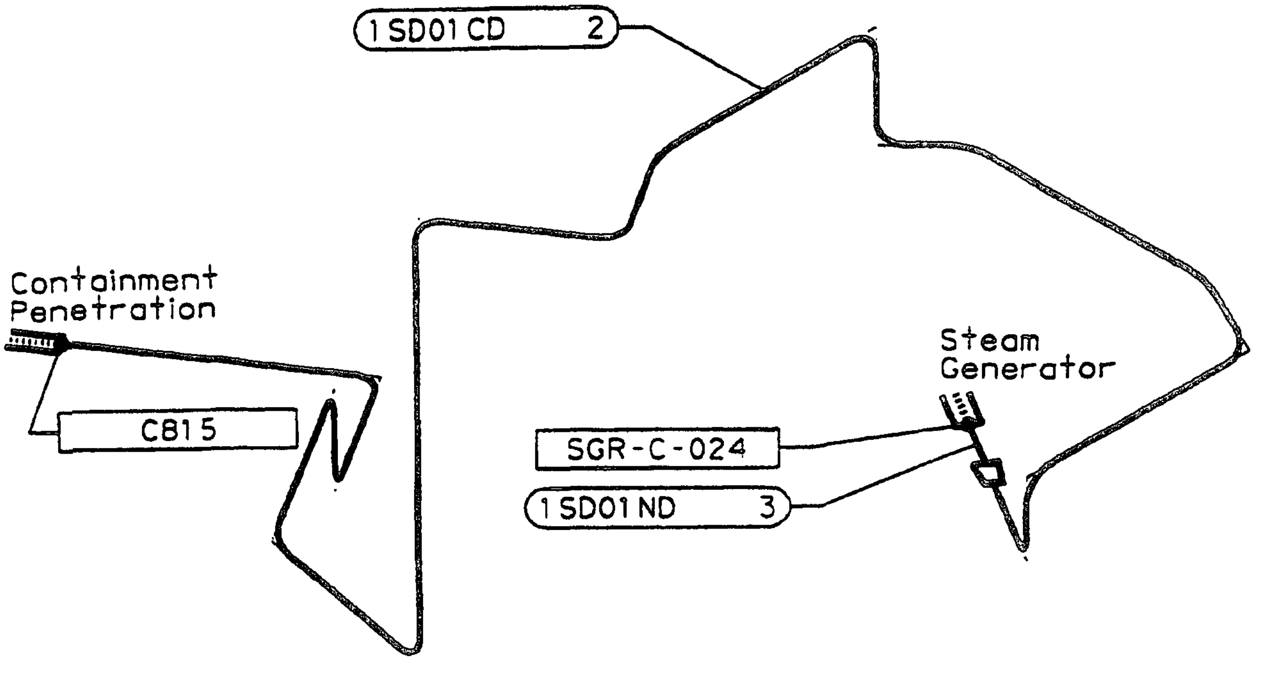


SUBSYSTEM 1 SD04

● Indicates Postulated Break

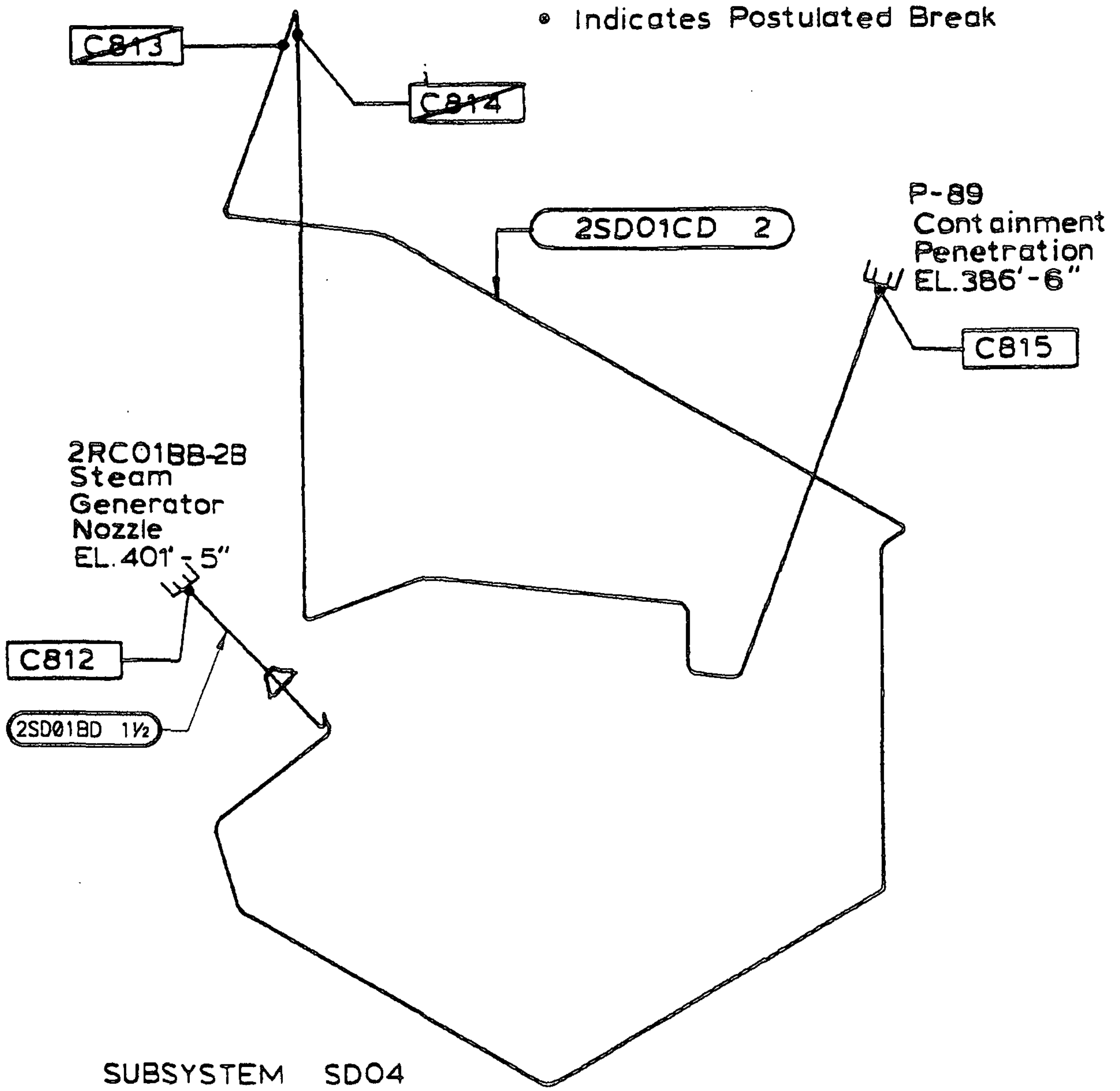
BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-83 STEAM GENERATOR BLOWDOWN LOOP 2 SUBSYSTEM 1SD04 POSTULATED BREAK LOCATIONS - UNIT 1



SUBSYSTEM 1SD04

● Indicates Postulated Break

BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-83a STEAM GENERATOR BLOWDOWN LOOP 2 SUBSYSTEM 1SD04 POSTULATED BREAK LOCATIONS - UNIT 1



SUBSYSTEM SD04

NOTE:
ARBITRARY INTERMEDIATE BREAK

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-83b
STEAM GENERATOR BLOWDOWN LOOP 2
SUBSYSTEM SD04
POSTULATED BREAK LOCATIONS (BYRON)
UNIT 2

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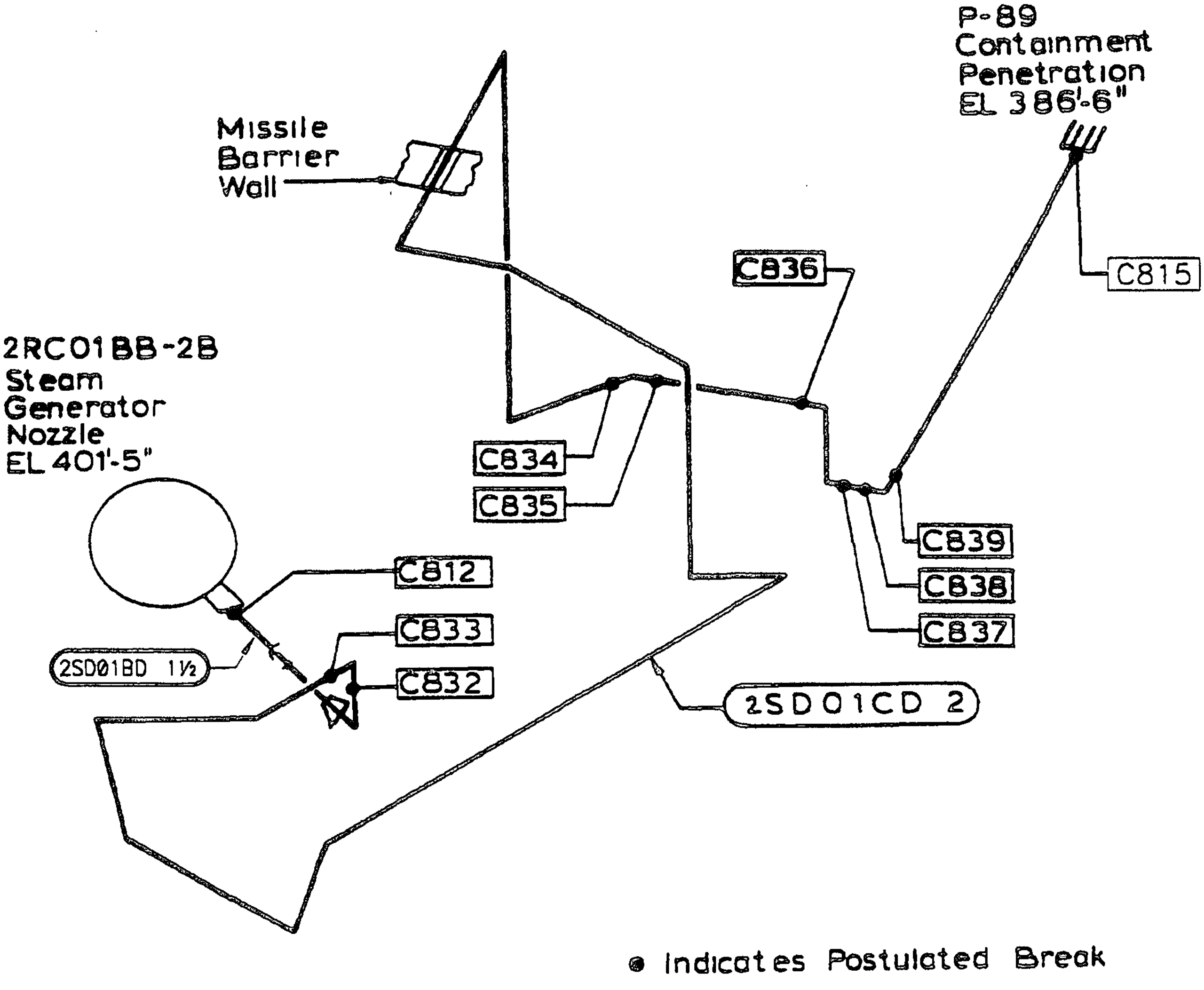

SUBSYSTEM SD04

BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-83c STEAM GENERATOR BLOWDOWN LOOP 2 SUBSYSTEM SD04 POSTULATED BREAK LOCATIONS (BRAIDWOOD) UNIT 2

REVISION 7
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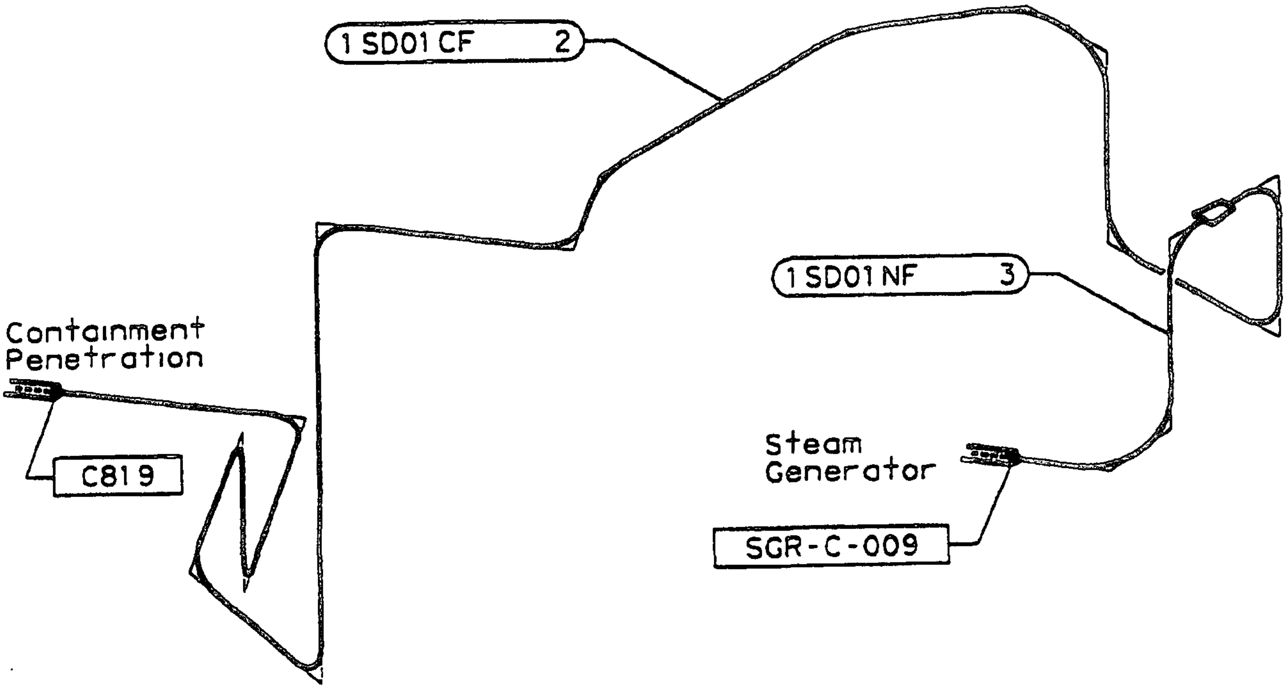

SUBSYSTEM 1SD05

⊛Indicates Postulated Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-84 STEAM GENERATOR BLOWDOWN LOOP 3 SUBSYSTEM 1SD05 POSTULATED BREAK LOCATIONS - UNIT 1

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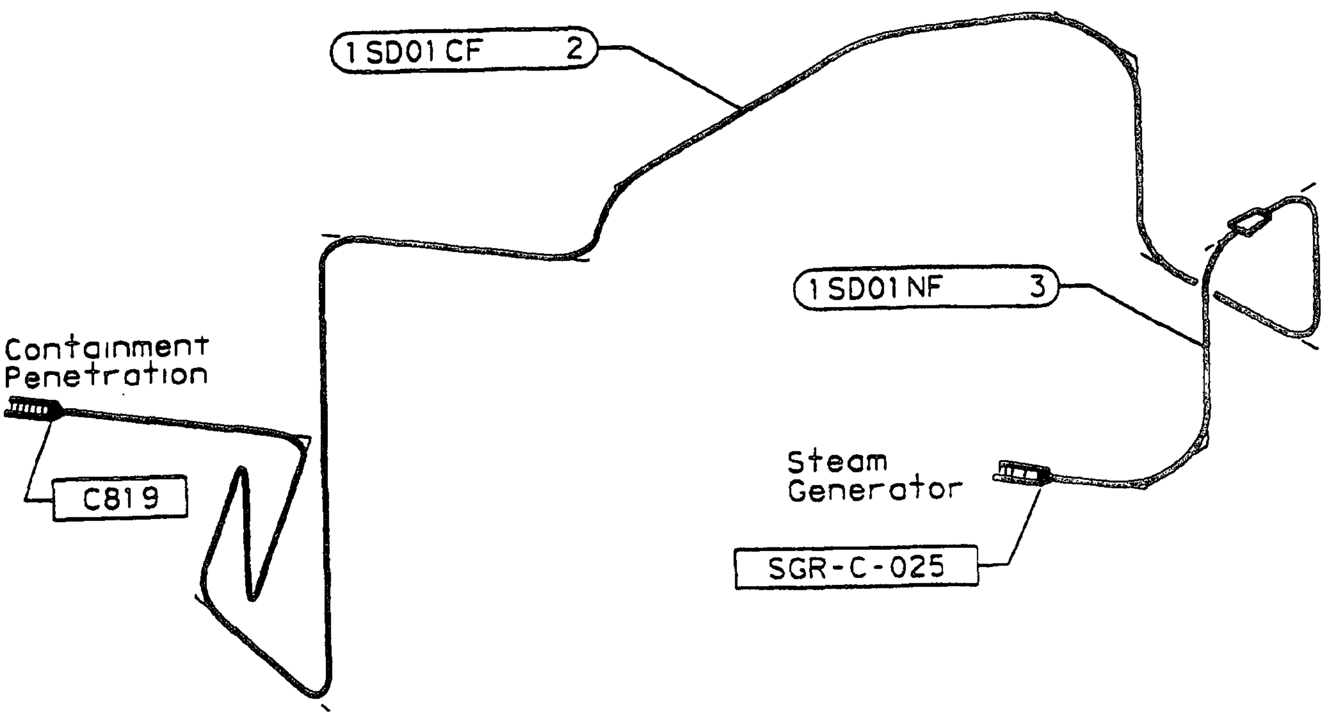
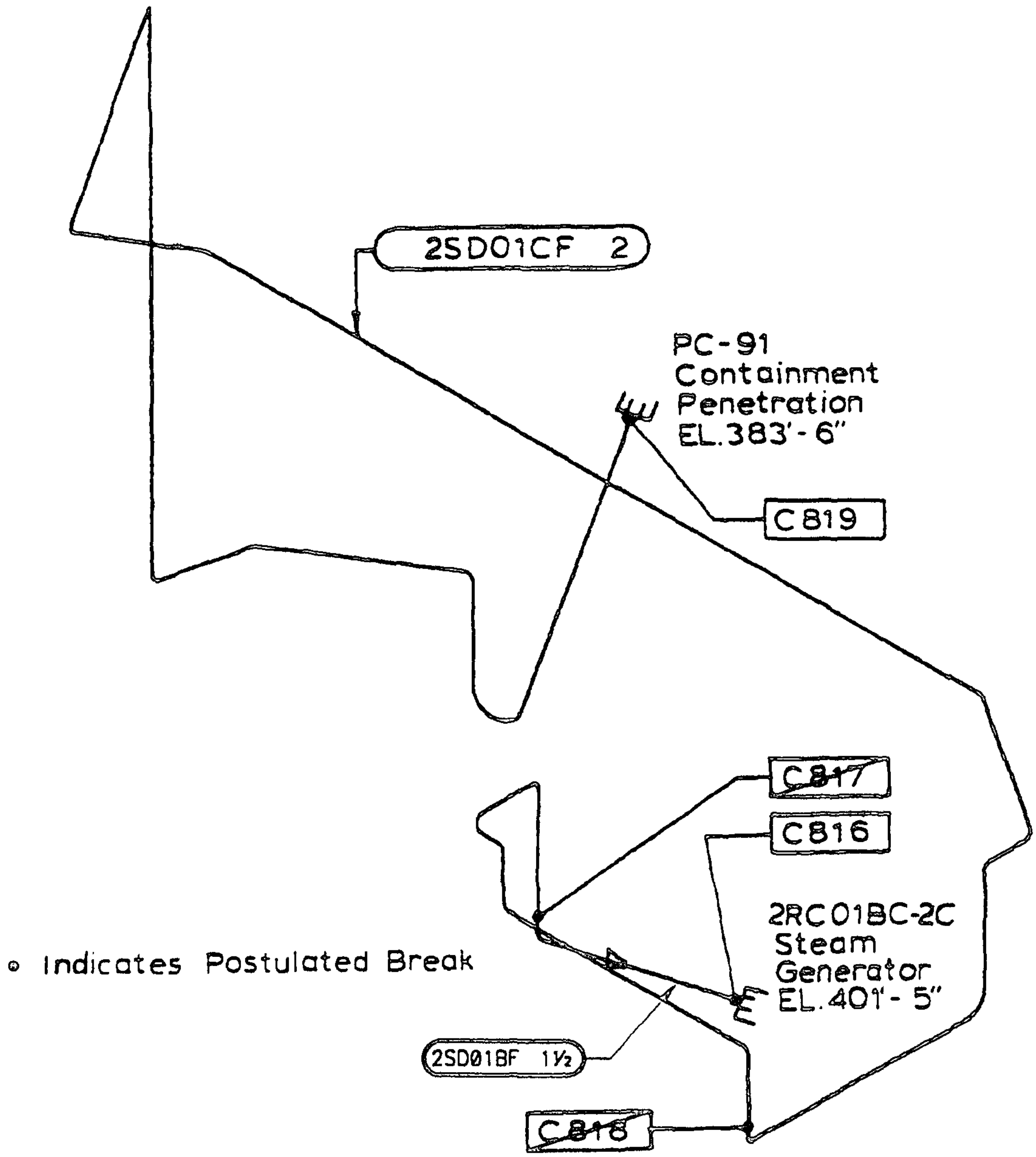


SUBSYSTEM 1SD05

⊛ Indicates Postulated Break

BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-84a STEAM GENERATOR BLOWDOWN LOOP 3 SUBSYSTEM 1SD05 POSTULATED BREAK LOCATIONS - UNIT 1



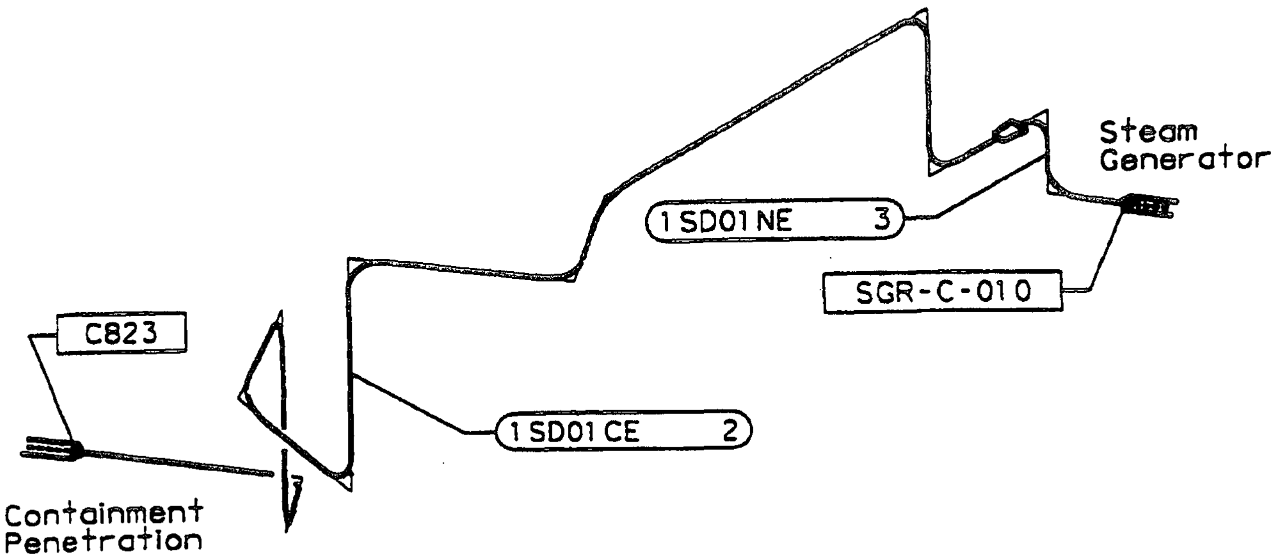
SUBSYSTEM SD05

Note:

Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-84b STEAM GENERATOR BLOWDOWN LOOP 3 SUBSYSTEM SD05 POSTULATED BREAK LOCATIONS - UNIT 2

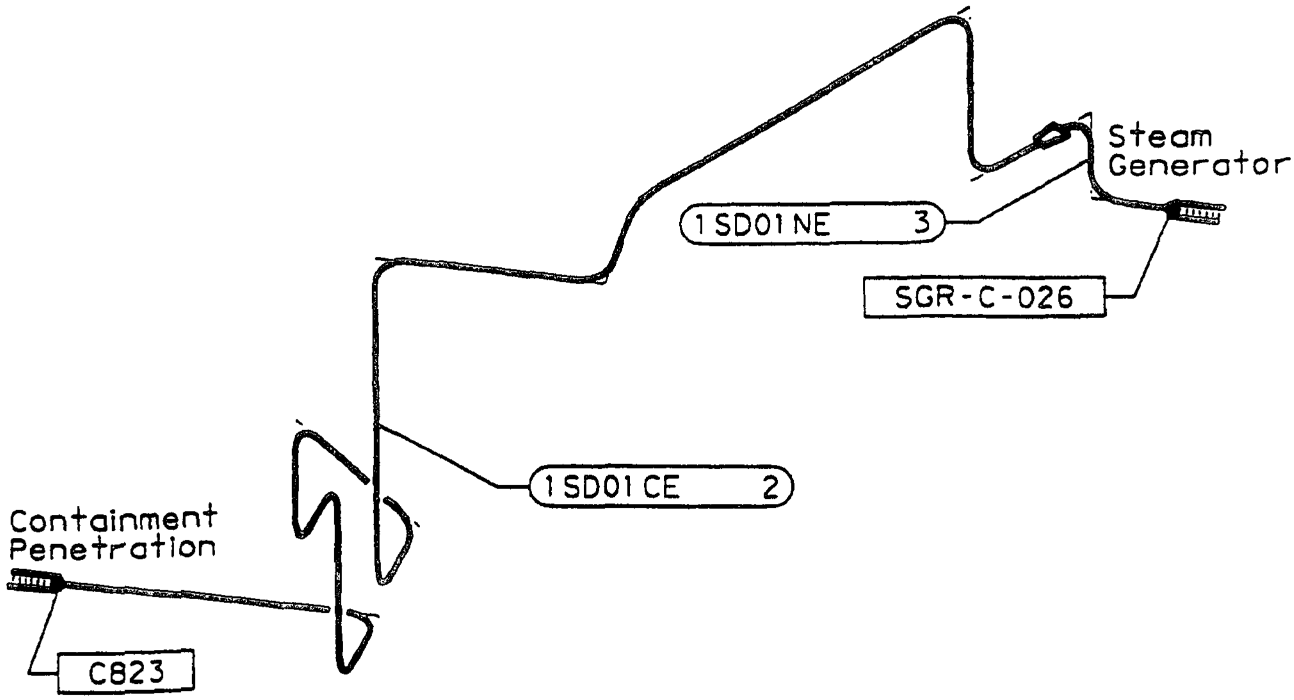
REVISION 7
DECEMBER 1998



SUBSYSTEM 1SD06

⊛ Indicates Postulated Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-85 STEAM GENERATOR BLOWDOWN LOOP 3 SUBSYSTEM 1SD06 POSTULATED BREAK LOCATIONS - UNIT 1

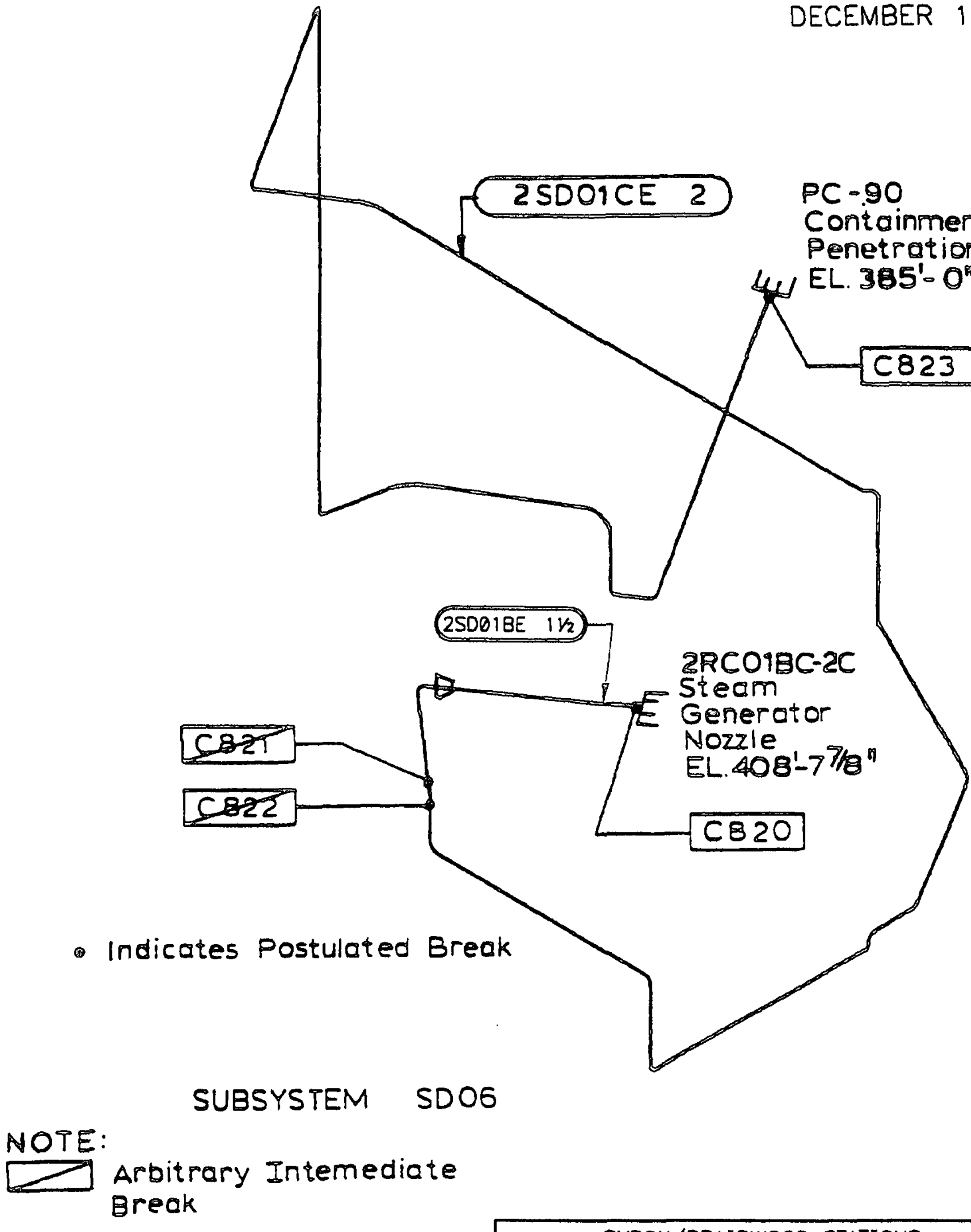


SUBSYSTEM 1SD06

● Indicates Postulated Break

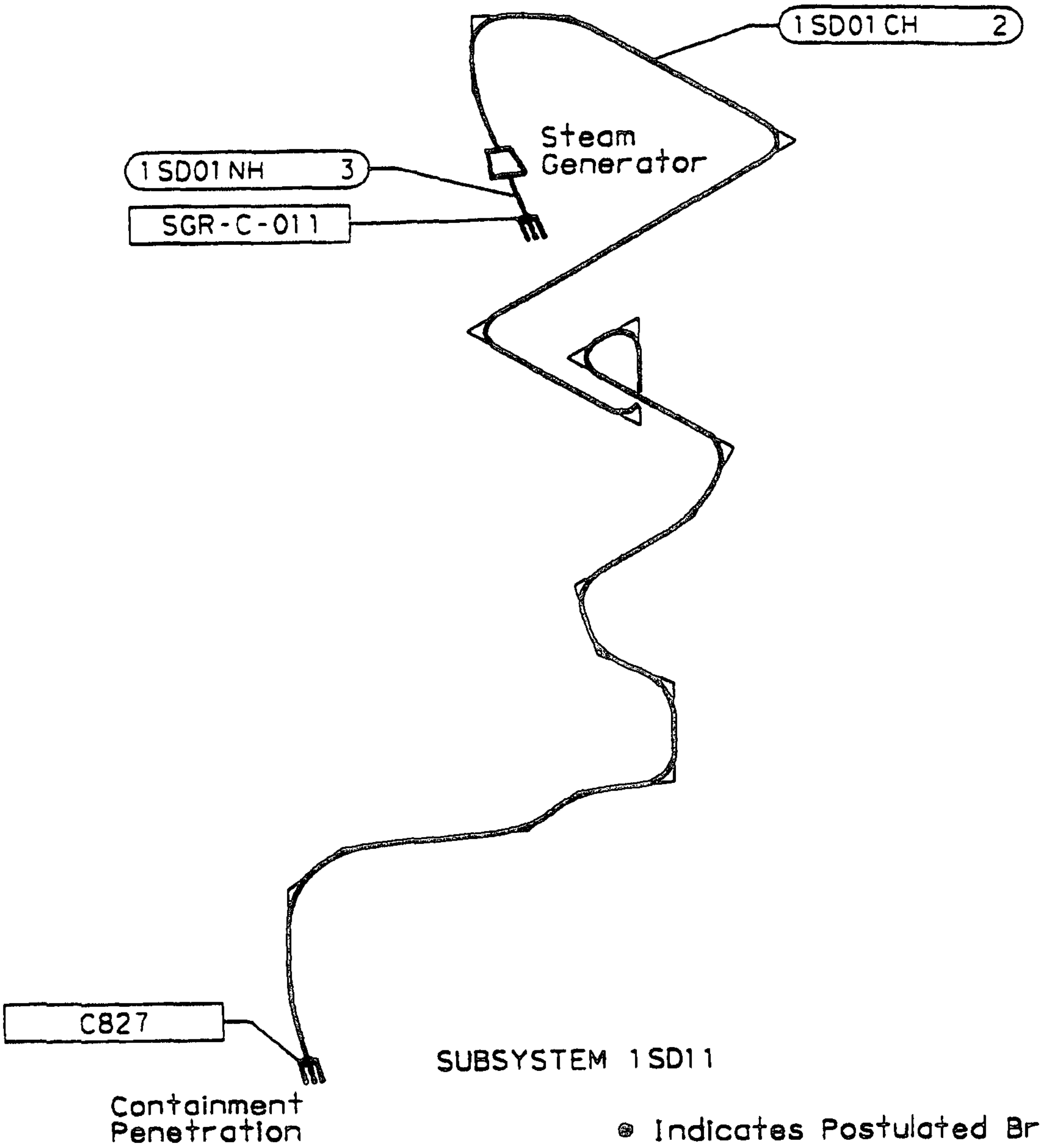
BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-85a STEAM GENERATOR BLOWDOWN LOOP 3 SUBSYSTEM 1SD06 POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998



BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-85b STEAM GENERATOR BLOWDOWN LOOP 3 SUBSYSTEM SD06 POSTULATED BREAK LOCATIONS - UNIT 2

REVISION 7
DECEMBER 1998



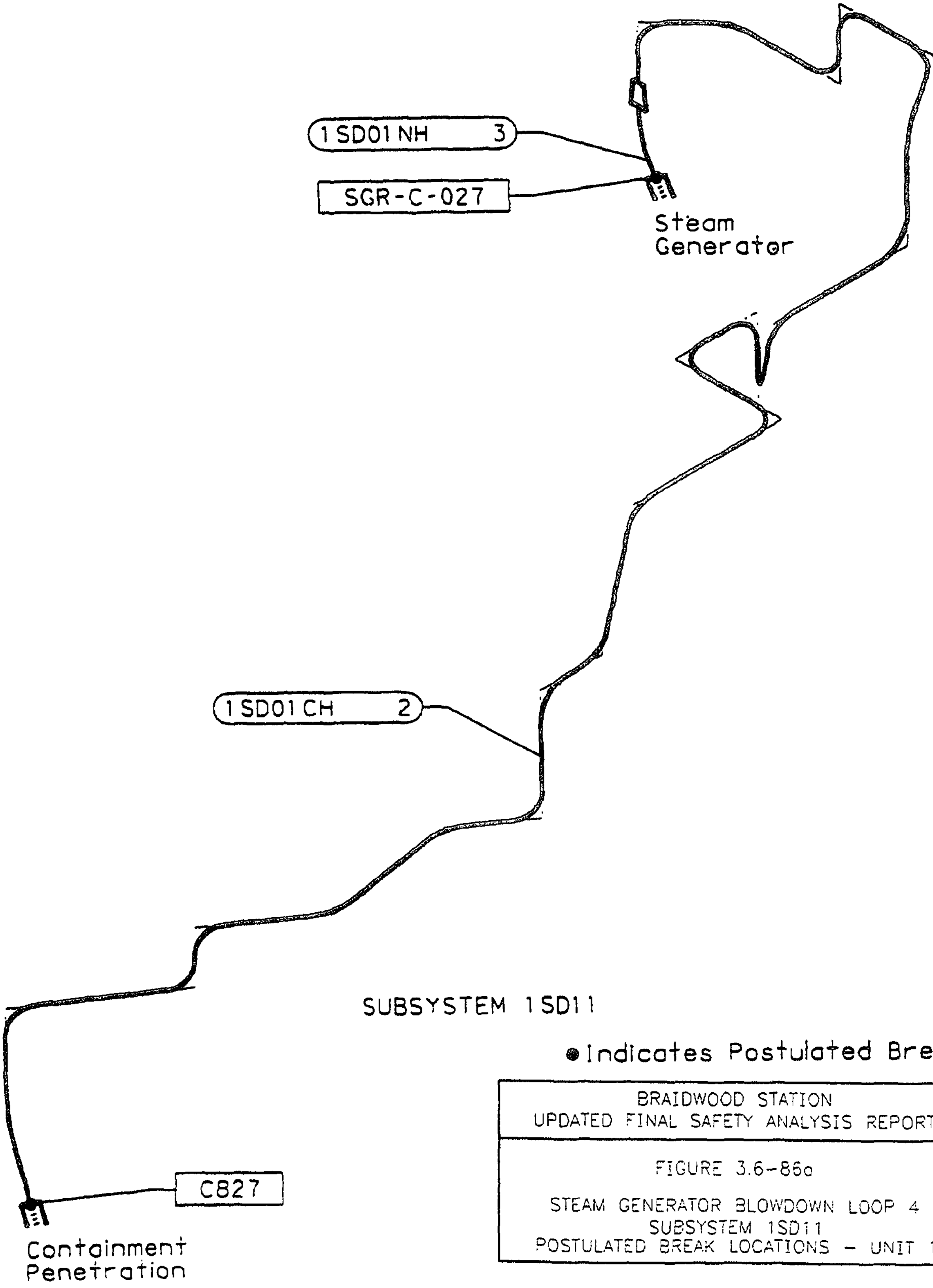
SUBSYSTEM 1SD11

⊛ Indicates Postulated Break

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-86

STEAM GENERATOR BLOWDOWN LOOP 4
SUBSYSTEM 1SD11
POSTULATED BREAK LOCATIONS – UNIT 1

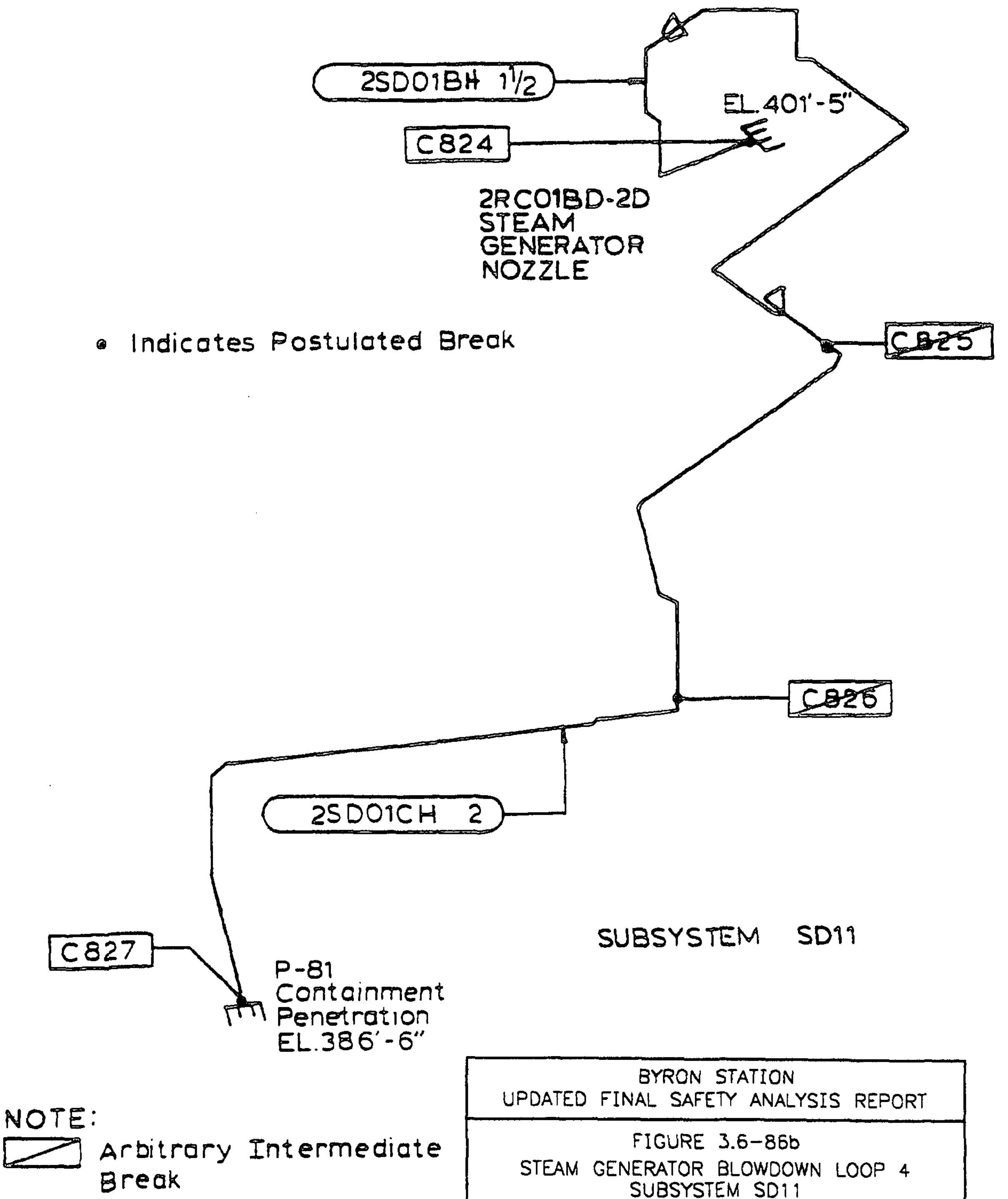


SUBSYSTEM 1SD11

● Indicates Postulated Break

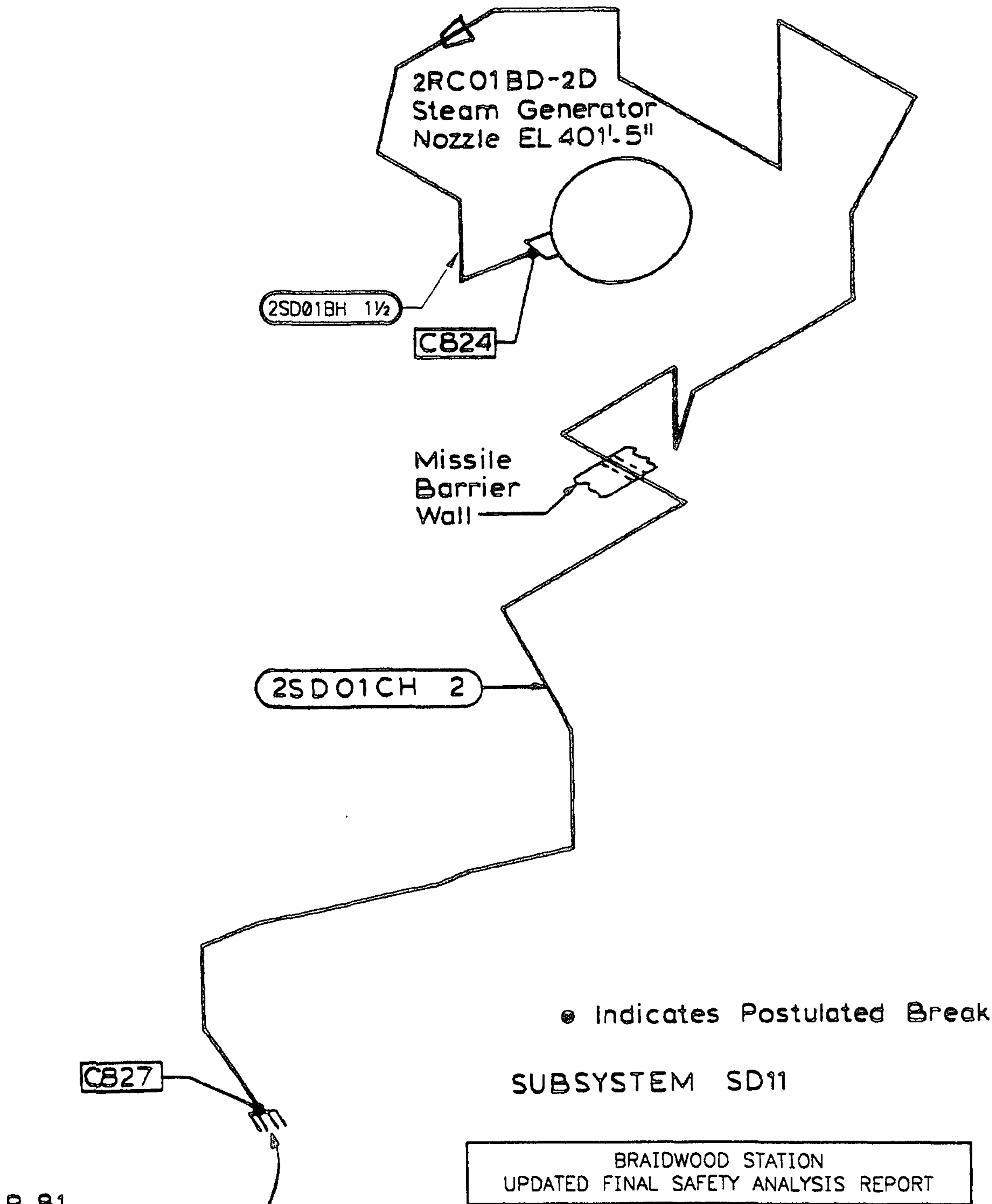
BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-86a STEAM GENERATOR BLOWDOWN LOOP 4 SUBSYSTEM 1SD11 POSTULATED BREAK LOCATIONS - UNIT 1

REVISION 7
DECEMBER 1998



BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

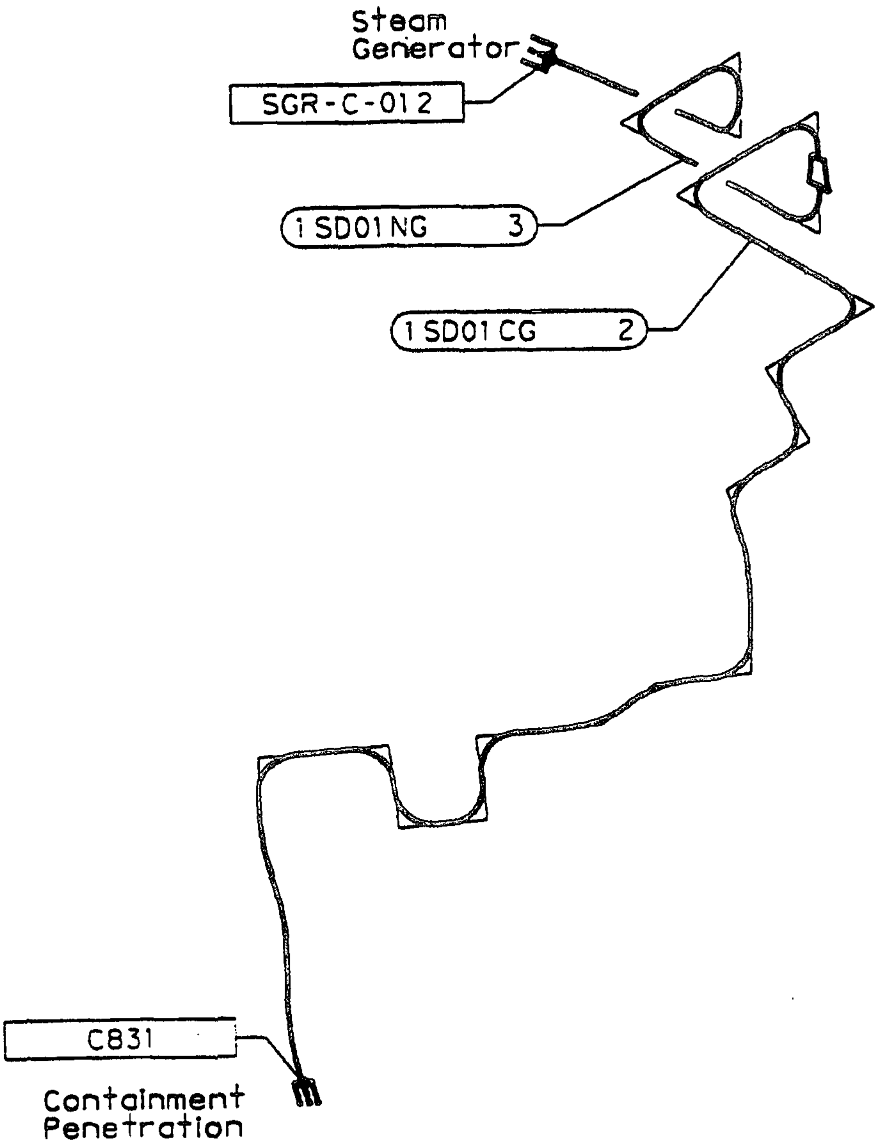
FIGURE 3.6-86b
STEAM GENERATOR BLOWDOWN LOOP 4
SUBSYSTEM SD11
POSTULATED BREAK LOCATIONS (BYRON)
UNIT 2



SUBSYSTEM SD11

BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-86c STEAM GENERATOR BLOWDOWN LOOP 4 SUBSYSTEM SD11 *POSTULATED BREAK LOCATIONS (BRAIDWOOD)* UNIT 2

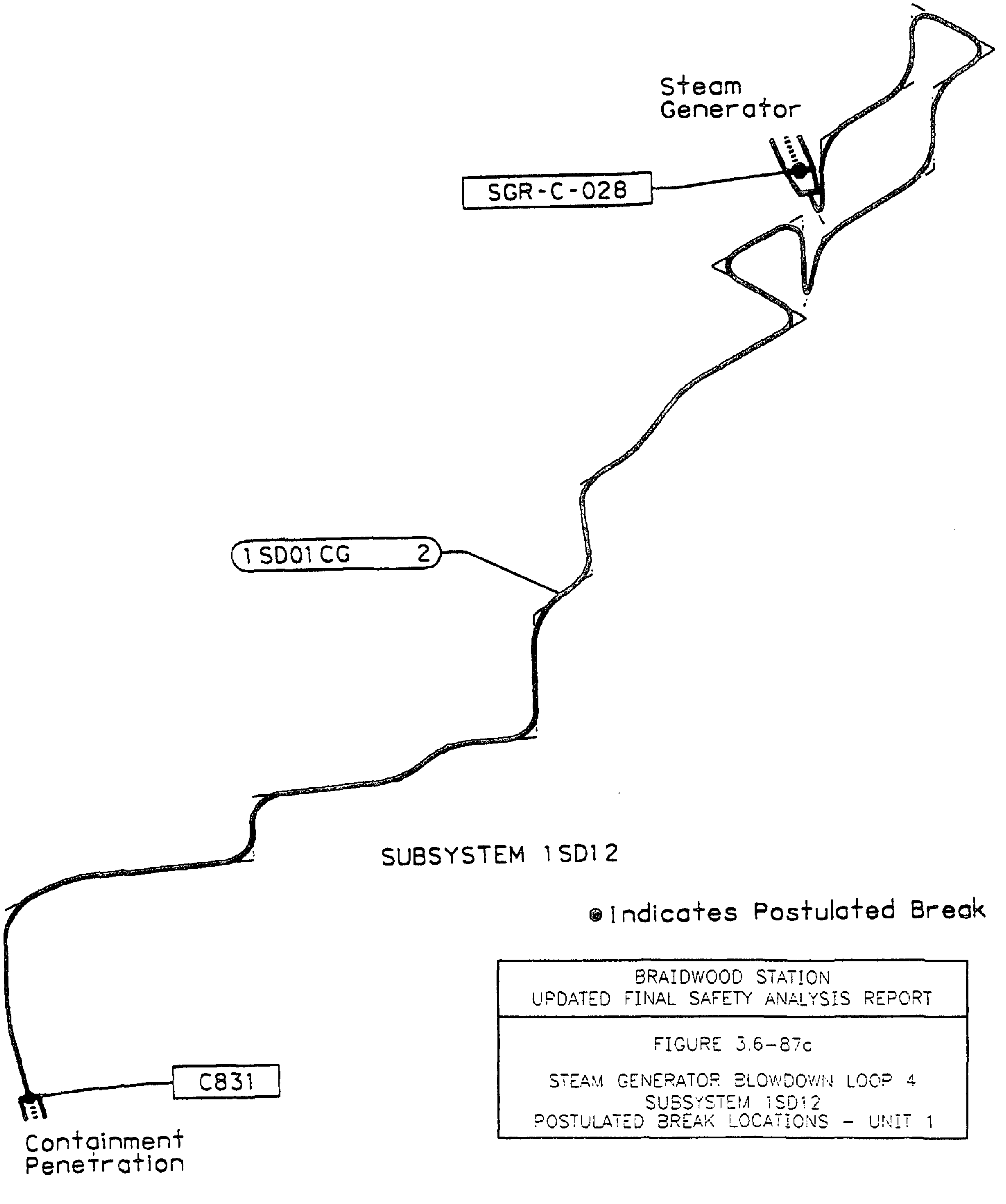
REVISION 7
DECEMBER 1998



SUBSYSTEM 1SD12

● Indicates Postulated Break

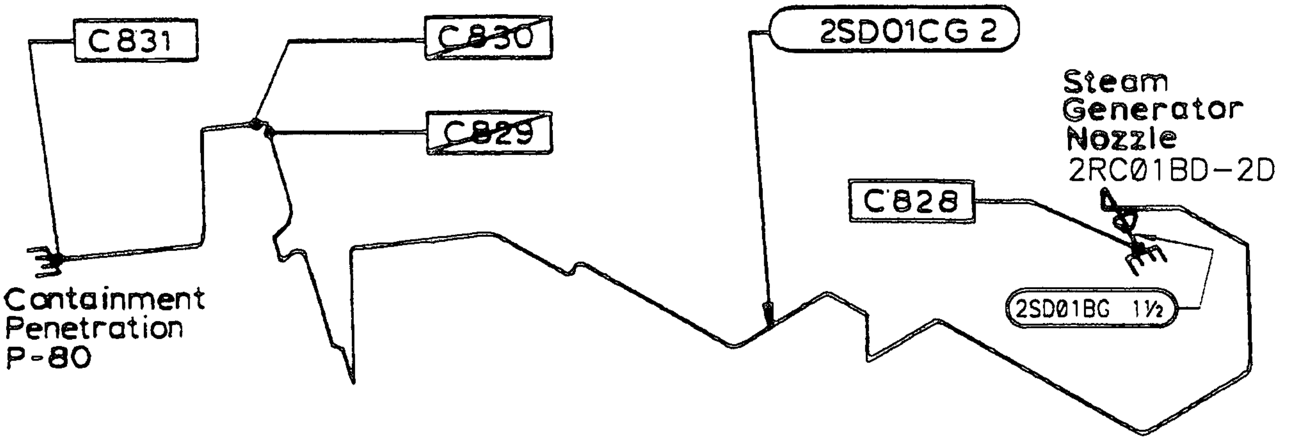
BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-87 STEAM GENERATOR BLOWDOWN LOOP 4 SUBSYSTEM 1SD12 POSTULATED BREAK LOCATIONS - UNIT 1



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-87c

STEAM GENERATOR BLOWDOWN LOOP 4
SUBSYSTEM 1SD12
POSTULATED BREAK LOCATIONS - UNIT 1



⊙ Indicates Postulated Break

SUBSYSTEM SD12

NOTE:

Arbitrary Intermediate Break

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-87b STEAM GENERATOR BLOWDOWN LOOP 4 SUBSYSTEM SD12 POSTULATED BREAK LOCATIONS (BYRON) UNIT 2

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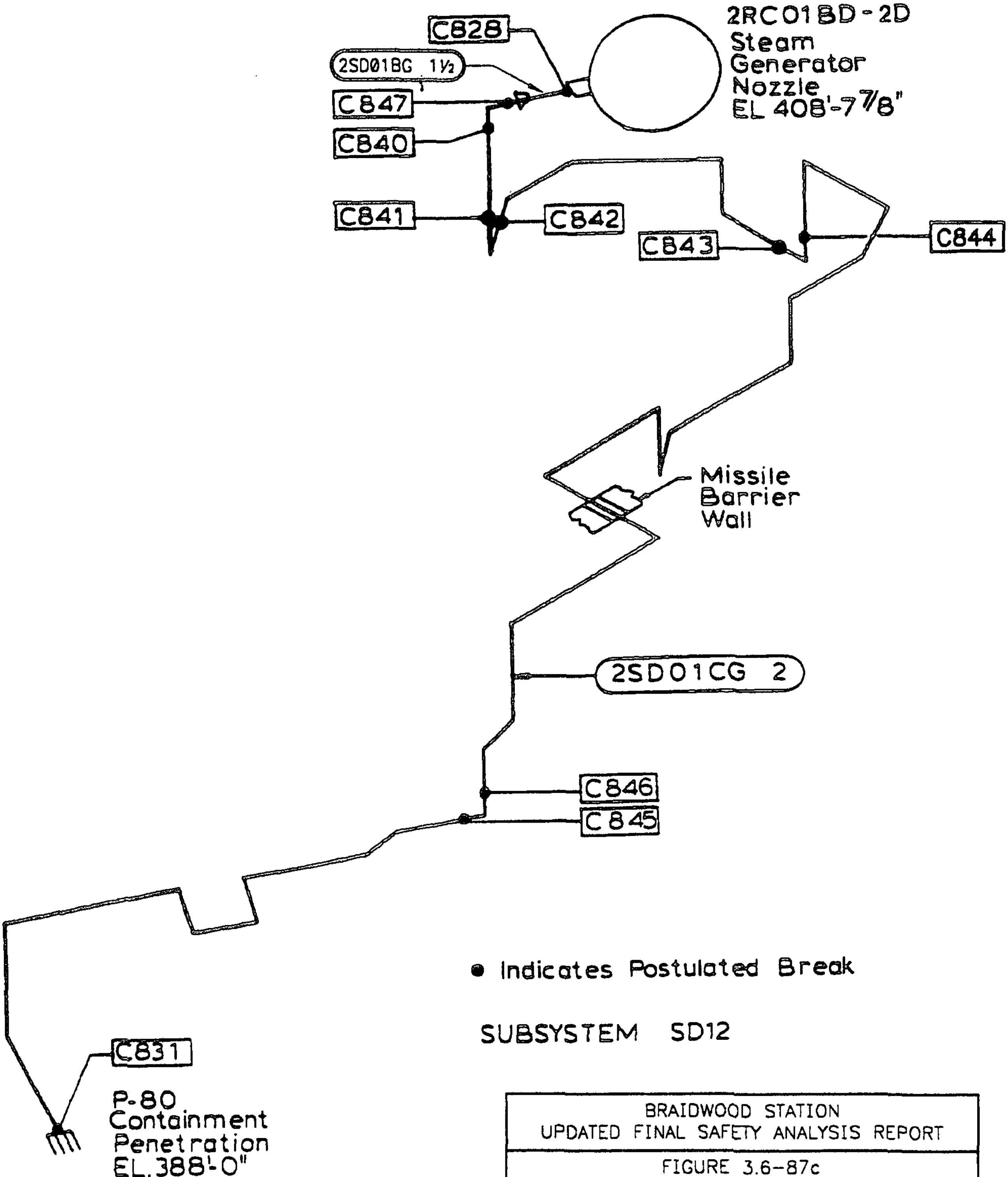
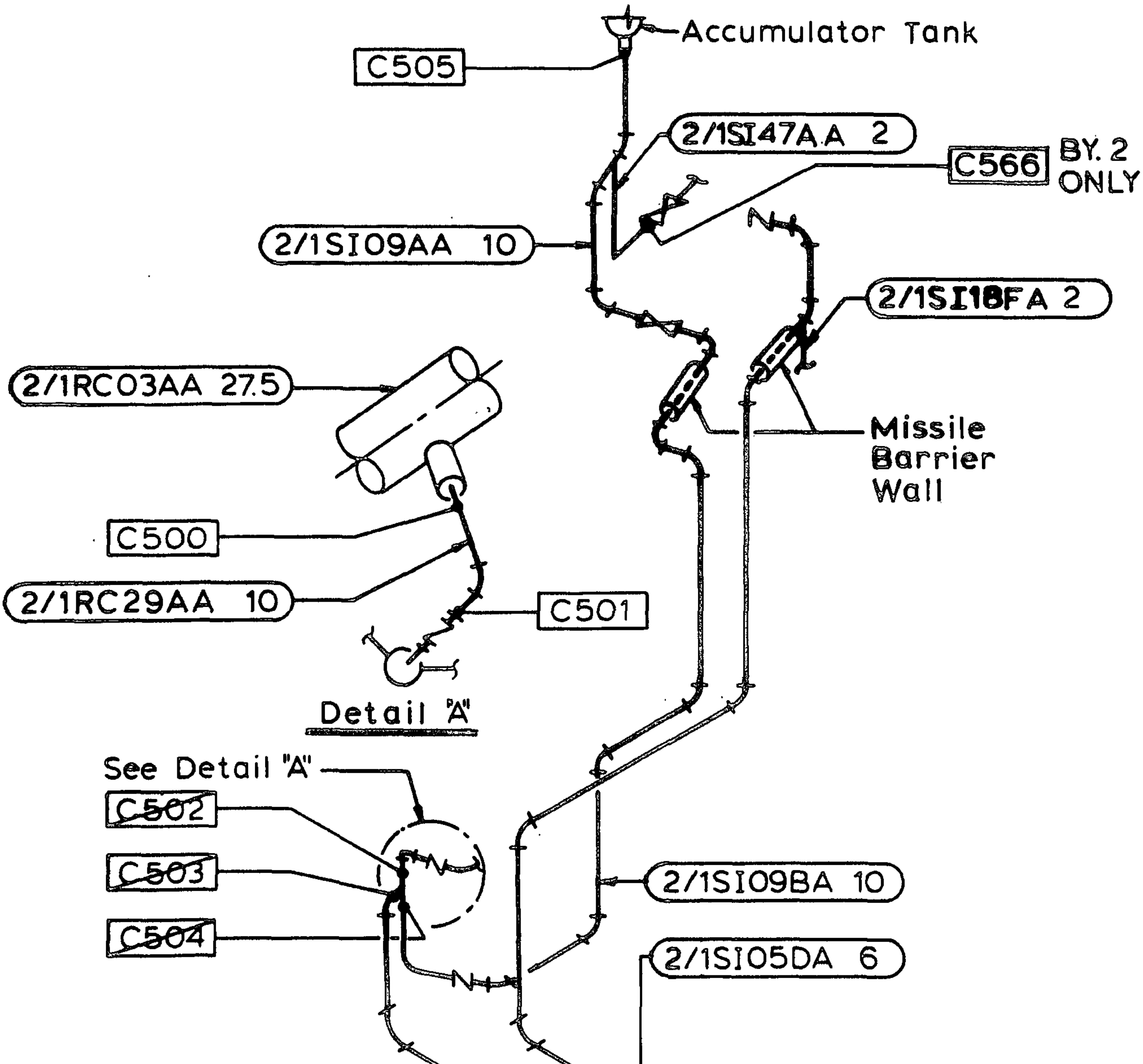
DECEMBER 1998



● Indicates Postulated Break

SUBSYSTEM SD12

BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-87c STEAM GENERATOR BLOWDOWN LOOP 4 SUBSYSTEM SD12 POSTULATED BREAK LOCATIONS (BRAIDWOOD) UNIT 2



• Indicates Postulated Break

SUBSYSTEM SI01

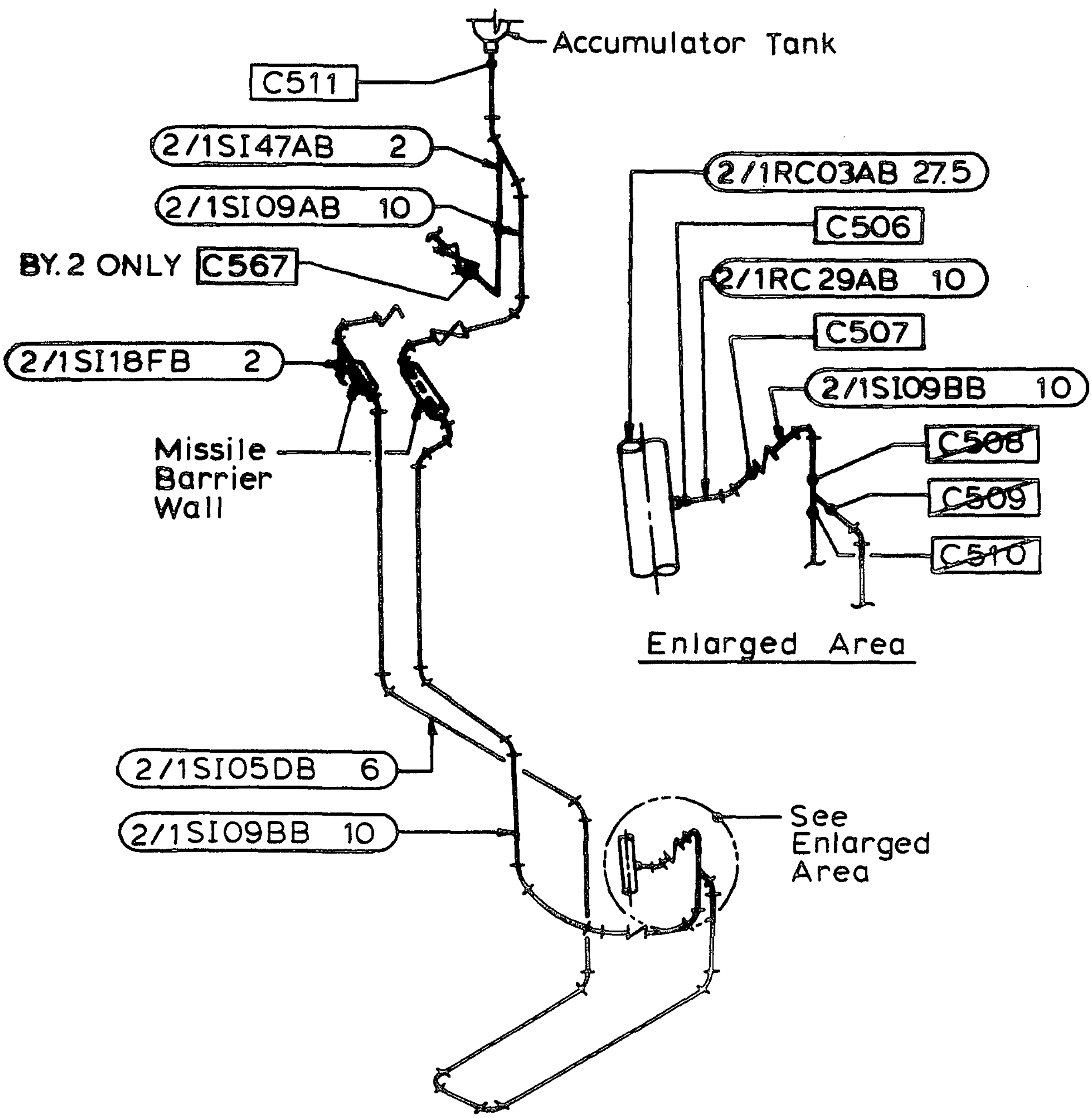
NOTE:

Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-88

SAFETY INJECTION LOOP 1
POSTULATED BREAK LOCATIONS



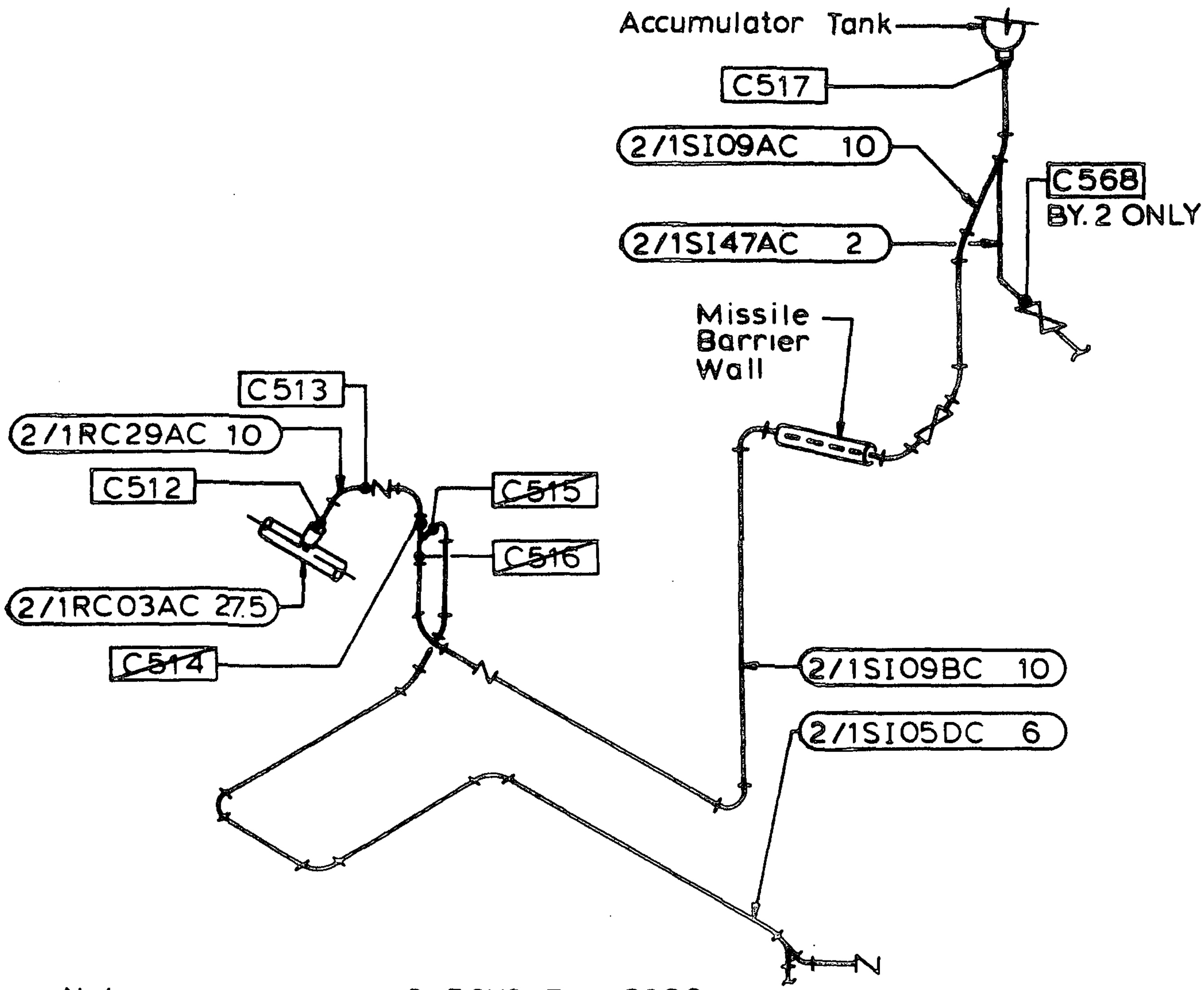
SUBSYSTEM 1SI04

Note:
Arbitrary Intermediate Break
• Indicates Postulated Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-89

SAFETY INJECTION LOOP 2
POSTULATED BREAK LOCATIONS



SUBSYSTEM SI09

Note:

Arbitrary Intermediate Break

® Indicates Postulated Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-90

SAFETY INJECTION LOOP 3
POSTULATED BREAK LOCATIONS

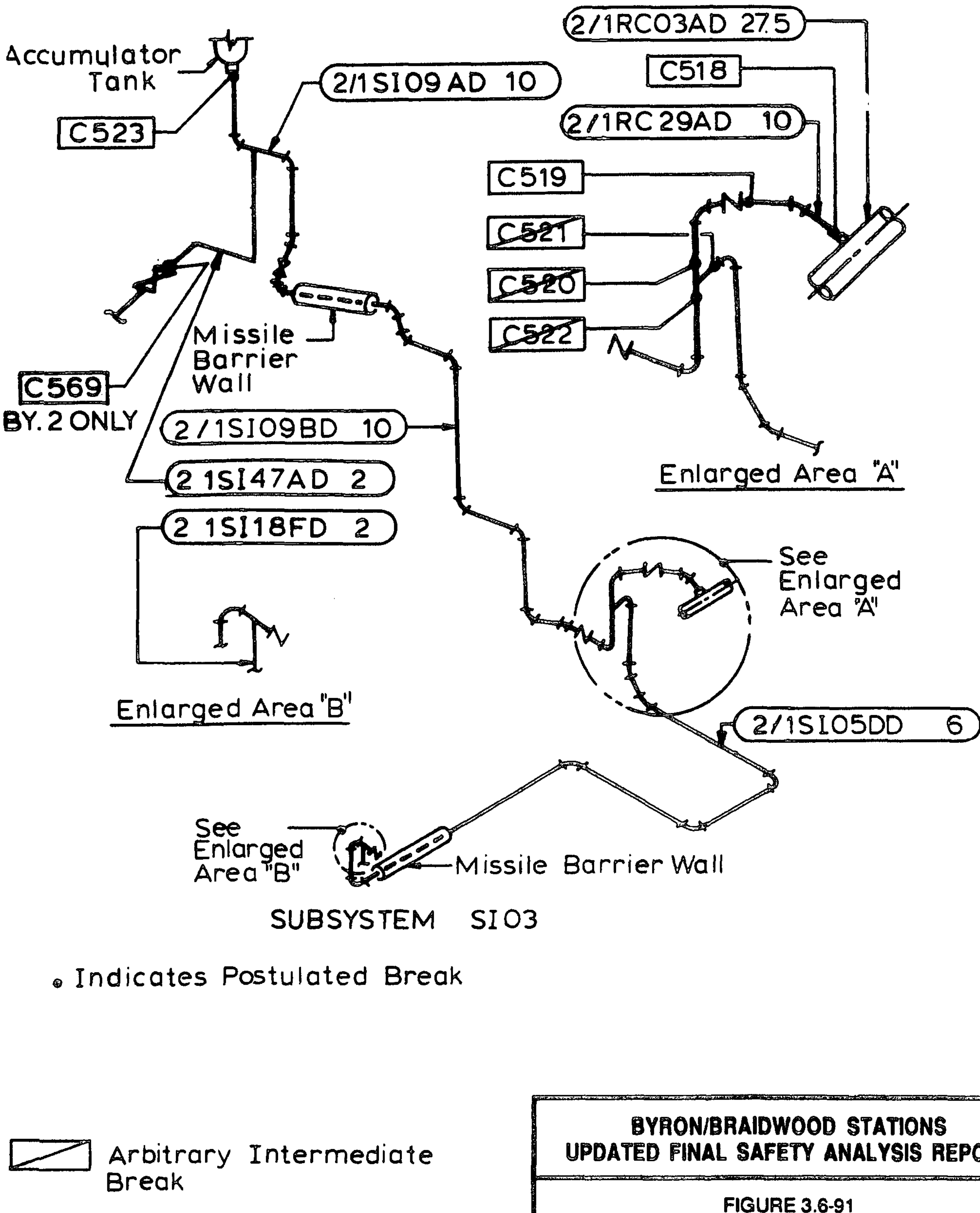


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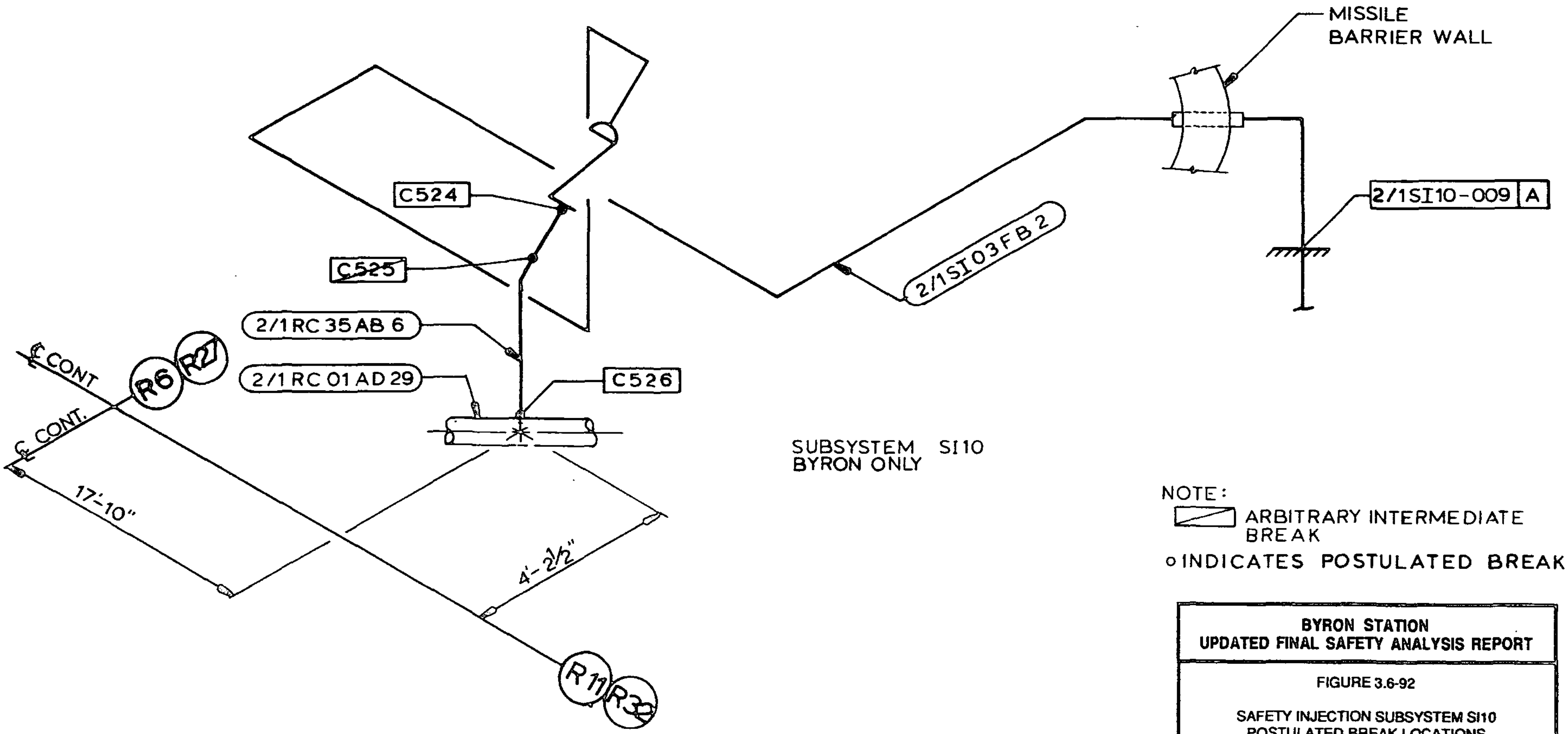
FIGURE 3.6-91 SAFETY INJECTION LOOP 4 POSTULATED BREAK LOCATIONS

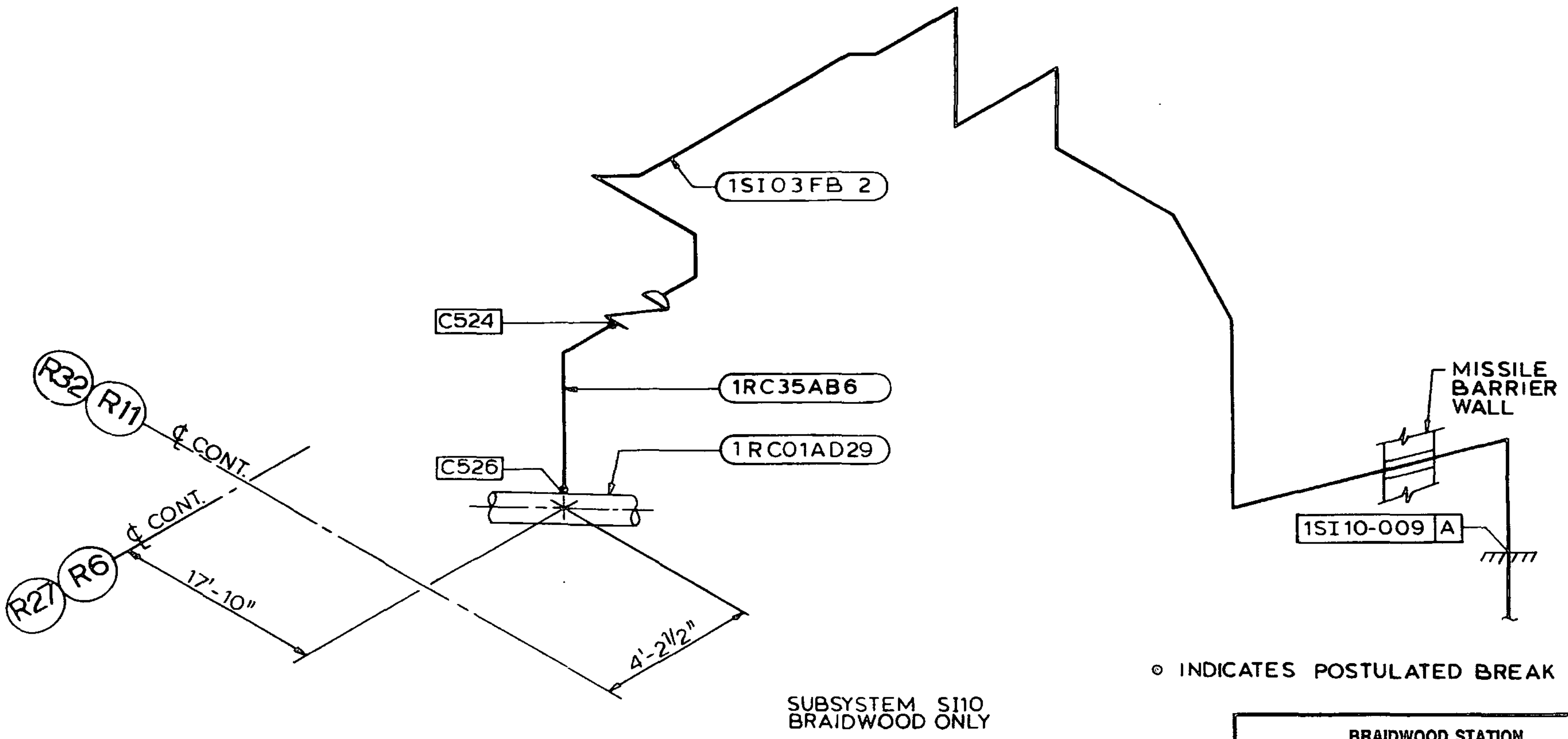


NOTE:

ARBITRARY INTERMEDIATE BREAK

o INDICATES POSTULATED BREAK

BYRON STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-92 SAFETY INJECTION SUBSYSTEM SI10 POSTULATED BREAK LOCATIONS (BYRON)



BRAIDWOOD STATION
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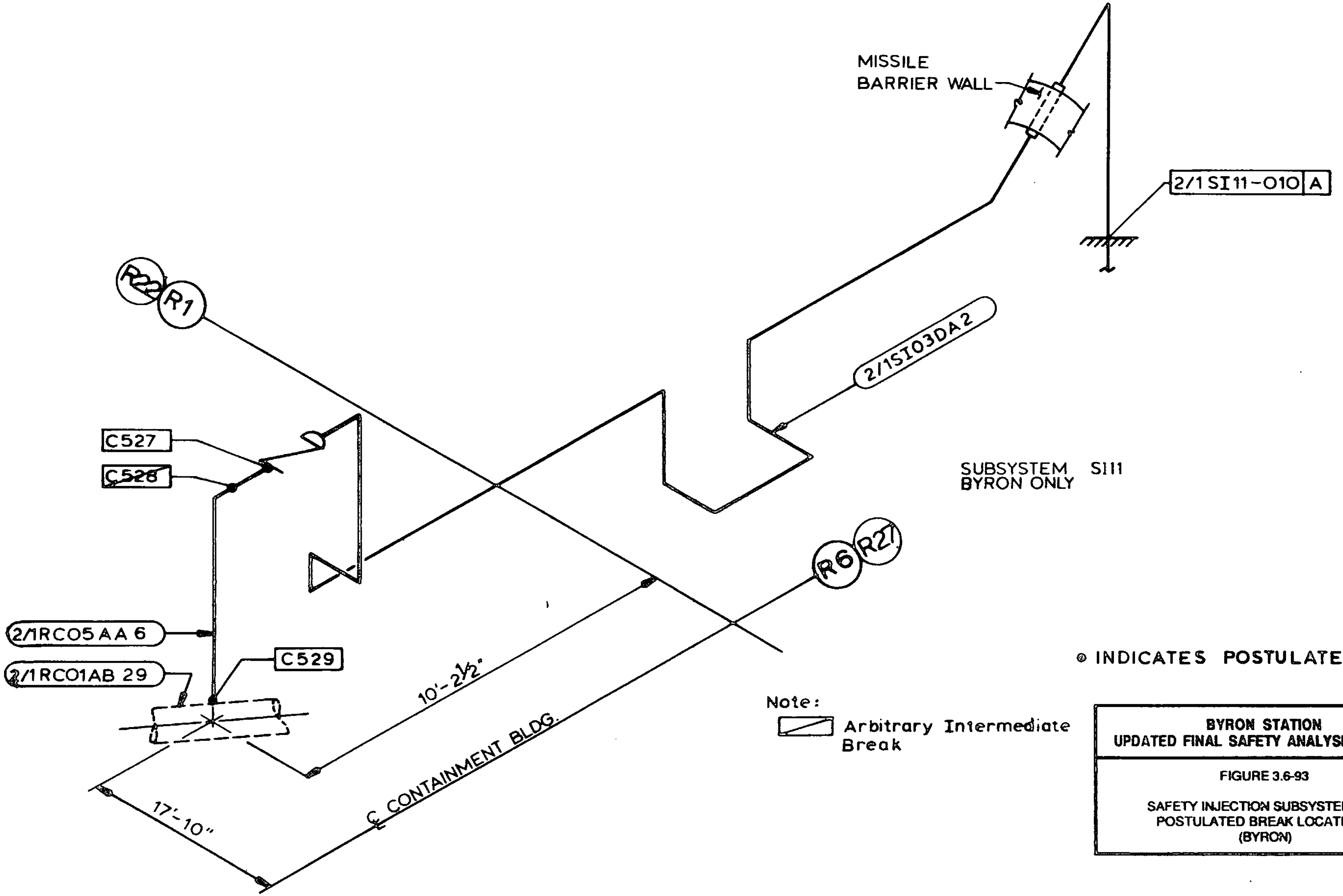
FIGURE 3.6-92a

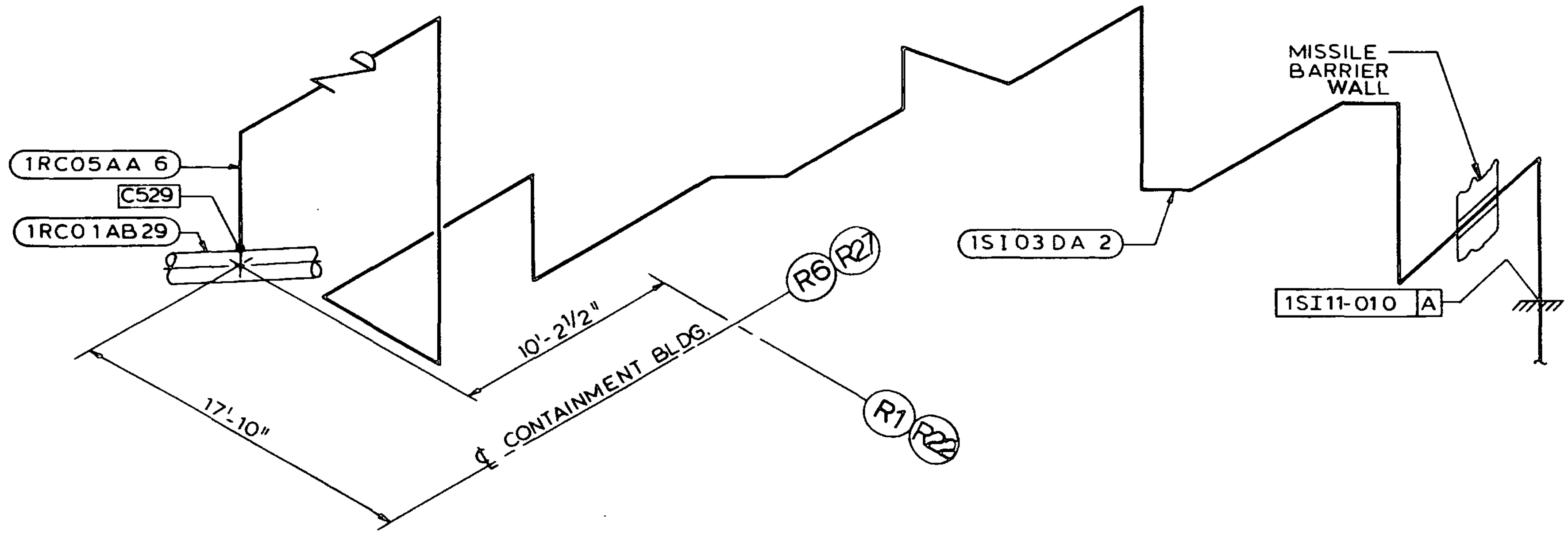
SAFETY INJECTION SUBSYSTEM SI10
POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



⊚ INDICATES POSTULATED BREAK

Note:
Arbitrary Intermediate Break

BYRON STATION **UPDATED FINAL SAFETY ANALYSIS REPORT**
FIGURE 3.6-93 SAFETY INJECTION SUBSYSTEM SI11 POSTULATED BREAK LOCATIONS (BYRON)



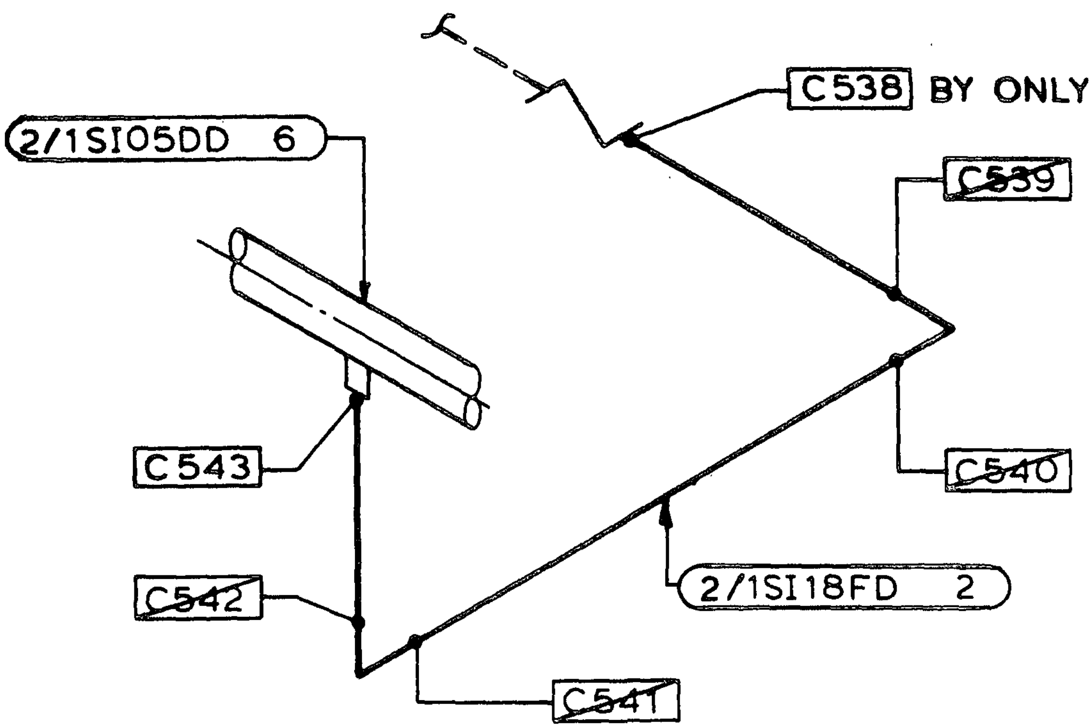
⊙ INDICATES POSTULATED BREAK

SUBSYSTEM SI11
BRAIDWOOD ONLY

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-93a

SAFETY INJECTION SUBSYSTEM SI11
POSTULATED BREAK LOCATIONS
(BRAIDWOOD)



• Indicates Postulated Break

SUBSYSTEM SI 16 (CONT'D)

Note:

Arbitrary Intermediate Break

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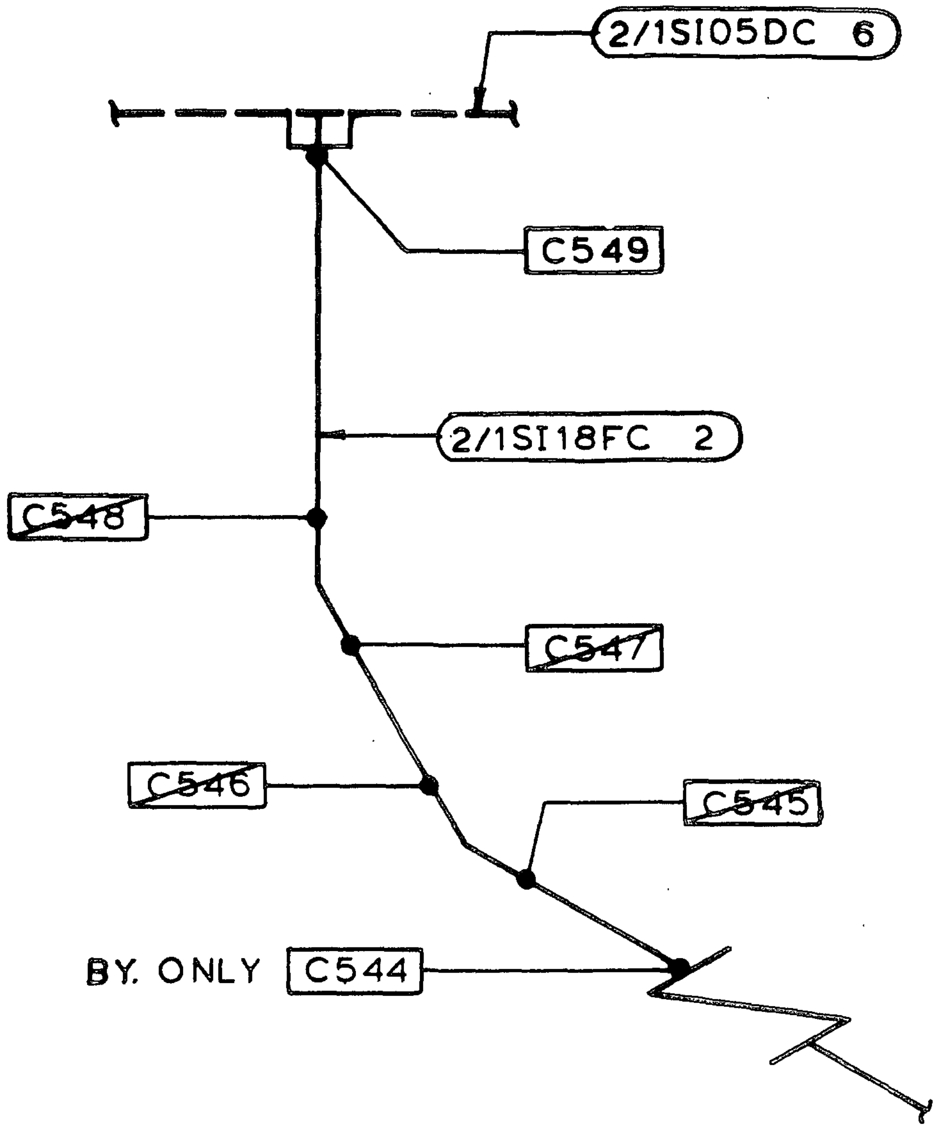
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-94

SAFETY INJECTION SUBSYSTEM SI16
POSTULATED BREAK LOCATIONS
(SHEET 1 OF 2)



SUBSYSTEM SI16 (Cont'd)

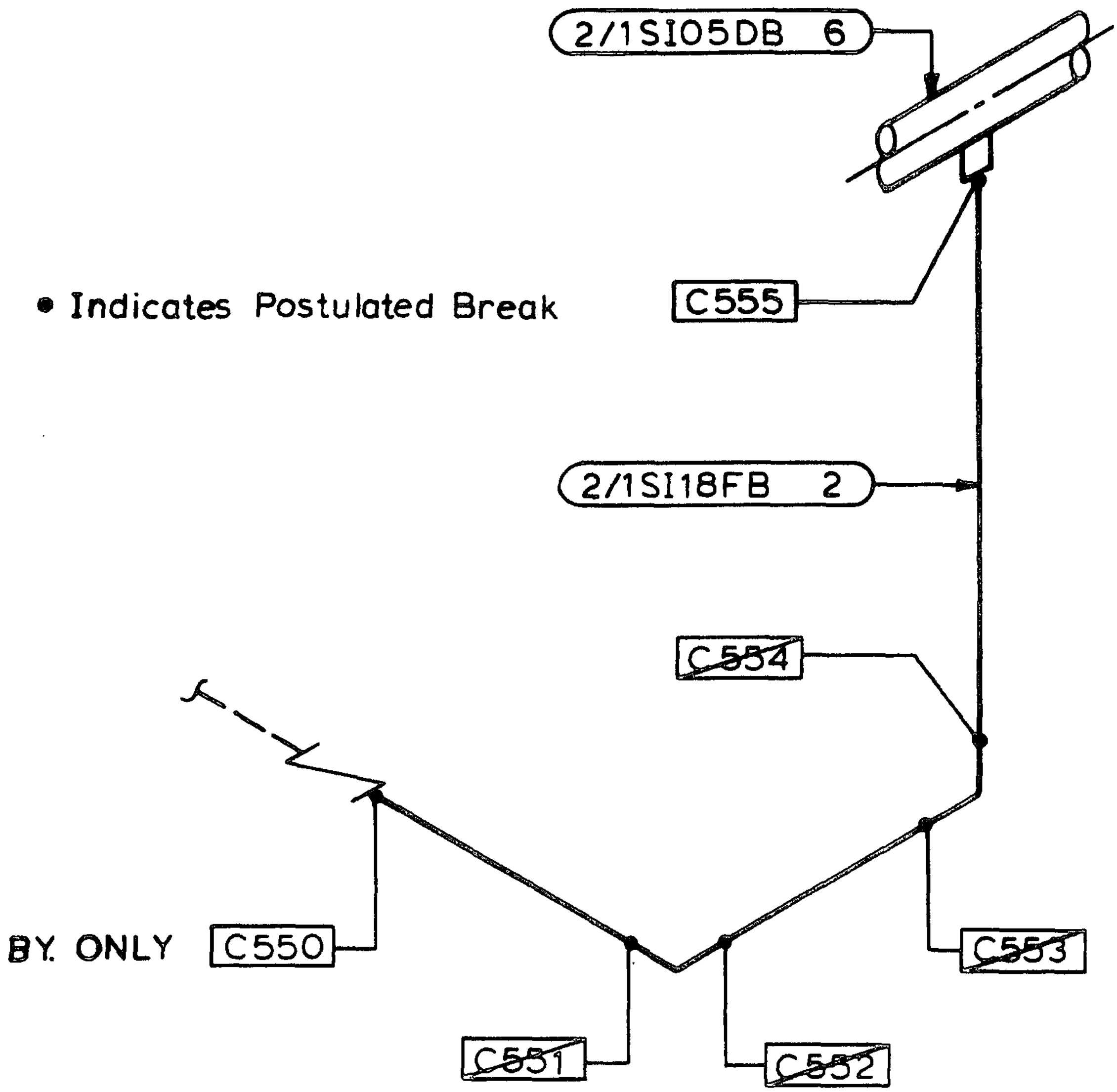
● Indicates Postulated Break

NOTE:
Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-94

SAFETY INJECTION SUBSYSTEM SI16
POSTULATED BREAK LOCATIONS
(SHEET 2 OF 2)



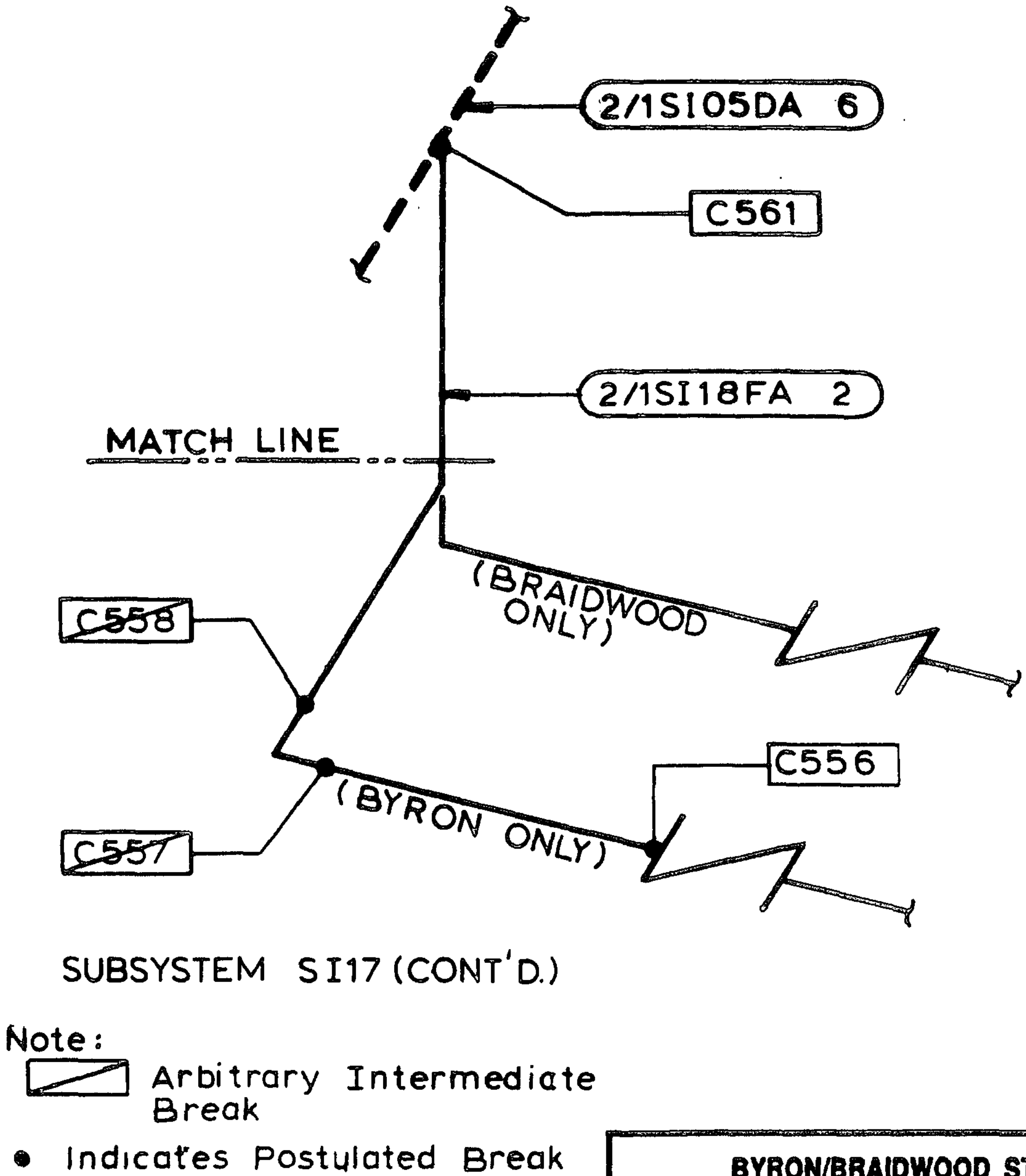
SUBSYSTEM SI17 (CONT'D.)

NOTE:
Arbitrary Intermediate Break

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-95

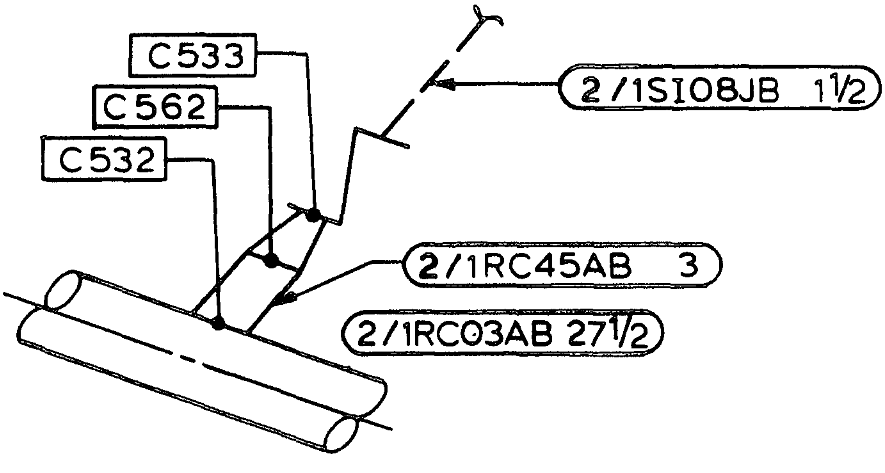
SAFETY INJECTION SUBSYSTEM SI17
POSTULATED BREAK LOCATIONS
(SHEET 1 OF 2)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-95

SAFETY INJECTION SUBSYSTEM SI17
POSTULATED BREAK LOCATIONS
(SHEET 2 OF 2)



SUBSYSTEM SI19

• Indicates Postulated Break

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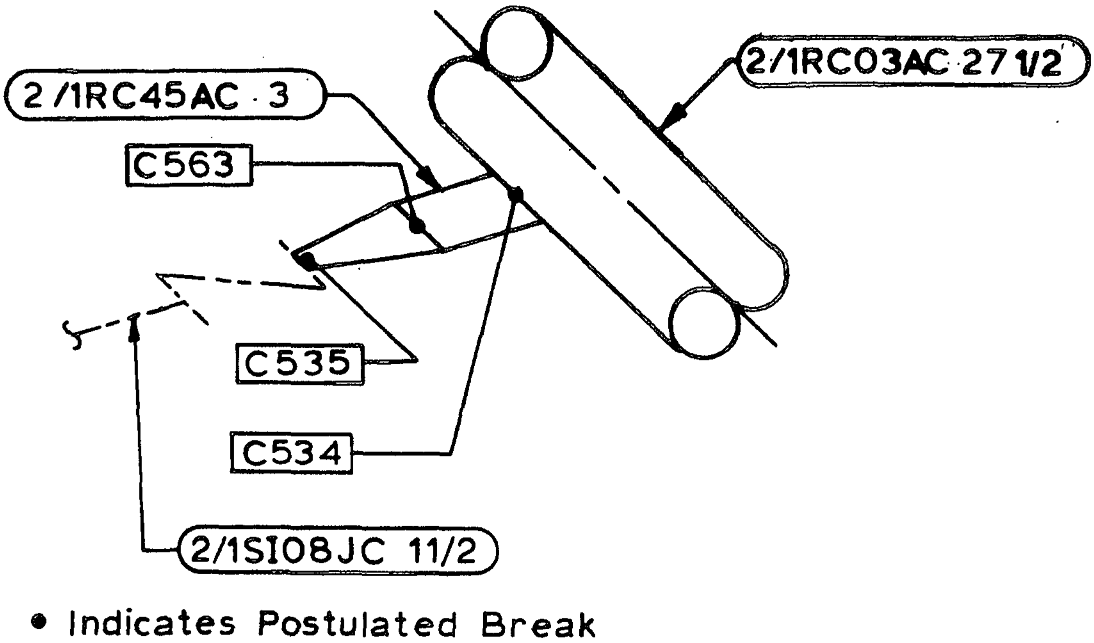
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-96

SAFETY INJECTION SUBSYSTEM SI19
POSTULATED BREAK LOCATIONS



SUBSYSTEM SI20

BYRON/BRAIDWOOD STATIONS
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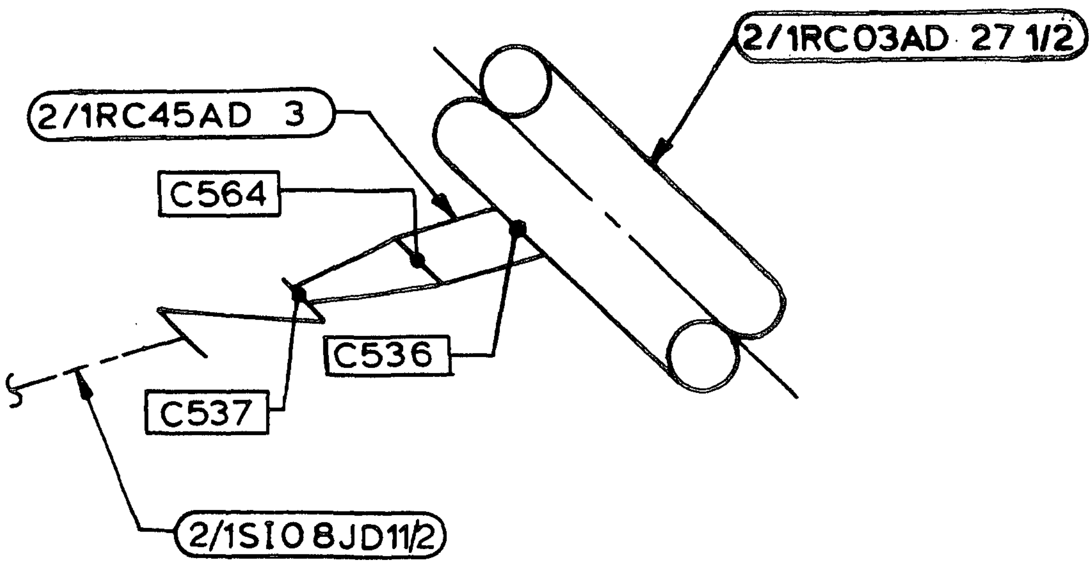
FIGURE 3.6-97

SAFETY INJECTION SUBSYSTEM SI20
POSTULATED BREAK LOCATIONS



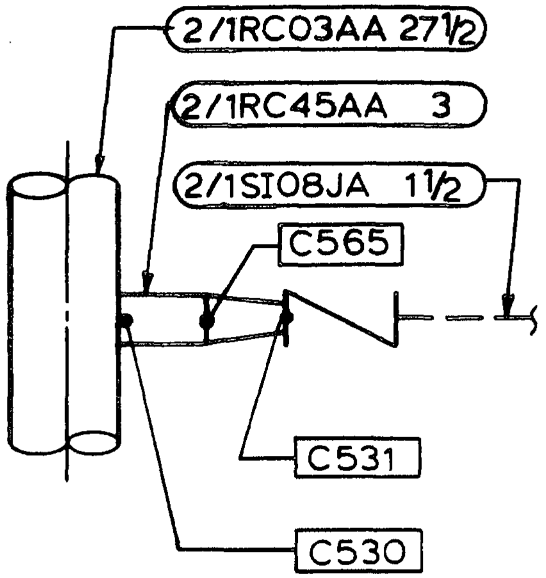
• Indicates Postulated Break

SUBSYSTEM SI22

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-98

SAFETY INJECTION SUBSYSTEM SI22
POSTULATED BREAK LOCATIONS



SUBSYSTEM SI24

● Indicates Postulated Break

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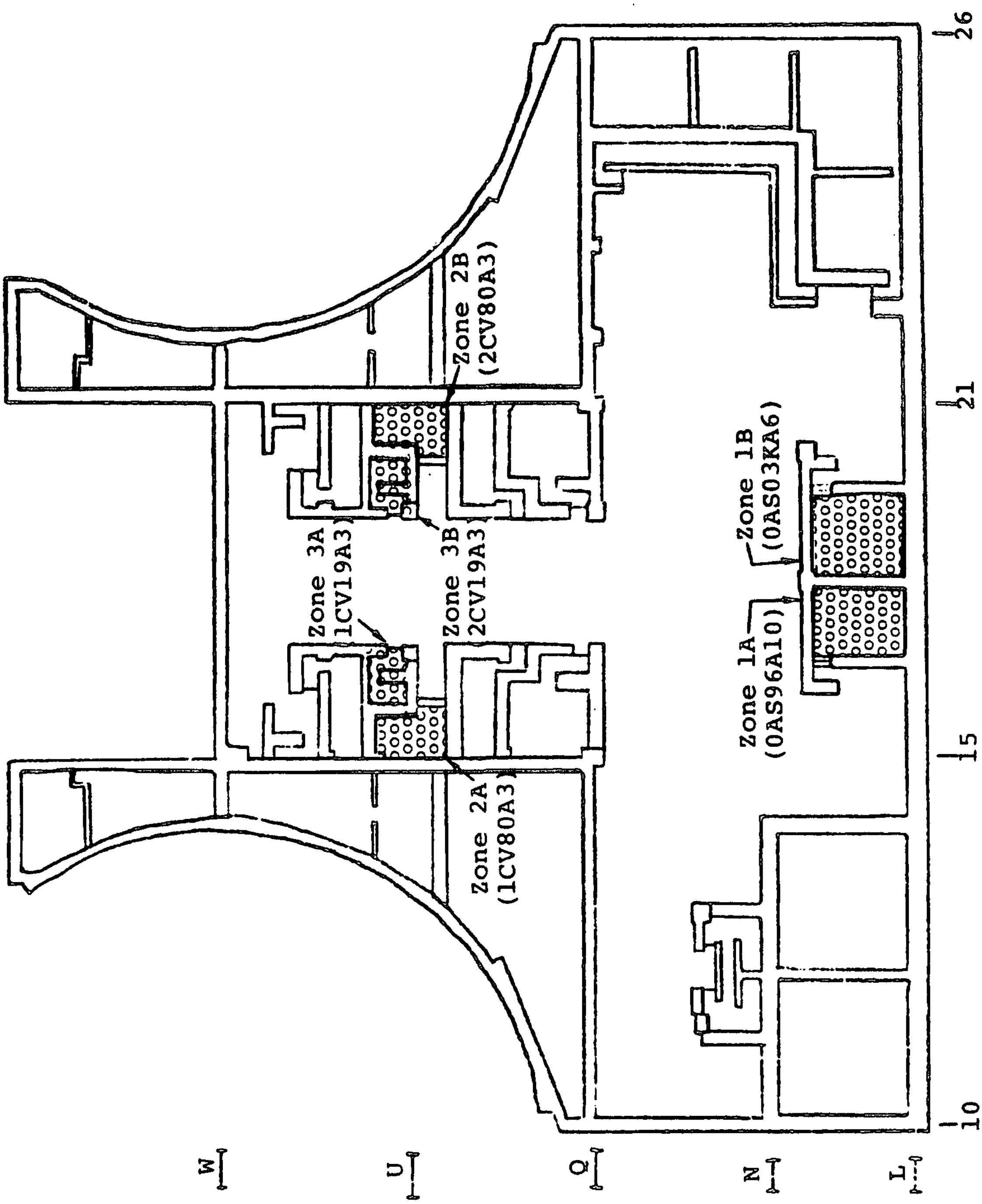
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-99

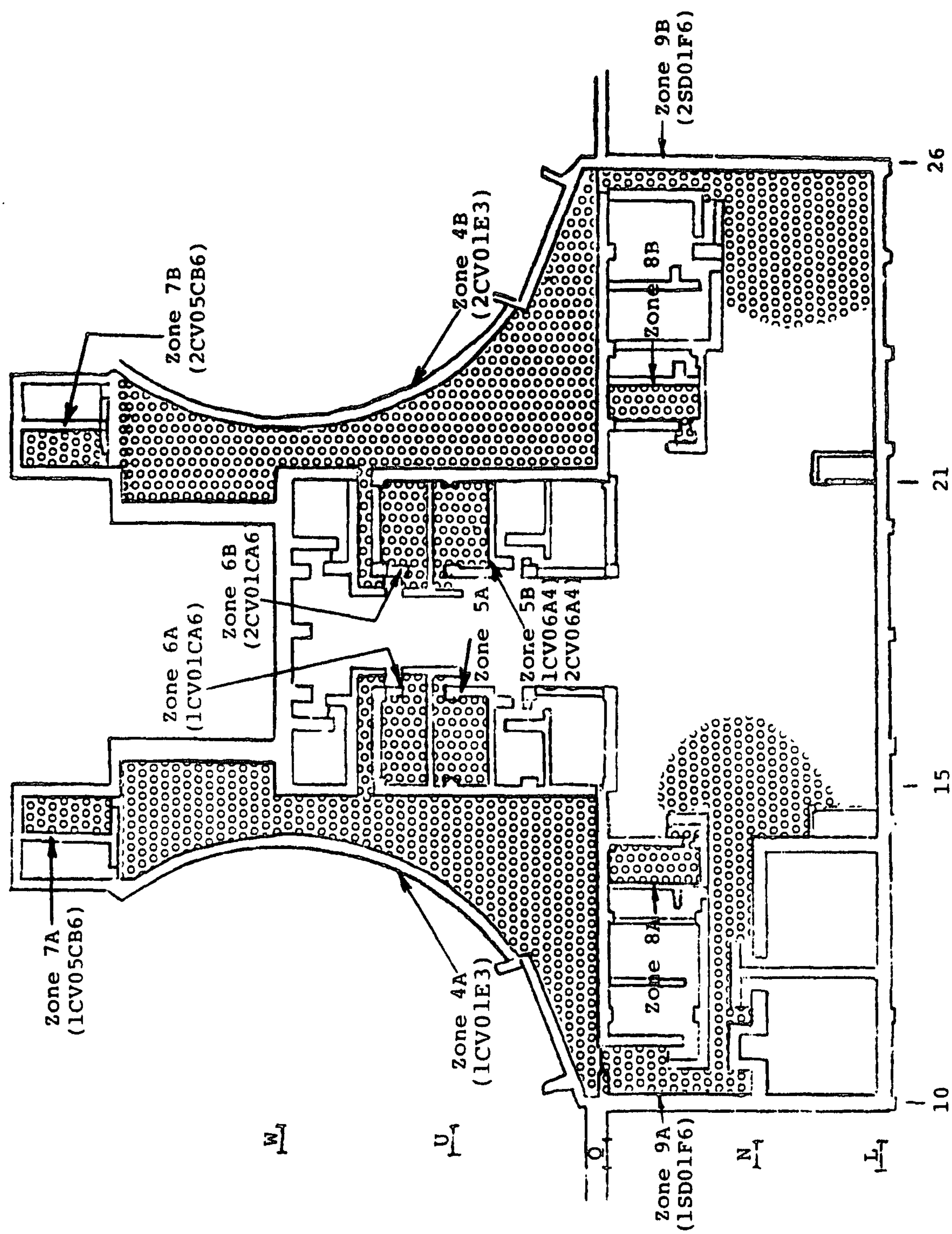
SAFETY INJECTION SUBSYSTEM SI24
POSTULATED BREAK LOCATIONS



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-100

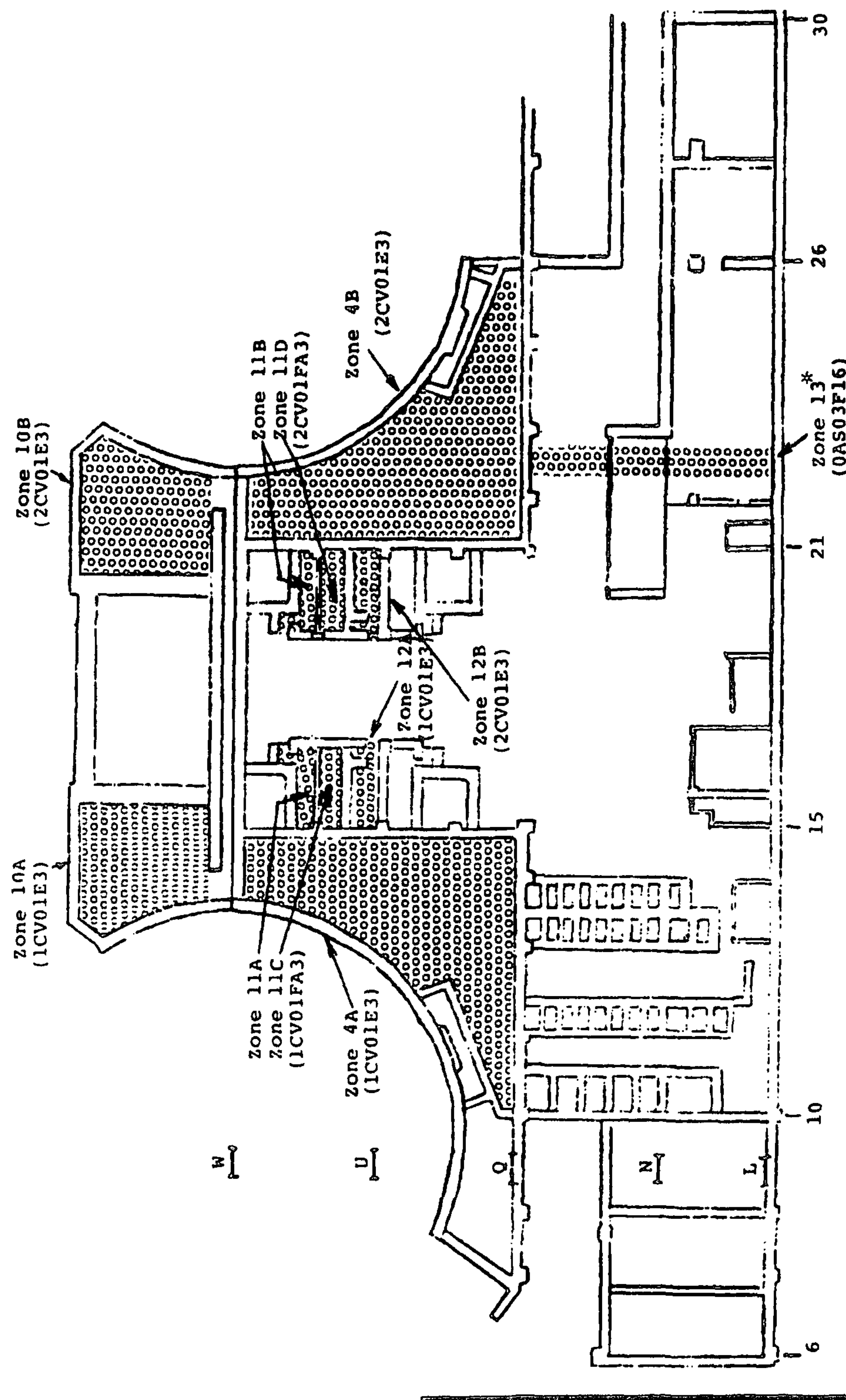
HELB ZONES OF INFLUENCE
ELEVATION 346 FT 0 IN.



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-101

HELB ZONES OF INFLUENCE
ELEVATION 364 FT 0 IN.



* AT BYRON, ZONE 13 IS NO LONGER A HELB ZONE DUE TO THE INSTALLATION OF A BLANK PLATE IN AUXILIARY STEAM SUPPLY PIPE 0AS03F-16 PRIOR TO ENTERING THE AUXILIARY BUILDING STEAM TUNNEL.

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FIGURE 3.6-102

HELB ZONES OF INFLUENCE
ELEVATION 383 FT 0 IN.

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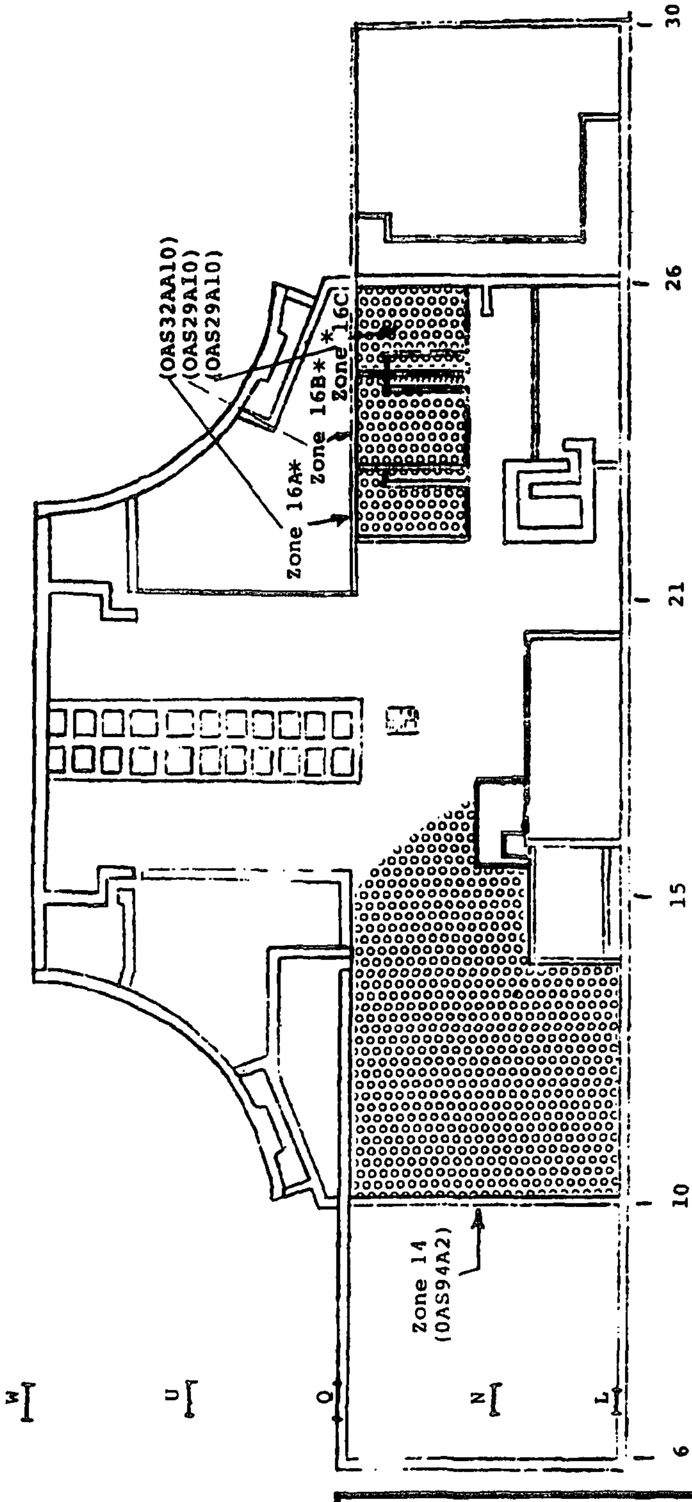
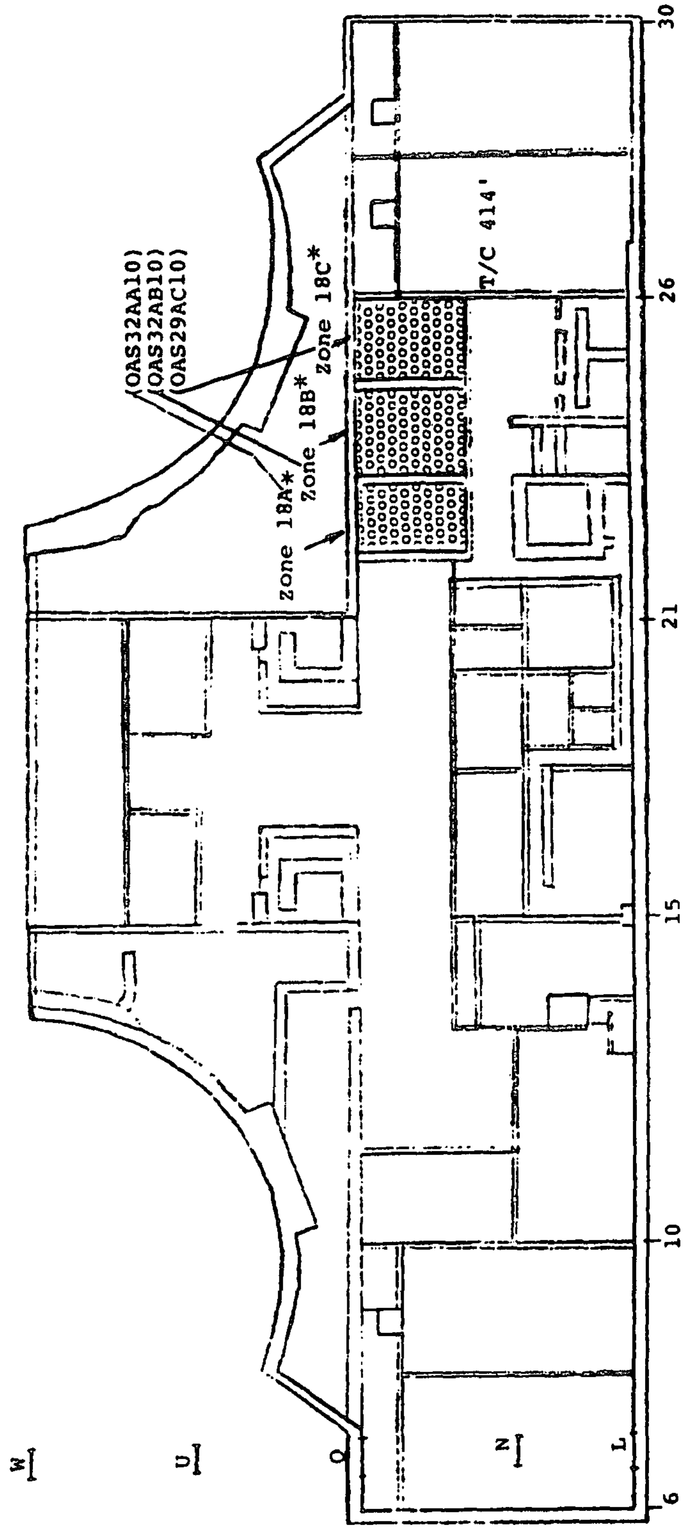


*AT BYRON, ZONES 16A, 16B, AND 16C ARE NO LONGER HELB ZONES DUE TO THE INSTALLATION OF A BLANK PLATE IN THE STEAM SUPPLY TO THE SURFACE CONDENSER ROOMS.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-103

HELB ZONES OF INFLUENCE
ELEVATION 401 FT 0 IN.



* AT BYRON, ZONES 18A, 18B, AND 18C ARE NO LONGER HELB ZONES DUE TO THE INSTALLATION OF A BLANK PLATE IN THE STEAM SUPPLY TO THE RADWASTE EVAPORATOR ROOMS.

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-104

HELB ZONES OF INFLUENCE
ELEVATION 426 FT 0 IN.

FLUID LINE * DESCRIPTIONS	LINE SIZE (IN.)	P (PSIG)	T (°F)	STEADY DISCHARGE RATE (LB/SEC.)
AUXILIARY STEAM LINE IN RECYCLE WASTE EVAPORATOR ROOM, EL. 346'0"	10.	50.	231.	CHOKED 109.
CV COOLANT LETDOWN LINE IN REHEAT HEAT EXCHANGER ROOM, EL. 346'0"	3.	600.	383.	LIMITED 27.
STEAM GENERATOR BLOWDOWN LINE IN BLOWDOWN CONDENSER ROOM, EL. 364'0"	14.	1093.	557.	CHOKED 50.
CV COOLANT LETDOWN LINE IN HEAT EXHANGER ROOM, EL. 383'0"	3.	600.	383.	LIMITED 27.
AUXILIARY STEAM LINE IN RADWASTE EVAPORATOR ROOM, EL. 414'0"; SURFACE CON-DENSER ROOM, EL. 401'0"; IN PIPING TUNNEL, EL. 394'0"**	16.	50.	297.7	CHOKED 174.4

*ASSUMING NO CUT-OFF VALVE

**AT BYRON, THE STEAM SUPPLY TO THE AUXILIARY STEAMLINE PIPING TUNNEL, SURFACE CONDENSER ROOMS, AND RADWASTE EVAPORATOR ROOMS HAS BEEN PERMANENTLY ISOLATED IN THE TURBINE BUILDING BY THE INSTALLATION OF A BLANK PLATE.

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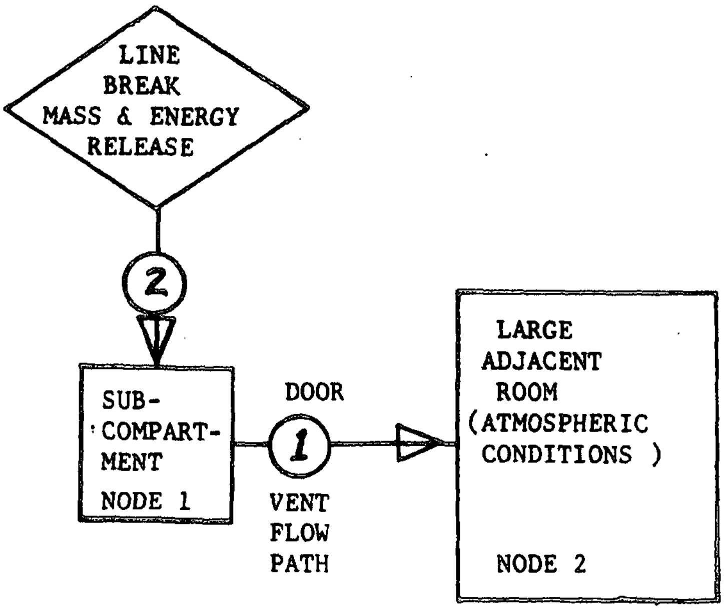
FIGURE A3.6-1

FLUID ENERGY LINES IN
AUXILIARY BUILDING SUBCOMPARTMENTS



FOR ELEVATIONS: 346'-0", 364'-0", 383'-0" and 401'-0"

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-2

BASIC FLOW SIMULATION MODEL

AIR TEMPERATURE = 114.°F
RELATIVE HUMIDITY = 0.%

DIMENSIONS AND INITIAL CONDITIONS OF CONTROL VOLUMES*

NODE NO.	AREA FT^2	HEIGHT FT.	EXIT ELEVATION FT.	INLET ELEVATION FT.	BOTTOM ELEVATION FT.	INITIAL PRESSURE PSIA
1	335.0	17.3	355.0	355.0	346.0	14.7
2	10^6	10^6	355.0	355.0	1.0	14.7

DIMENSIONS AND INITIAL CONDITIONS OF FLOW PATHS

PATH NO.	NODE FROM	TO	TYPE†	SUM L/A, Ft^{-1}	AREA FT^2	DIAMETER FT	k FACTOR
1	1	2	9	0.335	33.33	1.0	3.72
2		1	5	0.0			

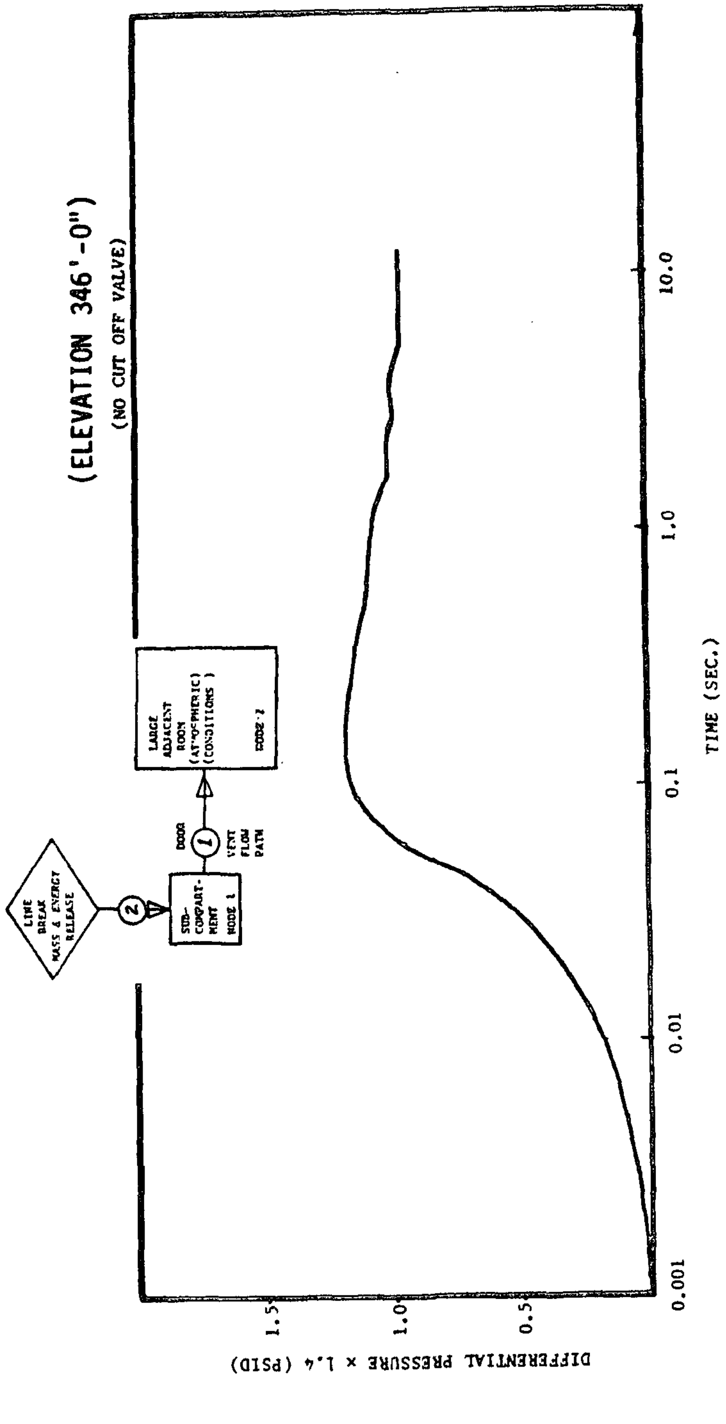
† TYPE 9 - FLOW PATH AREA AS A FUNCTION OF PRESSURE
TYPE 5 - TIME DEPENDANT INPUT RATE FLOW PATH

*ELEVATION 346'-0"

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-3

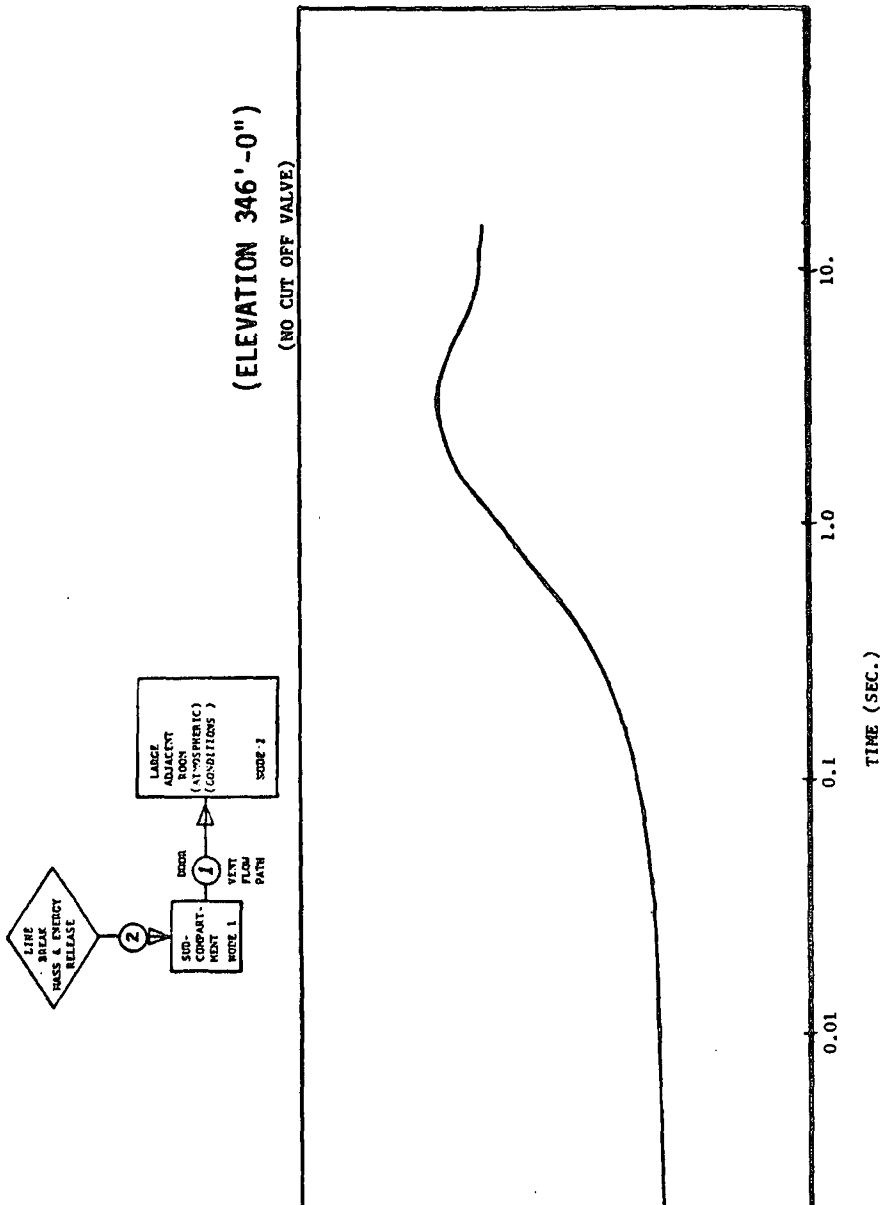
AUXILIARY STEAMLINE BREAK IN
RECYCLE WASTE EVAPORATOR ROOMS



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-4

DIFFERENTIAL PRESSURE VS. TIME—AUXILIARY STEAMLINE BREAK IN RECYCLE WASTE EVAPORATOR ROOM



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-5

TEMPERATURE VS. TIME—AUXILIARY STEAMLINE BREAK IN RECYCLE WASTE EVAPORATOR ROOM

AIR TEMPERATURE = 95.°F
RELATIVE HUMIDITY = 50.%

DIMENSIONS AND INITIAL CONDITIONS OF
CONTROL VOLUMES

NODE NO.	AREA FT^2	HEIGHT FT.	EXIT ELEVATION FT.	INLET ELEVATION FT.	BOTTOM ELEVATION FT.	INITIAL PRESSURE PSIA
1	97.977	6.54	349.0	349.0	346.0	14.7
2	10^6	10^6	349.0	349.0	1.0	14.7

DIMENSIONS AND INITIAL CONDITIONS OF
FLOW PATHS

PATH NO.	NODE FROM	TO	TYPE†	SUM L/A, Ft^{-1}	AREA FT^2	DIAMETER FT	k FACTOR
1	1	2	9	0.872	16.35	1.0	2.932
2		1	5				

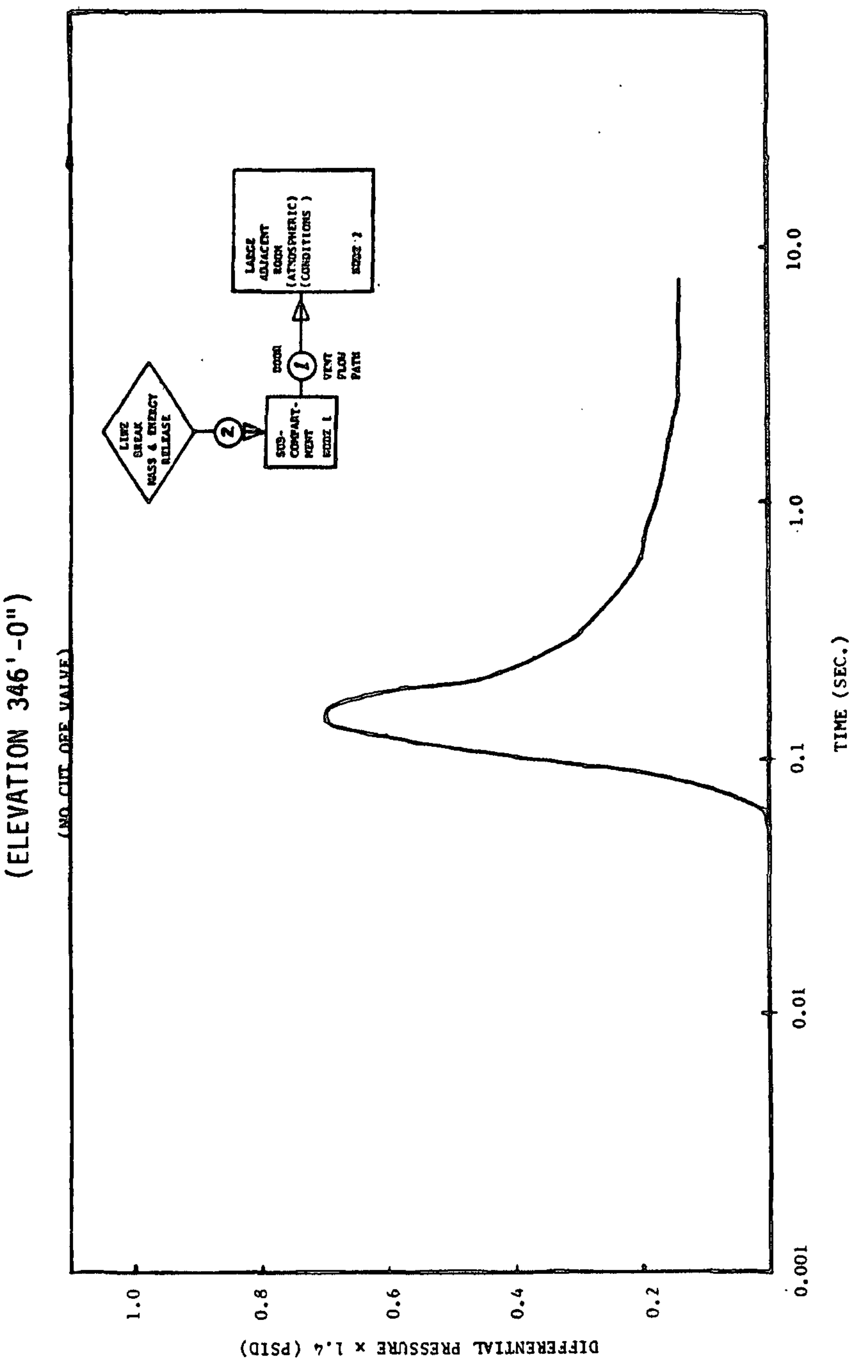
† TYPE 9 - FLOW PATH AREA AS A FUNCTION OF PRESSURE
TYPE 5 - TIME DEPENDANT INPUT RATE FLOW PATH

(ELEVATION 346'-0")

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-6

CV COOLANT LETDOWN LINE BREAK IN
LETDOWN REHEAT HEAT EXCHANGER ROOM



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-7

DIFFERENTIAL PRESSURE VS. TIME—
CV COOLANT LETDOWN LINE BREAK IN
LETDOWN REHEAT HEAT EXCHANGER ROOM

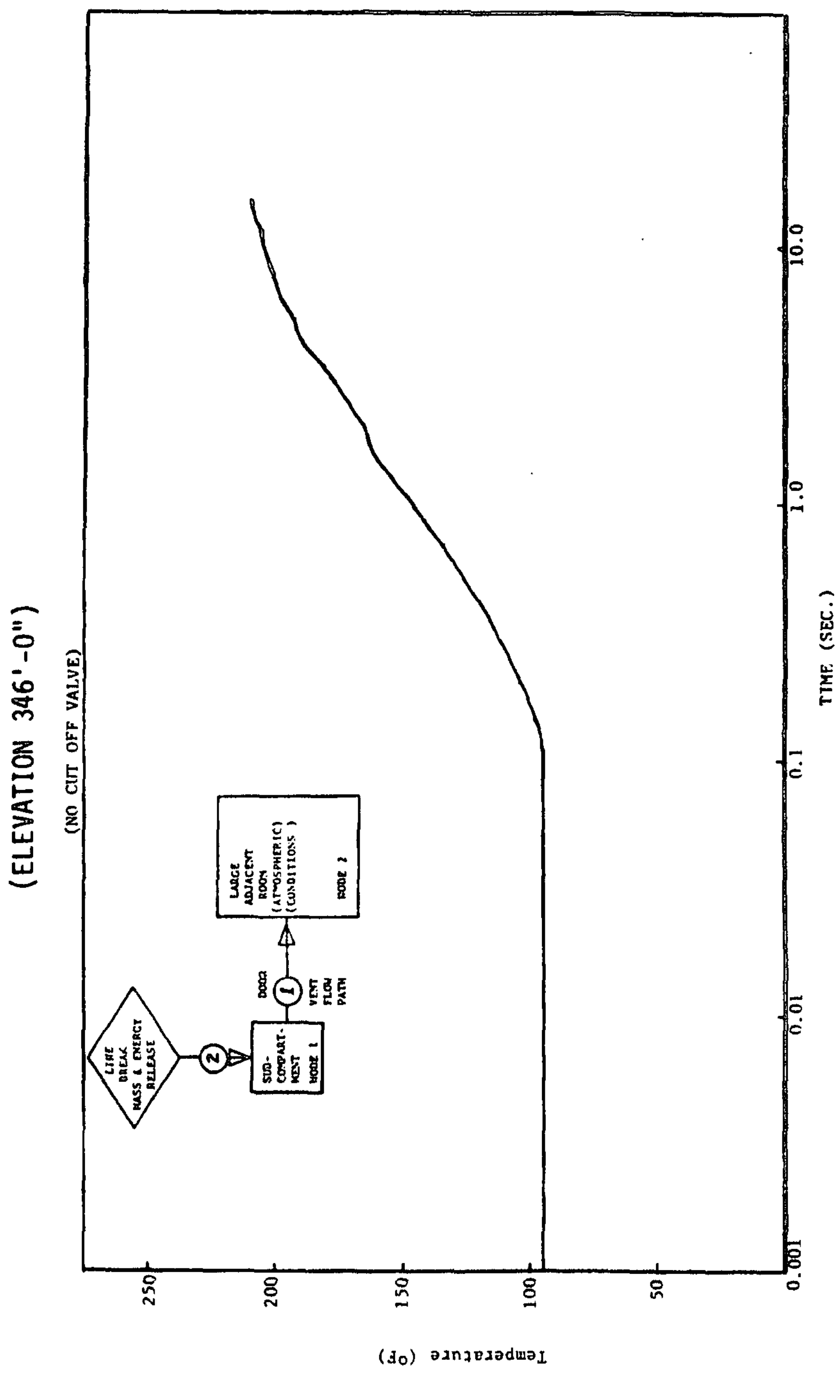


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FIGURE A3.6-8

TRANSIENT TEMPERATURE VS. TIME—
CV COOLANT LETDOWN LINE BREAK IN
LETDOWN REHEAT HEAT EXCHANGER ROOM

AIR TEMPERATURE = 112°F
RELATIVE HUMIDITY = 70%

DIMENSIONS AND INITIAL CONDITIONS OF
CONTROL VOLUMES

NODE NO.	AREA FT^2	HEIGHT FT.	EXIT ELEVATION FT.	INLET ELEVATION FT.	BOTTOM ELEVATION FT.	INITIAL PRESSURE PSIA
1	211.8	17.0	373.0	373.0	364.0	14.7
2	10^5	10	373.0	373.0	1.0	14.7

DIMENSIONS AND INITIAL CONDITIONS FOR
FLOW PATHS

NO.	NODE FROM	NODE TO	TYPE†	SUM L/A, Ft^{-1}	AREA FT^2	DIAMETER FT	k FACTOR
1	1	2	9	1.2	21.0	1.0	2.213
2		1	5				

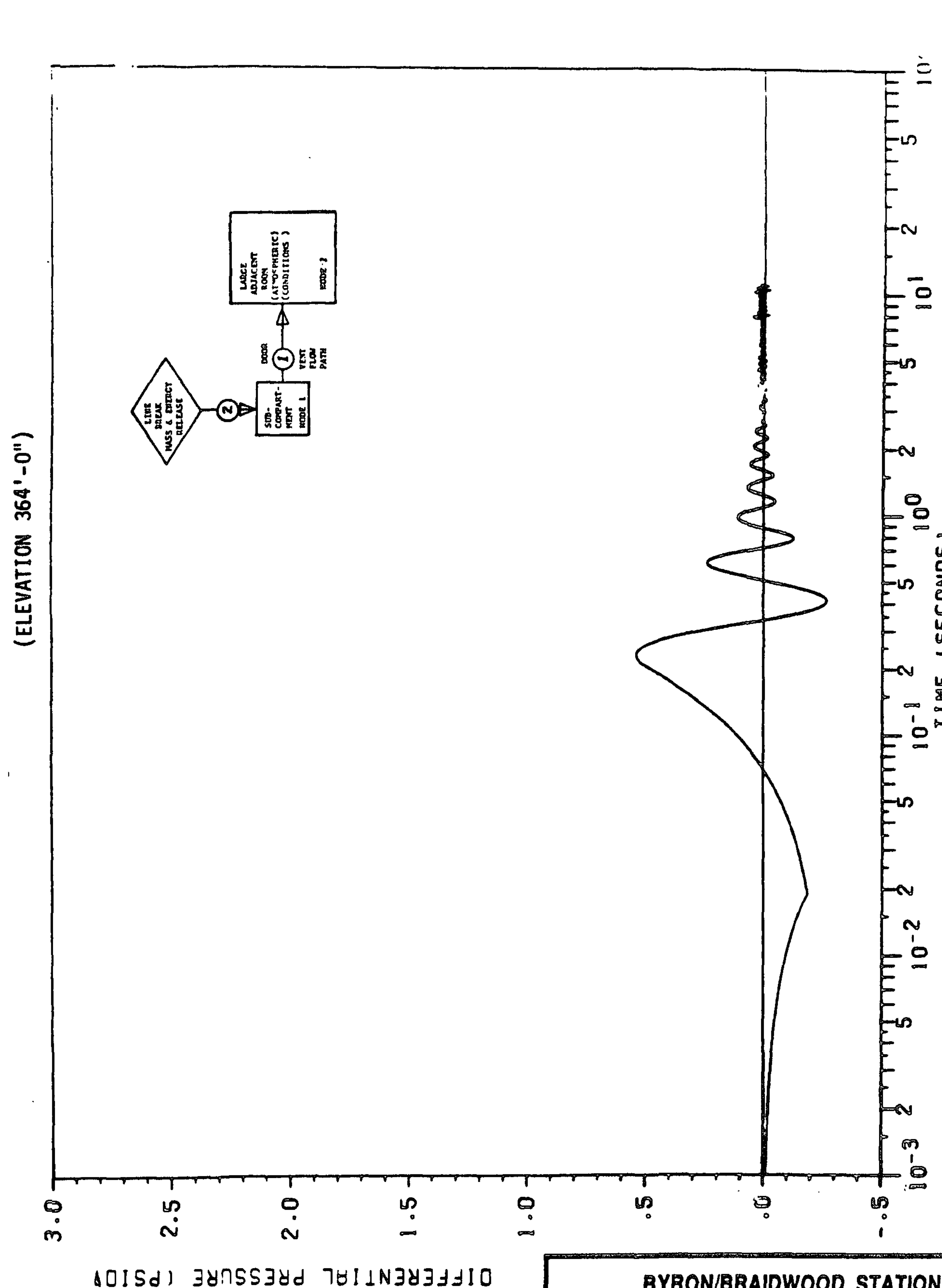
† TYPE 9 - FLOW PATH AREA AS A FUNCTION OF PRESSURE
TYPE 5 - TIME DEPENDANT INPUT RATE FLOW PATH

(ELEVATION 364'-0")

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-9

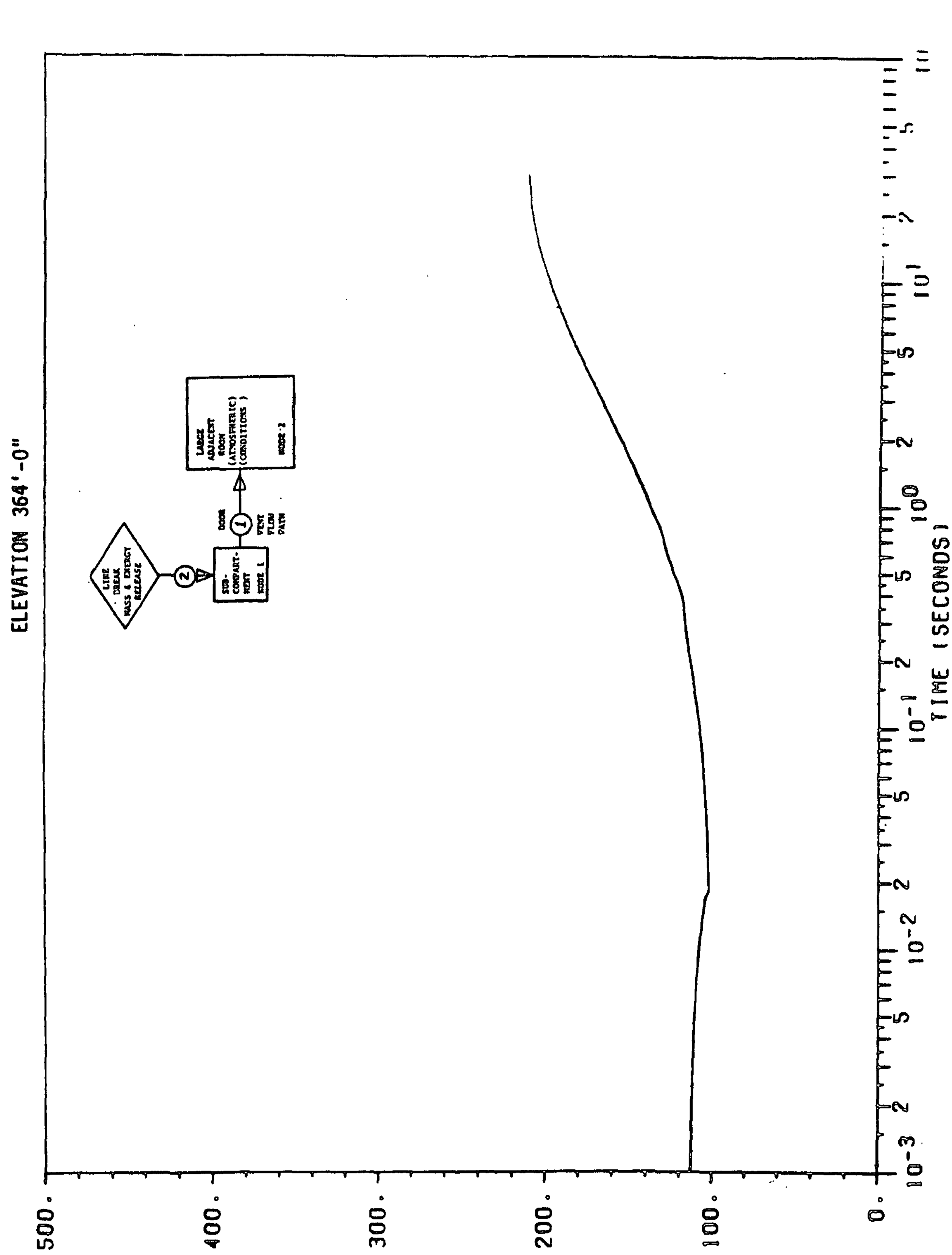
STEAM GENERATOR BLOWDOWN LINE BREAK
IN BLOWDOWN CONDENSER ROOM



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-10

DIFFERENTIAL PRESSURE VS. TIME—
STEAM GENERATOR BLOWDOWN LINE BREAK
IN BLOWDOWN CONDENSER ROOM



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-11

TEMPERATURE VS. TIME—
STEAM GENERATOR BLOWDOWN LINE BREAK
IN BLOWDOWN CONDENSER ROOM

DIMENSIONS AND INITIAL CONDITIONS OF
CONTROL VOLUMES

NODE NO.	AREA FT^2	HEIGHT FT.	EXIT ELEVATION FT.	INLET ELEVATION FT.	BOTTOM ELEVATION FT.	INITIAL PRESSURE PSIA
1	164.43	14.4	386.0	386.0	383.0	14.7
2	10^6	10^6	386.0	386.0	1.0	14.7

DIMENSIONS AND INITIAL CONDITIONS OF
FLOW PATHS

NO.	NODE FROM	TO	TYPE †	SUM L/A, Ft^{-1}	AREA FT^2	DIAMETER FT	k FACTOR
1	1	2	9	0.525	18.84	1.0	2.453
2		1	5				

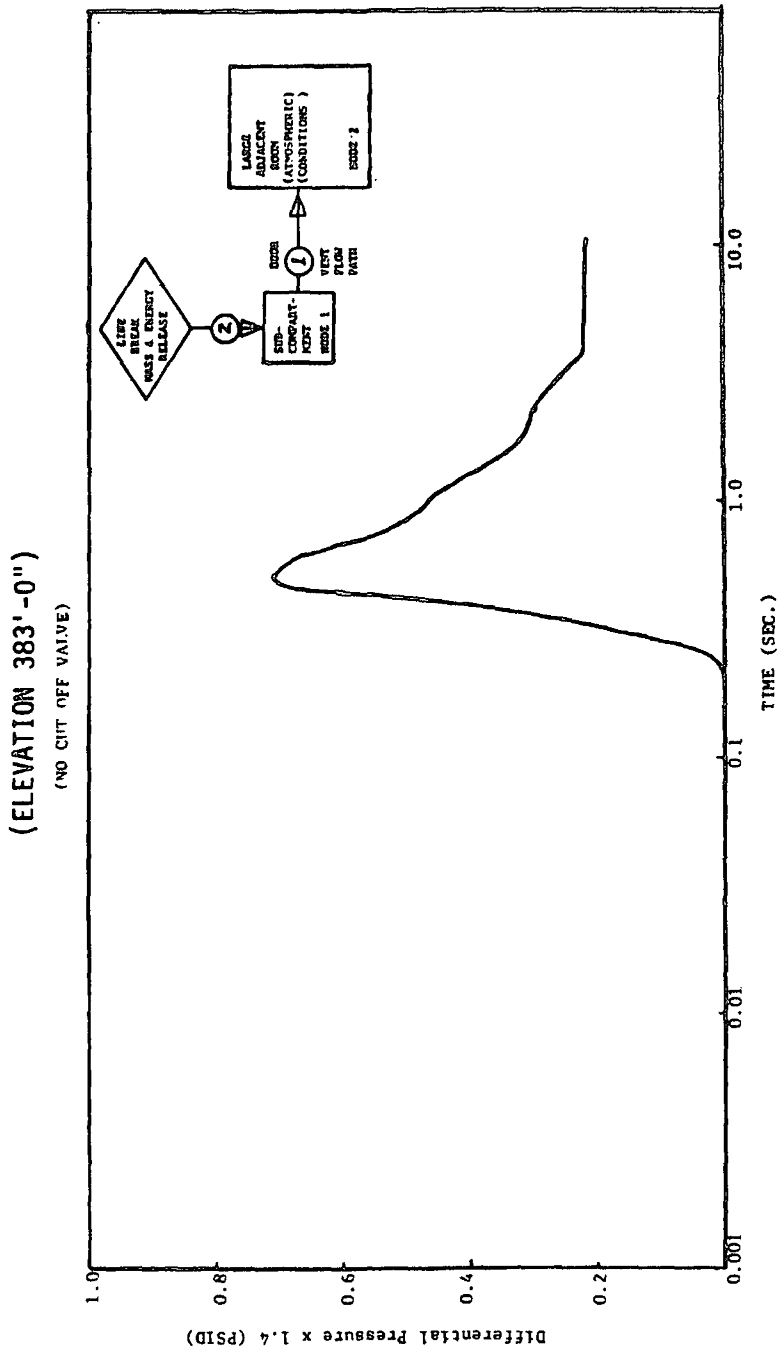
† TYPE 9 - FLOW PATH AREA AS A FUNCTION OF PRESSURE
TYPE 5 - TIME DEPENDANT INPUT RATE FLOW PATH

(ELEVATION 383'-0")

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-12

CV COOLANT LETDOWN LINE BREAK IN
LETDOWN HEAT EXCHANGER ROOM



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-13

DIFFERENTIAL PRESSURE VS. TIME—
CV COOLANT LETDOWN LINE BREAK IN
LETDOWN HEAT EXCHANGER ROOM

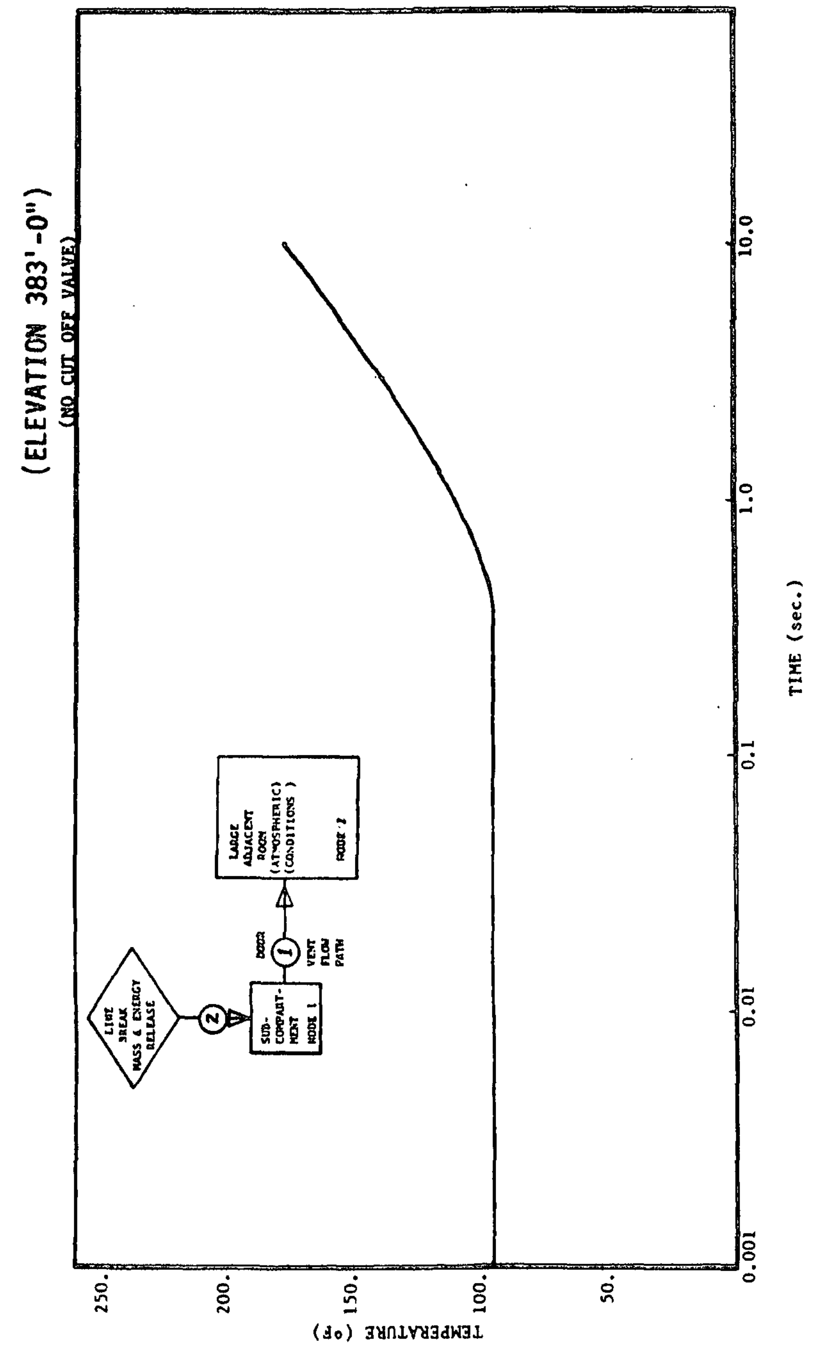


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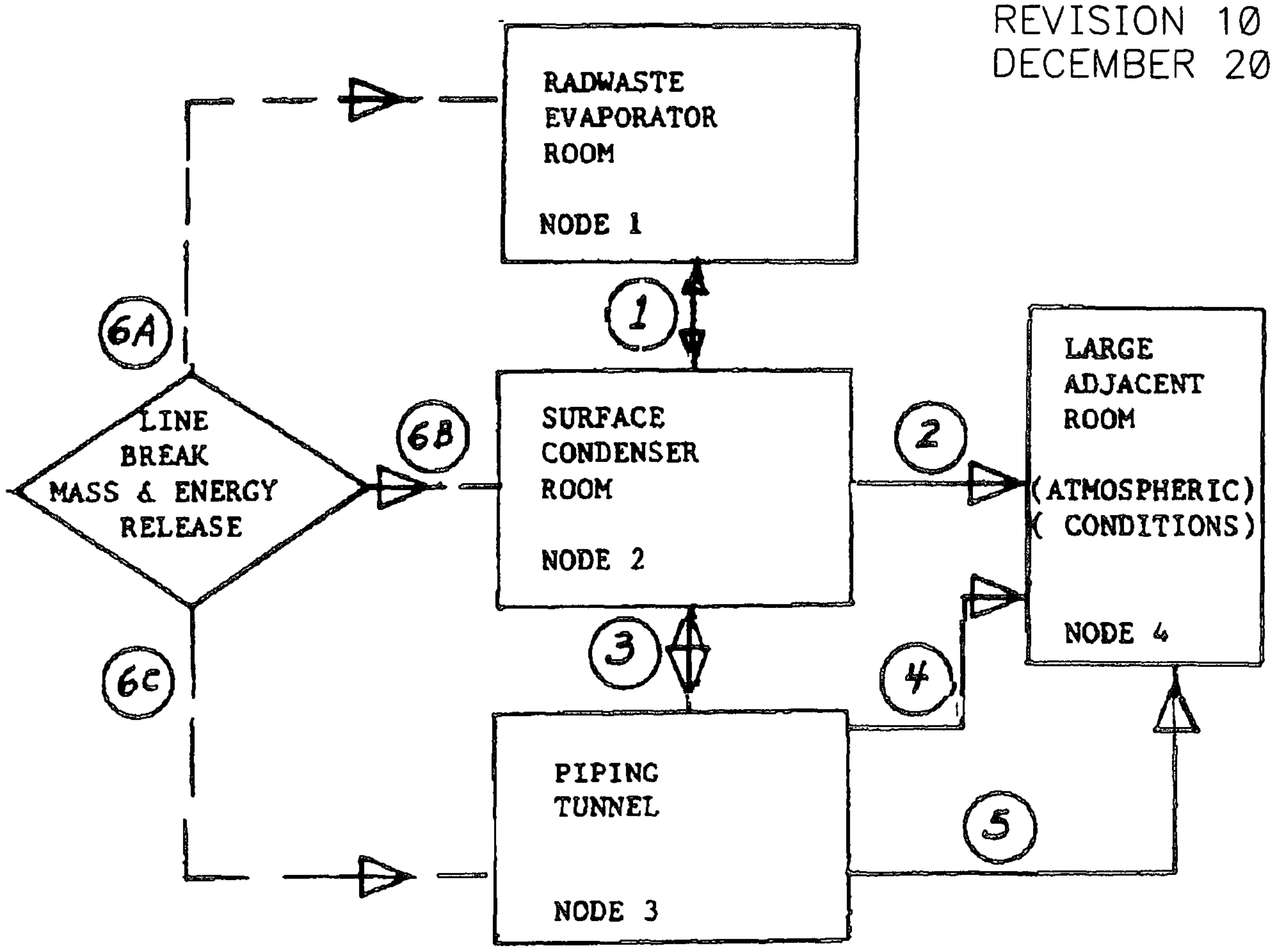
FIGURE A3.6-14

TEMPERATURE VS. TIME—
CV COOLANT LETDOWN LINE BREAK IN
LETDOWN HEAT EXCHANGER ROOM

Figures A3.6-15 through A3.6-17 have been deleted.



FOR ELEVATIONS: 394'-0", 401'-0" and 414'-0"

BRAIDWOOD STATION

UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE A3.6-18

BASIC FLOW SIMULATION MODEL
AUXILIARY STEAMLINE BREAK

AIR TEMPERATURE = 95.°F
RELATIVE HUMIDITY = 50.%

DIMENSIONS AND INITIAL CONDITIONS OF CONTROL VOLUMES

NODE NO.	AREA FT^2	HEIGHT FT.	EXIT ELEVATION FT.	INLET ELEVATION FT.	BOTTOM ELEVATION FT.	INITIAL PRESSURE PSIA
1	562.64	22.0	415.	424.	414.	14.7
2	611.3	10.0	402.	409.	401.	14.7
3	694.5	4.0	395.	396.	393.94	14.7
4	10^6	10^6	402.	402.	346.	14.7

DIMENSIONS AND INITIAL CONDITIONS FOR FLOW PATHS

PATH NO.	NODE FROM	TO	TYPE	SUM L/A, Ft^{-1}	AREA FT^2	DIAMETER FT.	k FACTOR
1	1	2	8*	1.	78.33	1.	3.40
2	2	4	9†	0.2555	63.02	1.	2.914
3	2	3	9	16.77	2.88	1.	2.732
4	3	4	9	0.556	9.	1.	2.914
5	3	4	9	0.479	8.0	1.	1.215
6		1	5				

* TYPE 8, Normal Flow Path
† TYPE 9 – FLOW PATH AREA AS A FUNCTION OF PRESSURE
TYPE 5 – TIME DEPENDANT INPUT RATE FLOW PATH

FOR BREAKS IN:

RADWASTE EVAPORATOR ROOM
EL. 414'–0"
SURFACE CONDENSER ROOM
EL. 401'–0"
PIPING TUNNEL
EL. 394'–0"

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FIGURE A3.6–19

AUXILIARY STEAMLINE BREAK

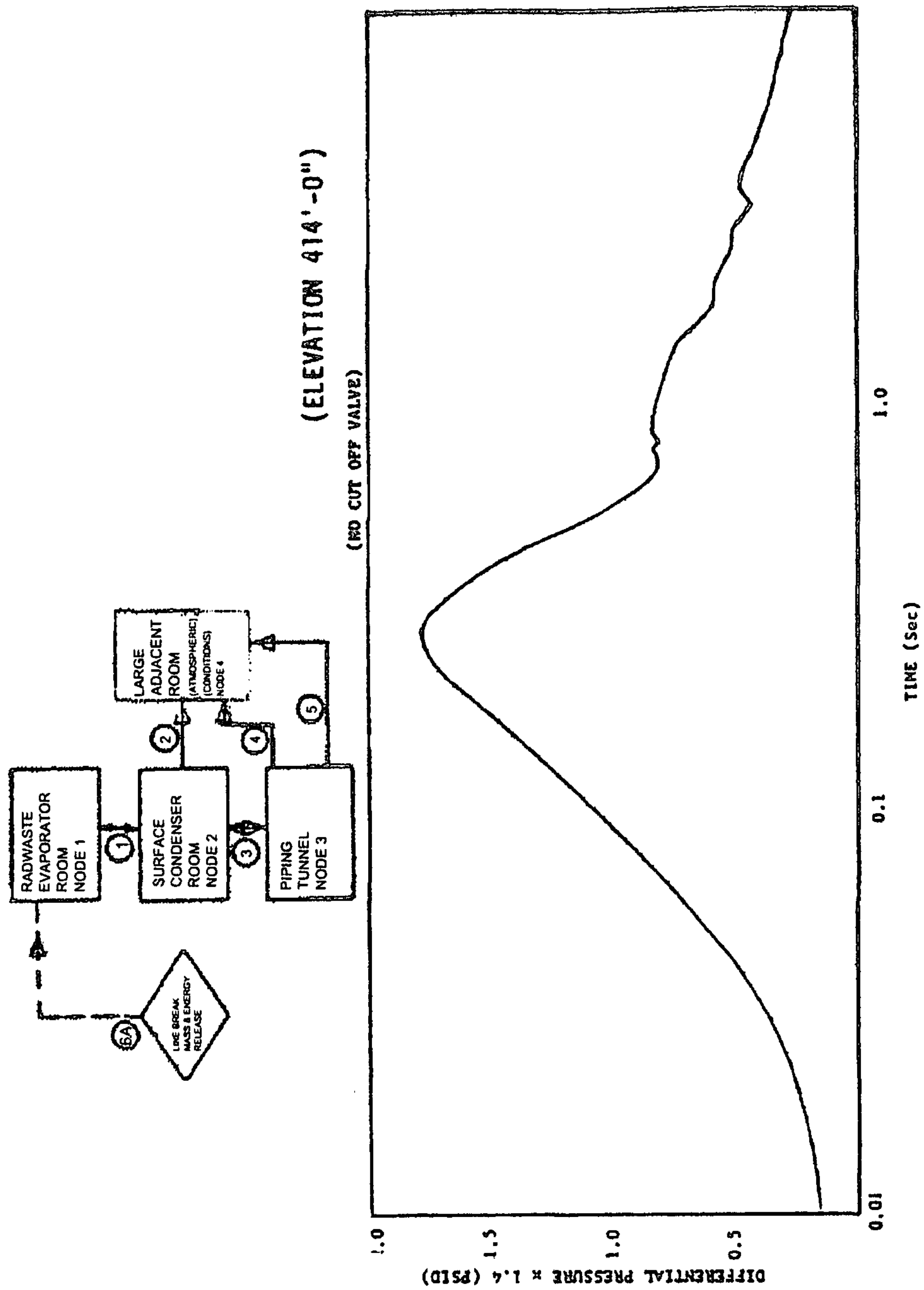

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FIGURE A3.6-20

DIFFERENTIAL PRESSURE VS. TIME—
AUXILIARY STEAMLINE BREAK IN
RADWASTE EVAPORATOR

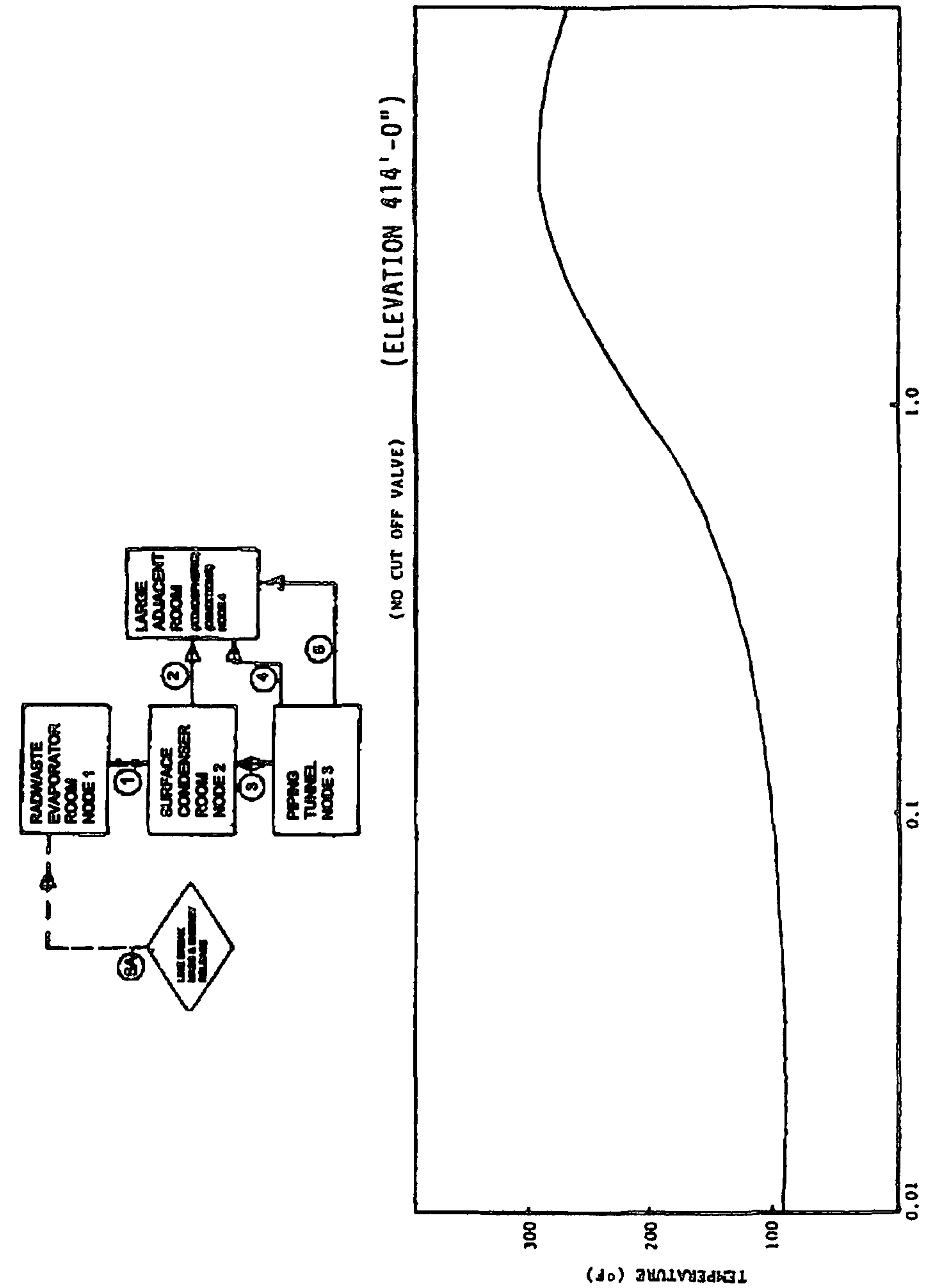

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FIGURE A3.6-21

TEMPERATURE VS. TIME—
AUXILIARY STEAMLINE BREAK IN
RADWASTE EVAPORATOR ROOM

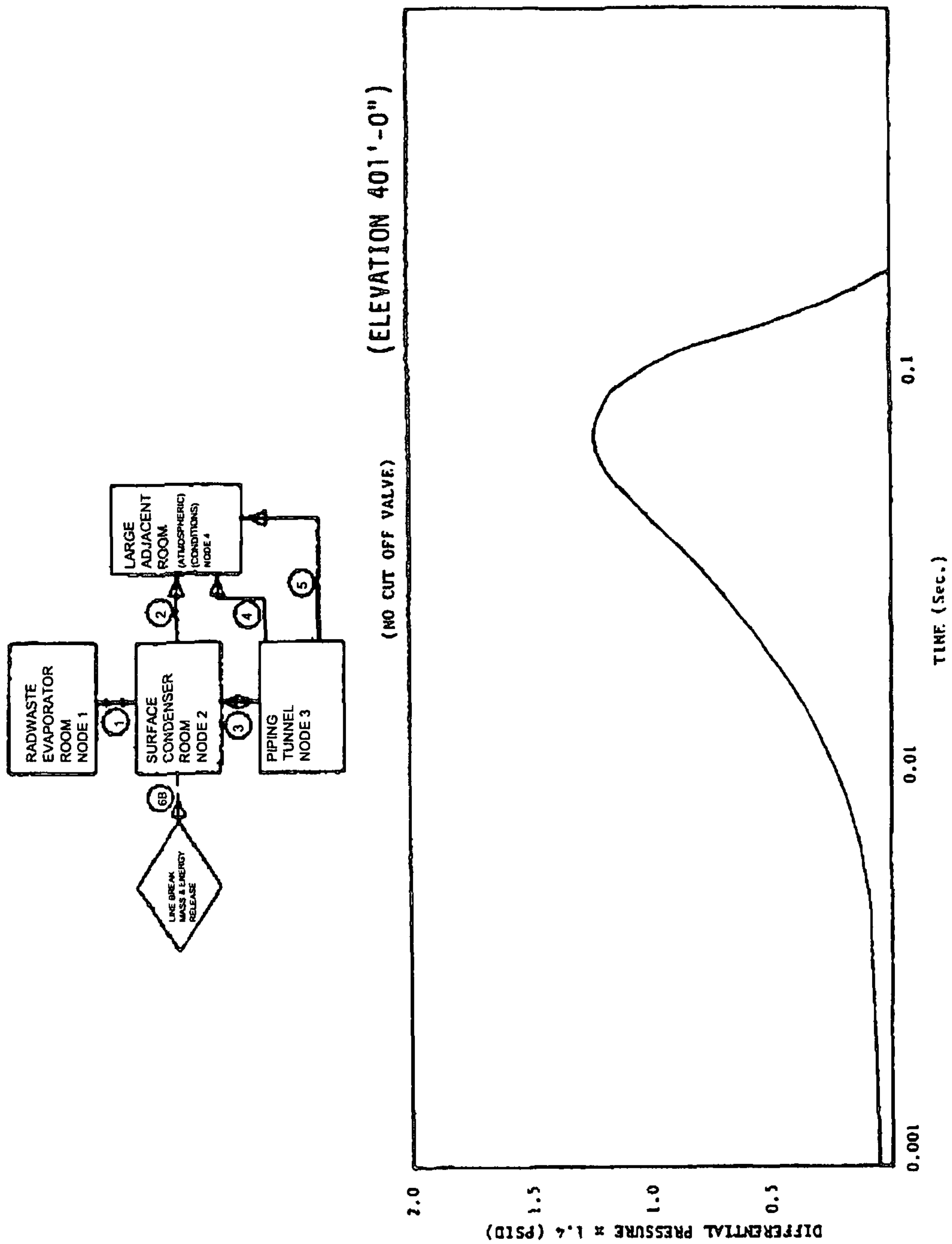

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FIGURE A3.6-22

DIFFERENTIAL PRESSURE VS. TIME—
AUXILIARY STEAMLINE BREAK IN
SURFACE CONDENSER ROOM

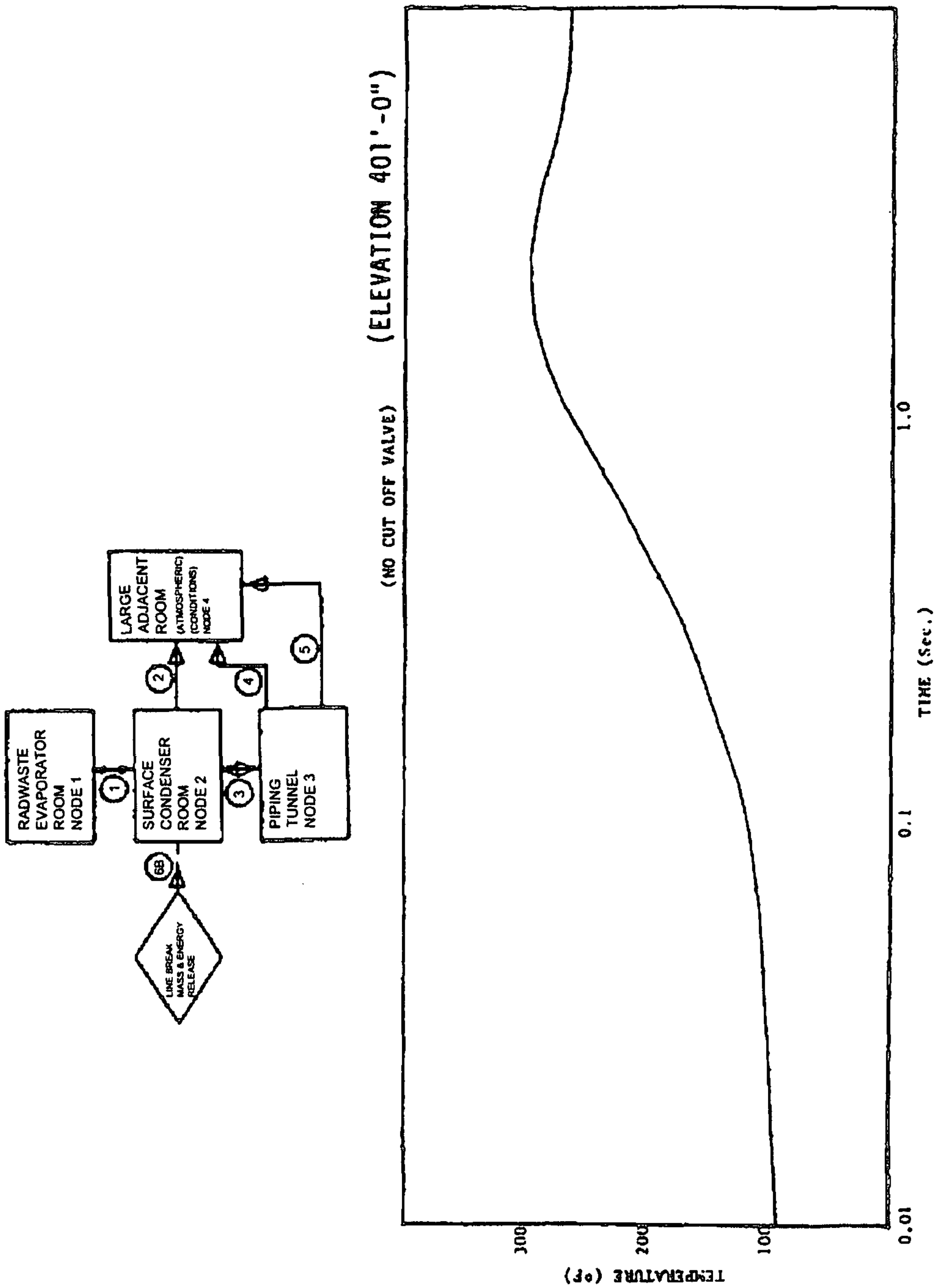

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FIGURE A3.6-23

TEMPERATURE VS. TIME—
AUXILIARY STEAMLINE BREAK IN
SURFACE CONDENSER ROOM

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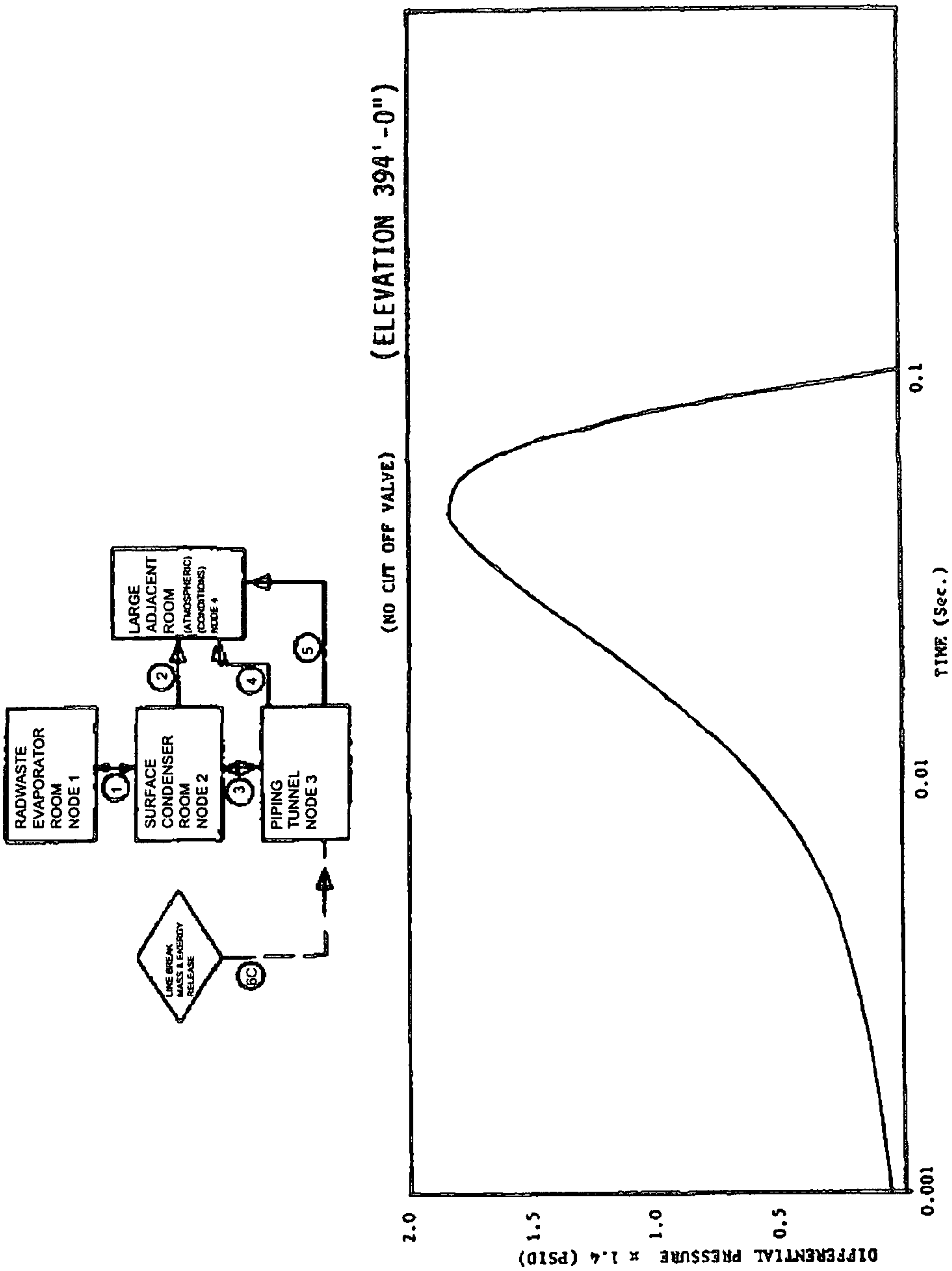

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FIGURE A3.6-24

DIFFERENTIAL PRESSURE VS. TIME—
AUXILIARY STEAMLINE BREAK IN
PIPE TUNNEL

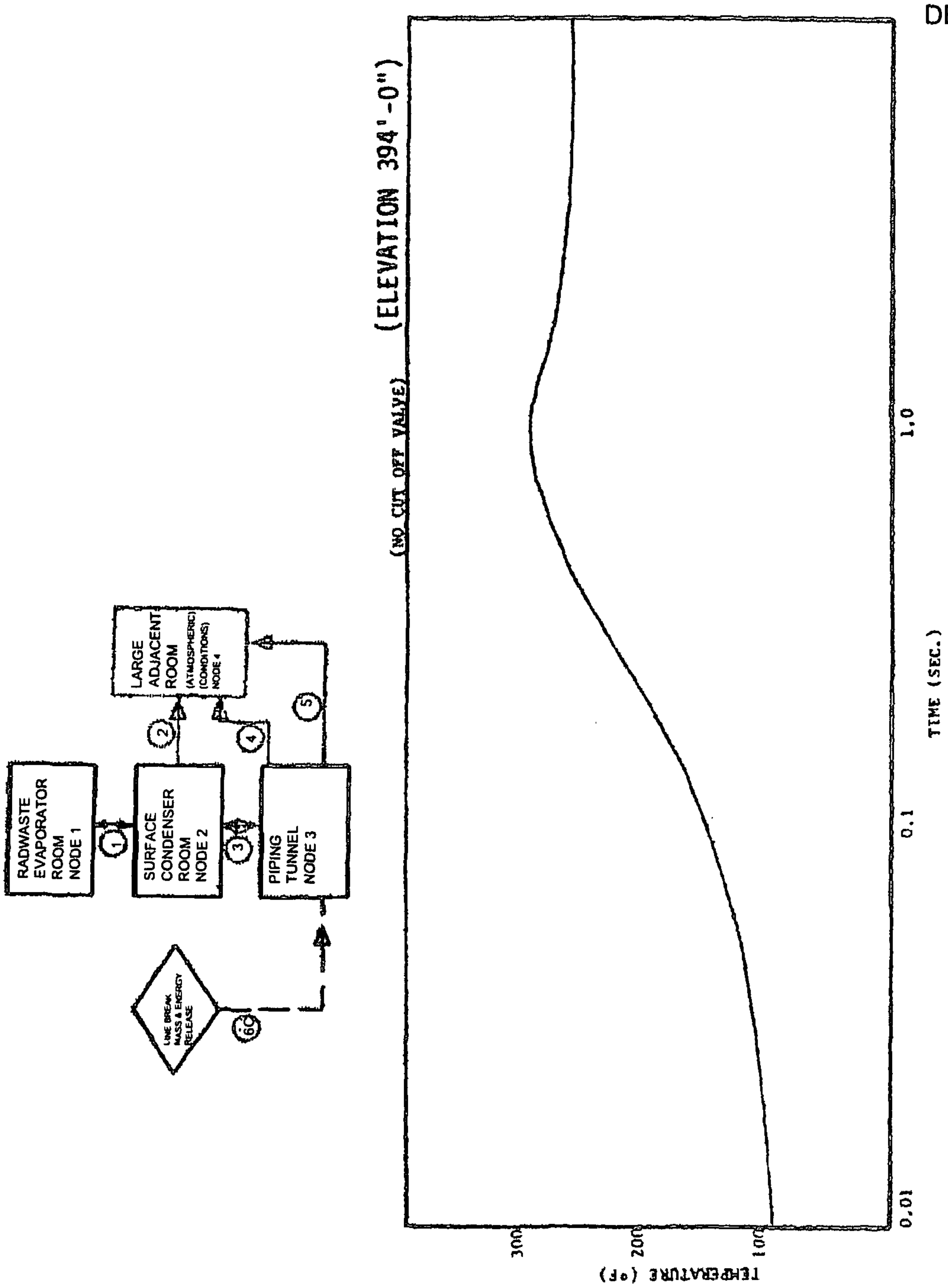

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FIGURE A3.6-25

TEMPERATURE VS. TIME—
AUXILIARY STEAMLINE BREAK IN
PIPE TUNNEL

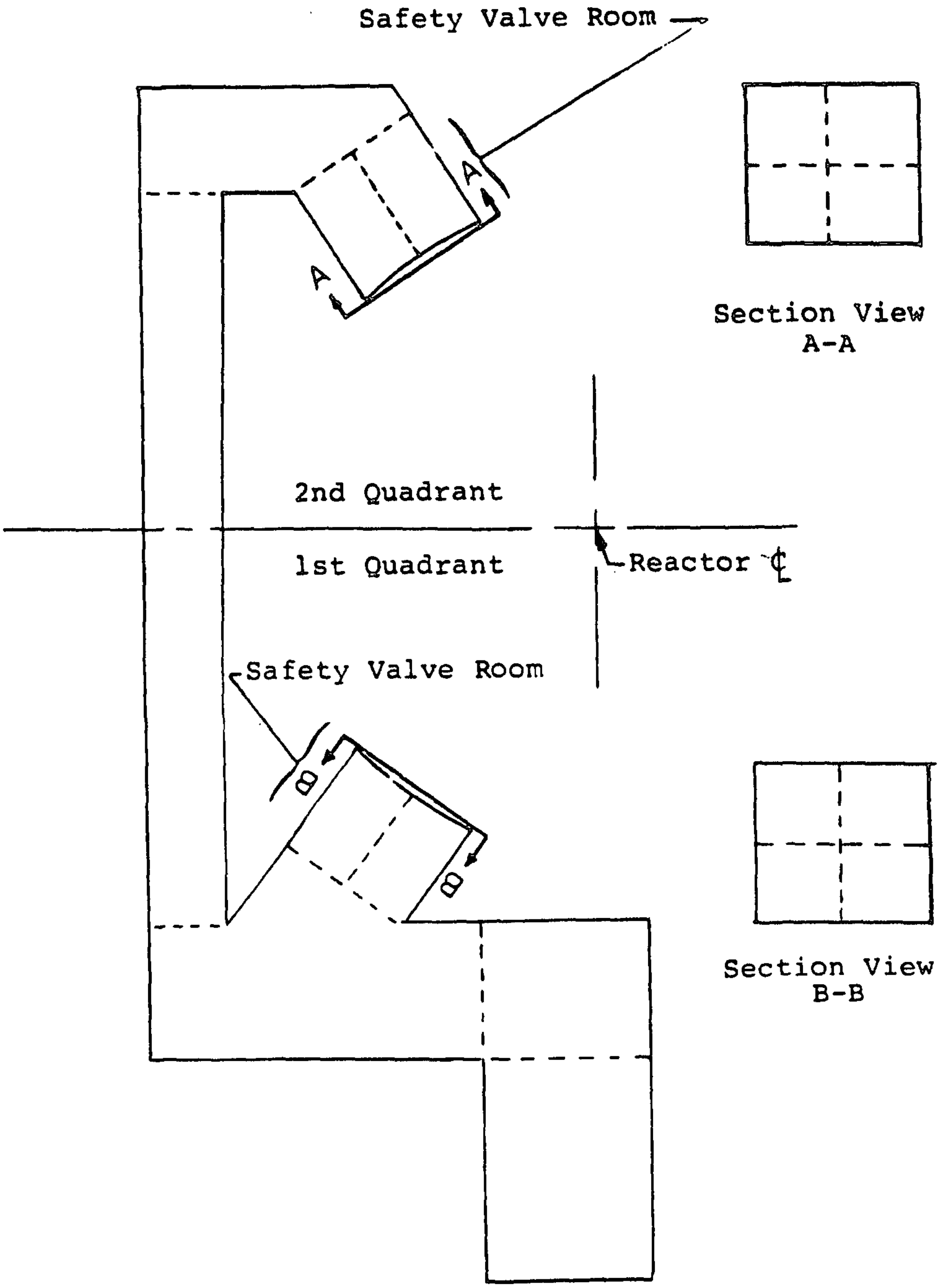

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FIGURE C3.6-1

MAIN STEAM TUNNEL VIEW PLAN

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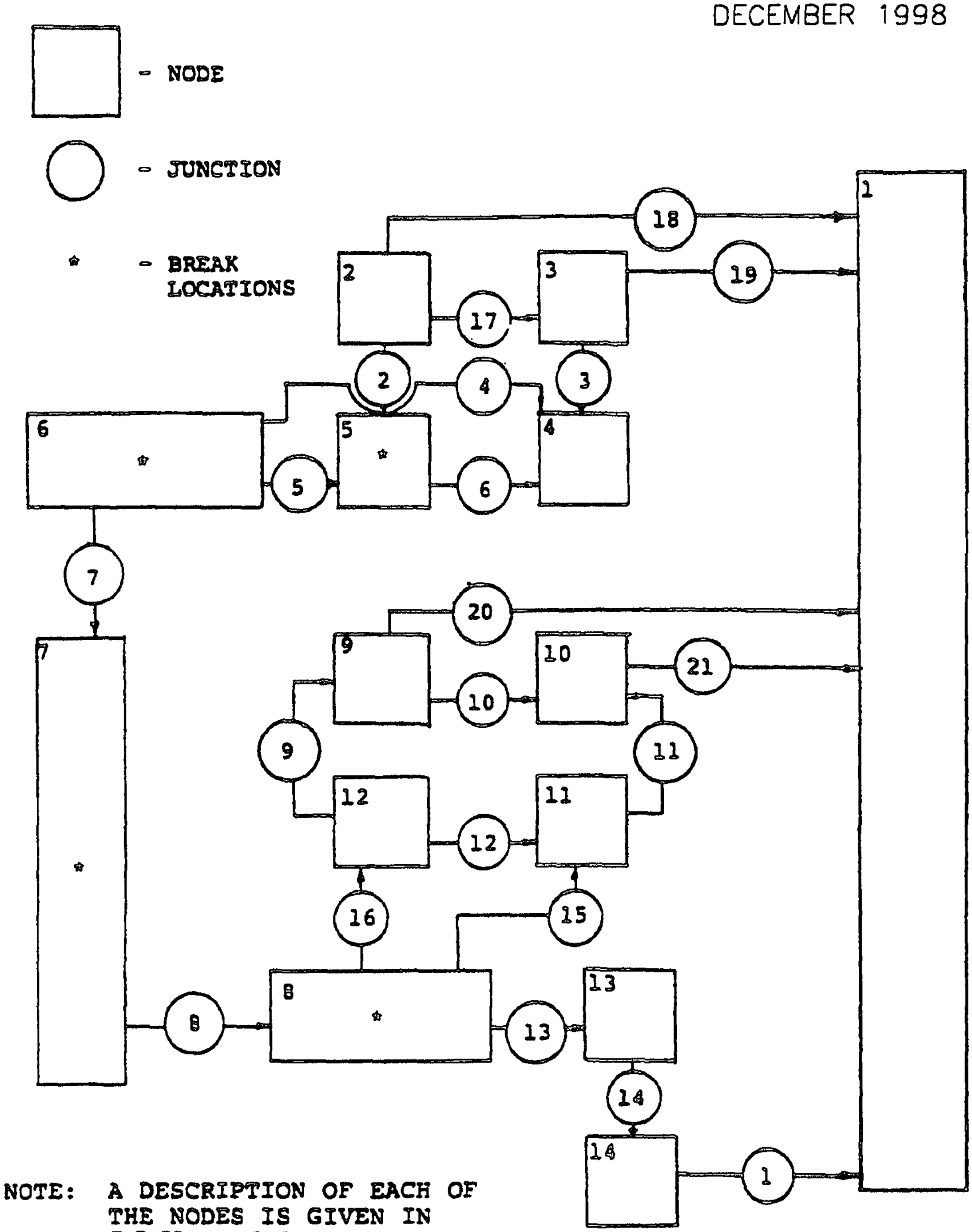

NOTE: A DESCRIPTION OF EACH OF THE NODES IS GIVEN IN TABLES 1 AND 2.

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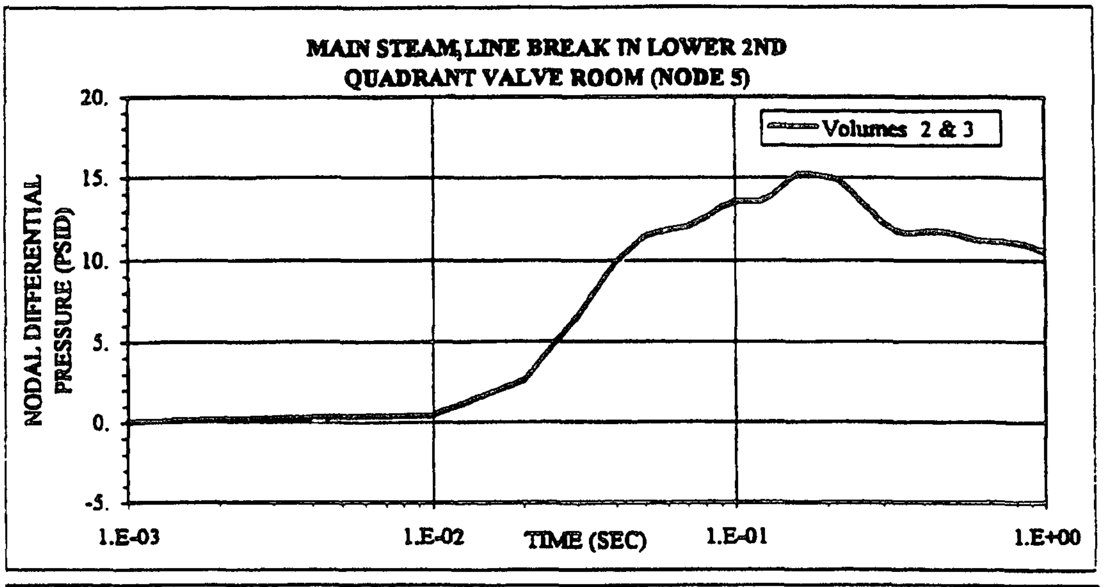
FIGURE C3.6-2

NODALIZATION SCHEMATIC



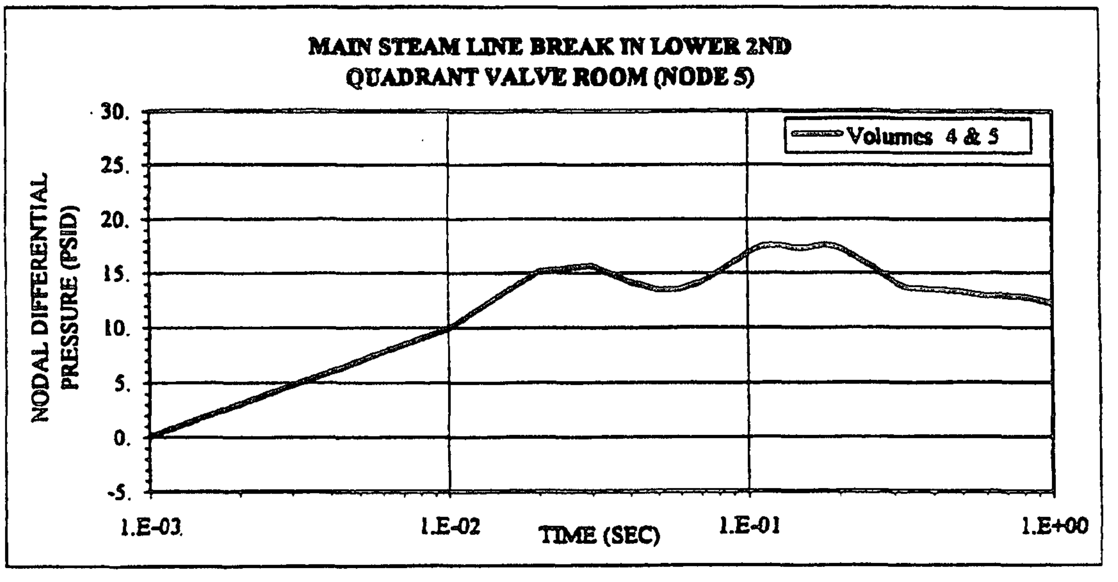


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FIGURE C3.6-3
UNIT 1
DIFFERENTIAL PRESSURE VS. TIME
FOR VOLUMES 2 AND 3 (BREAK IN NODE 5)
AND VOLUMES 4 AND 5 (BREAK IN NODE 5)

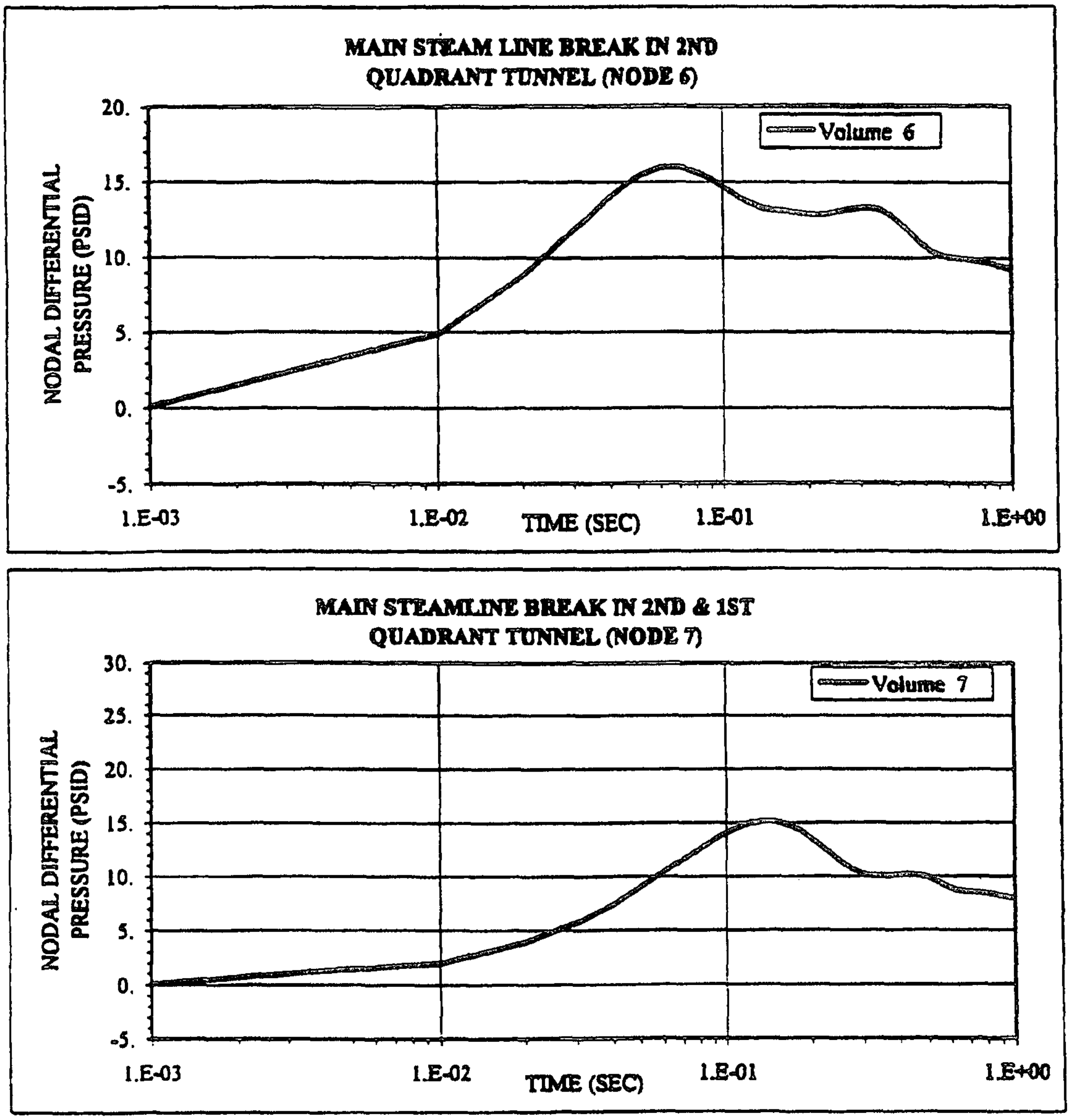

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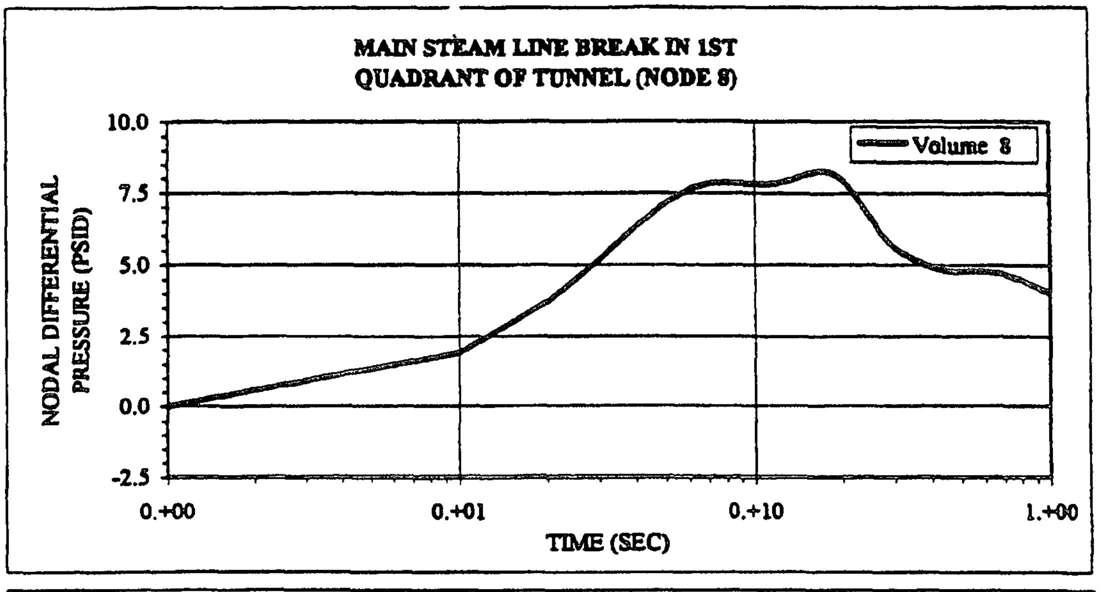
FIGURE C3.6-4

UNIT 1
DIFFERENTIAL PRESSURE VS. TIME
FOR VOLUME 6 (BREAK IN NODE 6)
AND VOLUME 7 (BREAK IN NODE 7)



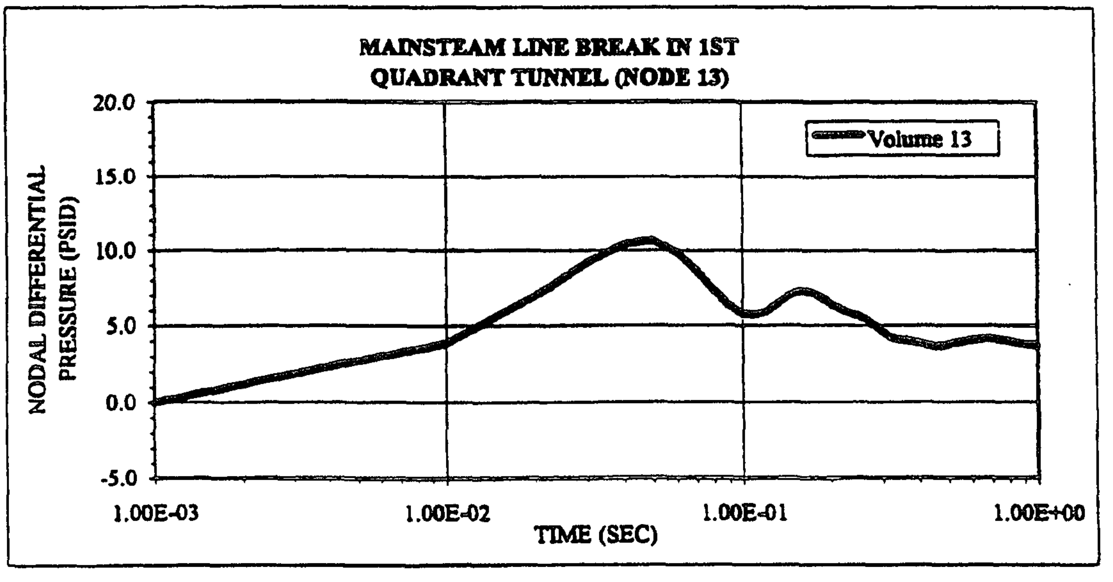


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FIGURE C3.6-5 UNIT 1 DIFFERENTIAL PRESSURE VS. TIME FOR VOLUME 8 (BREAK IN NODE 8) AND VOLUME 13 (BREAK IN NODE 13)

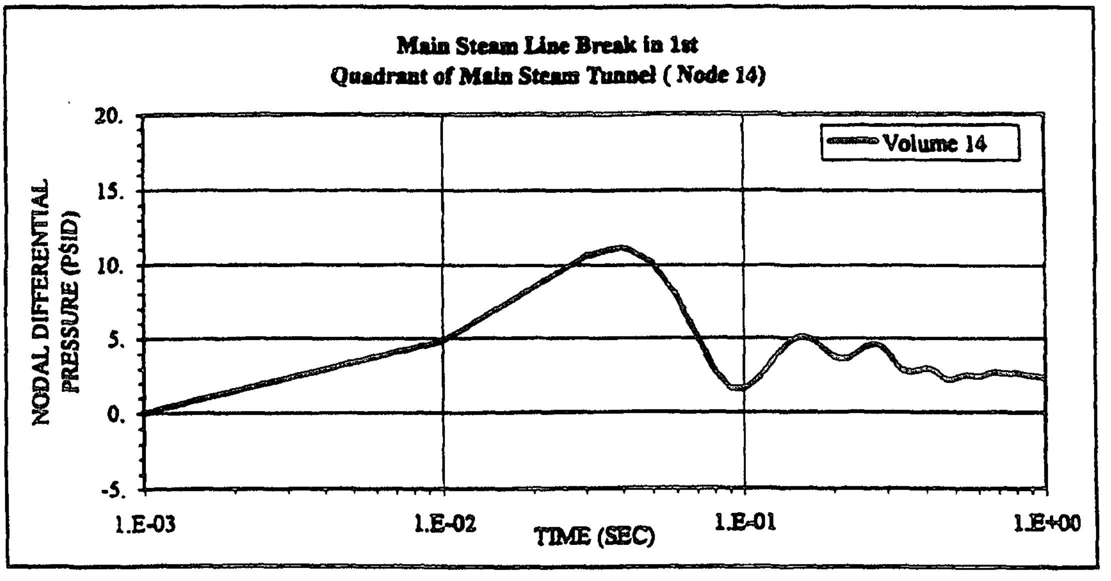

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FIGURE C3.6-6

UNIT 1
DIFFERENTIAL PRESSURE VS. TIME
FOR VOLUME 14 (BREAK IN NODE 14)

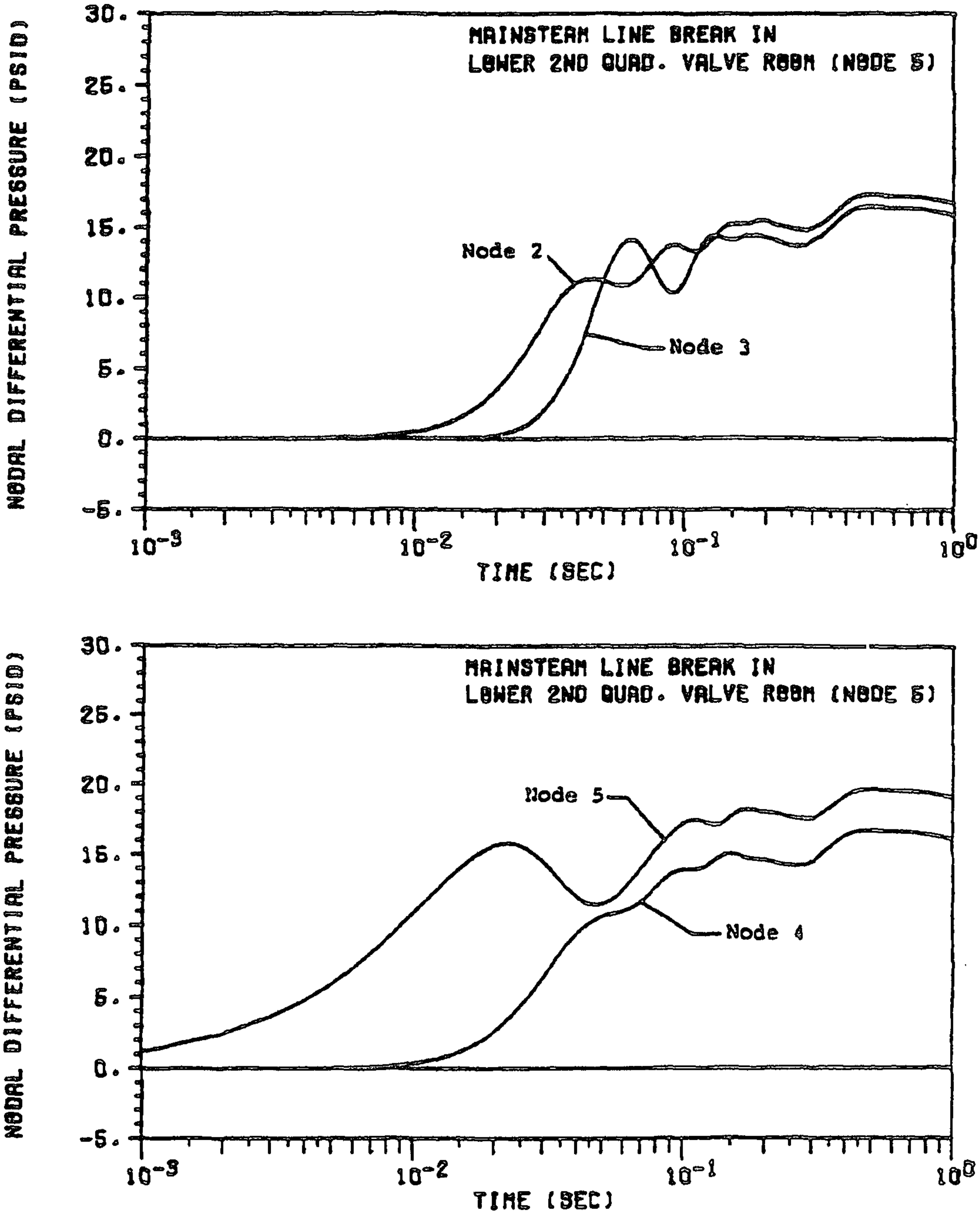

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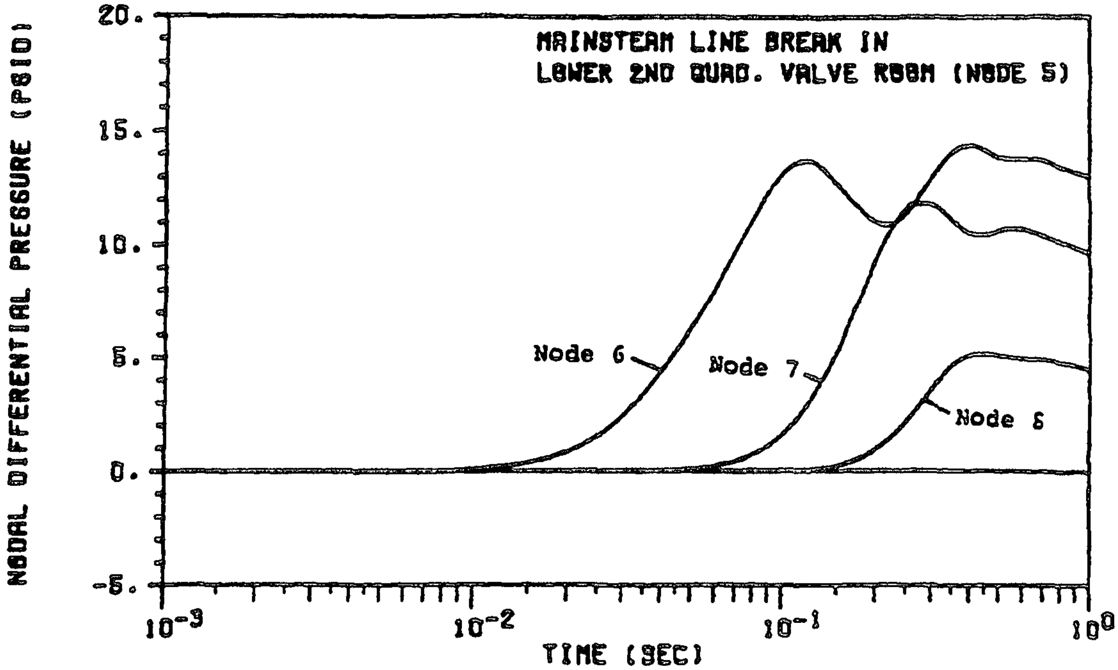
FIGURE C3.6-7 UNIT 2 DIFFERENTIAL PRESSURE VS. TIME FOR NODES 2,3,4, AND 5 (BREAK IN NODES 5)



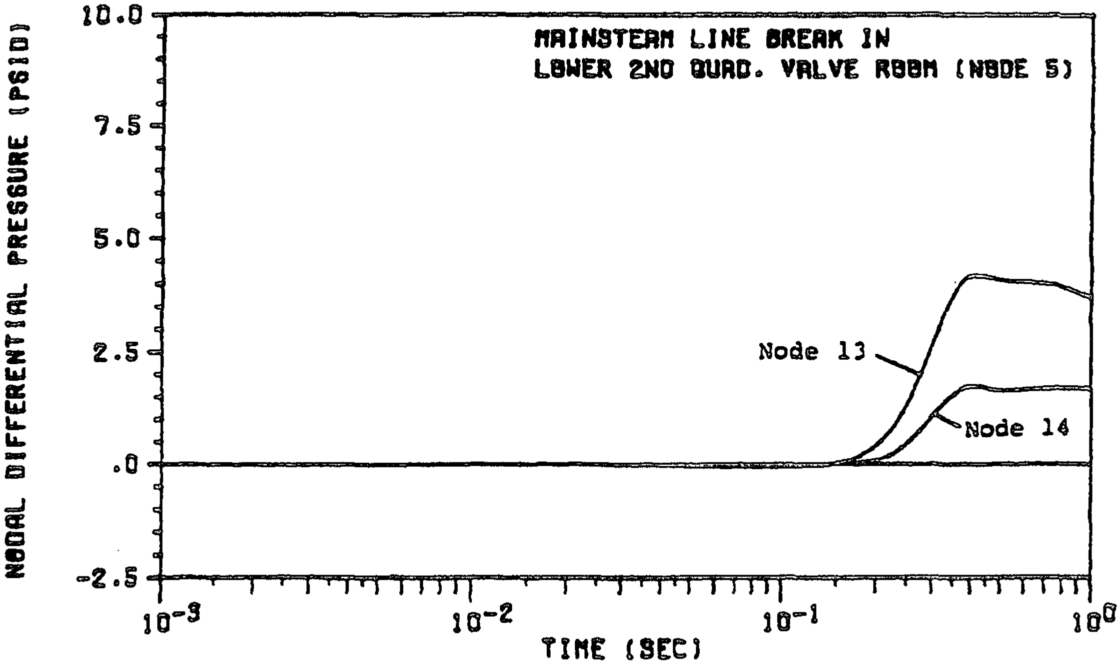


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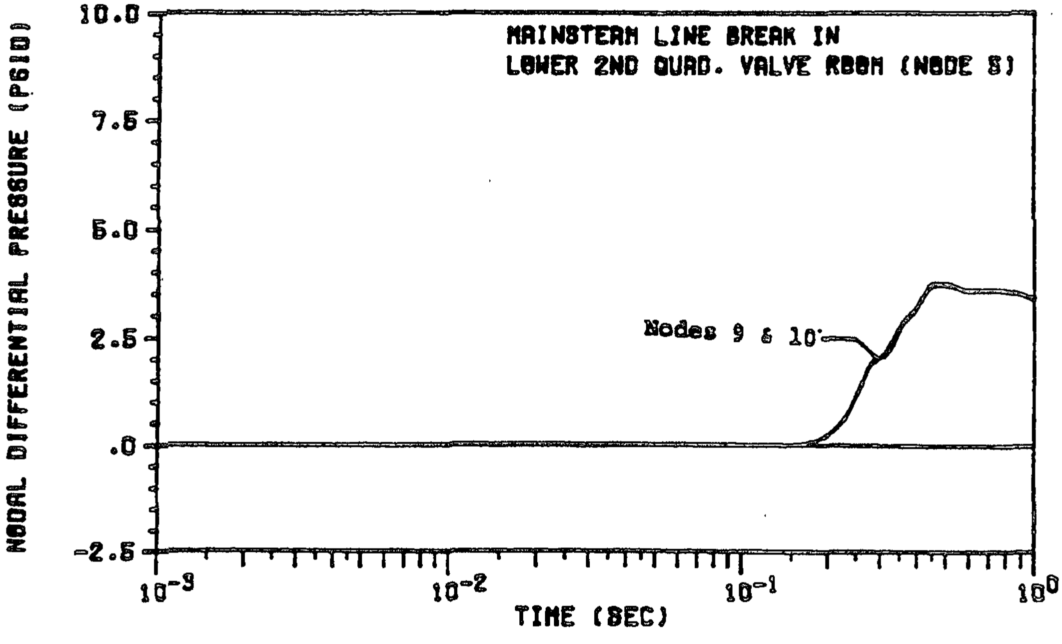
FIGURE C3.6-8 UNIT 2 DIFFERENTIAL PRESSURE VS. TIME FOR NODES 6,7,8,13 AND 14 (BREAK IN NODES 5)



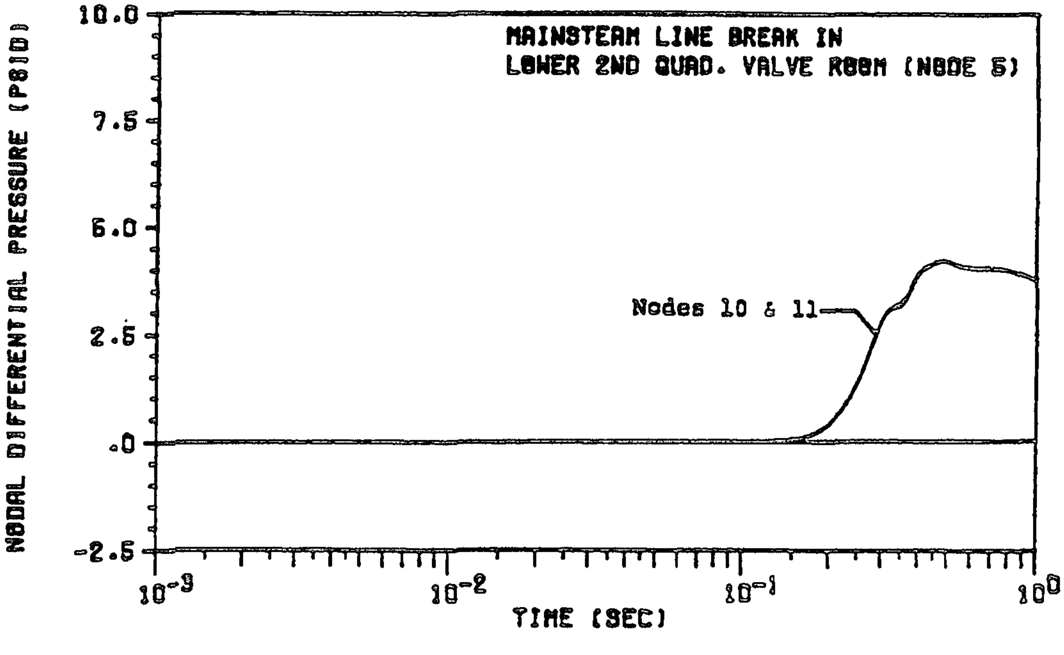


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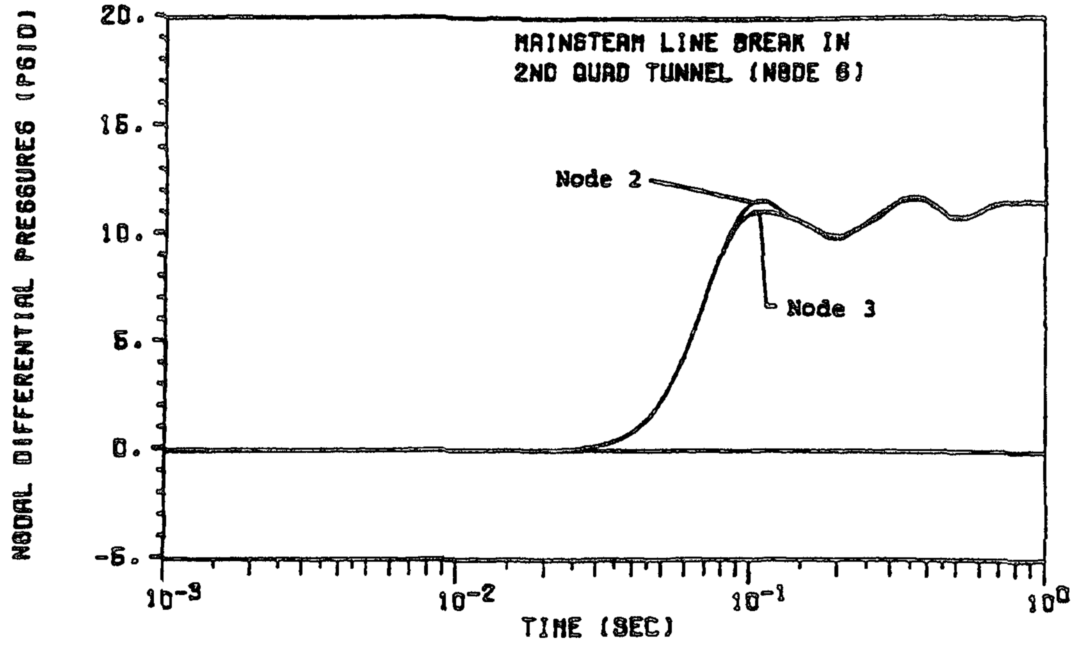
FIGURE C3.6-9
UNIT 2
DIFFERENTIAL PRESSURE VS. TIME
FOR NODES 9,10,11 AND 12
(BREAK IN NODE 5)



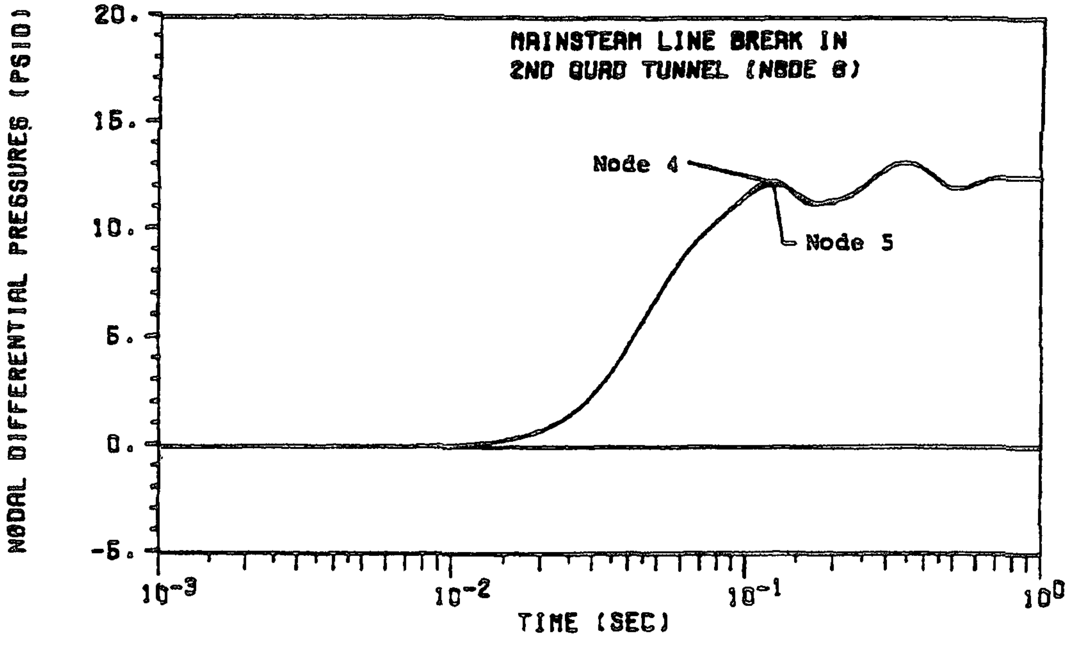


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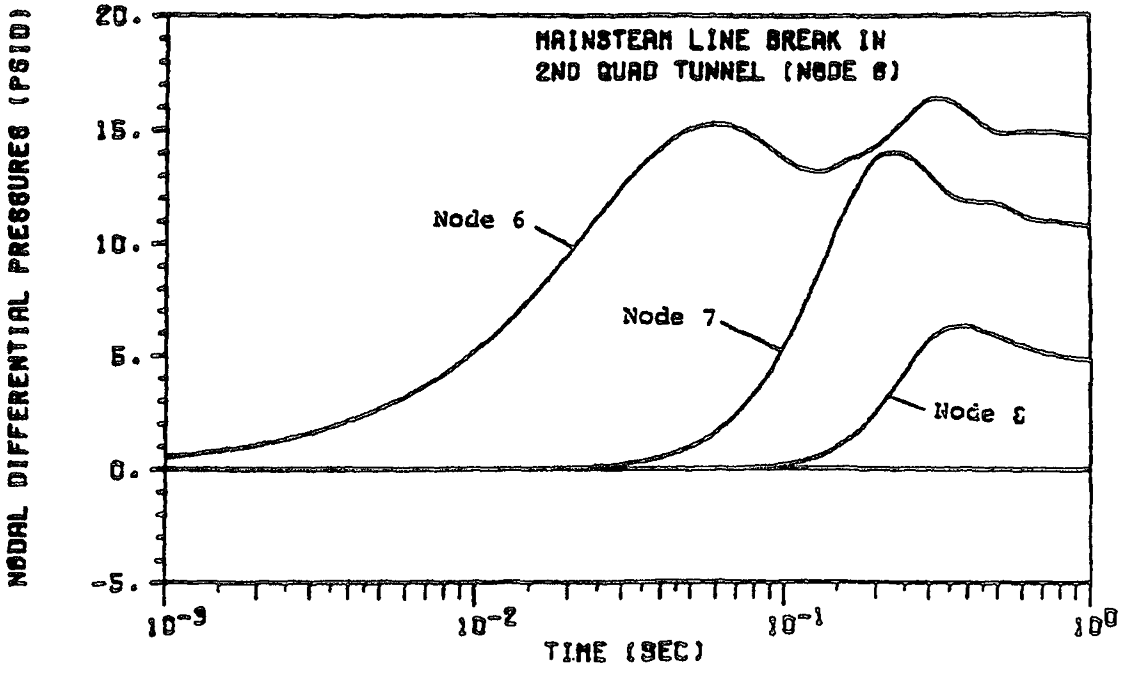
FIGURE C3.6-10
UNIT 2
DIFFERENTIAL PRESSURE VS. TIME
FOR NODES 2,3,4 AND 5
(BREAK IN NODE 6)



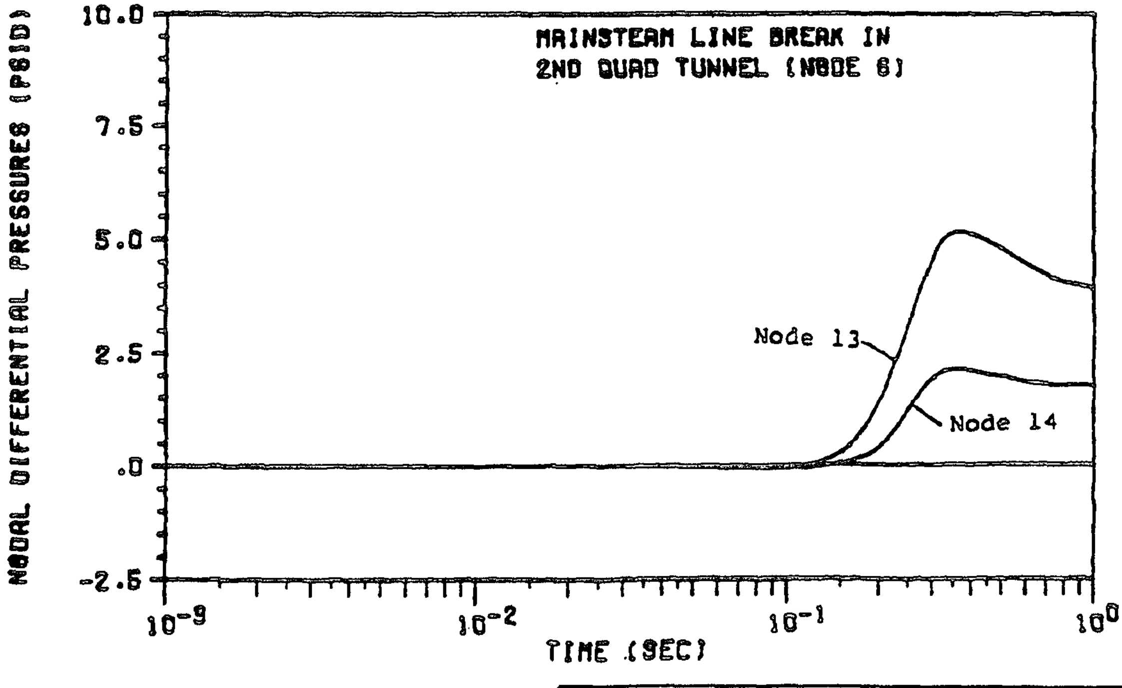


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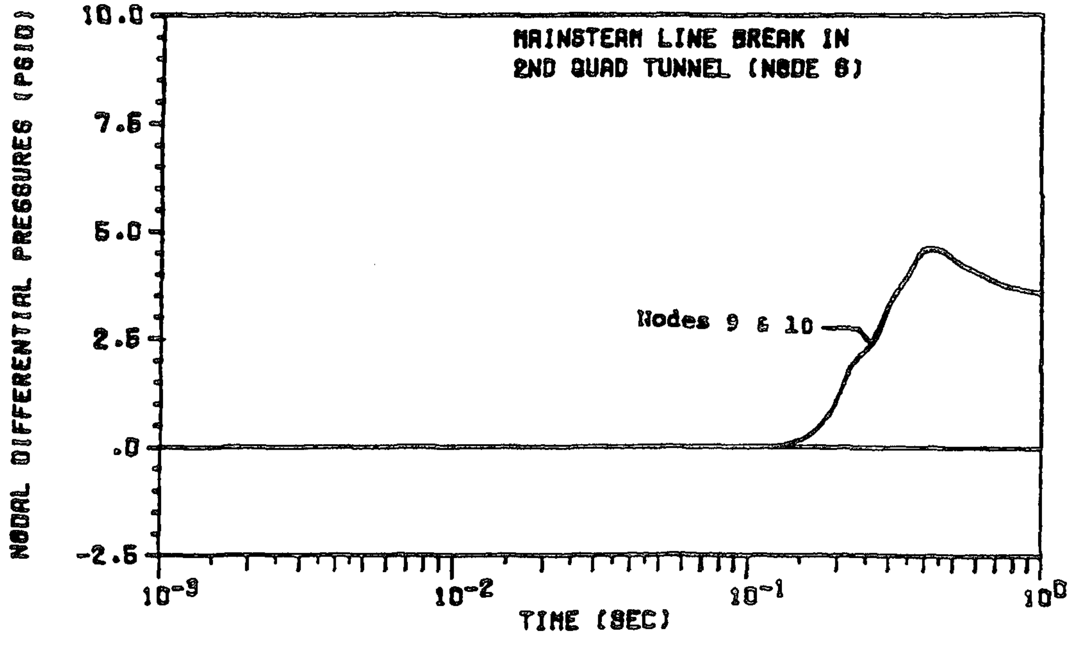
FIGURE C3.6-11 UNIT 2 DIFFERENTIAL PRESSURE VS. TIME FOR NODES 6,7,8,13 AND 14 (BREAK IN NODE 6)



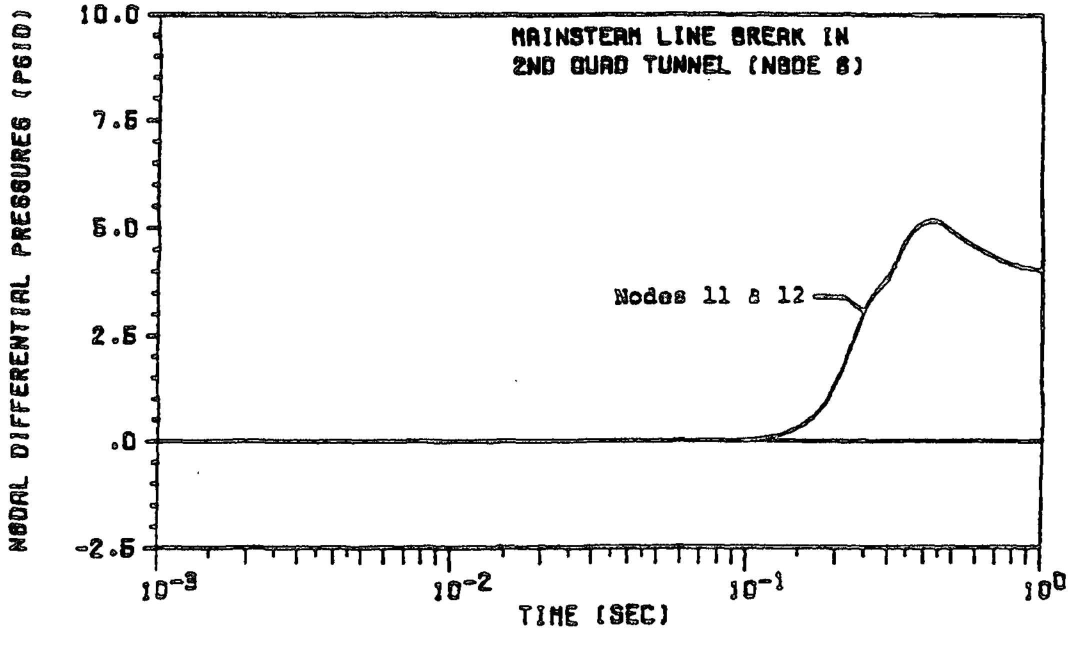


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FIGURE C3.6-12
UNIT 2
DIFFERENTIAL PRESSURE VS. TIME
FOR NODES 9,10,11 AND 12
(BREAK IN NODE 6)

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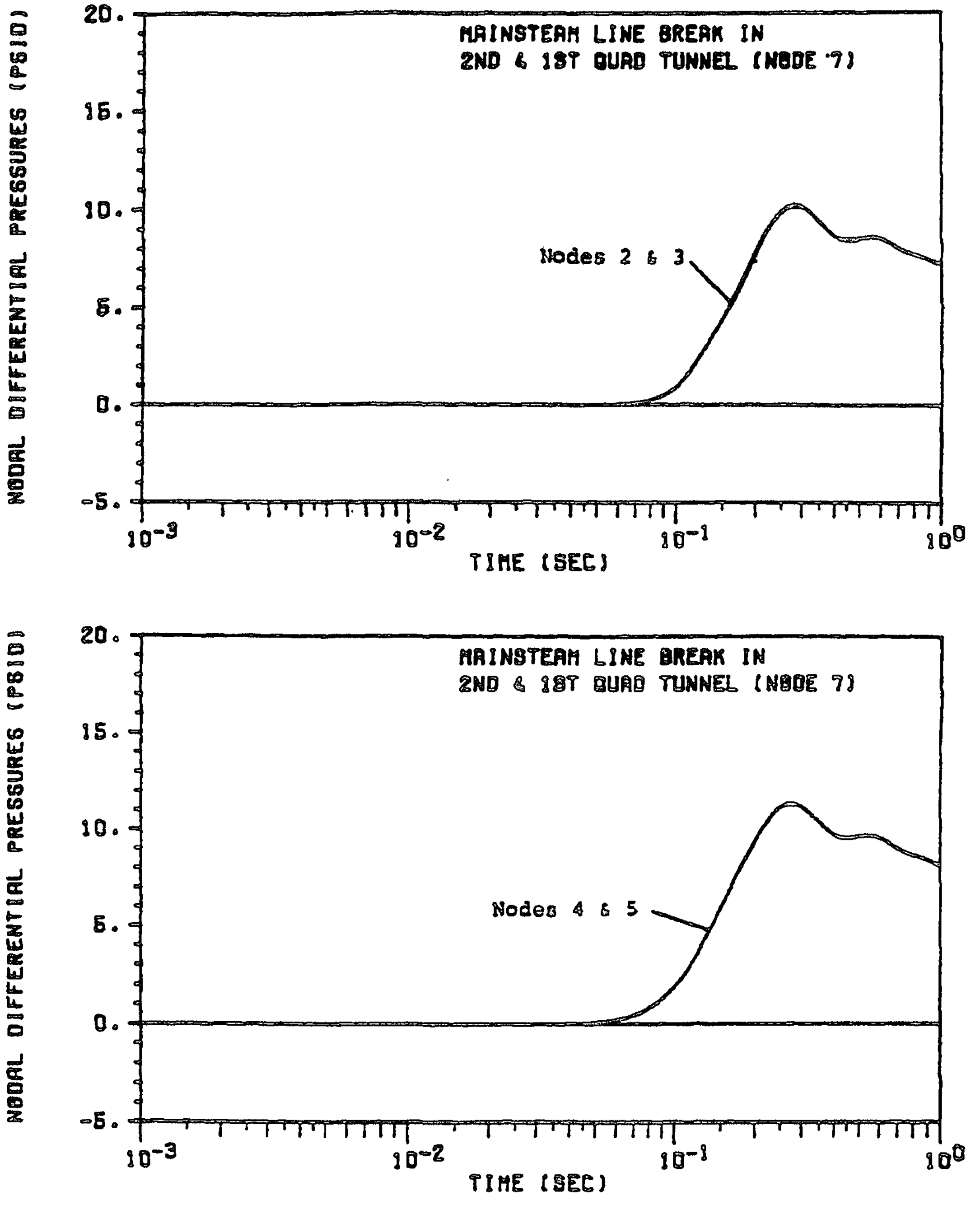

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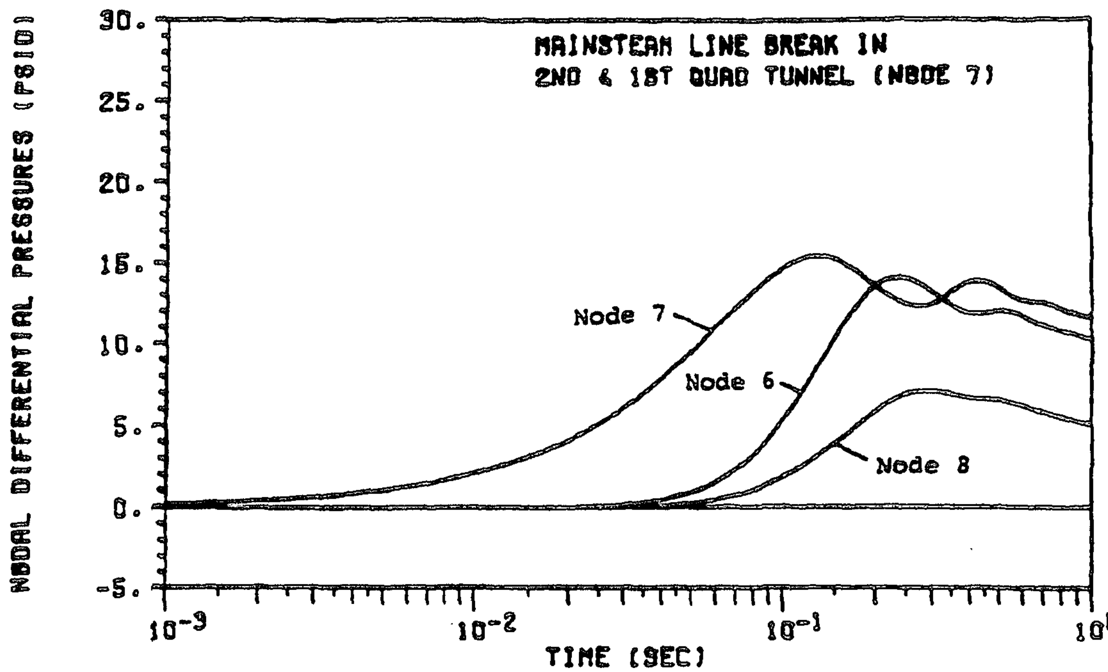
FIGURE C3.6-13
UNIT 2
DIFFERENTIAL PRESSURE VS. TIME
FOR NODES 2,3,4 AND 5
(BREAK IN NODE 7)



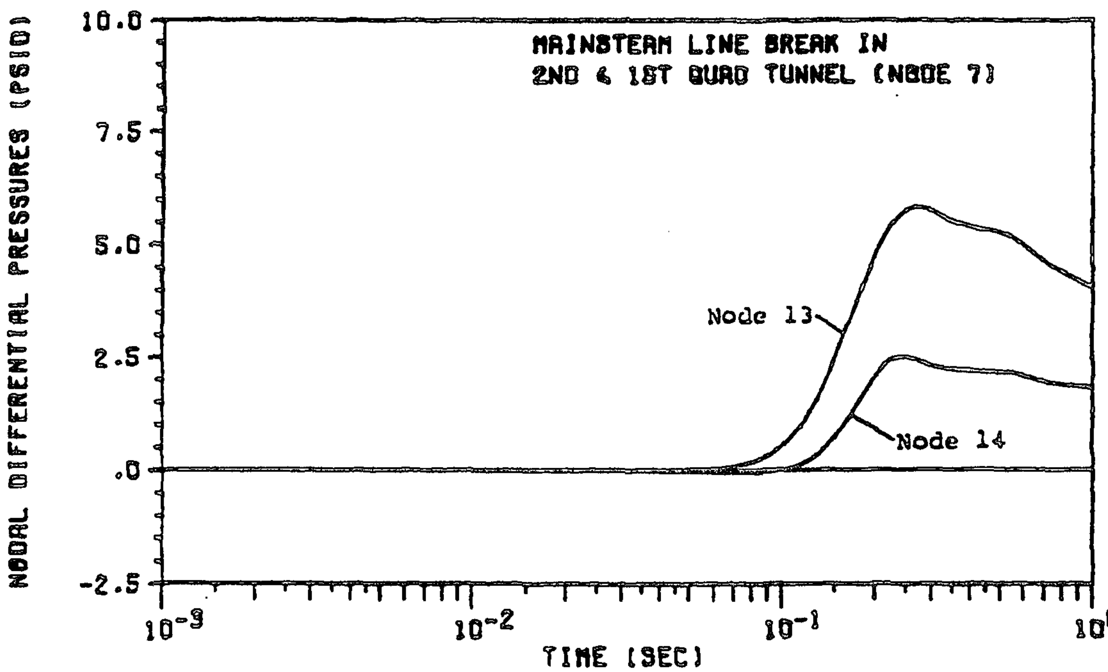


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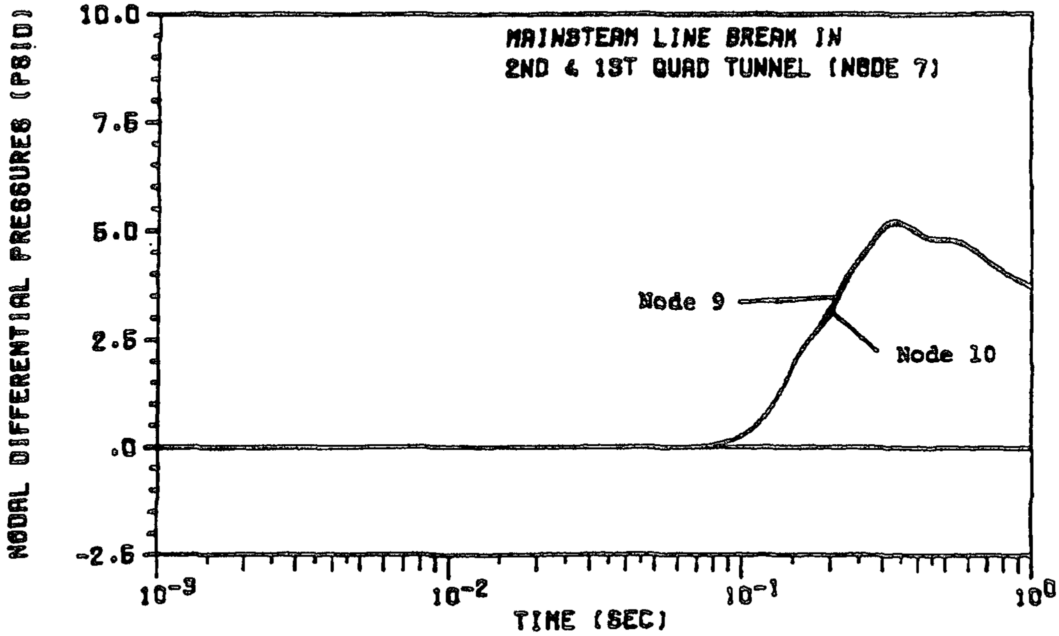
FIGURE C3.6-14
UNIT 2
DIFFERENTIAL PRESSURE VS. TIME
FOR NODES 6,7,8,13 AND 14
(BREAK IN NODE 7)



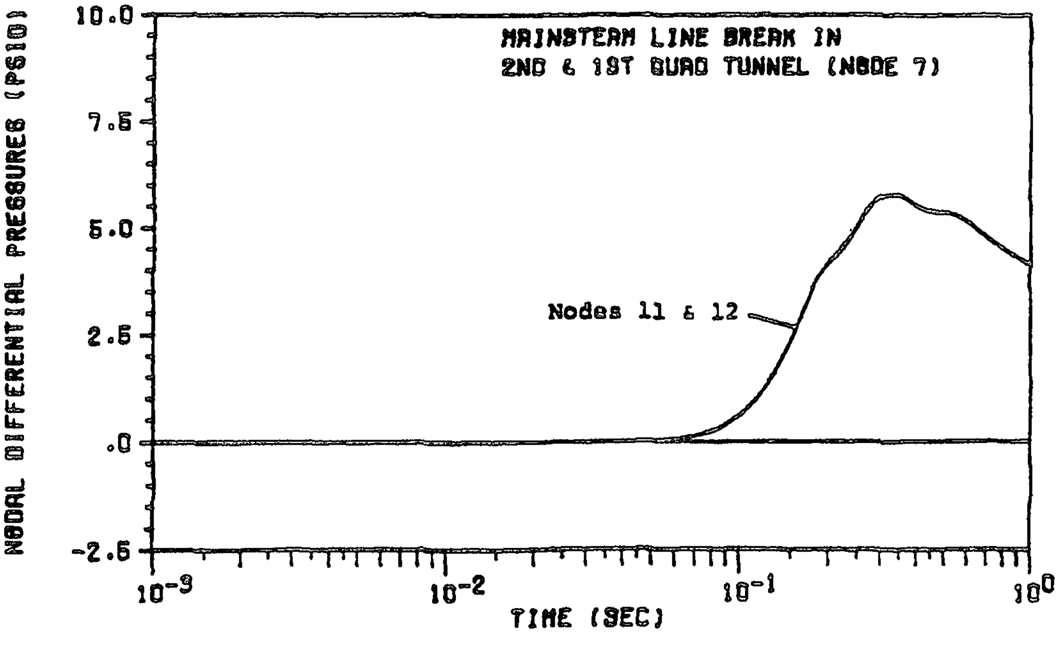


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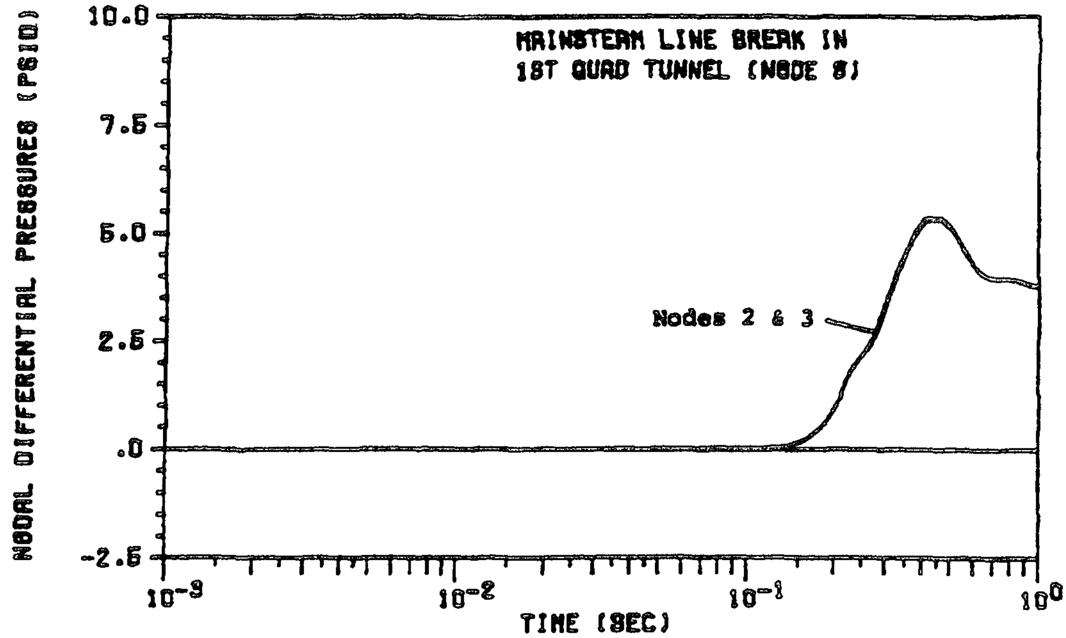
FIGURE C3.6-15
UNIT 2
DIFFERENTIAL PRESSURE VS. TIME
FOR NODES 9,10,11 AND 12
(BREAK IN NODE 7)



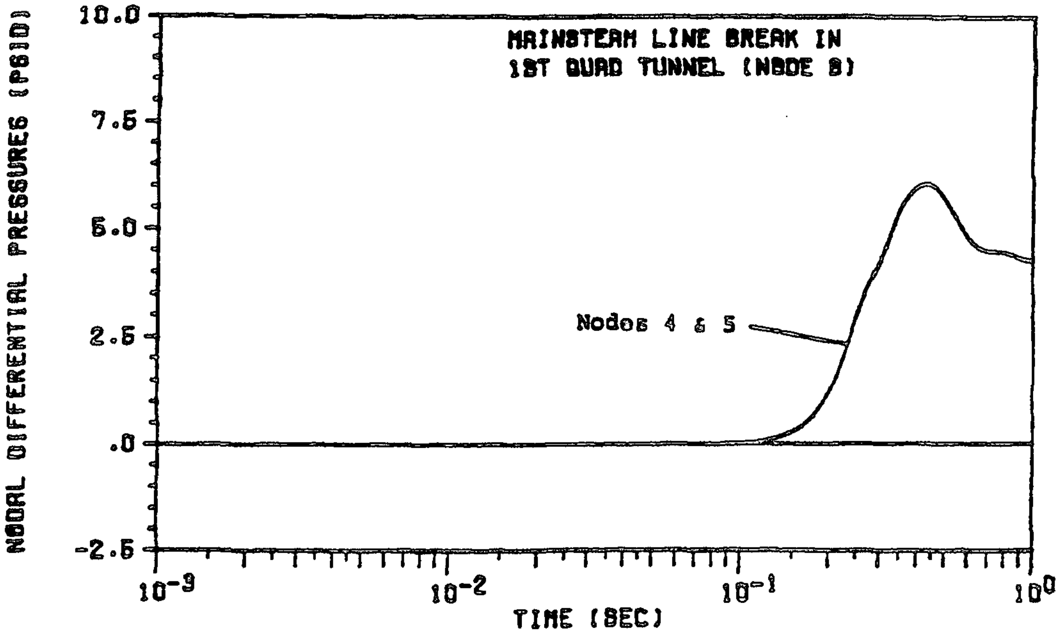


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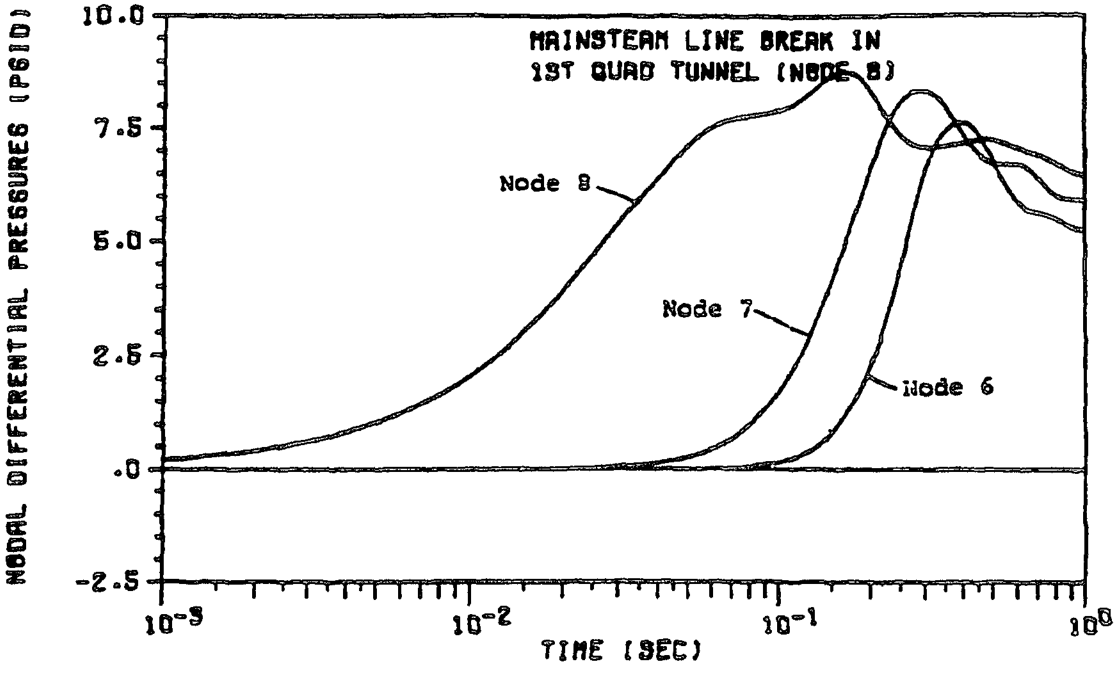
FIGURE C3.6-16 UNIT 2 DIFFERENTIAL PRESSURE VS. TIME FOR NODES 2,3,4 AND 5 (BREAK IN NODES 8)



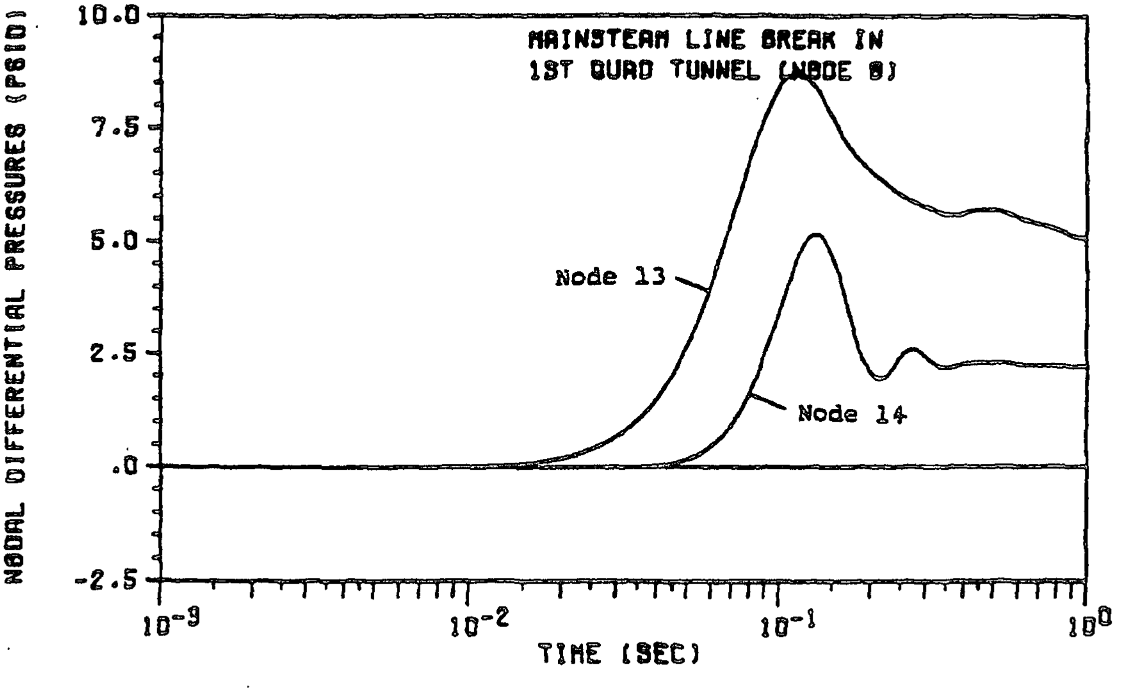


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FIGURE C3.6-17 UNIT 2 DIFFERENTIAL PRESSURE VS. TIME FOR NODES 6,7,8,13 AND 14 (BREAK IN NODE 8)

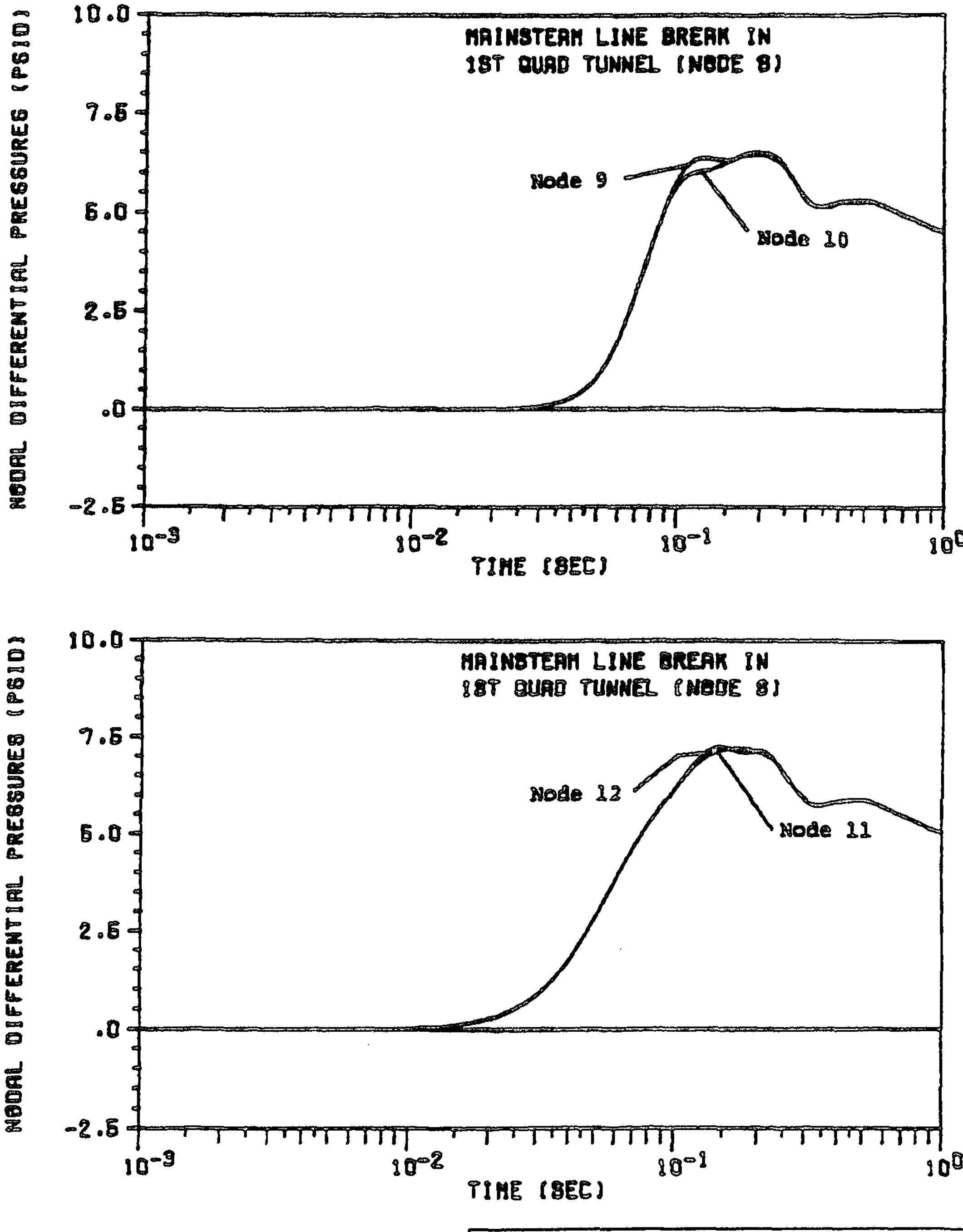

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FIGURE C3.6-18 UNIT 2 DIFFERENTIAL PRESSURE VS. TIME FOR NODES 9,10,11, AND 12 (BREAK IN NODE 8)

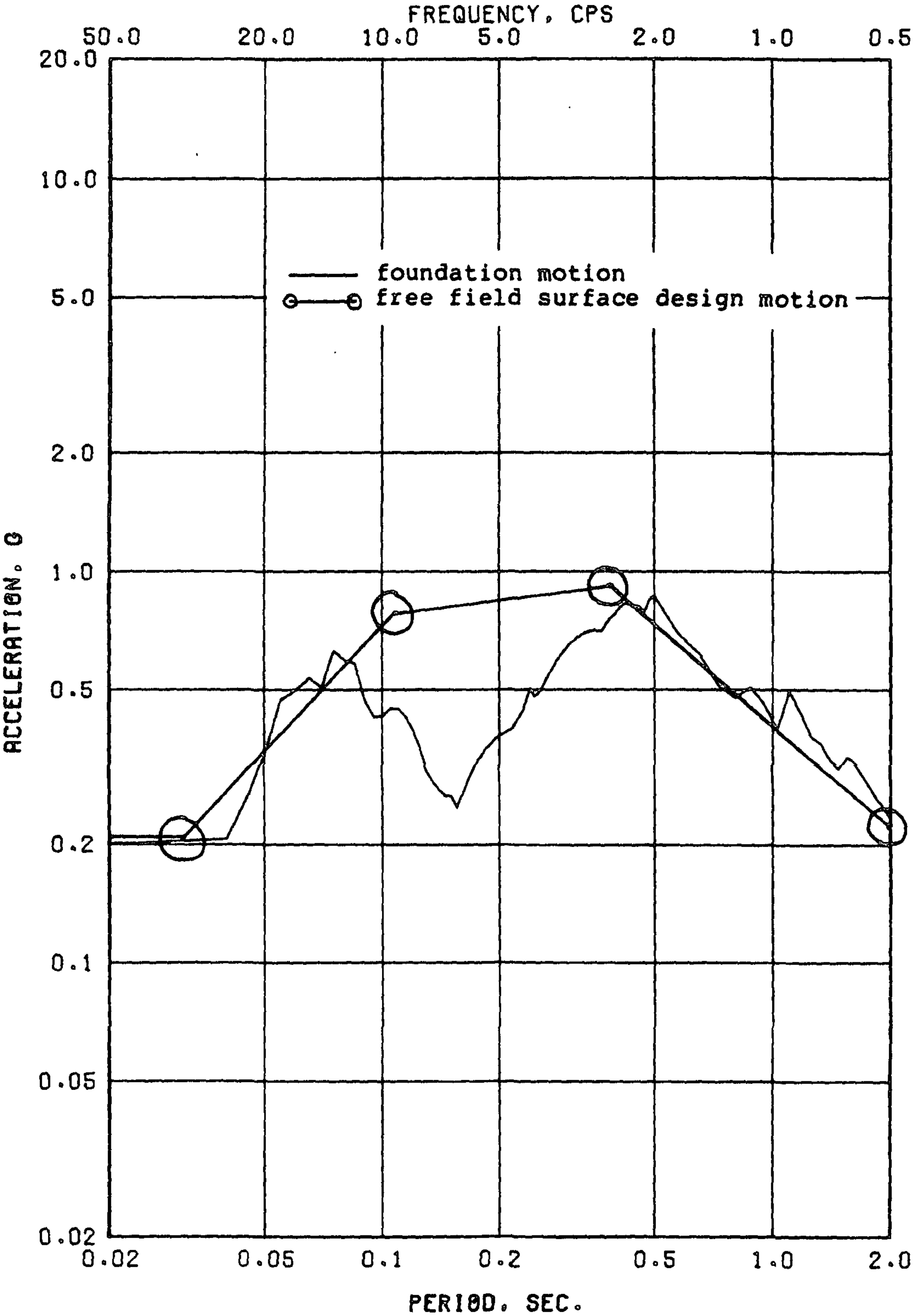

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FIGURE 3.7-1

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM SSE (2% DAMPING)

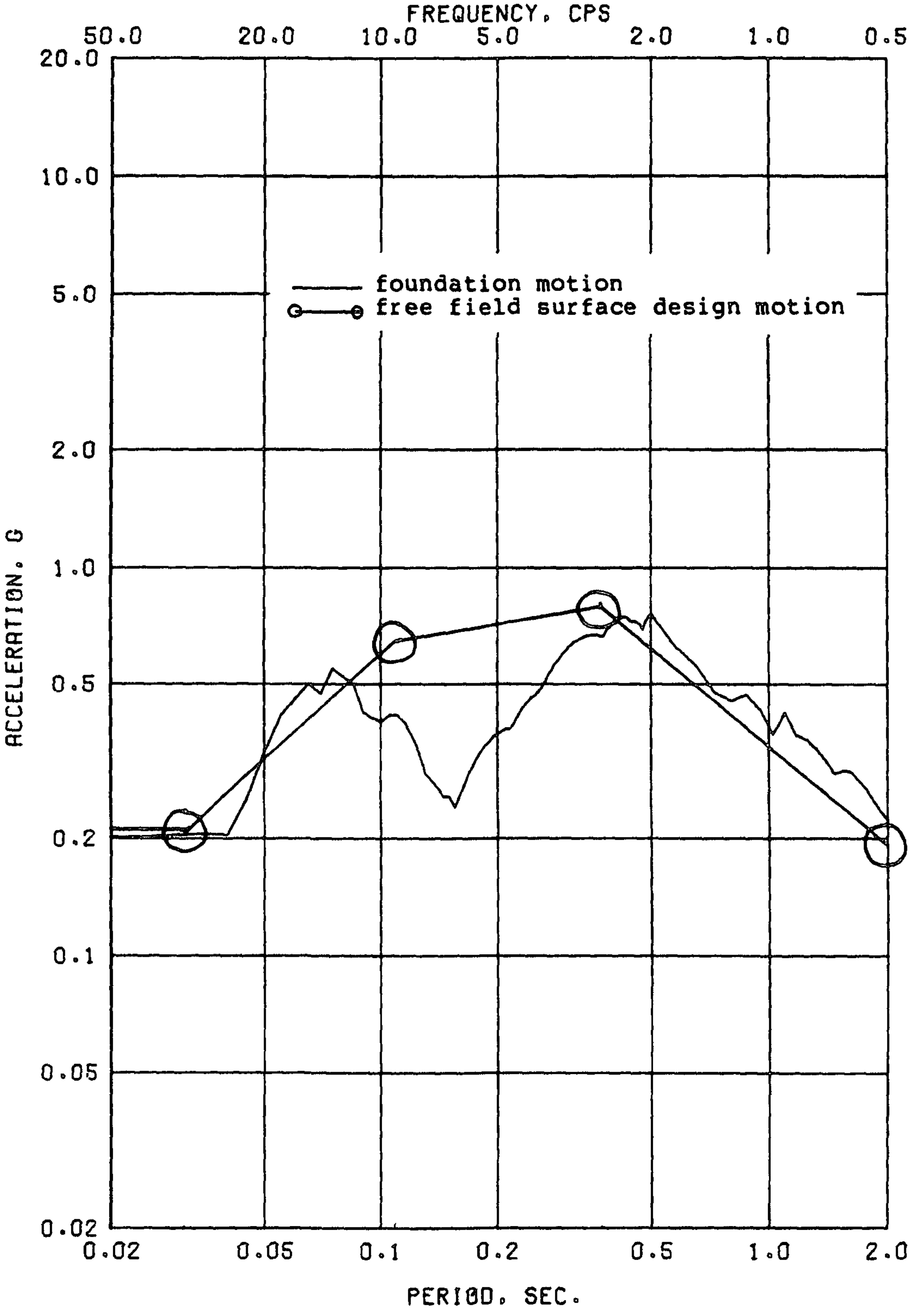

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FIGURE 3.7-2

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM SSE (3% DAMPING)

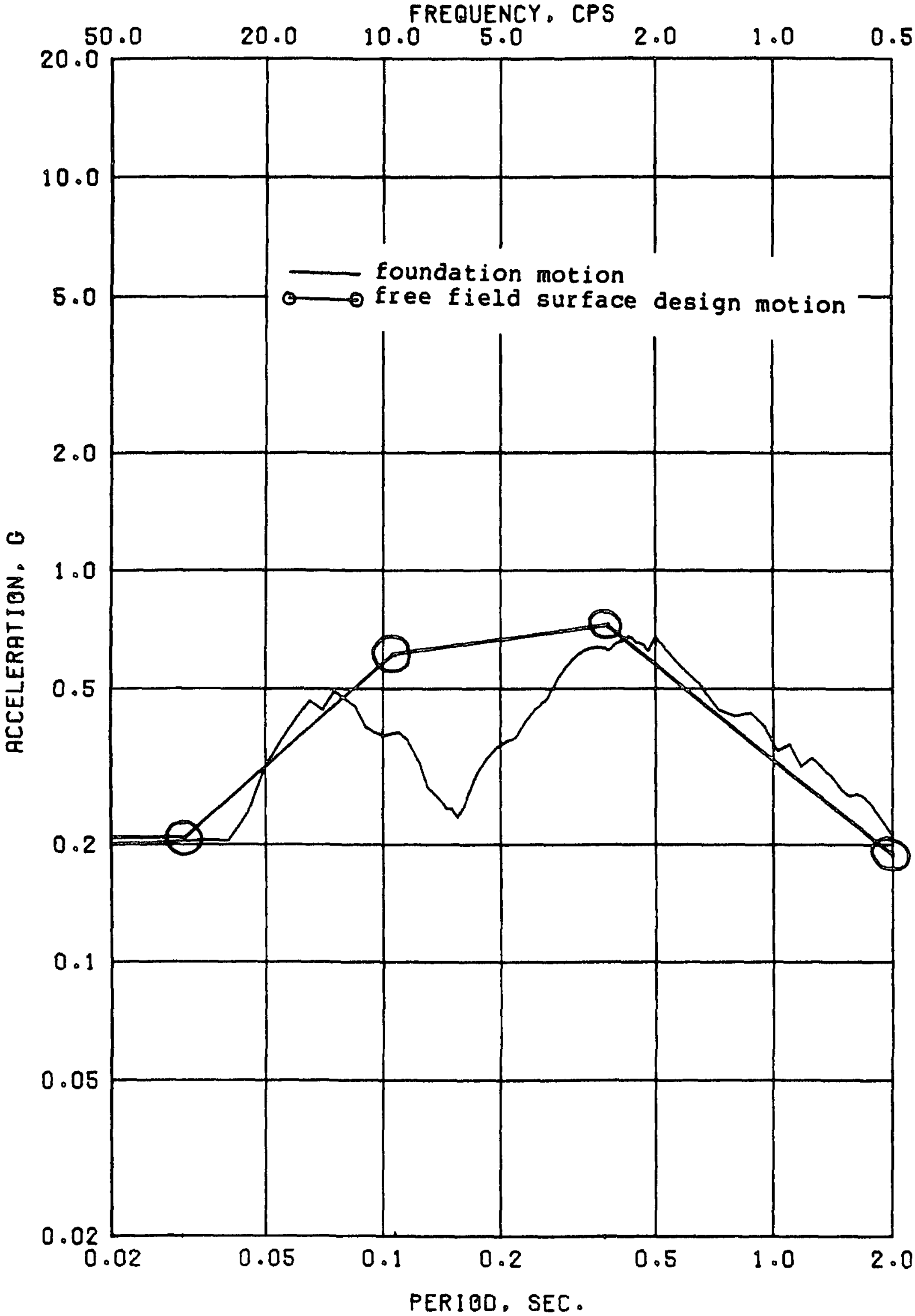

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FIGURE 3.7-3

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM SSE (4% DAMPING)

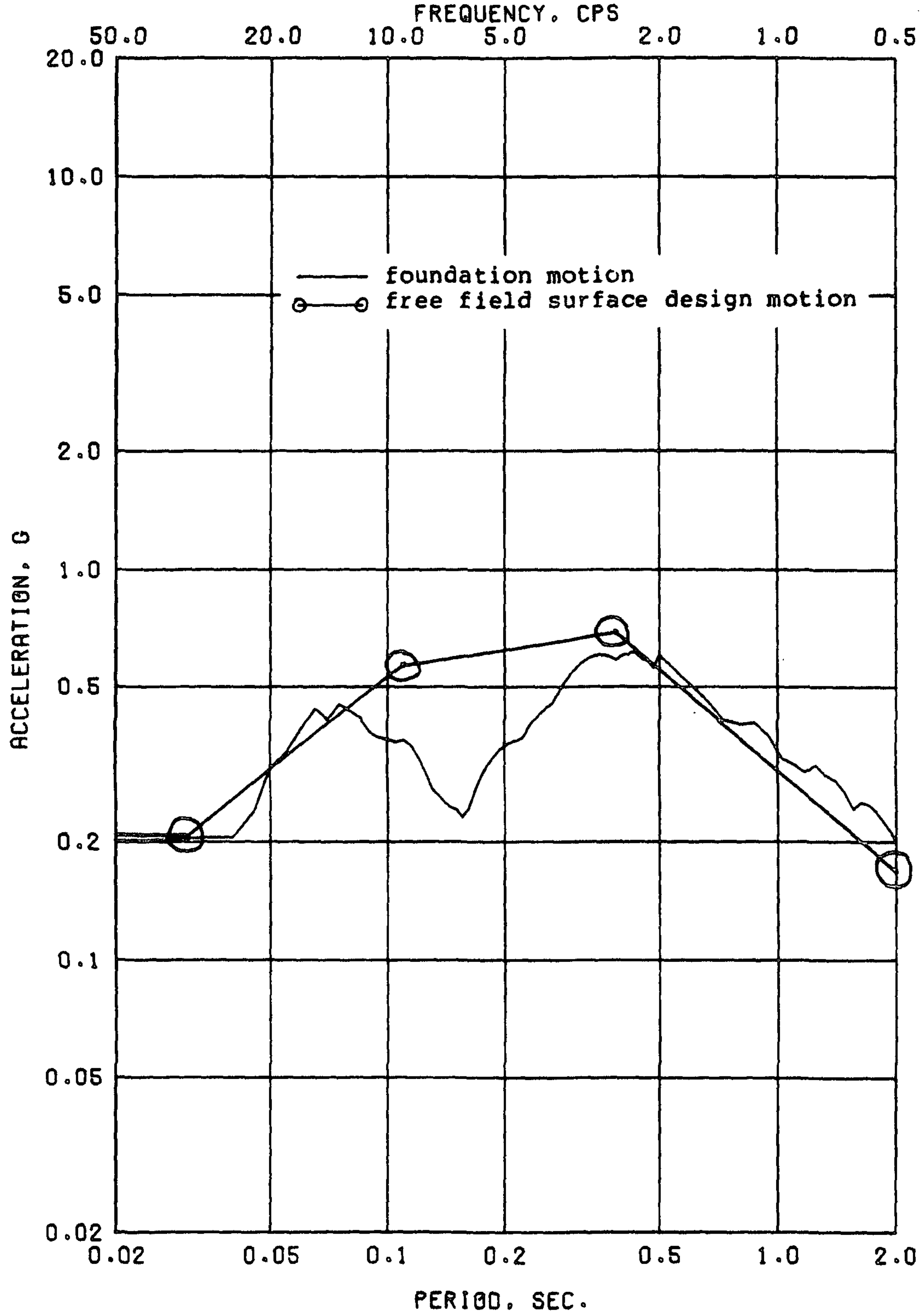

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FIGURE 3.7-4

HORIZONTAL FOUNDATION RESPONSE SPECTRUM SSE (5% DAMPING)

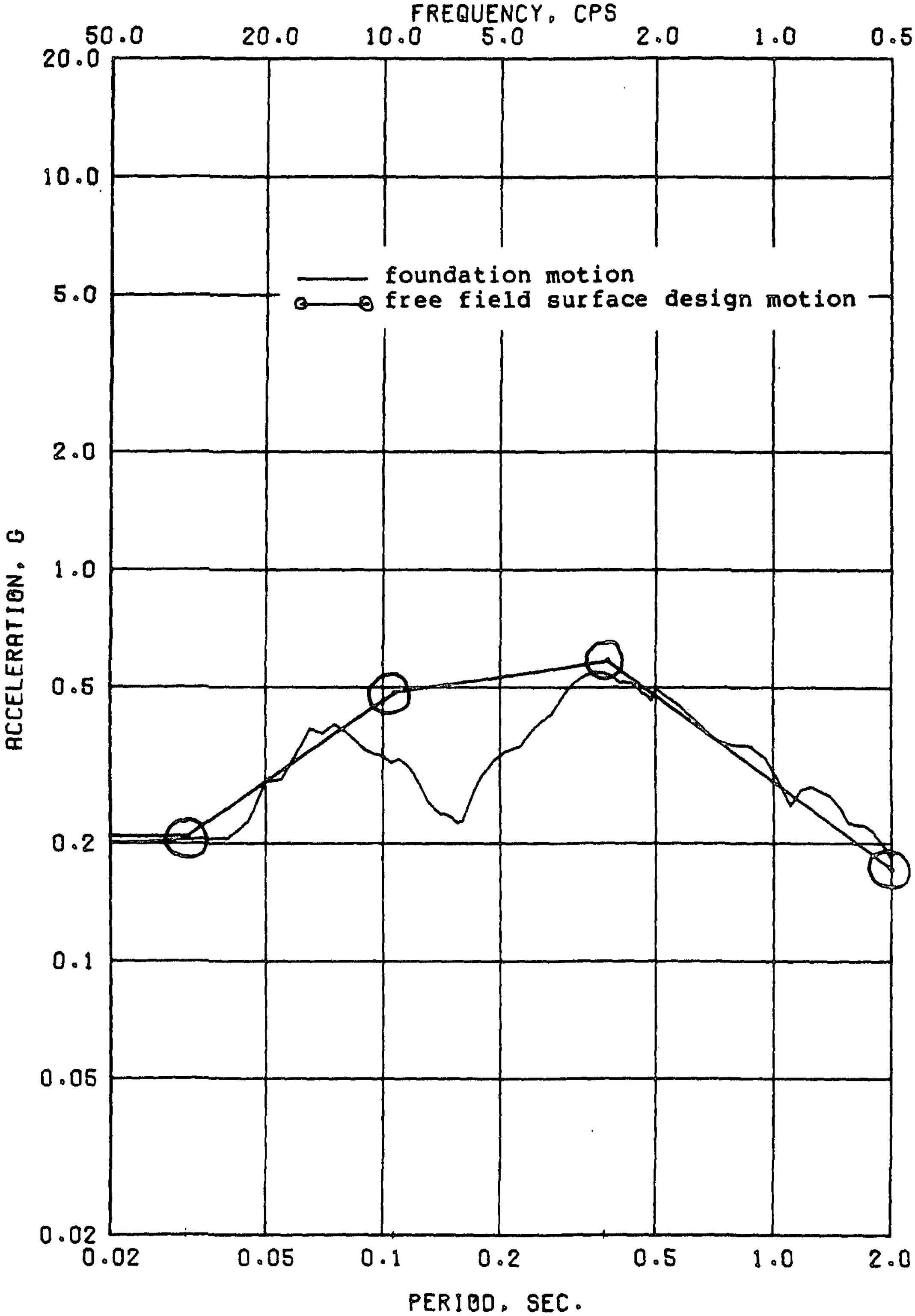

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FIGURE 3.7-5

HORIZONTAL FOUNDATION RESPONSE SPECTRUM SSE (7% DAMPING)

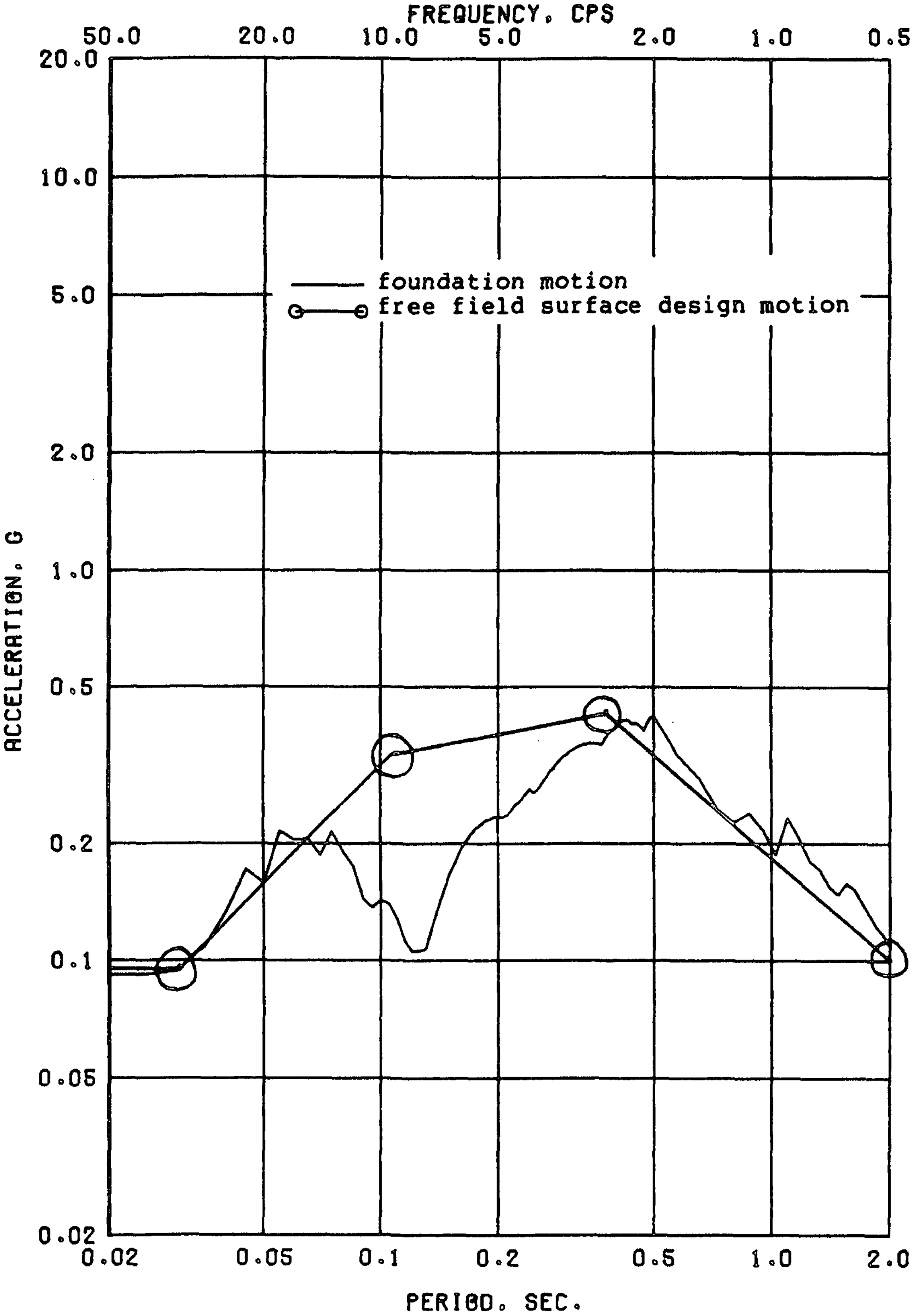

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FIGURE 3.7-6

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM OBE (2% DAMPING)

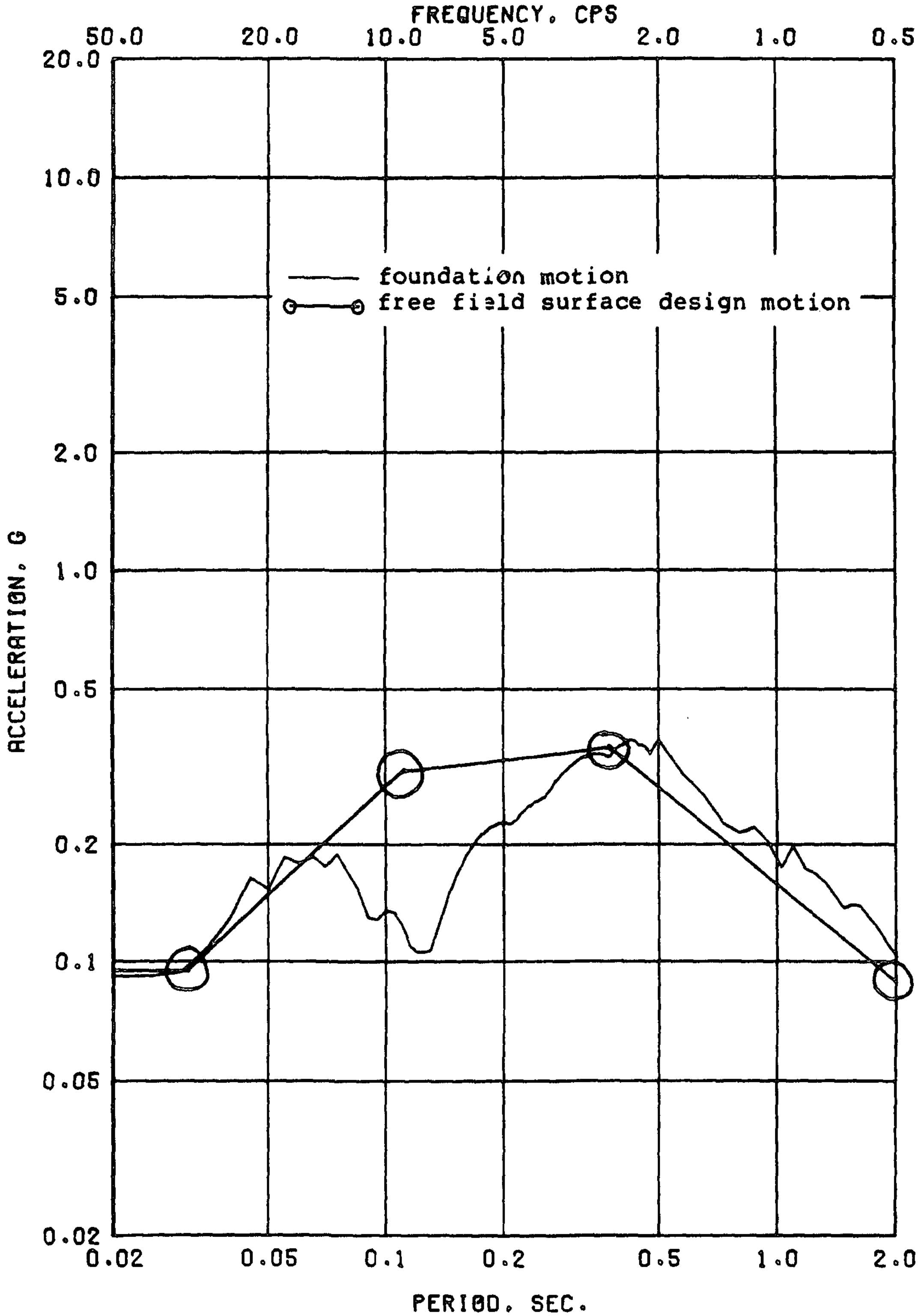

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FIGURE 3.7-7

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM OBE (3% DAMPING)

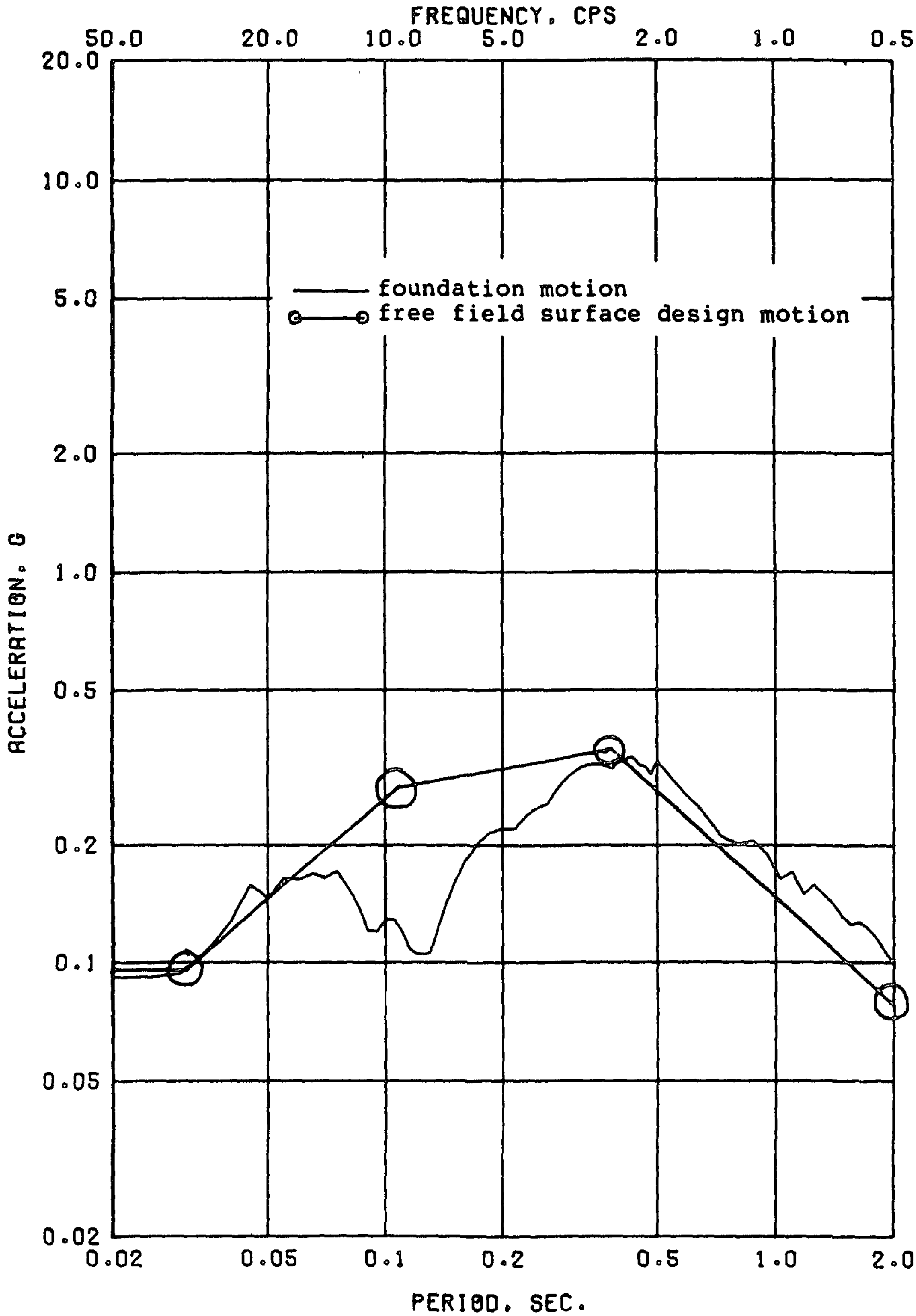

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FIGURE 3.7-8

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM OBE (4% DAMPING)

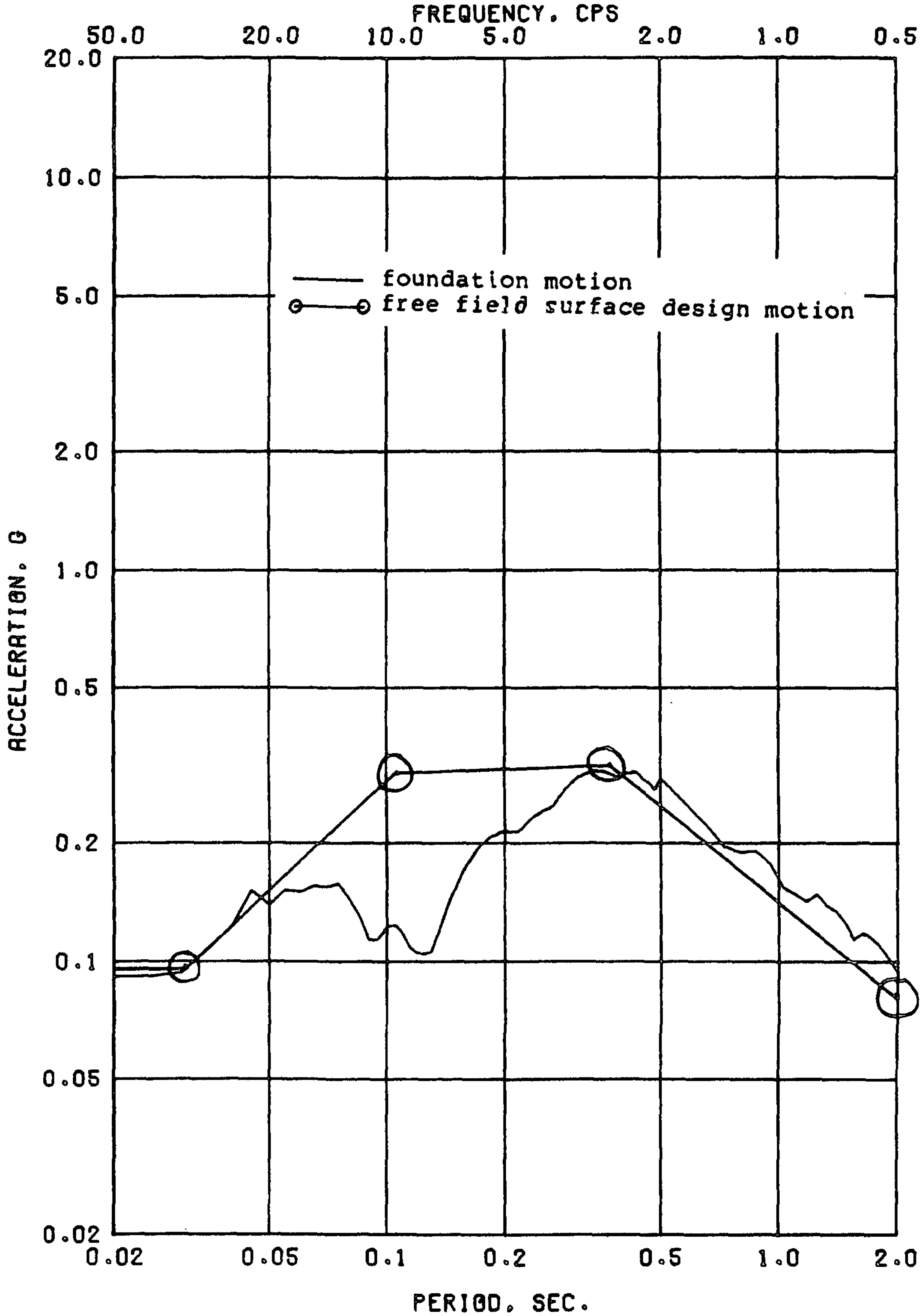

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FIGURE 3.7-9

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM OBE (5% DAMPING)

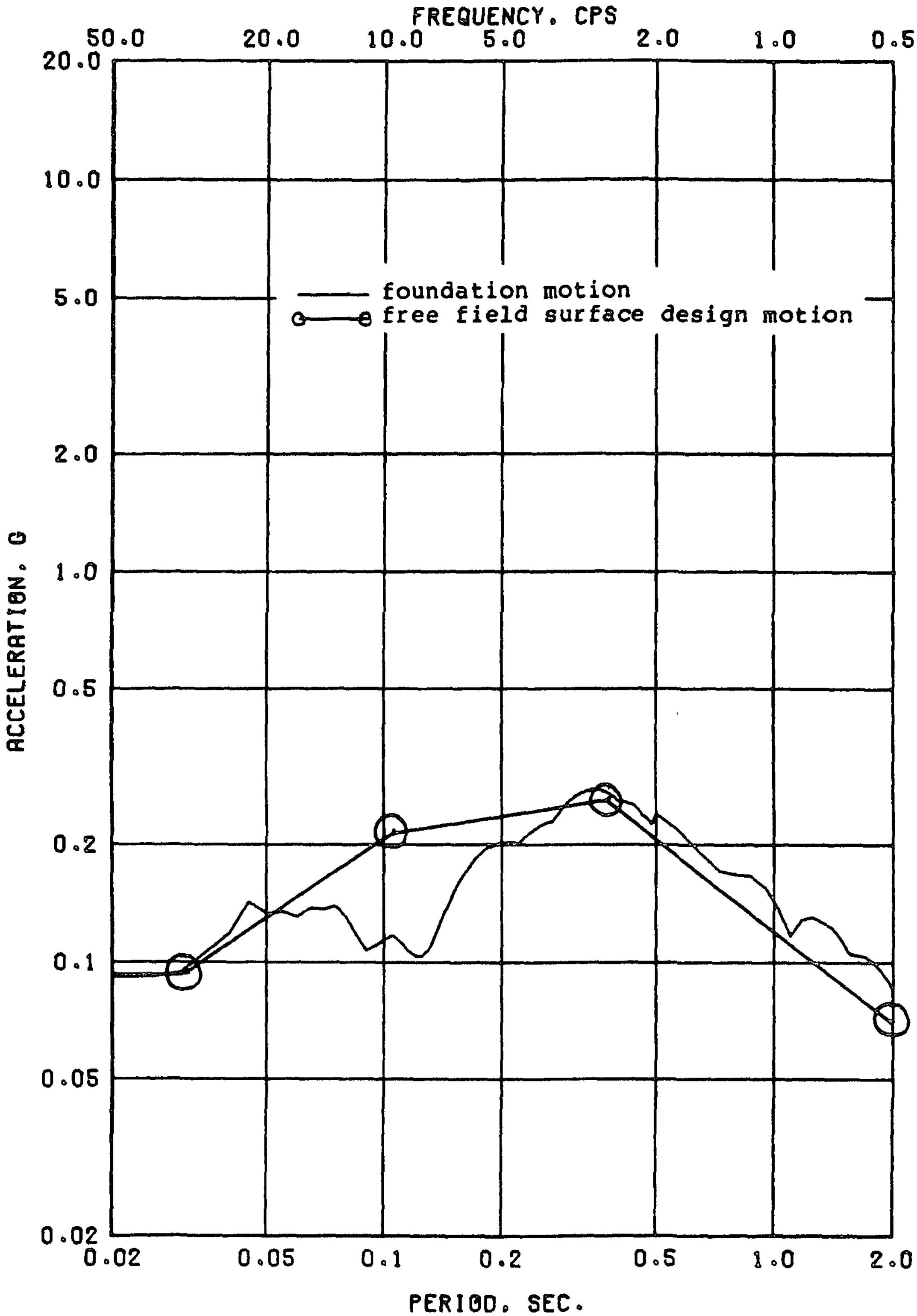

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FIGURE 3.7-10

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM OBE (7% DAMPING)

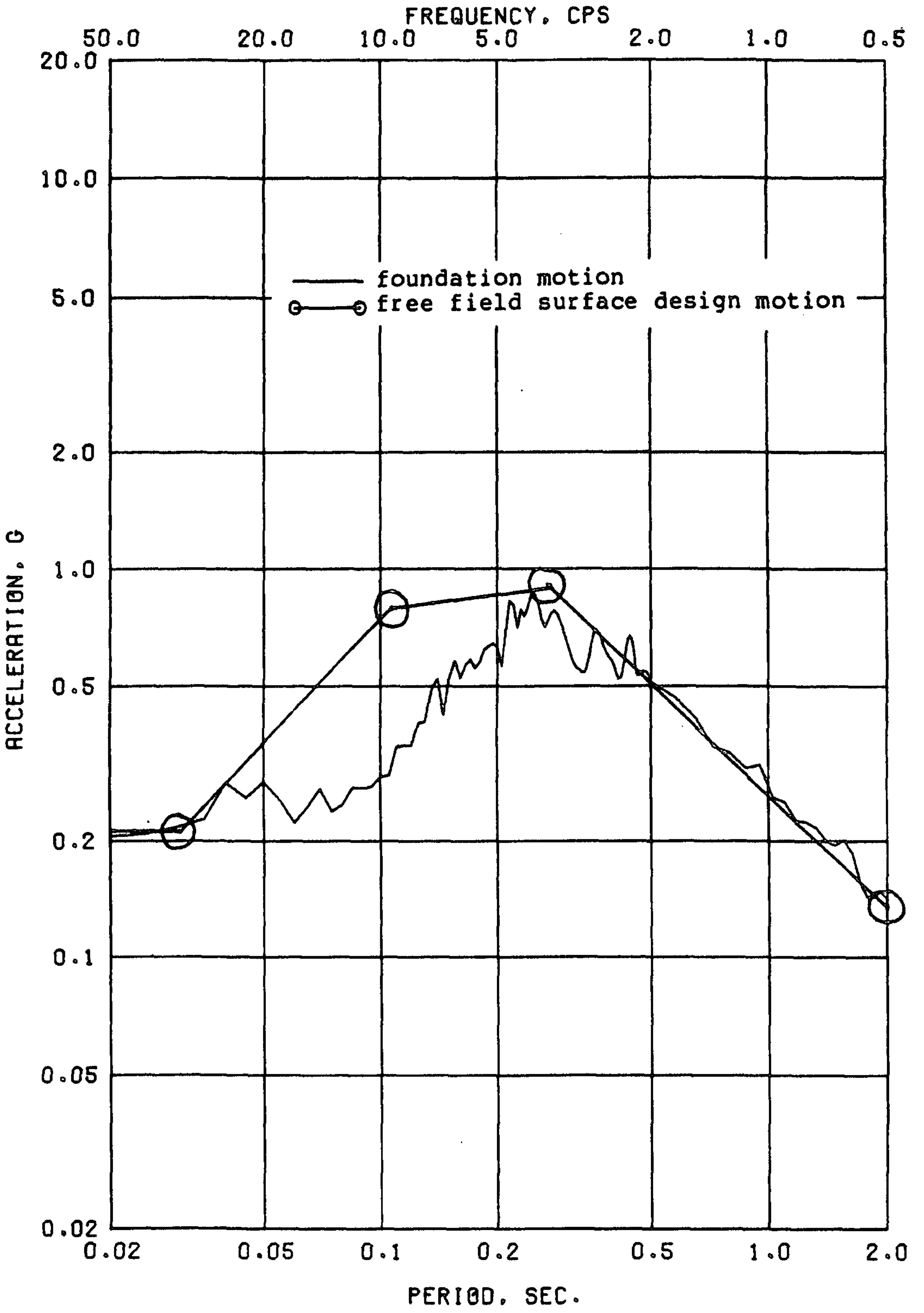

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FIGURE 3.7-11

VERTICAL FOUNDATION RESPONSE
SPECTRUM SSE (2% DAMPING)

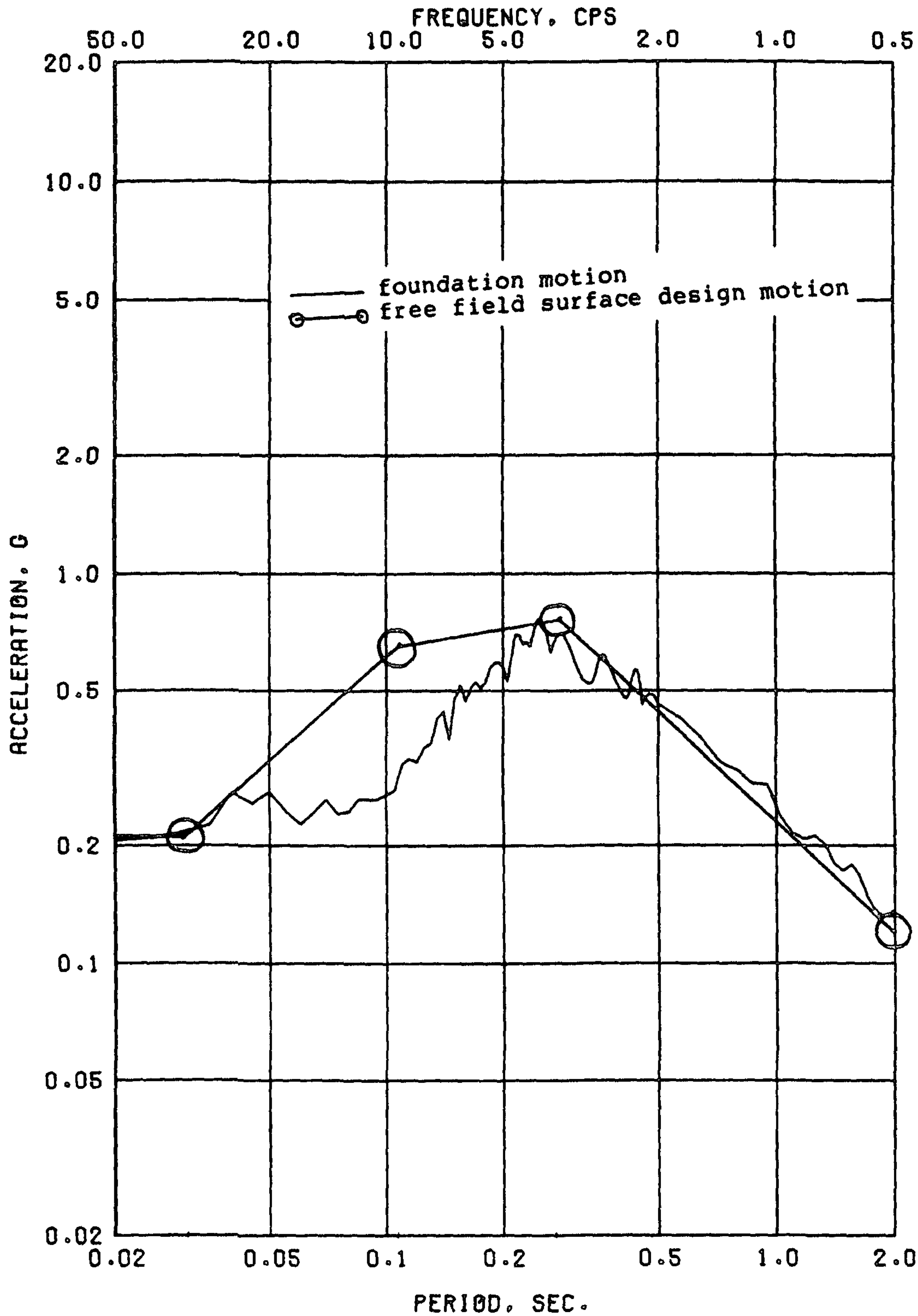

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FIGURE 3.7-12

VERTICAL FOUNDATION RESPONSE SPECTRUM SSE (3% DAMPING)

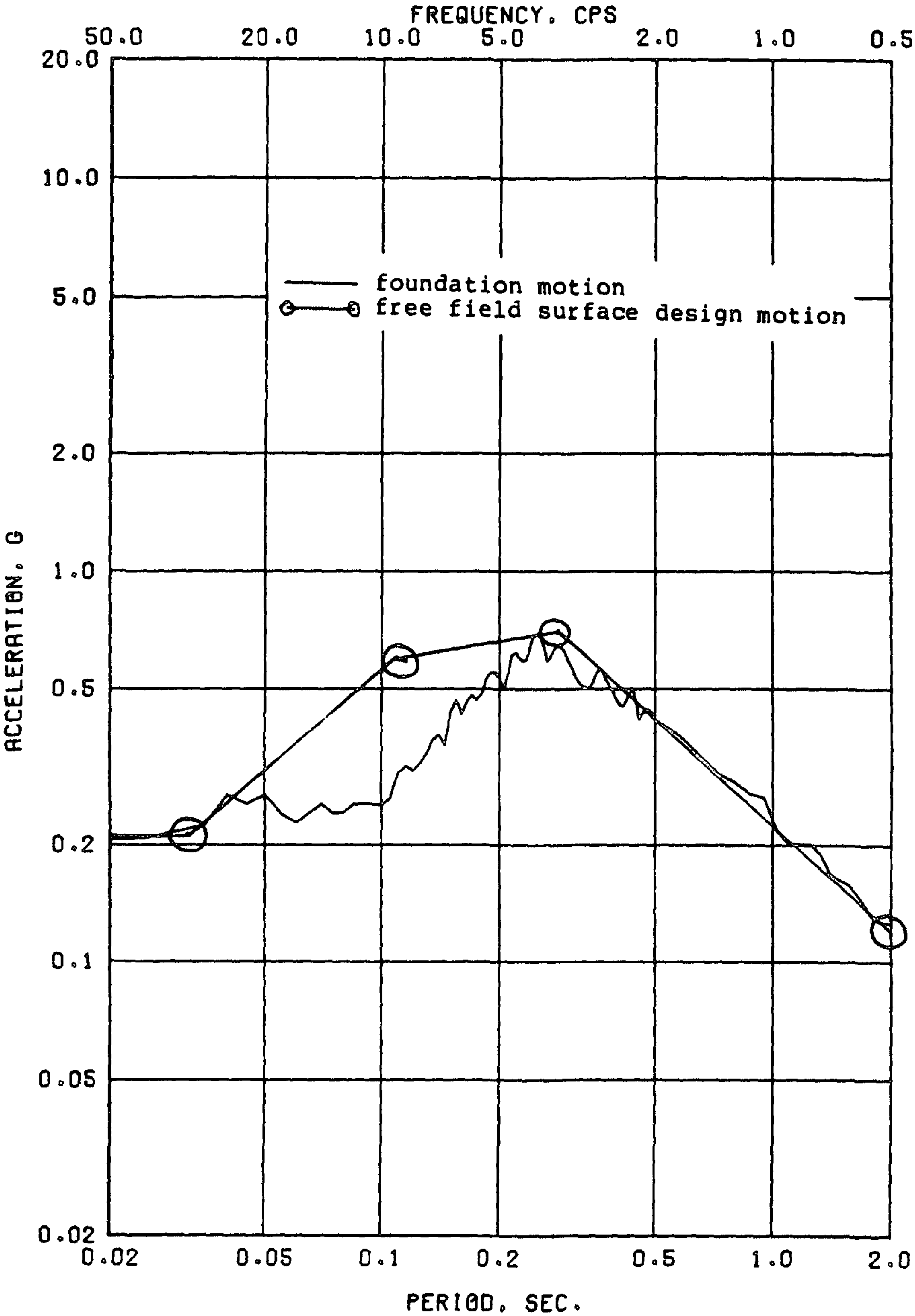

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FIGURE 3.7-13

VERTICAL FOUNDATION RESPONSE SPECTRUM SSE (4% DAMPING)

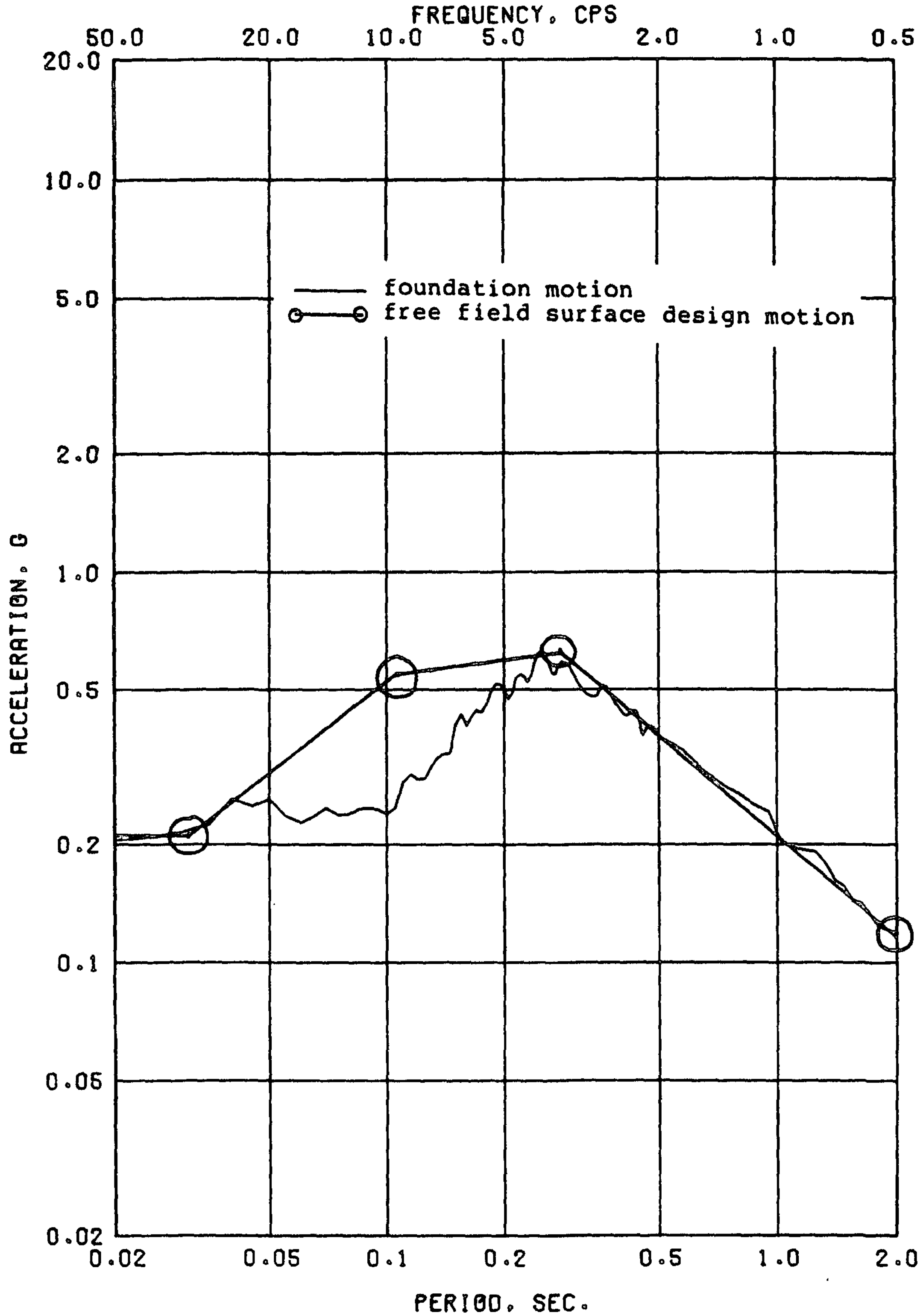

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FIGURE 3.7-14

VERTICAL FOUNDATION RESPONSE
SPECTRUM SSE (5% DAMPING)

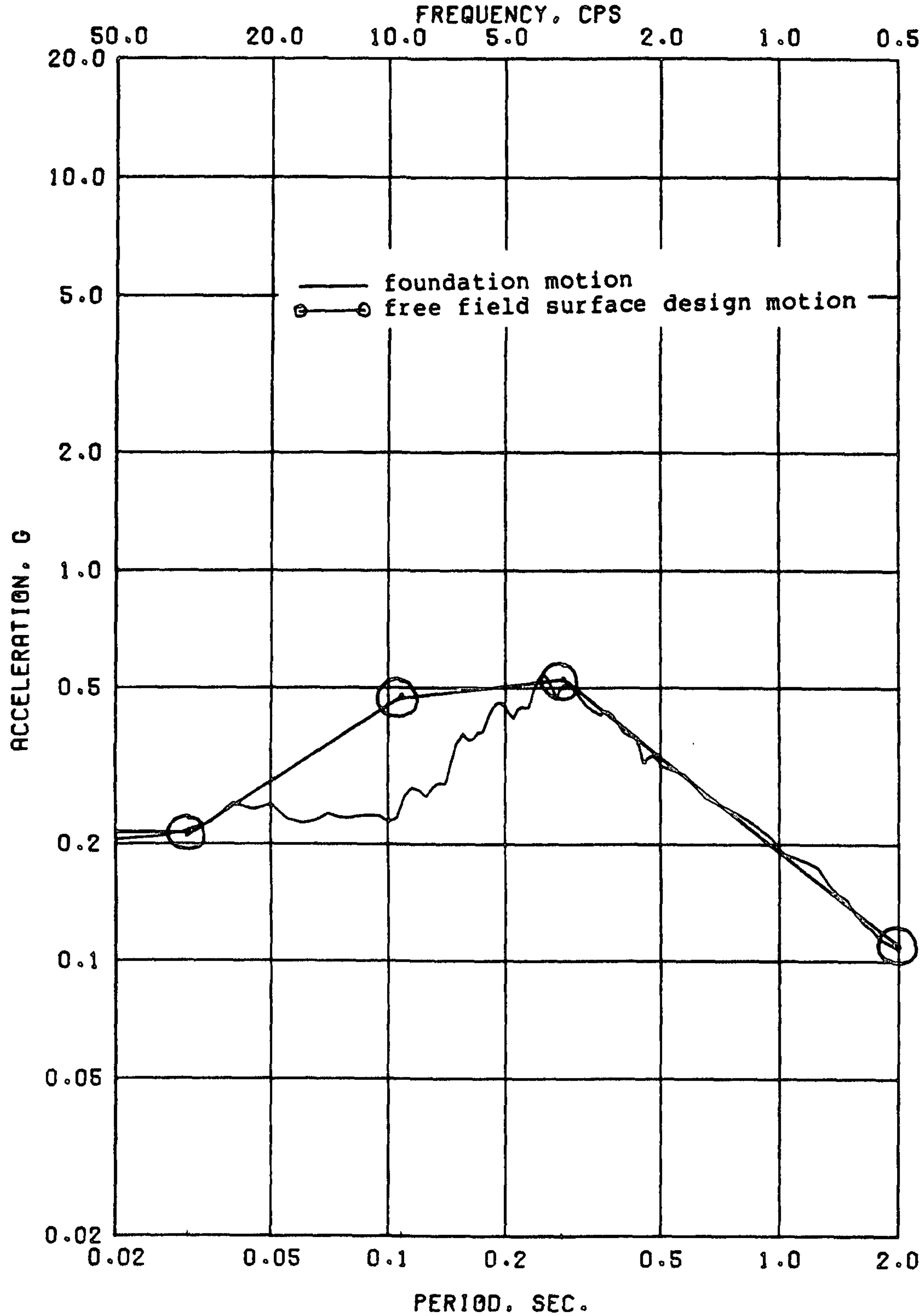

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FIGURE 3.7-15

VERTICAL FOUNDATION RESPONSE
SPECTRUM SSE (7% DAMPING)

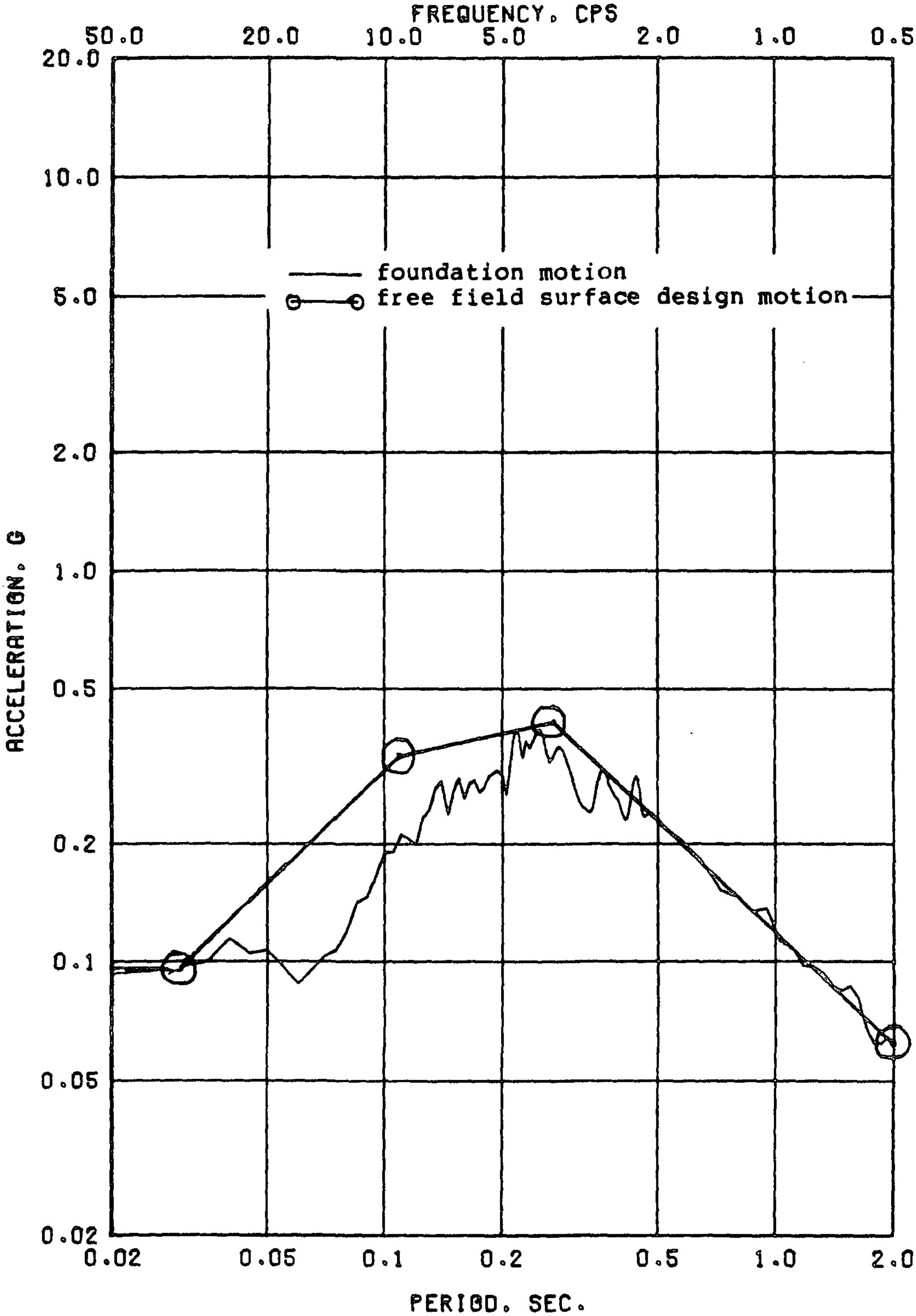

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FIGURE 3.7-16

VERTICAL FOUNDATION RESPONSE SPECTRUM OBE (2% DAMPING)

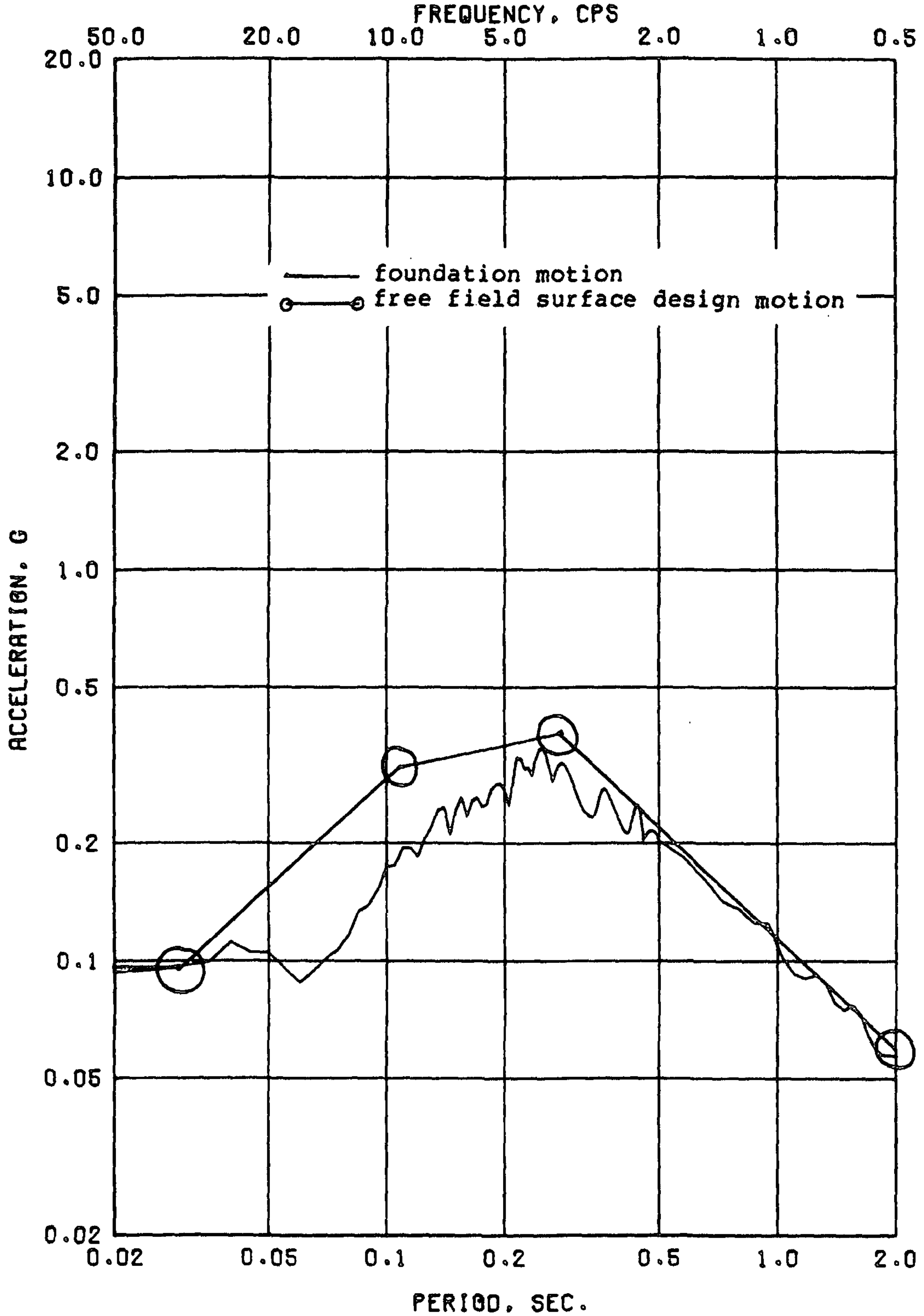

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FIGURE 3.7-17

VERTICAL FOUNDATION RESPONSE
SPECTRUM OBE (3% DAMPING)

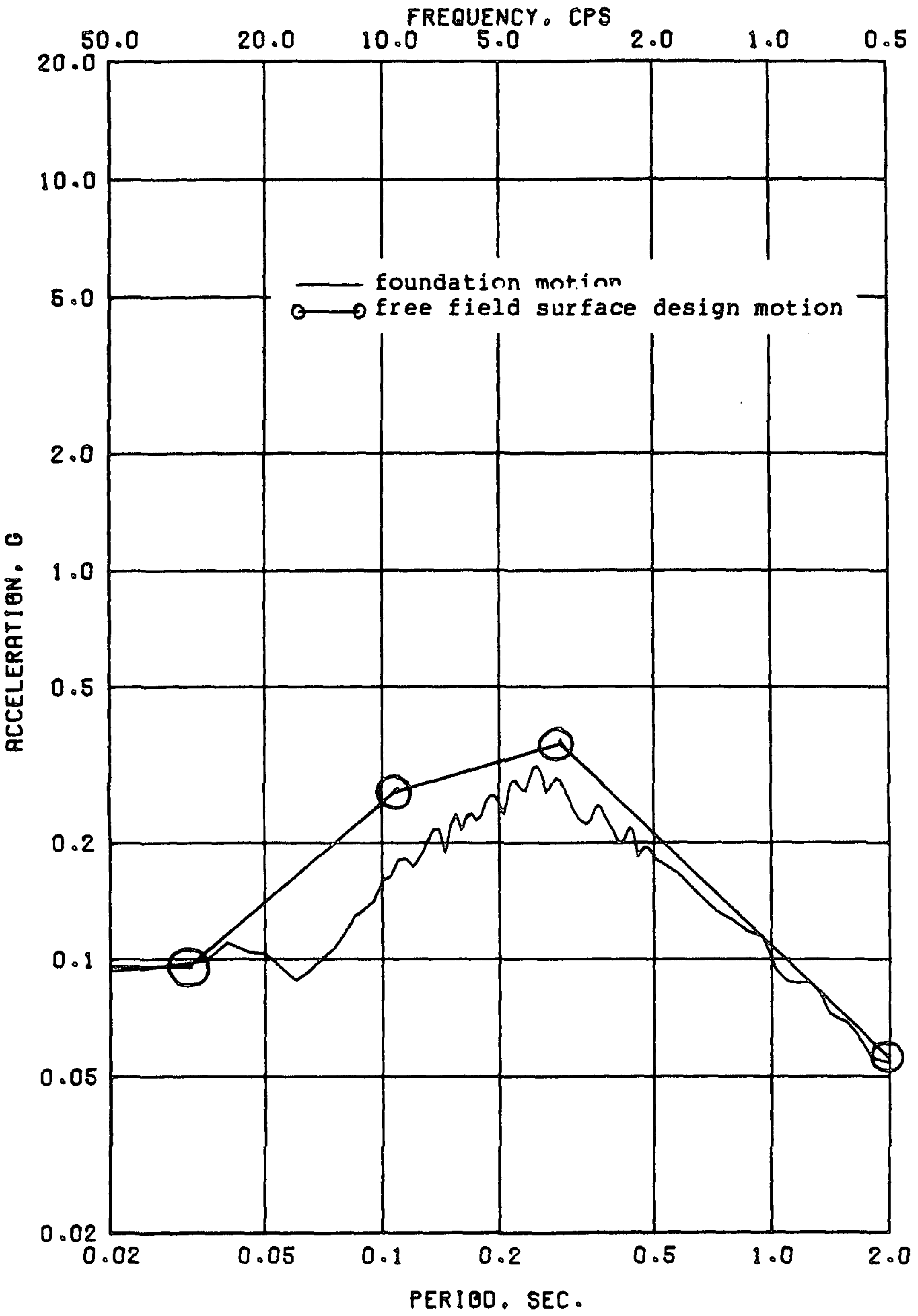

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FIGURE 3.7-18

VERTICAL FOUNDATION RESPONSE
SPECTRUM OBE (4% DAMPING)

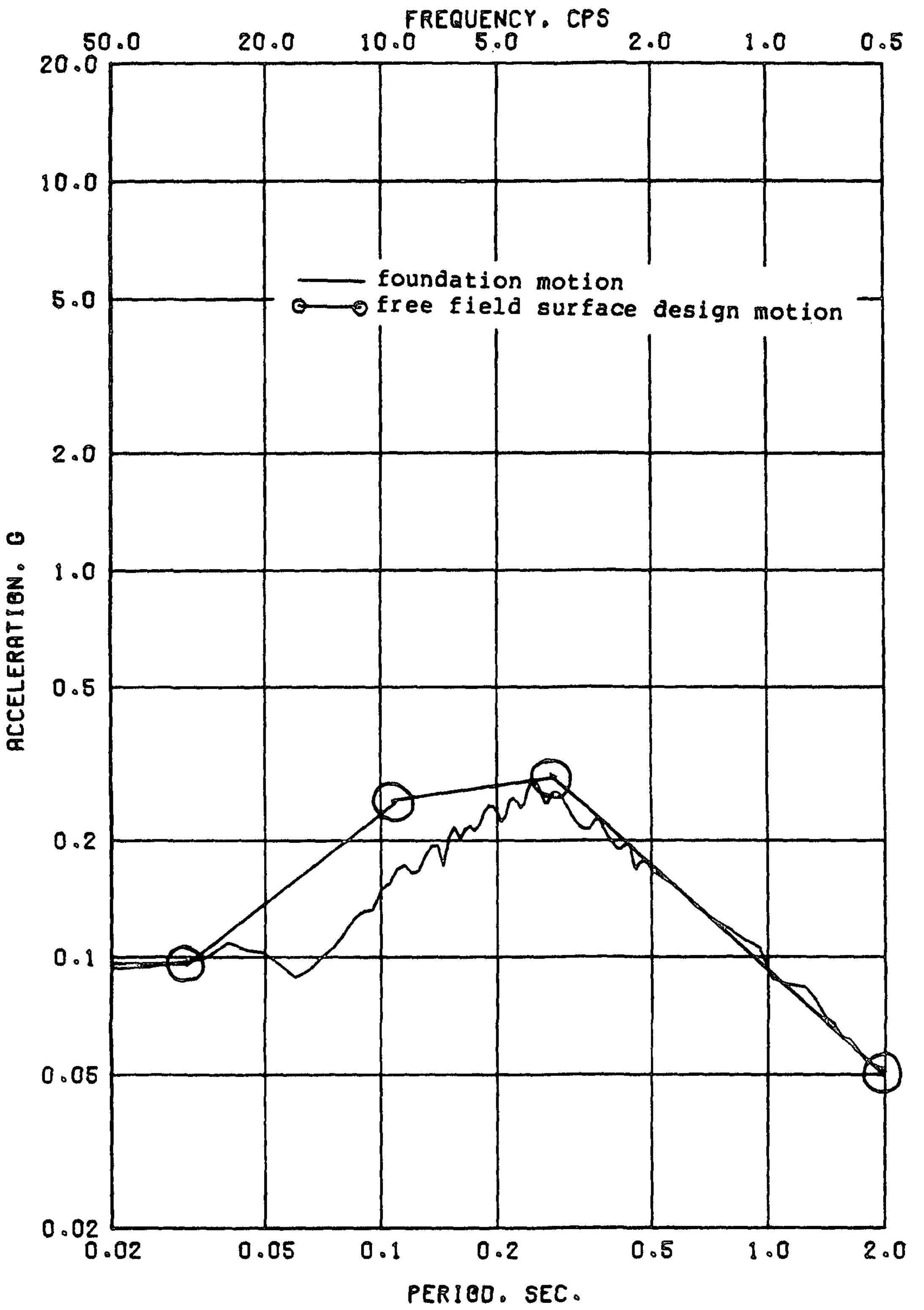

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FIGURE 3.7-19

VERTICAL FOUNDATION RESPONSE SPECTRUM OBE (5% DAMPING)

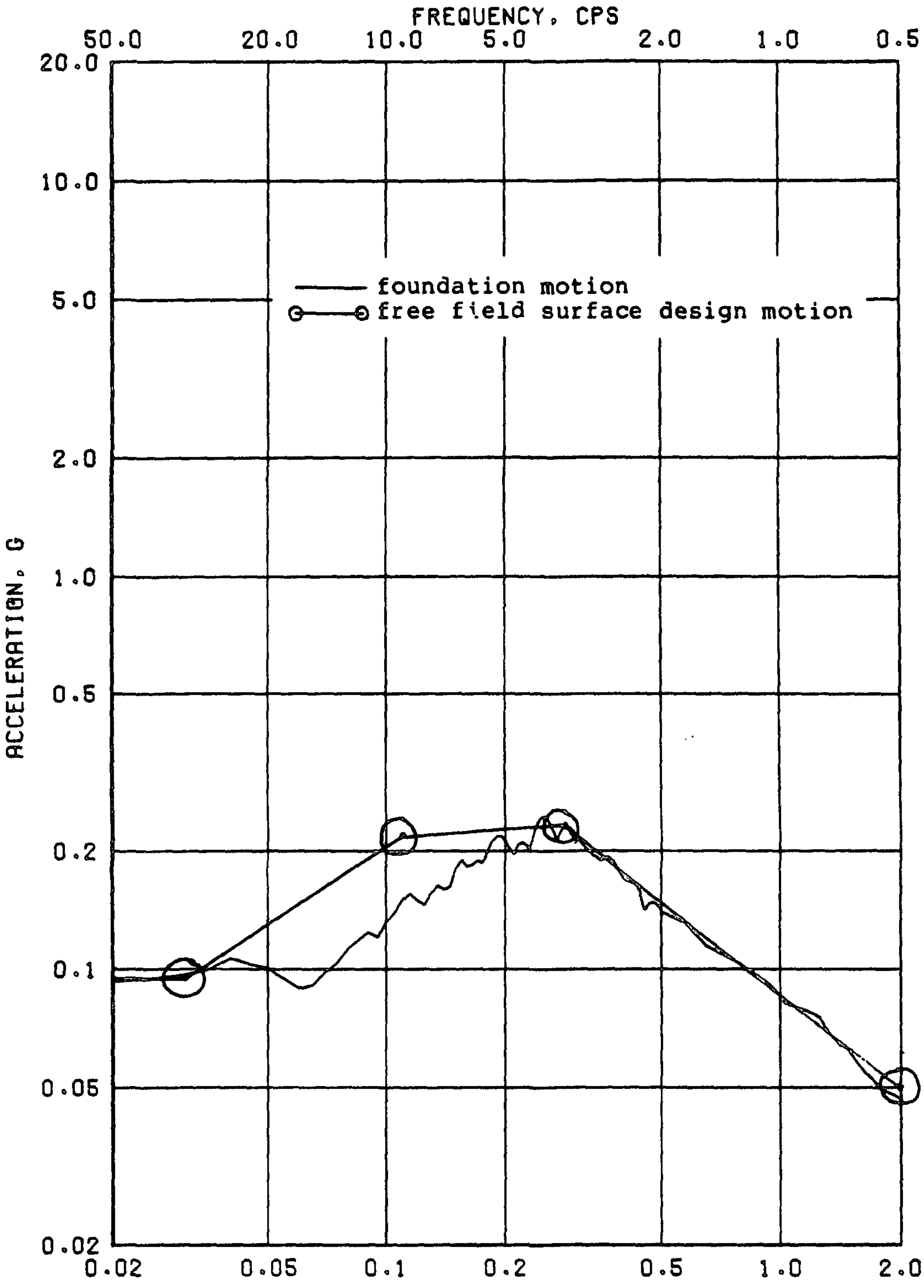

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FIGURE 3.7-20

VERTICAL FOUNDATION RESPONSE
SPECTRUM OBE (7% DAMPING)

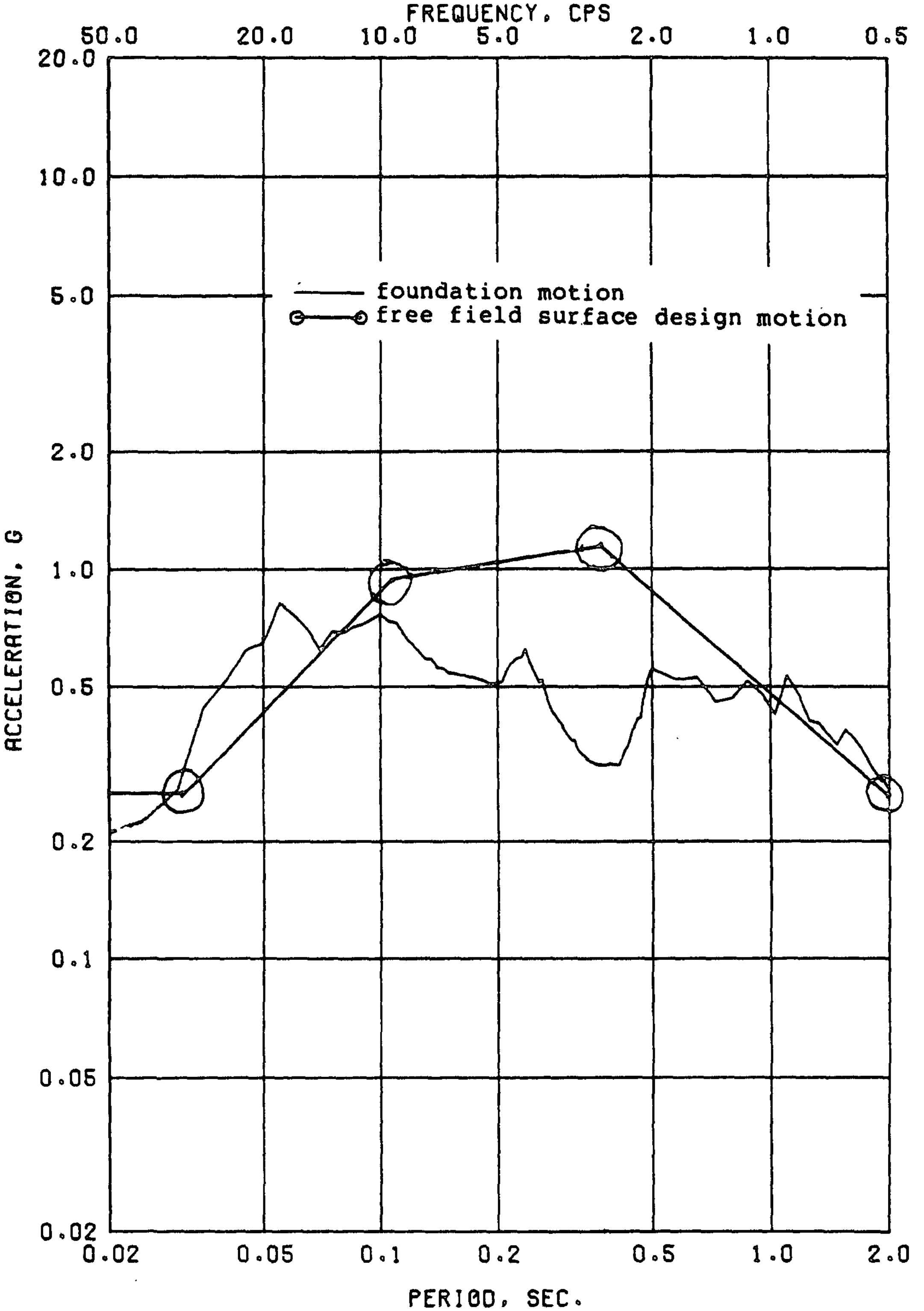

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FIGURE 3.7-21

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM SSE (2% DAMPING)

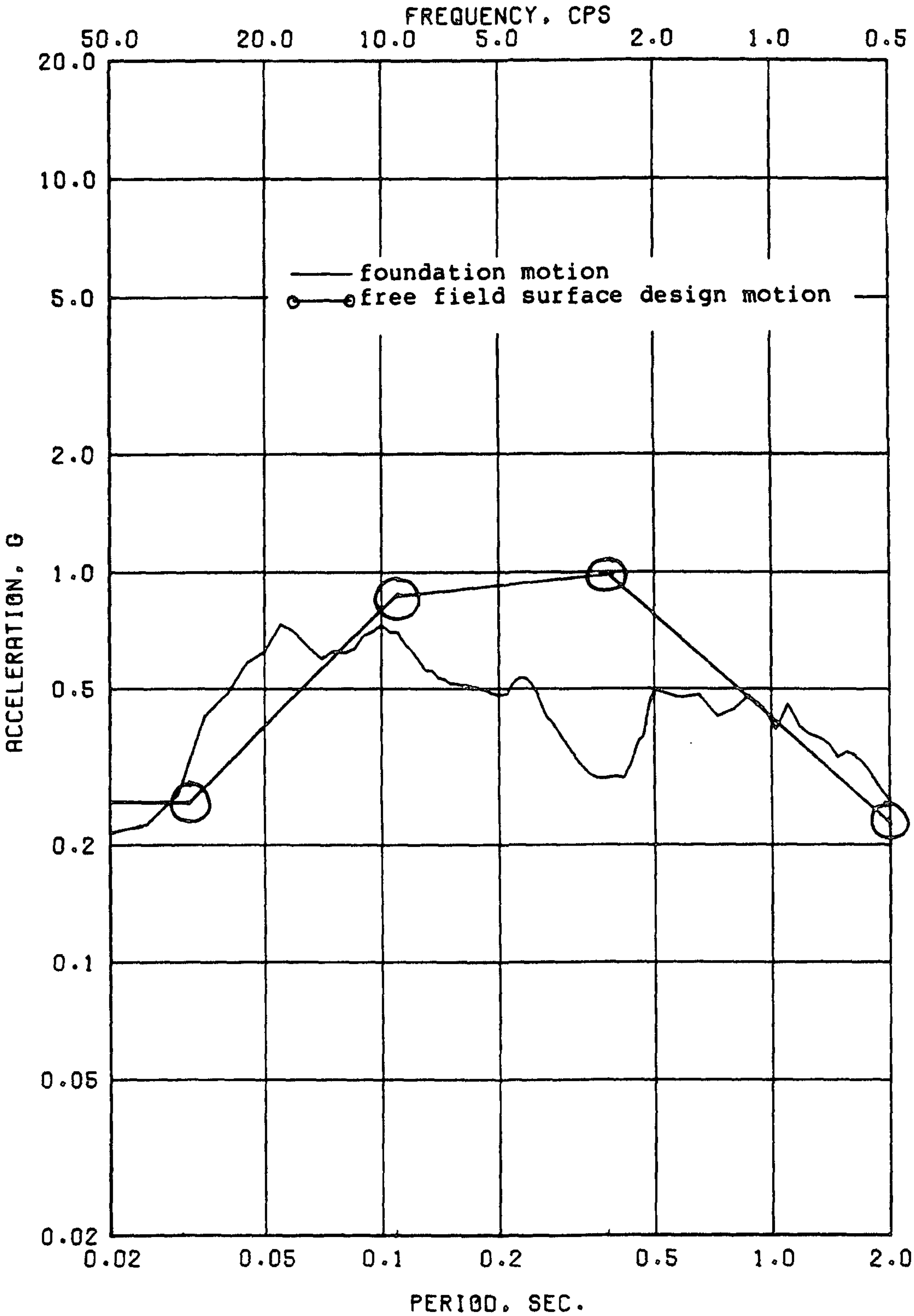

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FIGURE 3.7-22

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM SSE (3% DAMPING)

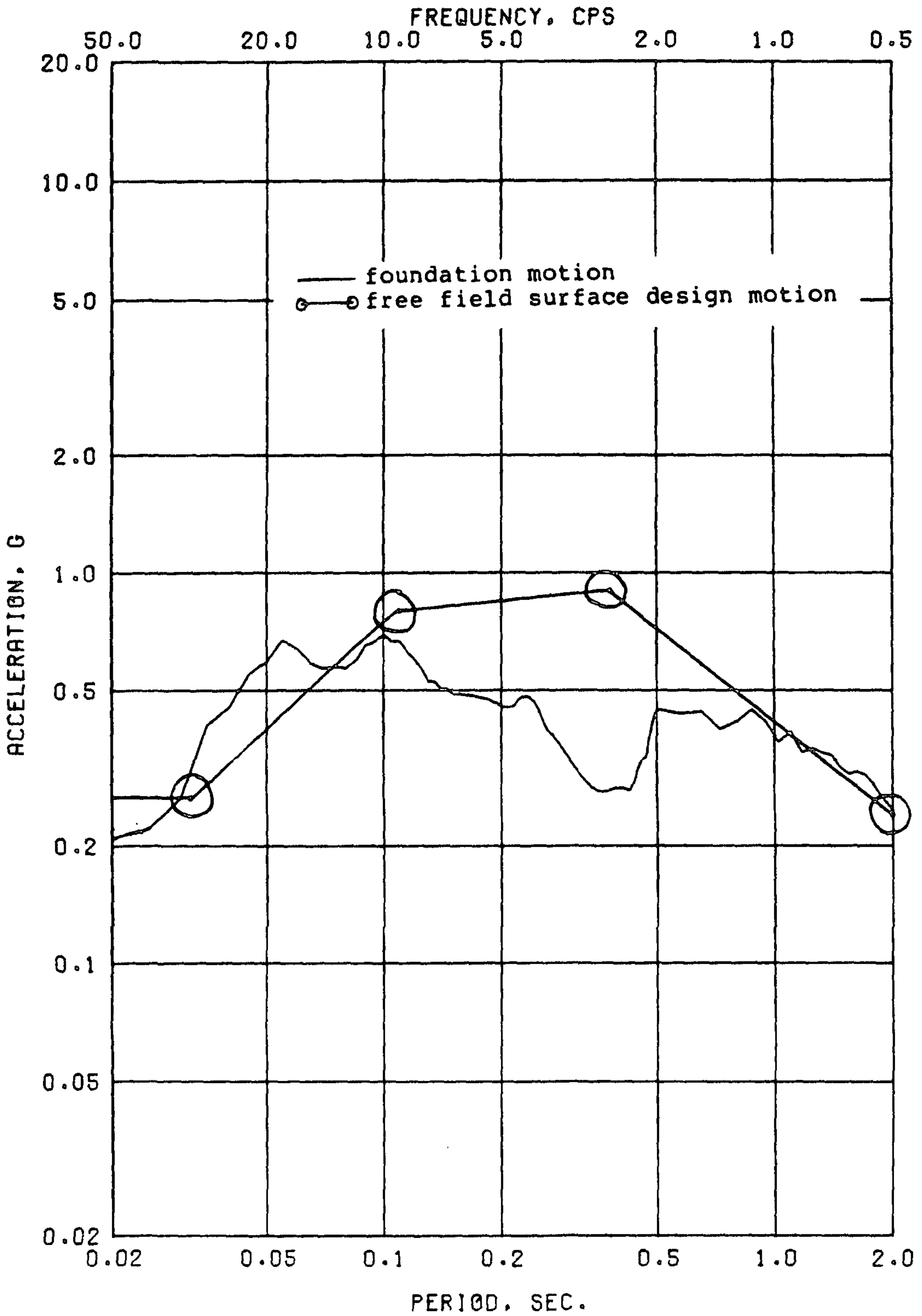

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FIGURE 3.7-23

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM SSE (4% DAMPING)

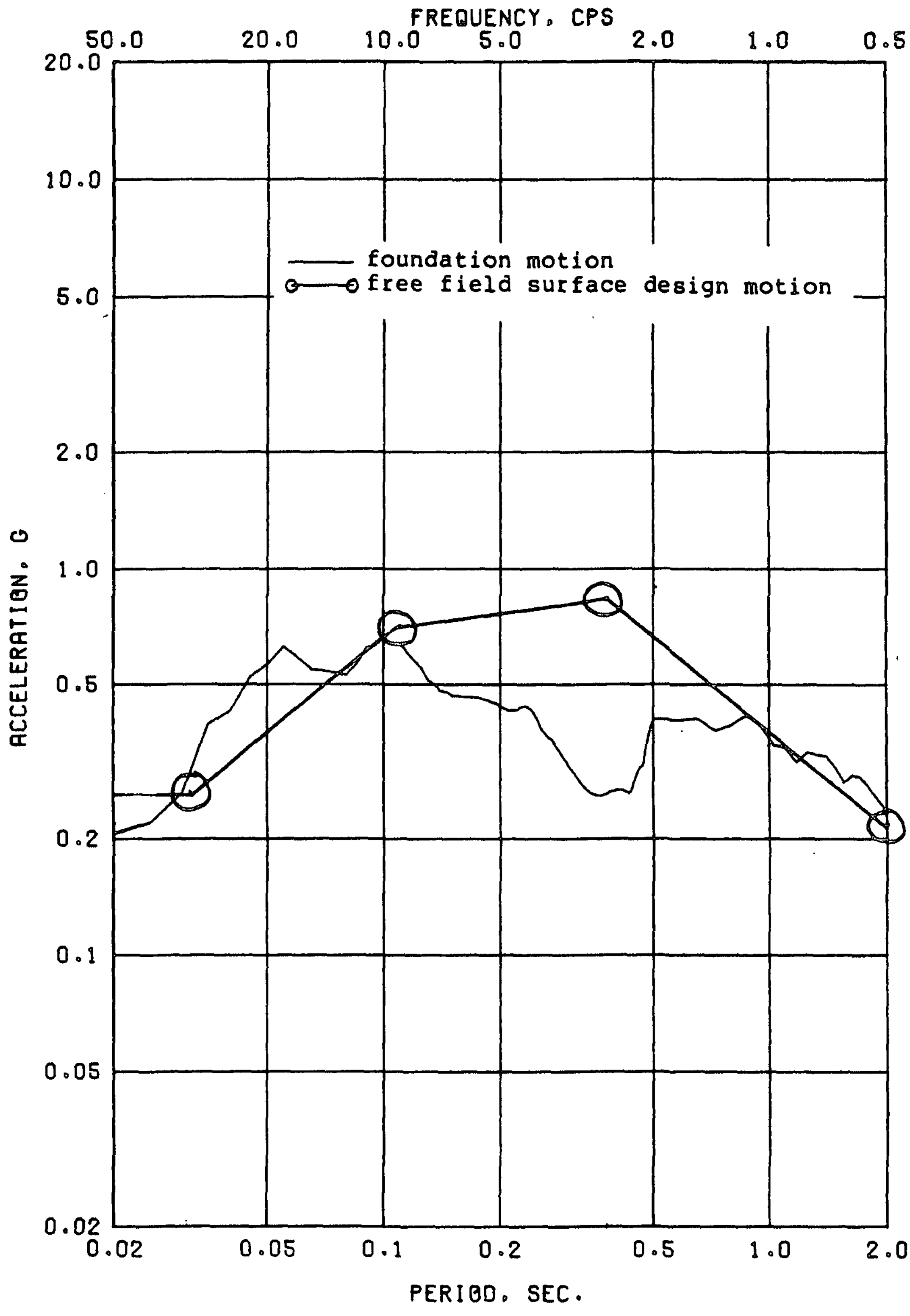

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FIGURE 3.7-24

HORIZONTAL FOUNDATION RESPONSE SPECTRUM SSE (5% DAMPING)

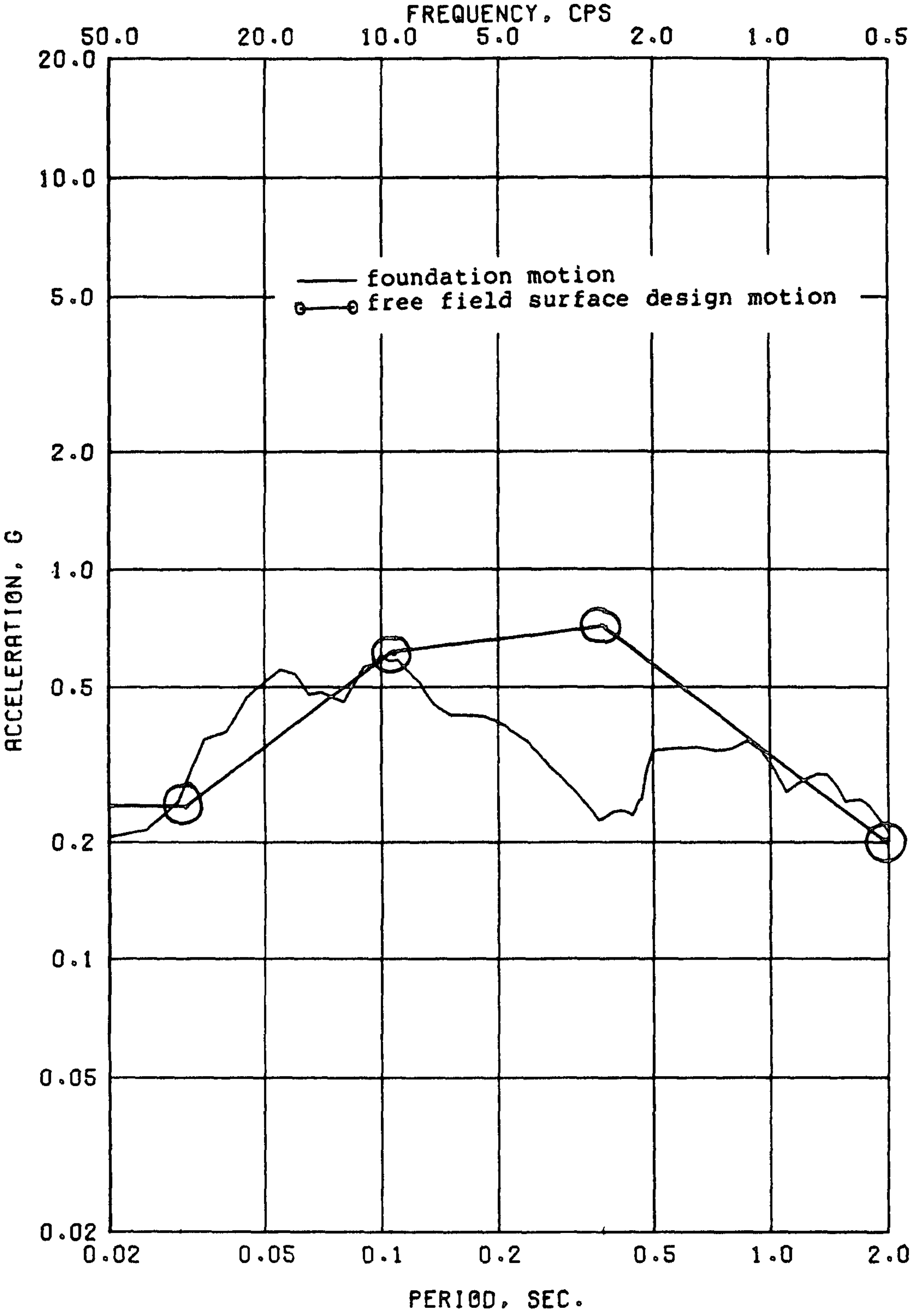

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FIGURE 3.7-25

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM SSE (7% DAMPING)

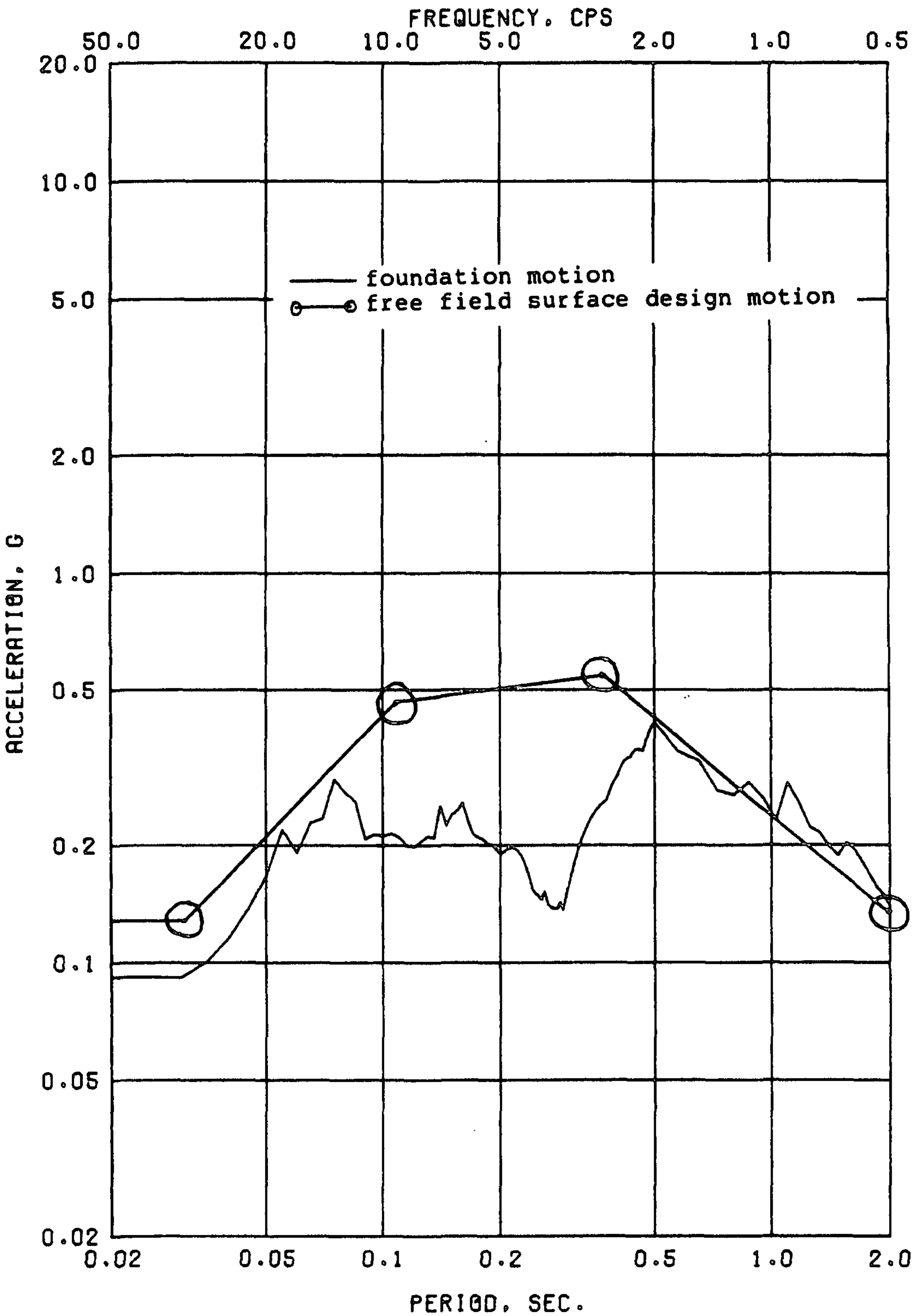

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FIGURE 3.7-26

HORIZONTAL FOUNDATION RESPONSE SPECTRUM OBE (2% DAMPING)

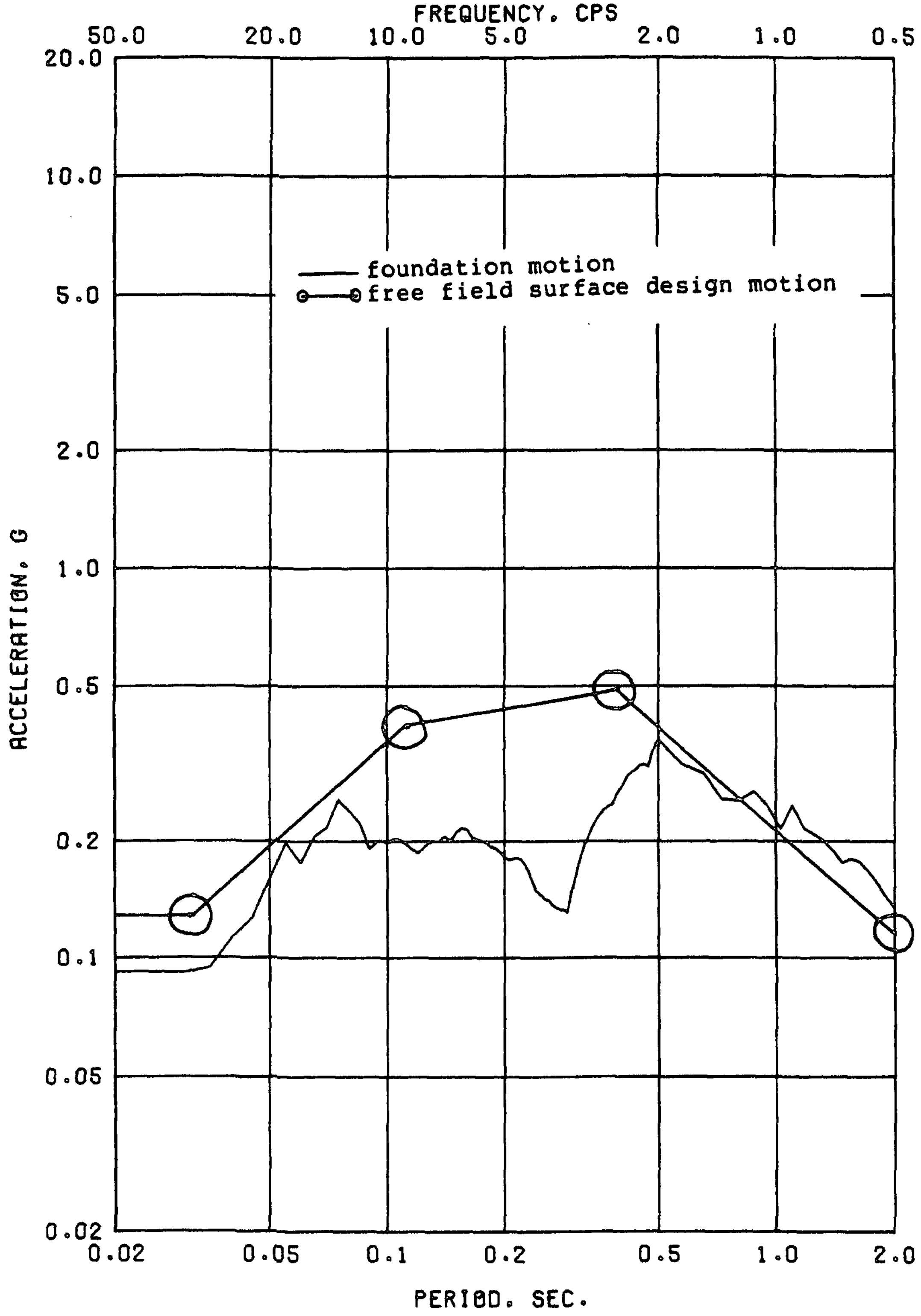

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FIGURE 3.7-27

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM OBE (3% DAMPING)

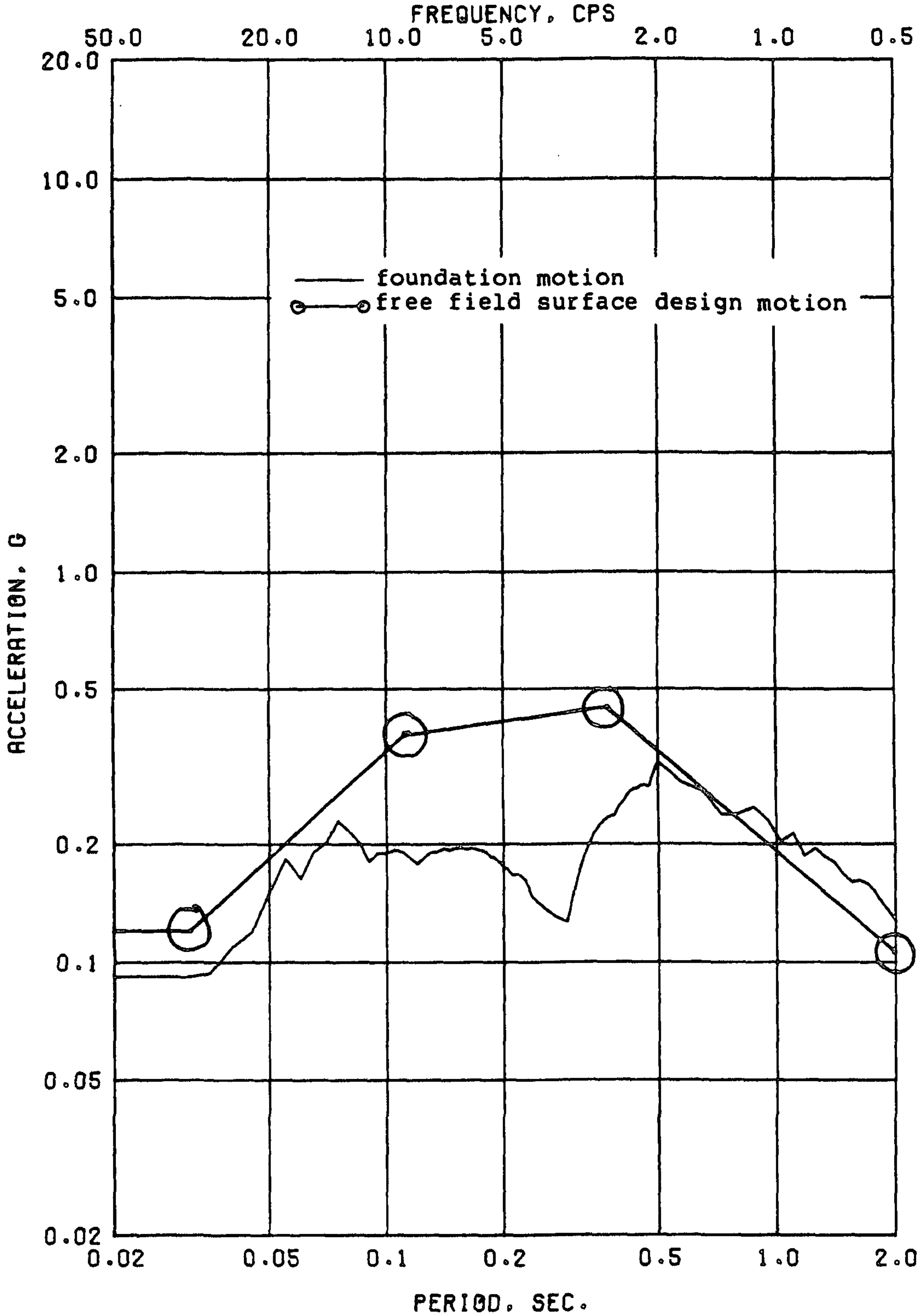

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FIGURE 3.7-28

HORIZONTAL FOUNDATION RESPONSE SPECTRUM OBE (4% DAMPING)

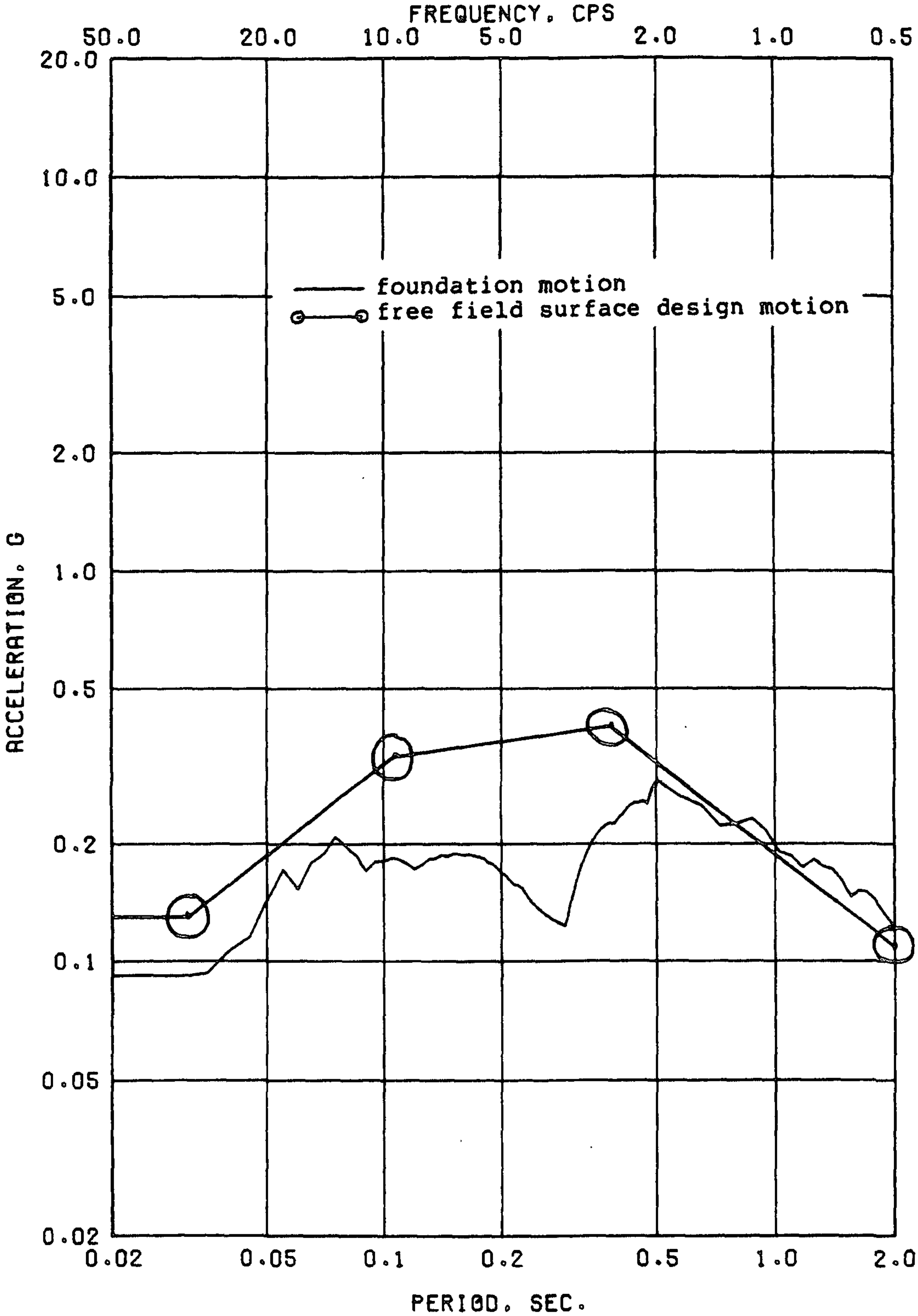

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FIGURE 3.7-29

HORIZONTAL FOUNDATION RESPONSE
SPECTRUM OBE (5% DAMPING)

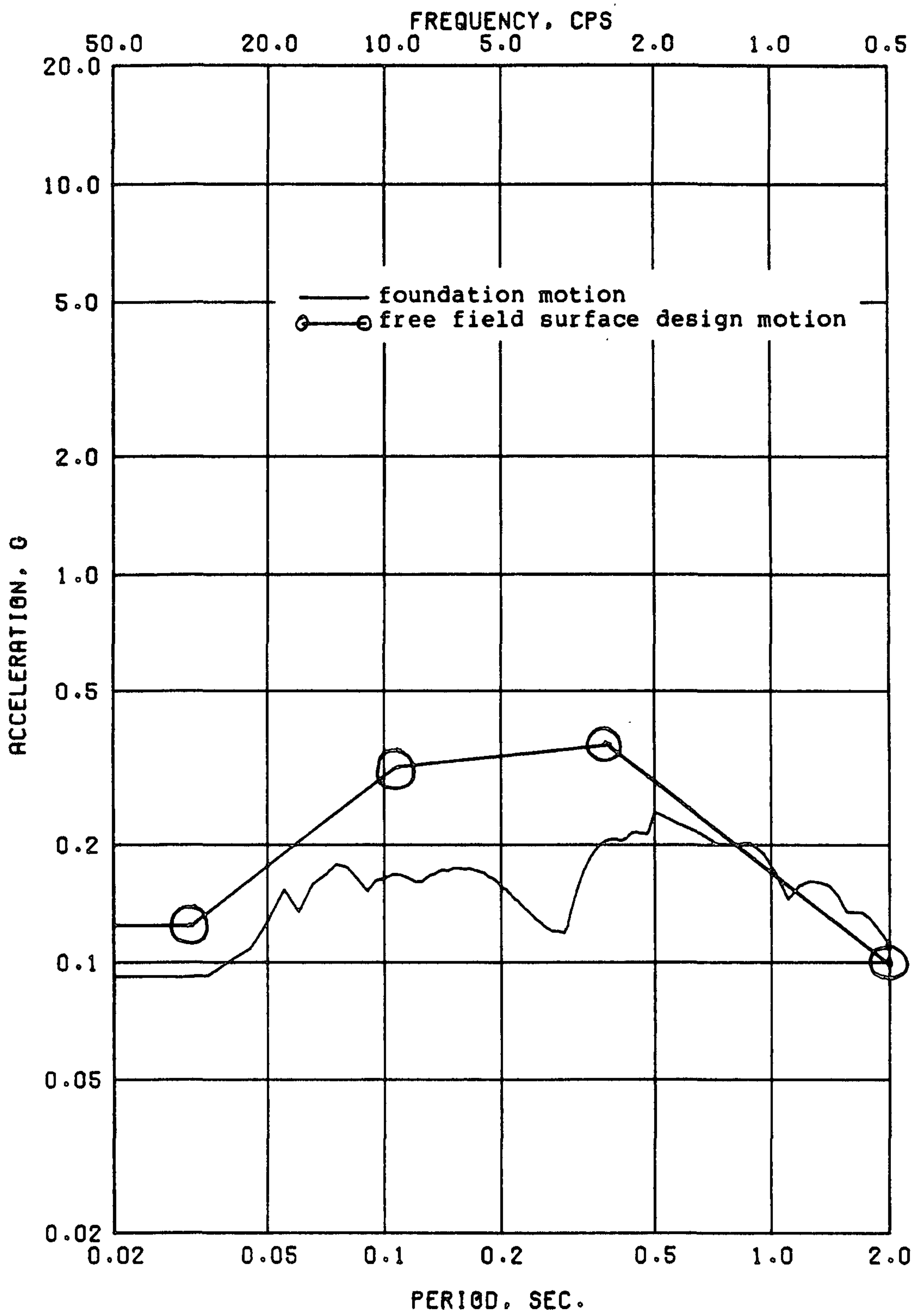

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FIGURE 3.7-30

HORIZONTAL FOUNDATION RESPONSE SPECTRUM OBE (7% DAMPING)

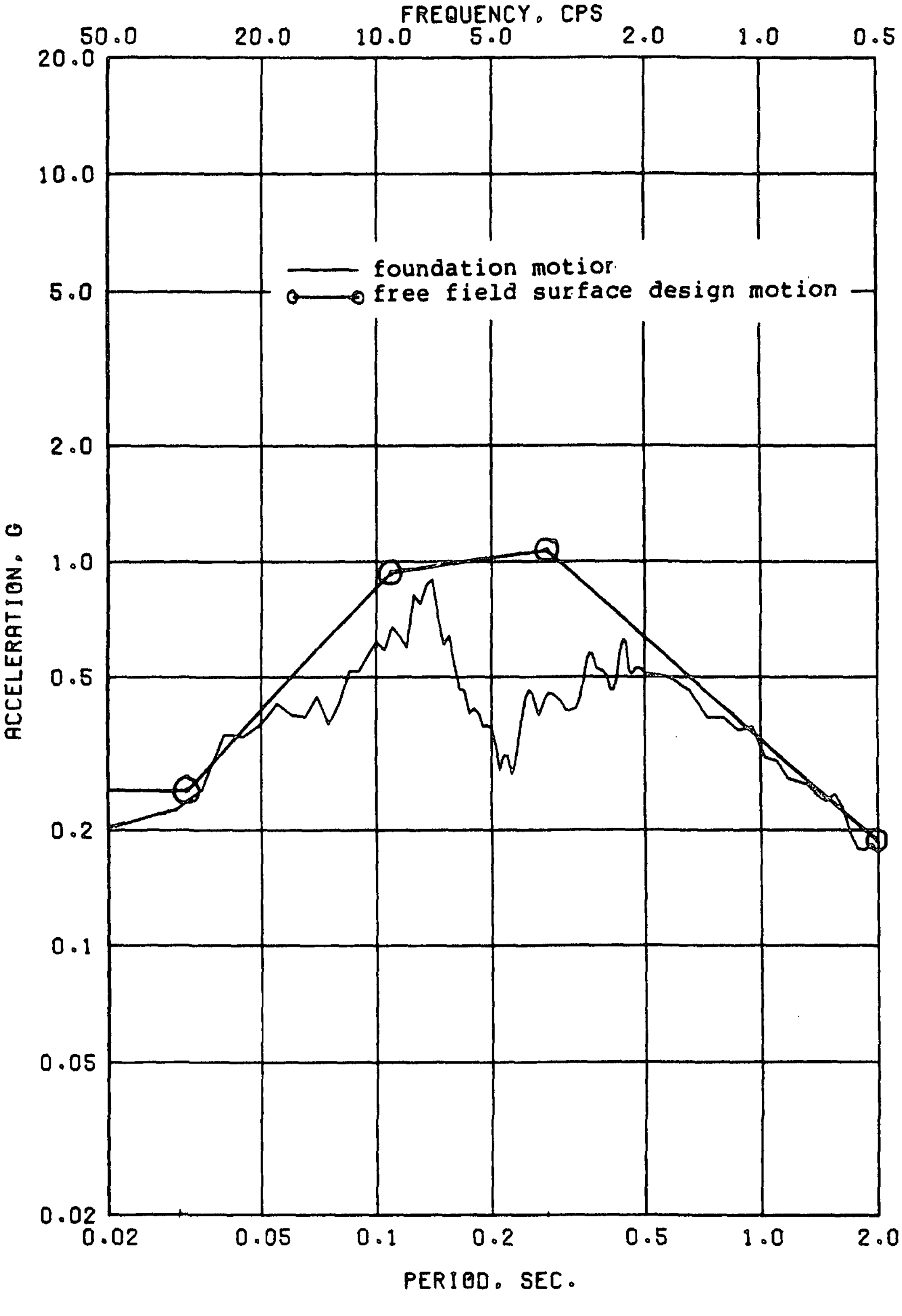

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FIGURE 3.7-31

VERTICAL FOUNDATION RESPONSE SPECTRUM SSE (2% DAMPING)

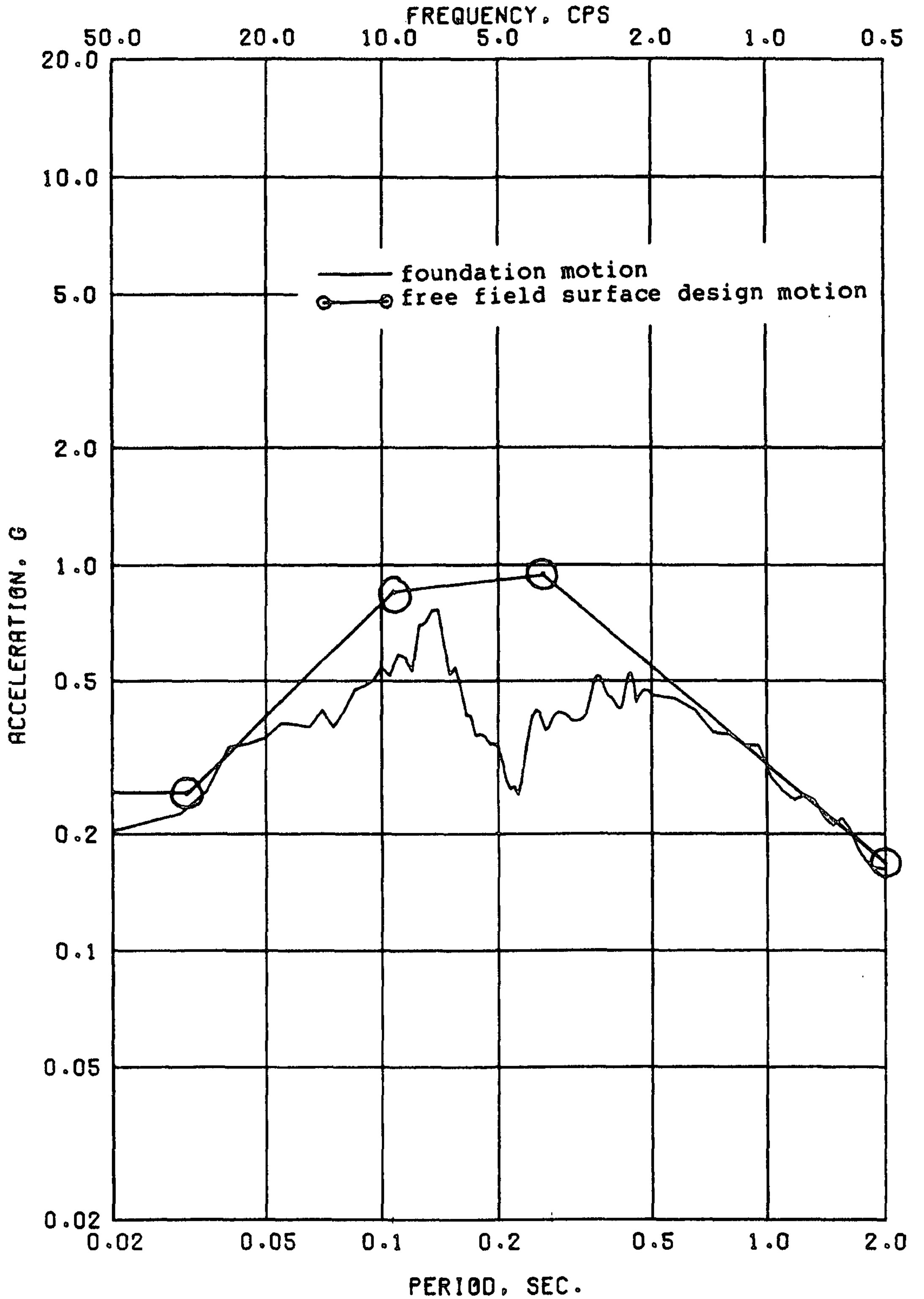

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FIGURE 3.7-32

VERTICAL FOUNDATION RESPONSE
SPECTRUM SSE (3% DAMPING)

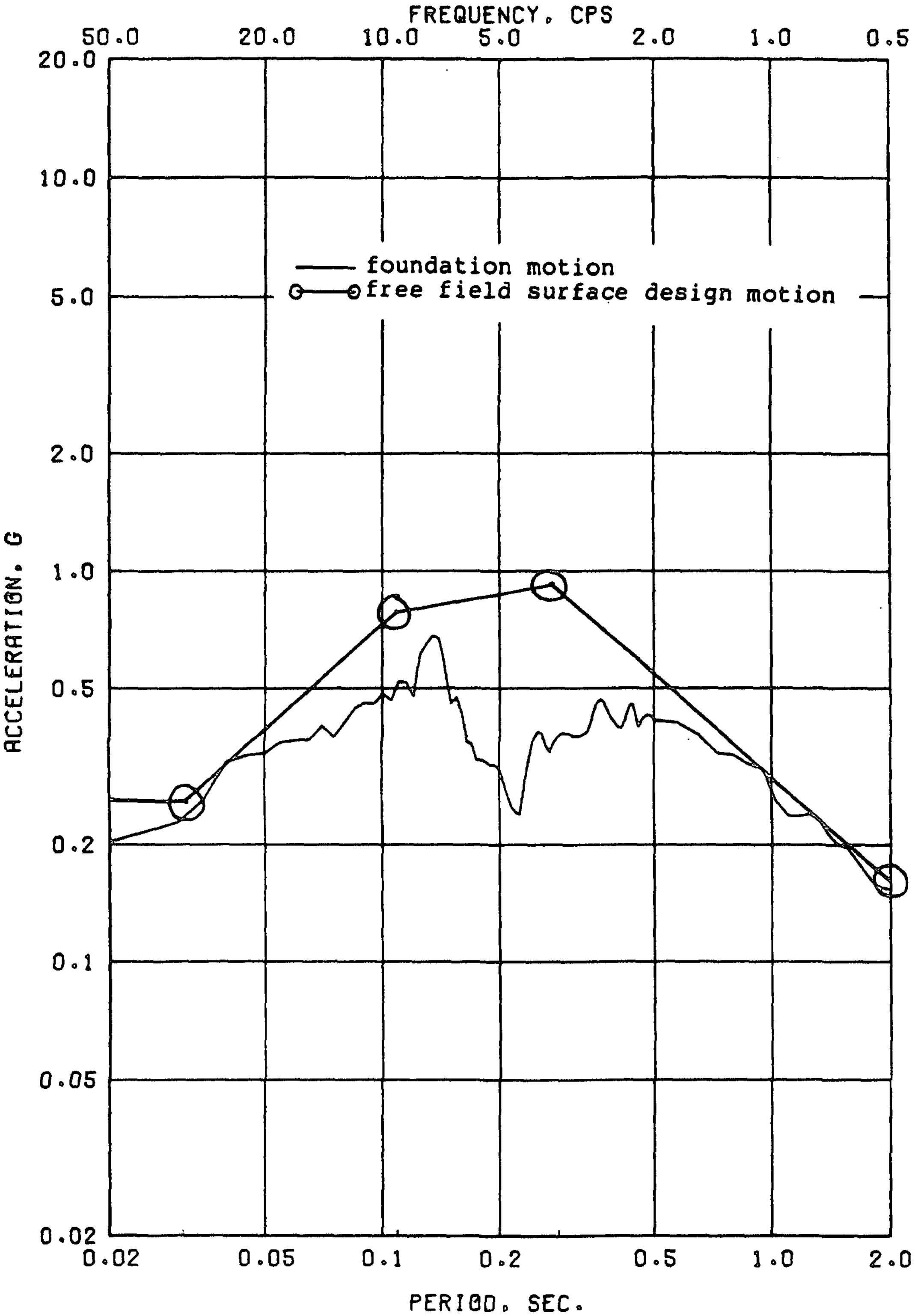

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FIGURE 3.7-33

VERTICAL FOUNDATION RESPONSE
SPECTRUM SSE (4% DAMPING)

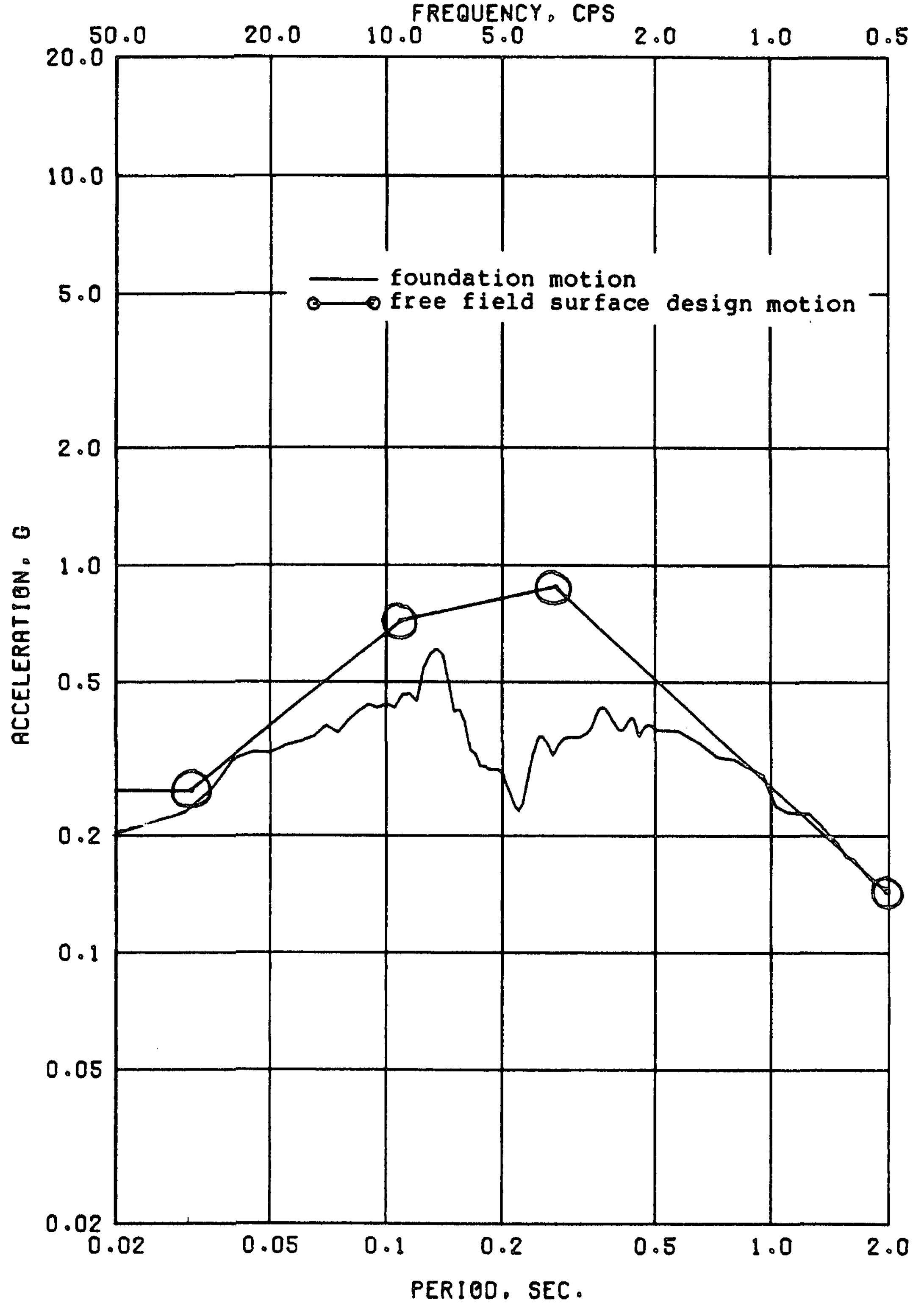

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FIGURE 3.7-34

VERTICAL FOUNDATION RESPONSE SPECTRUM SSE (5% DAMPING)

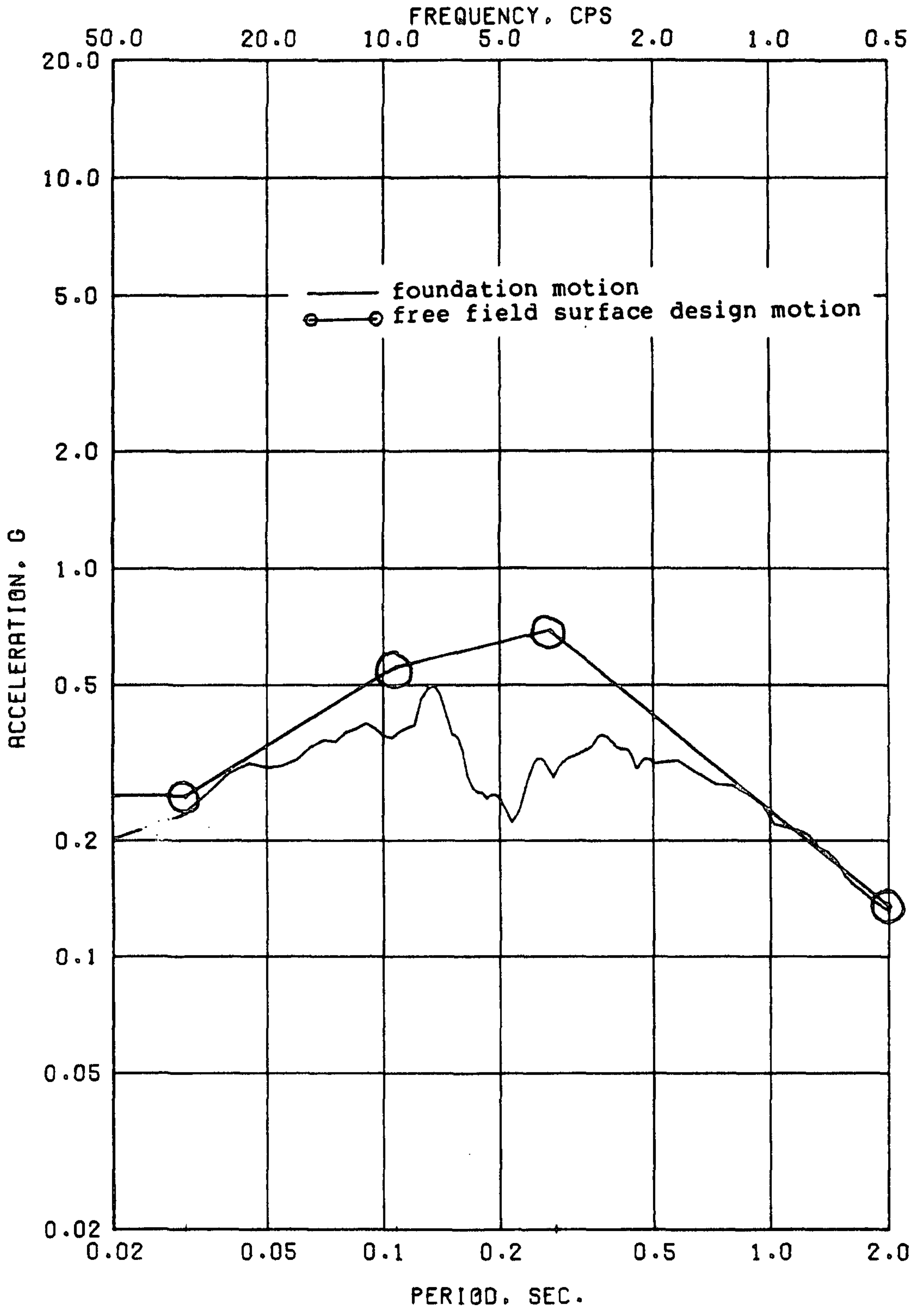

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FIGURE 3.7-35

VERTICAL FOUNDATION RESPONSE
SPECTRUM SSE (7% DAMPING)

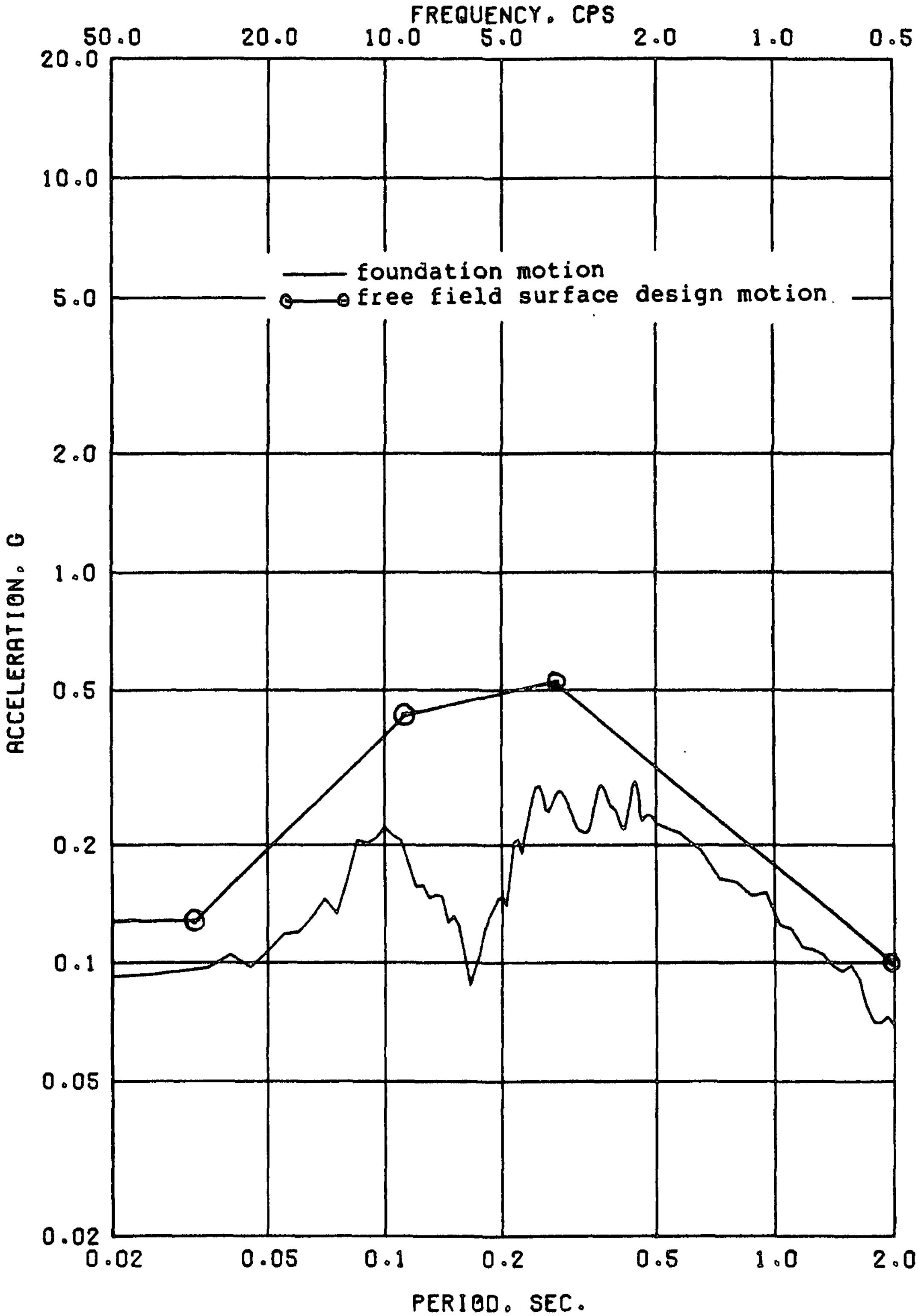

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FIGURE 3.7-36

VERTICAL FOUNDATION RESPONSE
SPECTRUM OBE (2% DAMPING)

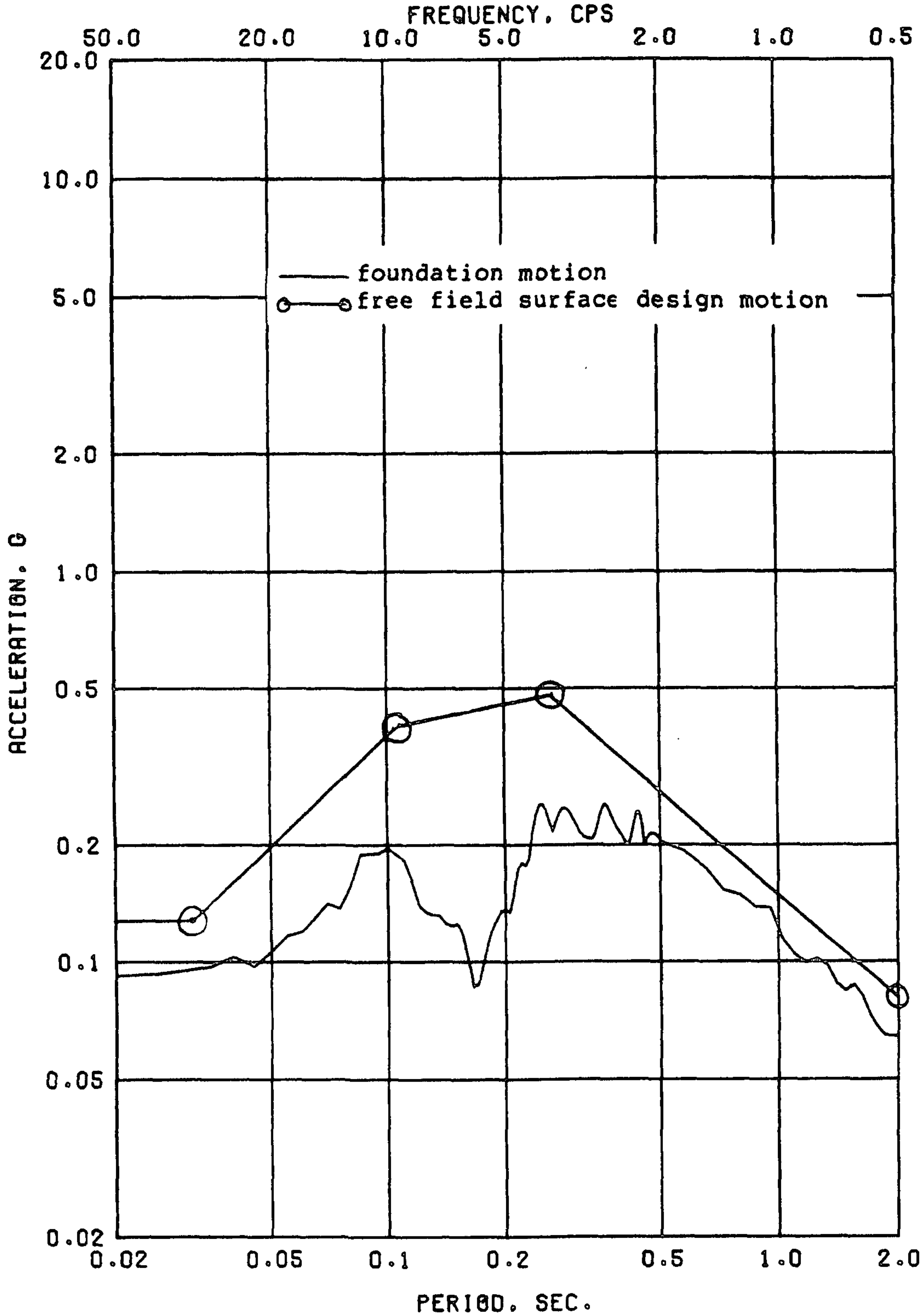

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FIGURE 3.7-37

VERTICAL FOUNDATION RESPONSE
SPECTRUM OBE (3% DAMPING)

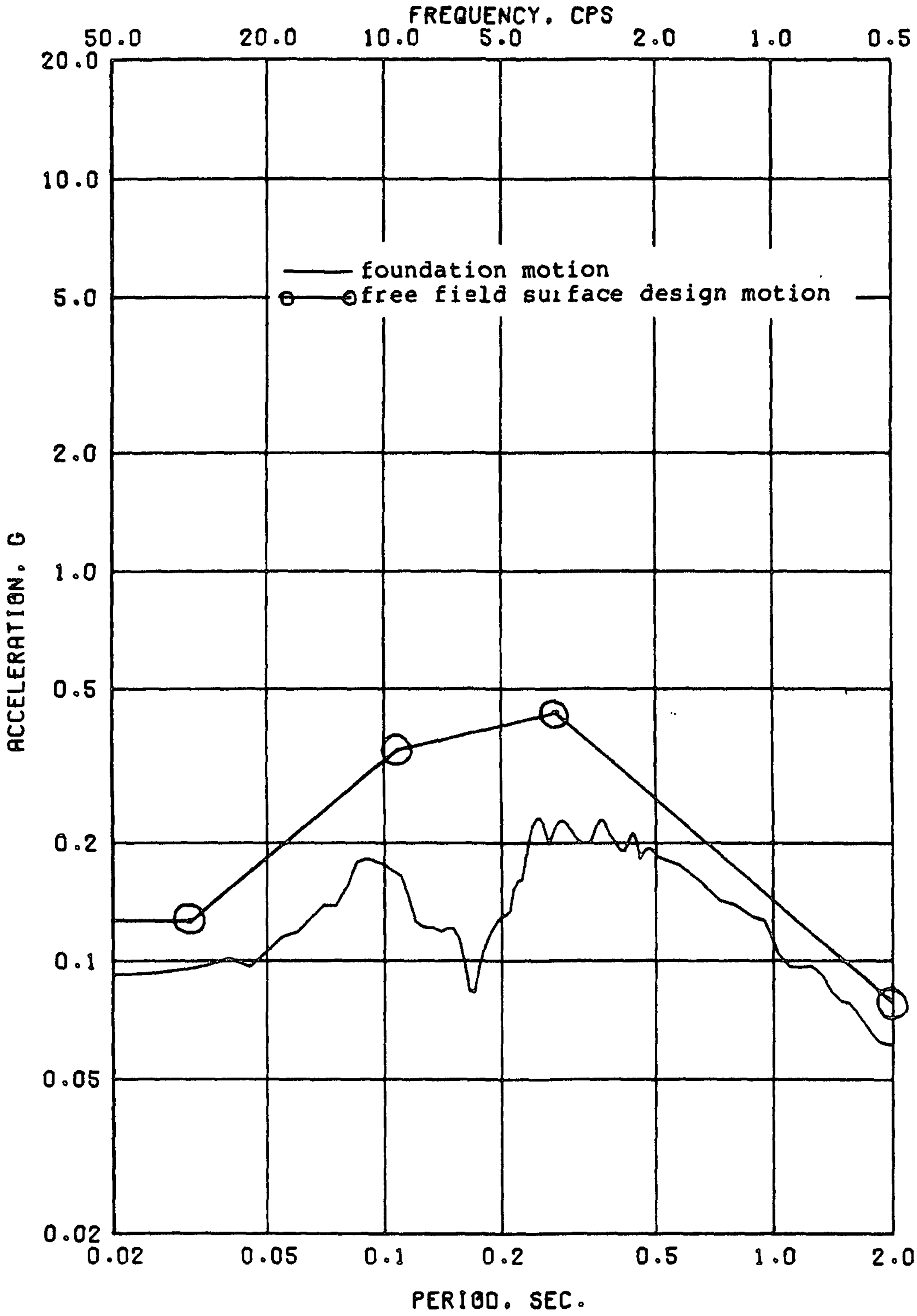

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FIGURE 3.7-38

VERTICAL FOUNDATION RESPONSE SPECTRUM OBE (4% DAMPING)

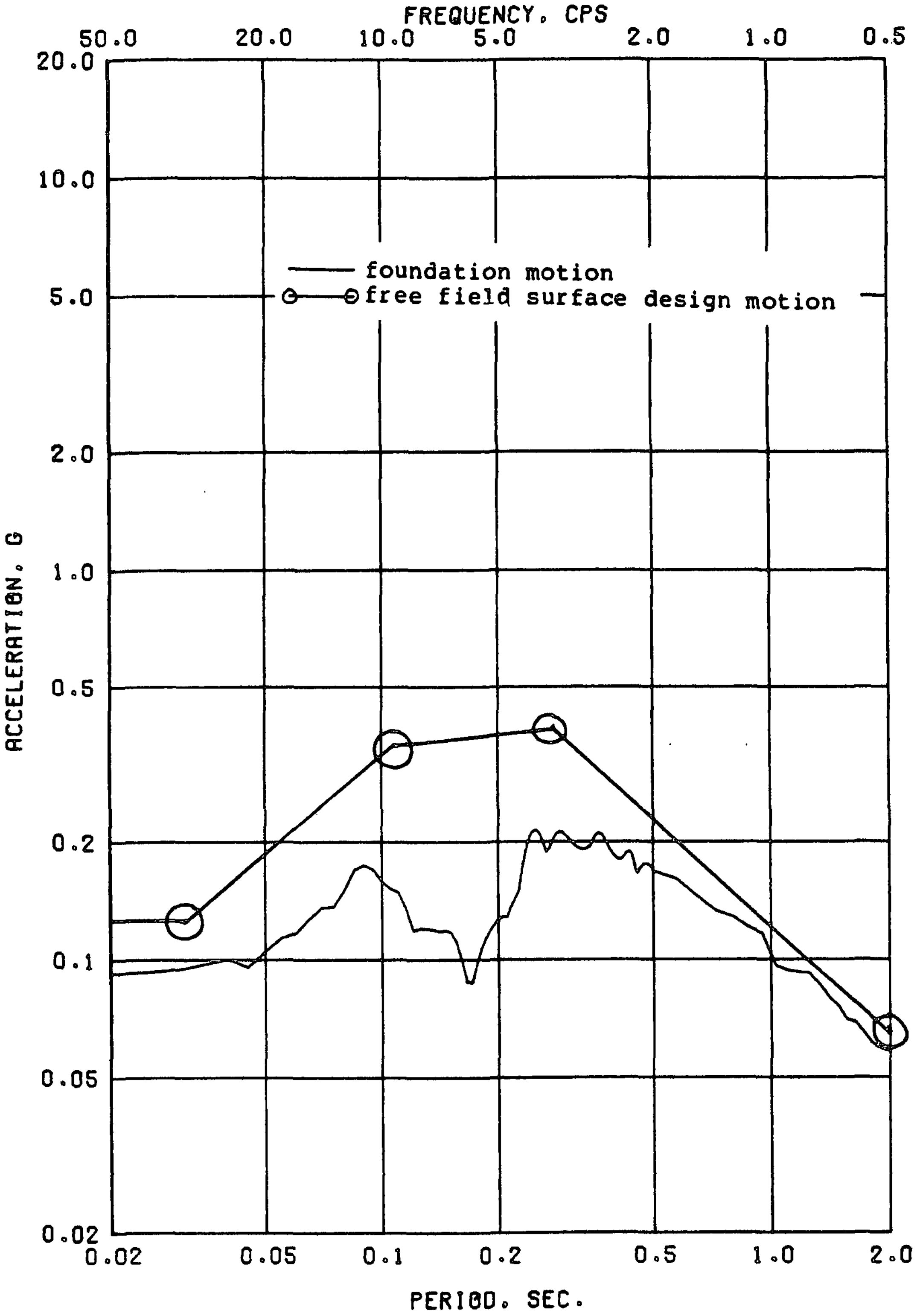

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FIGURE 3.7-39

VERTICAL FOUNDATION RESPONSE
SPECTRUM OBE (5% DAMPING)

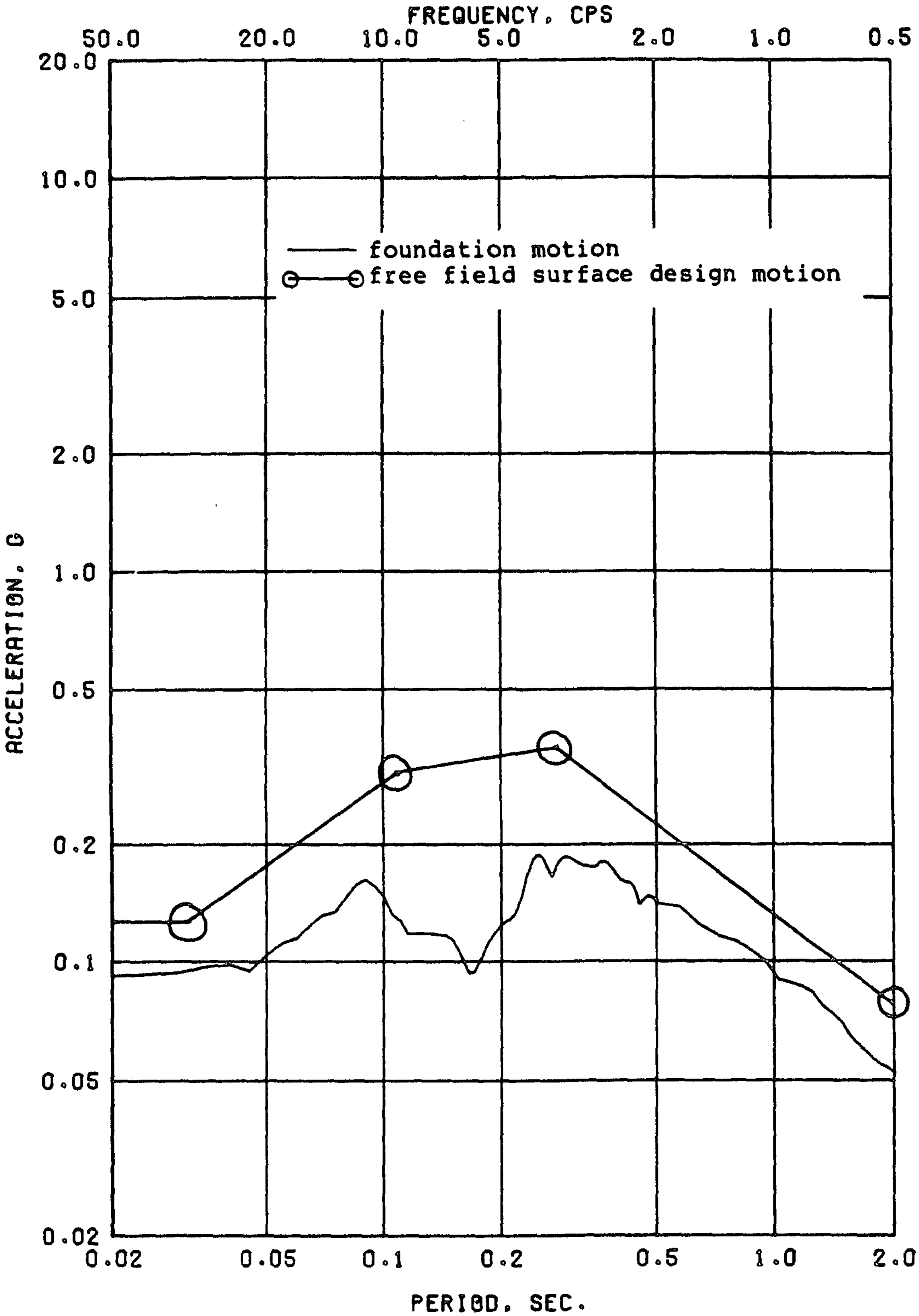

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FIGURE 3.7-40

VERTICAL FOUNDATION RESPONSE SPECTRUM OBE (7% DAMPING)

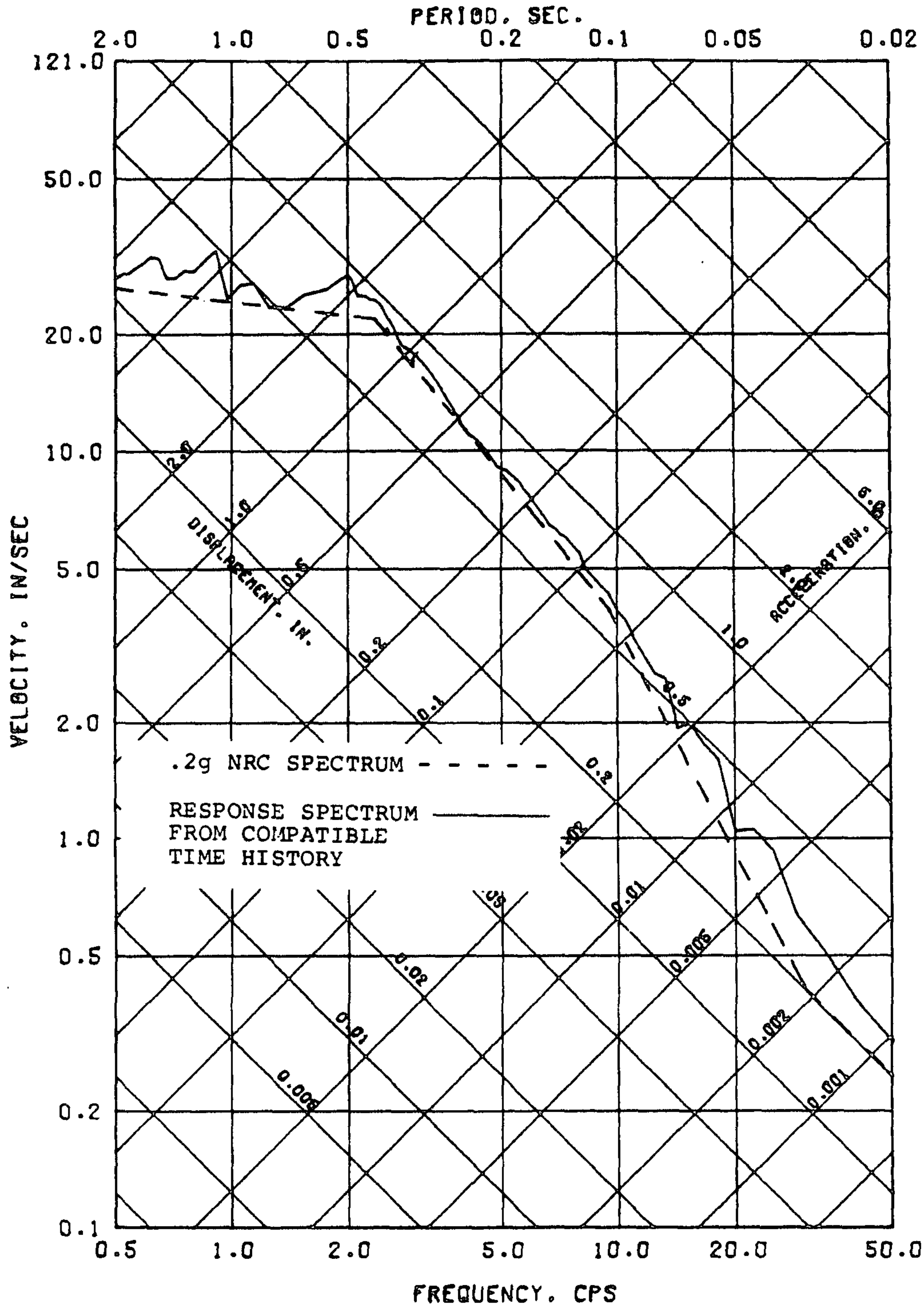

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FIGURE 3.7-41

HORIZONTAL RESPONSE SPECTRA
(2% DAMPING)

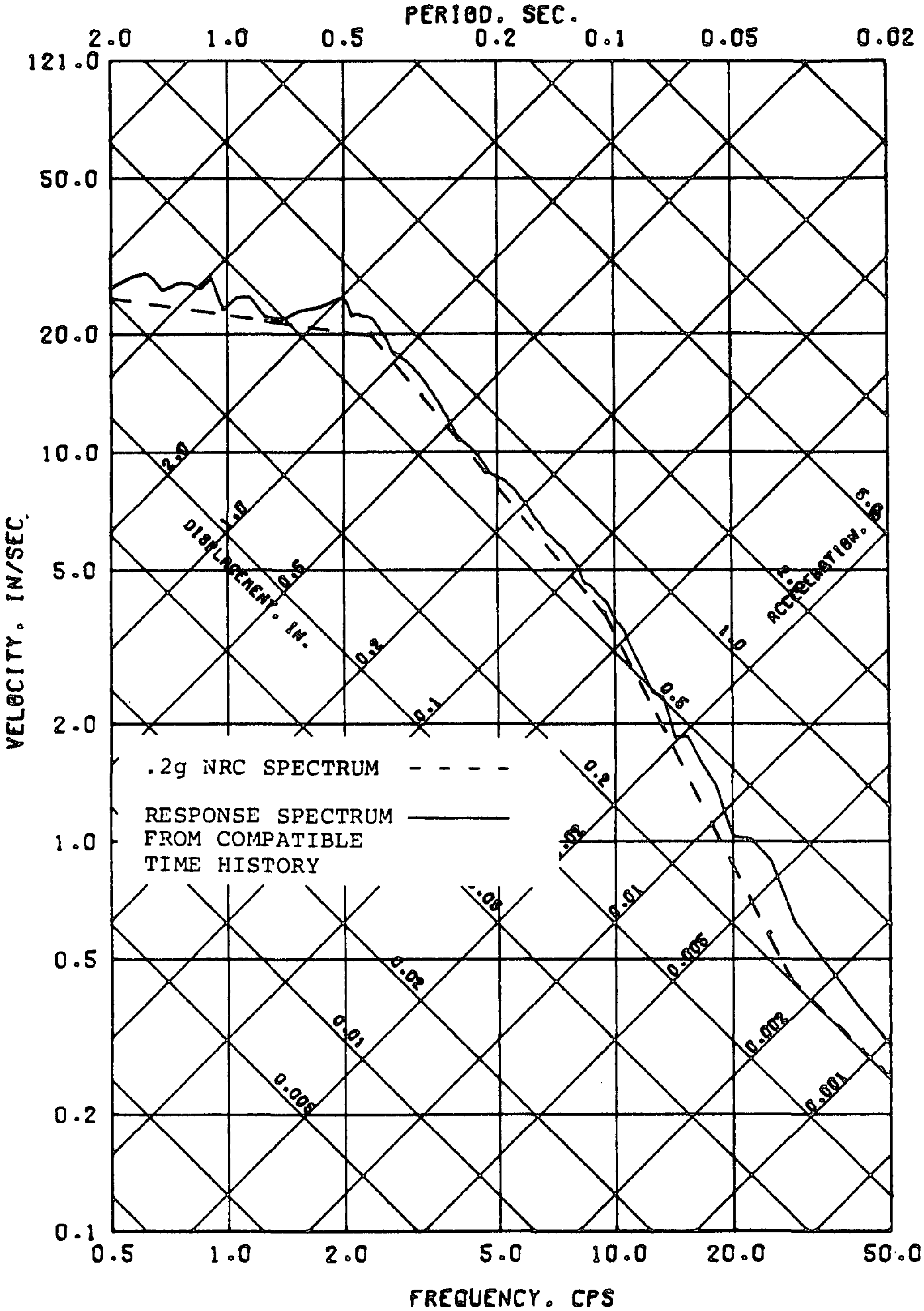

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FIGURE 3.7-42

HORIZONTAL RESPONSE SPECTRA
(3% DAMPING)

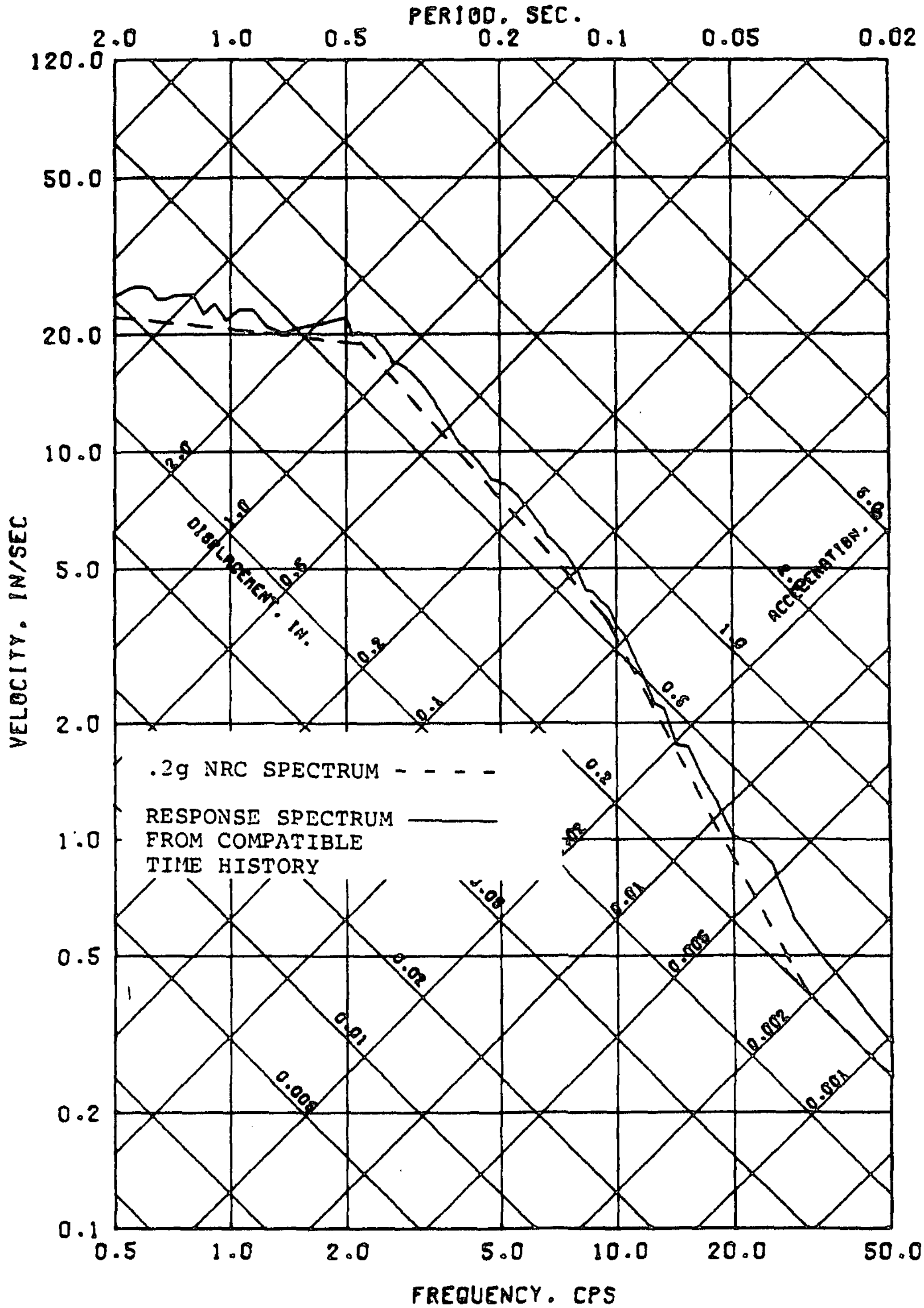

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FIGURE 3.7-43

HORIZONTAL RESPONSE SPECTRA
(4% DAMPING)

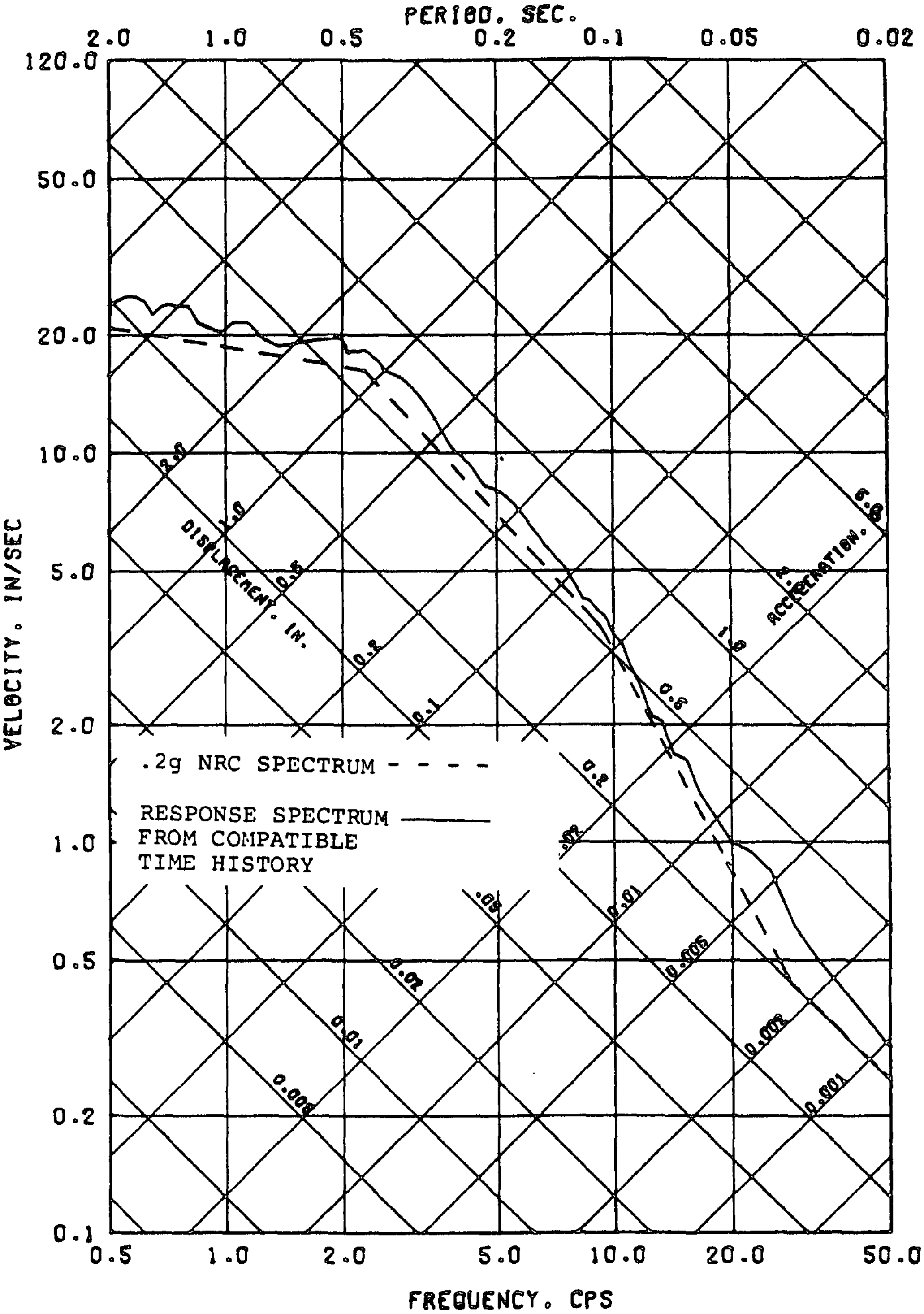

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FIGURE 3.7-44

HORIZONTAL RESPONSE SPECTRA
(5% DAMPING)

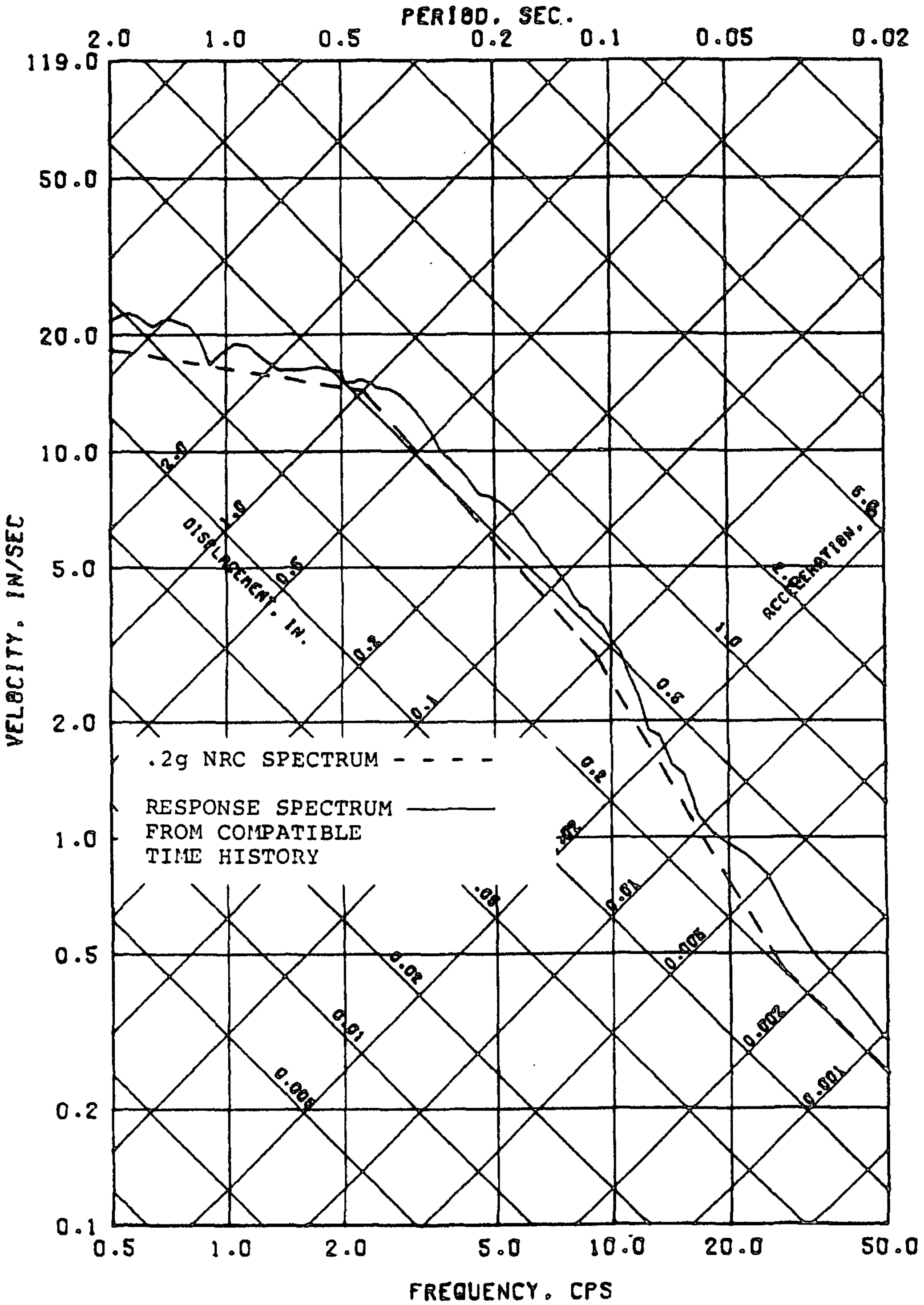

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FIGURE 3.7-45

HORIZONTAL RESPONSE SPECTRA
(7% DAMPING)

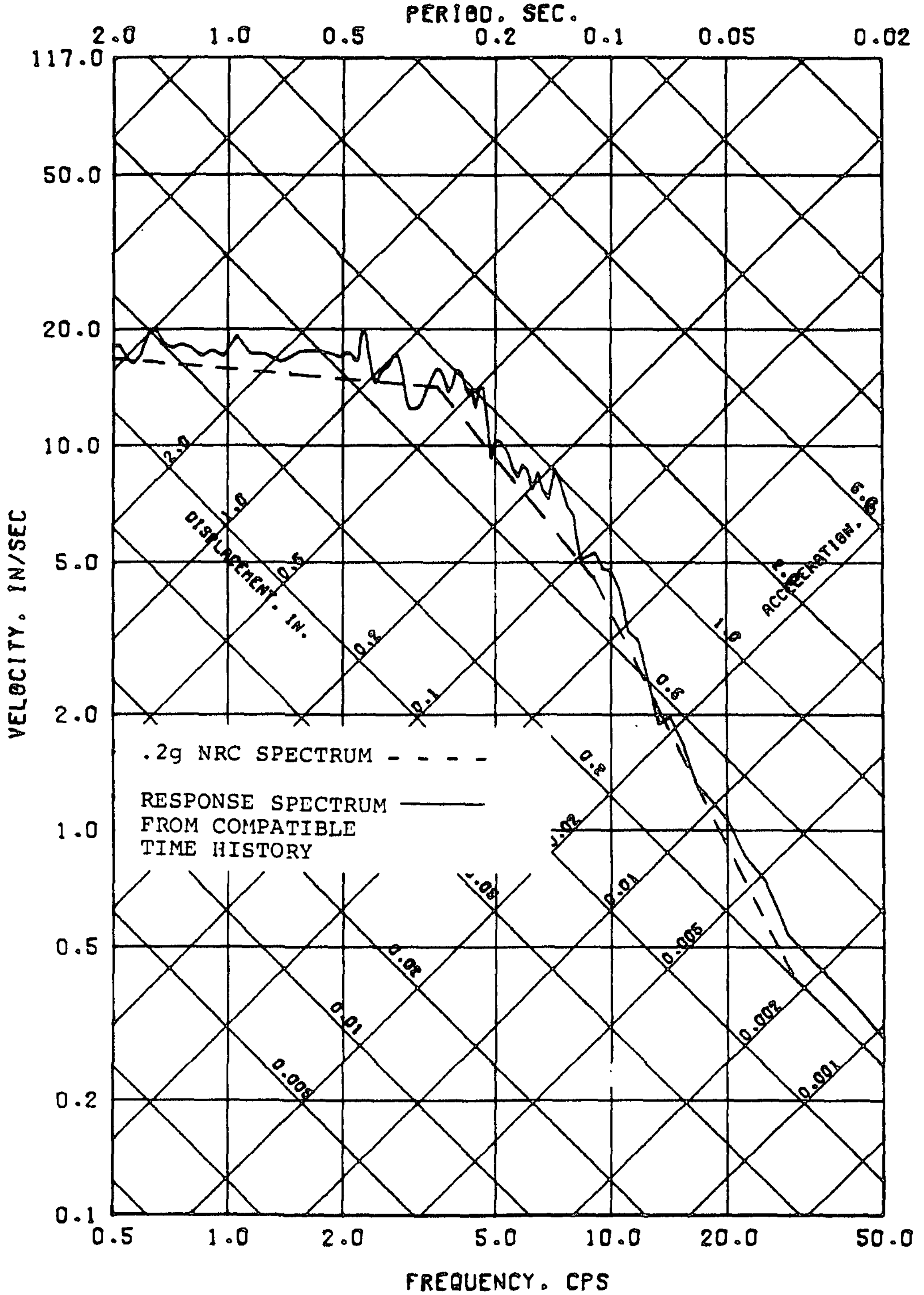

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FIGURE 3.7-46

VERTICAL RESPONSE SPECTRA
(2% DAMPING)

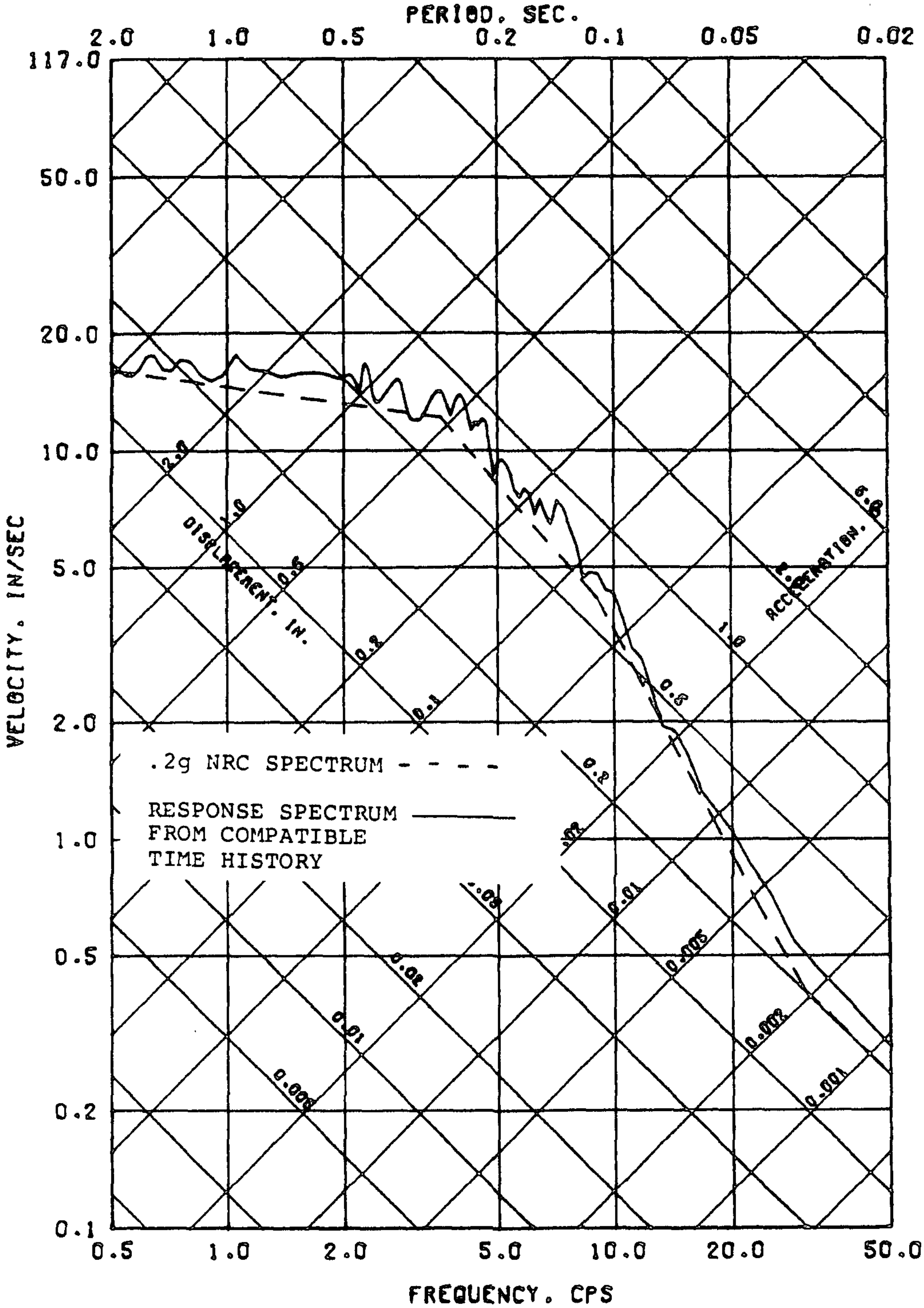

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FIGURE 3.7-47

VERTICAL RESPONSE SPECTRA
(3% DAMPING)

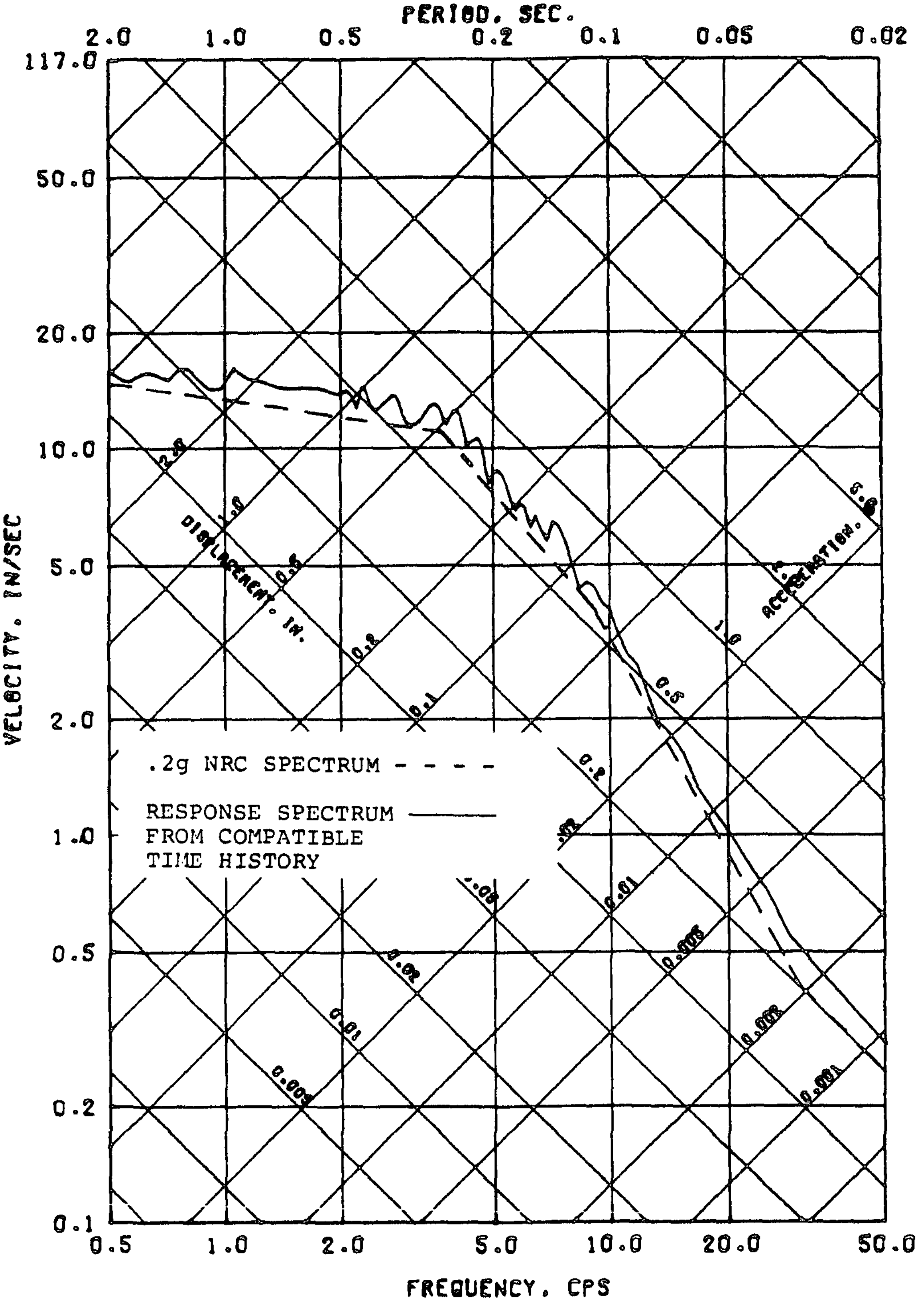

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FIGURE 3.7-48

VERTICAL RESPONSE SPECTRA
(4% DAMPING)

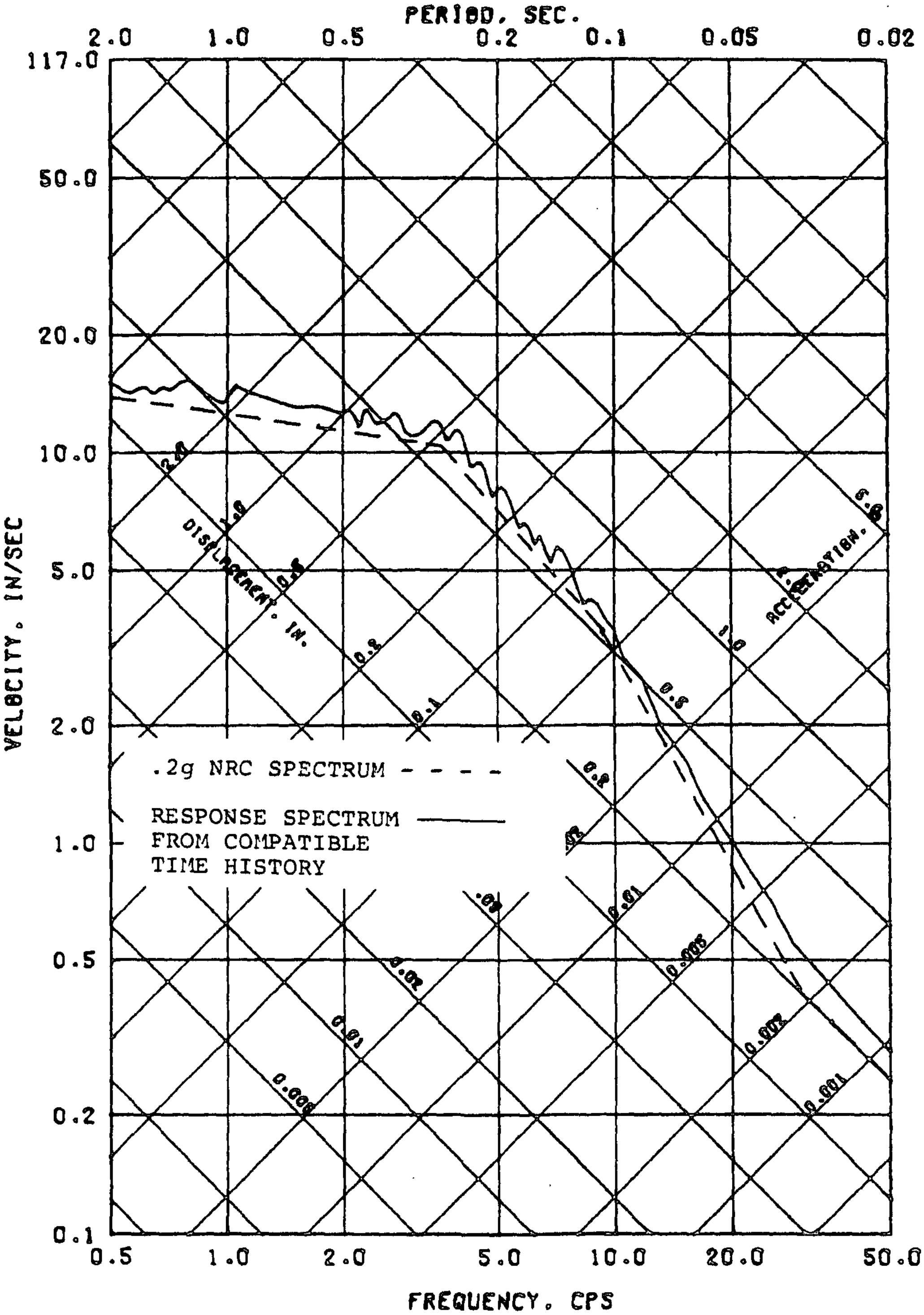

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FIGURE 3.7-49

VERTICAL RESPONSE SPECTRA
(5% DAMPING)

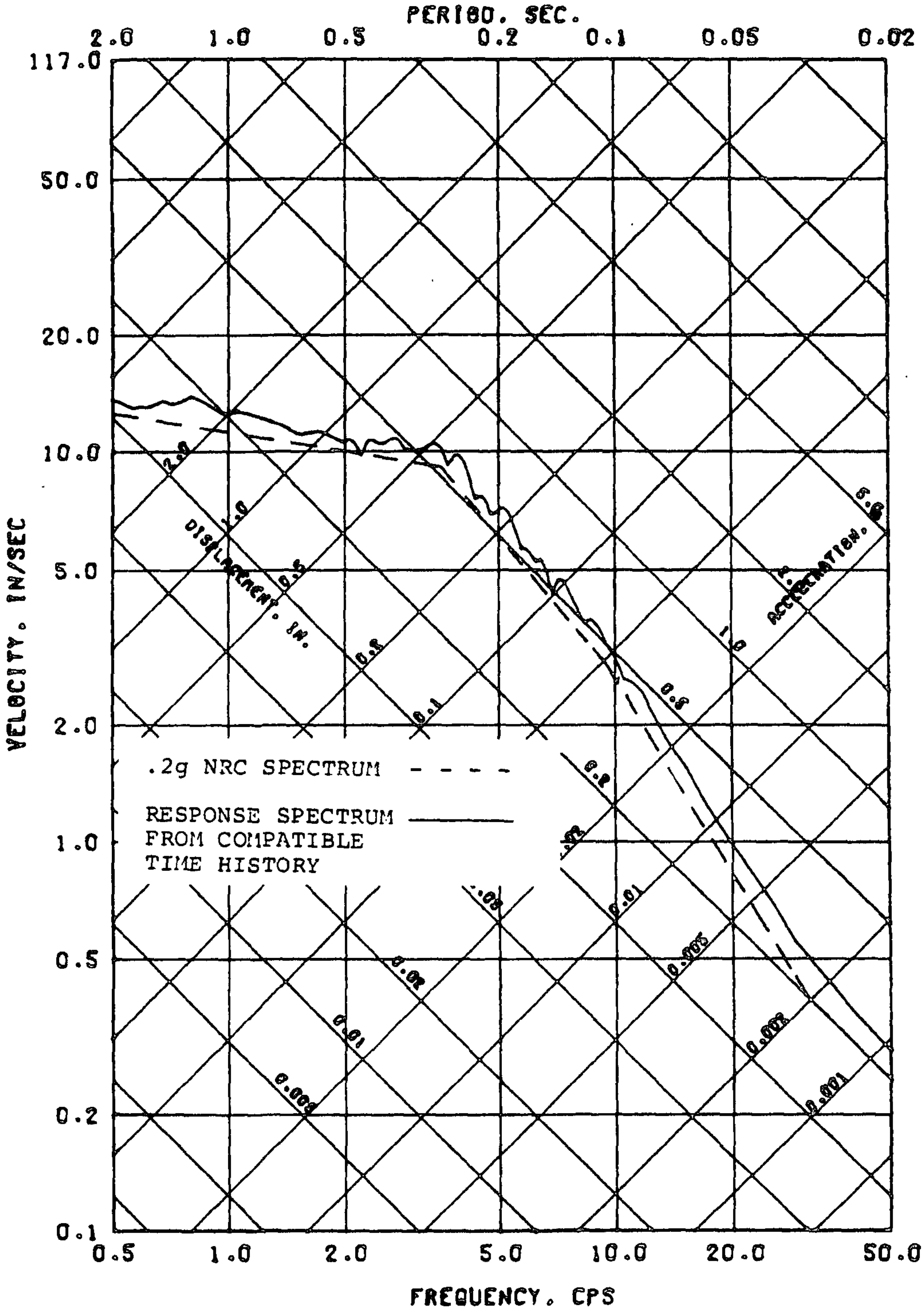

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FIGURE 3.7-50

VERTICAL RESPONSE SPECTRA
(7% DAMPING)

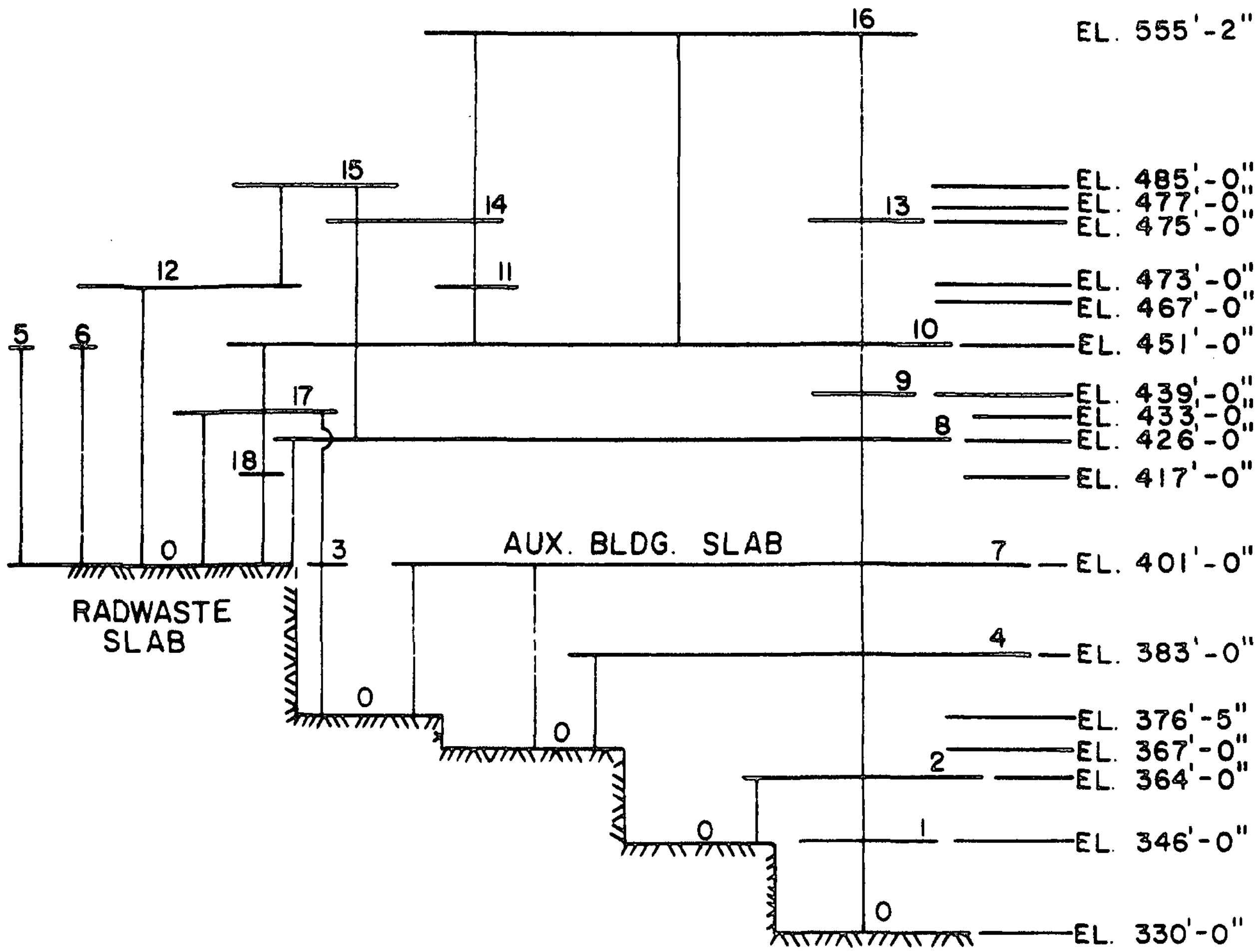

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FIGURE 3.7-51

BYRON SEISMIC MODEL
(HORIZONTAL)

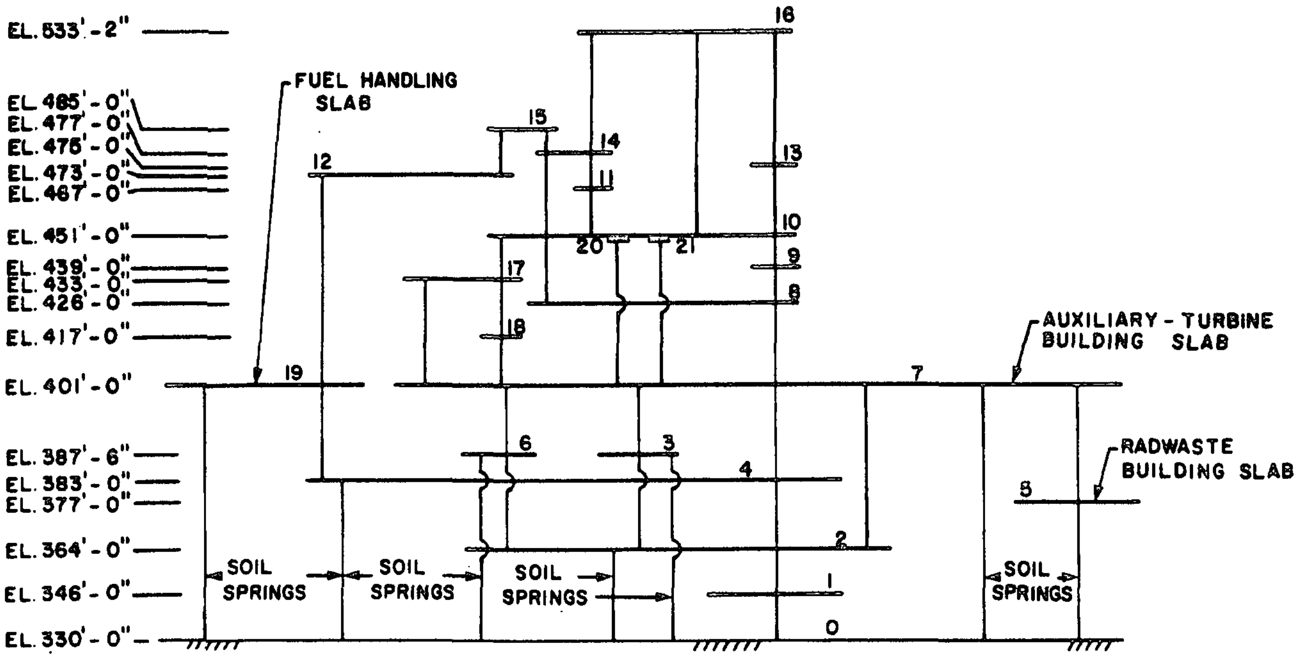

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FIGURE 3.7-52

BRAIDWOOD SEISMIC MODEL
(HORIZONTAL)

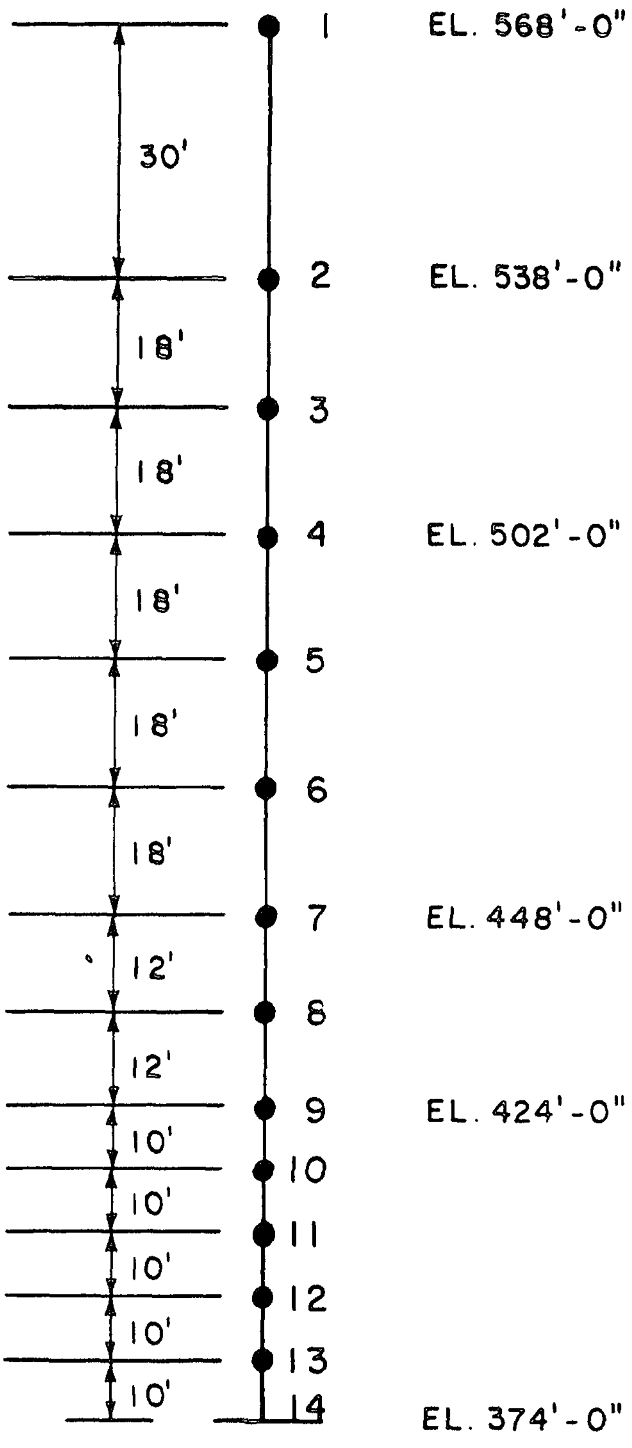

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FIGURE 3.7-53

CONTAINMENT BUILDING SEISMIC MODEL
(BYRON/BRAIDWOOD)

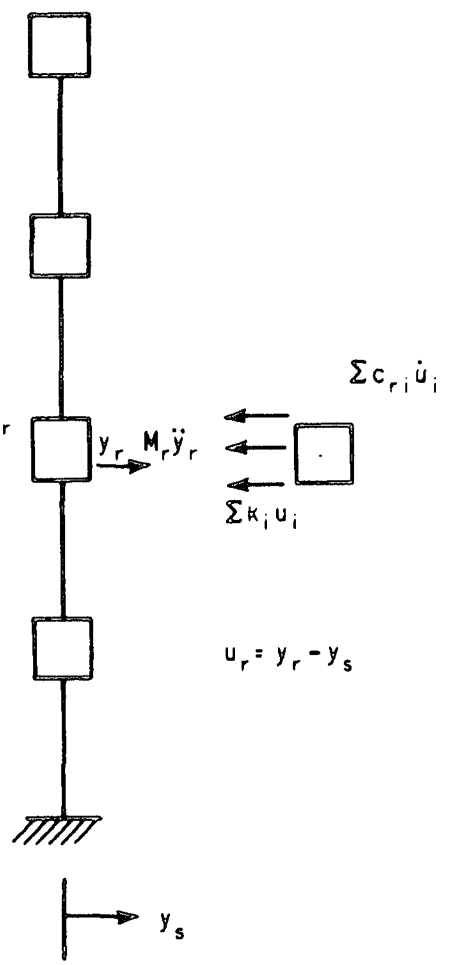

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FIGURE 3.7-54

MULTI-DEGREE OF FREEDOM SYSTEM

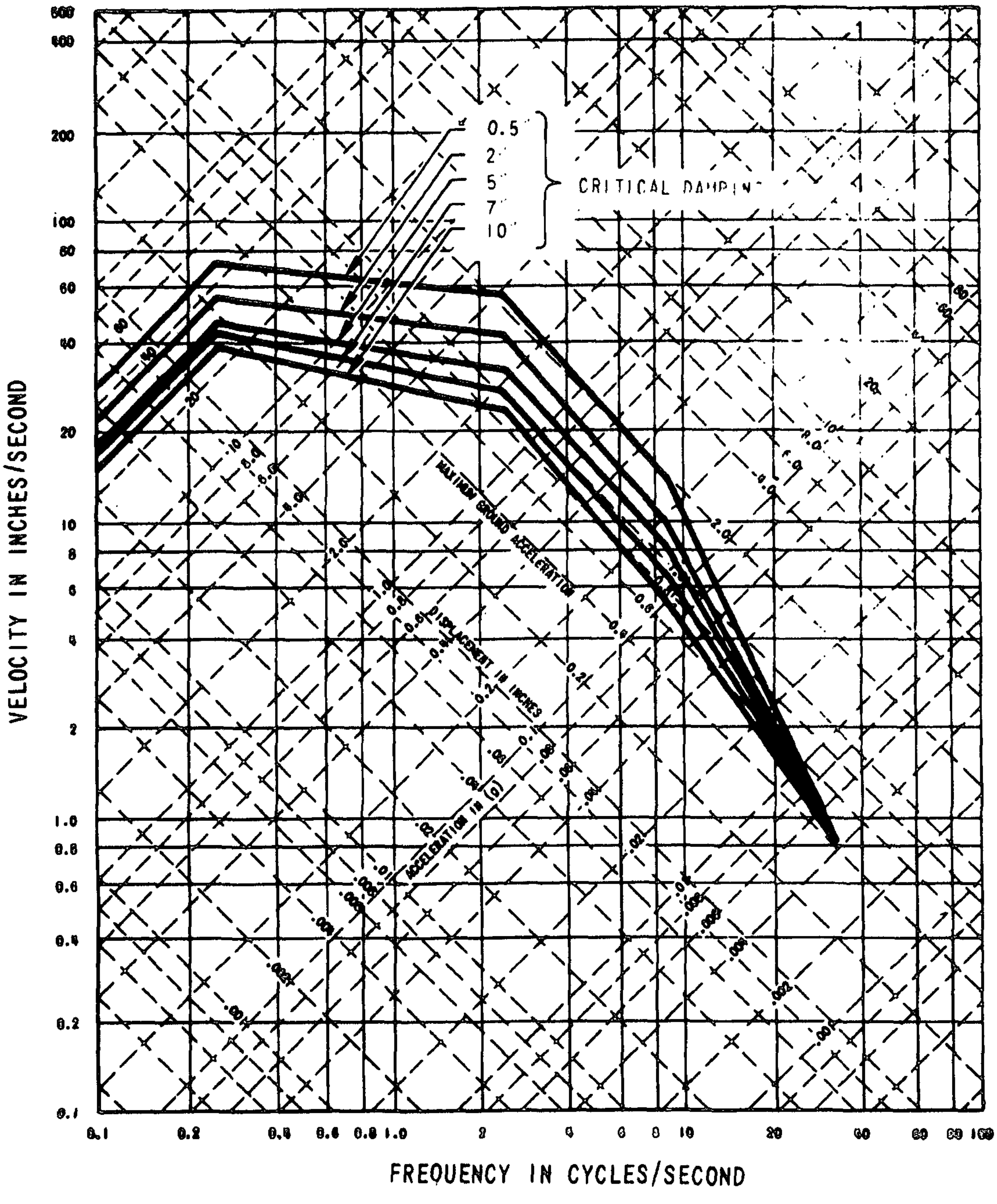

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FIGURE 3.7-55

HORIZONTAL DESIGN RESPONSE SPECTRA
SCALED TO 0.4 G HORIZONTAL
GROUND ACCELERATION

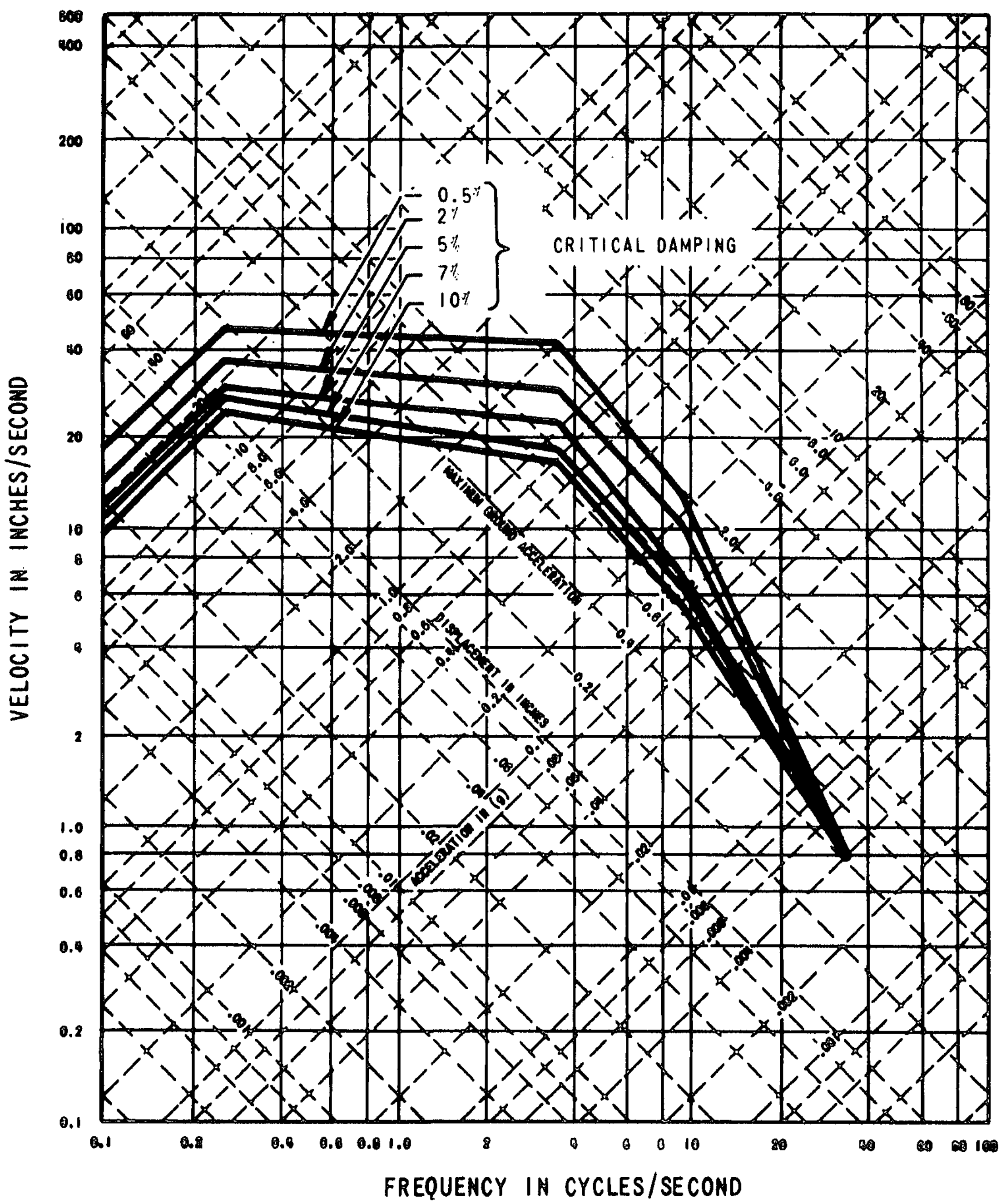

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FIGURE 3.7-56

VERTICAL DESIGN RESPONSE SPECTRA
SCALED TO 0.4 G HORIZONTAL
GROUND ACCELERATION

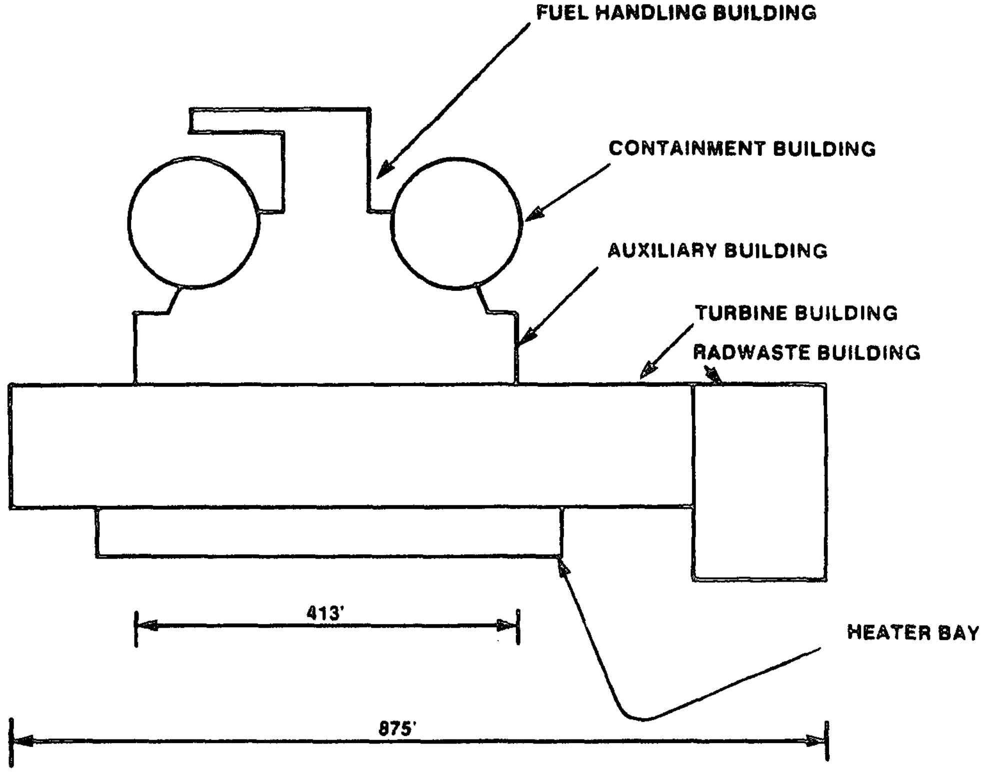

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FIGURE 3.7-57

PLANT LAYOUT

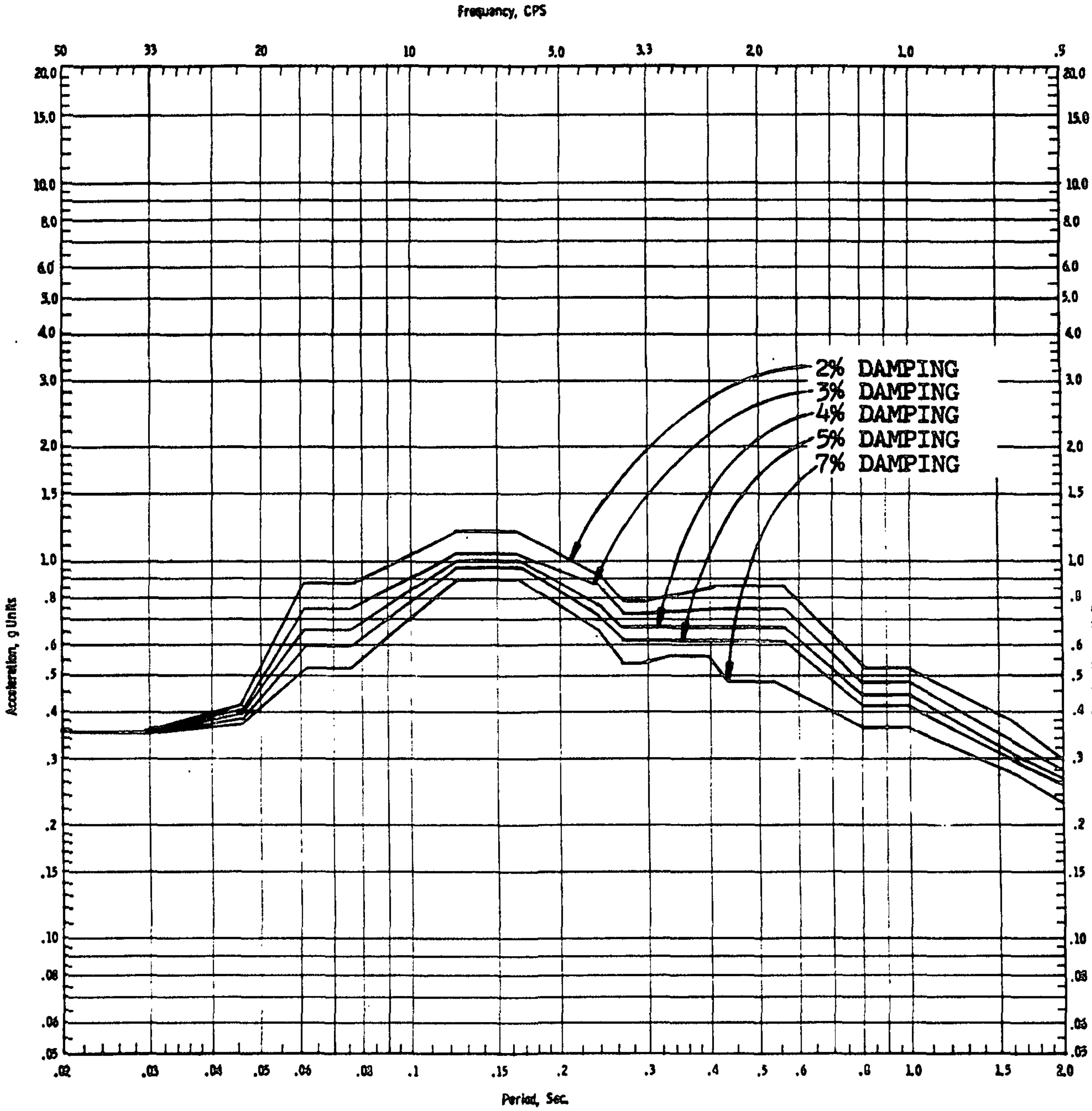

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FIGURE 3.7-58

HORIZONTAL FLOOR RESPONSE SPECTRA
N-S COMPONENT (SSE/BLAST) EL. 401'-0"
AUXILIARY—HEATER BAY—TURBINE

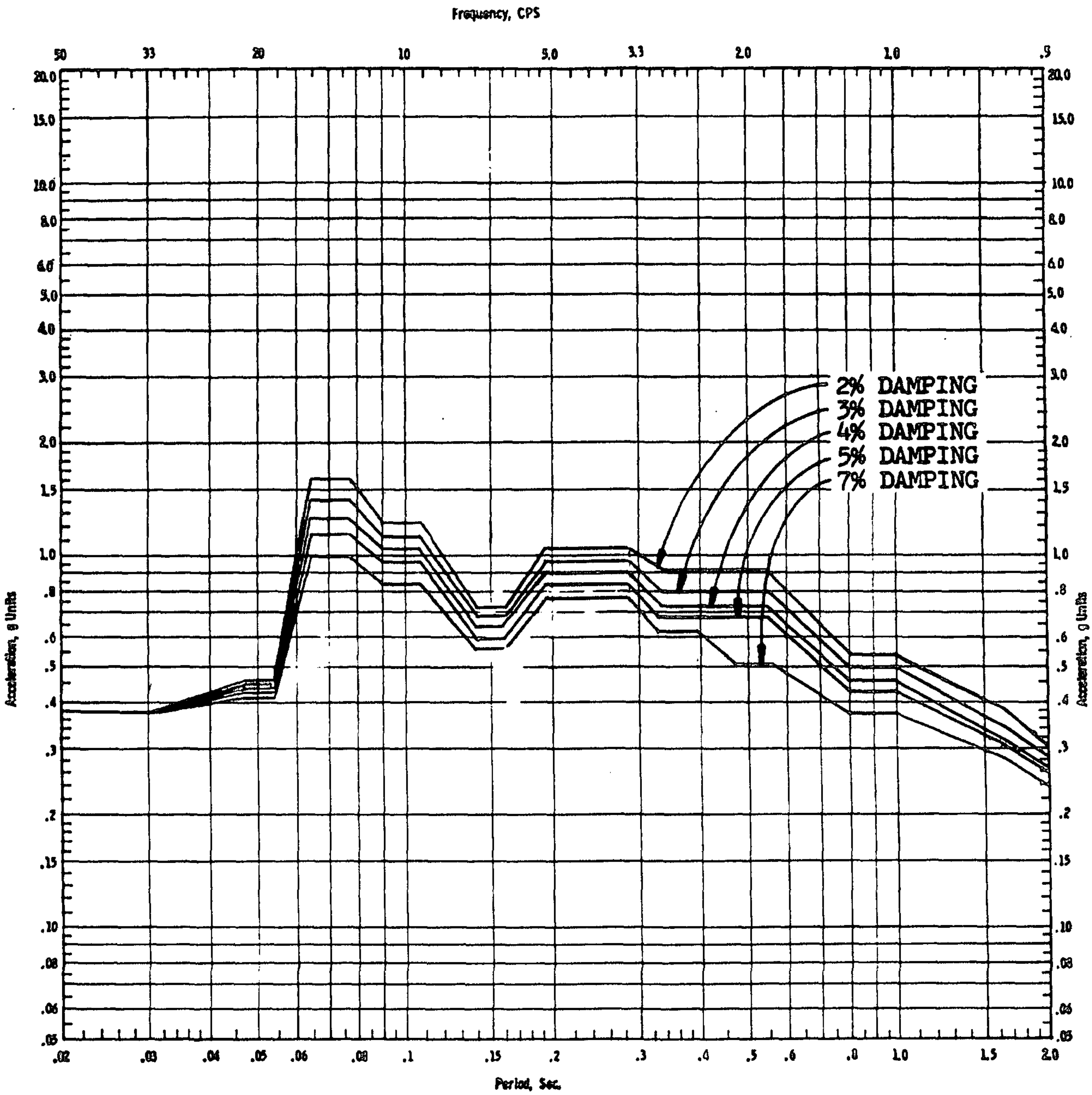

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FIGURE 3.7-59

HORIZONTAL FLOOR RESPONSE SPECTRA
E-W COMPONENT (SSE/BLAST) EL. 401'-0"
AUXILIARY—HEATER BAY—TURBINE

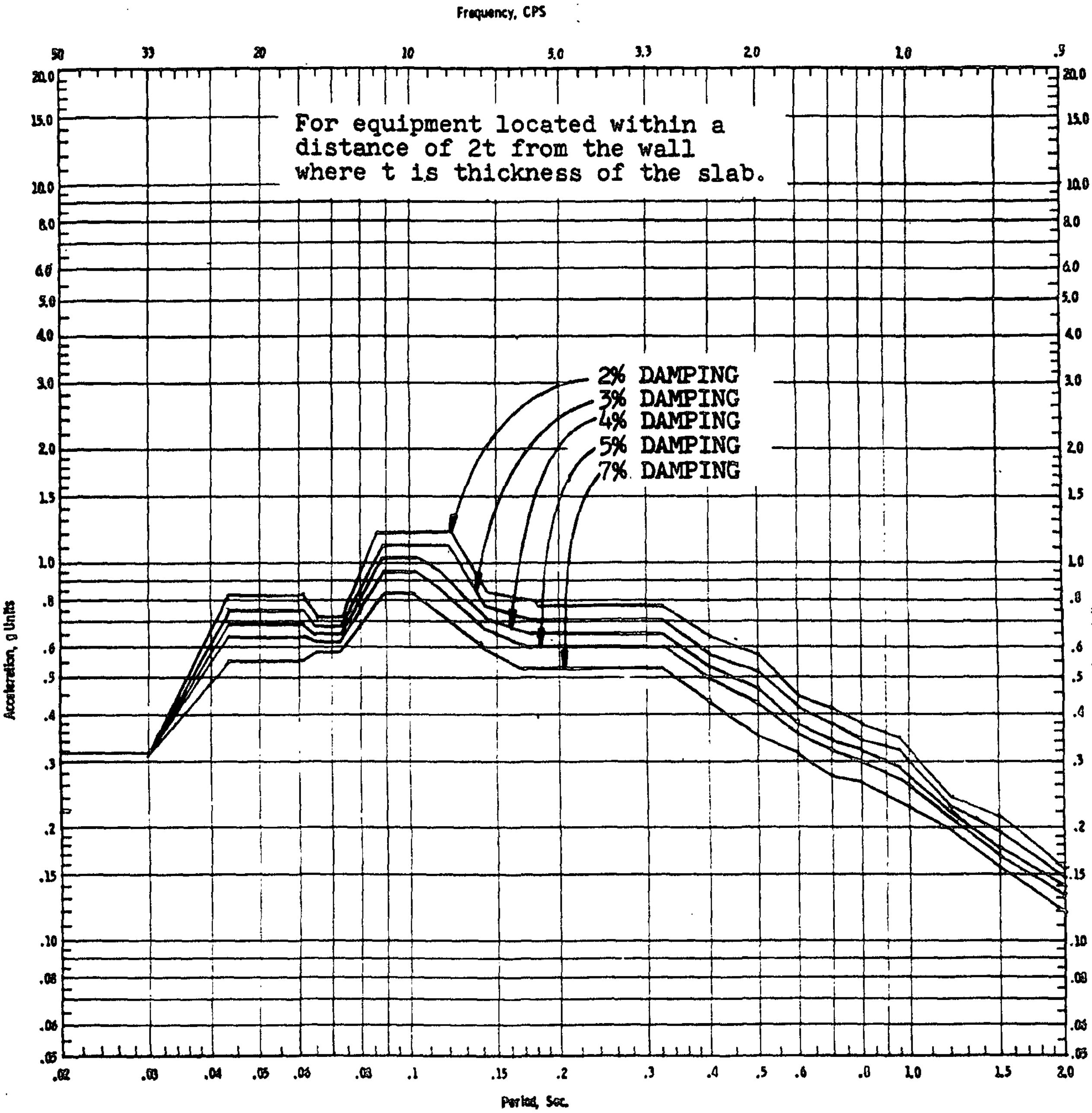

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FIGURE 3.7-60

VERTICAL RESPONSE SPECTRA (SSE/BLAST)
EL. 346'-0", 364'-0", 383'-0", & 401'-0"
AUXILIARY BUILDING WALL

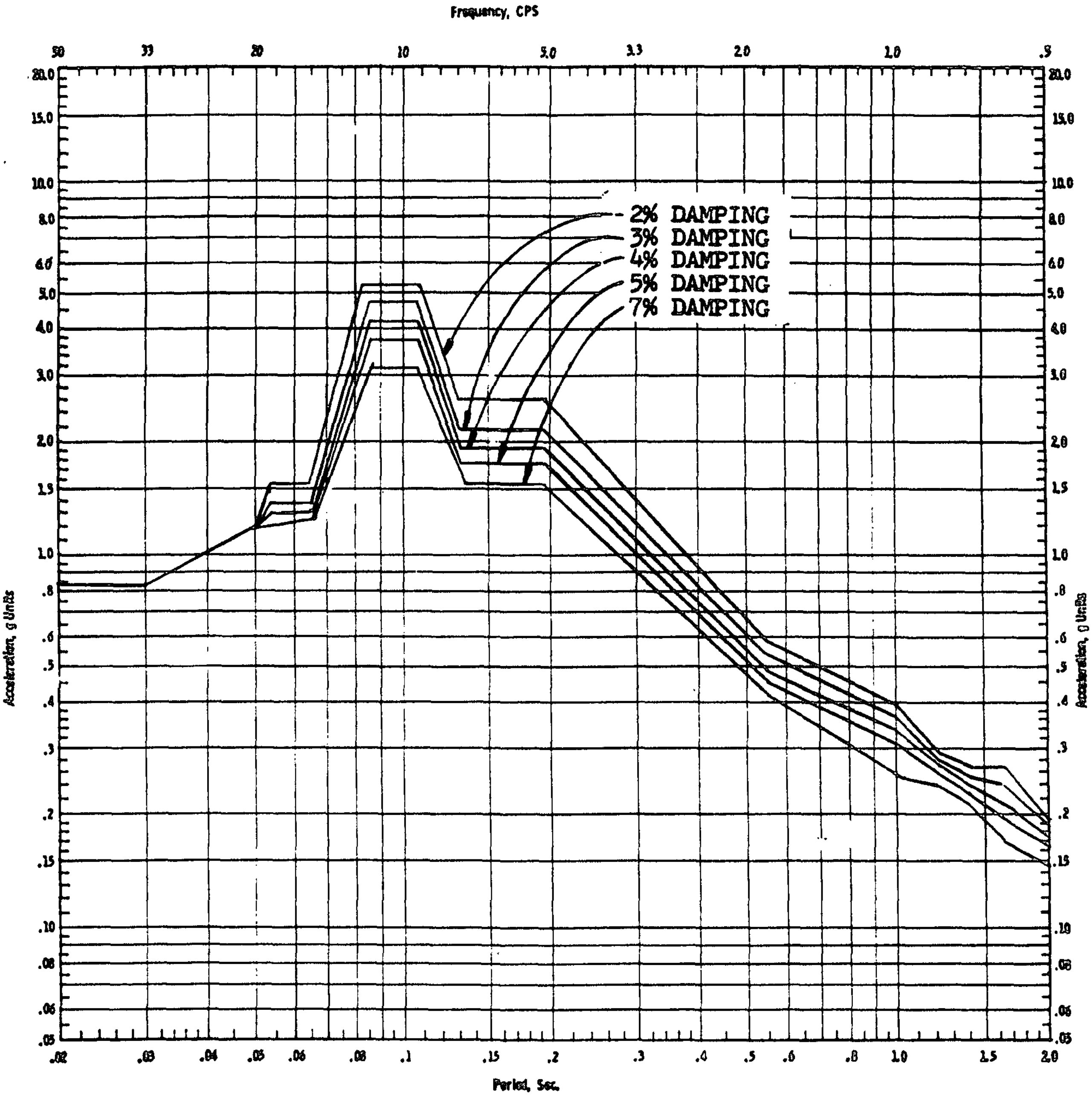

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FIGURE 3.7-61

VERTICAL RESPONSE SPECTRA (SSE/BLAST)
EL. 346'-0", 364'-0", 383'-0", & 401'-0"
AUXILIARY BUILDING SLAB

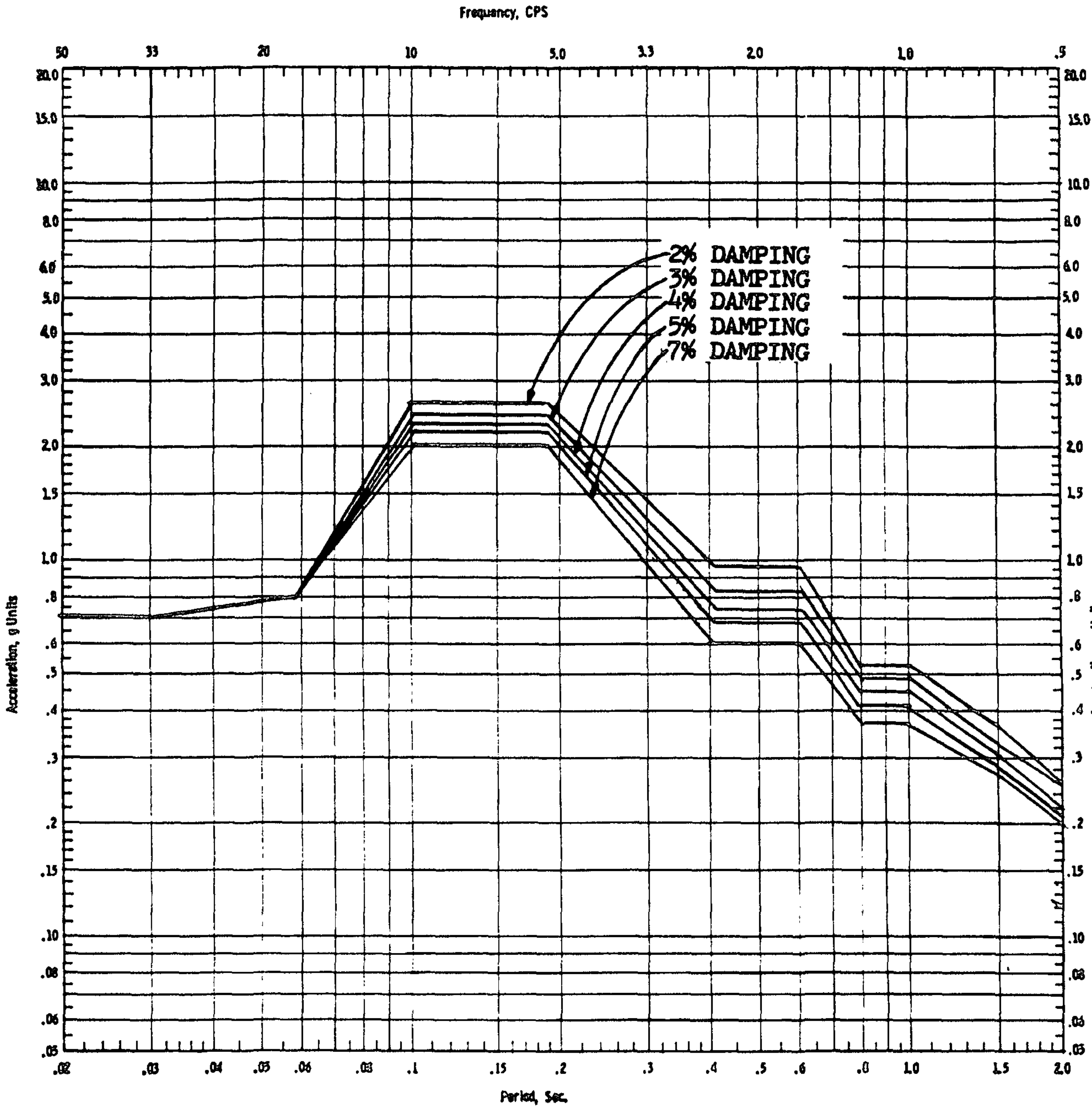

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FIGURE 3.7-62

HORIZONTAL FLOOR RESPONSE SPECTRA
N-S COMPONENT (SSE/BLAST) EL. 451'-0"
AUXILIARY—FUEL HANDLING BUILDING

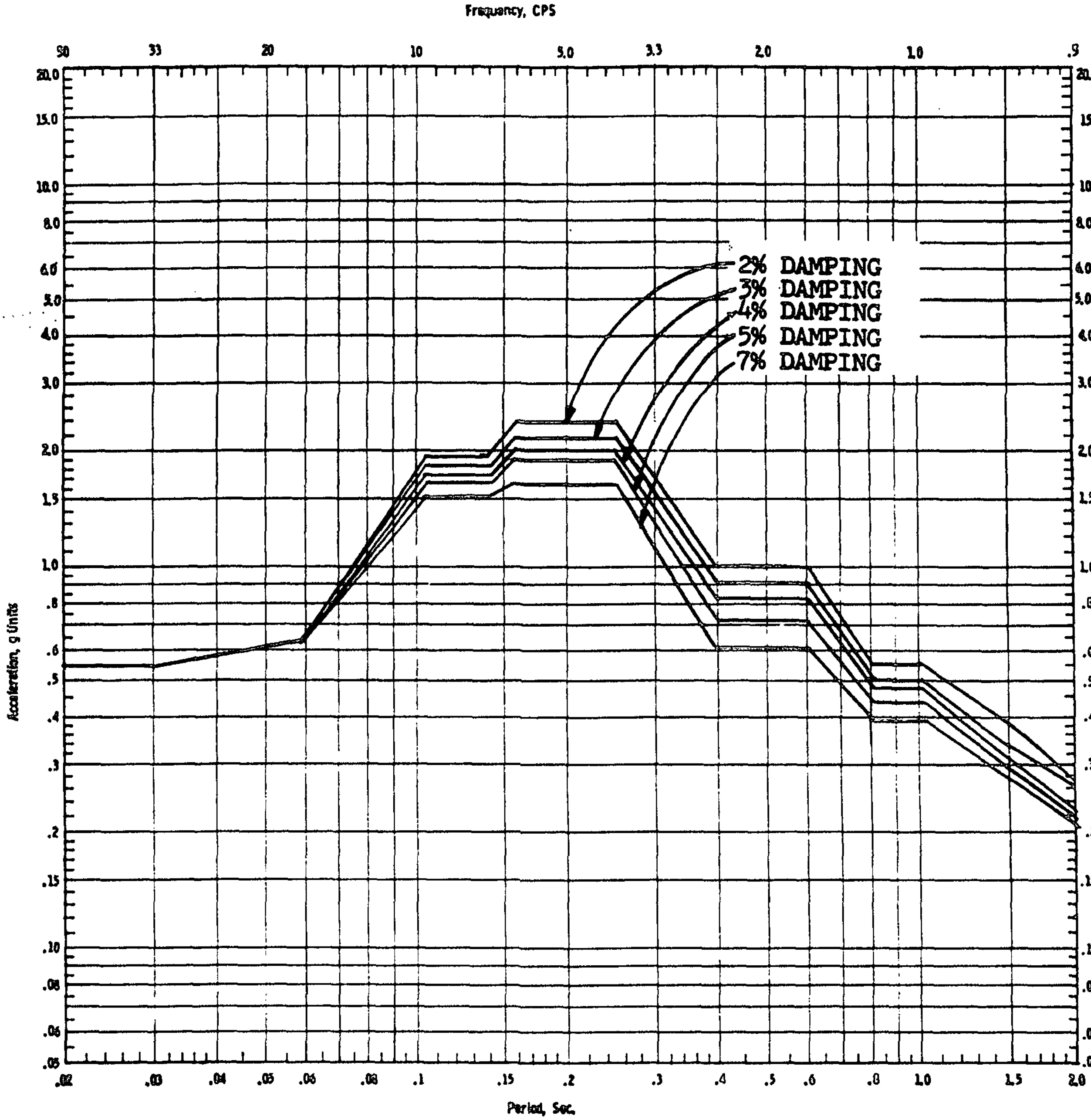

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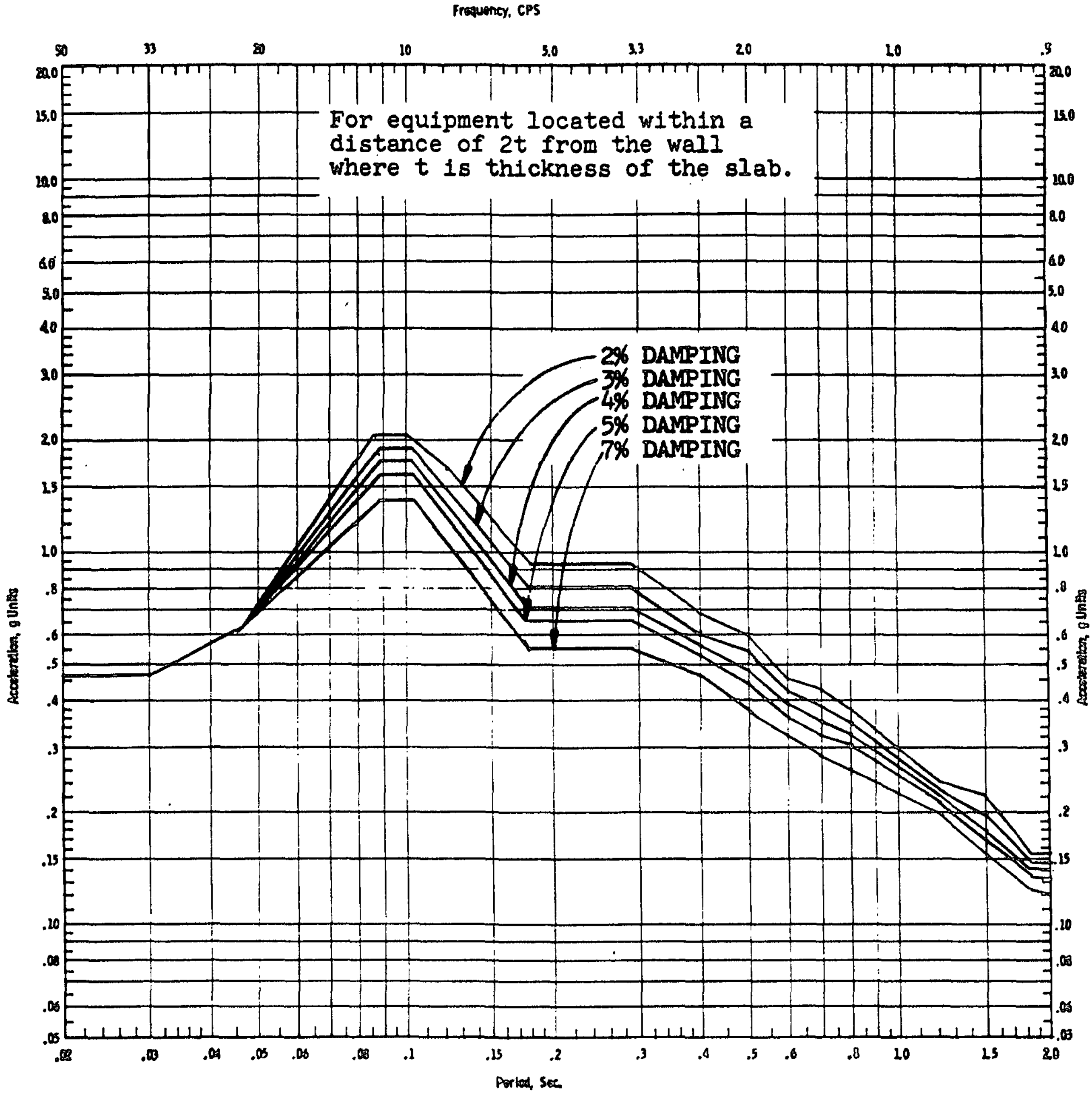
FIGURE 3.7-63

HORIZONTAL FLOOR RESPONSE SPECTRA
E-W COMPONENT (SSE/BLAST) EL. 451'-0"
AUXILIARY—FUEL HANDLING BUILDING



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7-64

VERTICAL RESPONSE SPECTRA (SSE/BLAST)
EL. 426'-0", 439'-0", & 451'-0"
AUXILIARY BUILDING WALL

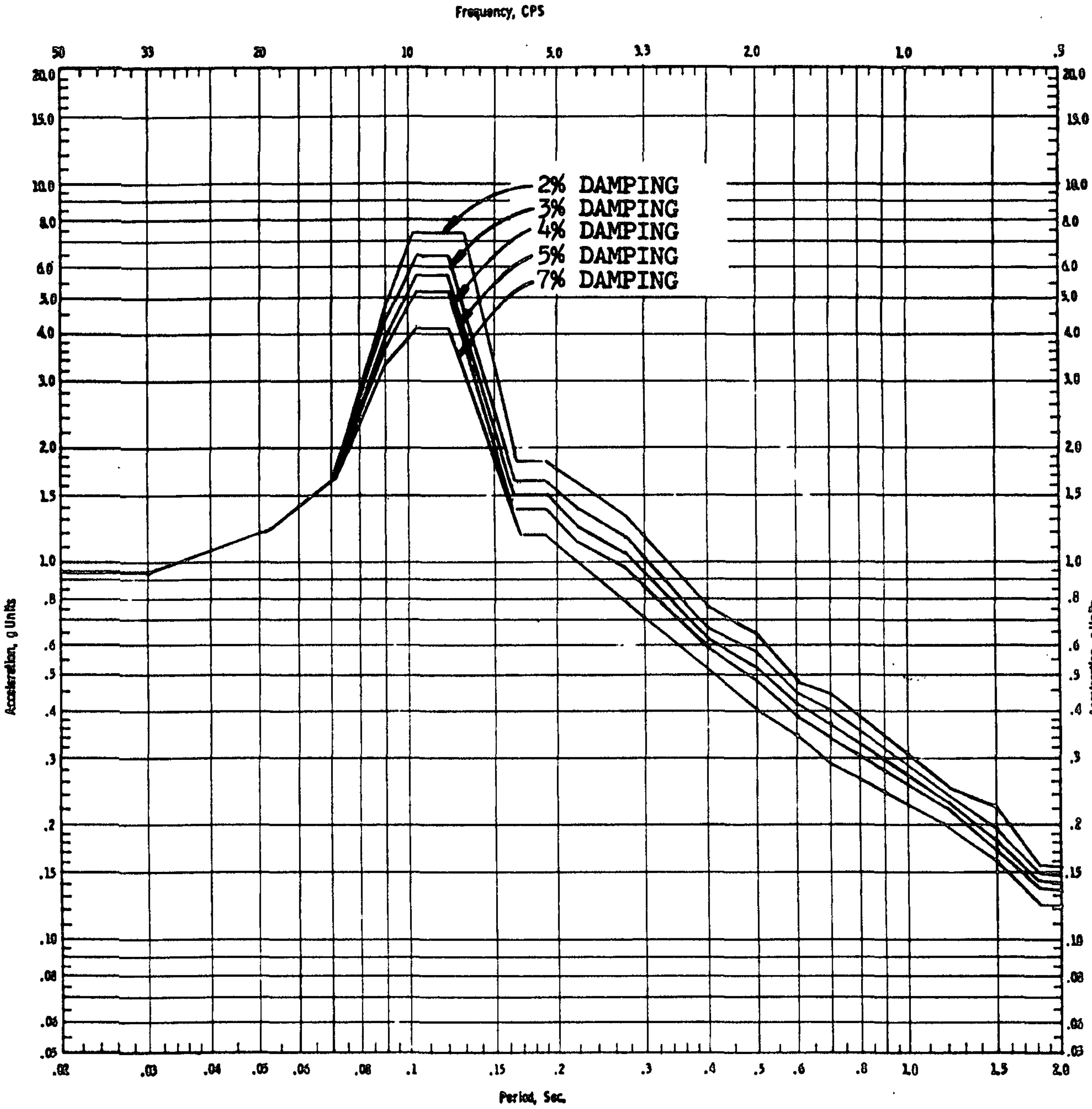

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FIGURE 3.7-65

VERTICAL RESPONSE SPECTRA (SSE/BLAST)
EL. 426'-0", 439'-0", & 451'-0"
AUXILIARY BUILDING SLAB

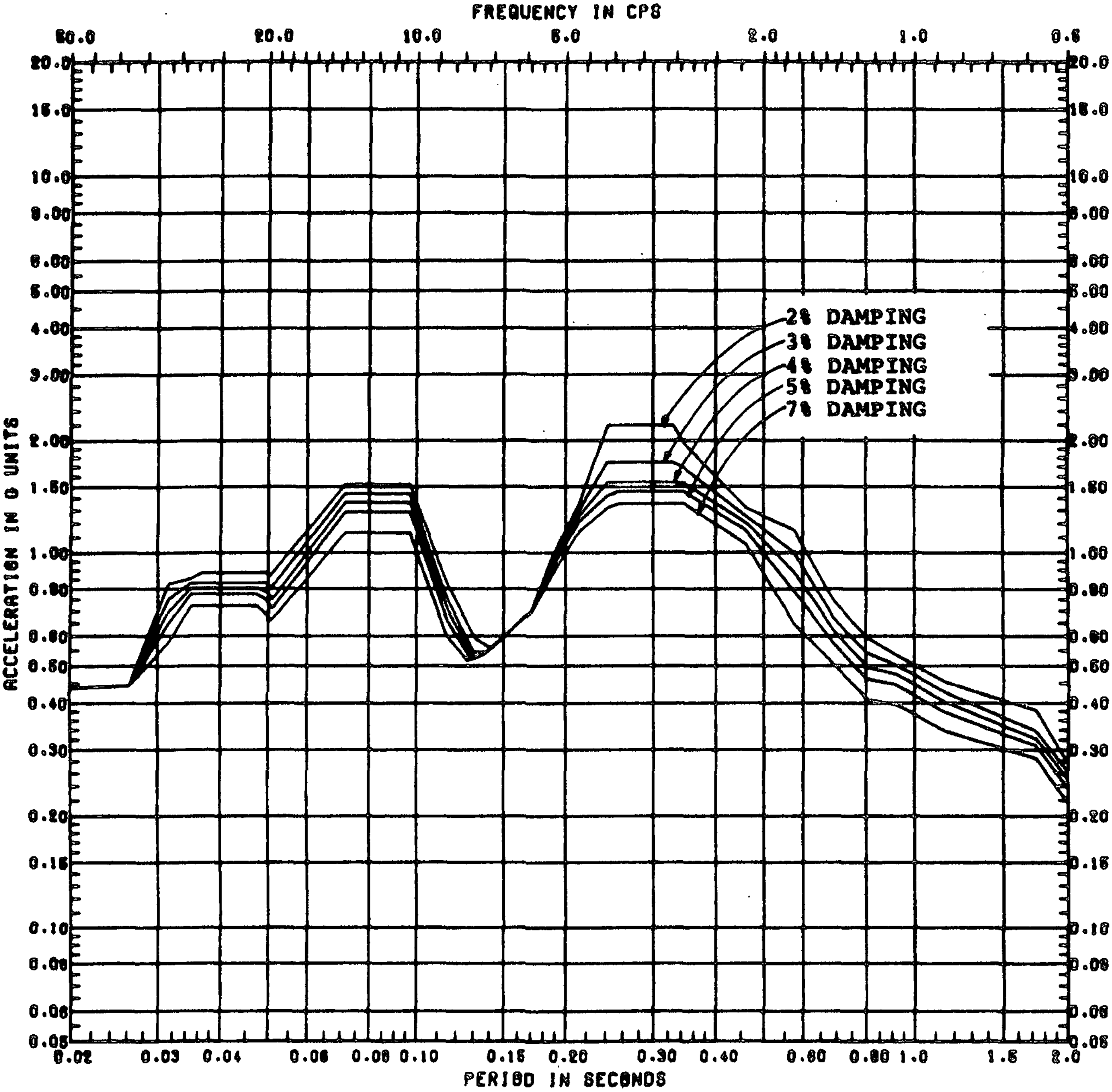

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FIGURE 3.7-66

HORIZONTAL RESPONSE SPECTRA SSE
EL. 500'-0"
CONTAINMENT BUILDING CRANE SUPPORT

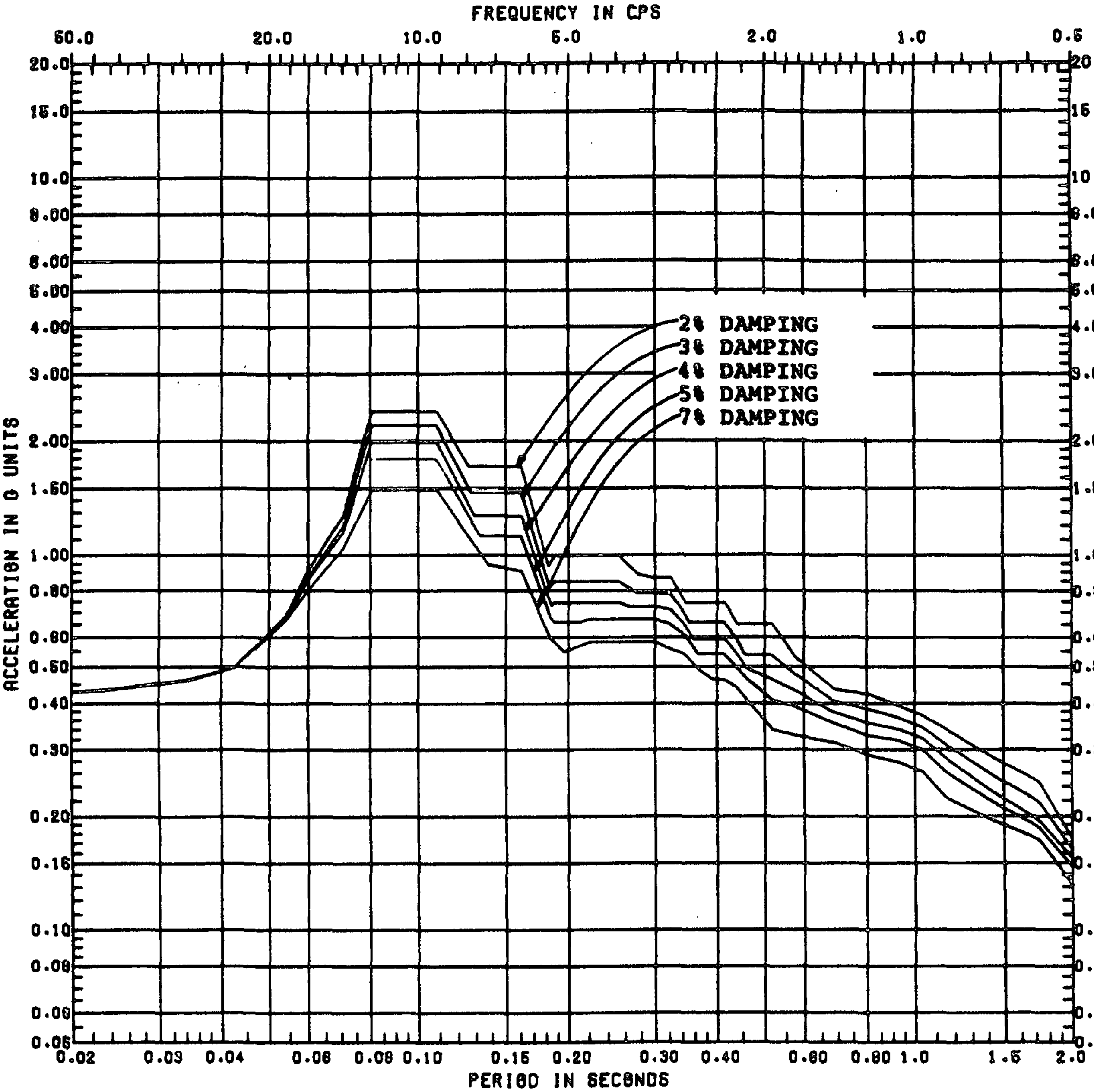

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FIGURE 3.7-67

VERTICAL RESPONSE SPECTRA SSE
EL. 500'-0"
CONTAINMENT BUILDING CRANE SUPPORT

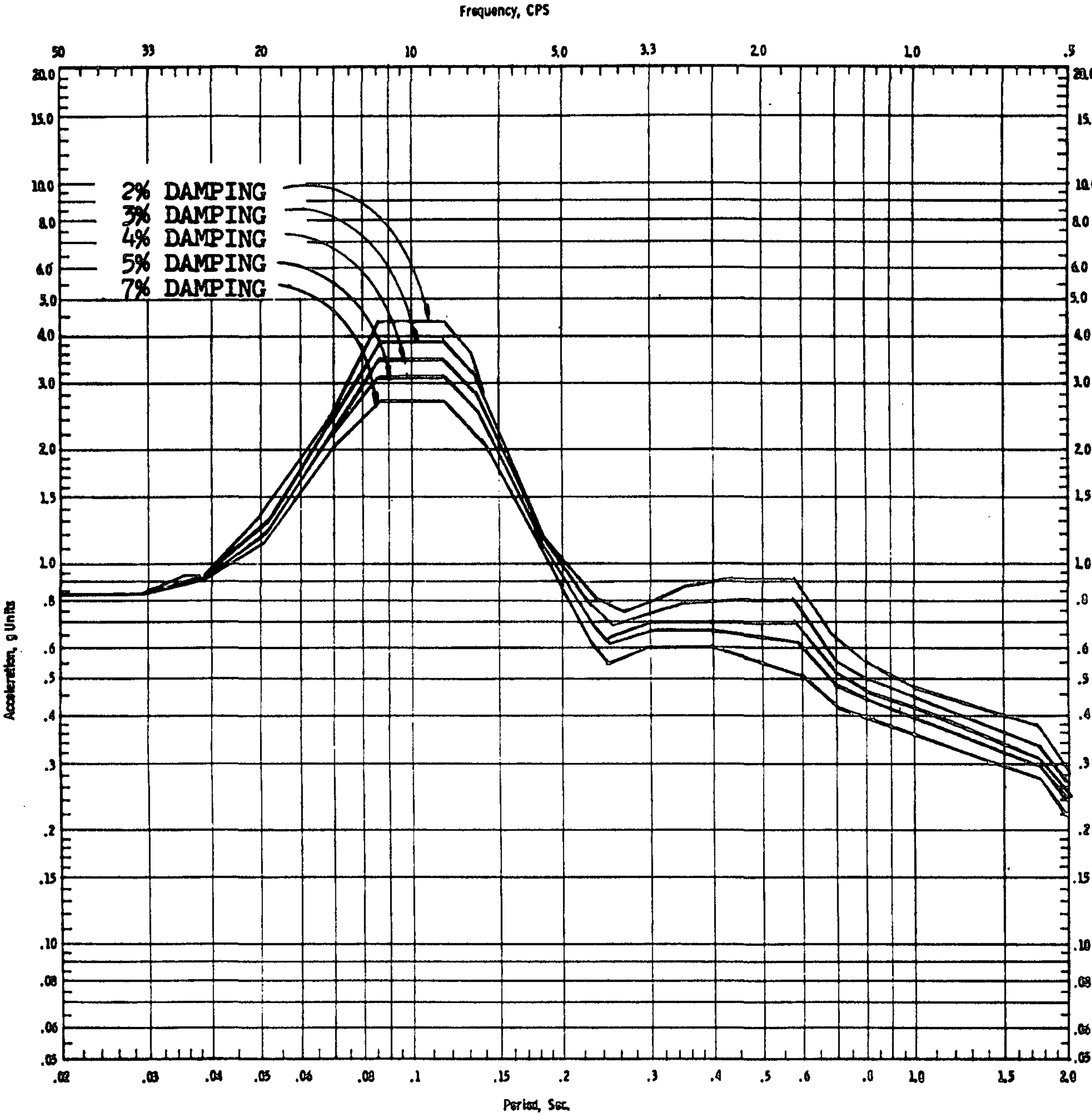

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FIGURE 3.7-68

HORIZONTAL FLOOR RESPONSE SPECTRA
N-S COMPONENT (SSE/BLAST) EL. 426'-0"
CONTAINMENT BUILDING (INNER STRUCTURE)

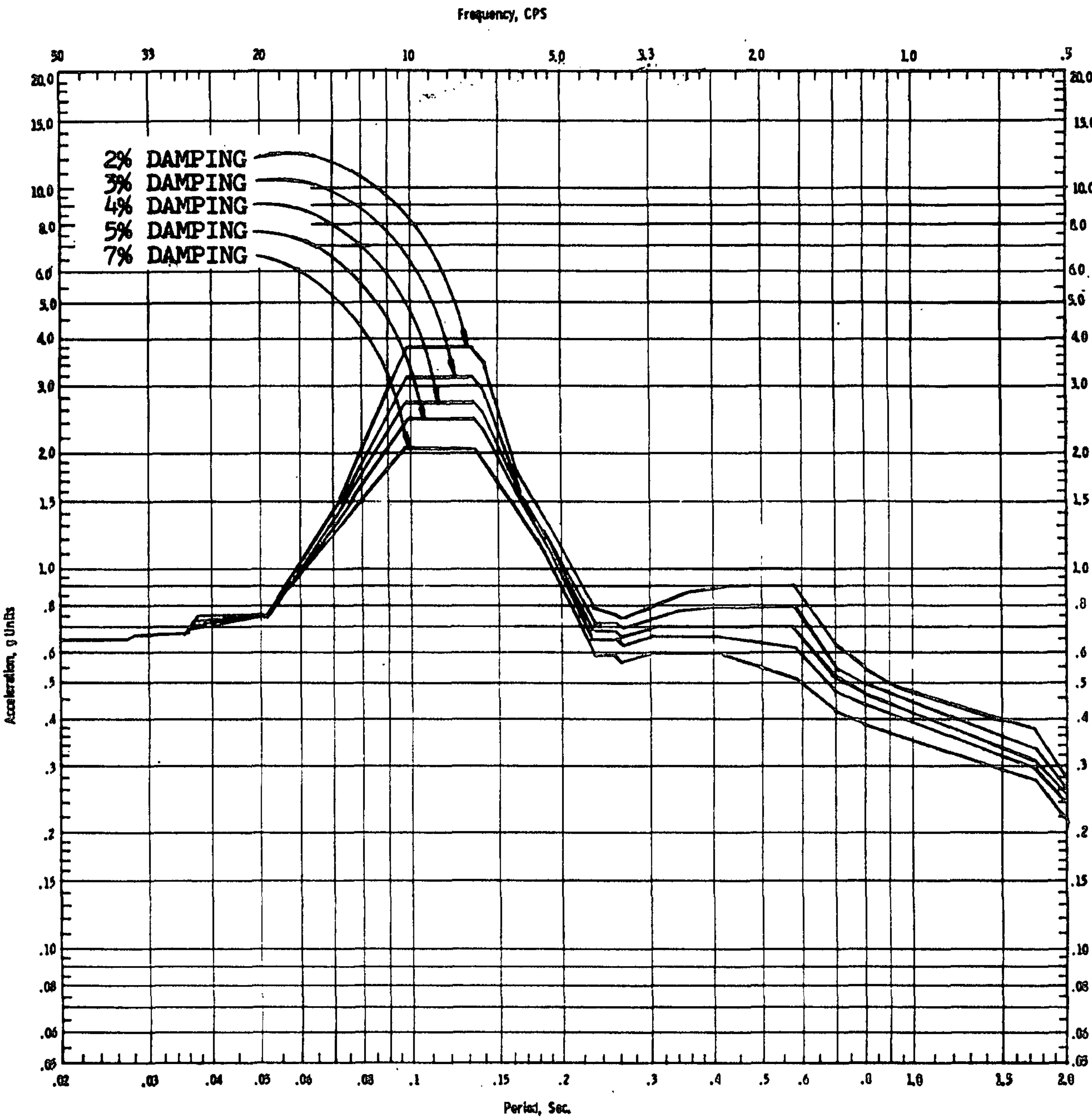

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FIGURE 3.7-69

HORIZONTAL FLOOR RESPONSE SPECTRA
E-W COMPONENT (SSE/BLAST) EL. 426'-0"
CONTAINMENT BUILDING (INNER STRUCTURE)

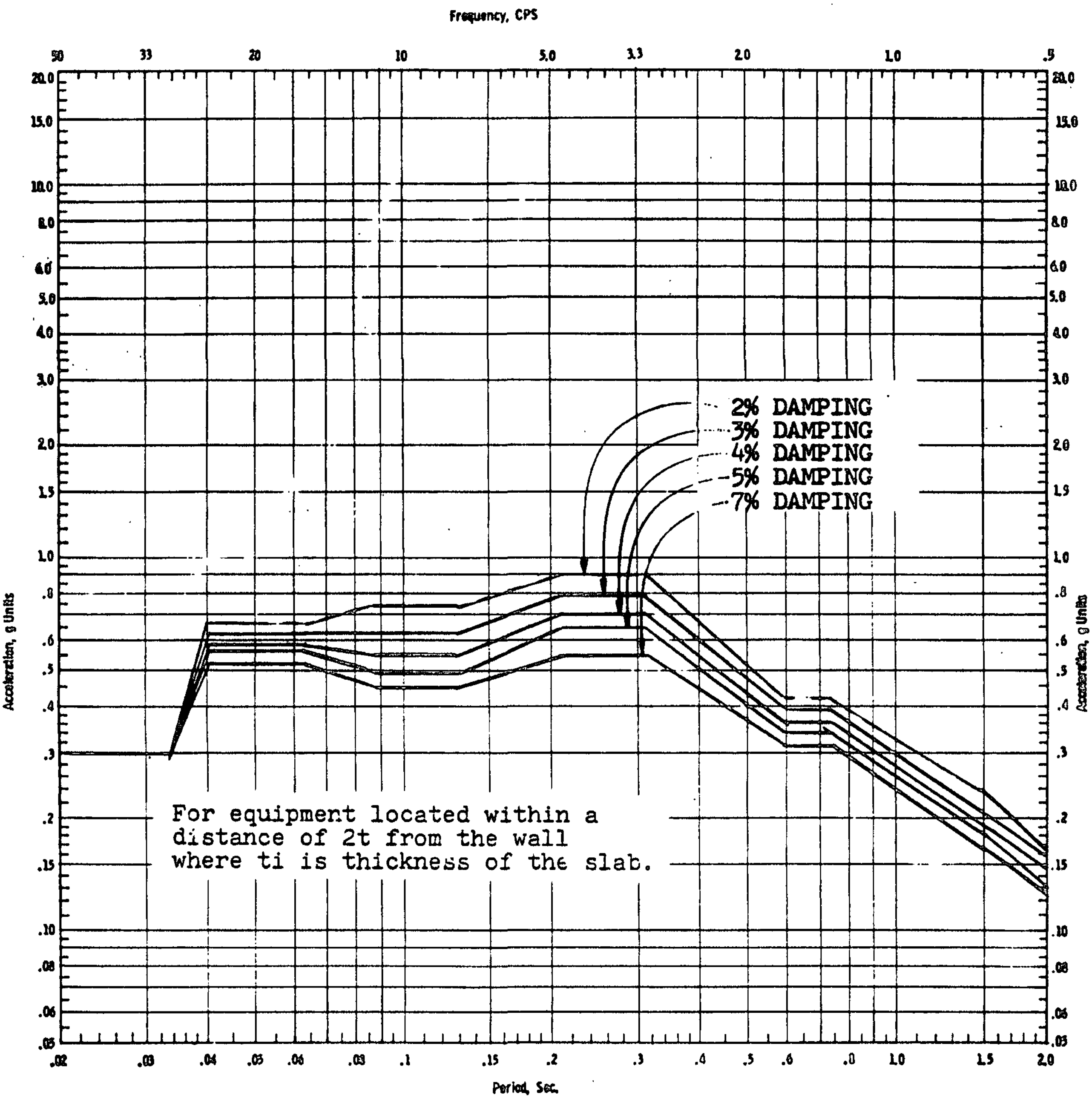

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FIGURE 3.7-70

VERTICAL RESPONSE SPECTRA (SSE/BLAST)
EL. 426'-0" & 412'-0"
CONTAINMENT BUILDING (INNER STRUCTURE) SLAB

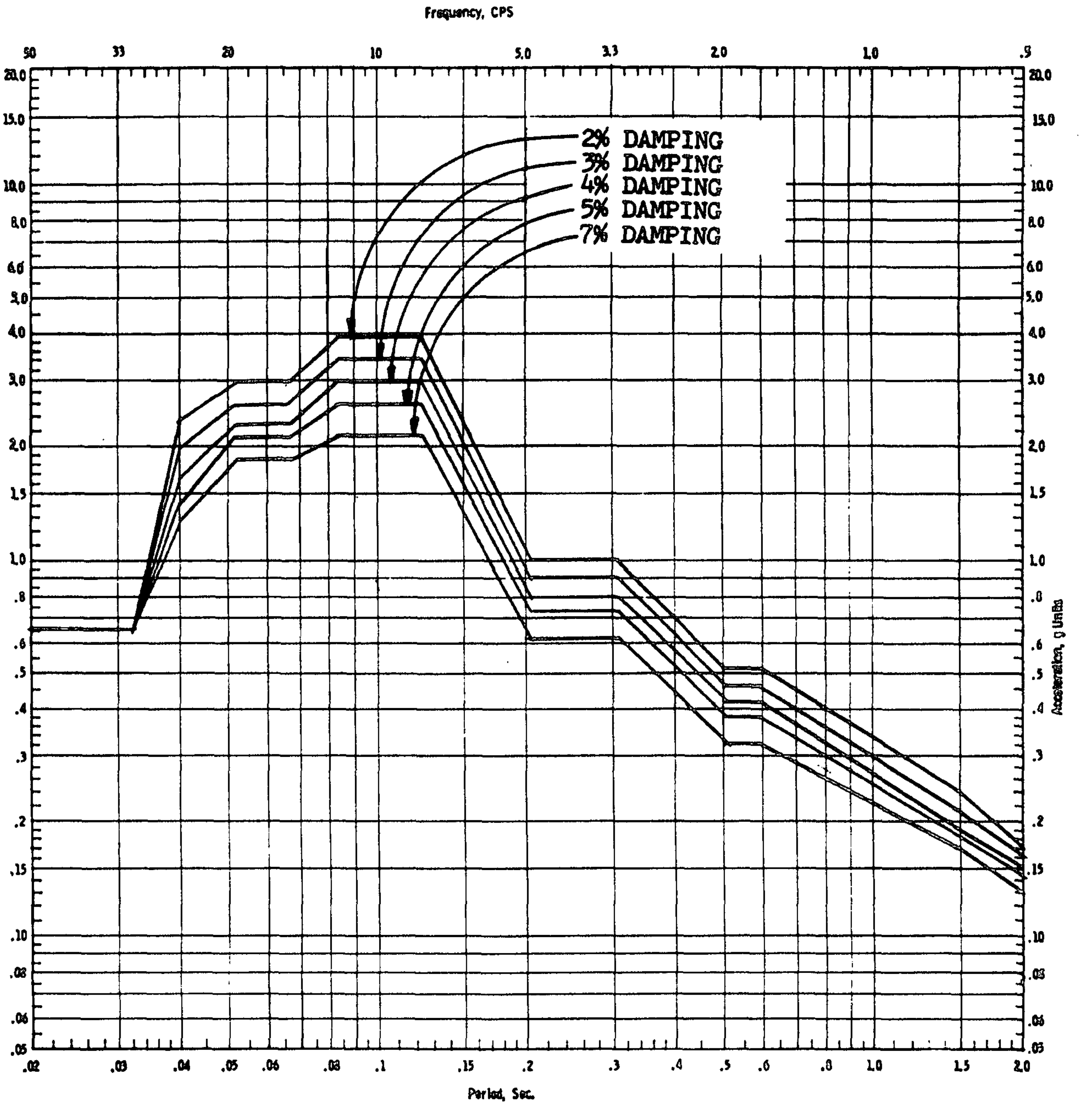

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FIGURE 3.7-71

VERTICAL RESPONSE SPECTRA (SSE/BLAST)
EL. 426'-0", 412'-0", 401'-0", & 390'-0"
CONTAINMENT BUILDING (INNER STRUCTURE) SLAB

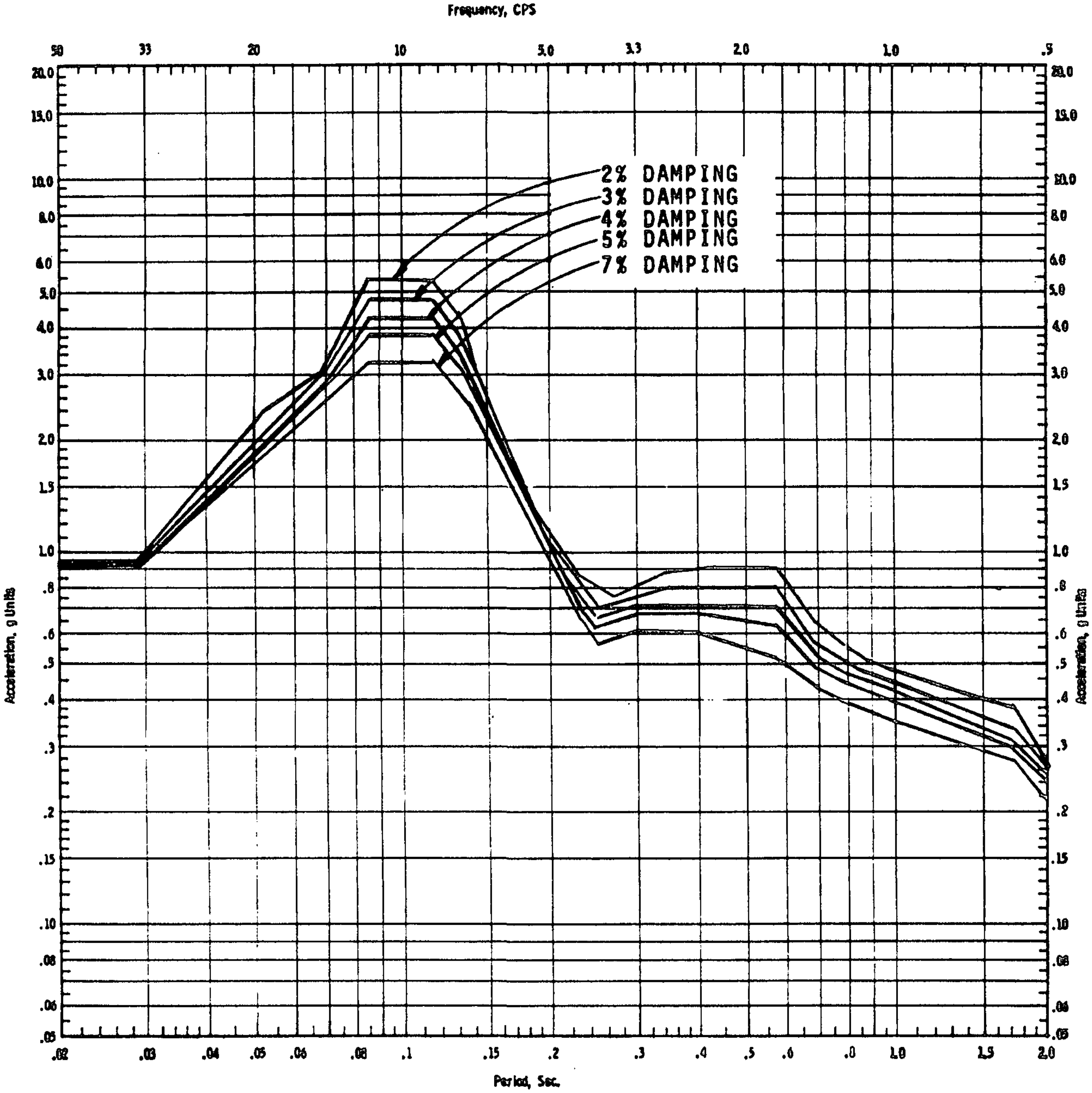

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FIGURE 3.7-72

HORIZONTAL FLOOR RESPONSE SPECTRA
N-S COMPONENT (SSE) EL. 448'-0"
CONTAINMENT BUILDING (INNER STRUCTURE)

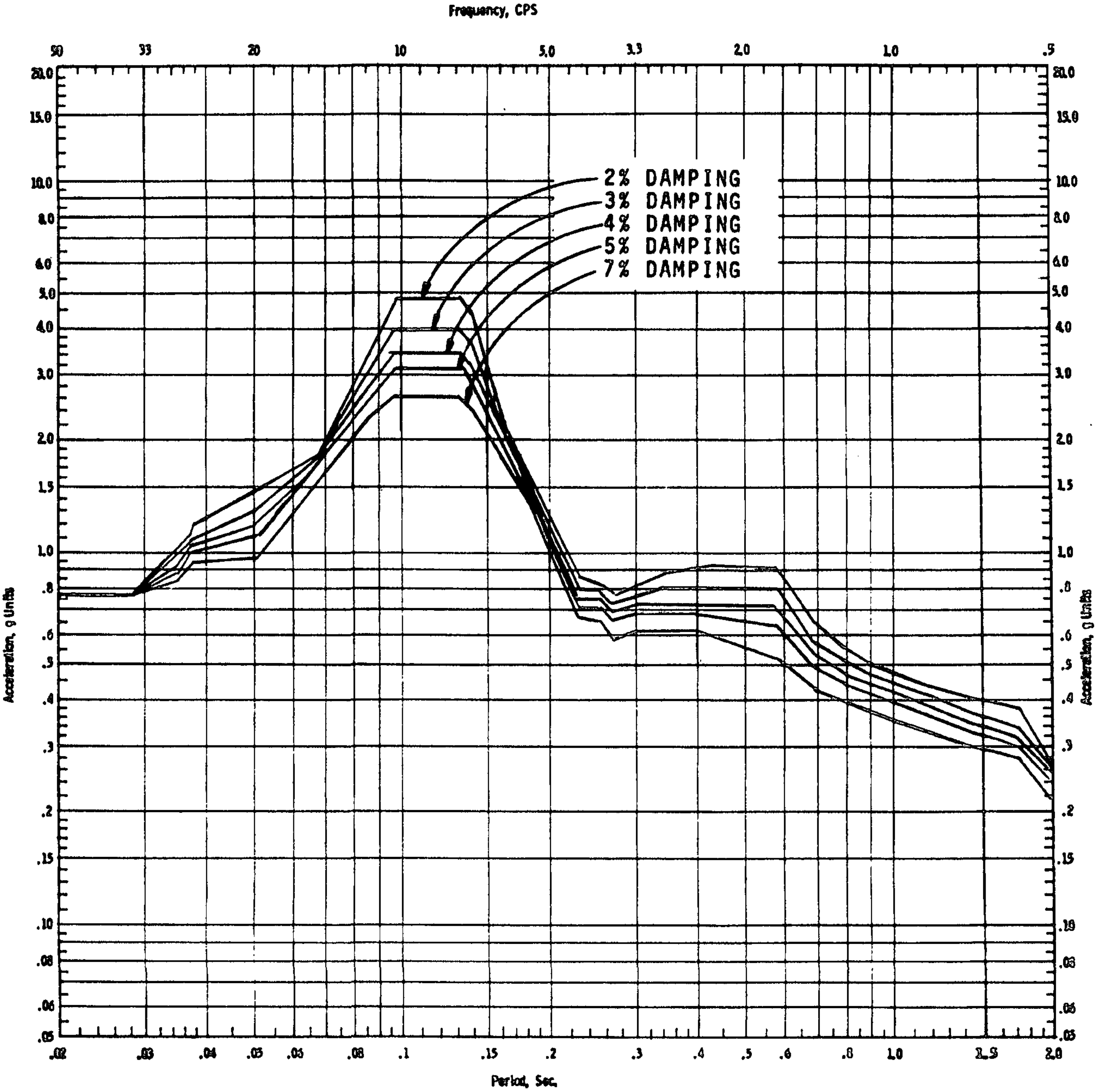

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FIGURE 3.7-73

HORIZONTAL FLOOR RESPONSE SPECTRA
E-W COMPONENT (SSE) EL. 448'-0"
CONTAINMENT BUILDING (INNER STRUCTURE)

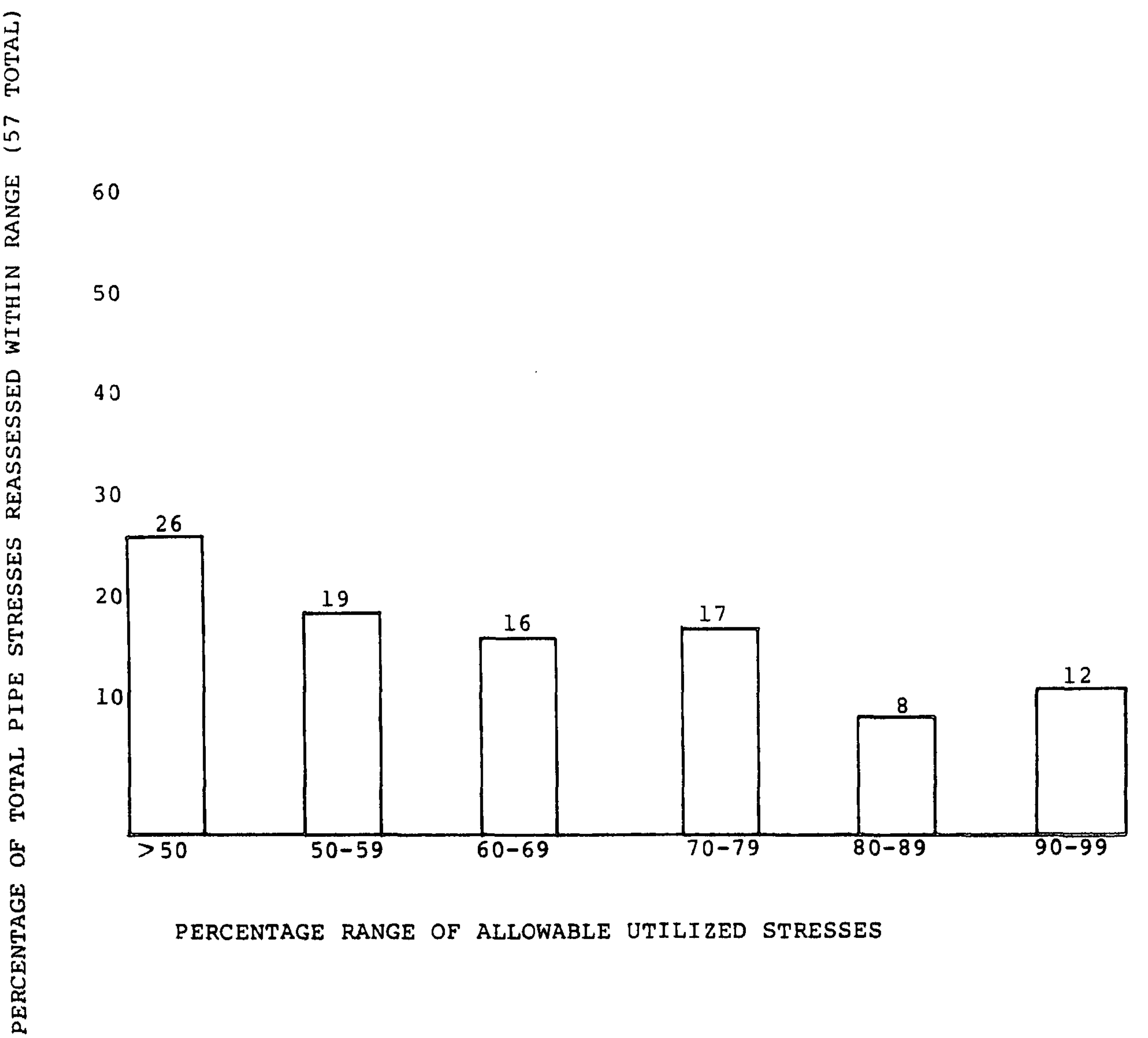

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FIGURE Q110.66-1

PIPING PROBLEMS

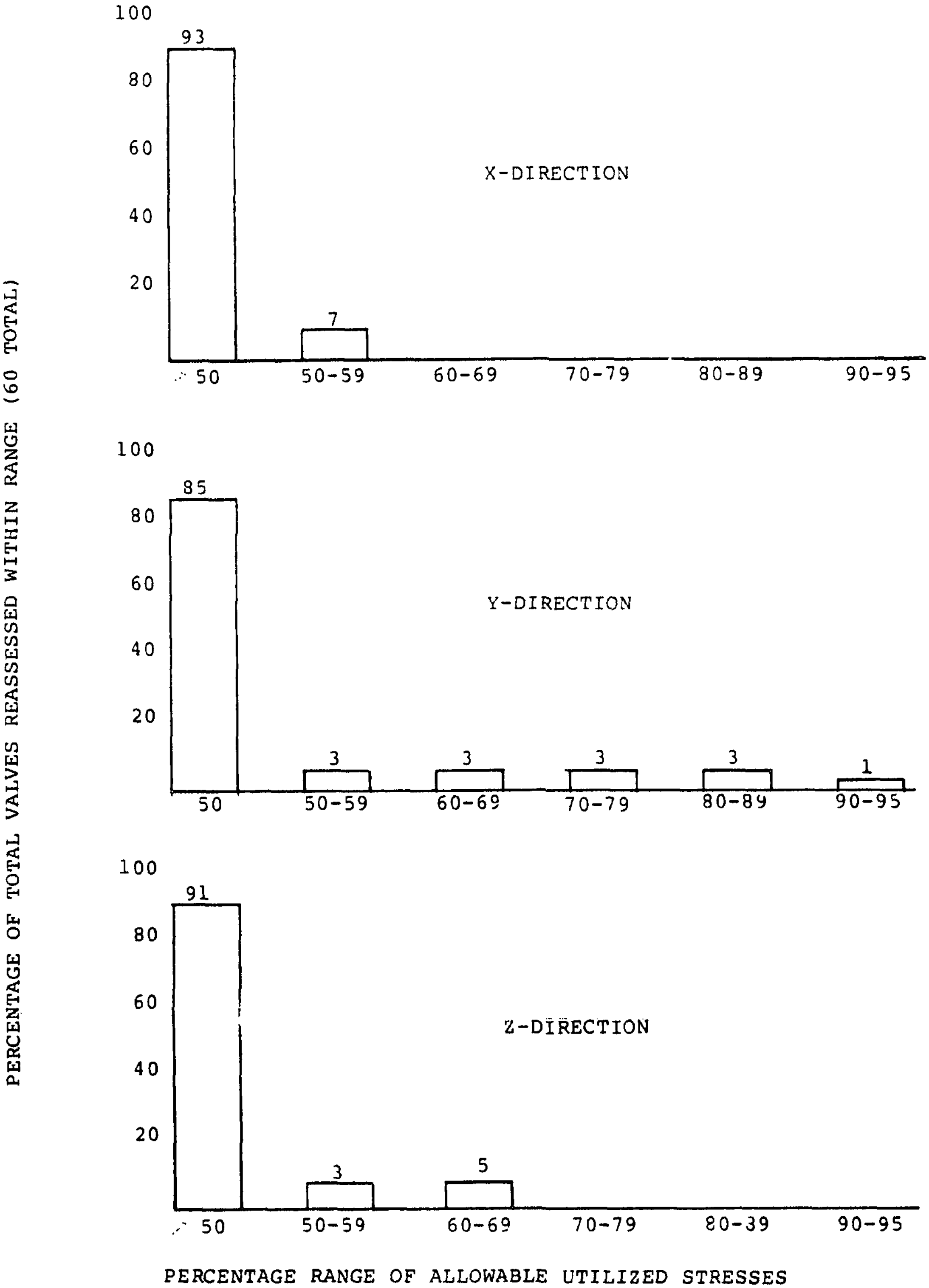

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FIGURE Q110.66-2

VALVE ACCELERATIONS GROUP I

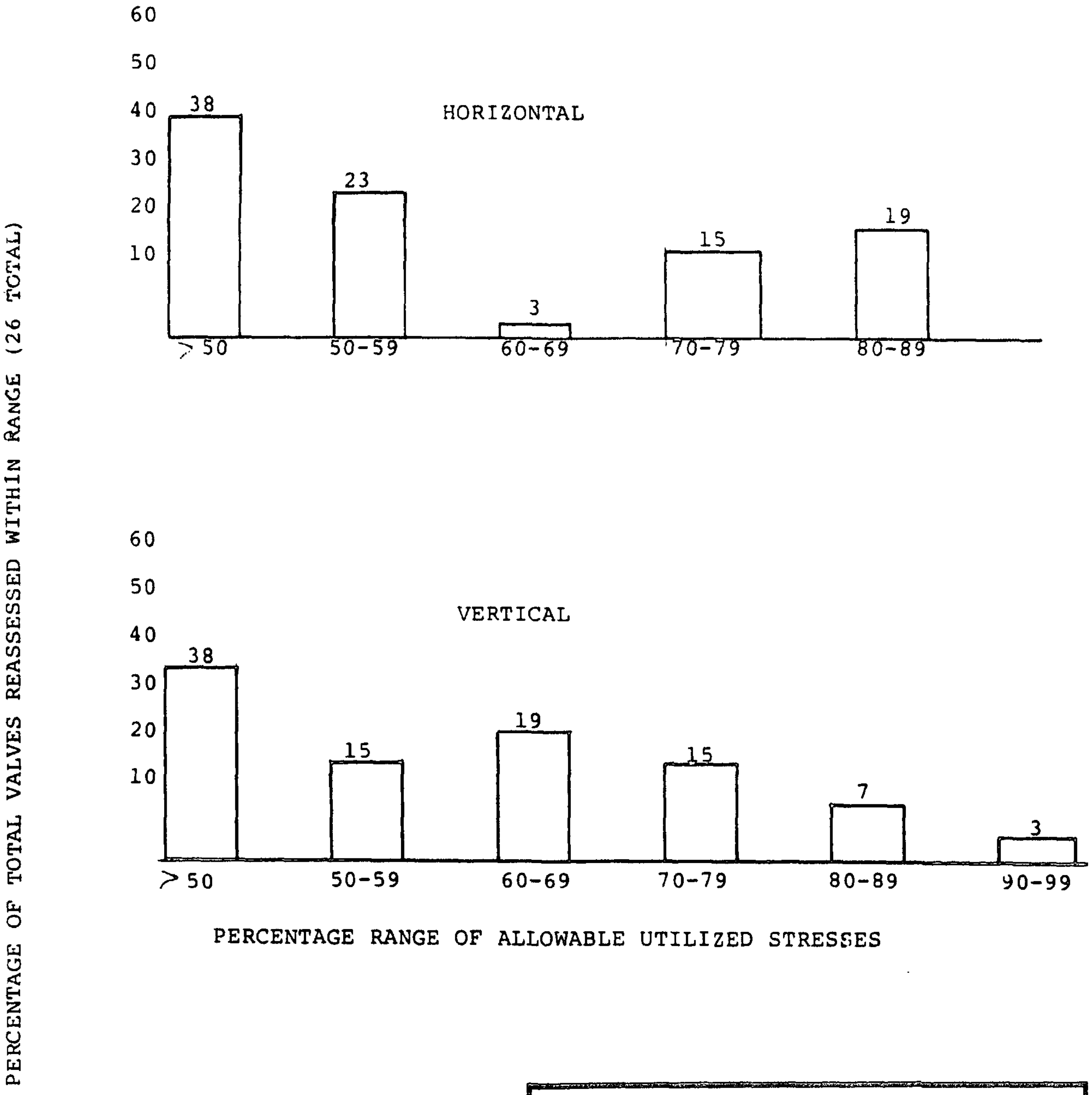

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FIGURE Q110.66-3

VALVE ACCELERATIONS GROUP II

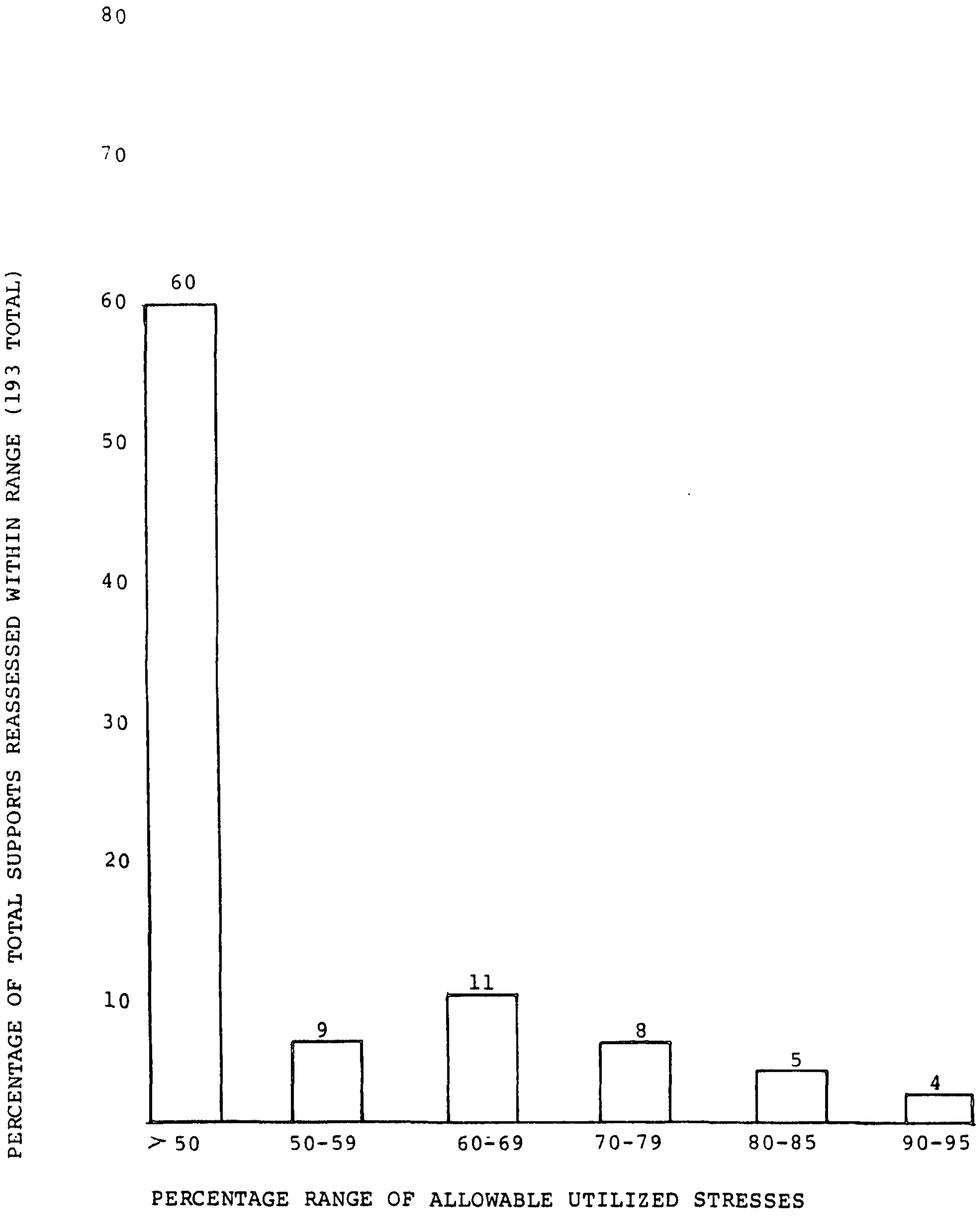

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FIGURE Q110.66-4

SUPPORT LOADS

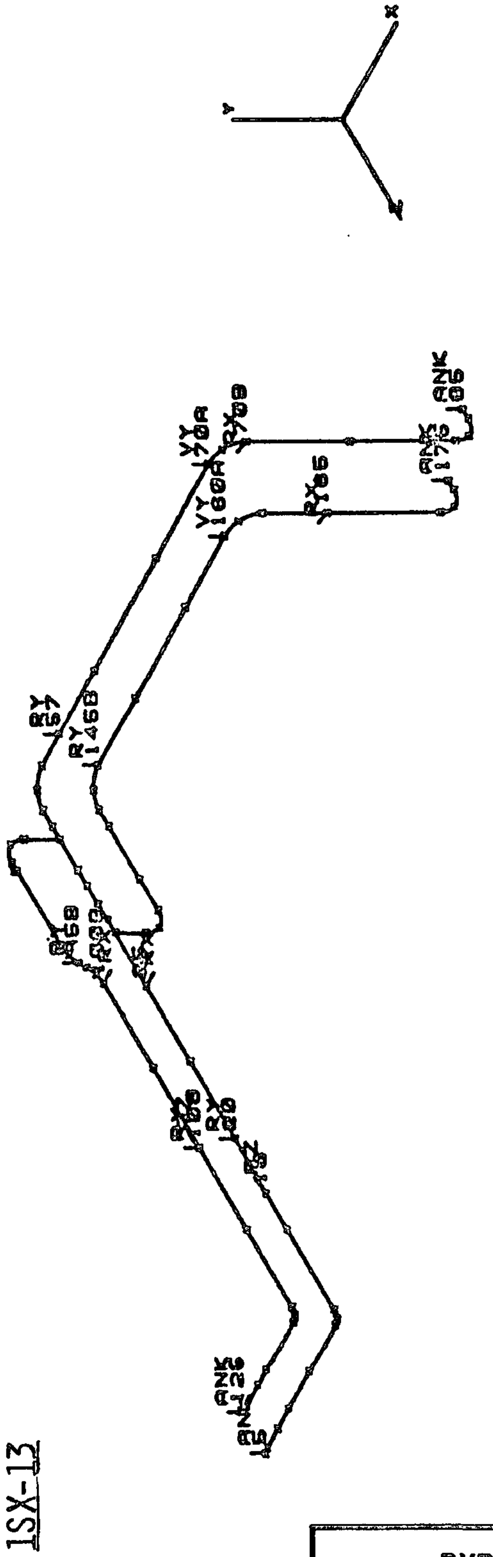

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FIGURE Q110.67-1

PIPING PROBLEM 1SX-13

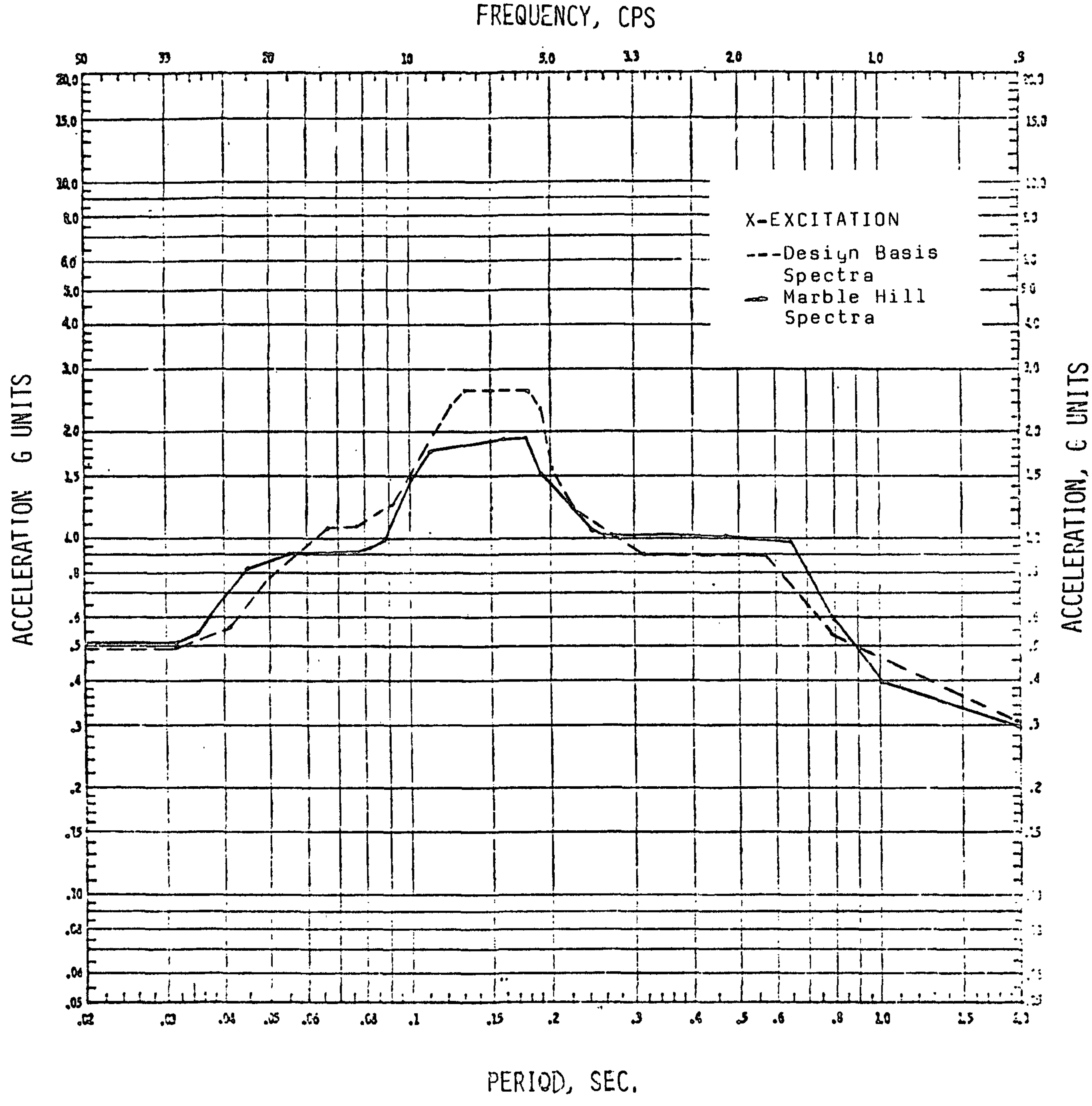

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FIGURE Q110.67-2

COMPARISON SPECTRA FOR
PIPING PROBLEM 1SX-13
(SHEET 1 OF 3)

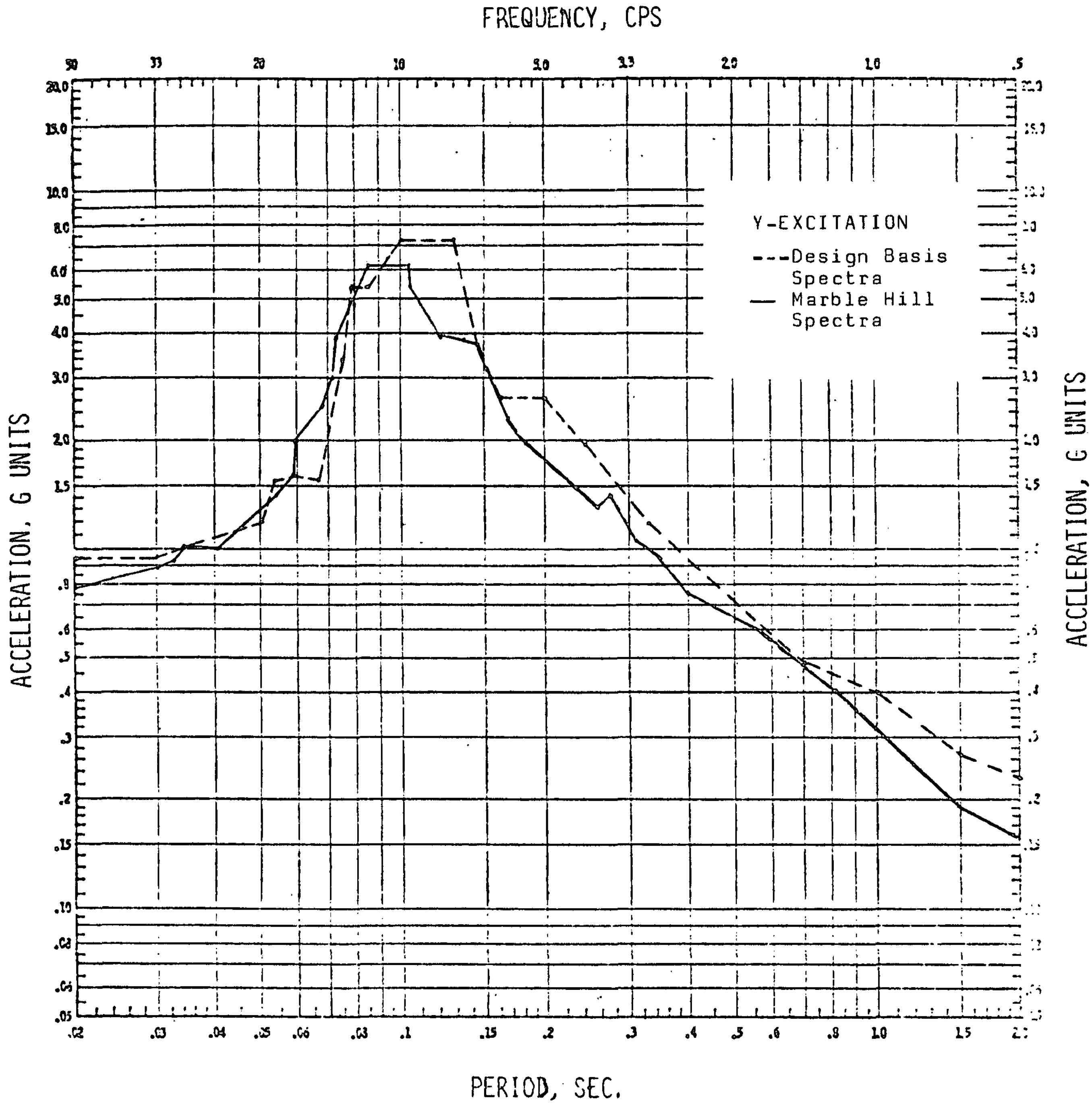

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FIGURE Q110.67-2

COMPARISON SPECTRA FOR
PIPING PROBLEM 1SX-13
(SHEET 2 OF 3)

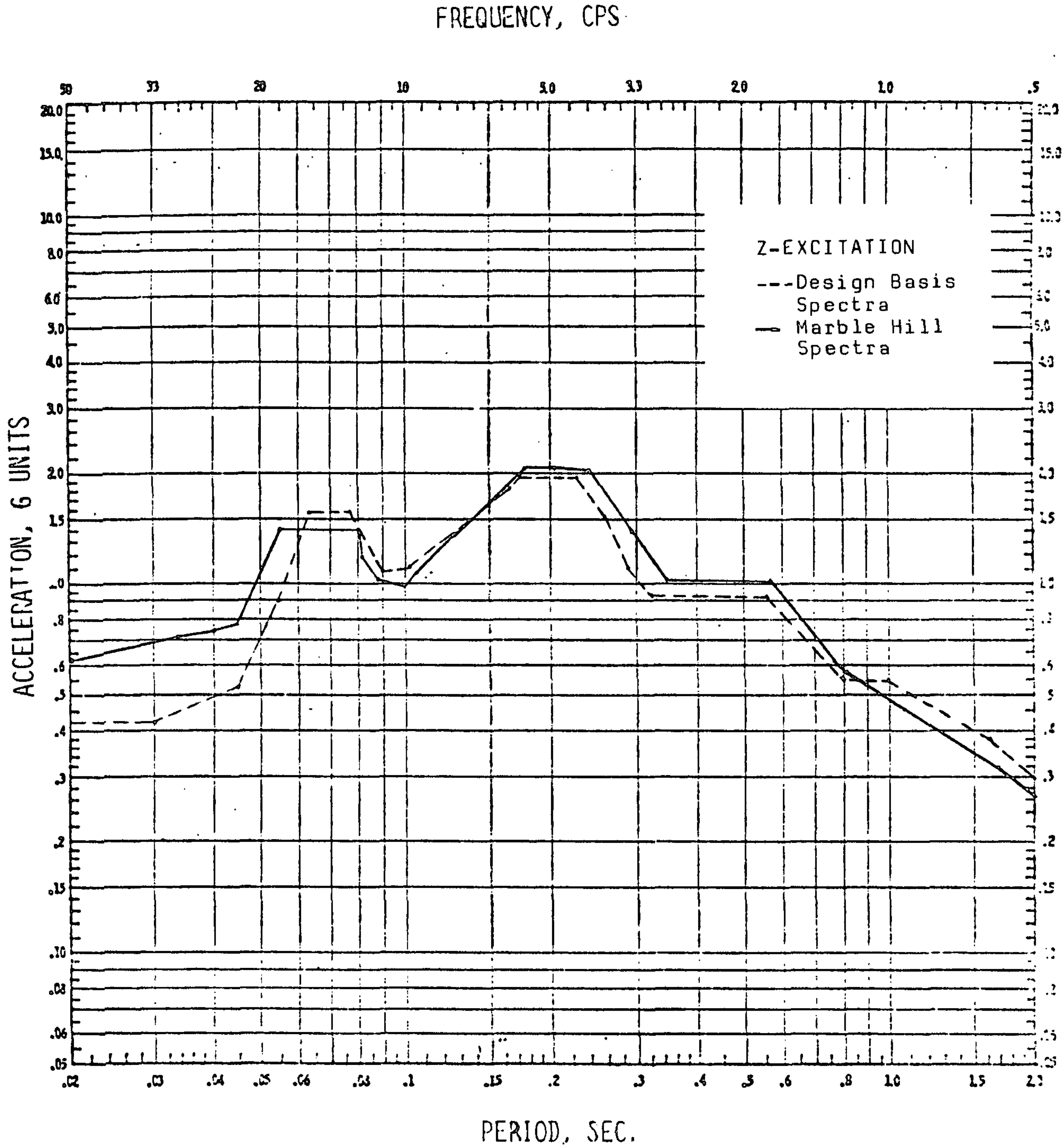

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FIGURE Q110.67-2

COMPARISON SPECTRA FOR
PIPING PROBLEM 1SX-13
(SHEET 3 OF 3)

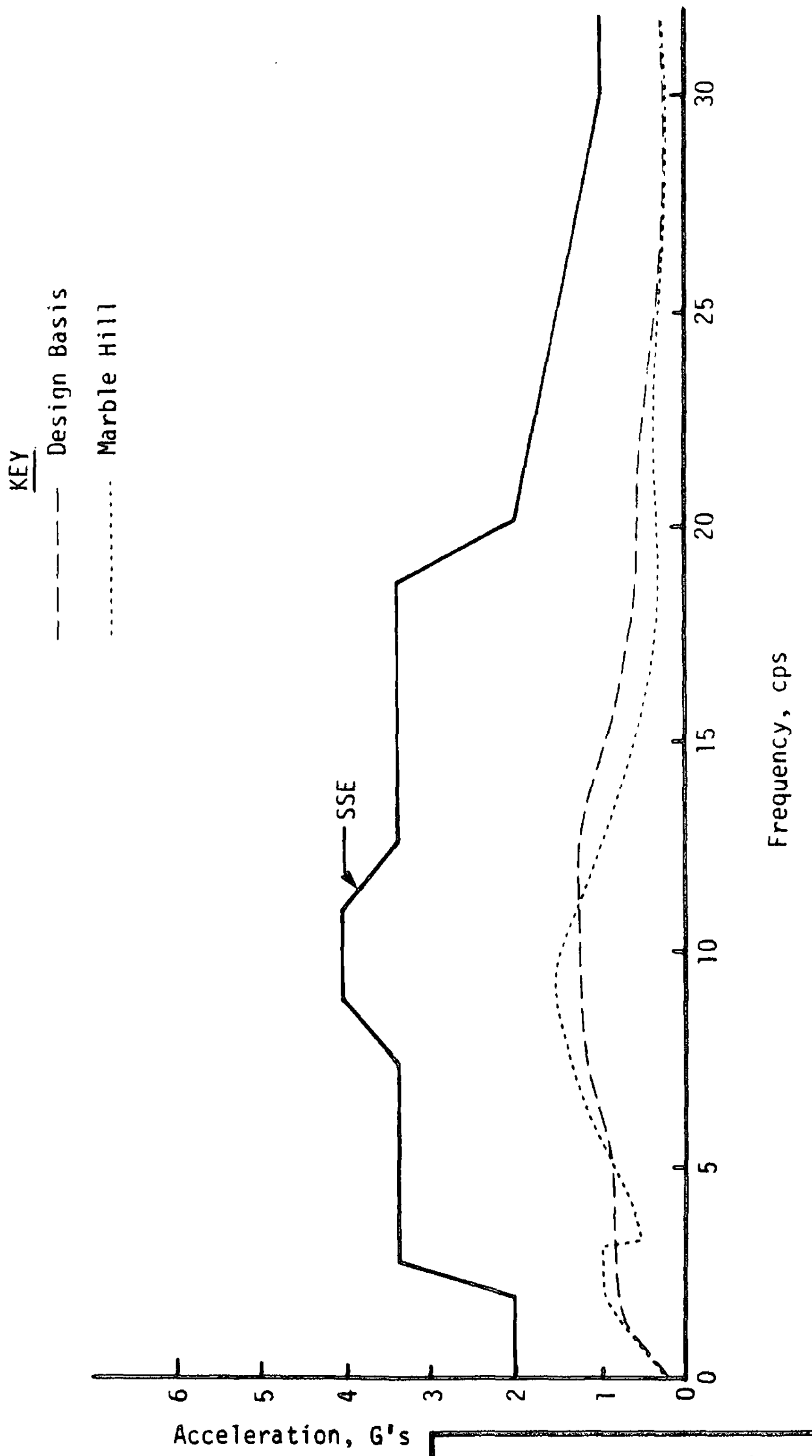

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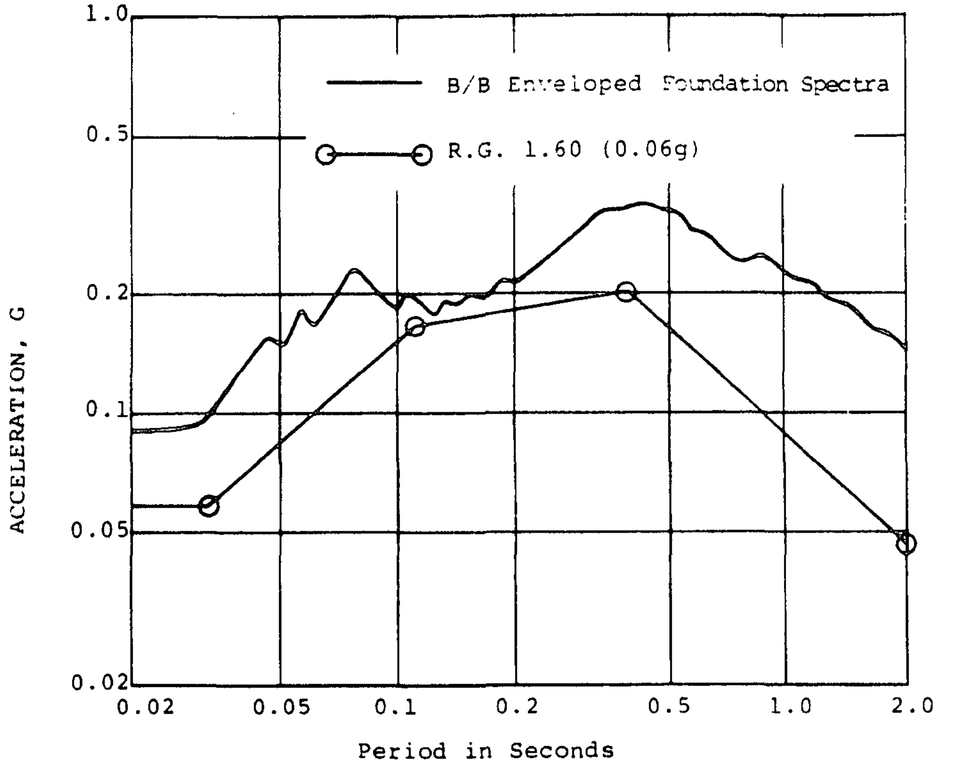
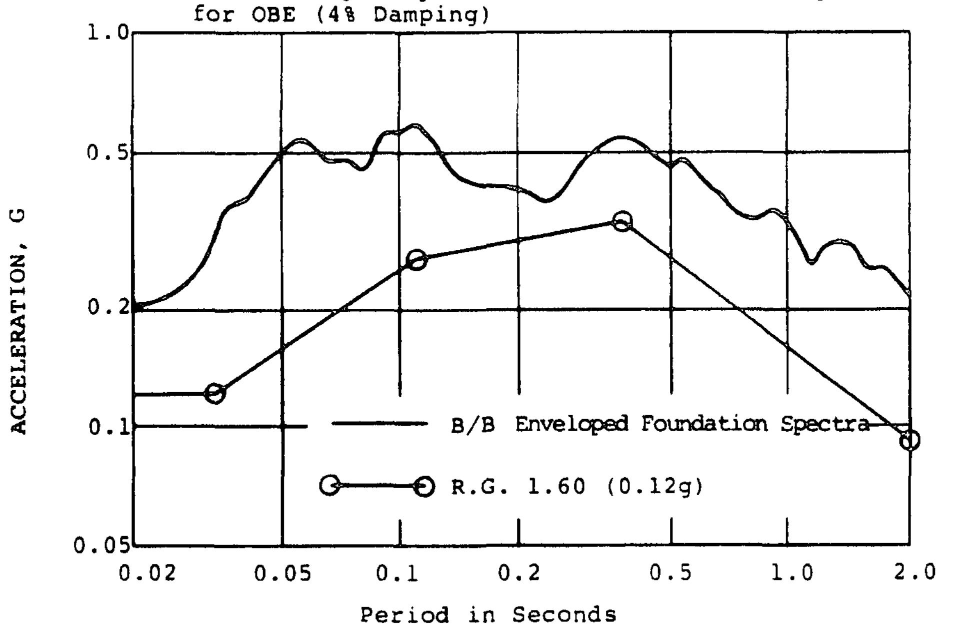
FIGURE Q110.70-1

COMPARISON SPECTRA FOR
RPV INTERNALS



(a) B/B Enveloped Spectra and R.G. 1.60 (0.06g) Spectra for OBE (4% Damping)



(b) B/B Enveloped Spectra and R.G. 1.60 (0.12g) Spectra for SSE (7% Damping)

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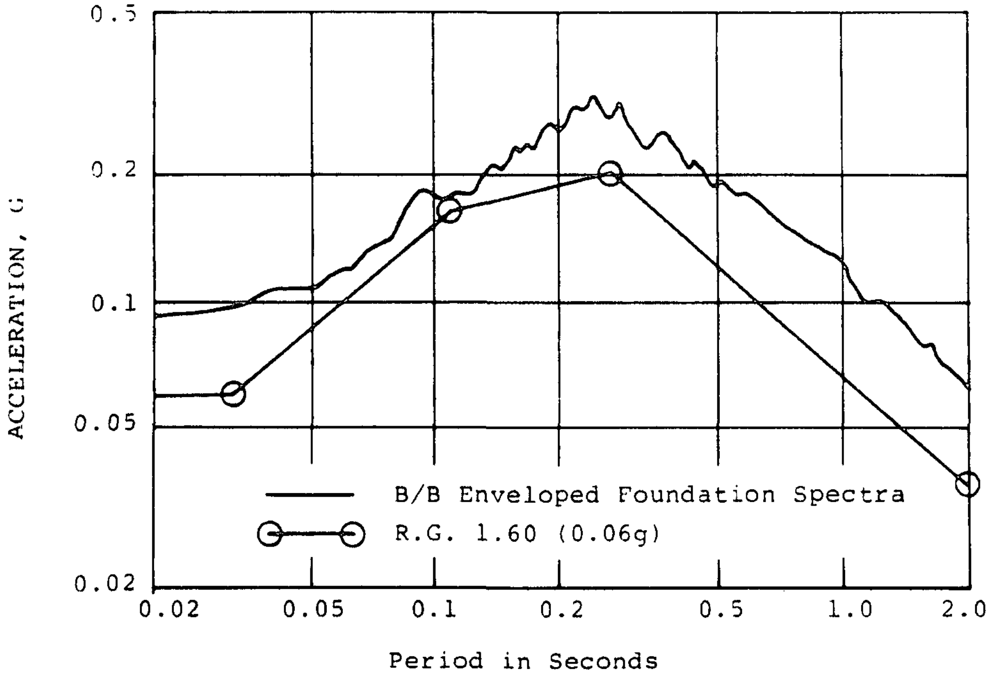
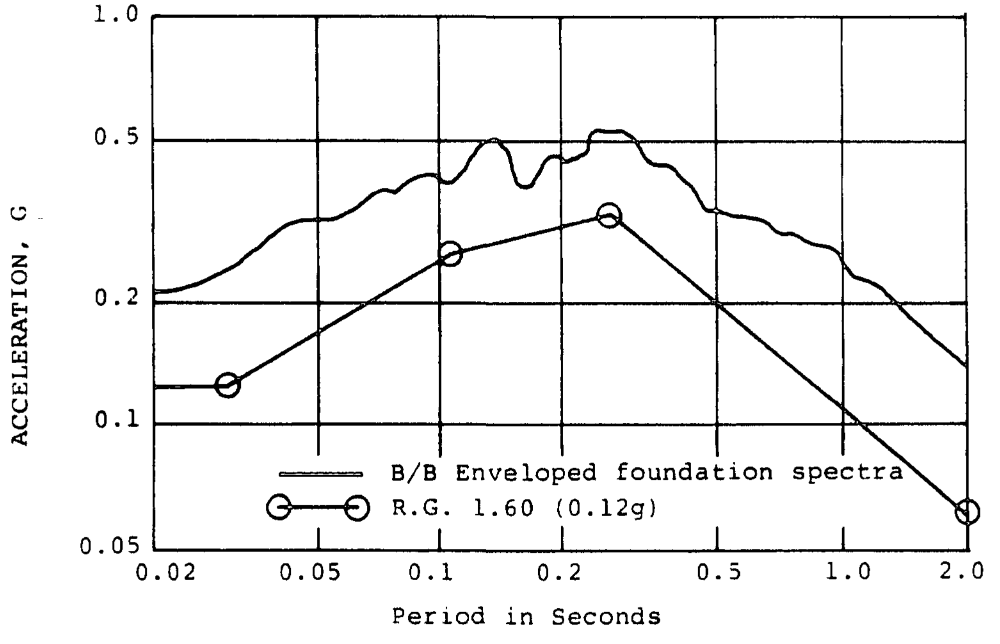
FIGURE Q130.6-1

COMPARISON OF HORIZONTAL SPECTRA



(a) B/B Enveloped spectra and R.G. 1.60 (0.06g) spectra
for OBE (4% Damping)



(b) B/B Enveloped spectra and R.G. 1.60 (0.12g) spectra
for SSE (7% Damping)

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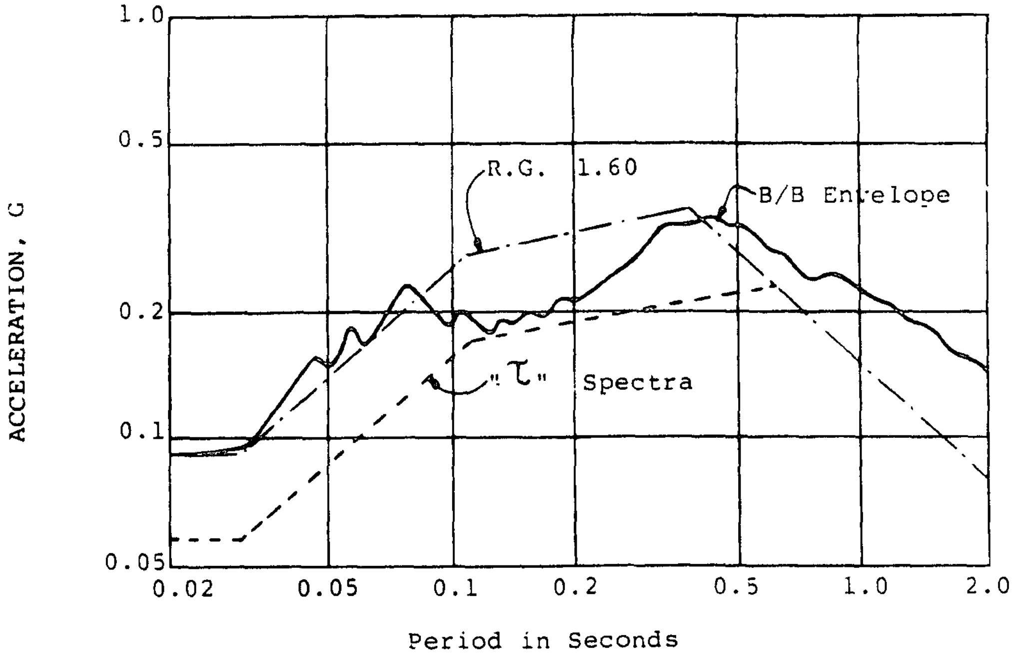
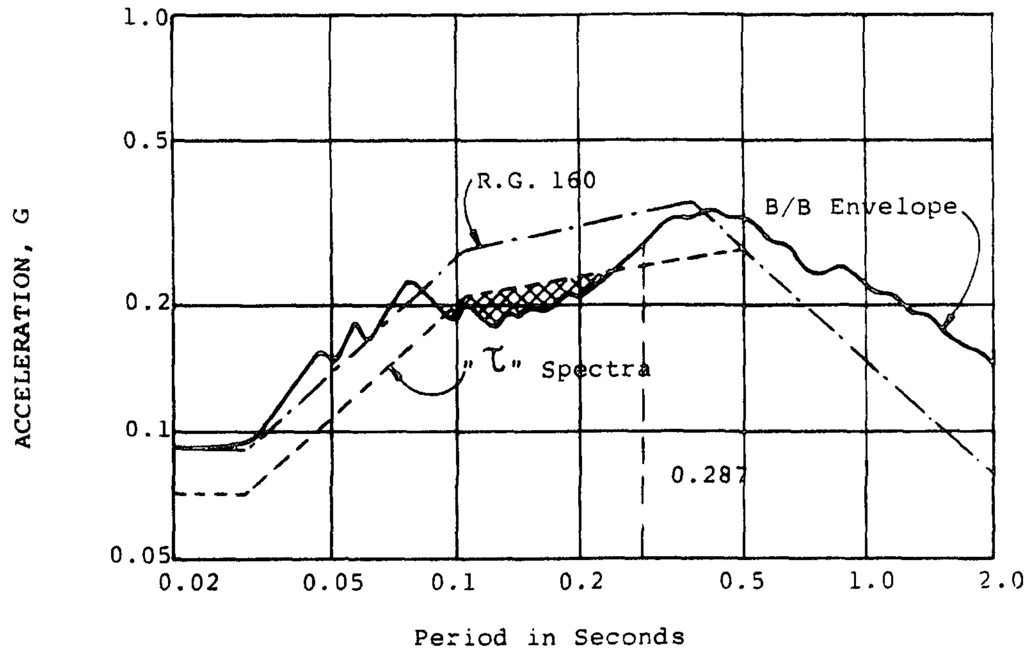
FIGURE Q130.6-2

COMPARISON OF VERTICAL SPECTRA



(a) OBE (4%) Spectra for AUX-TUR complex (τ = 0.067 sec.)



(b) OBE (4%) Spectra for Containment (τ = 0.04 sec.)

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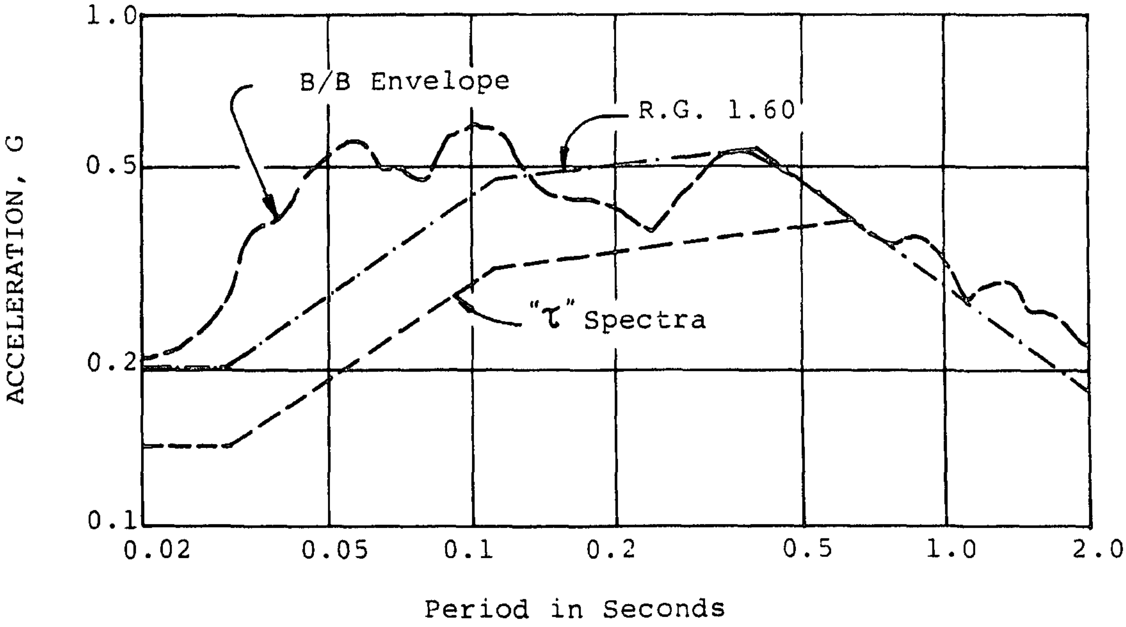
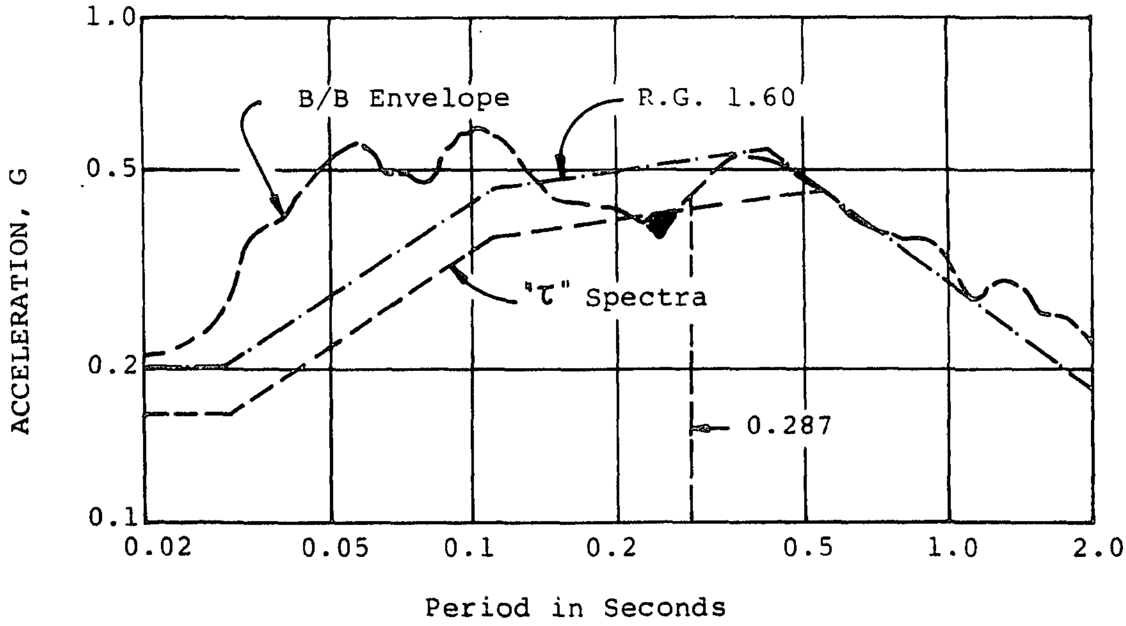
FIGURE Q130.6-3 HORIZONTAL OBE (4%) SPECTRA COMPARISON



(a) SSE (7%) spectra for AUX-TUR complex (τ = 0.067 sec.)



(b) SSE (7%) spectra for containment (τ = 0.04 sec.)

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FIGURE Q130.6-4

HORIZONTAL SSE (7%) SPECTRA COMPARISON

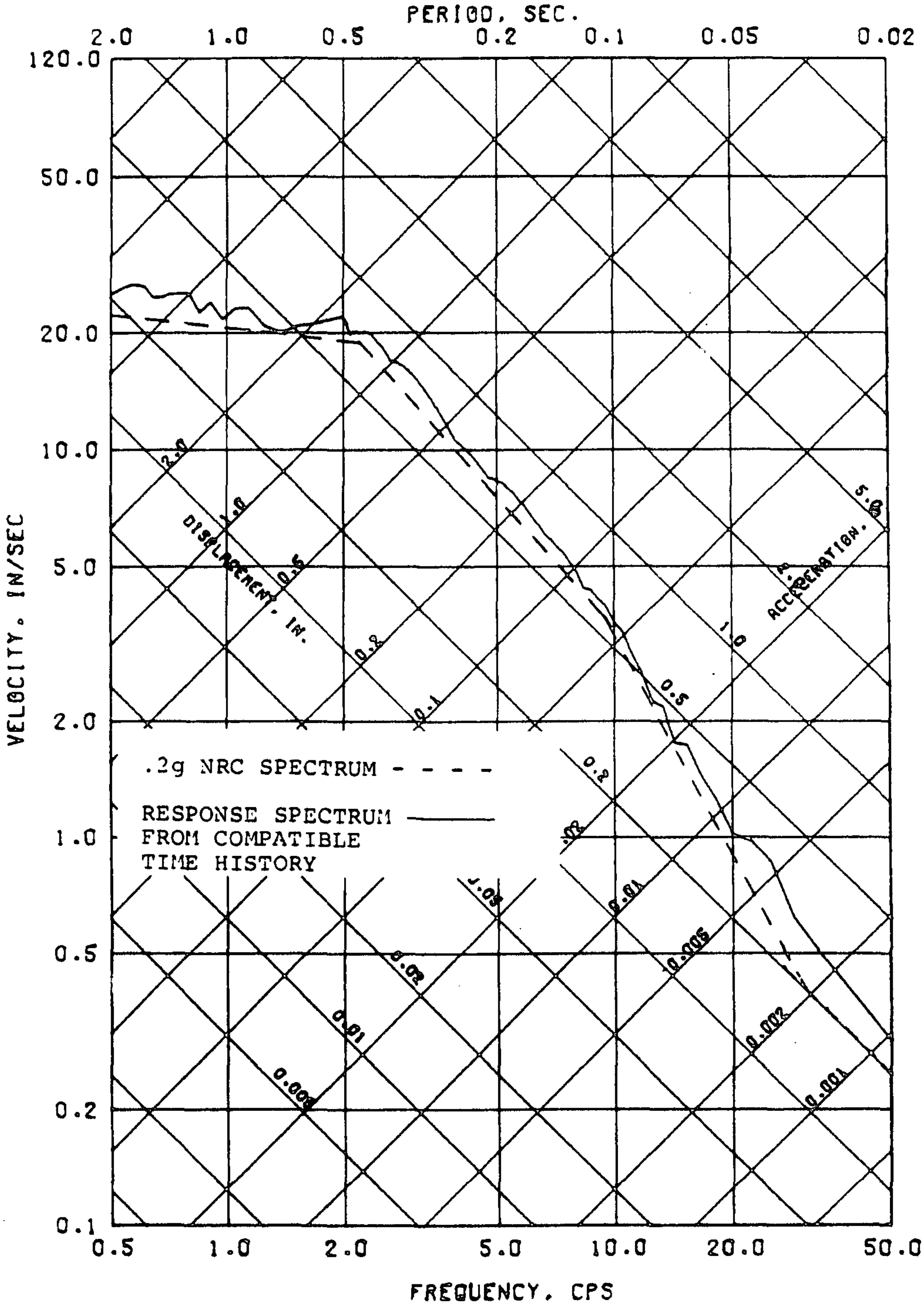

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FIGURE Q130.6-5

HORIZONTAL RESPONSE SPECTRA
(4% DAMPING)

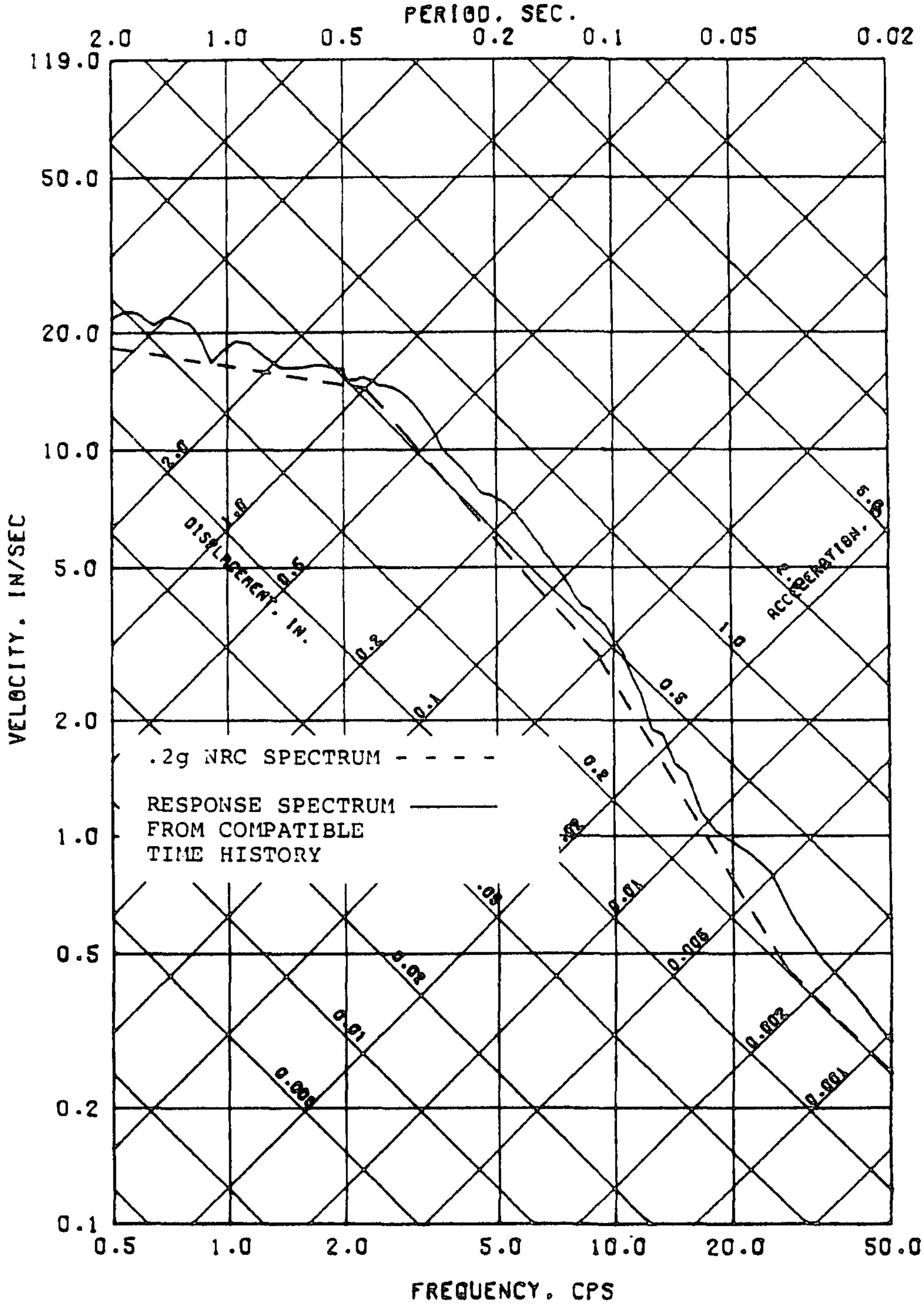

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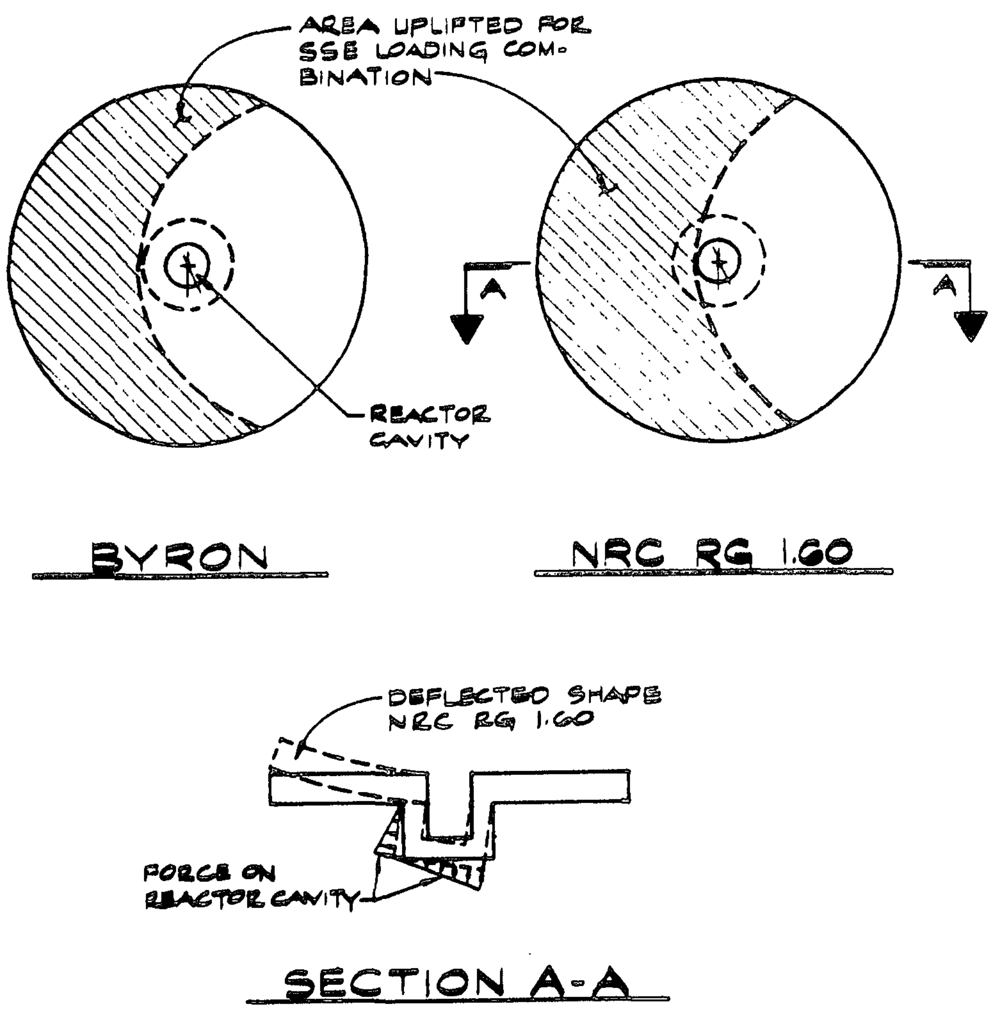
FIGURE Q130.6-6

HORIZONTAL RESPONSE SPECTRA
(7% DAMPING)



NOTE: THE UPLIFT CALCULATIONS DO NOT ACCOUNT FOR THE SIDE SOIL/ROCK RESISTANCE WHICH WOULD BE MOBILIZED

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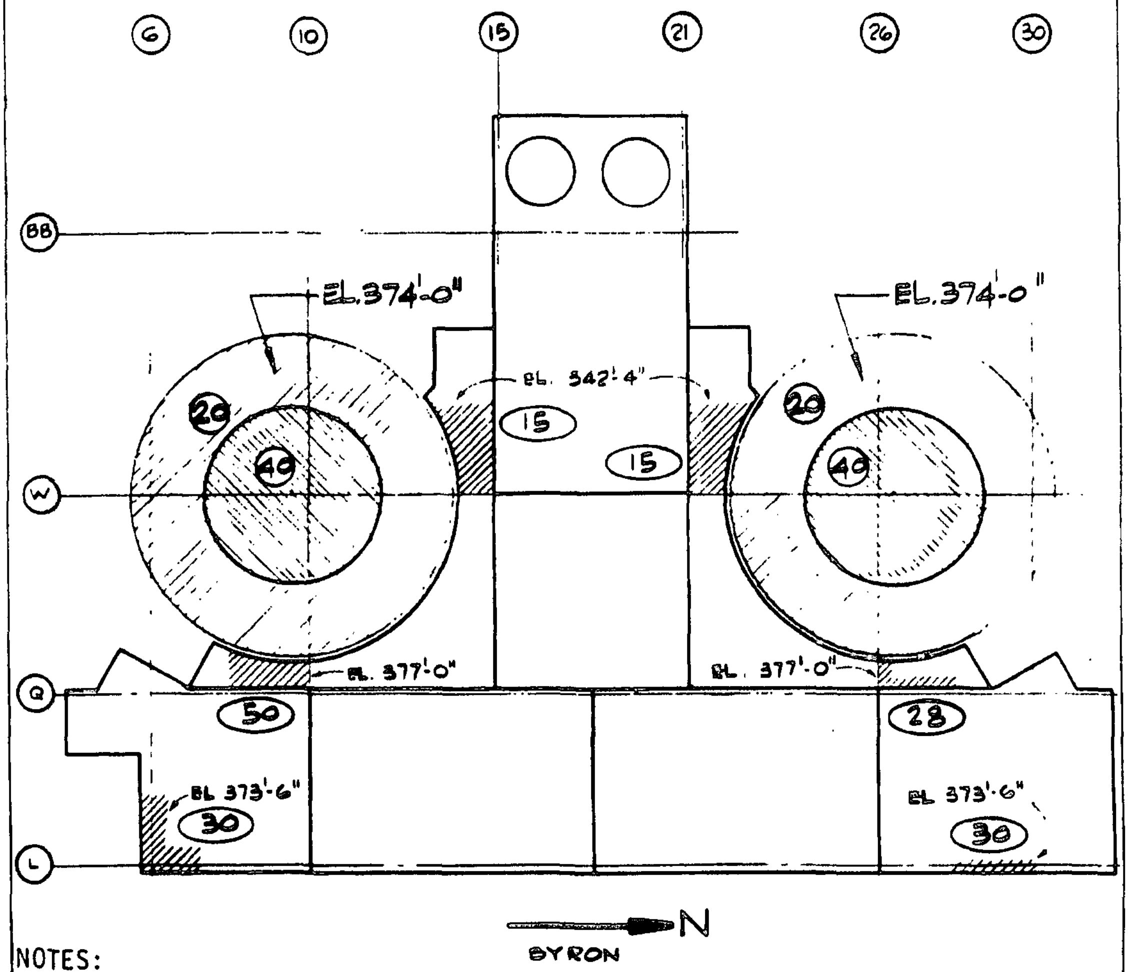
FIGURE Q130.6-7

CONTAINMENT BASE MAT UPLIFT COMPARISON



NOTES:

1. THE CHANGES IN THE FORCES ARE PRIMARILY DUE TO INCREASED UPLIFT OF THE MAT.
2. THE UPLIFT CALCULATIONS DO NOT ACCOUNT FOR THE SIDE SOIL/ROCK RESISTANCE WHICH WOULD BE MOBILIZED.

LEGEND:

INDICATES AVERAGE % INCREASE IN SEISMIC FORCES IN BASE MAT.

INDICATES AFFECTED AREAS

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FIGURE Q130.6-8

CONTAINMENT—AUXILIARY —FUEL HANDLING BUILDING COMPLEX AT MAT ELEVATION

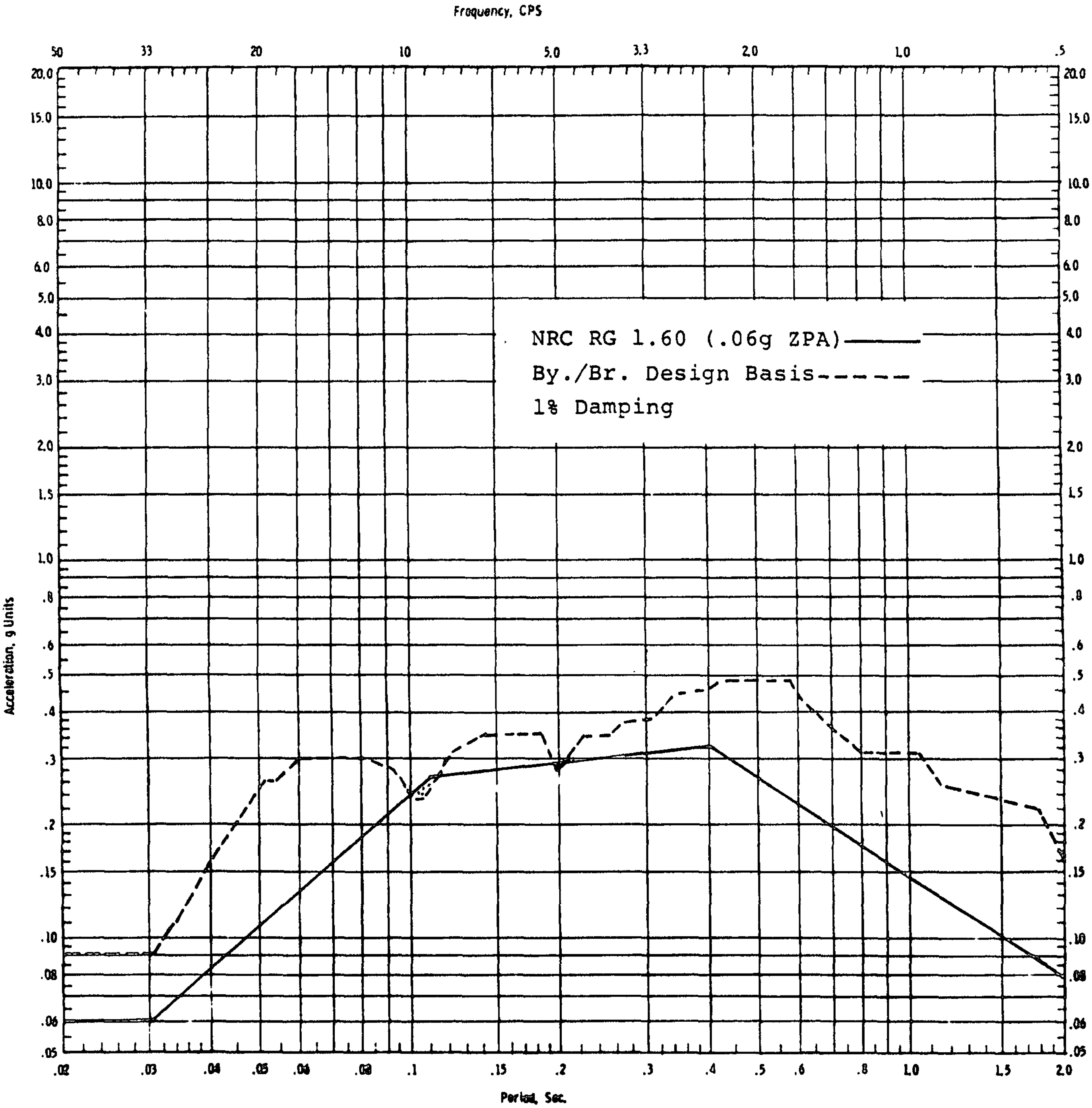

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FIGURE Q130.6-9
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS AND EW
LOCATION: AUXILIARY AND CONTAINMENT BLDG.
ELEVATION: 330'-0"; 374'-0"

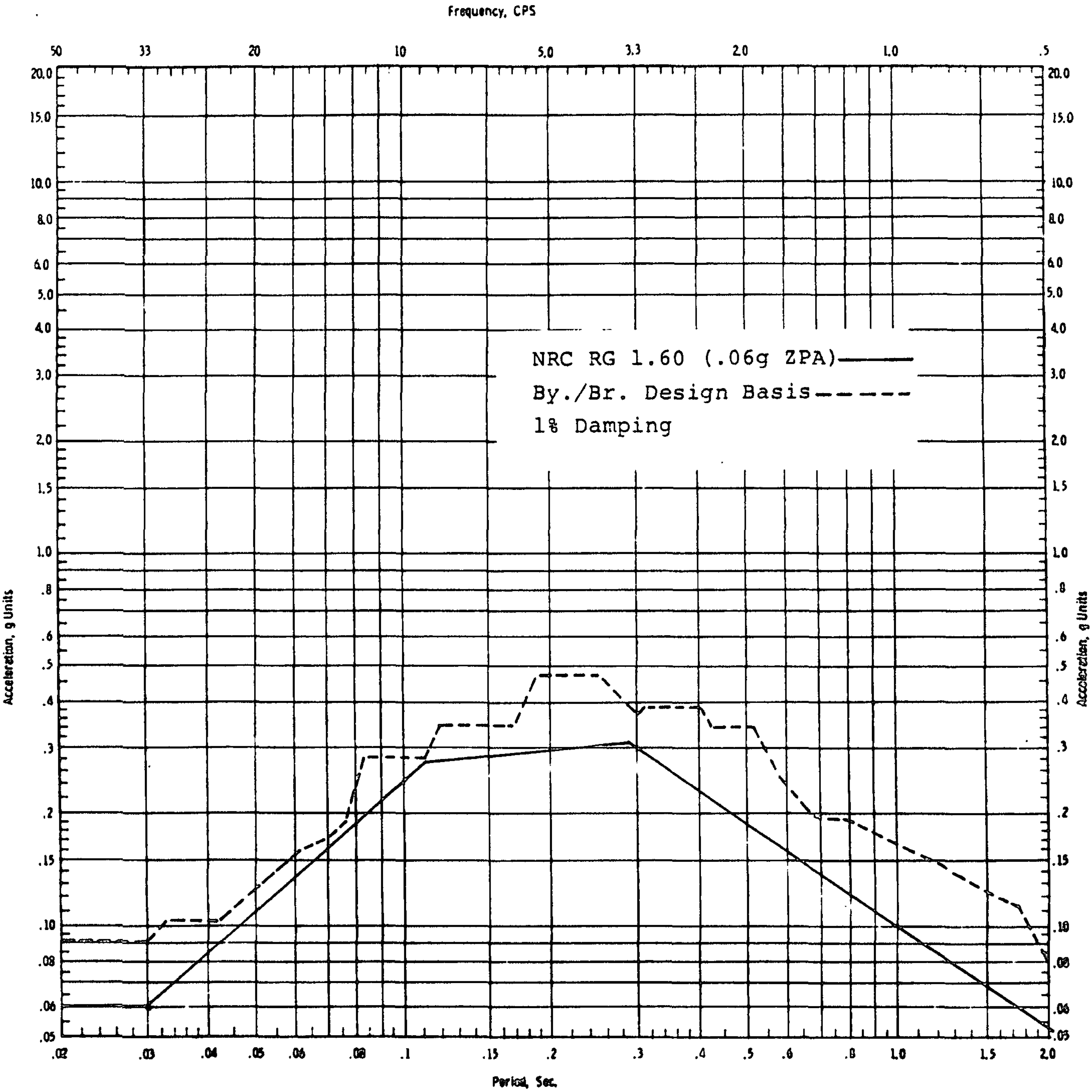

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FIGURE Q130.6-10
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL AND SLAB
LOCATION: AUXILIARY AND CONTAINMENT BLDG.
ELEVATION: 330'-0"; 374'-0"

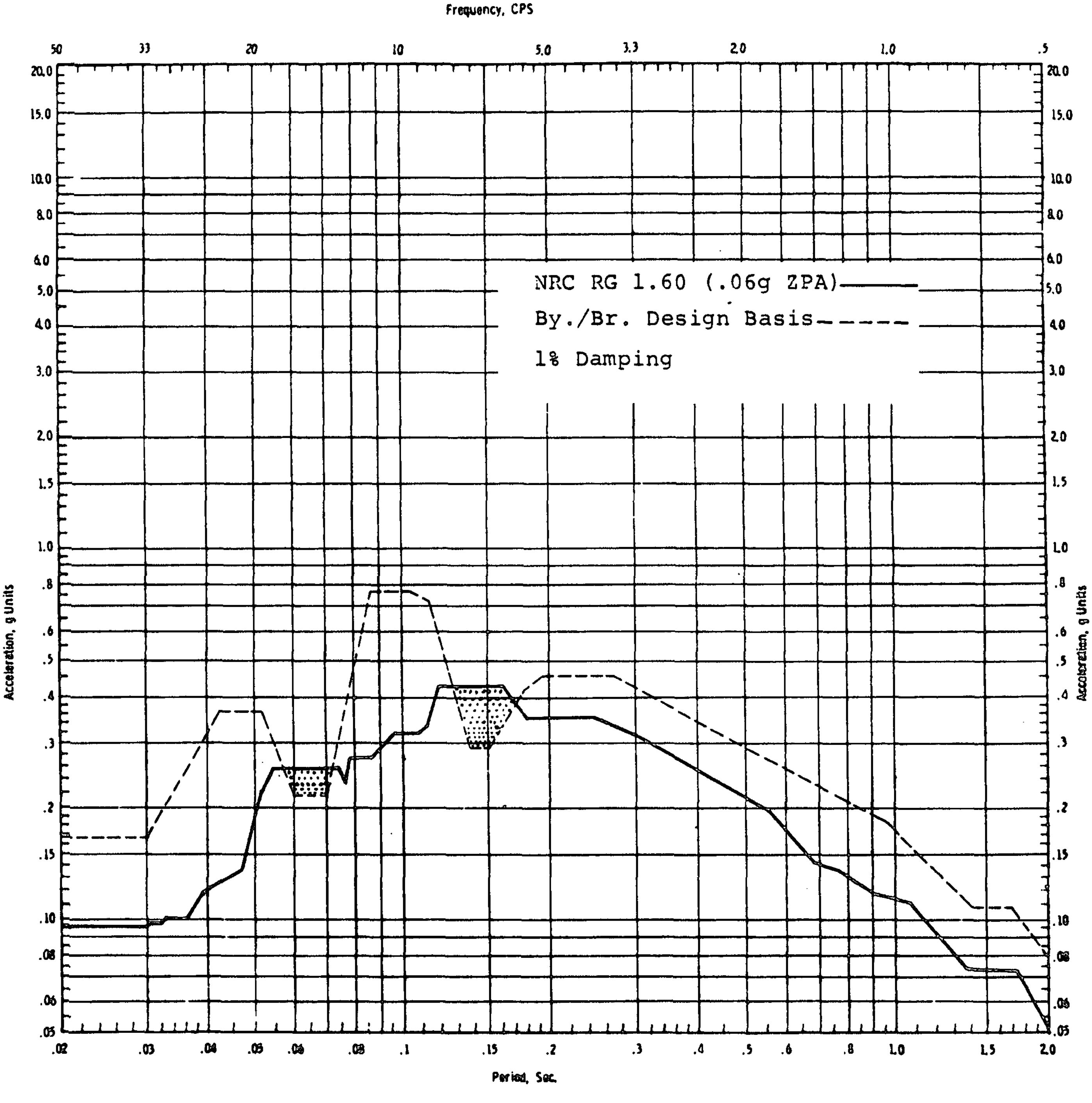

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FIGURE Q130.6-11
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 346'-0"; 364'-0"; 383'-0"; 401'-0"

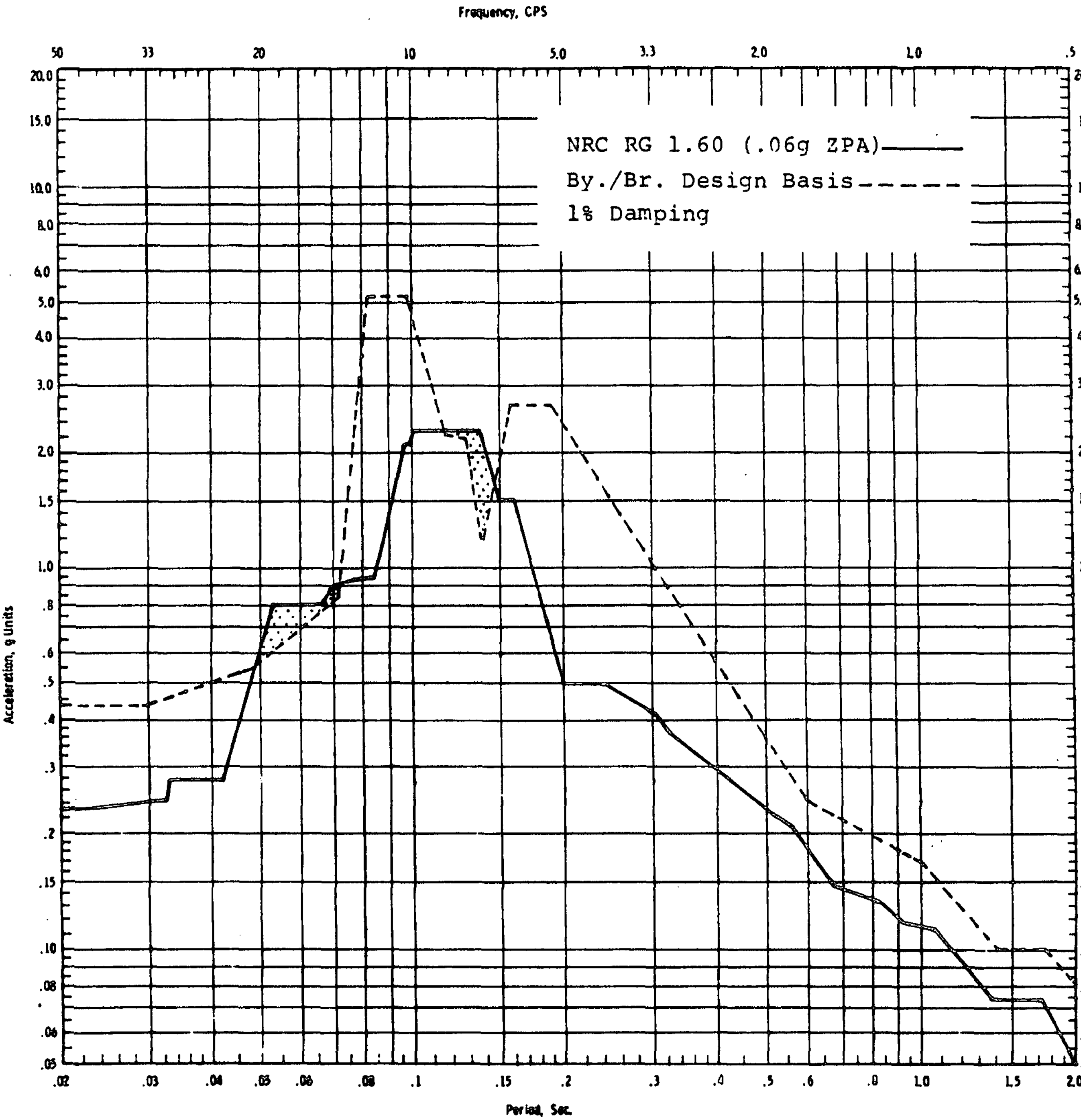

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FIGURE Q130.6-12
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 346'-0"; 364'-0"; 383'-0"; 401'-0"

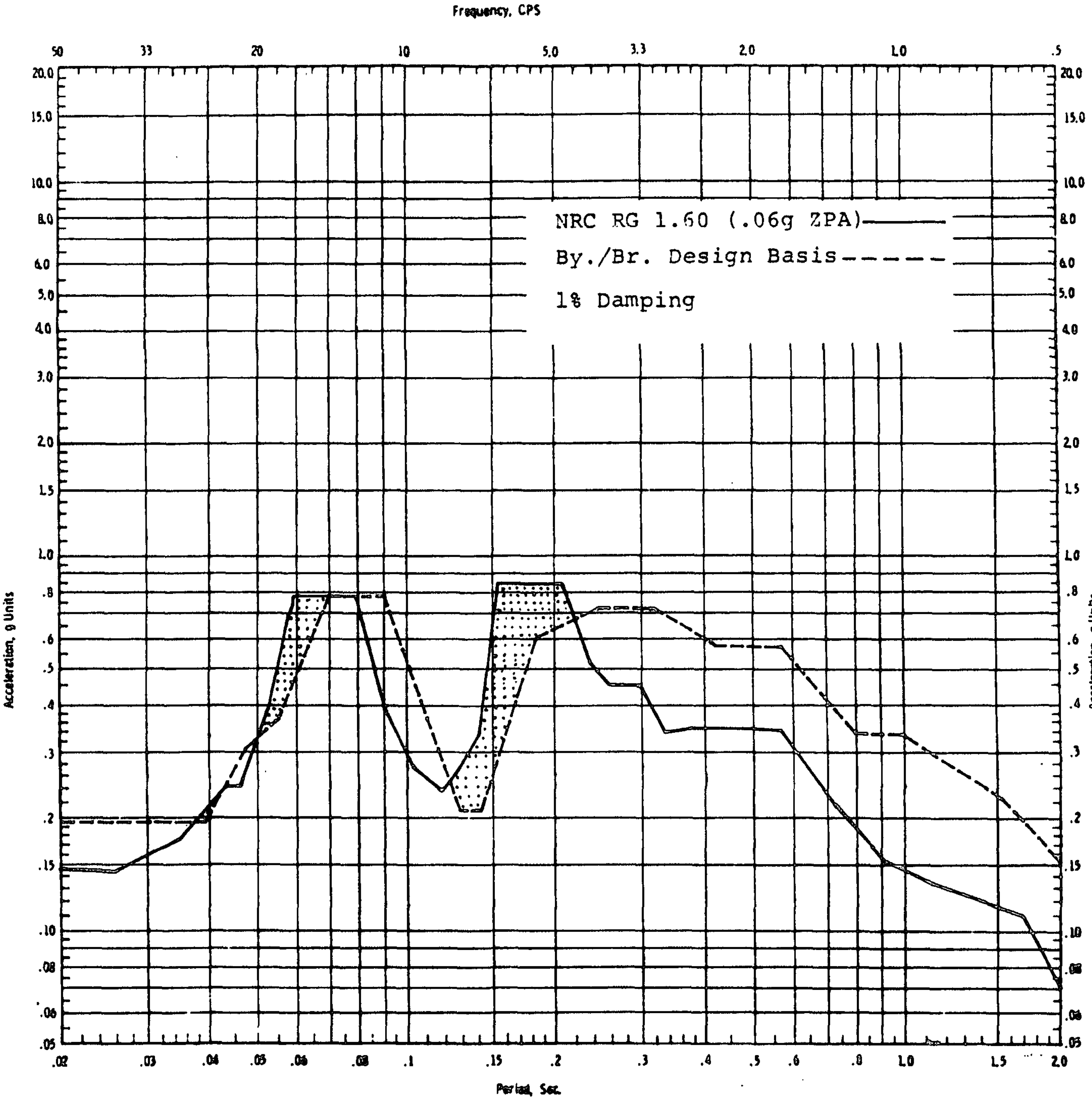

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FIGURE Q130.6-13
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, EW
LOCATION: AUXILIARY BUILDING
ELEVATION: 401'-0"

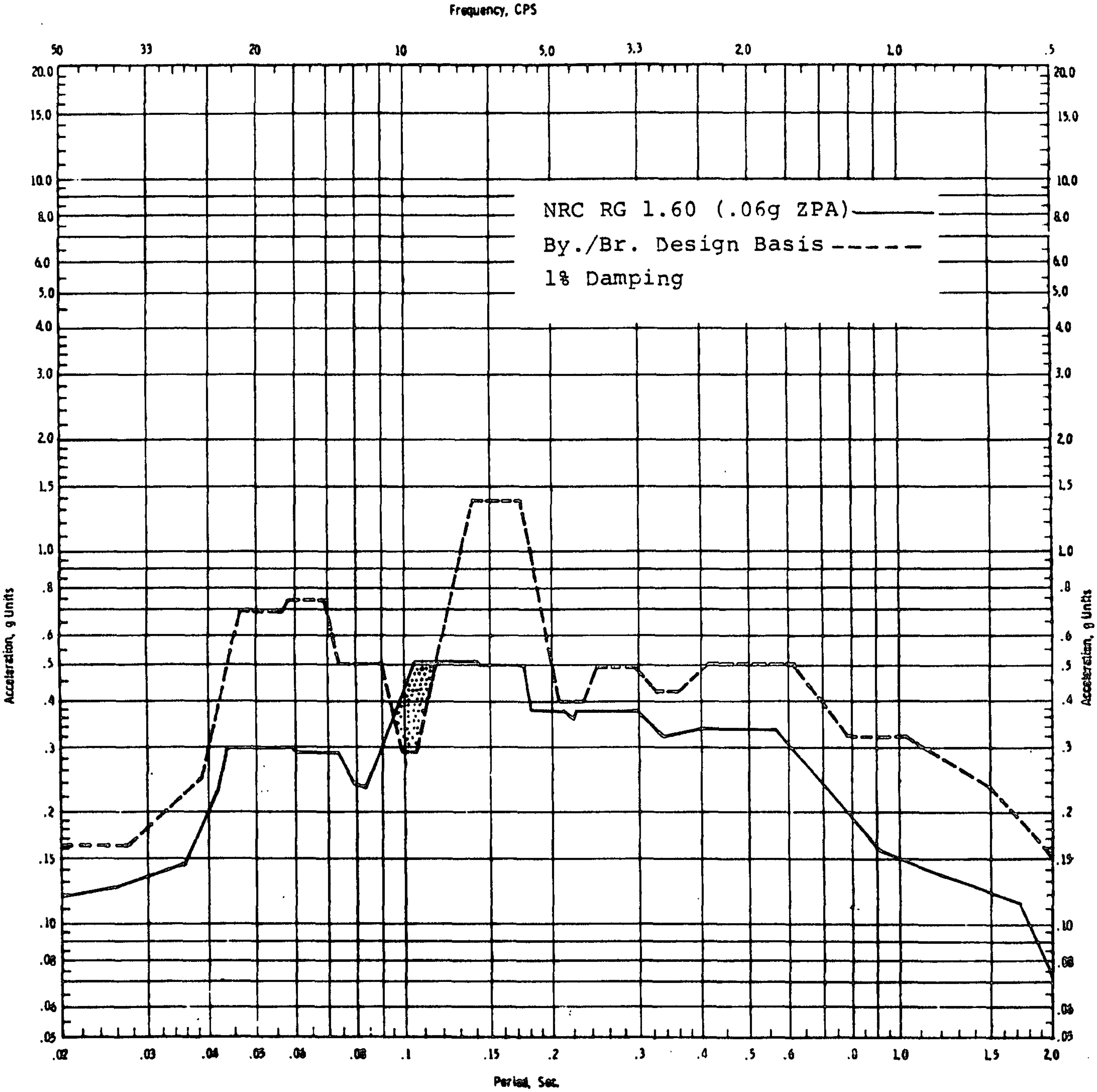

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FIGURE Q130.6-14
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: AUXILIARY BUILDING
ELEVATION: 401'-0"

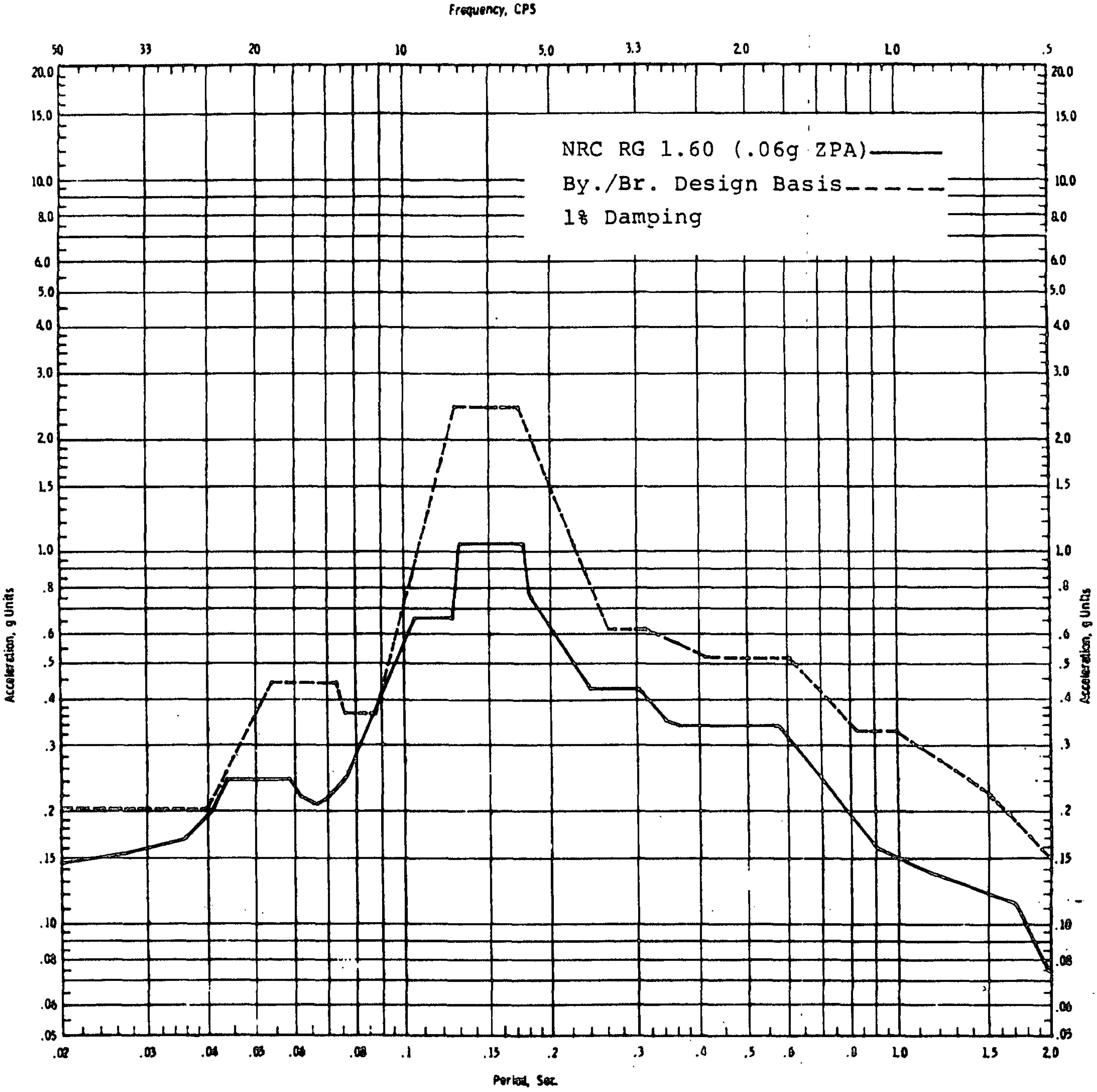

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FIGURE Q130.6-15
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: AUXILIARY BLDG., TURBINE BLDG.,
HEATER BAY—ELEVATION: 426'-0"

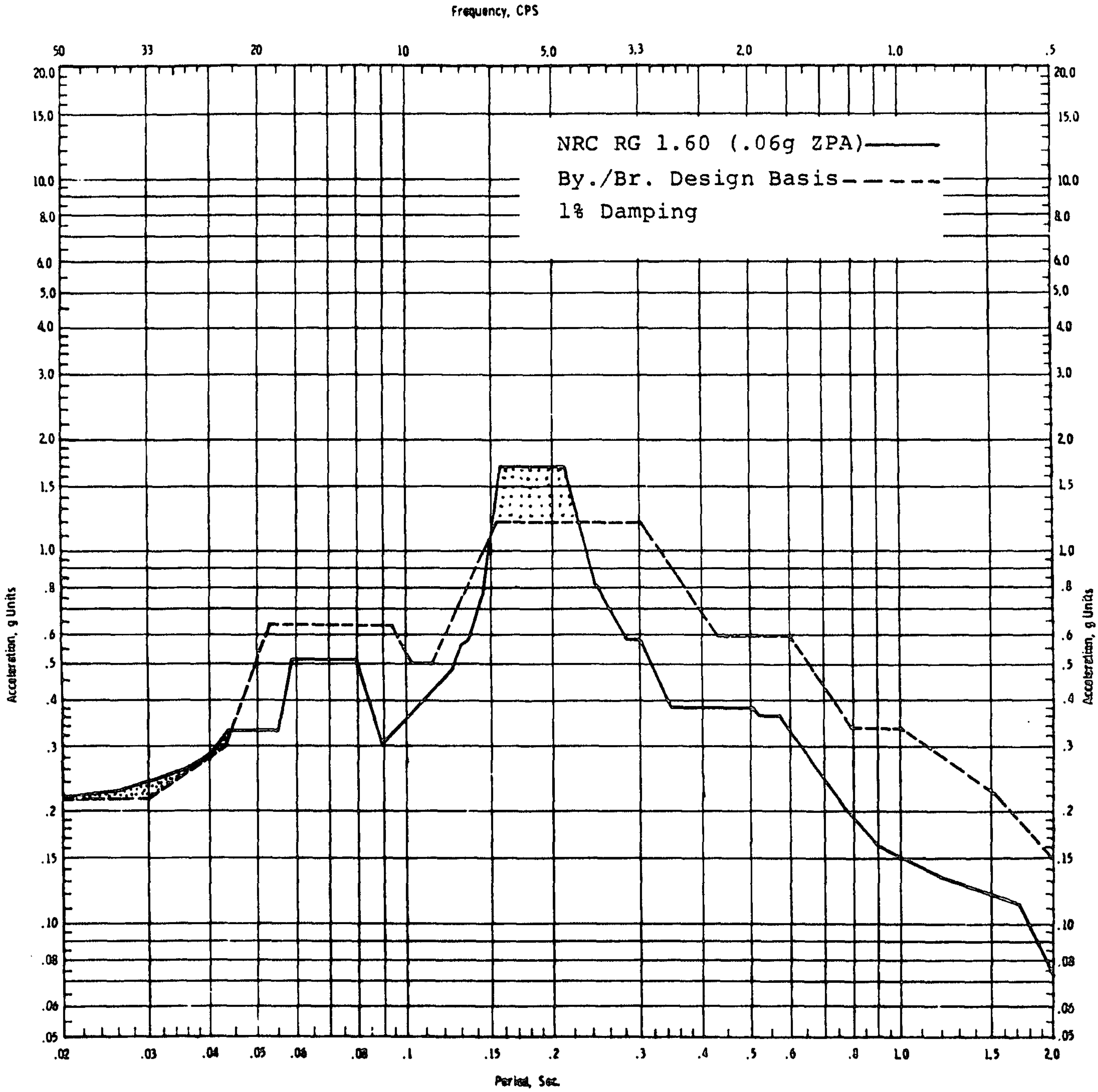

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FIGURE Q130.6-16
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, EW
LOCATION: AUXILIARY BLDG., TURBINE BLDG., HEATER BAY—ELEVATION: 426'-0"

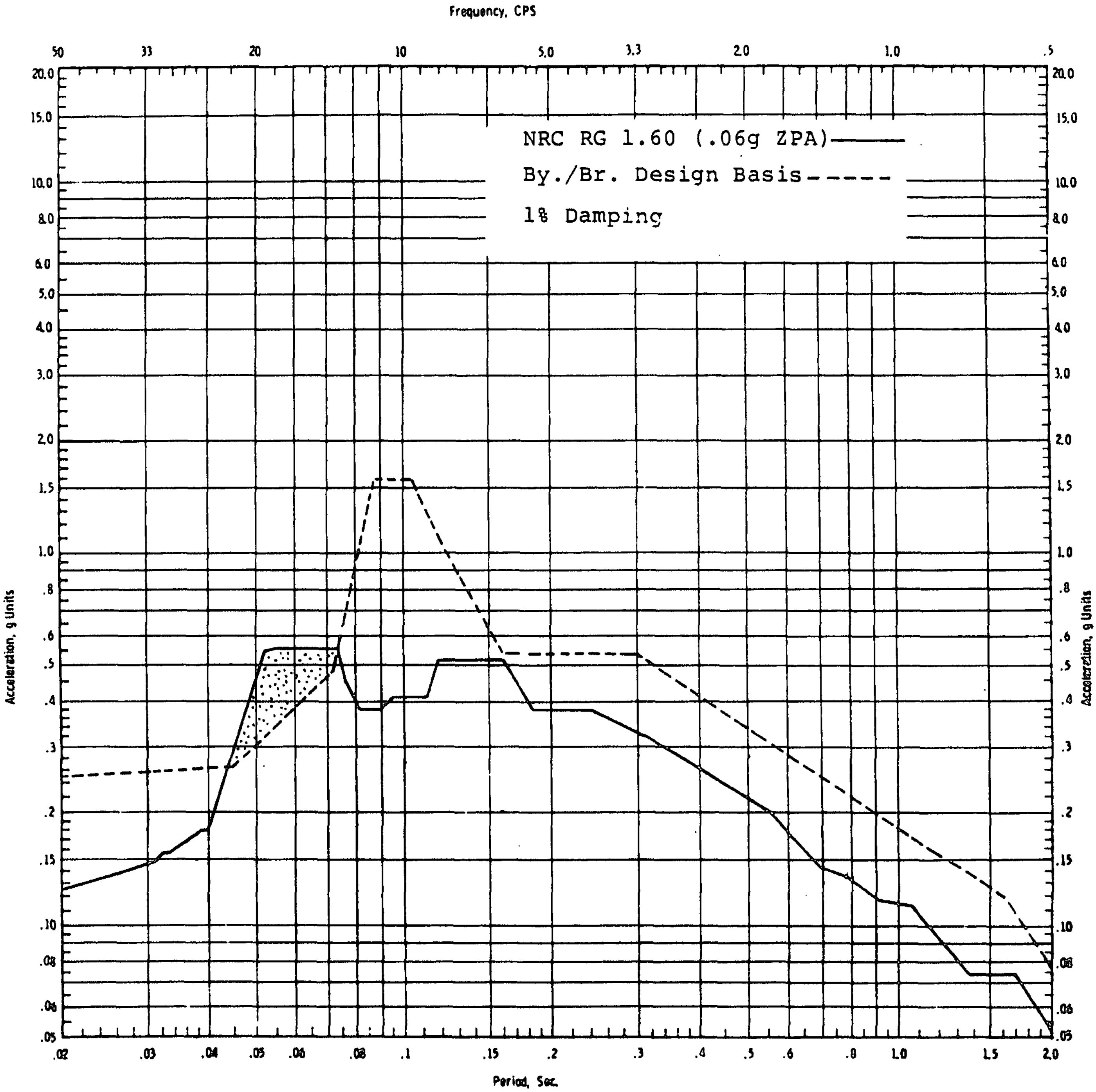

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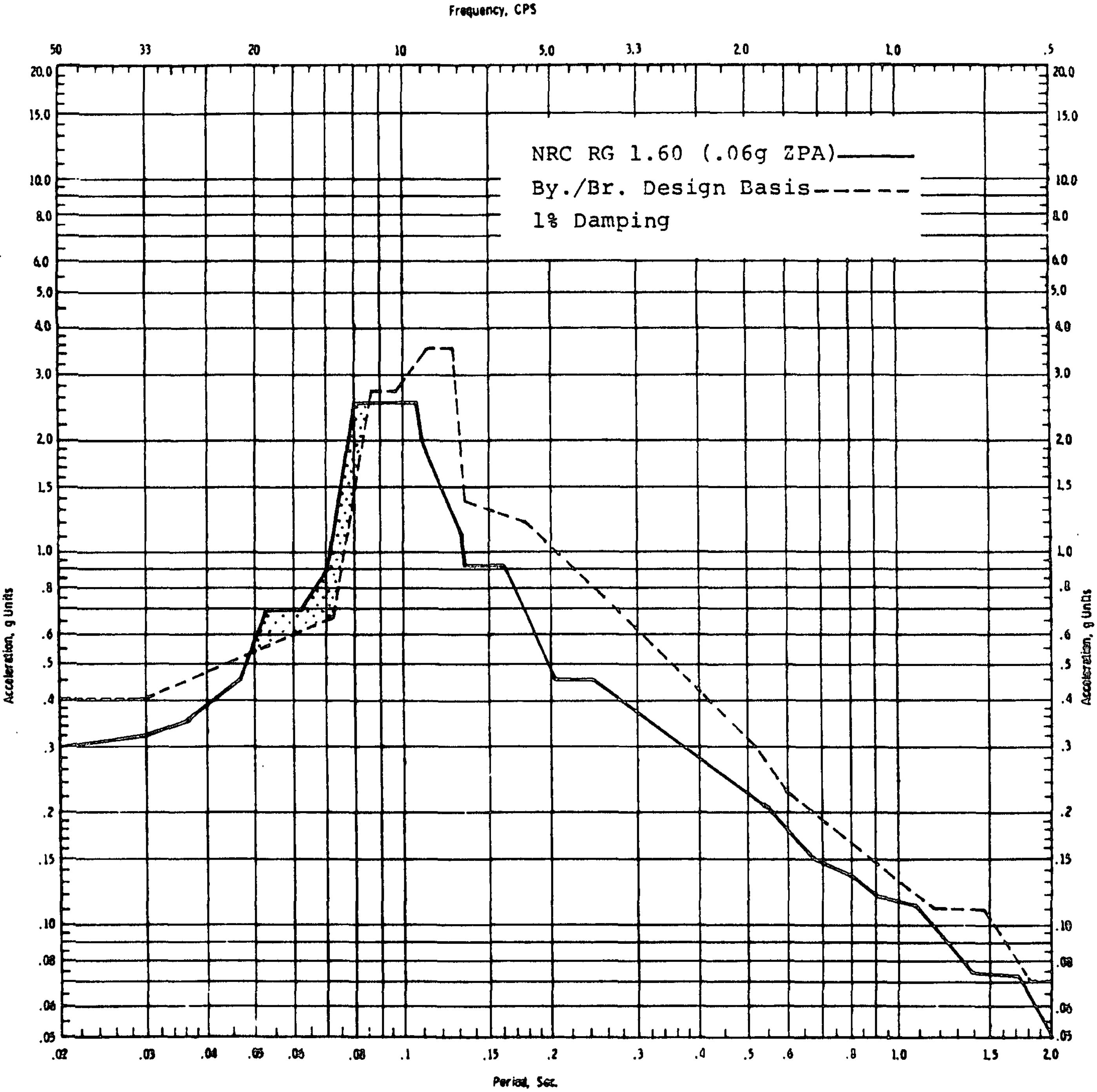
FIGURE Q130.6-17
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 426'-0"; 439'-0"; 451'-0"



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6-18
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 426'-0"; 439'-0"; 451'-0"

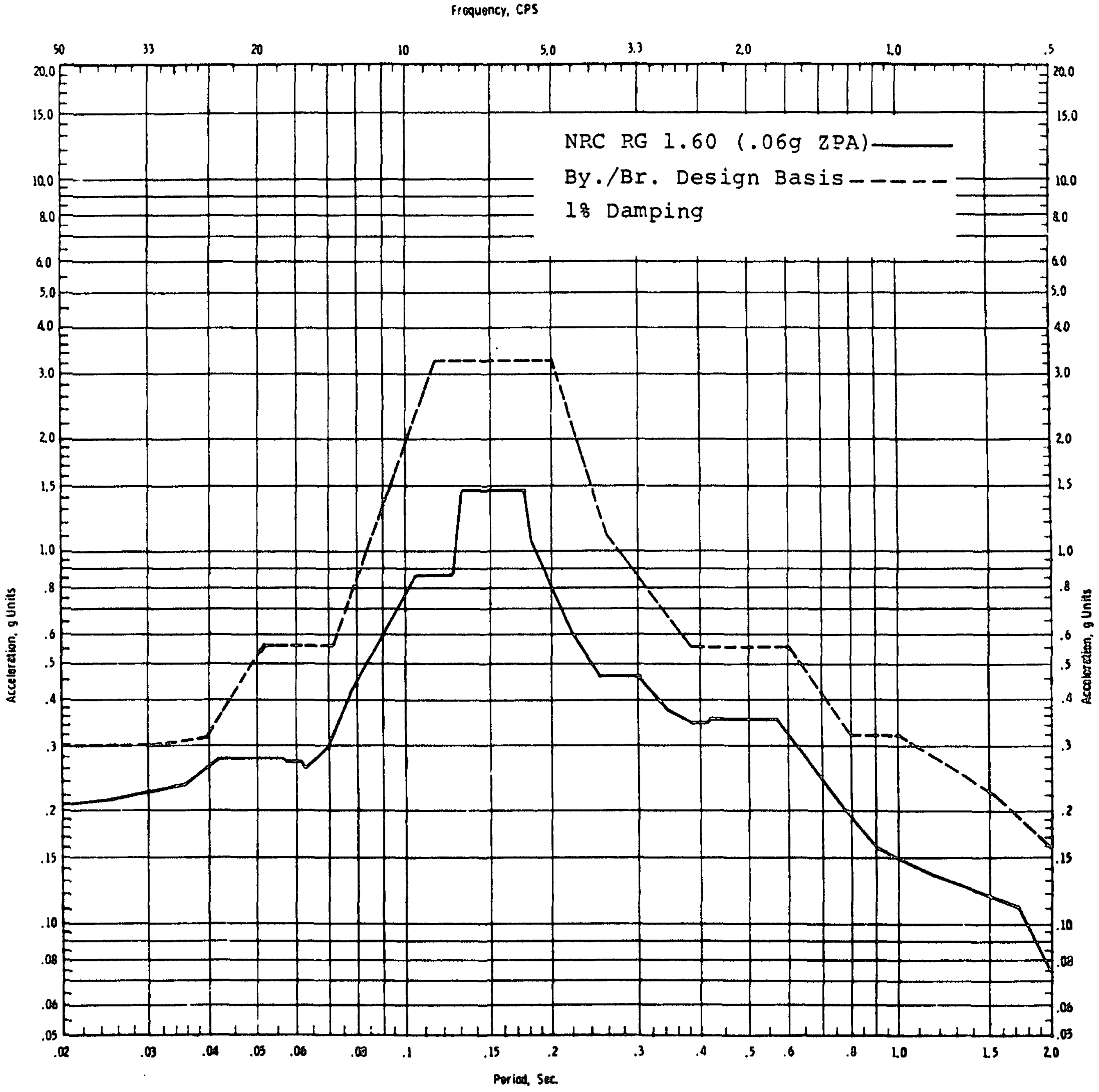

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FIGURE Q130.6-19
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: AUXILIARY BLDG., TURBINE BLDG., HEATER BAY—ELEVATION: 451'-0"

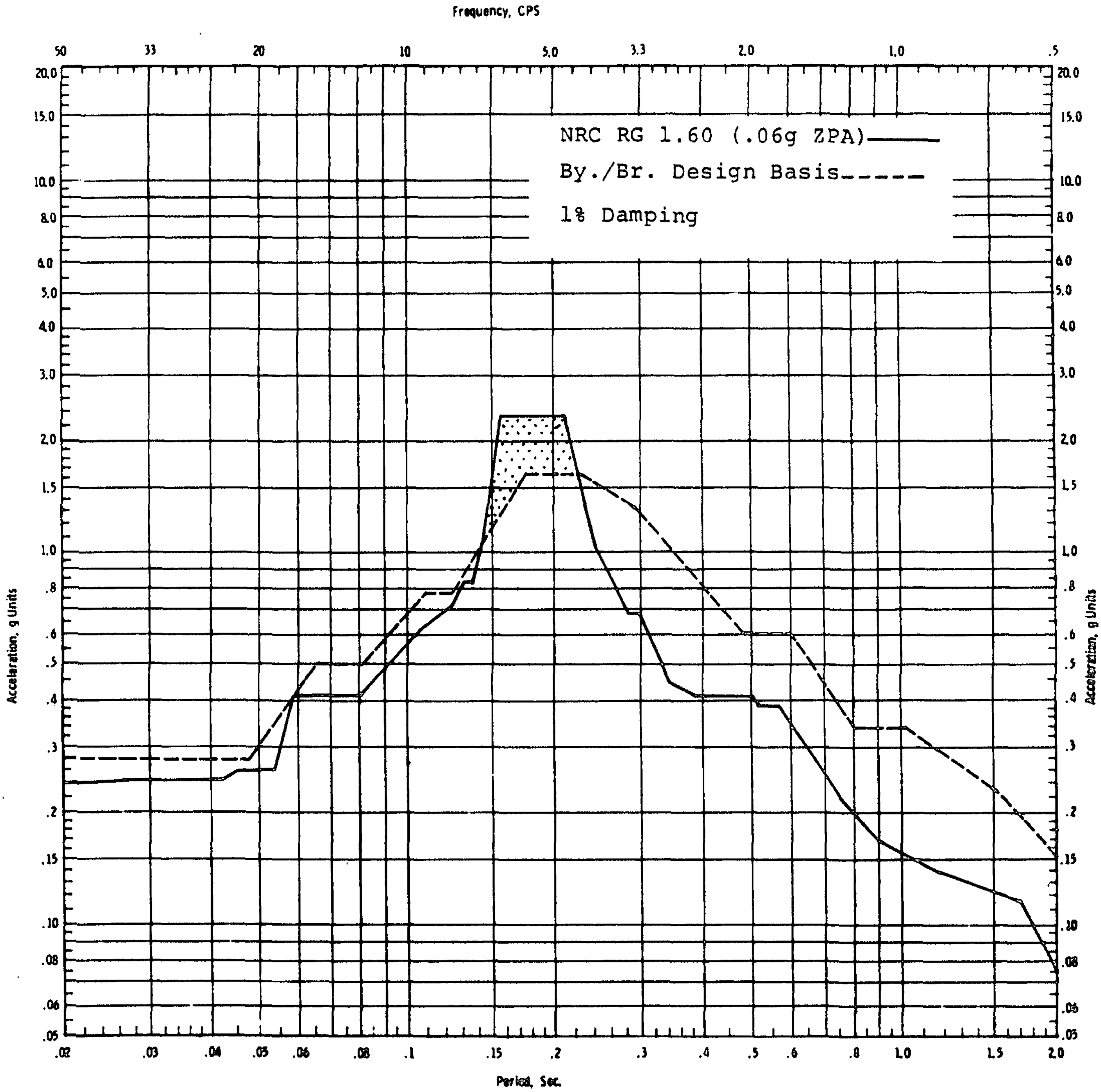

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FIGURE Q130.6-20
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, EW
LOCATION: AUXILIARY BLDG., TURBINE BLDG.,
HEATER BAY—ELEVATION: 451'-0"

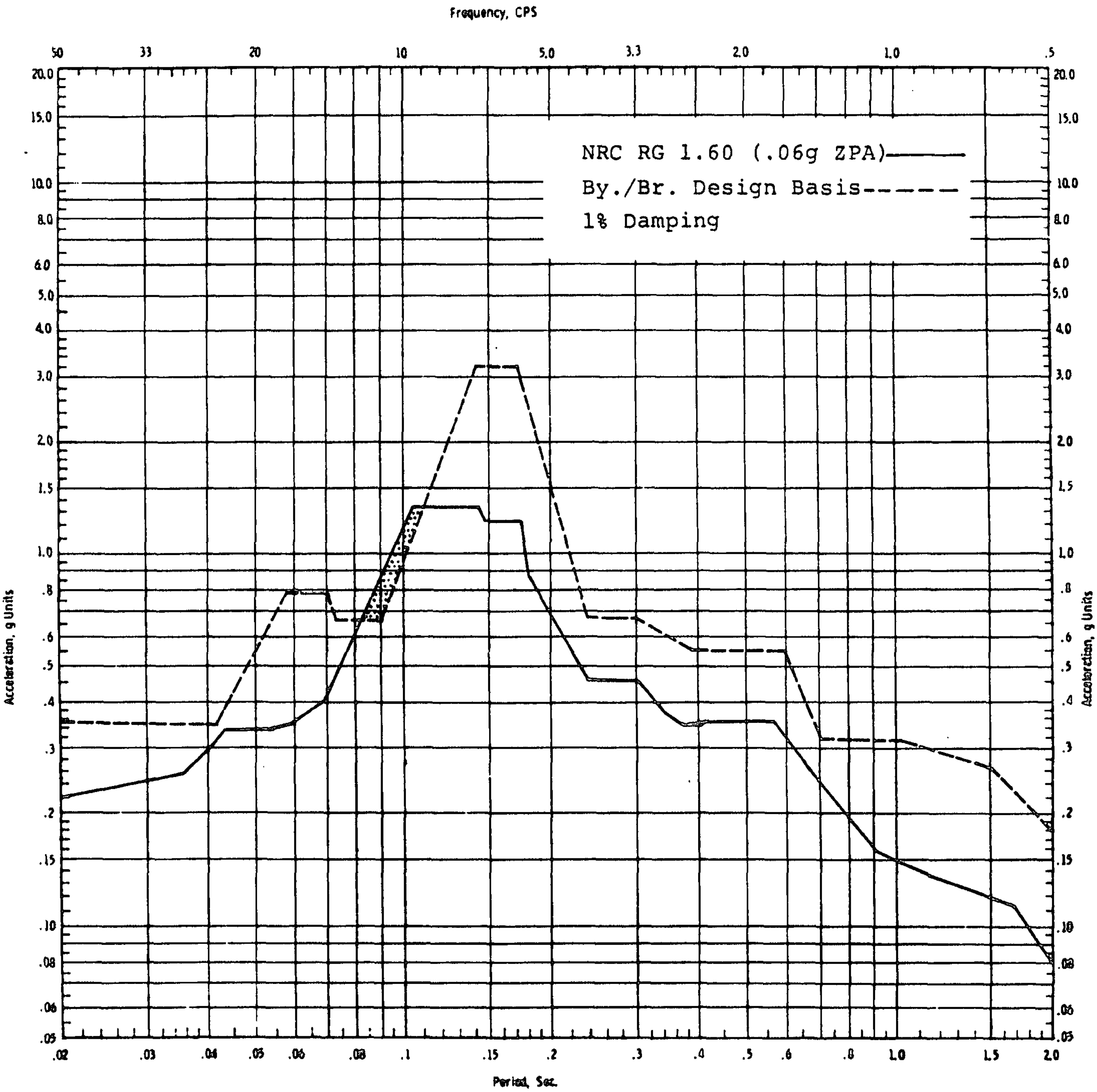

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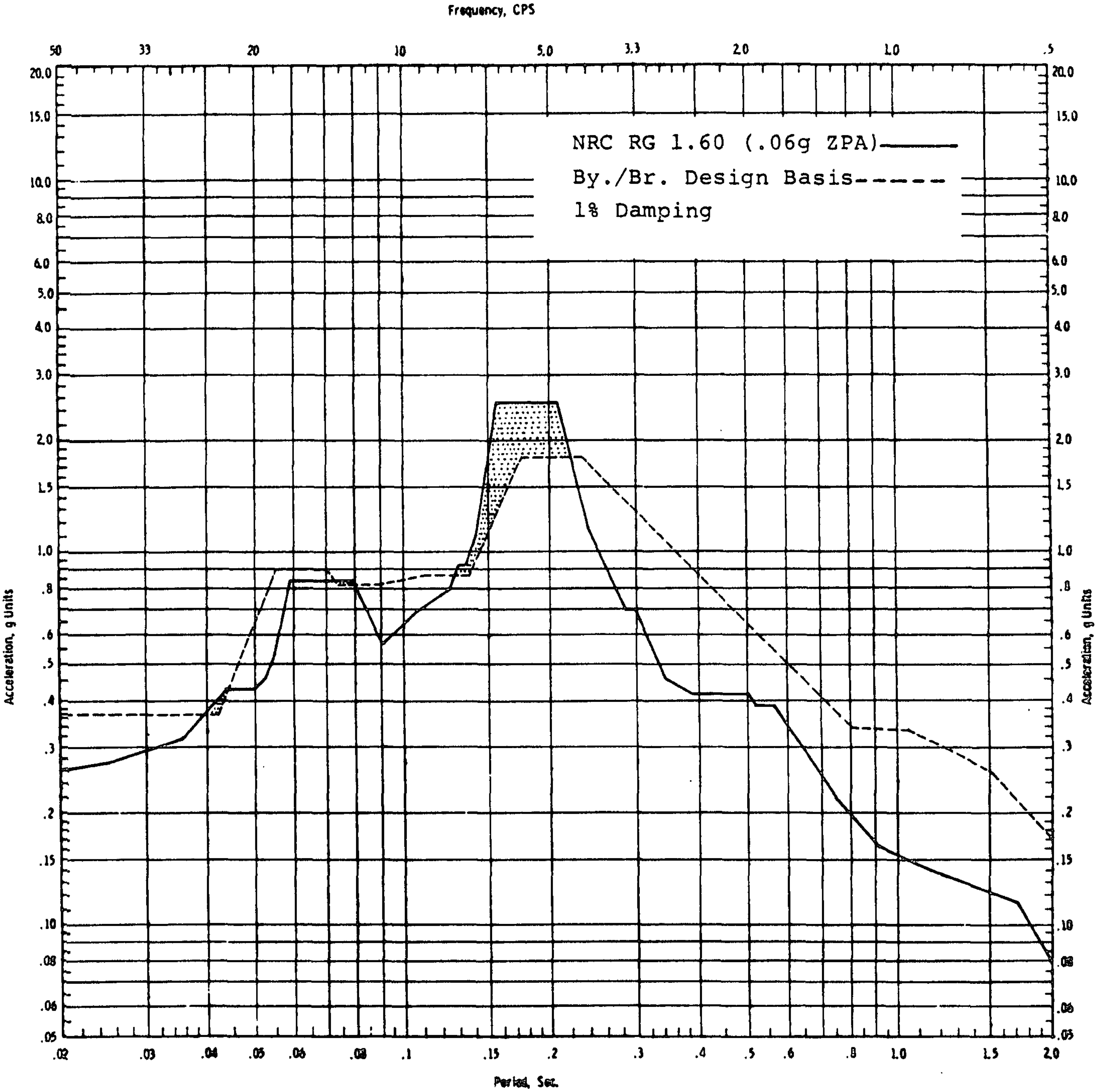
FIGURE Q130.6-21
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: AUXILIARY BUILDING
ELEVATION: 477'-0"



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6-22
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, EW
LOCATION: AUXILIARY BUILDING
ELEVATION: 477'-0"

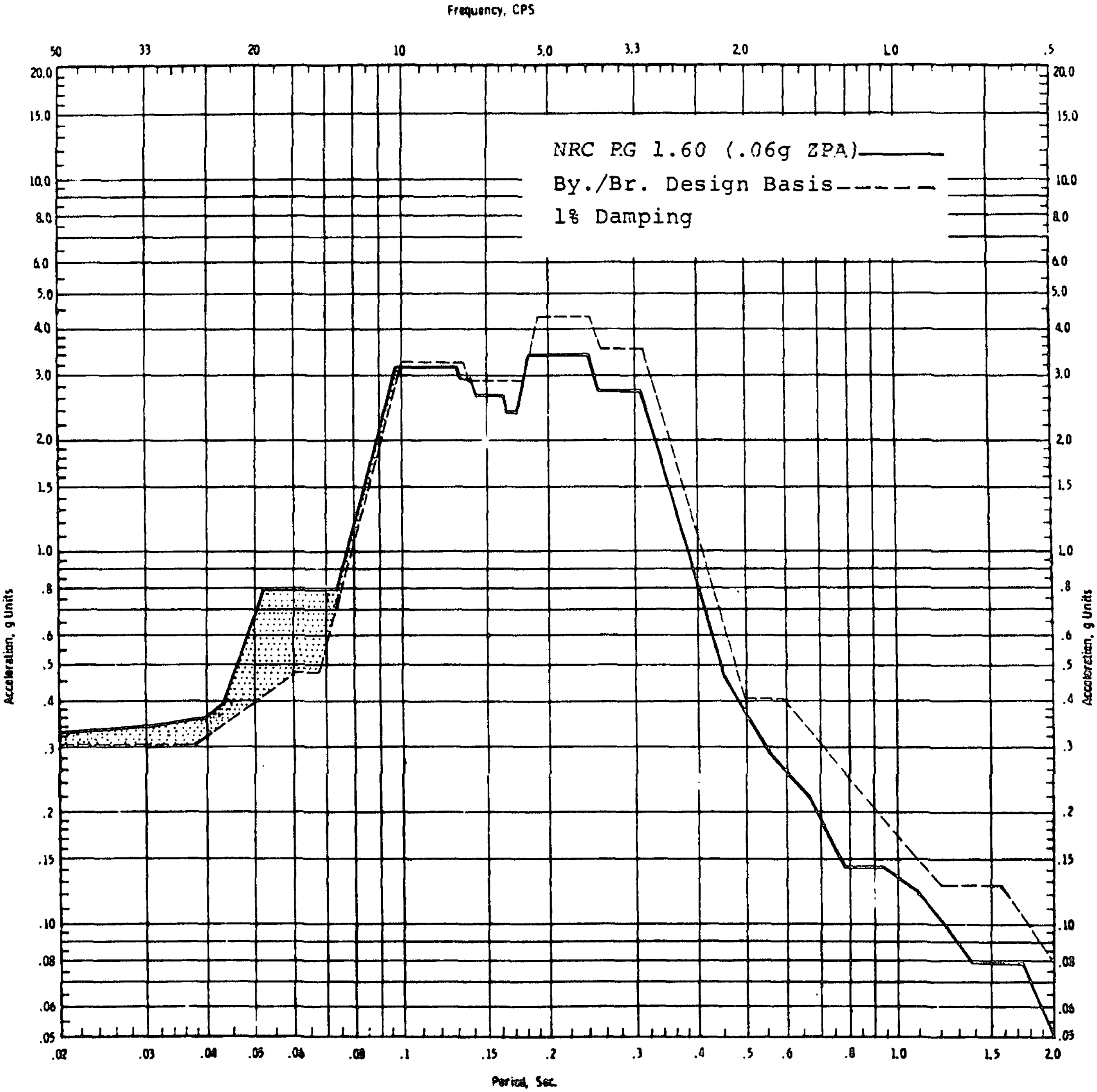

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FIGURE Q130.6-23
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 467'-0"; 477'-0"

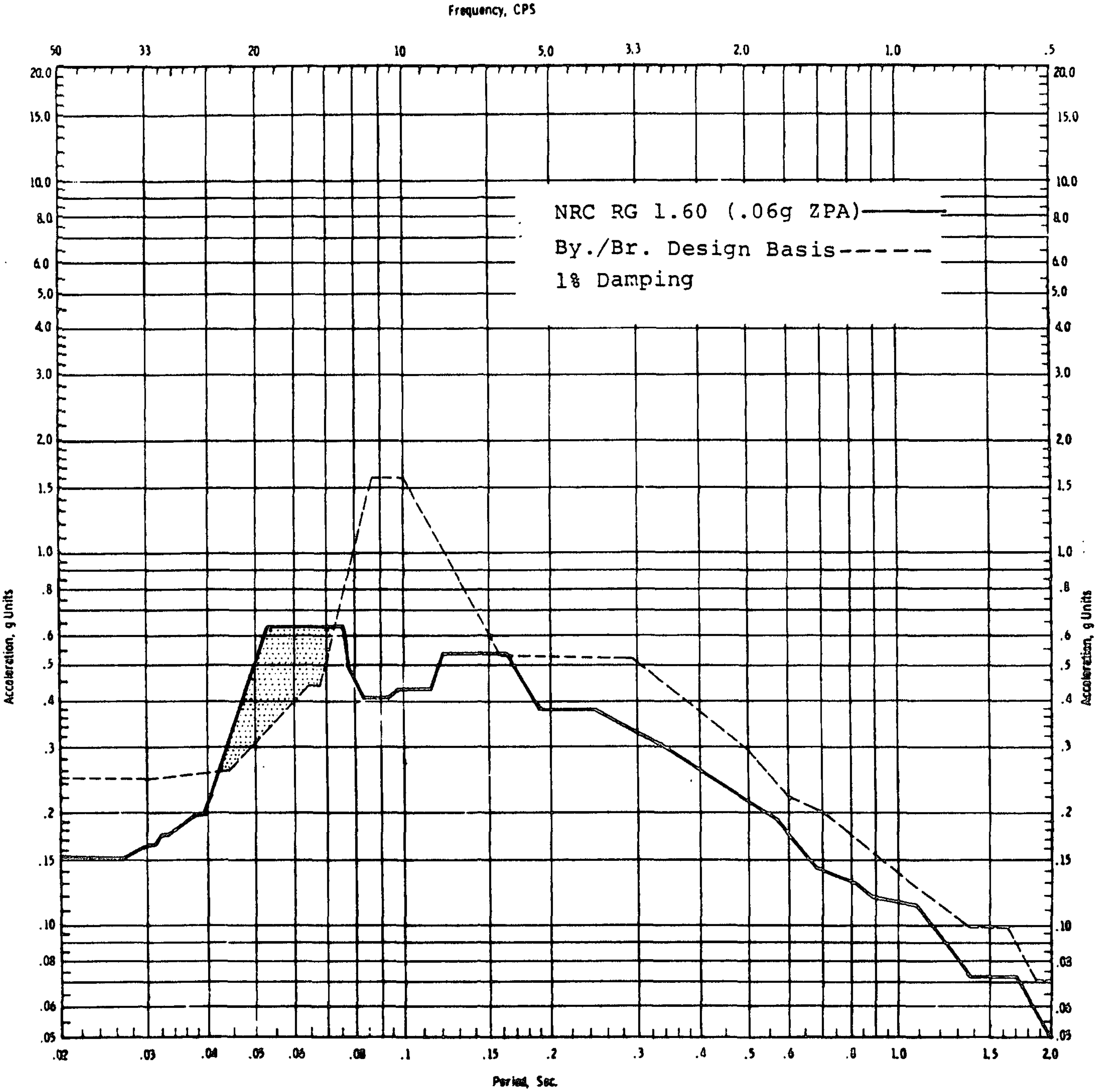

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FIGURE Q130.6-24
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 467'-0"; 477'-0"; 473'-0"; 485'-0"

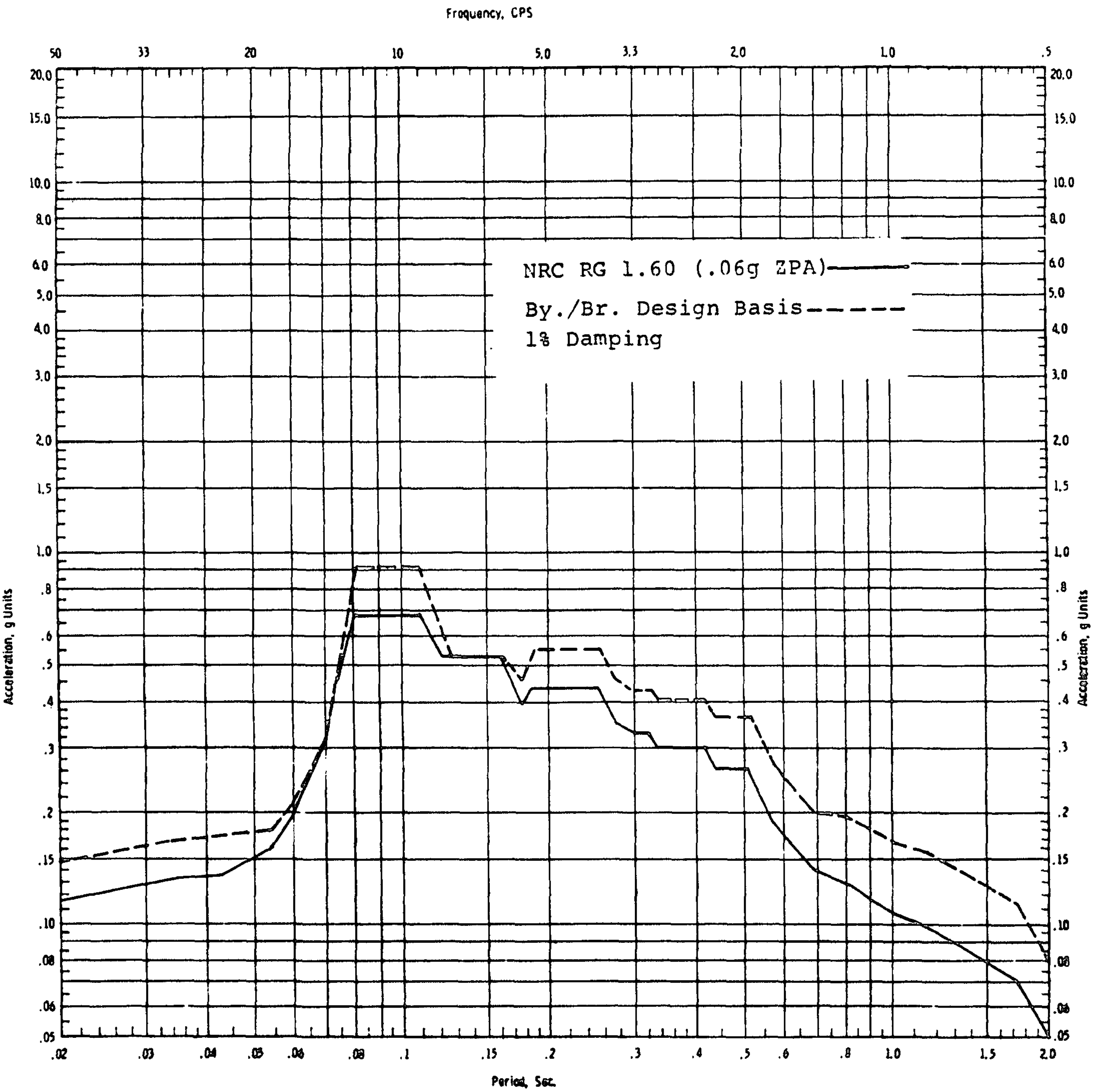

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FIGURE Q130.6-25
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: CONTAINMENT BUILDING WALL
ELEVATION: 424'-0"; 436'-0"

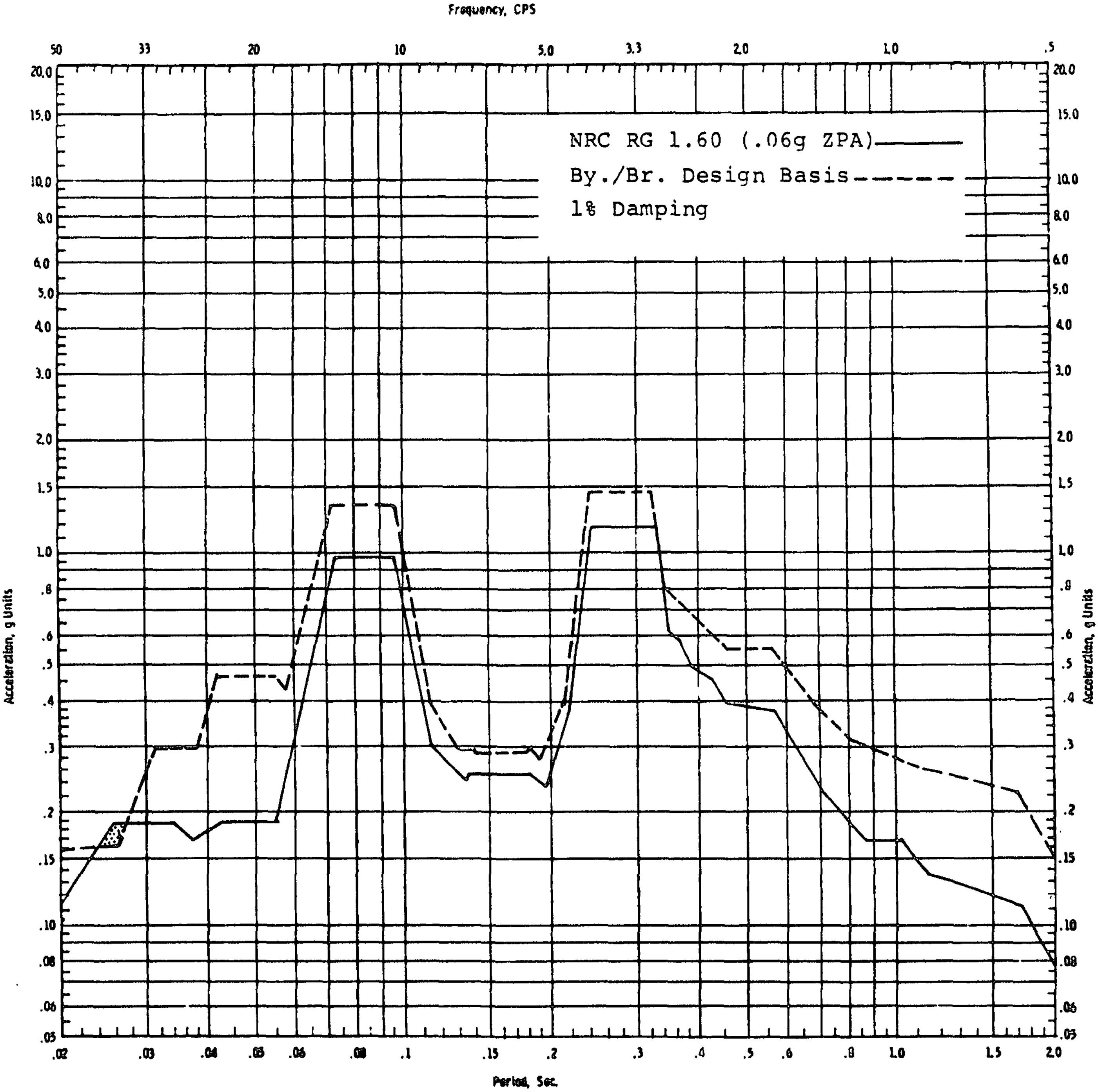

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FIGURE Q130.6-26
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: CONTAINMENT BUILDING
ELEVATION: 424'-0"; 436'-0"

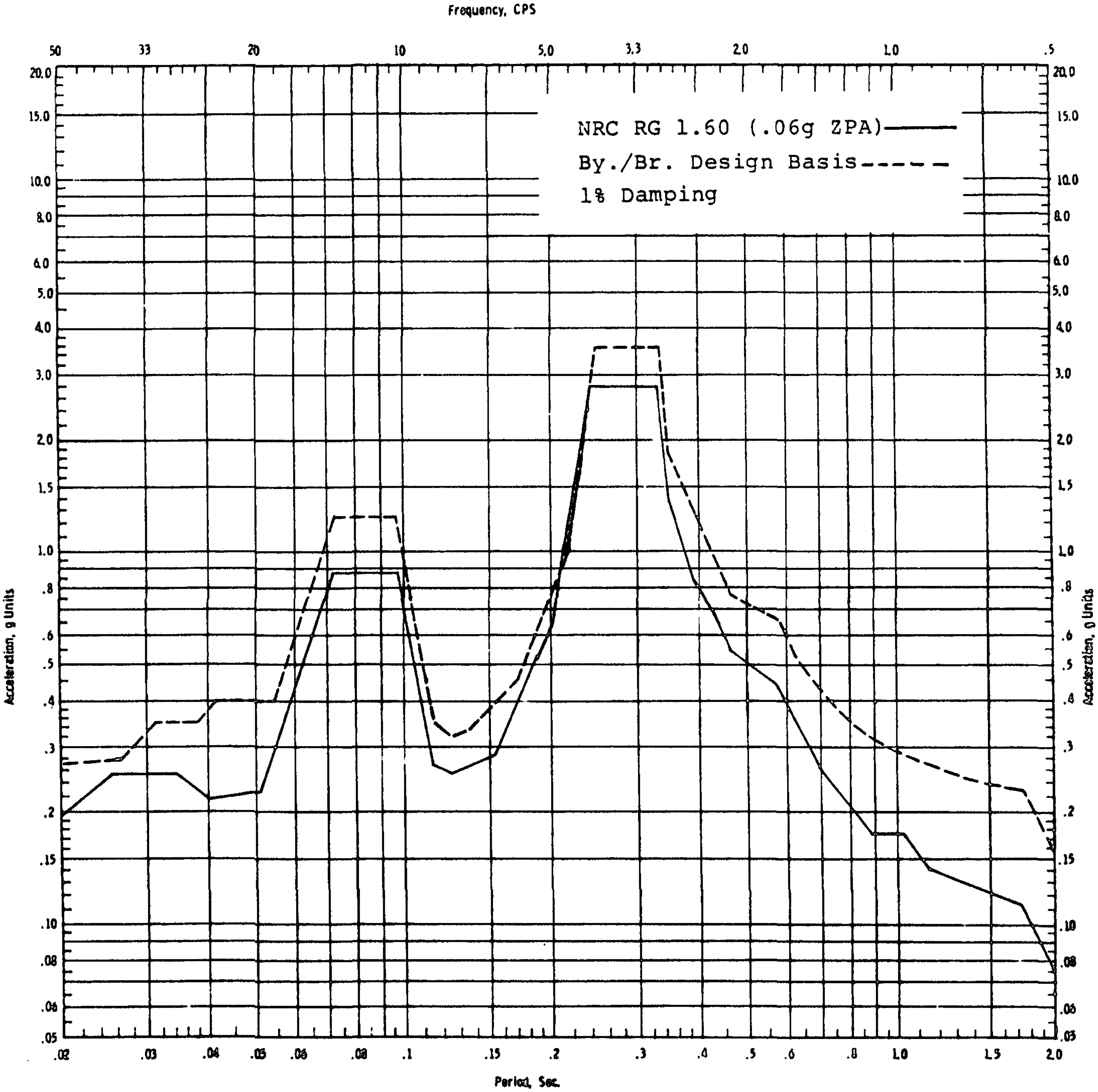

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FIGURE Q130.6-27
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS AND EW
LOCATION: CONTAINMENT BUILDING
ELEVATION: 496'-0"

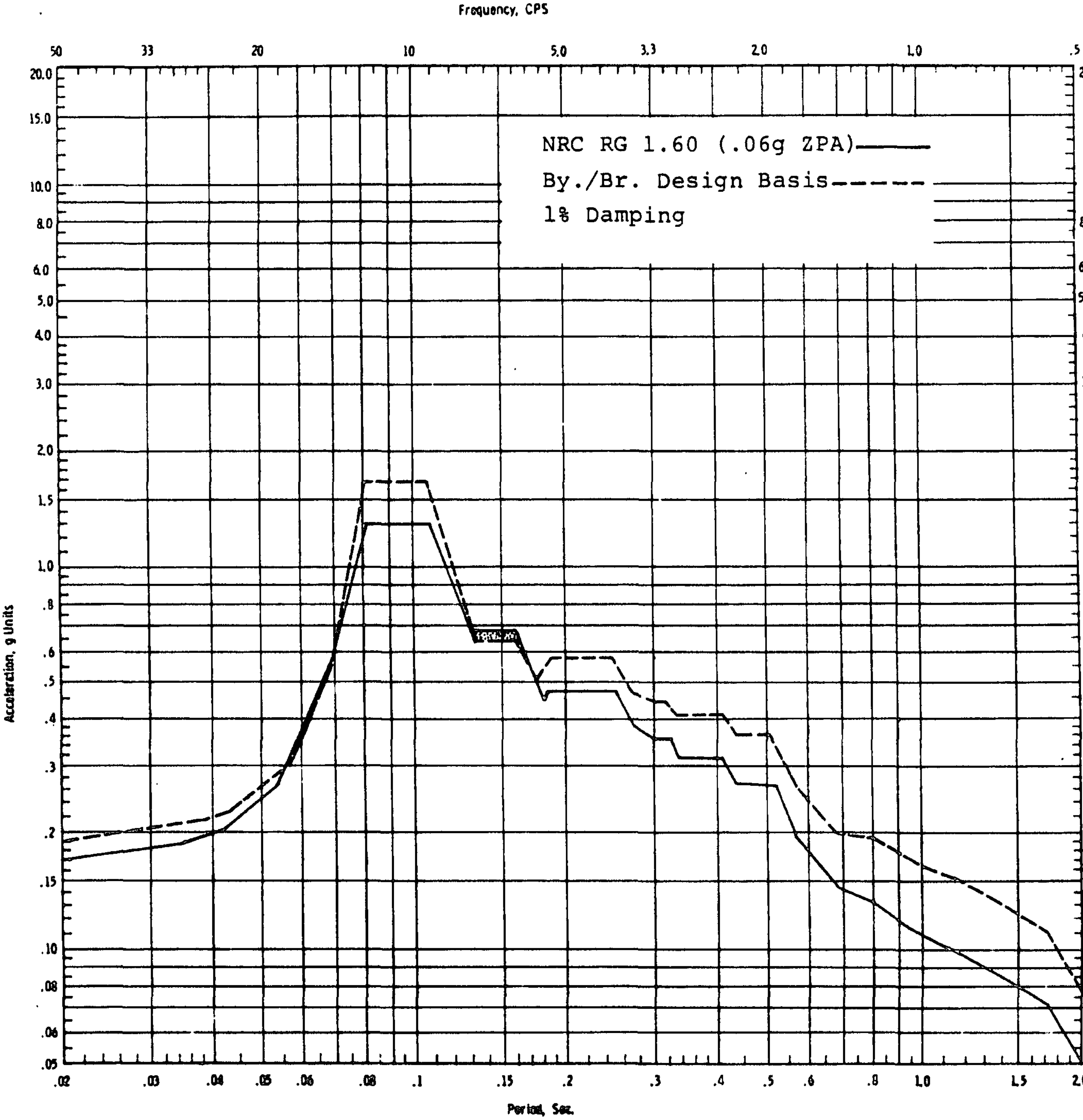

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FIGURE Q130.6-28
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: CONTAINMENT BUILDING
ELEVATION: 496'-0"

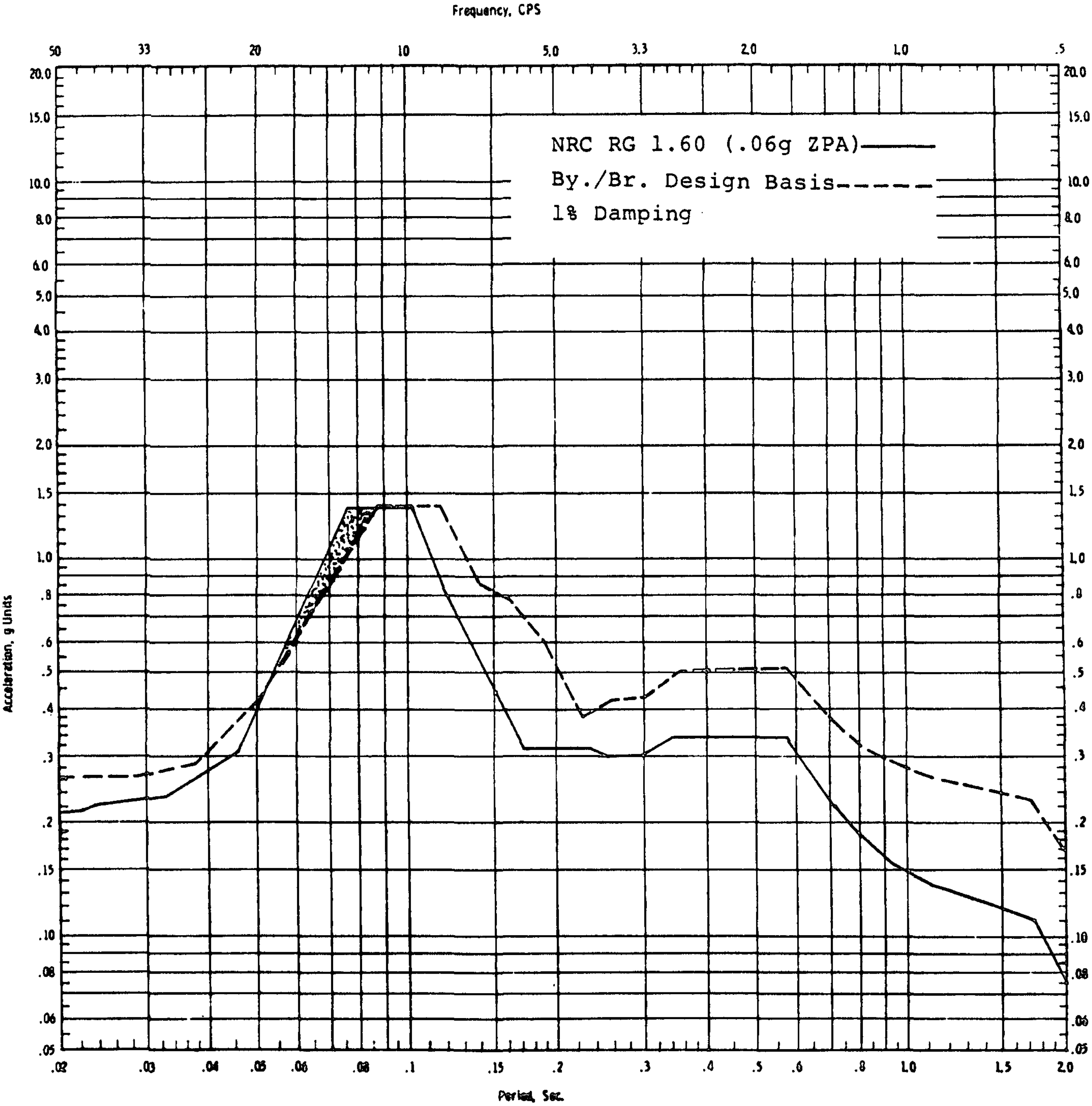

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FIGURE Q130.6-29
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: CONTAINMENT INNER STRUCTURE
ELEVATION: 426'-0"

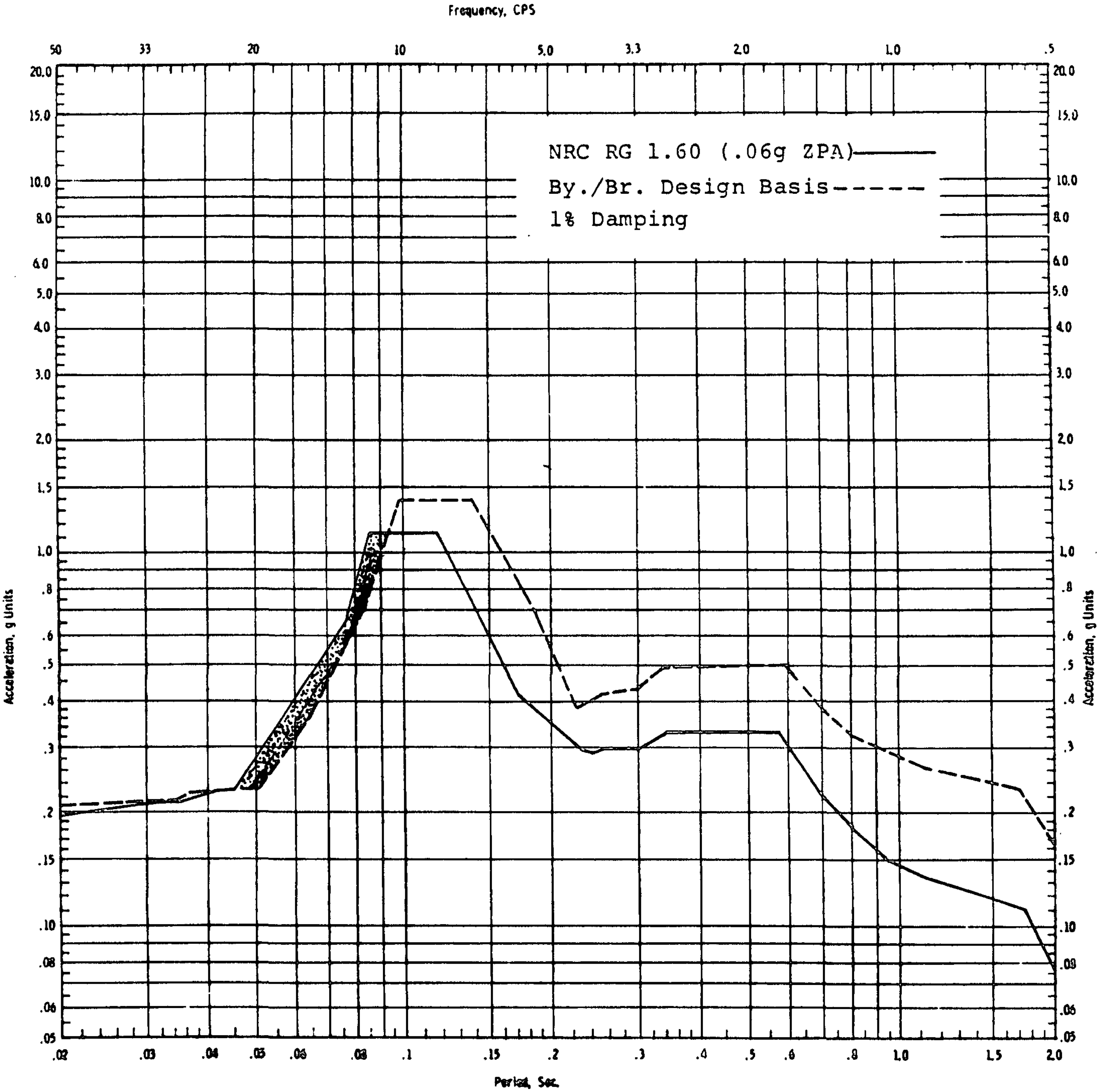

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FIGURE Q130.6-30
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, EW
LOCATION: CONTAINMENT INNER STRUCTURE
ELEVATION: 426'-0"

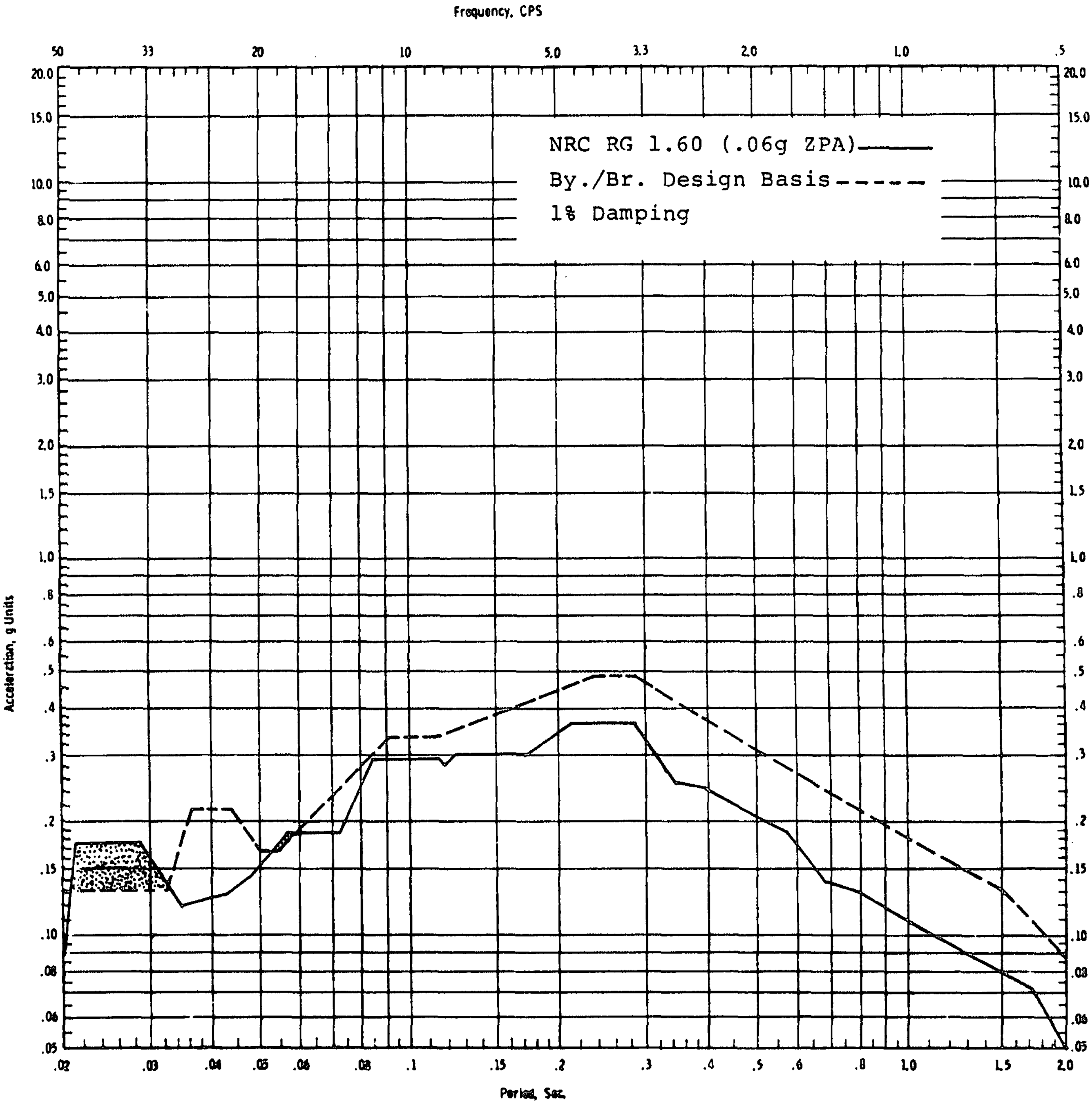

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FIGURE Q130.6-31
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: CONTAINMENT INNER STRUCTURE WALL
ELEVATION: 412'-0"; 426'-0"

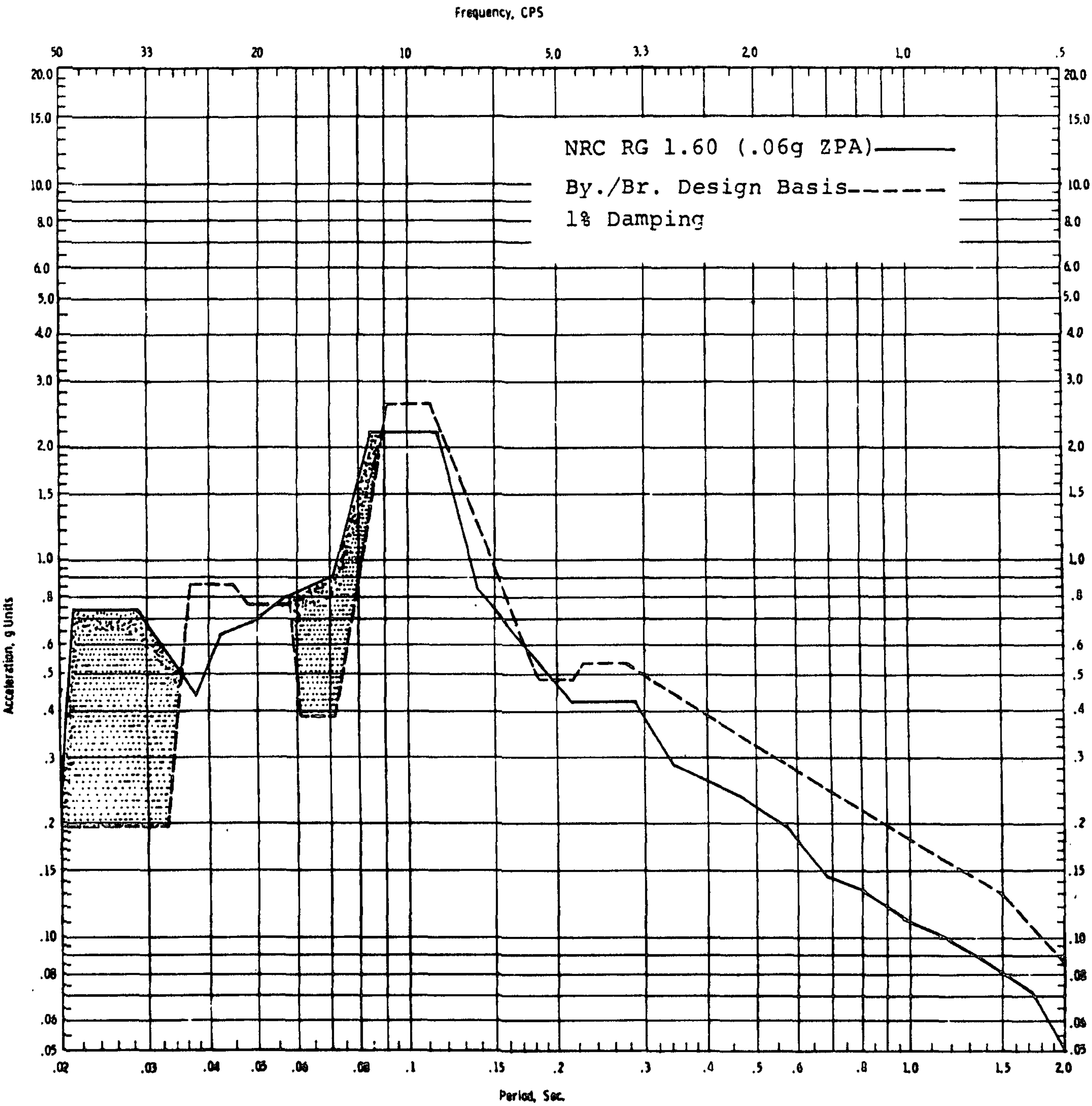

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FIGURE Q130.6-32
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, SLAB
LOCATION: CONTAINMENT INNER STRUCTURE SLAB
ELEVATION: 390'-0"; 401'-0"; 412'-0"; 426'-0"

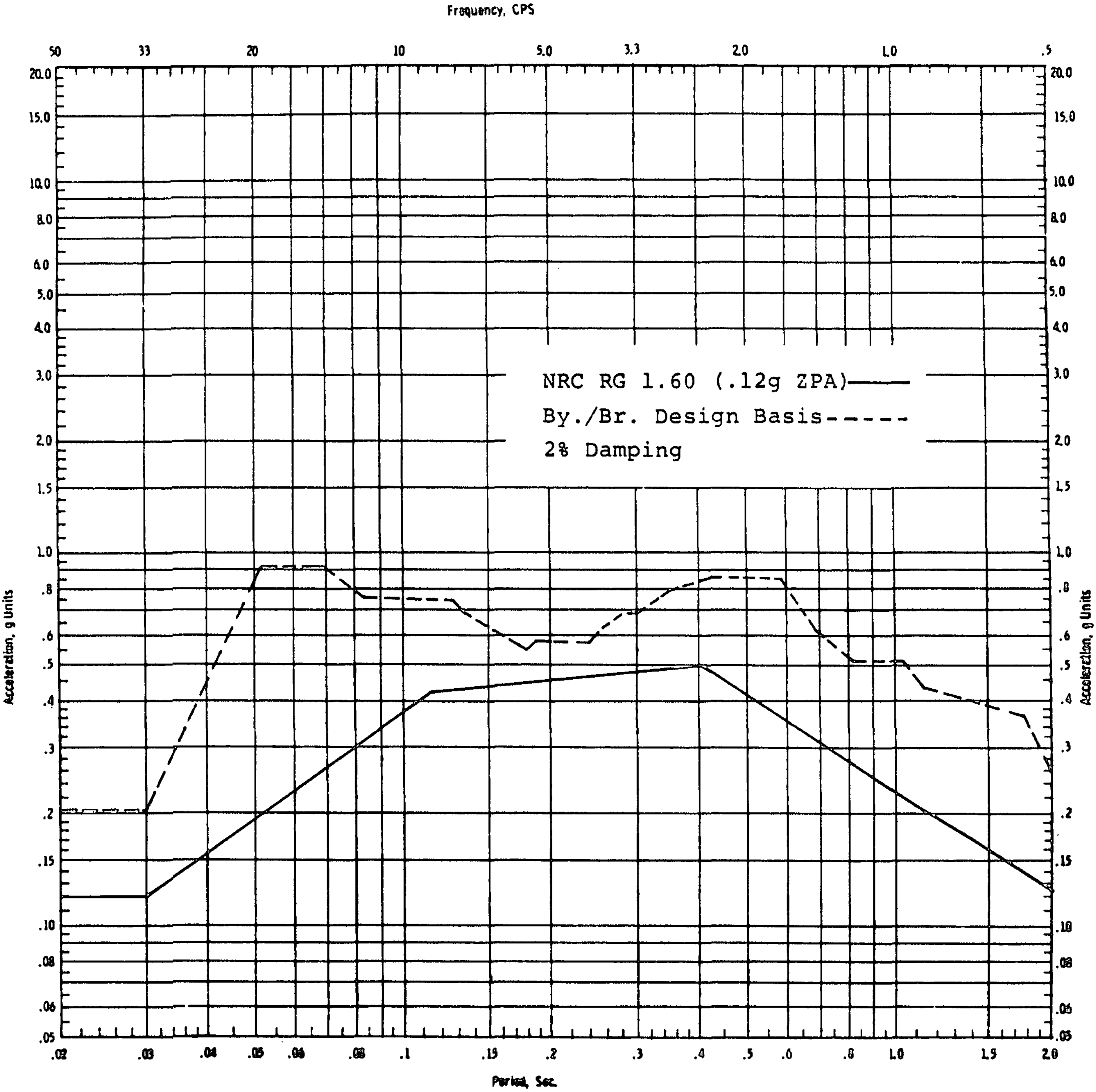

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FIGURE Q130.6-33
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS AND EW
LOCATION: AUXILIARY AND CONTAINMENT BLDG.
ELEVATION: 330'-0"; 374'-0"

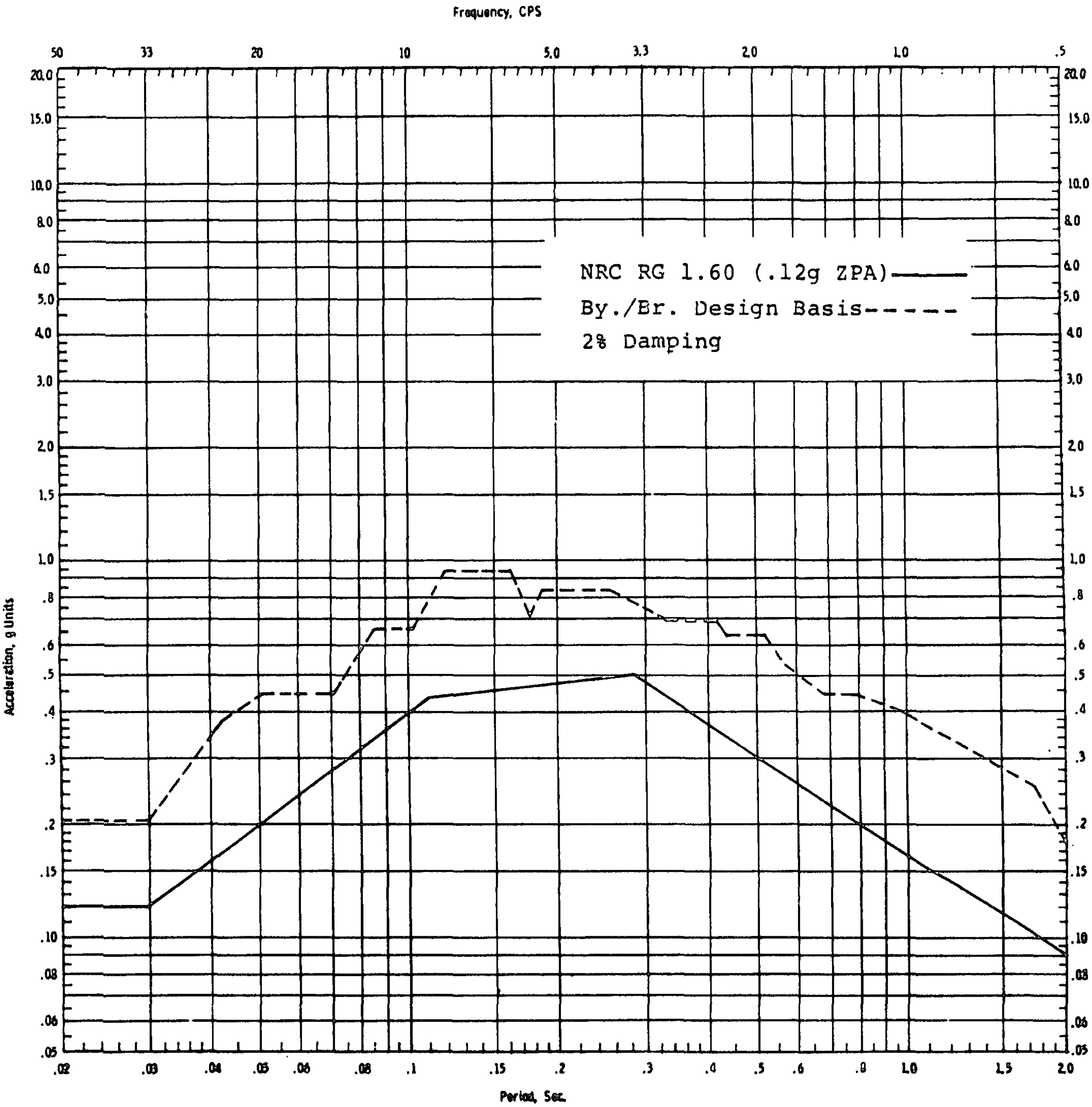

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FIGURE Q130.6-34
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL AND SLAB
LOCATION: AUXILIARY AND CONTAINMENT BLDG.
ELEVATION: 330'-0"; 374'-0"

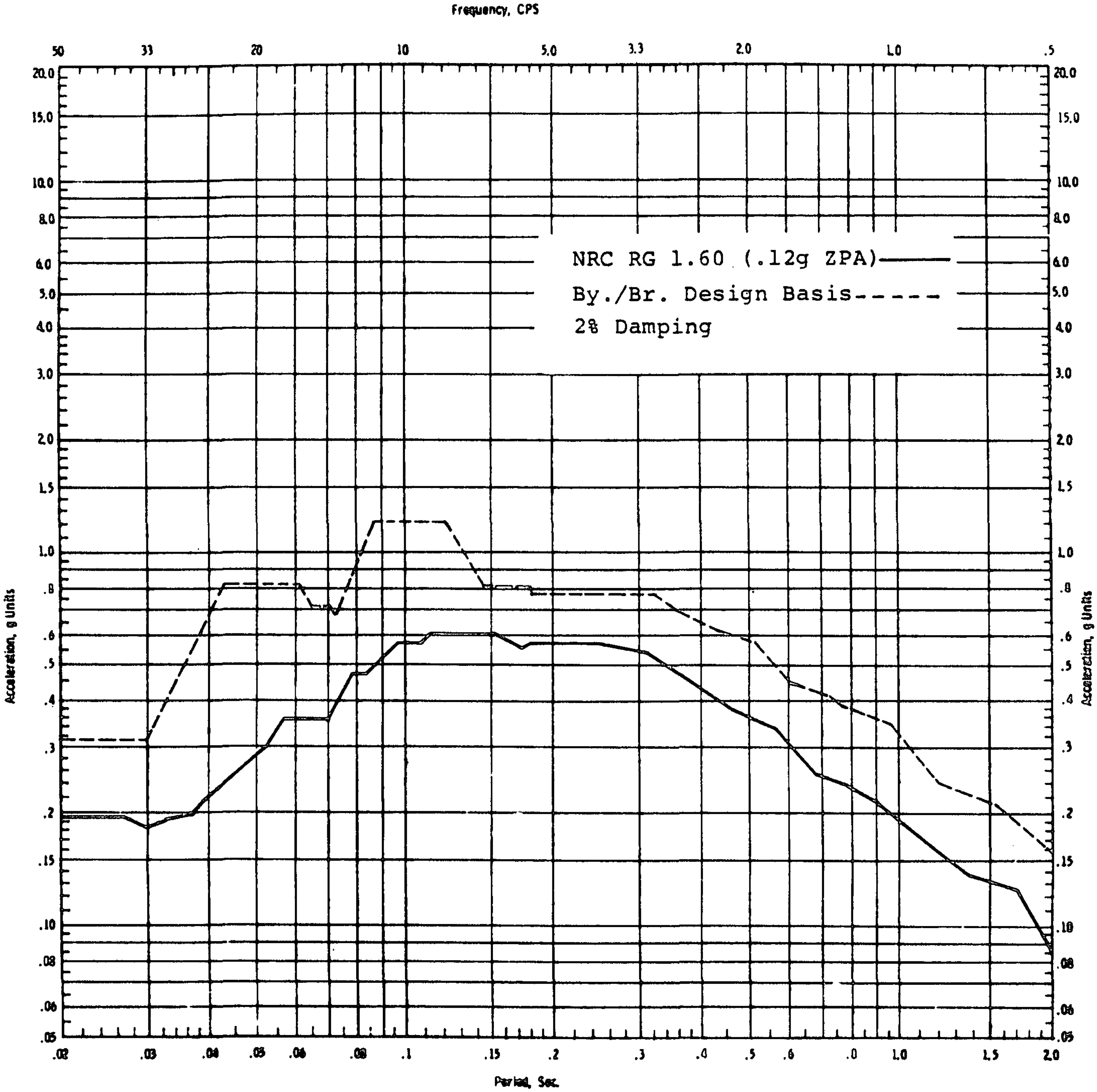

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FIGURE Q130.6-35
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 346'-0"; 364'-0"; 383'-0"; 401'-0"

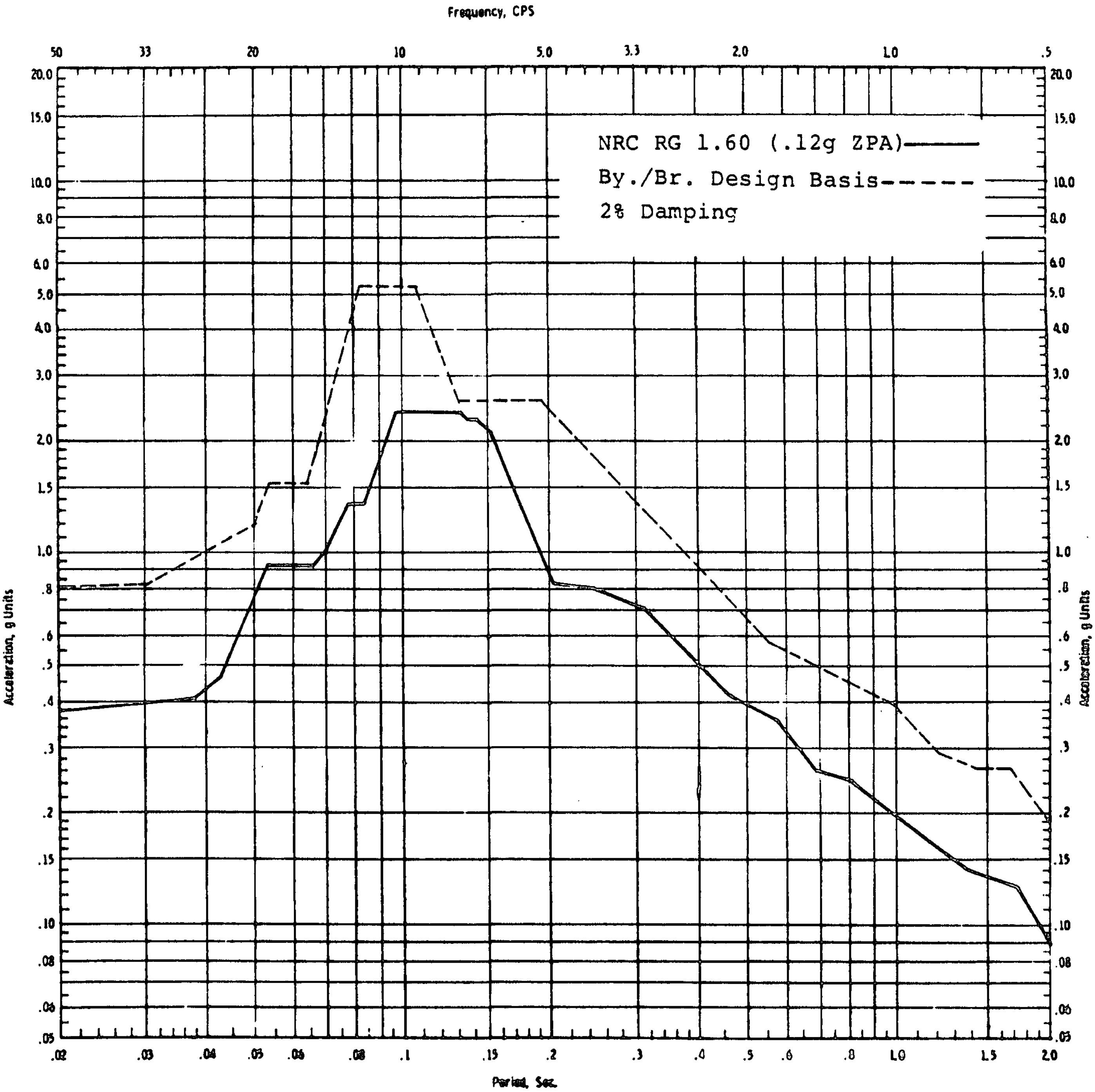

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FIGURE Q130.6-36
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 346'-0"; 364'-0"; 383'-0"; 401'-0"

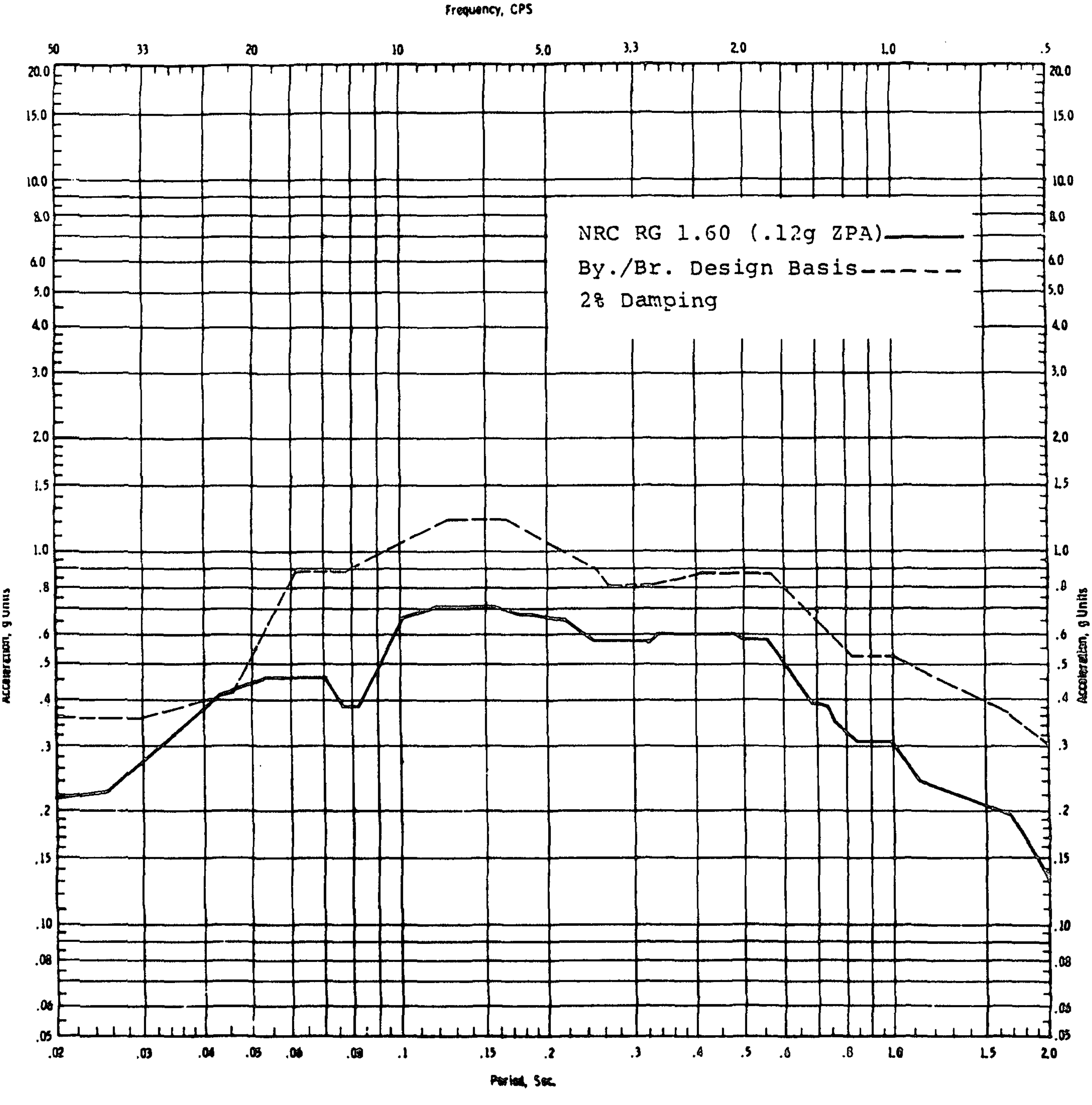

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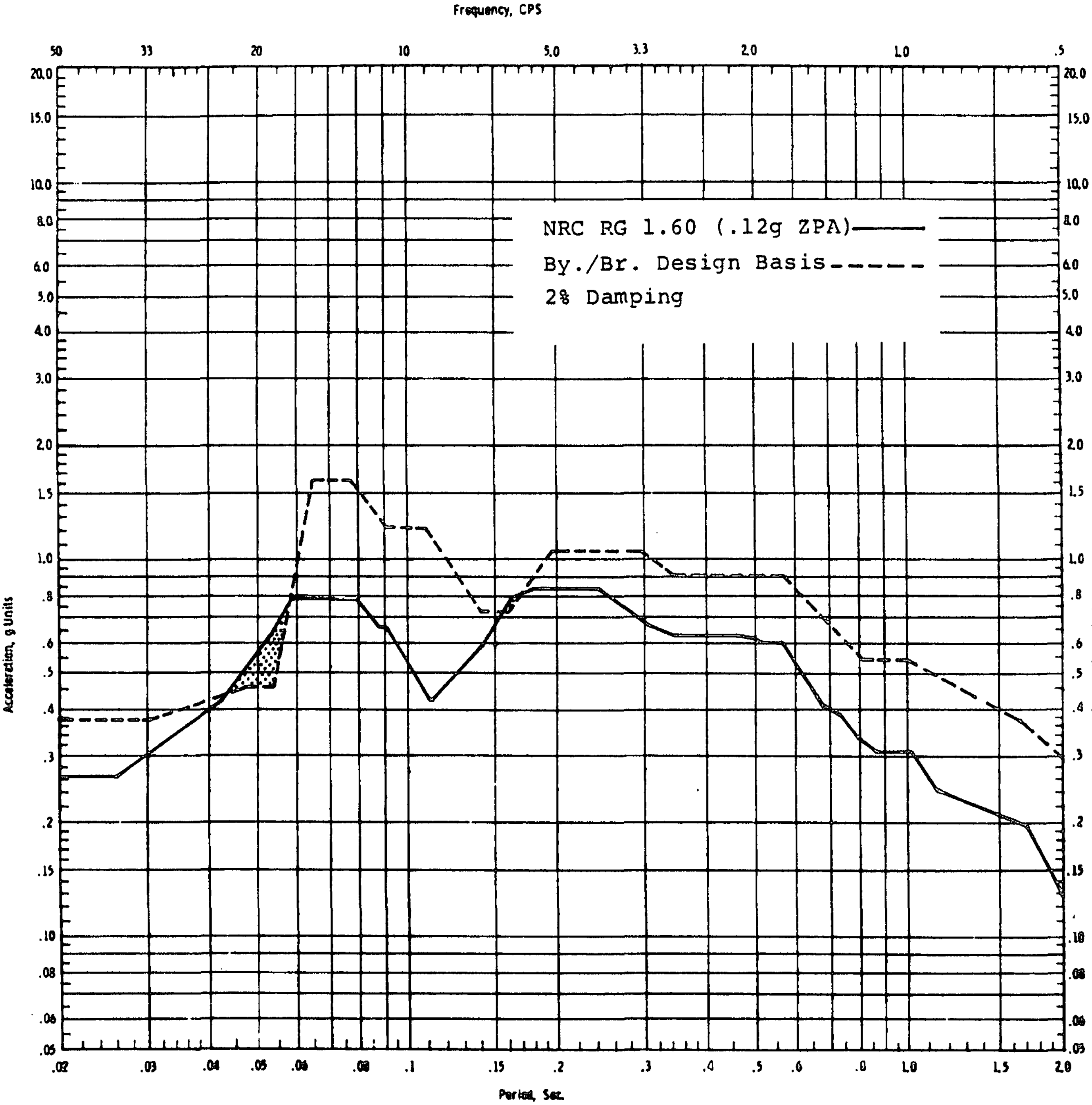
FIGURE Q130.6-37
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS
LOCATION: AUXILIARY BUILDING
ELEVATION: 401'-0"



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6-38
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, EW
LOCATION: AUXILIARY BUILDING
ELEVATION: 401'-0"

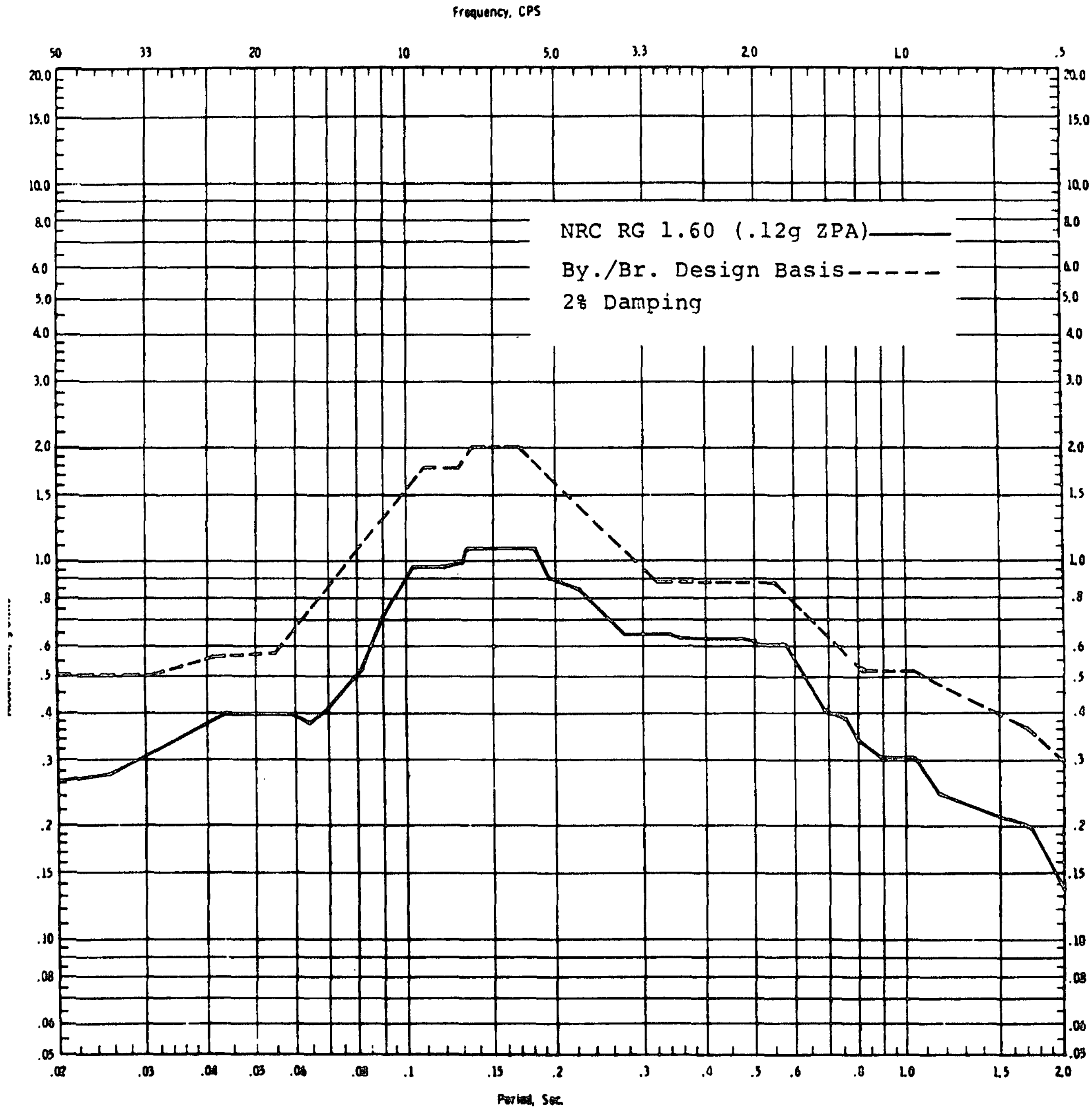

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FIGURE Q130.6-39
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS
LOCATION: AUXILIARY BLDG., TURBINE BLDG., HEATER BAY—ELEVATION: 426'-0"

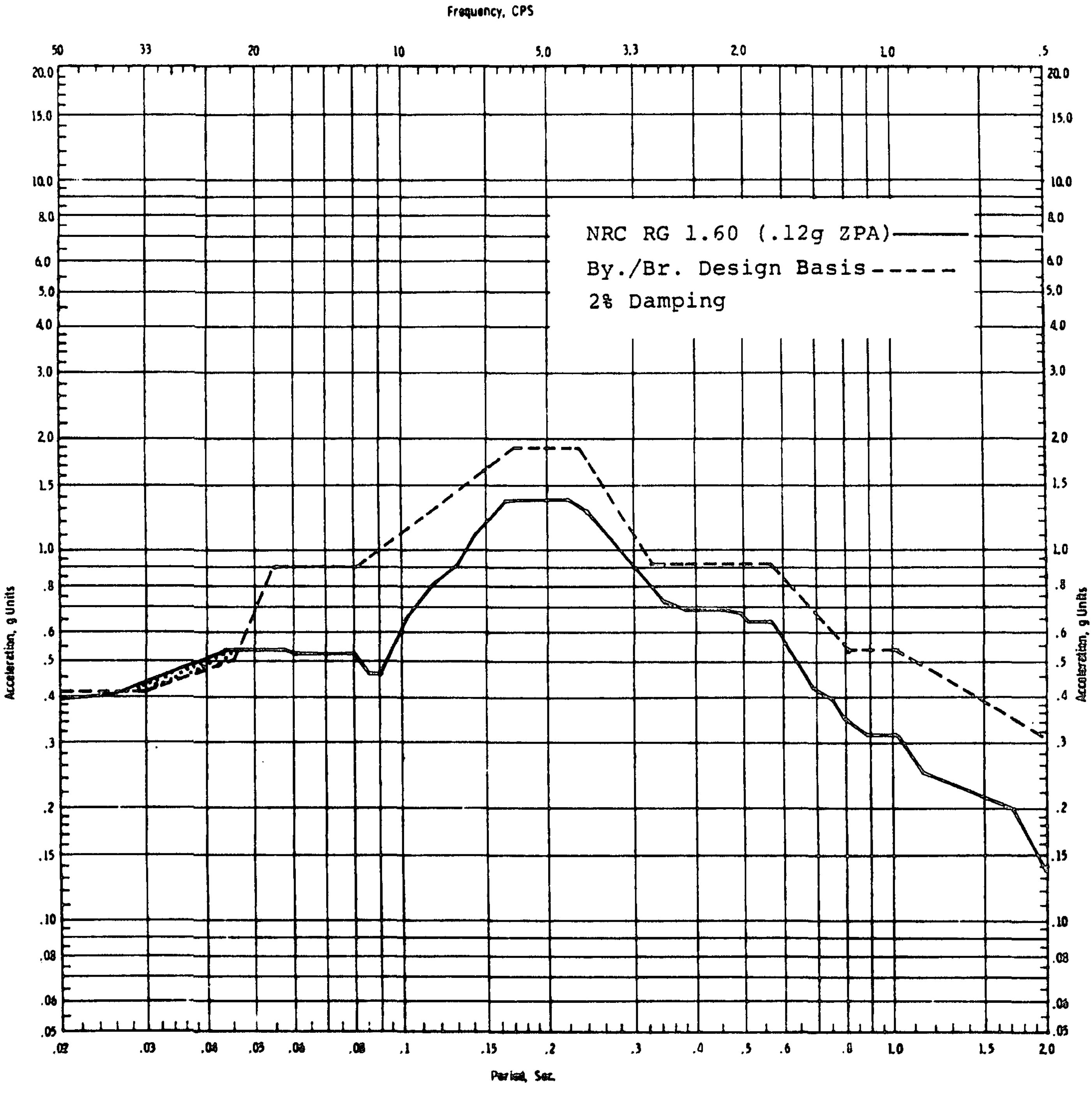

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FIGURE Q130.6-40
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, EW
LOCATION: AUXILIARY BLDG., TURBINE BLDG., HEATER BAY—ELEVATION: 426'-0"

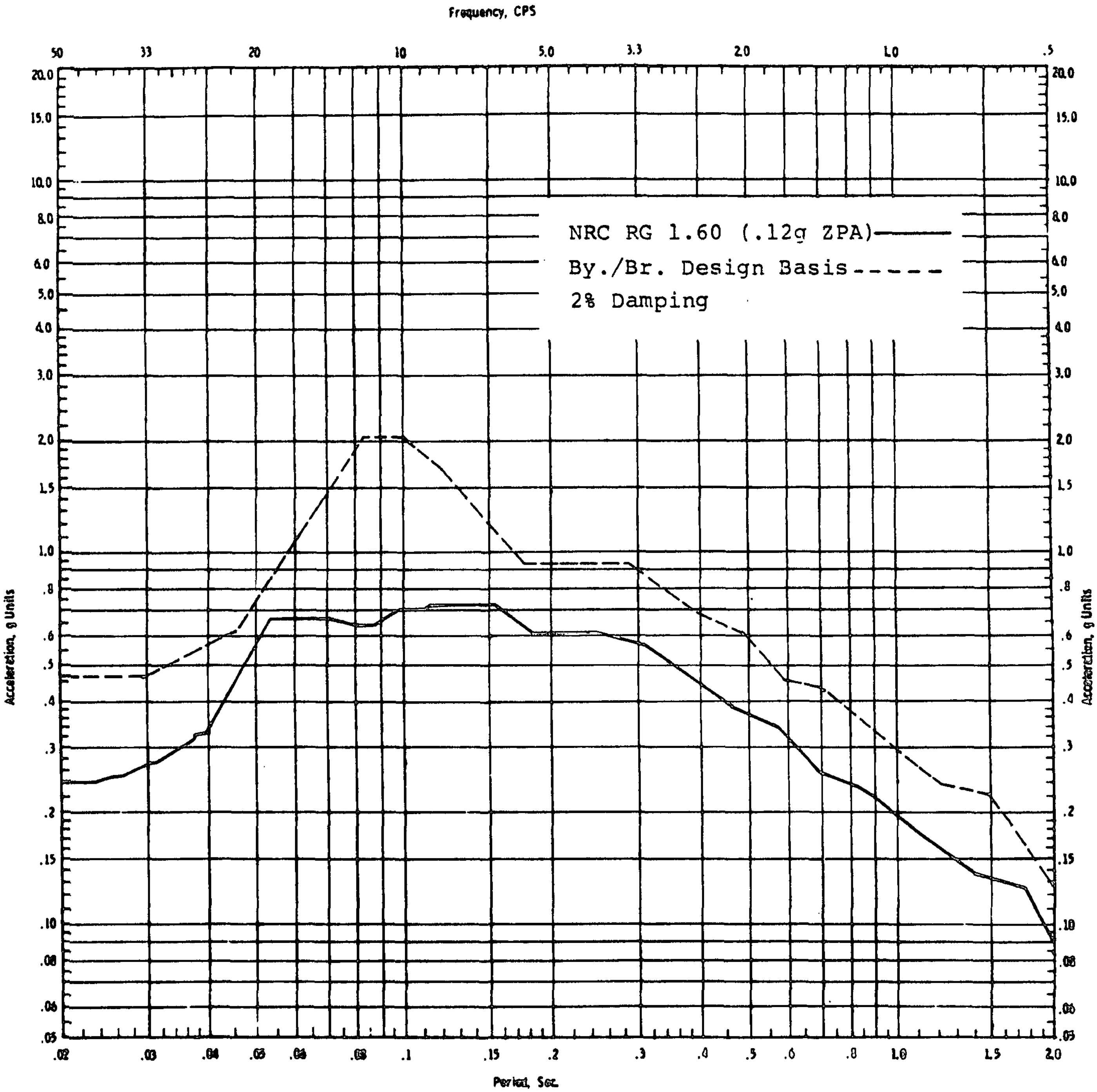

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FIGURE Q130.6-41
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 426'-0"; 439'-0"; 451'-0"

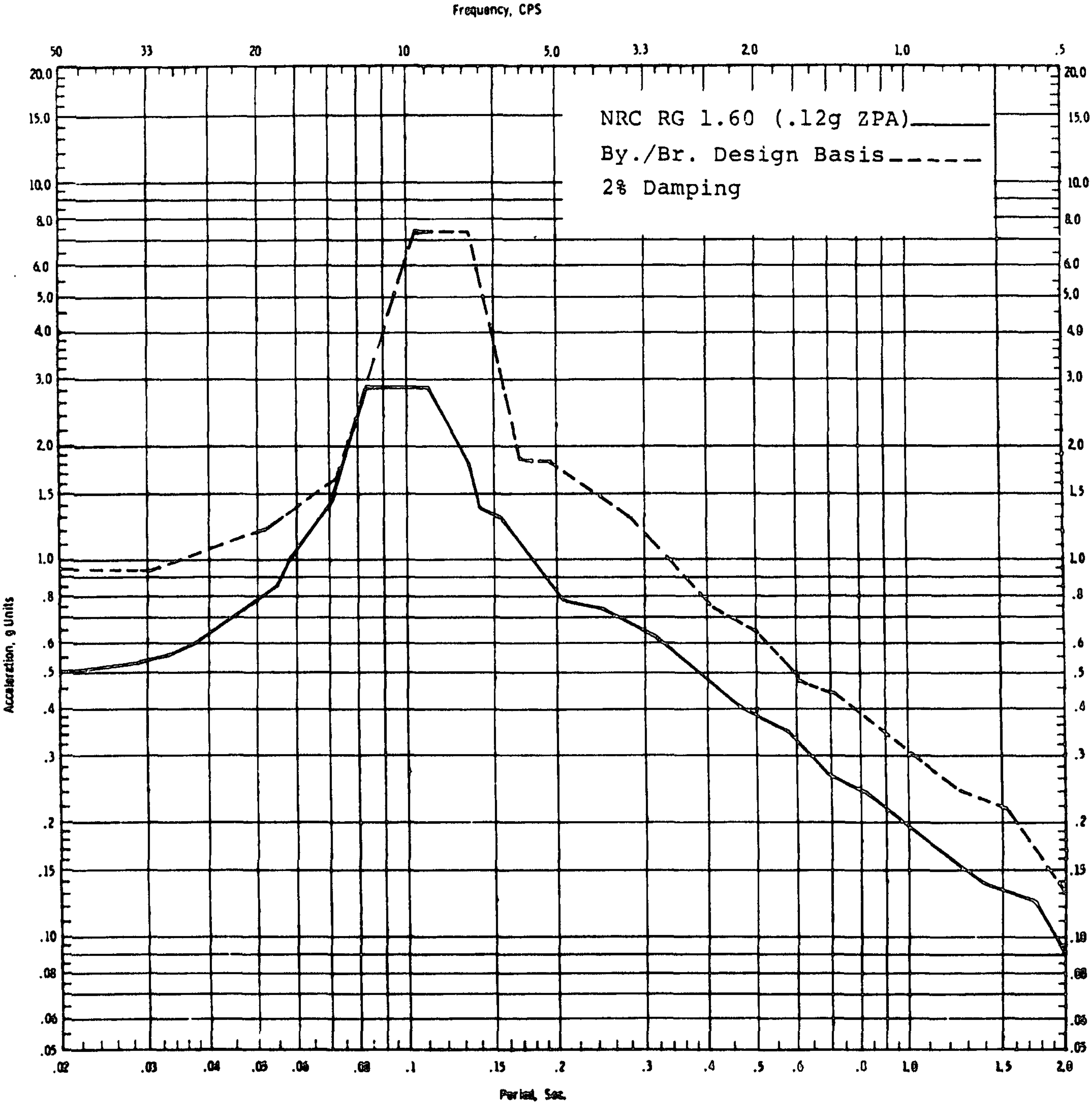

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FIGURE Q130.6-42
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 426'-0"; 439'-0"; 451'-0"

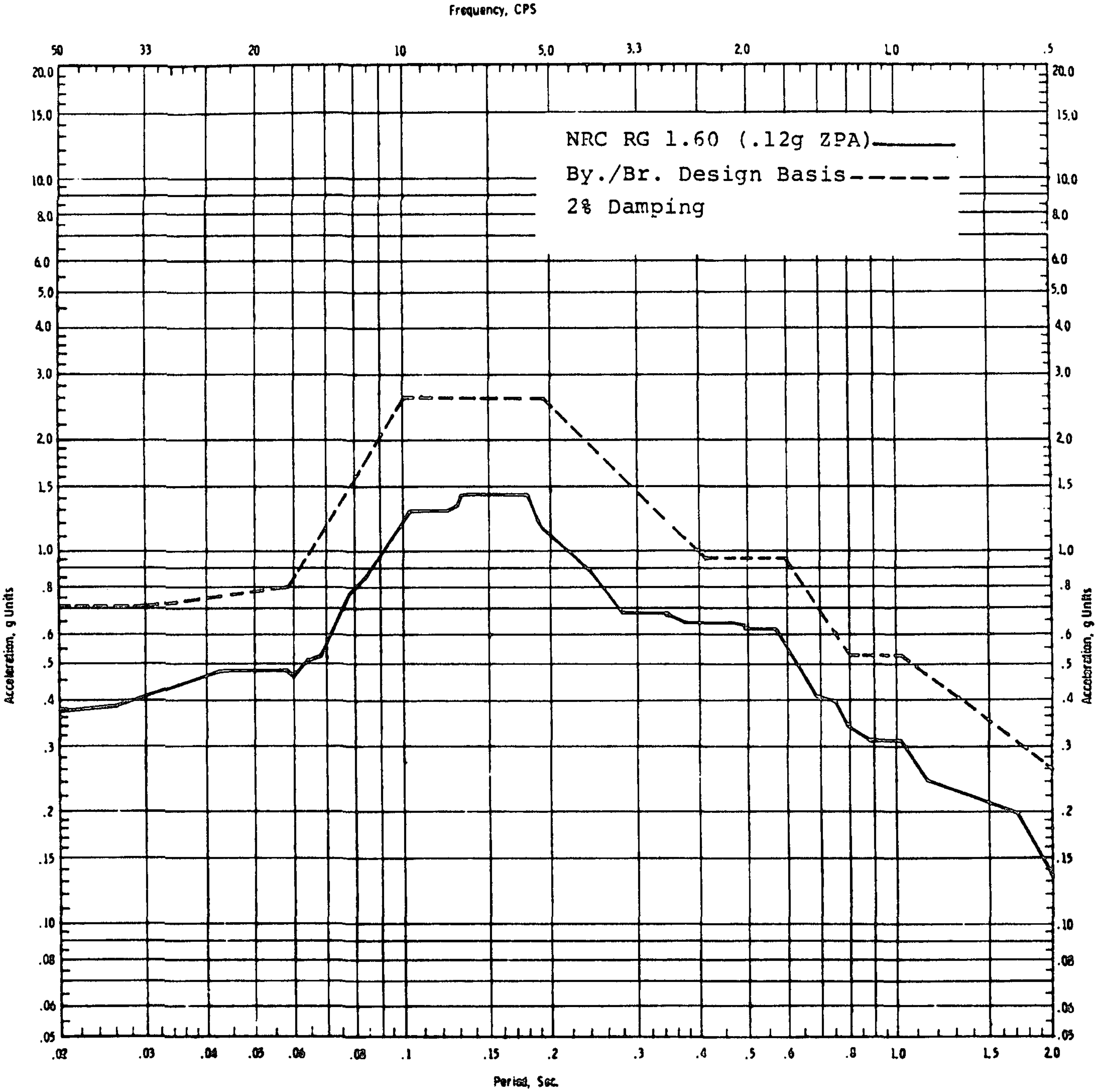

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FIGURE Q130.6-43
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS
LOCATION: AUXILIARY BLDG., TURBINE BLDG., HEATER BAY—ELEVATION: 451'-0"

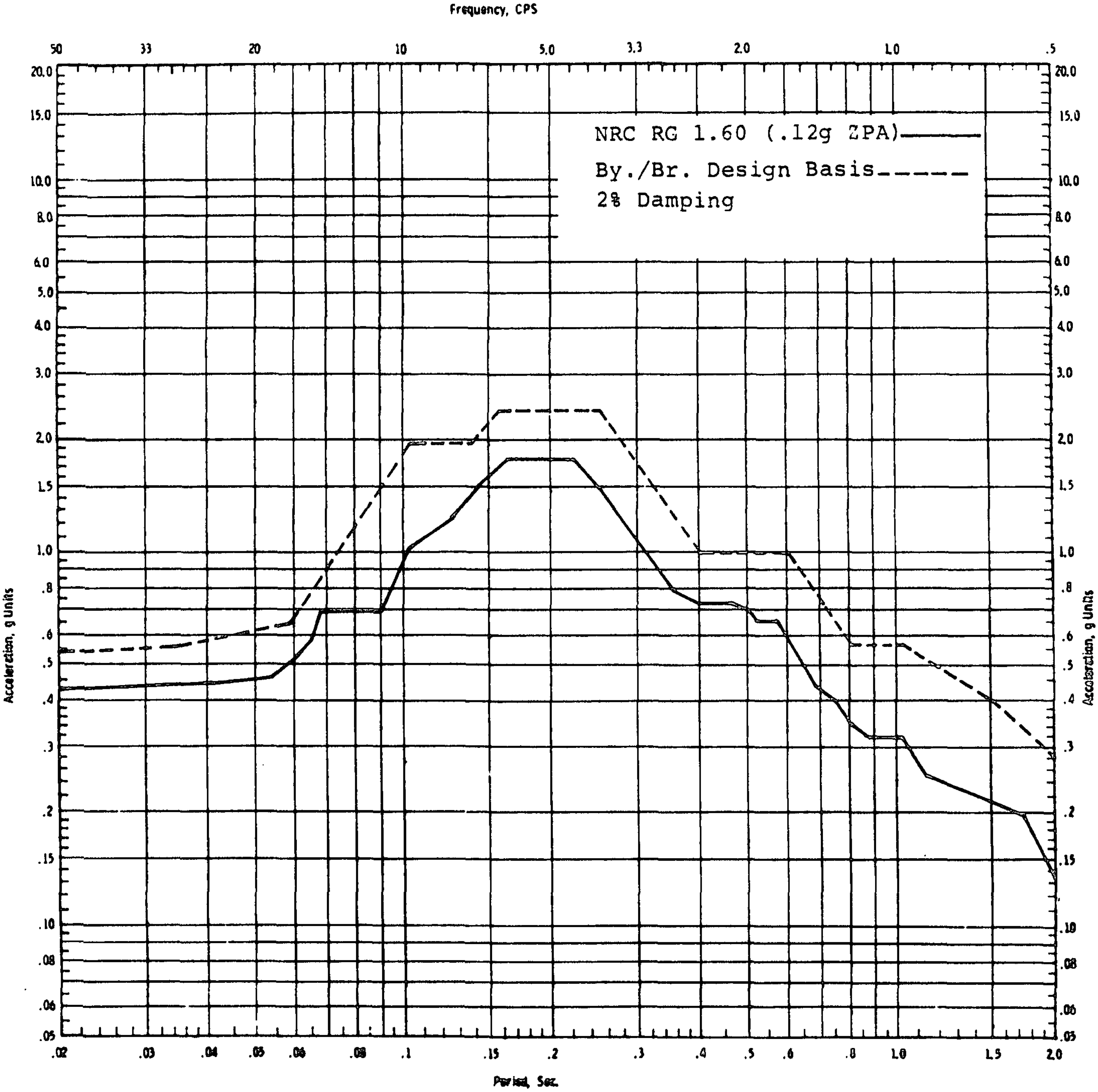

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FIGURE Q130.6-44
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, EW
LOCATION: AUXILIARY BLDG., TURBINE BLDG.,
HEATER BAY—ELEVATION: 451'-0"

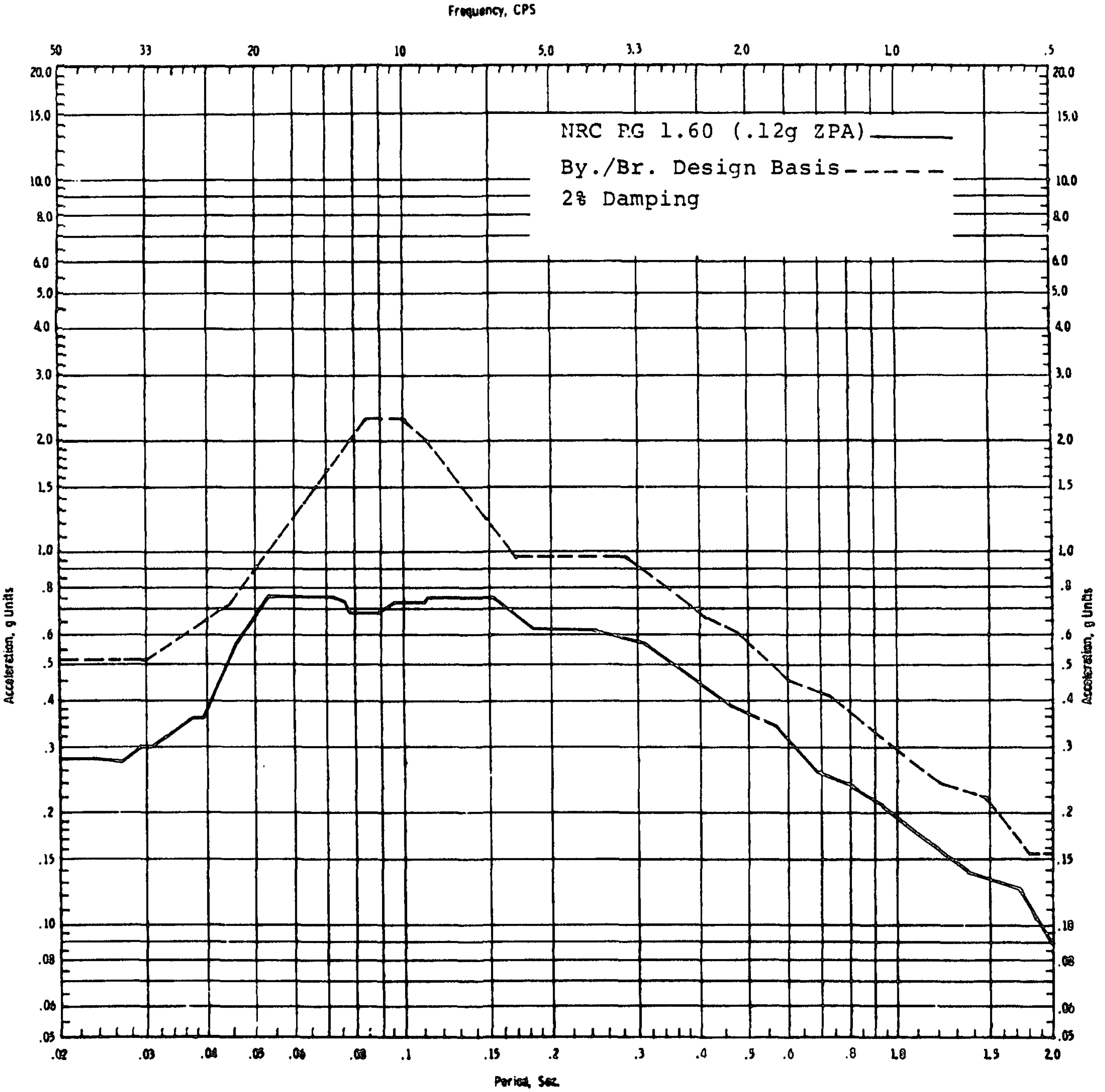

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FIGURE Q130.6-45
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 467'-0"; 473'-0"; 477'-0"; 485'-0"

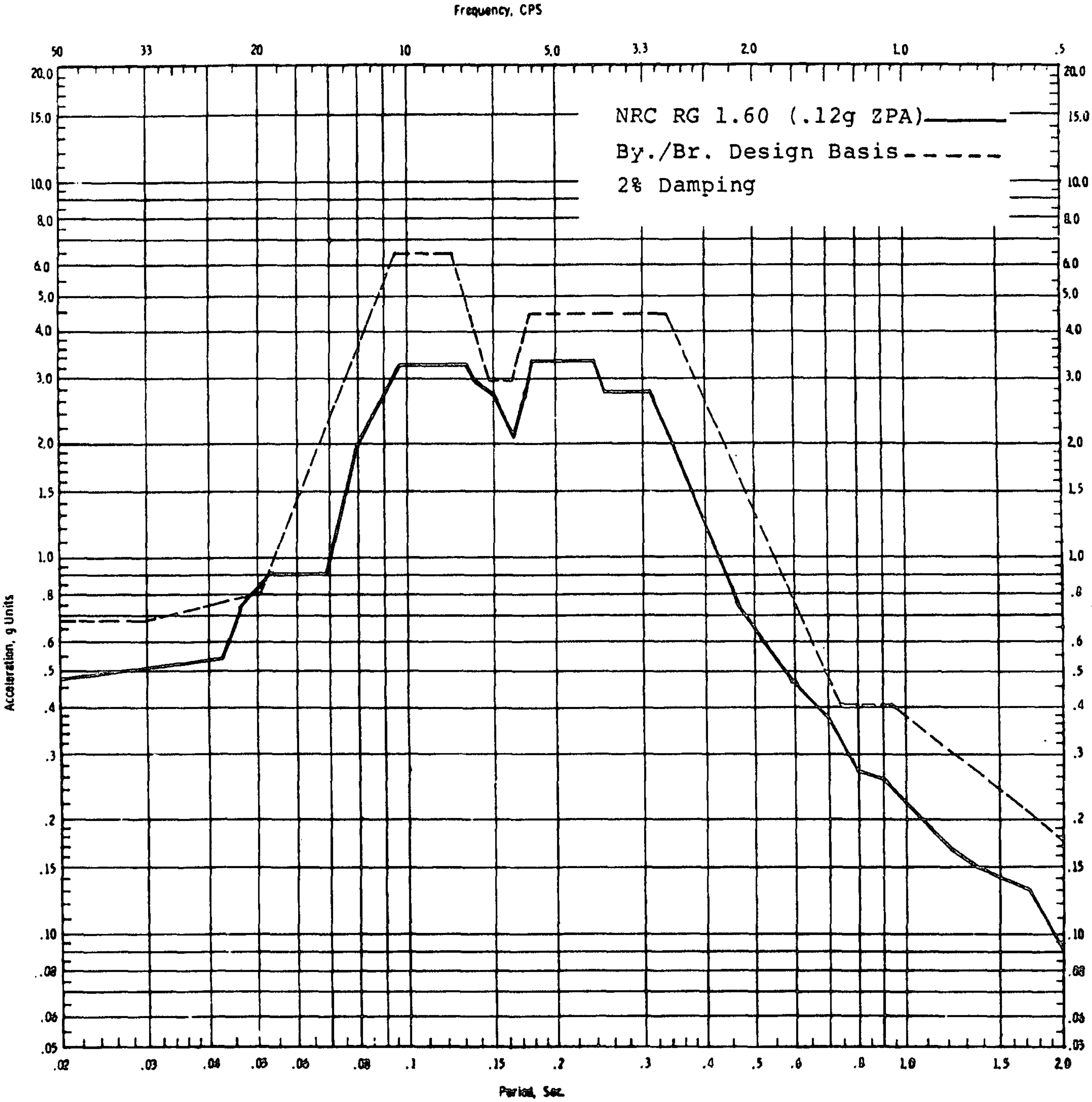

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FIGURE Q130.6-46
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 467'-0"; 477'-0"

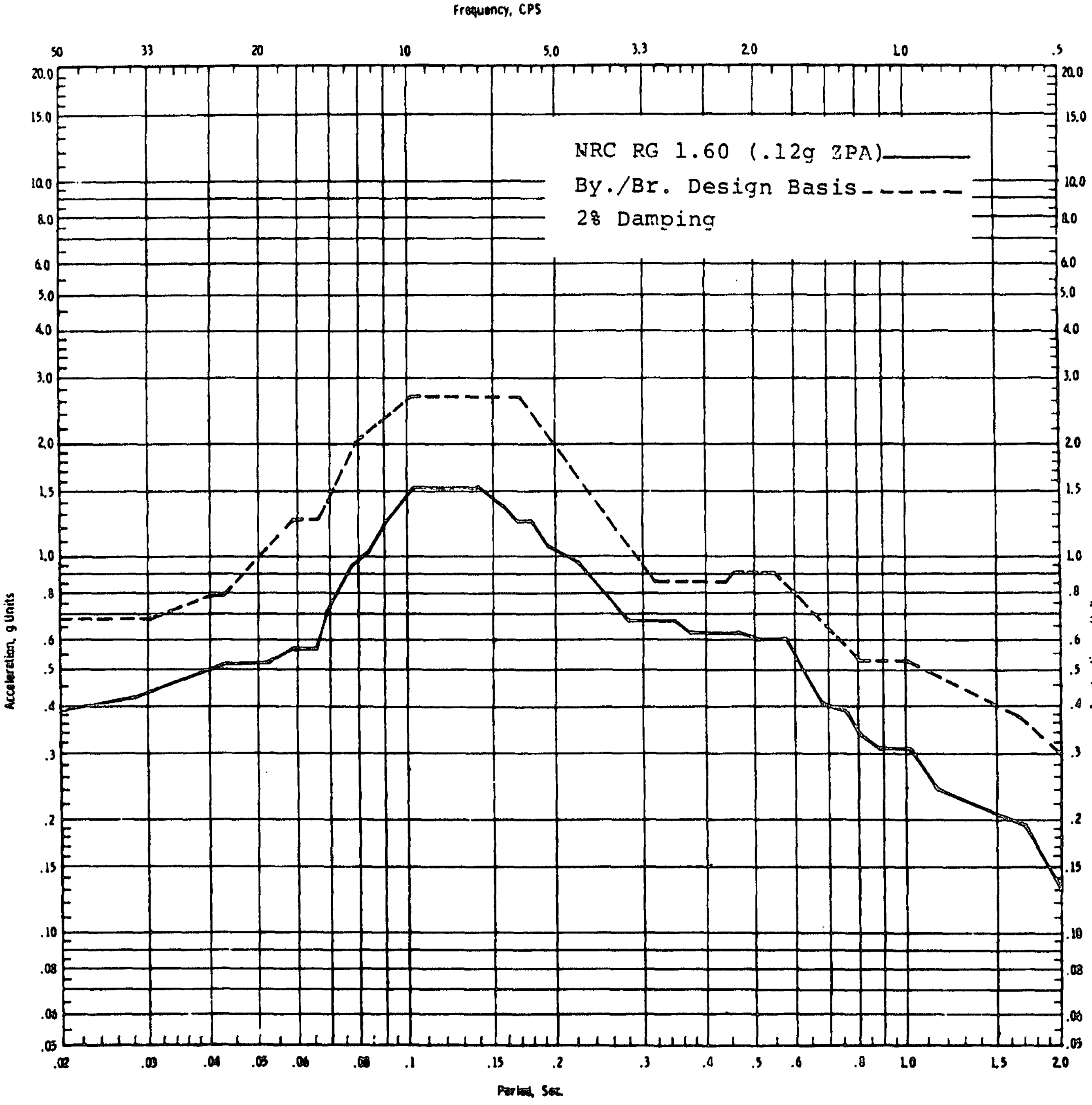

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FIGURE Q130.6-47
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS
LOCATION: AUXILIARY BUILDING
ELEVATION: 477'-0"

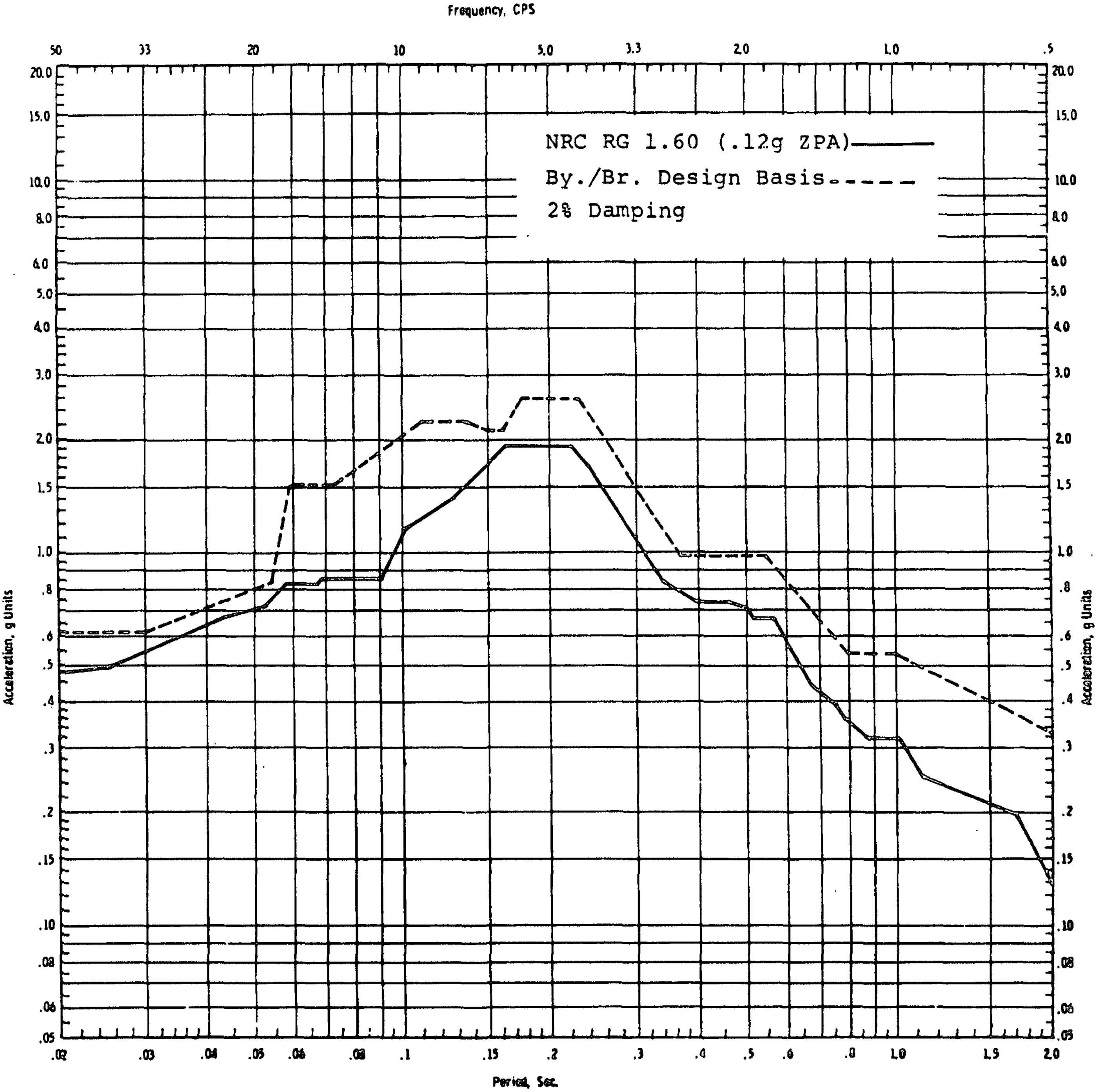

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FIGURE Q130.6-48
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, EW
LOCATION: AUXILIARY BUILDING
ELEVATION: 477'-0"

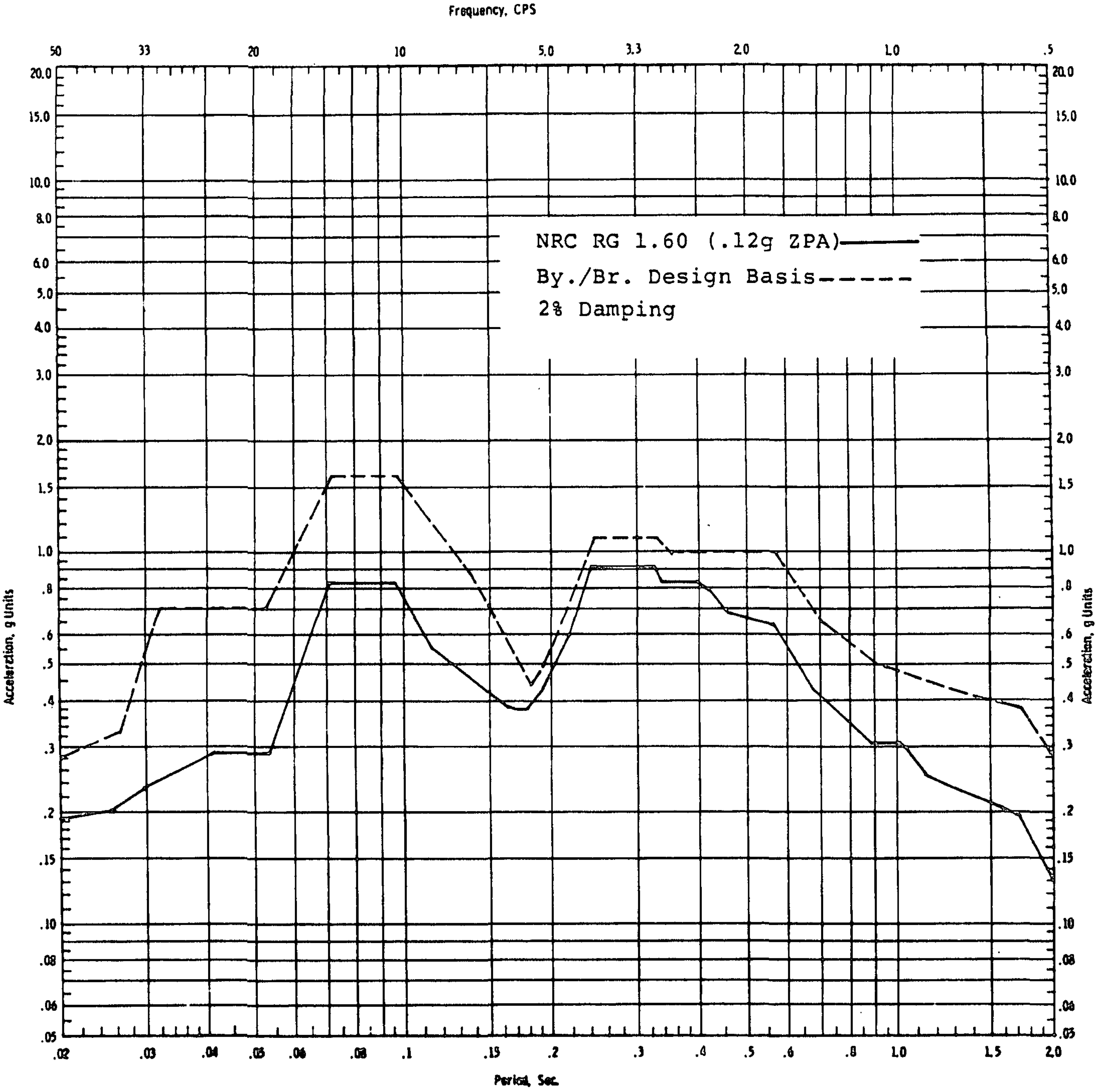

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FIGURE Q130.6-49
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS AND EW
LOCATION: CONTAINMENT BUILDING
ELEVATION: 424'-0"; 436'-0"

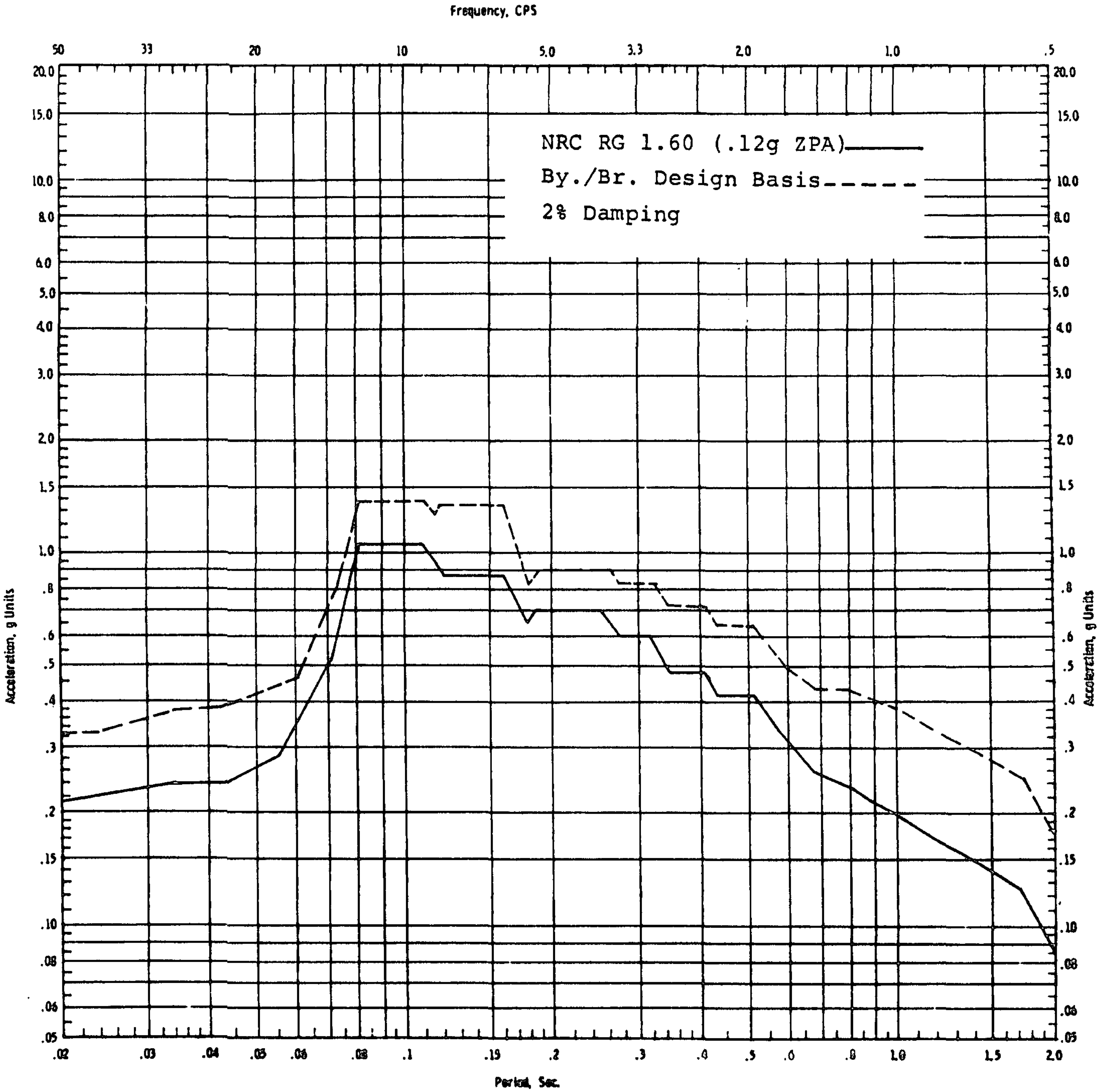

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FIGURE Q130.6-50
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL
LOCATION: CONTAINMENT BUILDING WALL
ELEVATION: 424'-0"; 436'-0"

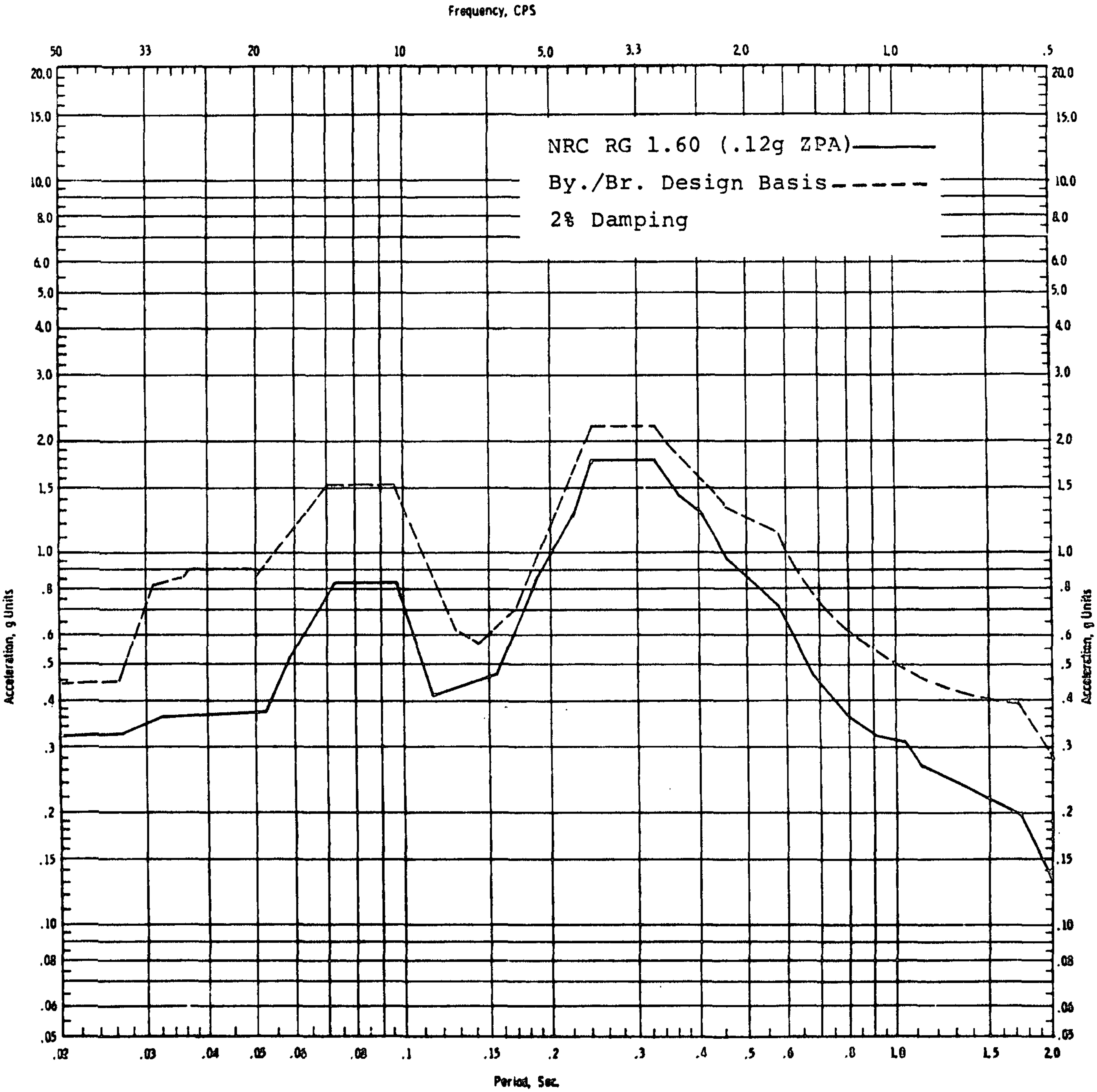

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FIGURE Q130.6-51
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS AND EW
LOCATION: CONTAINMENT BUILDING
ELEVATION: 496'-0"

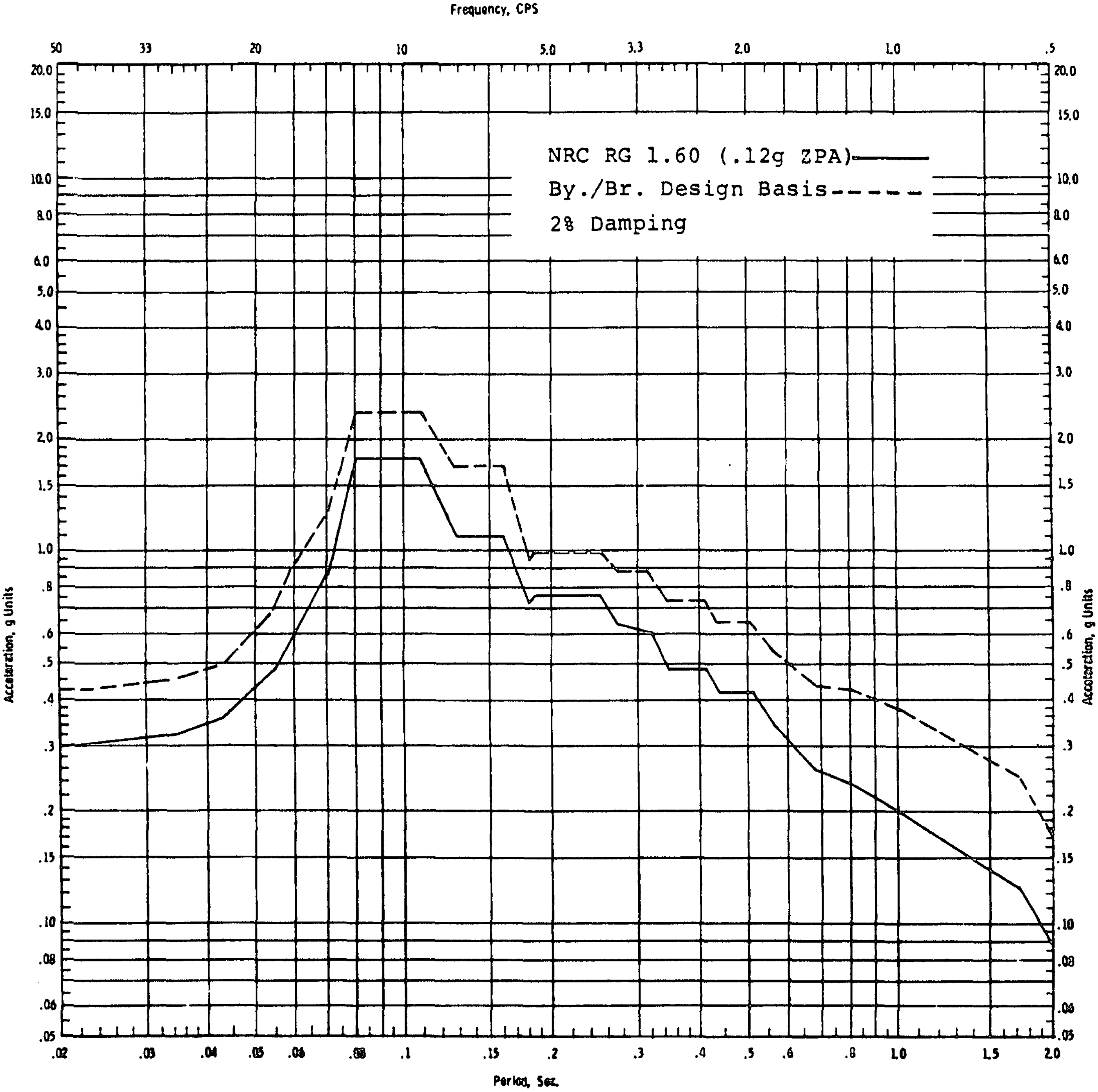

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FIGURE Q130.6-52
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL
LOCATION: CONTAINMENT BUILDING WALL
ELEVATION: 496'-0"

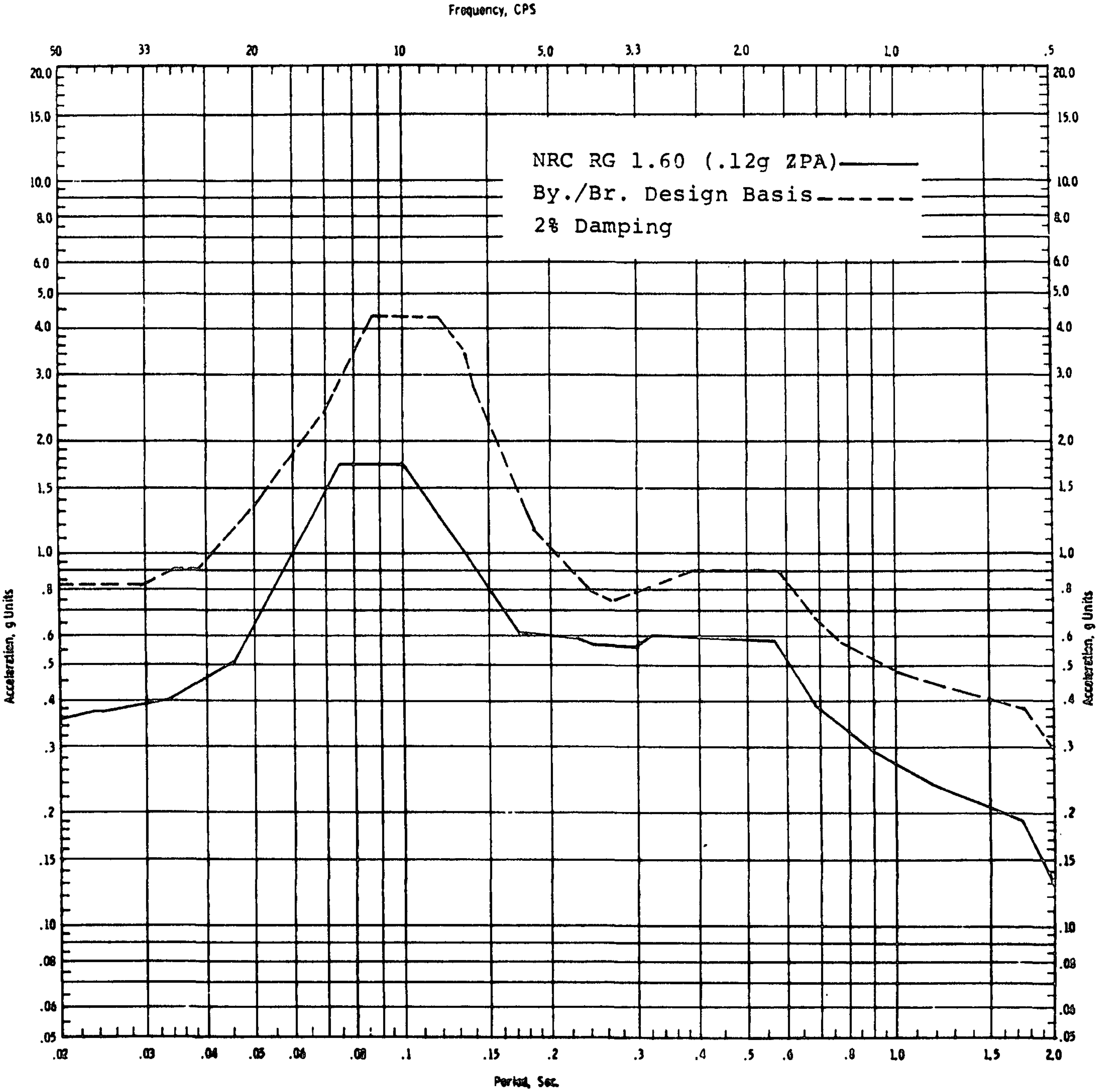

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FIGURE Q130.6-53
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS
LOCATION: CONTAINMENT INNER STRUCTURE
ELEVATION: 426'-0"

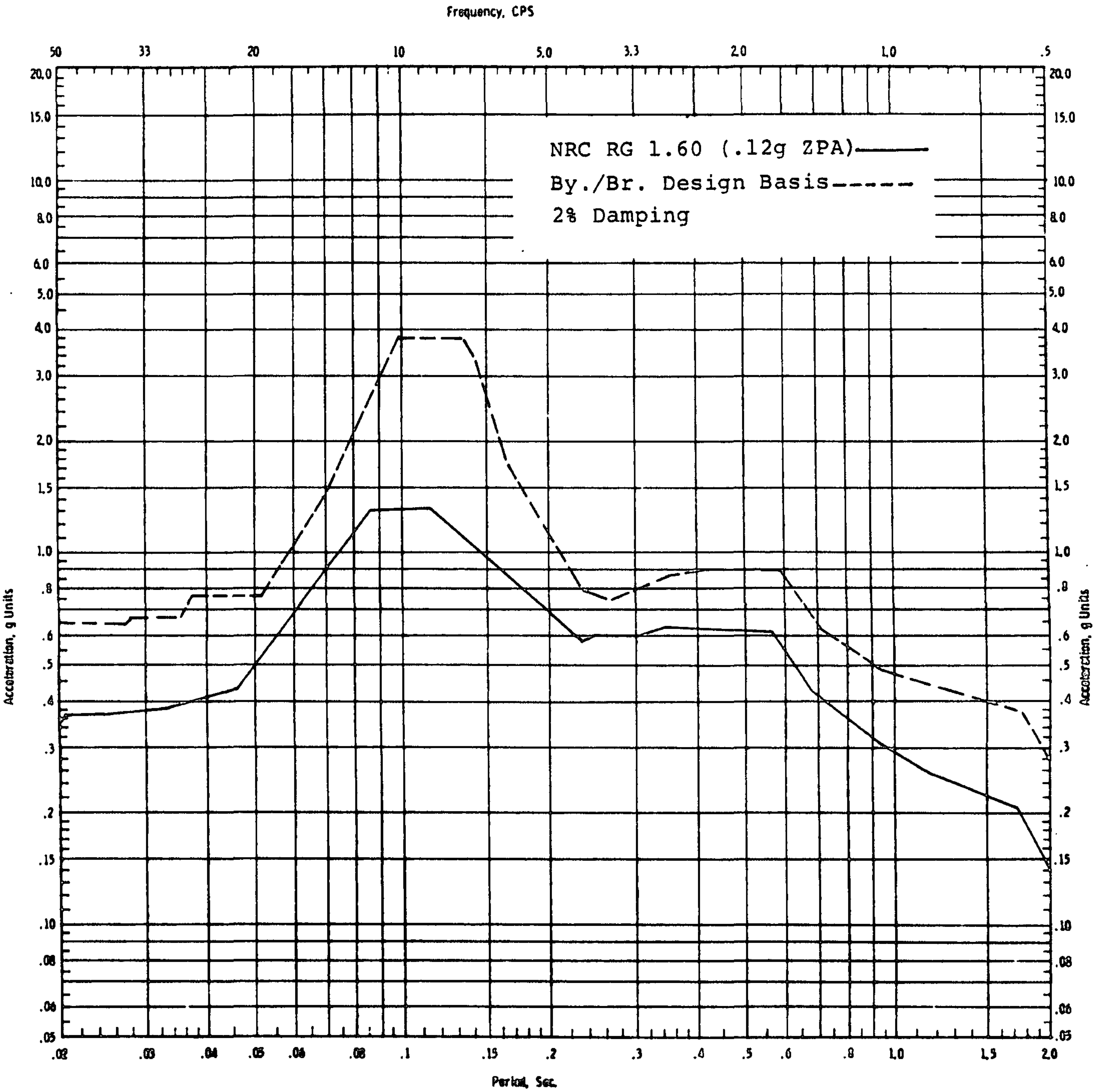

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FIGURE Q130.6-54
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, EW
LOCATION: CONTAINMENT INNER STRUCTURE
ELEVATION: 426'-0"

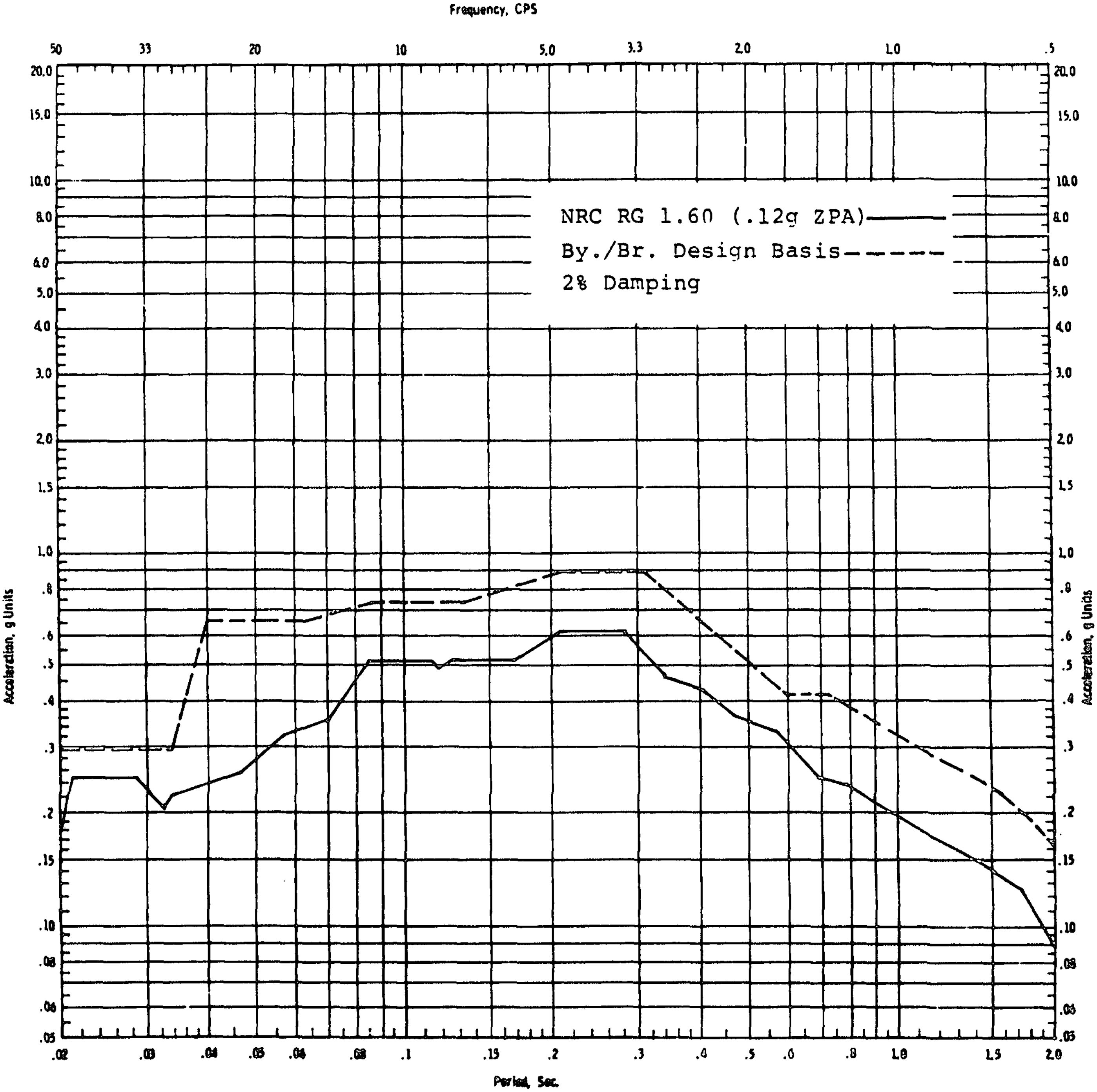

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FIGURE Q130.6-55
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL
LOCATION: CONTAINMENT INNER STRUCTURE WALL
ELEVATION: 412'-0"; 426'-0"

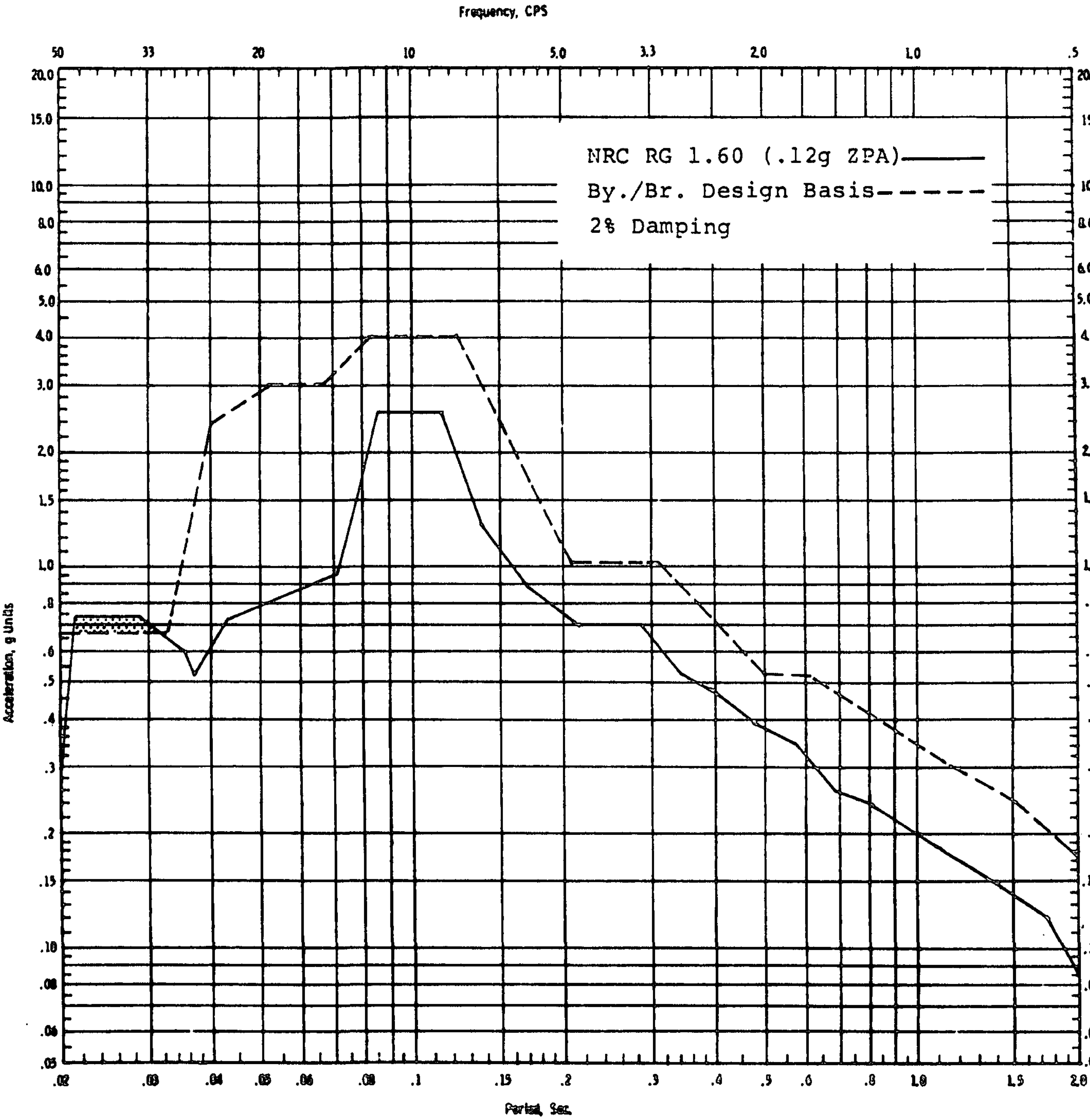

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FIGURE Q130.6-56
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, SLAB
LOCATION: CONTAINMENT INNER STRUCTURE SLAB
ELEVATION: 390'-0"; 401'-0"; 412'-0"; 426'-0"

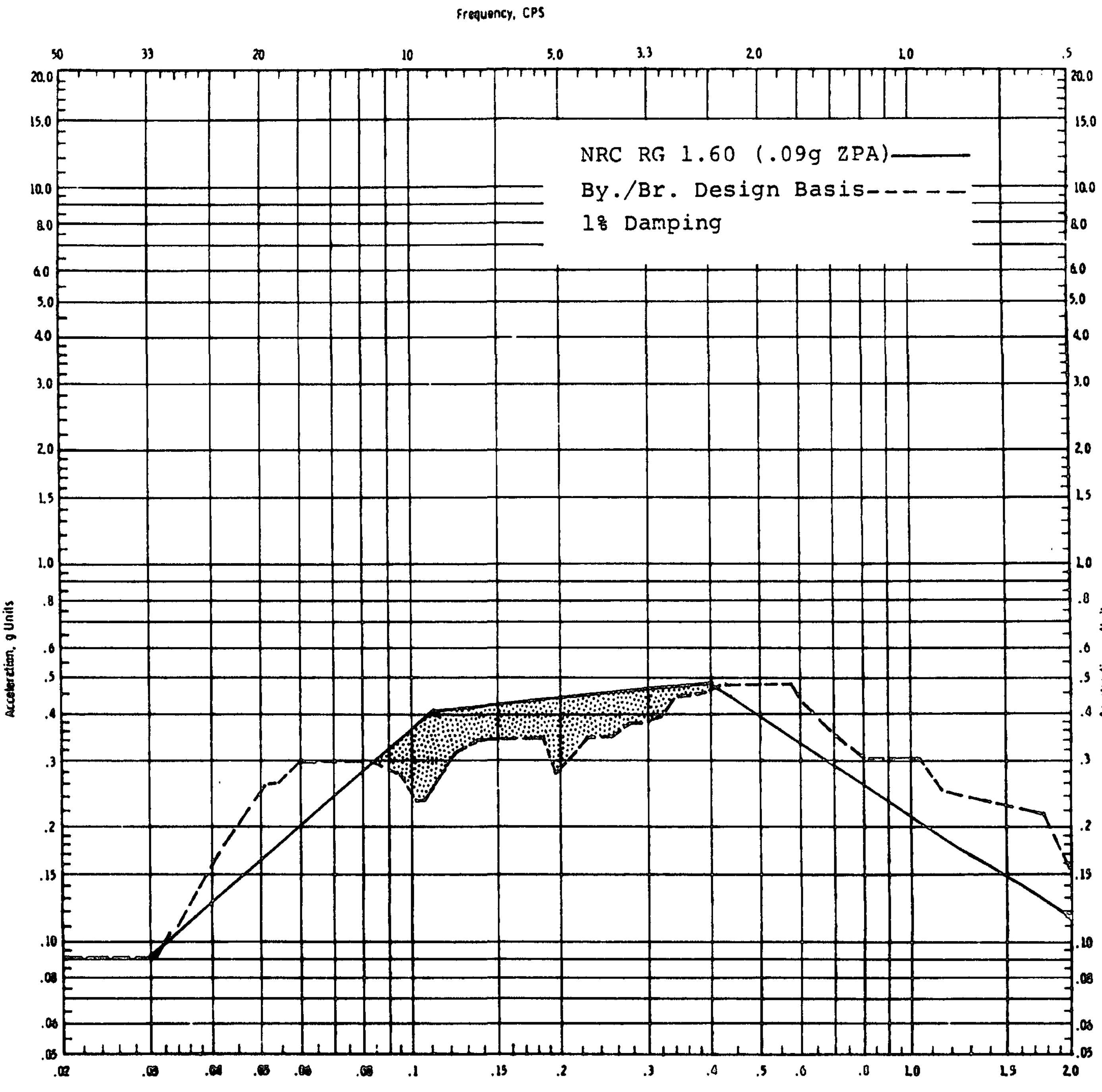

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FIGURE Q130.6-57
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, EW AND NS
LOCATION: AUXILIARY AND CONTAINMENT BLDG.
ELEVATION: 330'-0"; 374'-0"

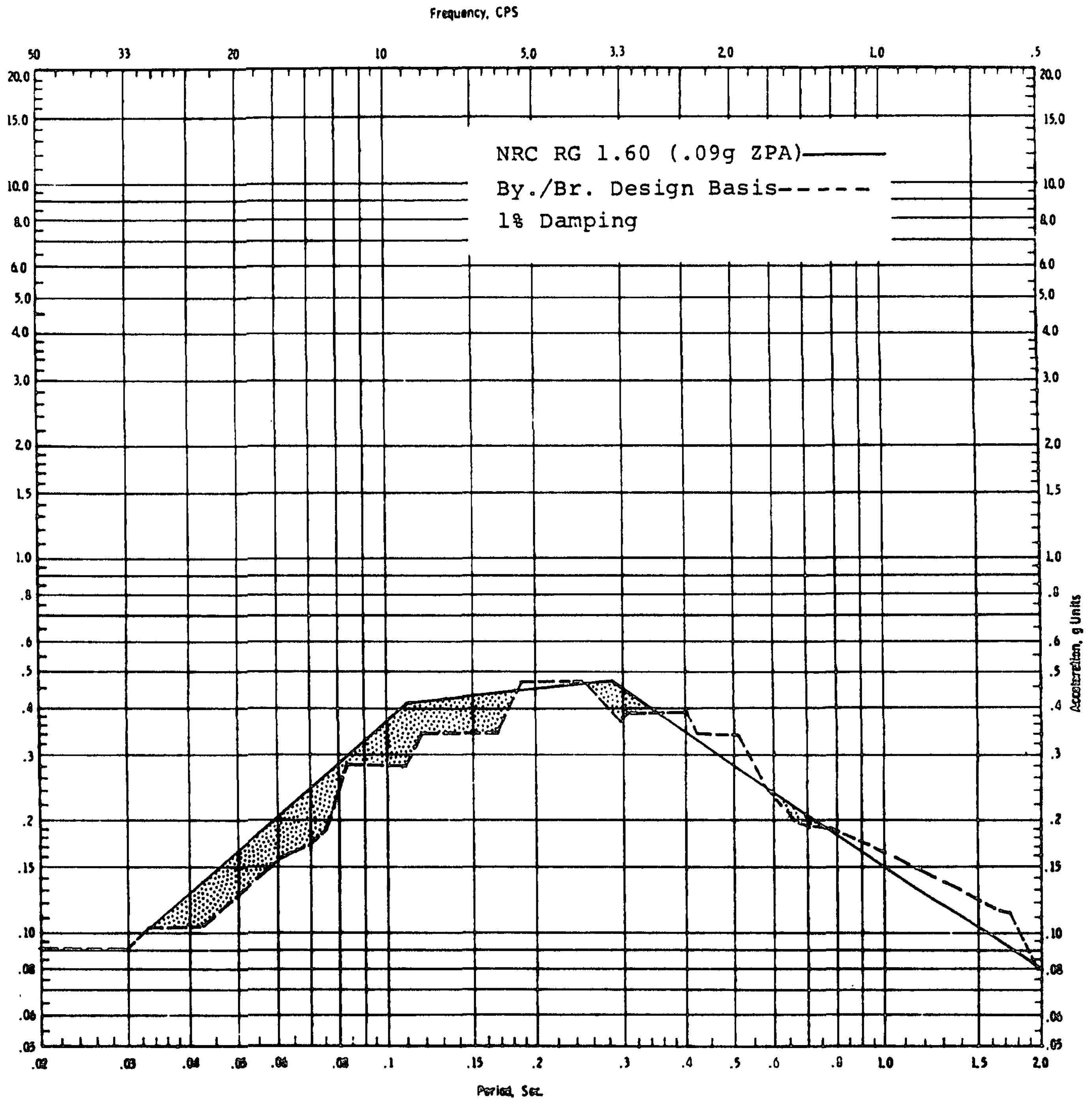

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FIGURE Q130.6-58
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL AND SLAB
LOCATION: AUXILIARY AND CONTAINMENT BLDG.
ELEVATION: 330'-0"; 374'-0"

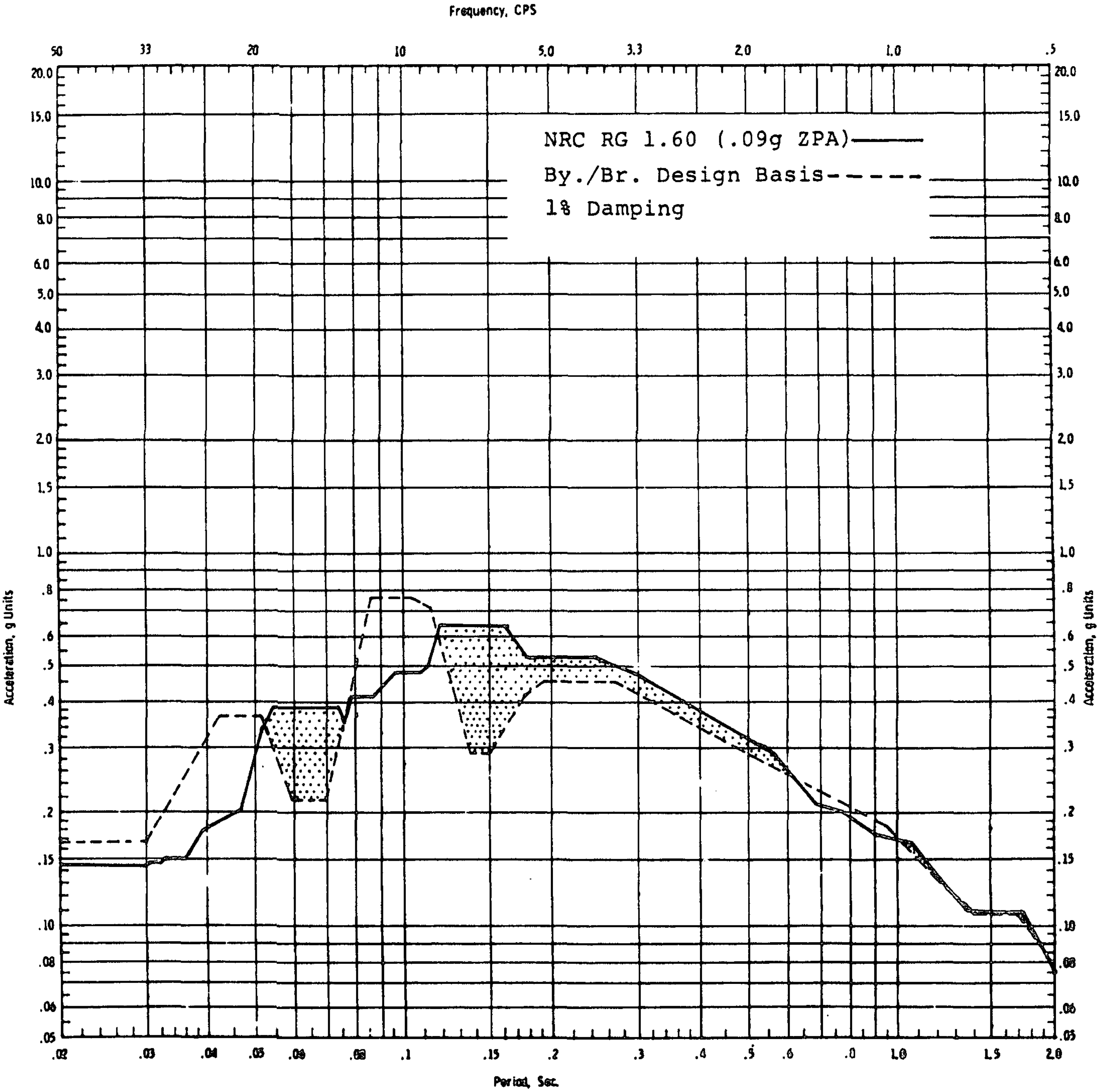

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FIGURE Q130.6-59
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 346'-0"; 364'-0"; 383'-0"; 401'-0"

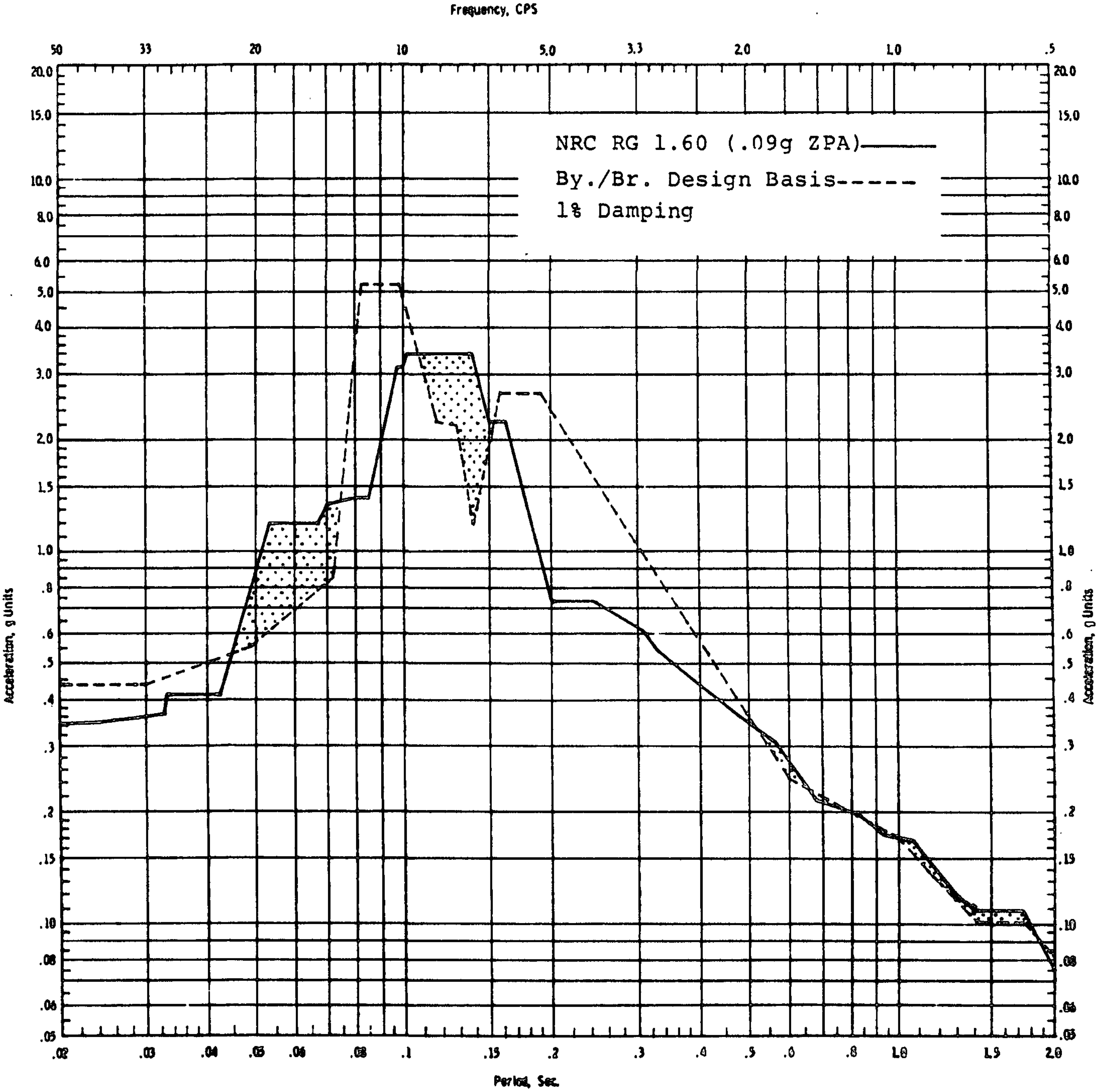

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FIGURE Q130.6-60
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 346'-0"; 364'-0"; 383'-0"; 401'-0"

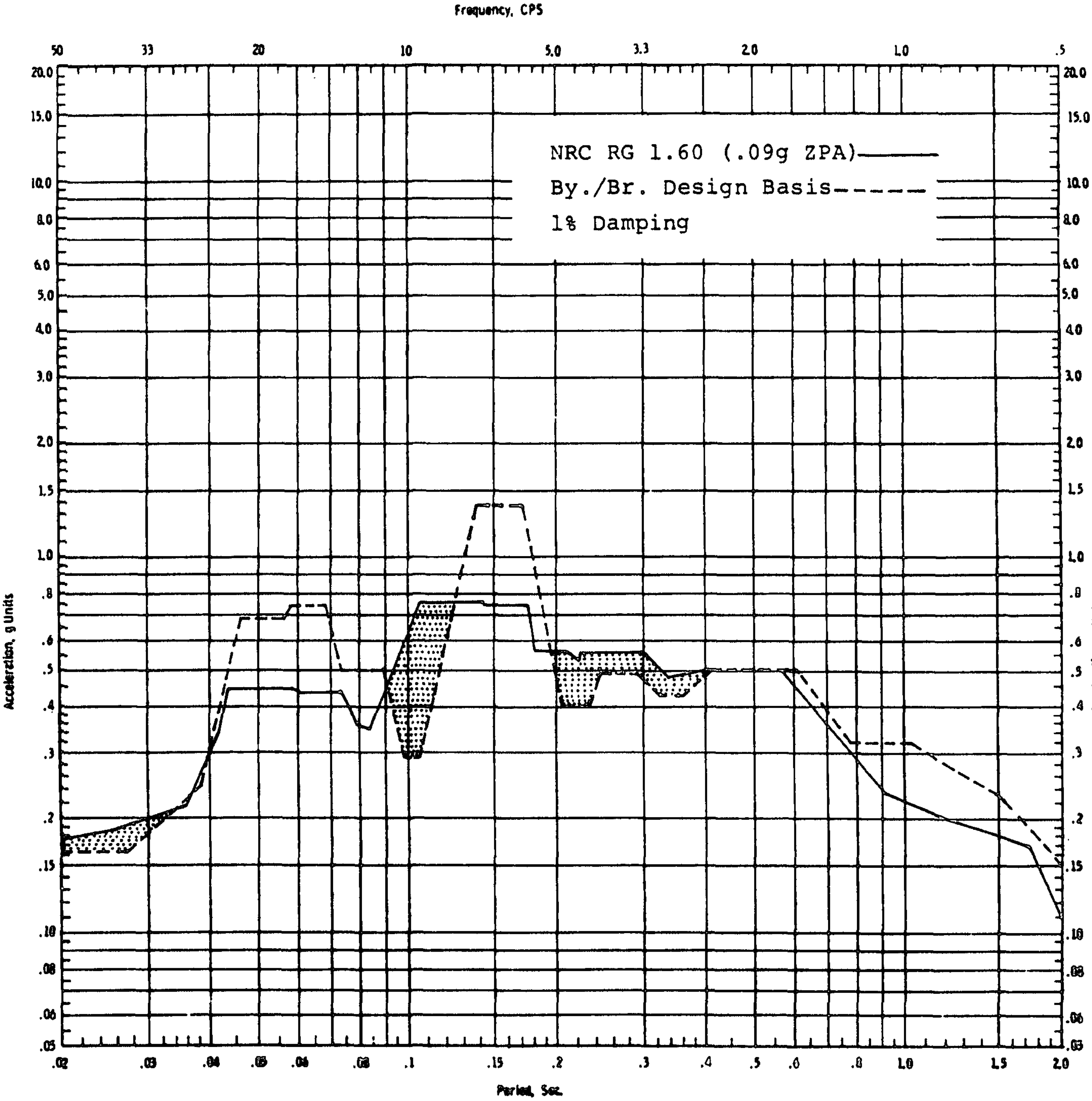

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FIGURE Q130.6-61
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: AUXILIARY BUILDING
ELEVATION: 401'-0"

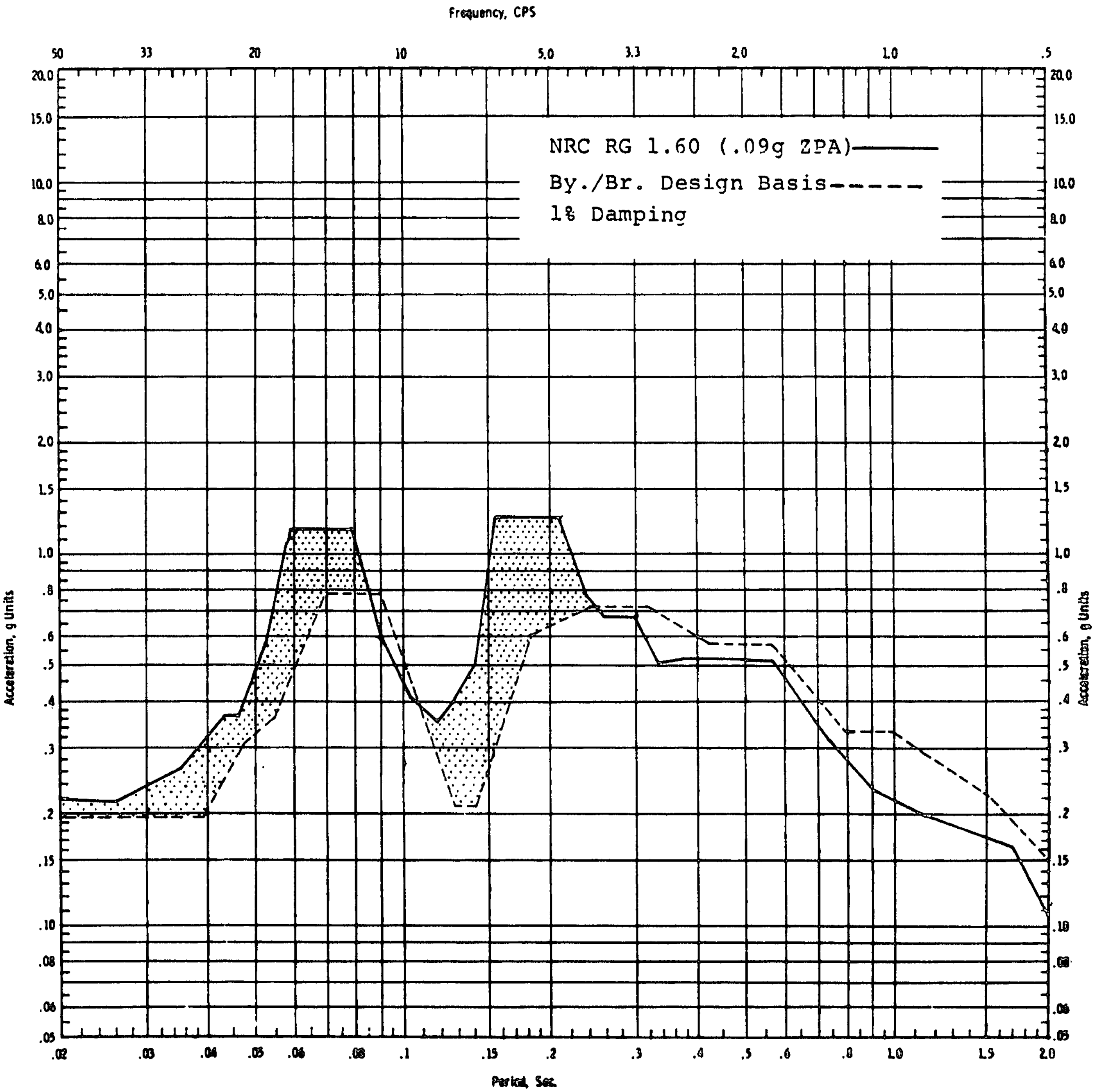

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FIGURE Q130.6-62
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, EW
LOCATION: AUXILIARY BUILDING
ELEVATION: 401'-0"

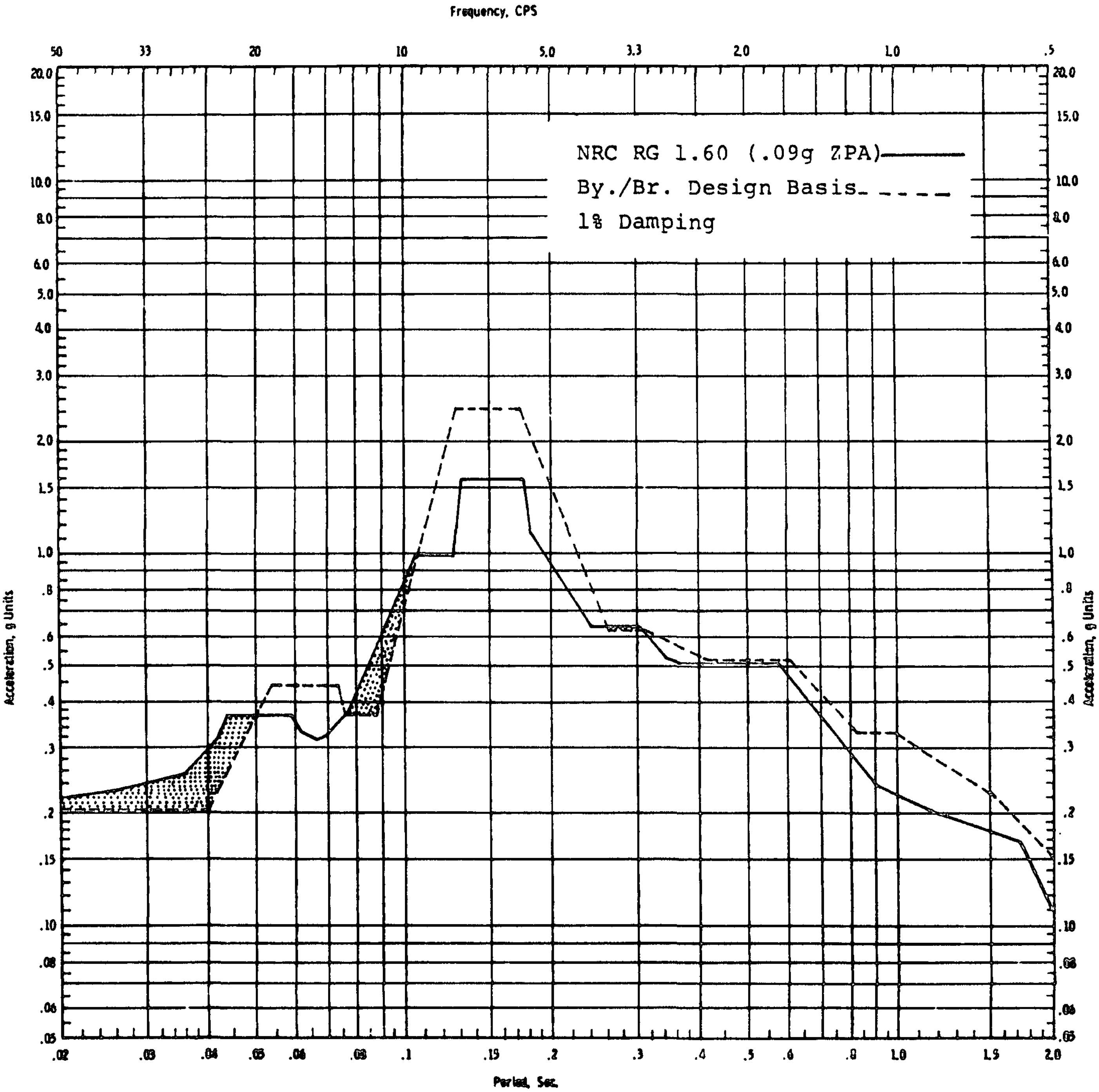

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FIGURE Q130.6-63
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: AUXILIARY BLDG., TURBINE BLDG., HEATER BAY—ELEVATION: 426'-0"

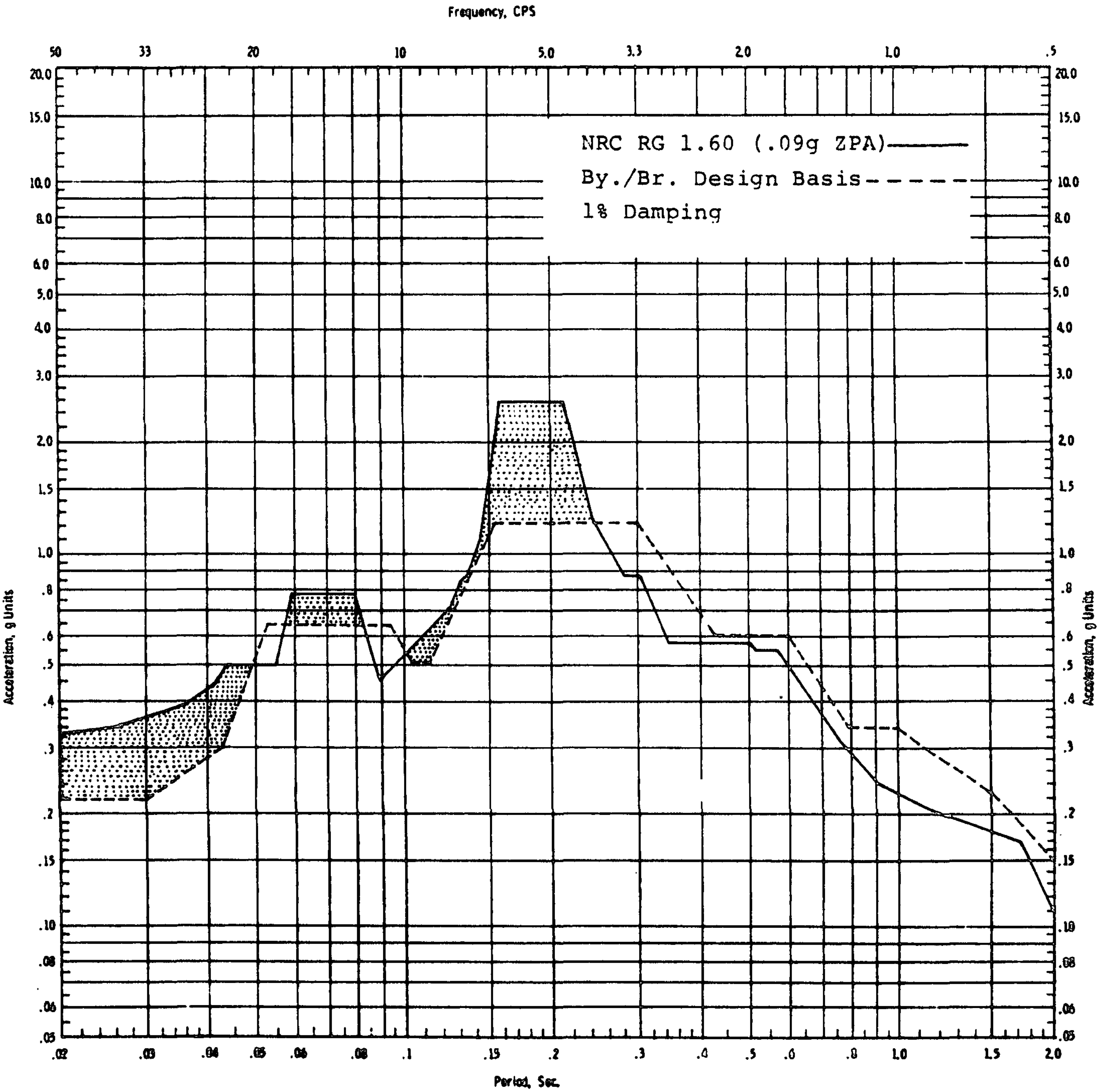

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FIGURE Q130.6-64
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, EW
LOCATION: AUXILIARY BLDG., TURBINE BLDG., HEATER BAY—ELEVATION: 426'-0"

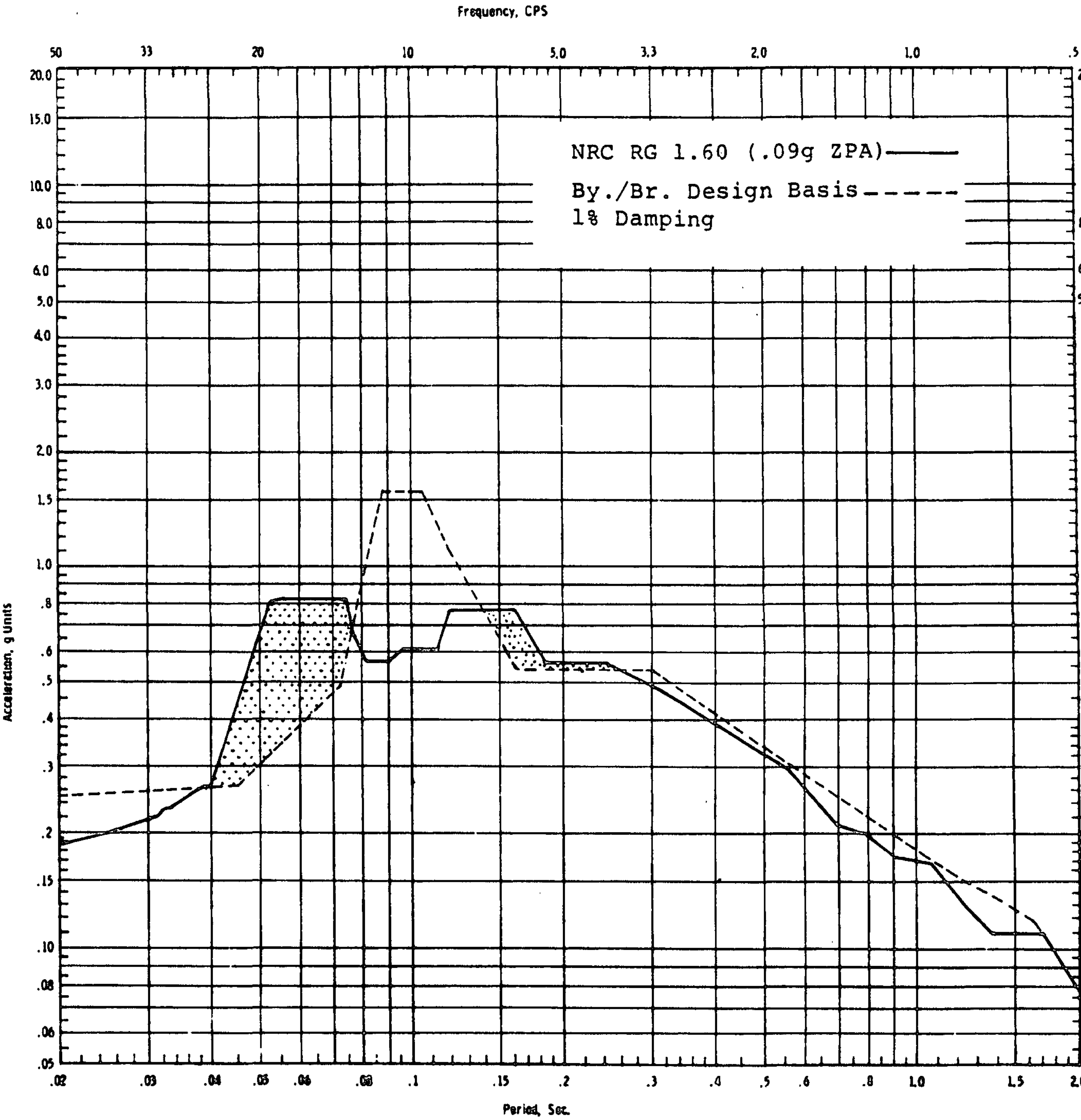

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FIGURE Q130.6-65
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 426'-0"; 439'-0"; 451'-0"

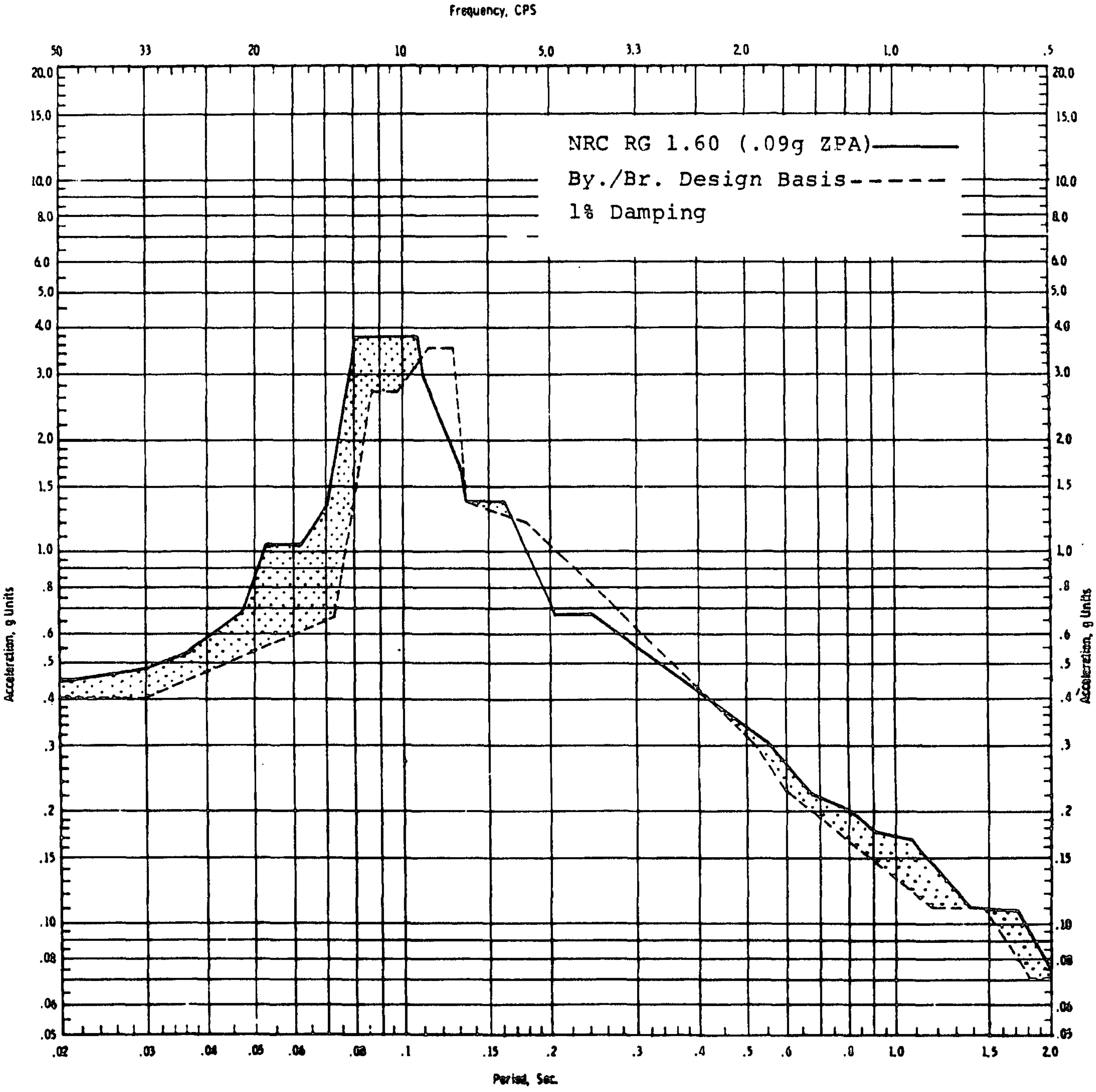

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FIGURE Q130.6-66
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 426'-0"; 439'-0"; 451'-0"

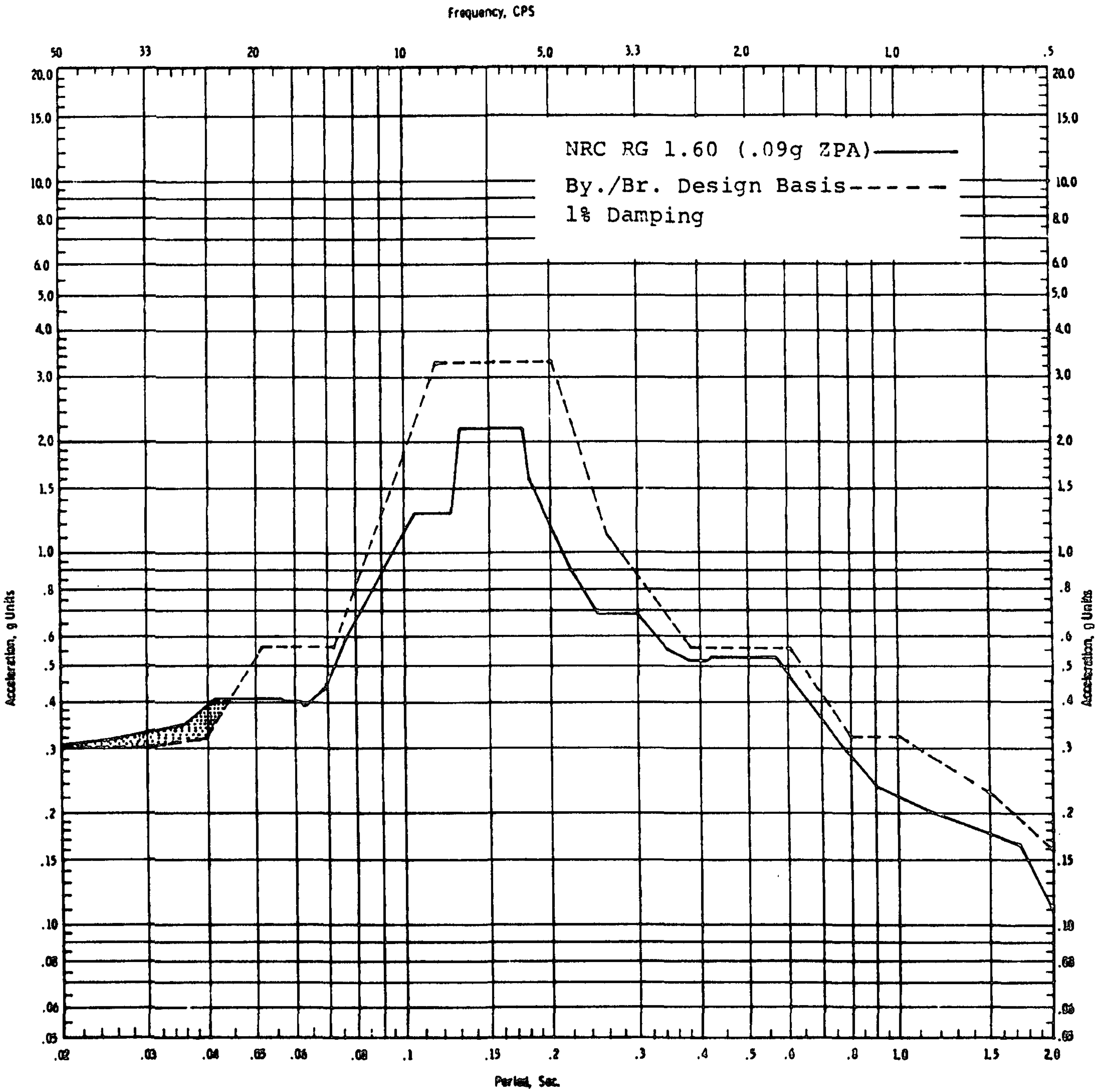

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FIGURE Q130.6-67
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: AUXILIARY BLDG., TURBINE BLDG., HEATER BAY—ELEVATION: 451'-0"

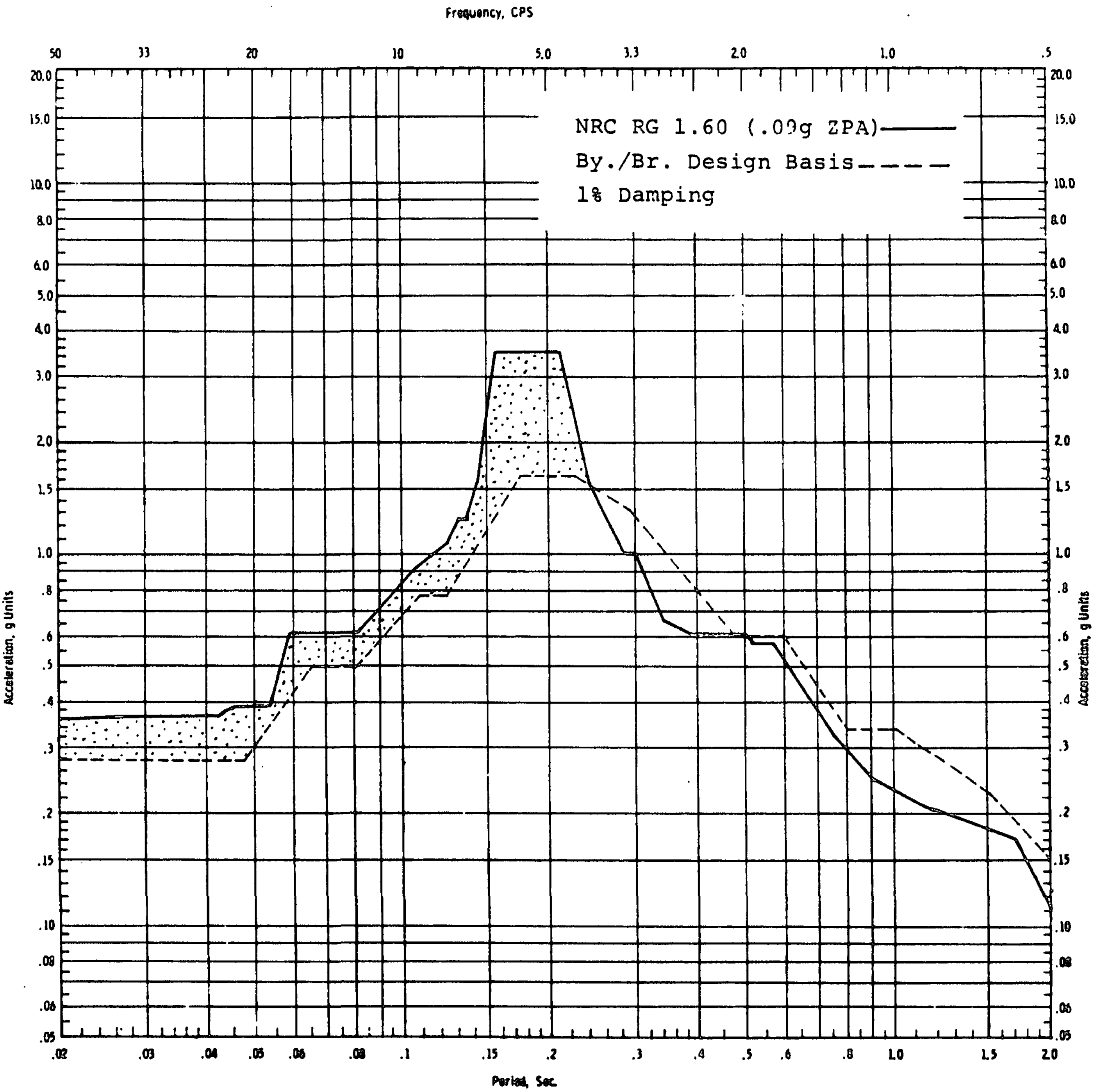

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FIGURE Q130.6-68
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, EW
LOCATION: AUXILIARY BLDG., TURBINE BLDG., HEATER BAY—ELEVATION: 451'-0"

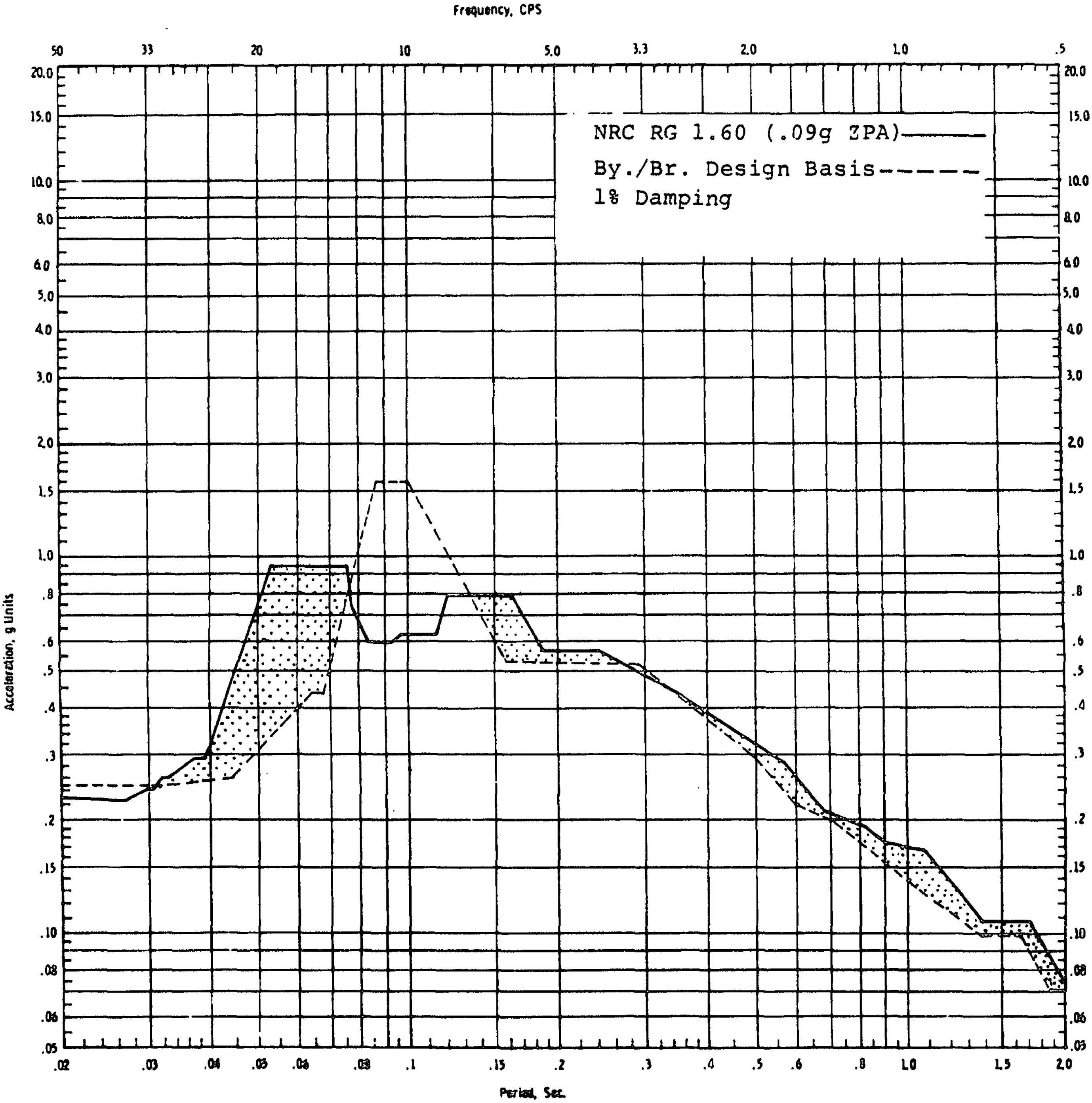

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FIGURE Q130.6-69
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 467'-0"; 473'-0"; 477'-0"; 485'-0"

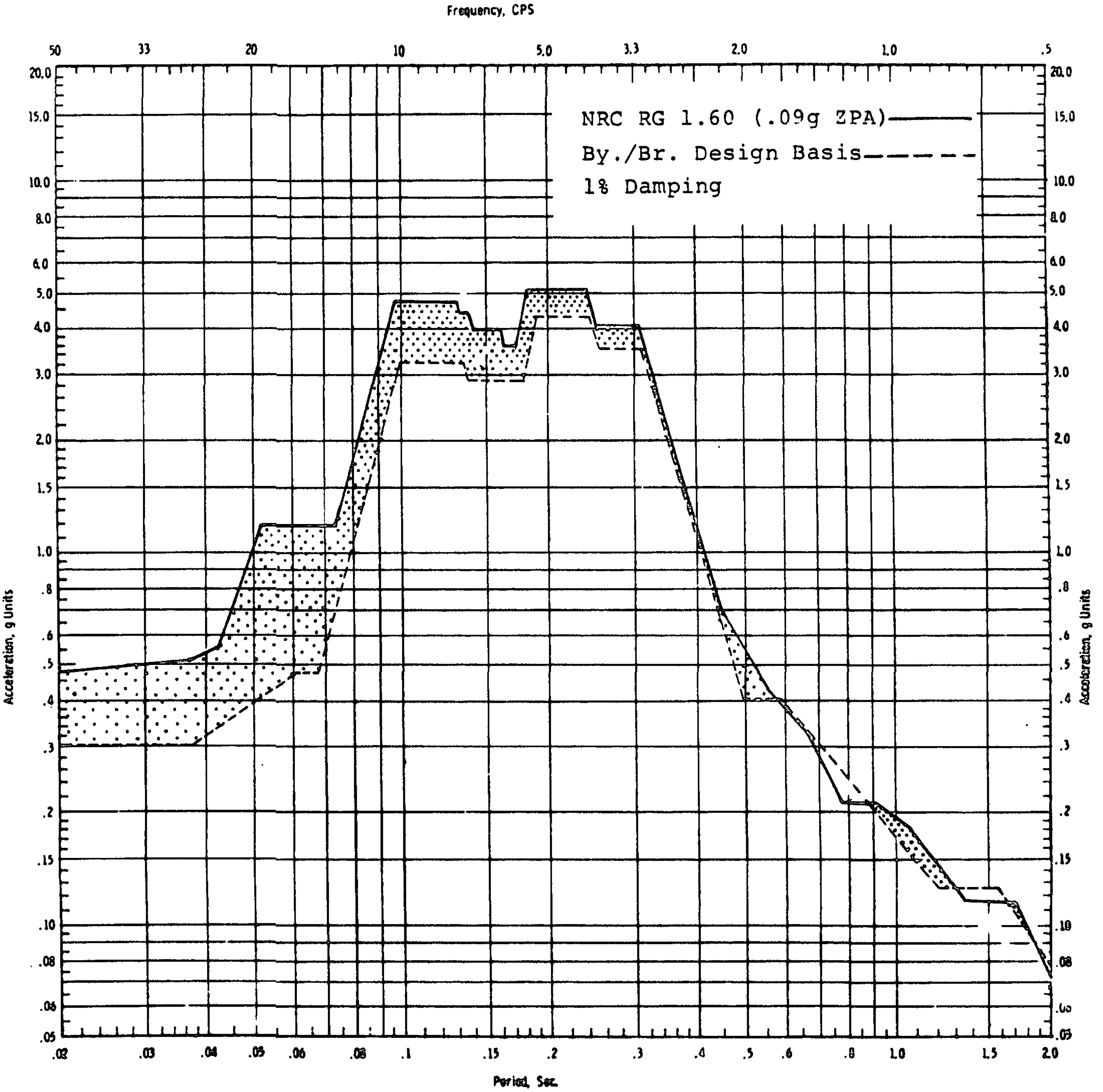

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FIGURE Q130.6-70
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 467'-0"; 477'-0"

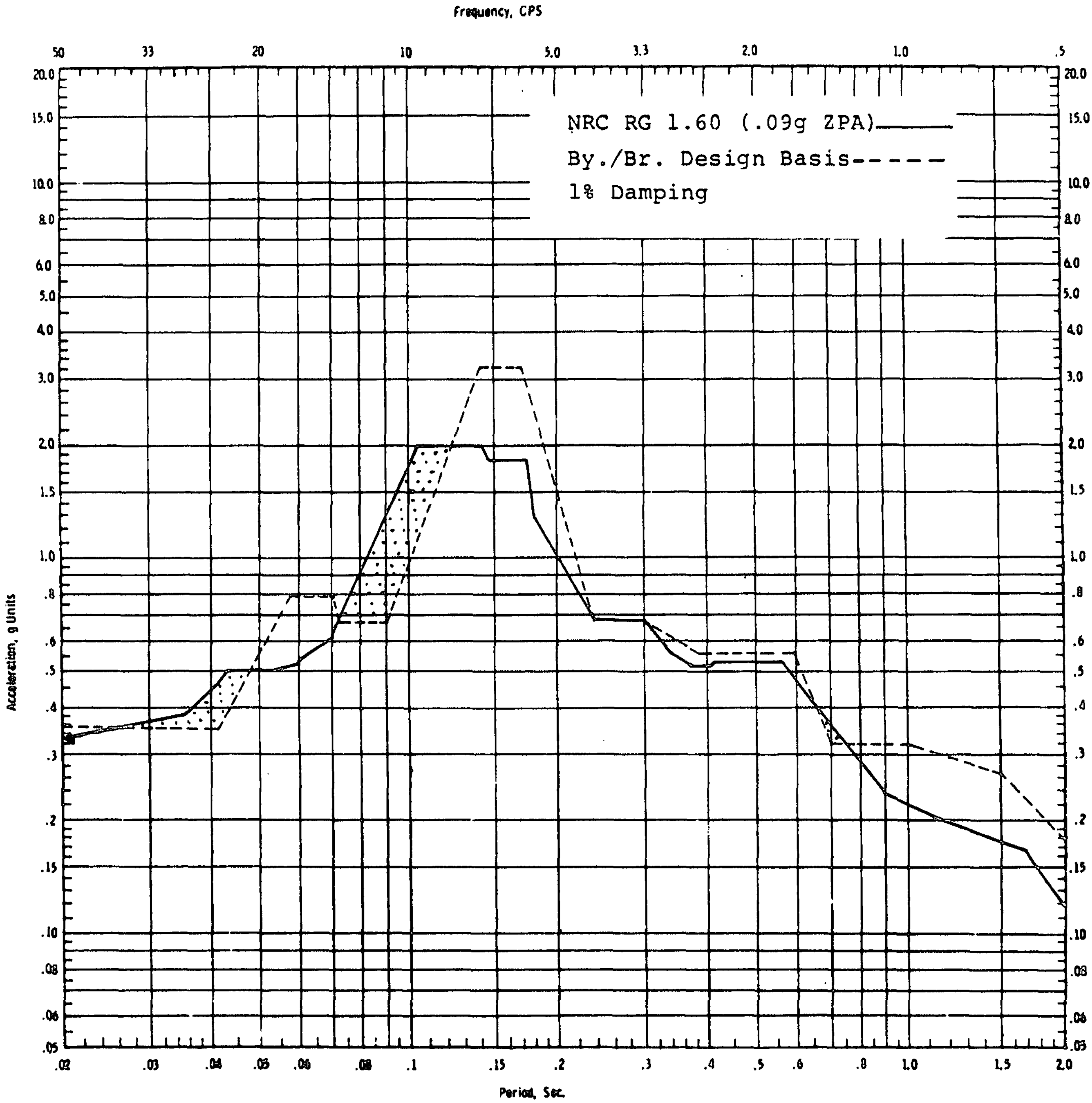

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FIGURE Q130.6-71
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: AUXILIARY BUILDING
ELEVATION: 477'-0"

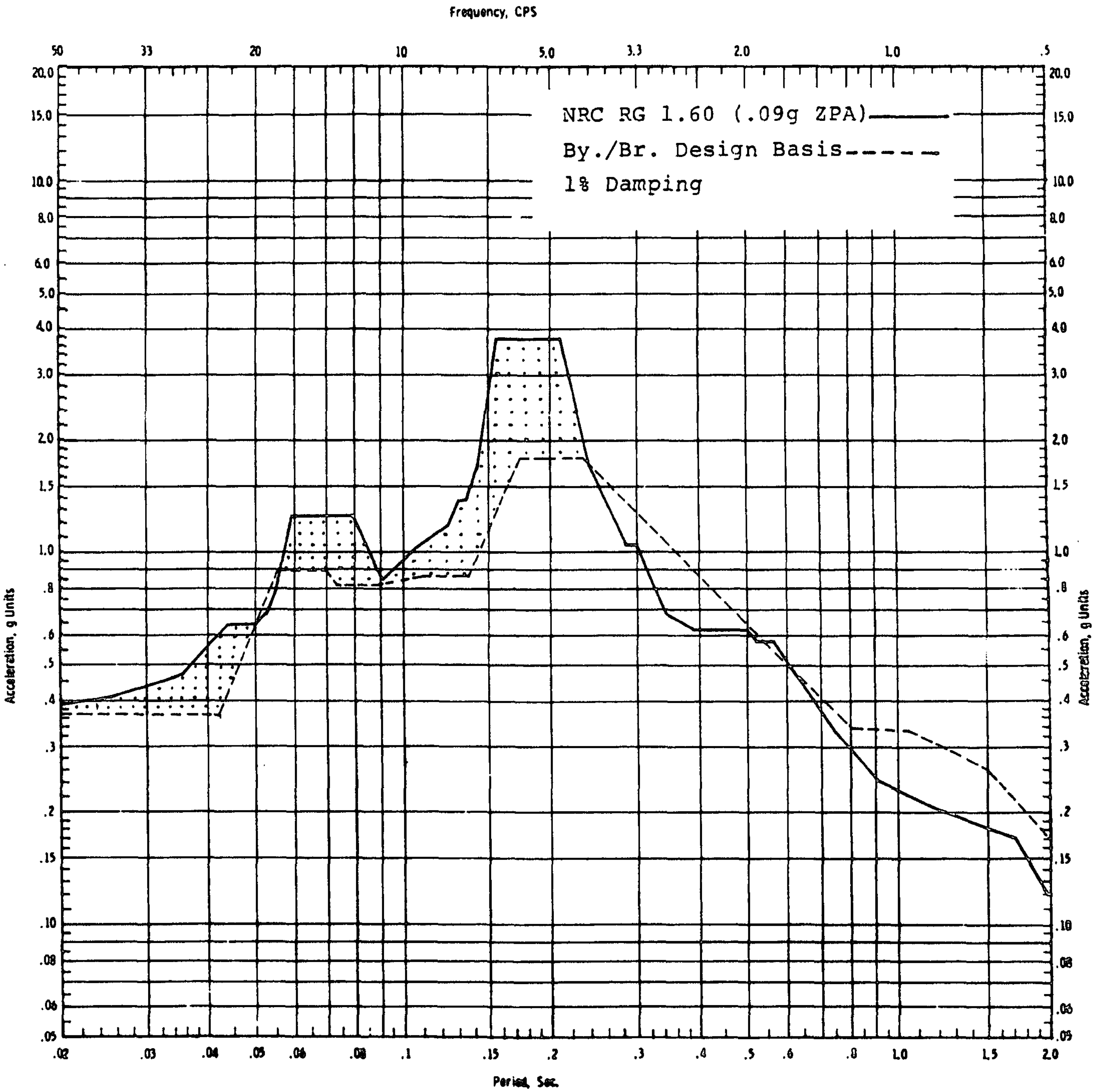

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FIGURE Q130.6-72
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, EW
LOCATION: AUXILIARY BUILDING
ELEVATION: 477'-0"

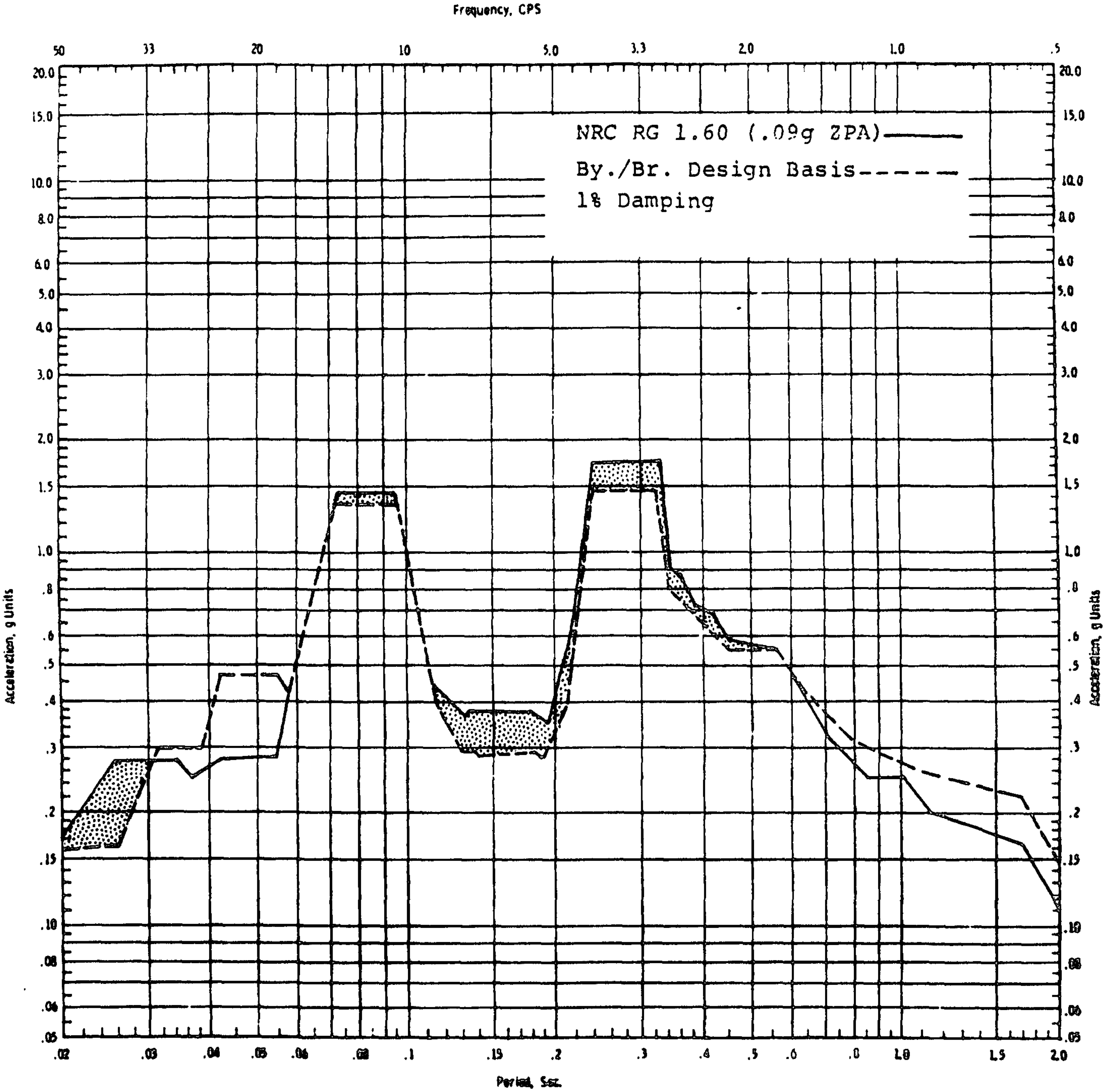

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FIGURE Q130.6-73
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: CONTAINMENT BUILDING
ELEVATION: 424'-0"; 436'-0"

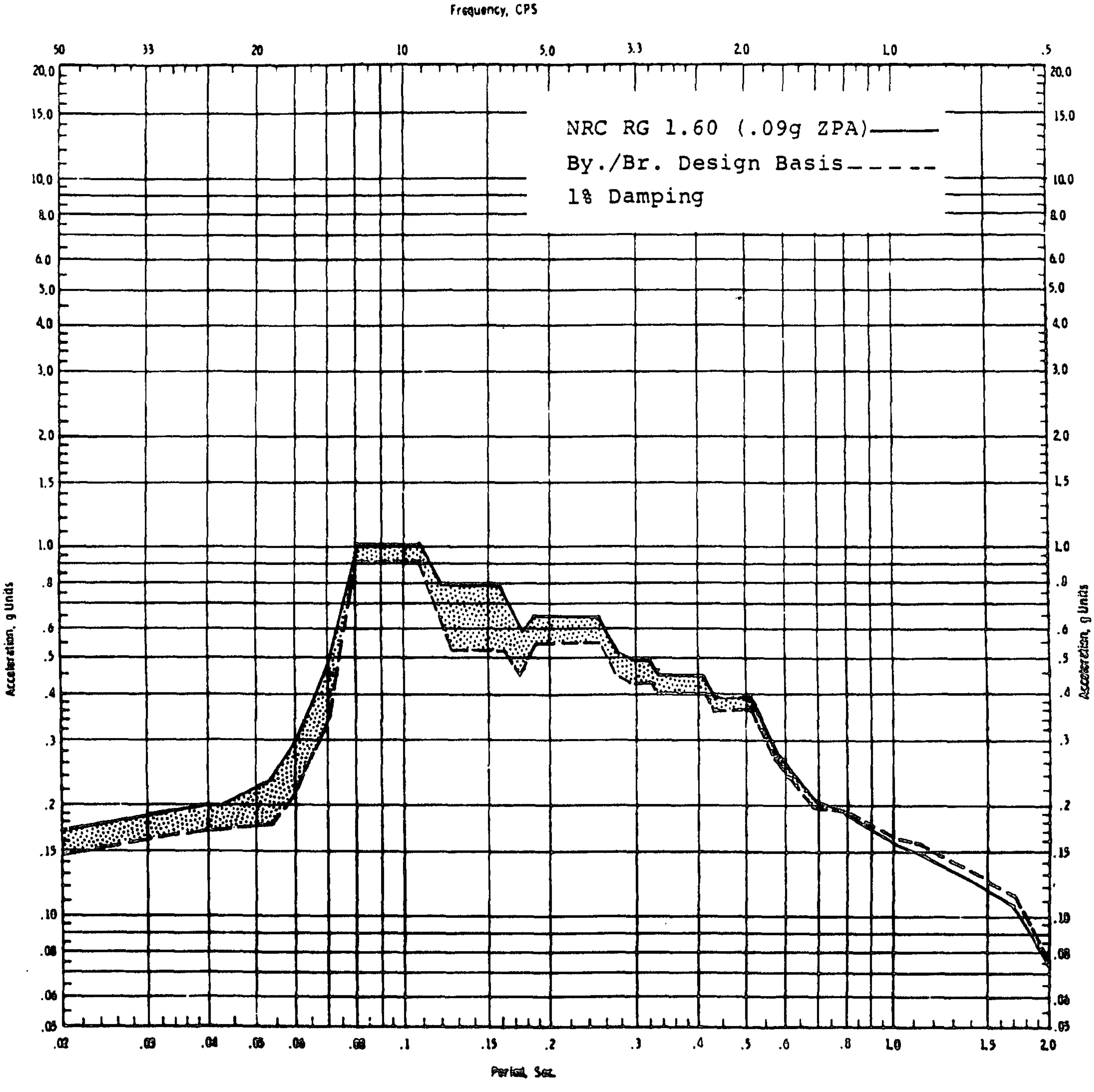

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FIGURE Q130.6-74
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: CONTAINMENT BUILDING WALL
ELEVATION: 424'-0"; 436'-0"

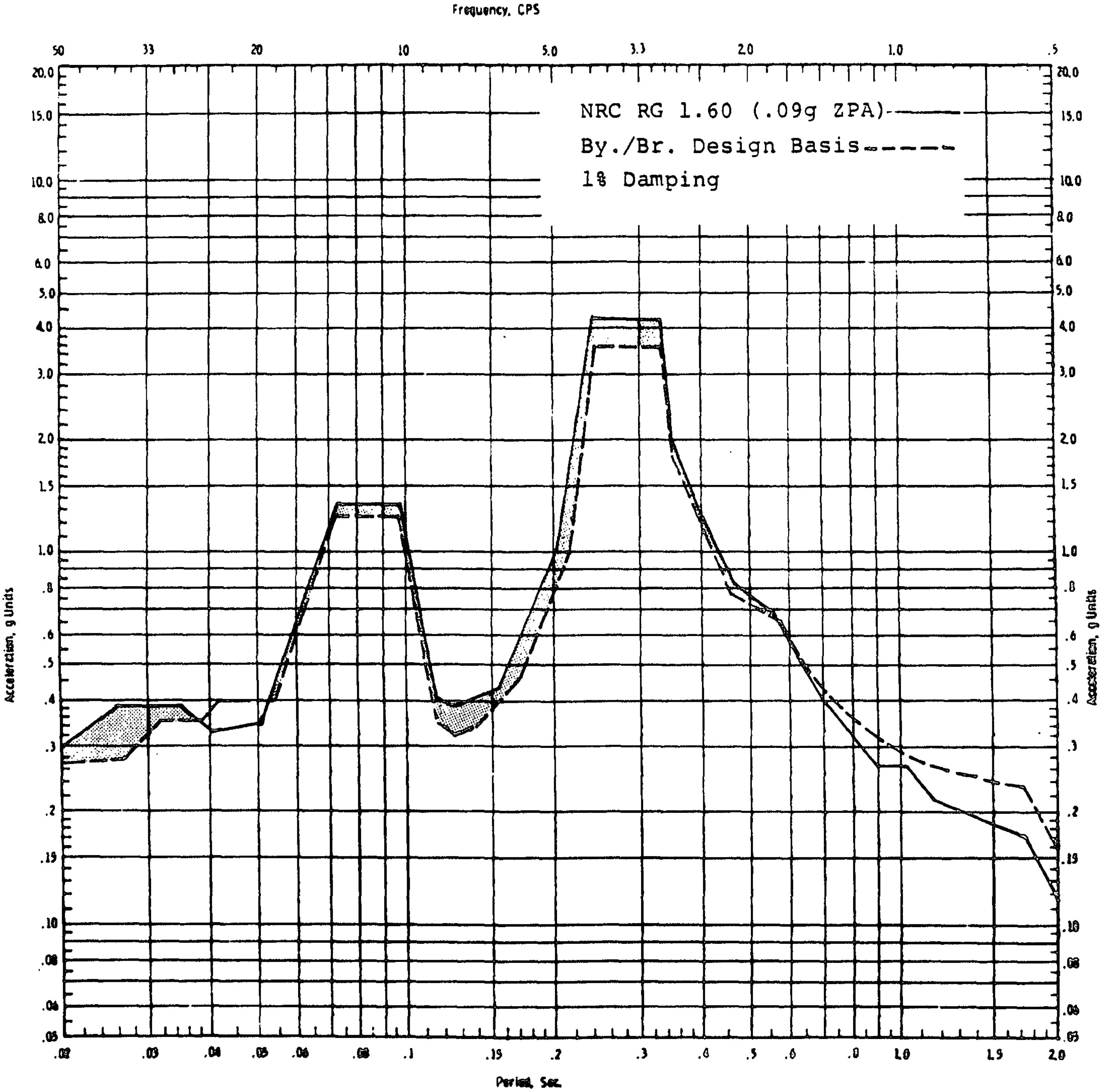

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FIGURE Q130.6-75
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS AND EW
LOCATION: CONTAINMENT BUILDING
ELEVATION: 496'-0"

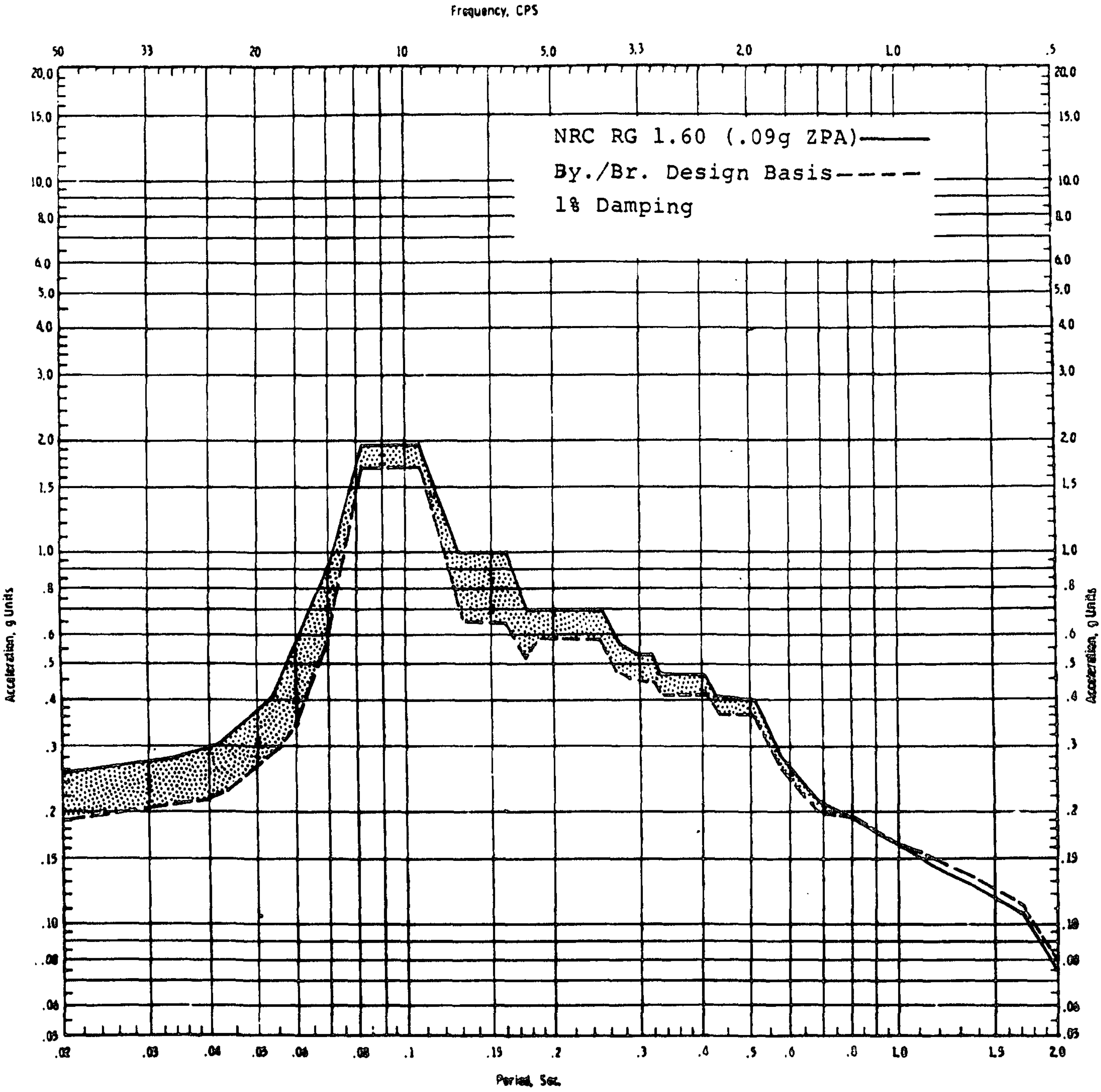

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FIGURE Q130.6-76
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: CONTAINMENT BUILDING WALL
ELEVATION: 496'-0"

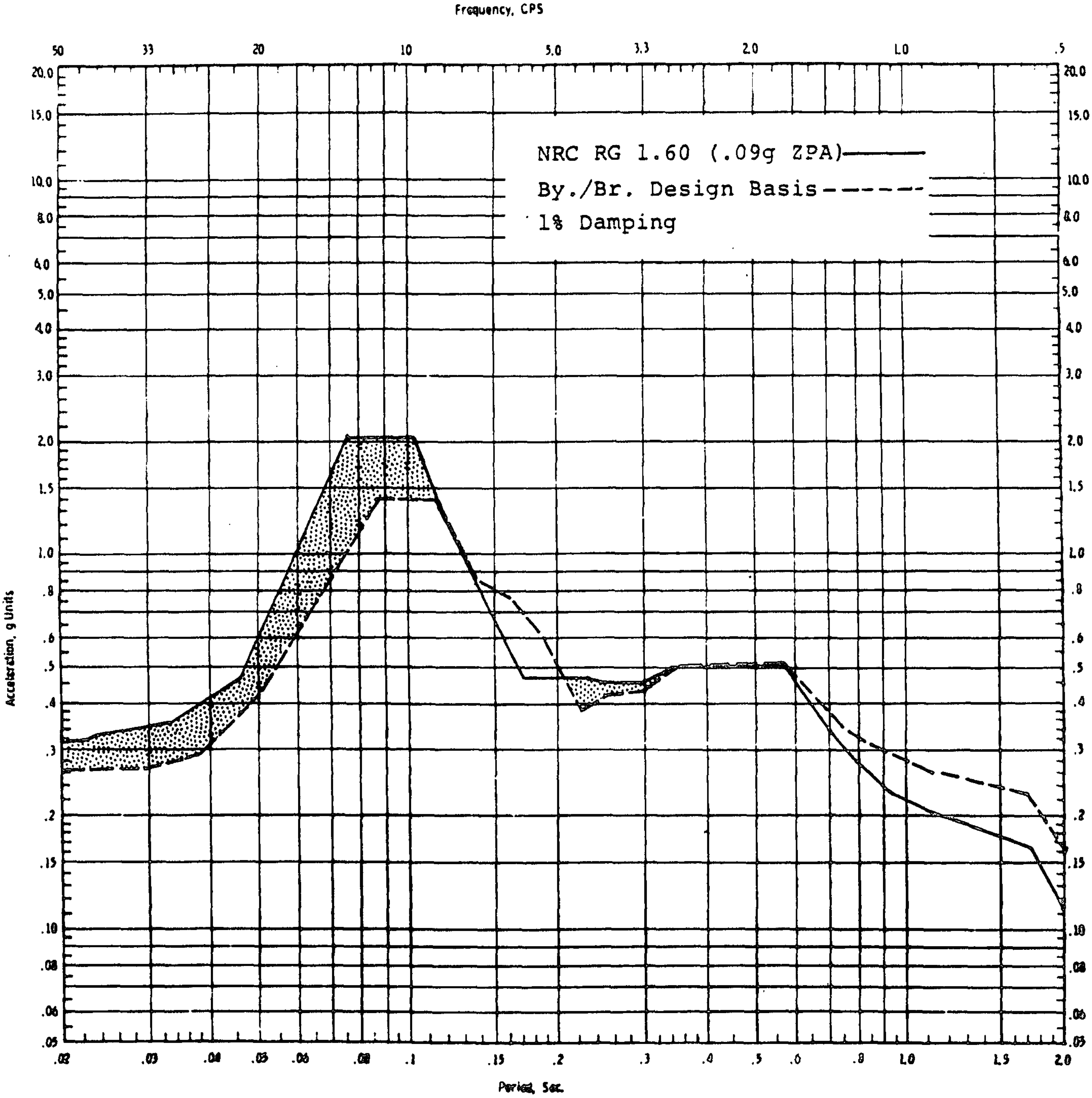

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FIGURE Q130.6-77
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, NS
LOCATION: CONTAINMENT INNER STRUCTURE
ELEVATION: 426'-0"

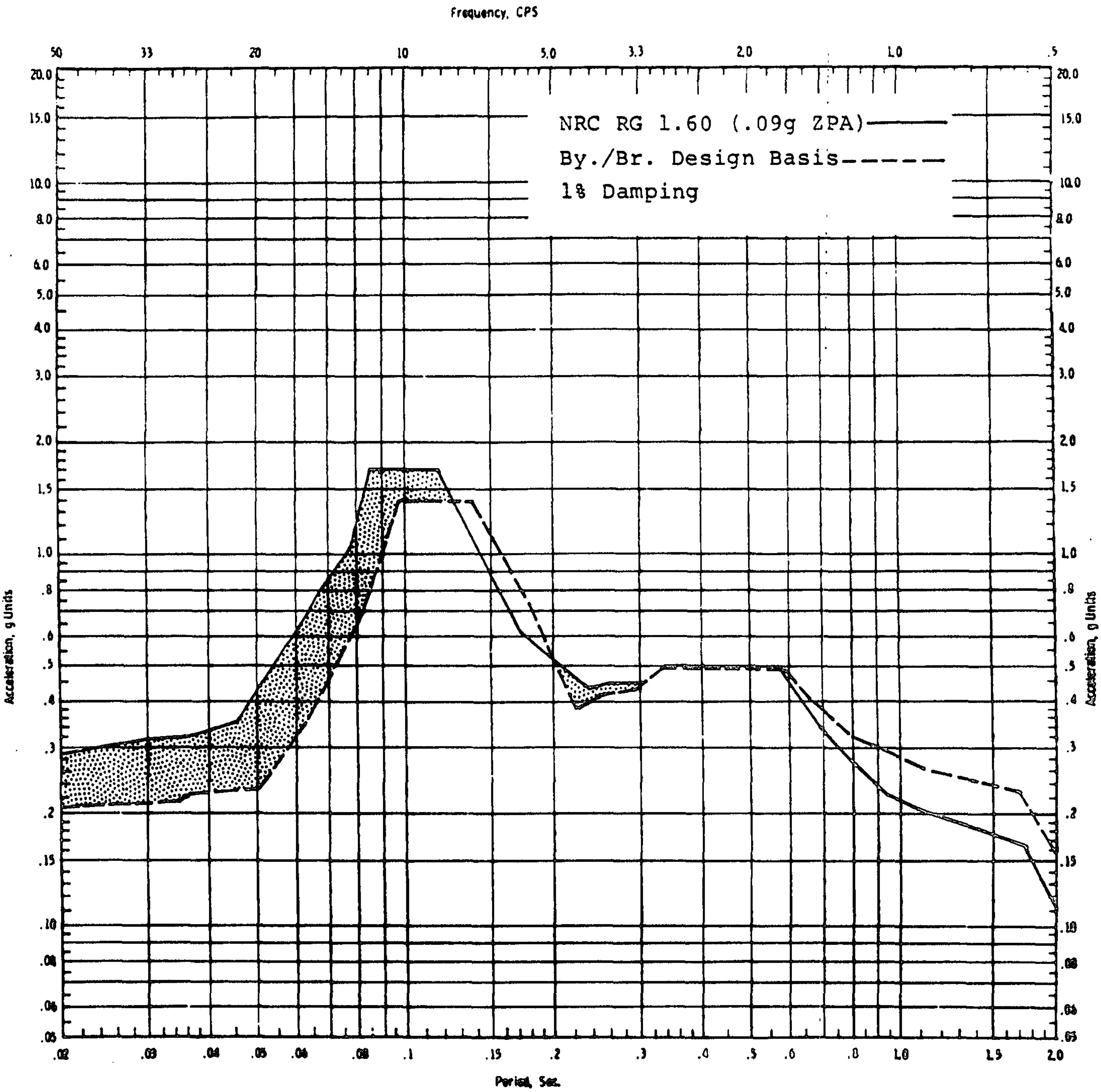

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FIGURE Q130.6-78
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, HORIZONTAL, EW
LOCATION: CONTAINMENT INNER STRUCTURE
ELEVATION: 426'-0"

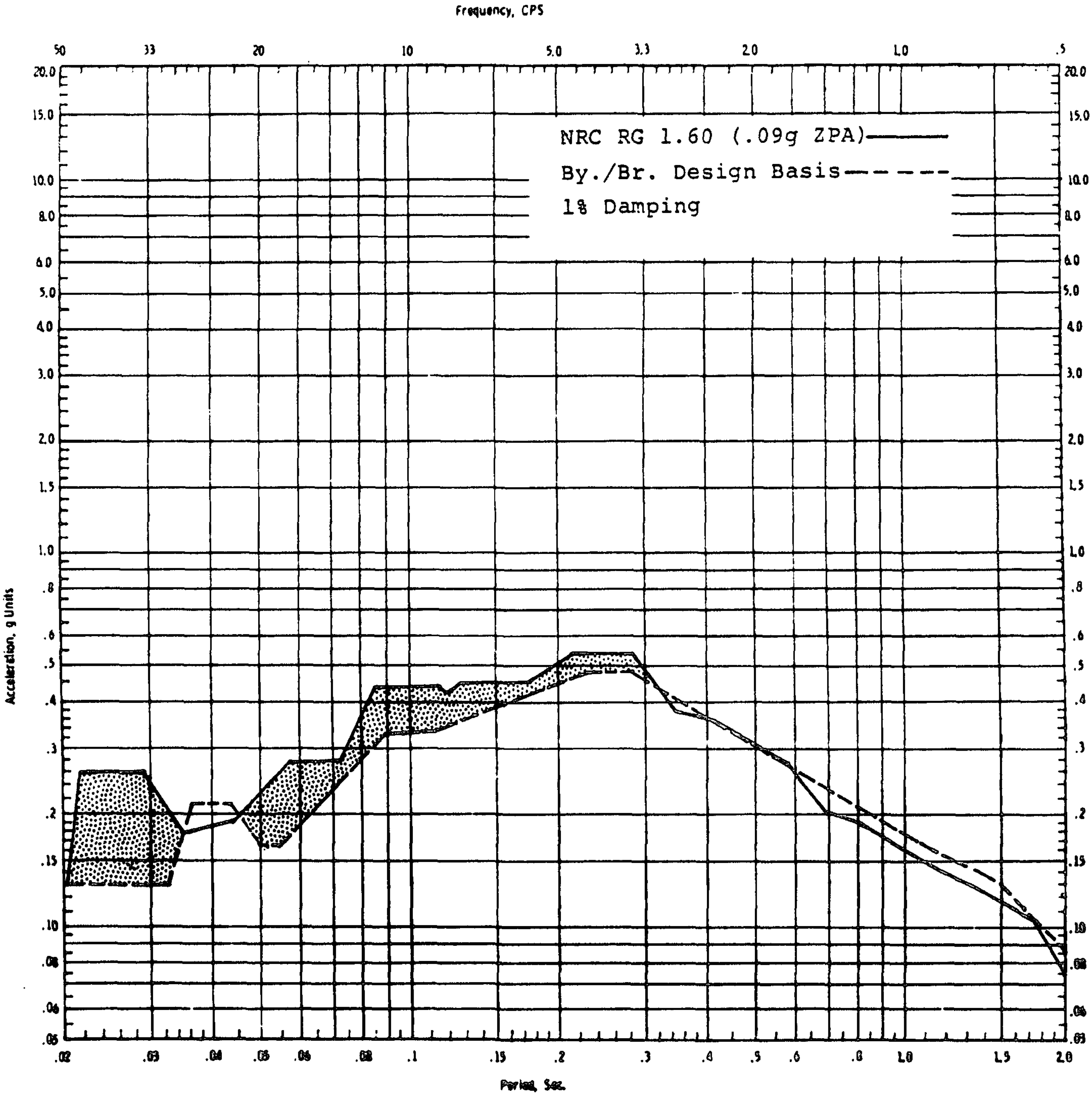

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FIGURE Q130.6-79
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, WALL
LOCATION: CONTAINMENT INNER STRUCTURE WALL
ELEVATION: 412'-0"; 426'-0"

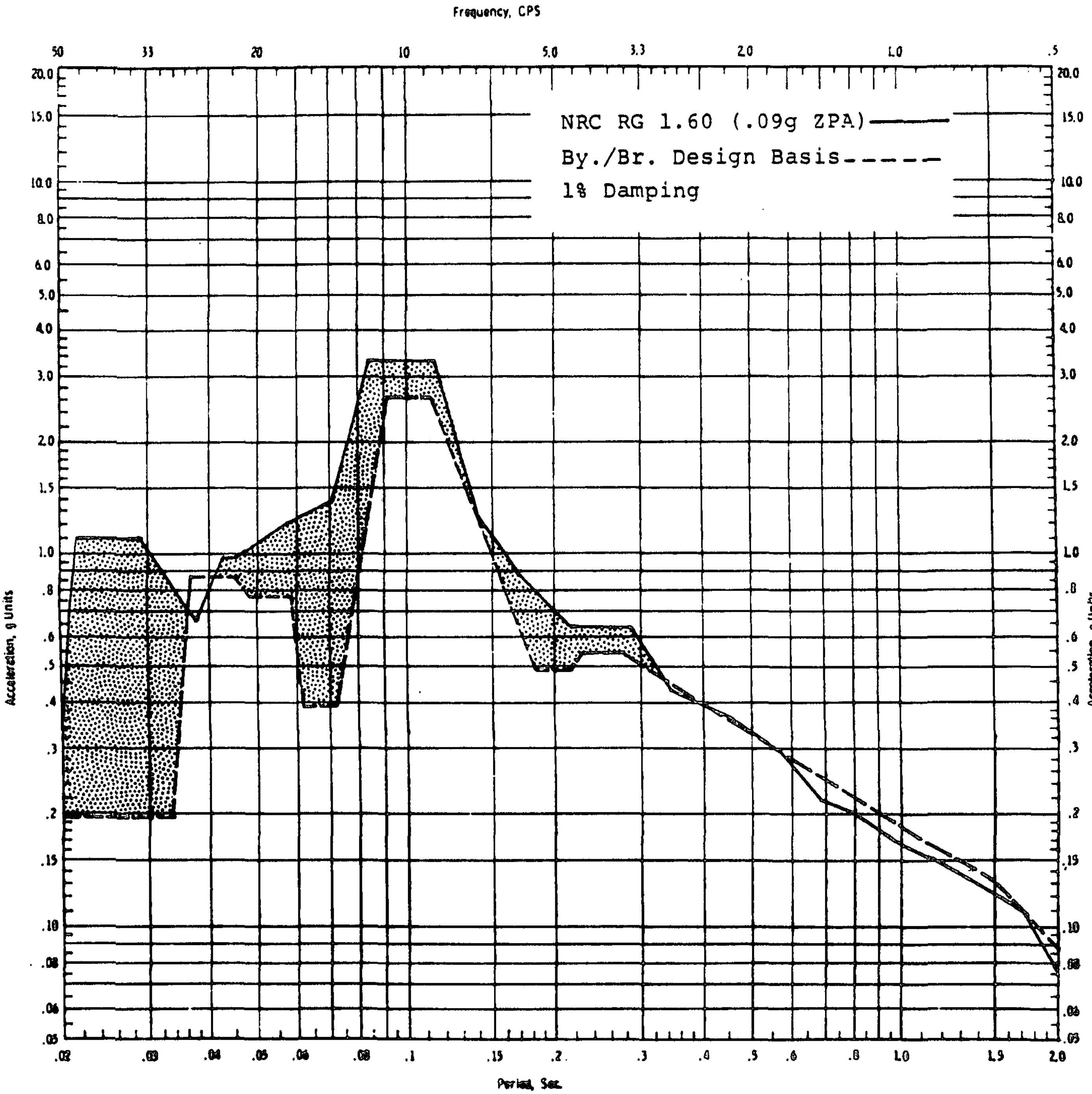

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FIGURE Q130.6-80
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: OBE, VERTICAL, SLAB
LOCATION: CONTAINMENT INNER STRUCTURE SLAB
ELEVATION: 390'-0"; 401'-0"; 412'-0"; 426'-0"

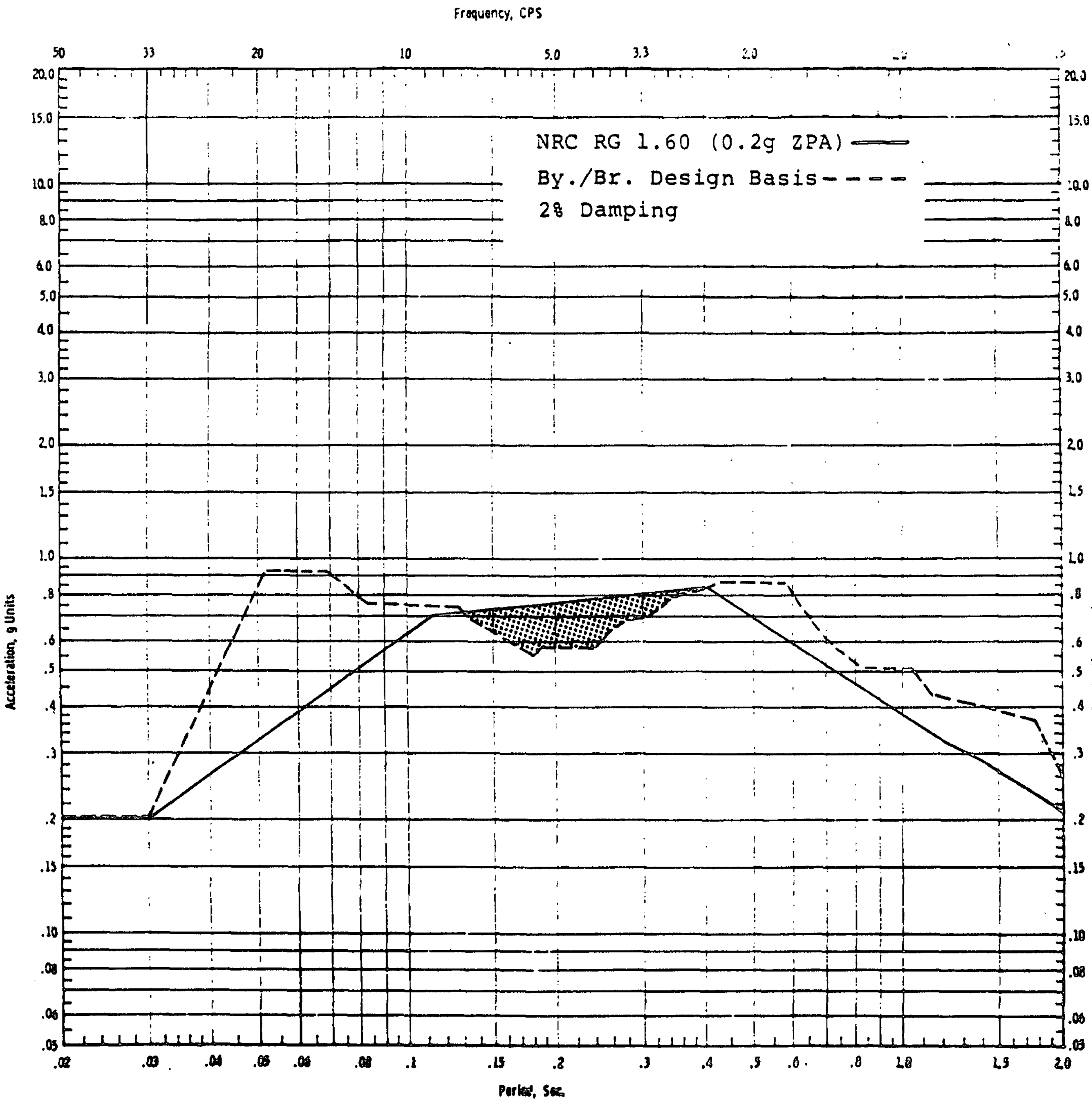

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FIGURE Q130.6-81
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS AND EW
LOCATION: AUXILIARY AND CONTAINMENT BLDG.
ELEVATION: 330'-0"; 374'-0"

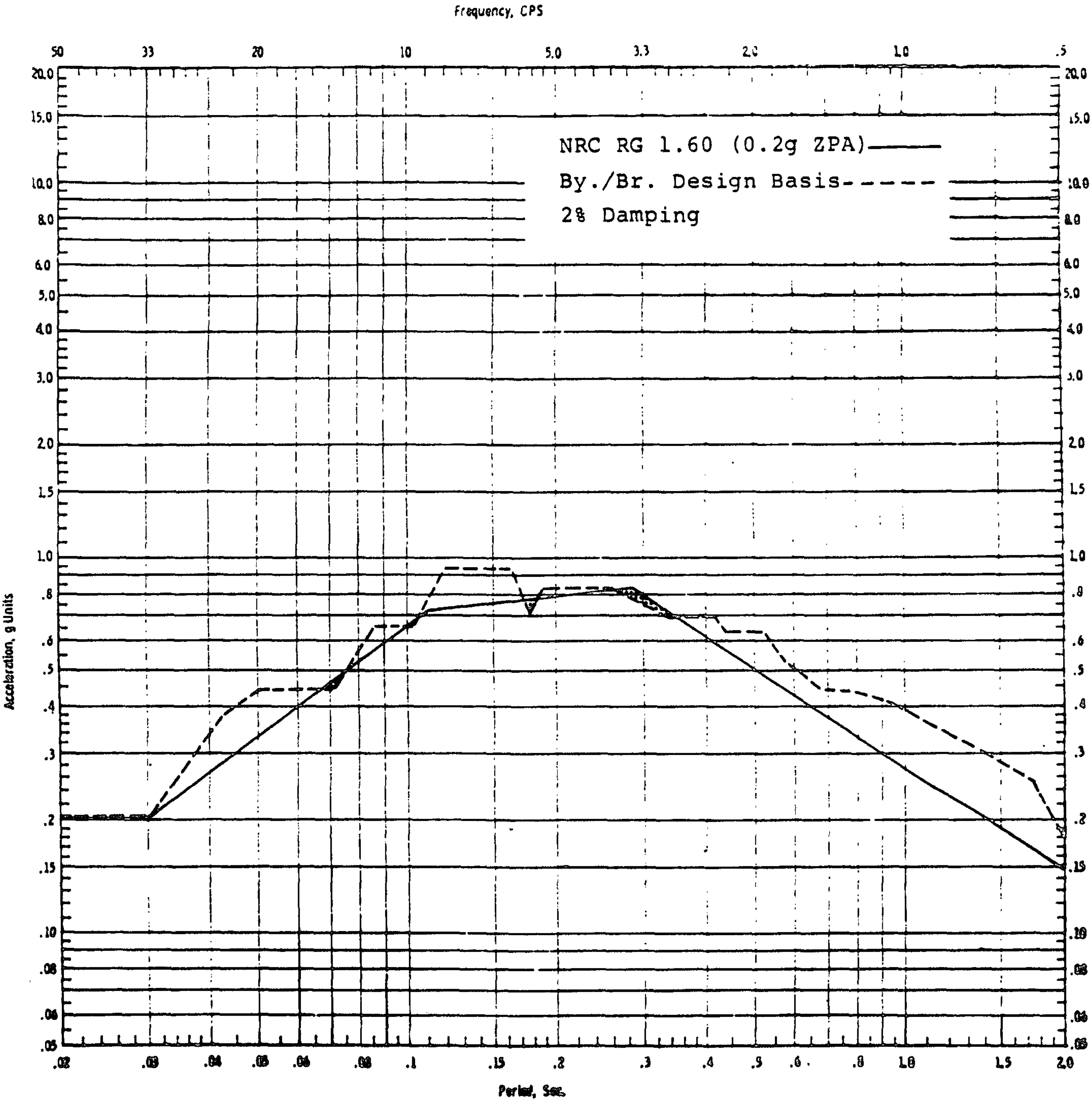

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FIGURE Q130.6-82
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL AND SLAB
LOCATION: AUXILIARY AND CONTAINMENT BLDG.
ELEVATION: 330'-0"; 374'-0"

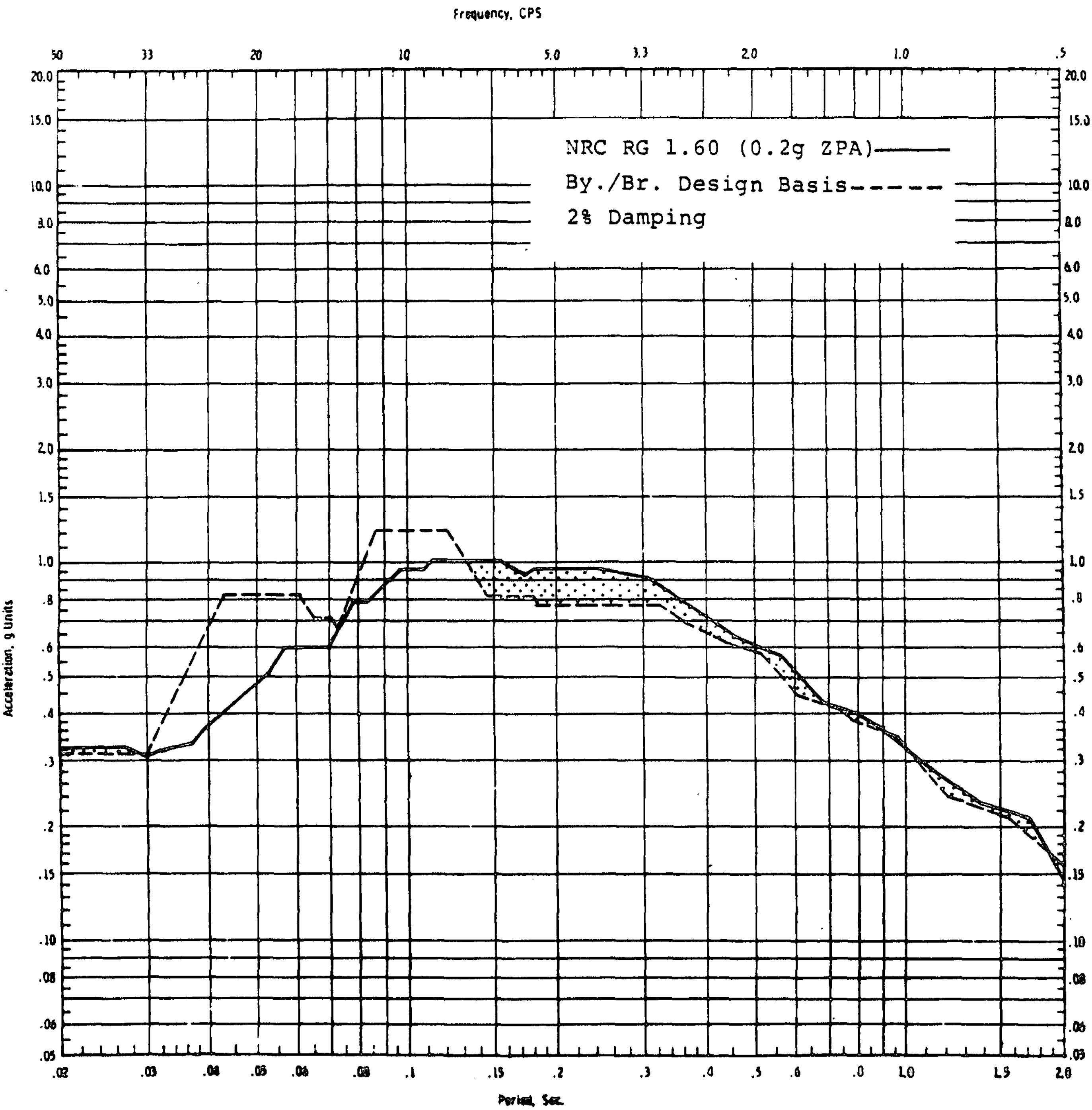

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FIGURE Q130.6-83
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 346'-0"; 364'-0"; 383'-0"; 401'-0"

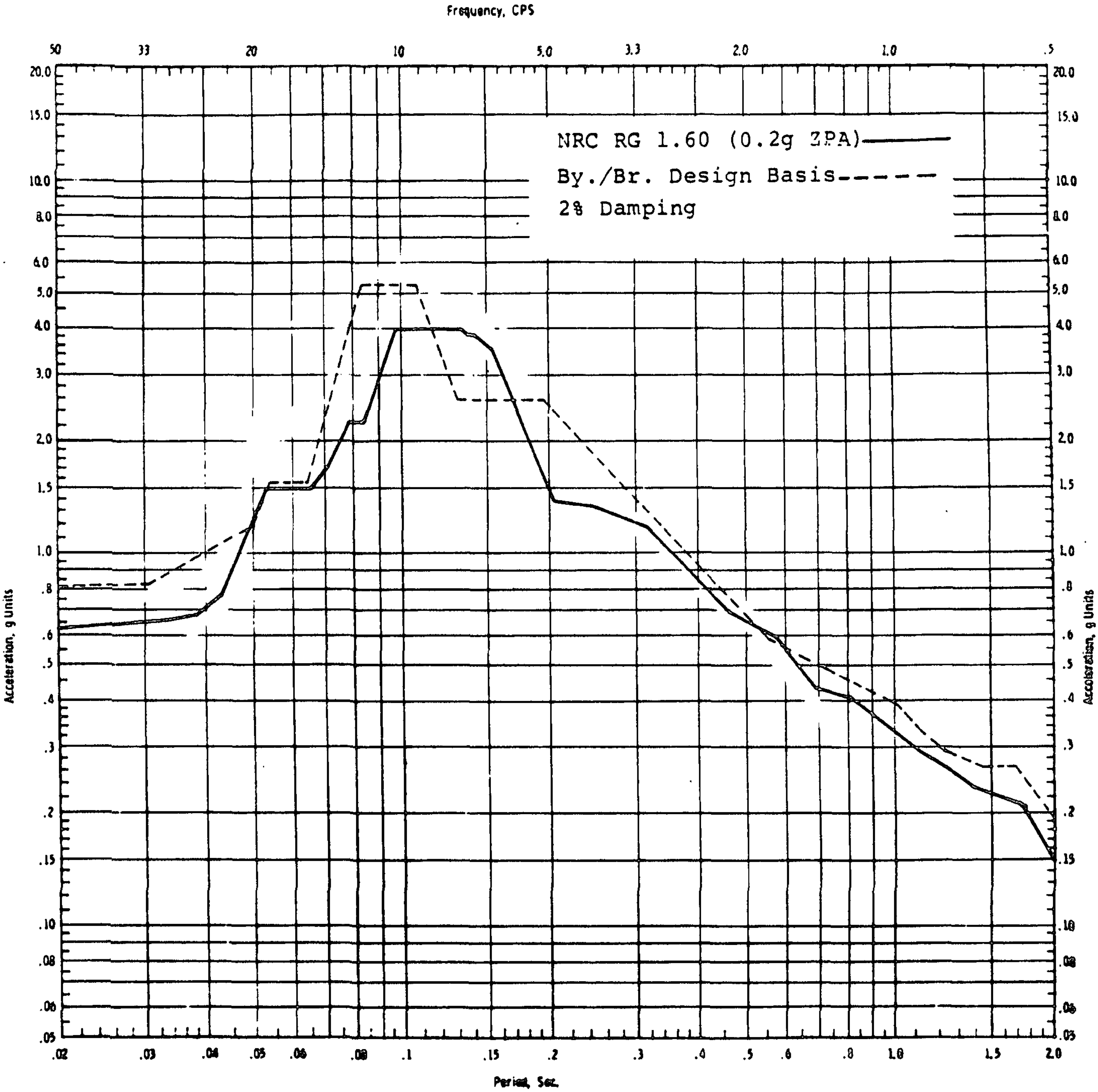

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FIGURE Q130.6-84
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 346'-0"; 364'-0"; 383'-0"; 401'-0"

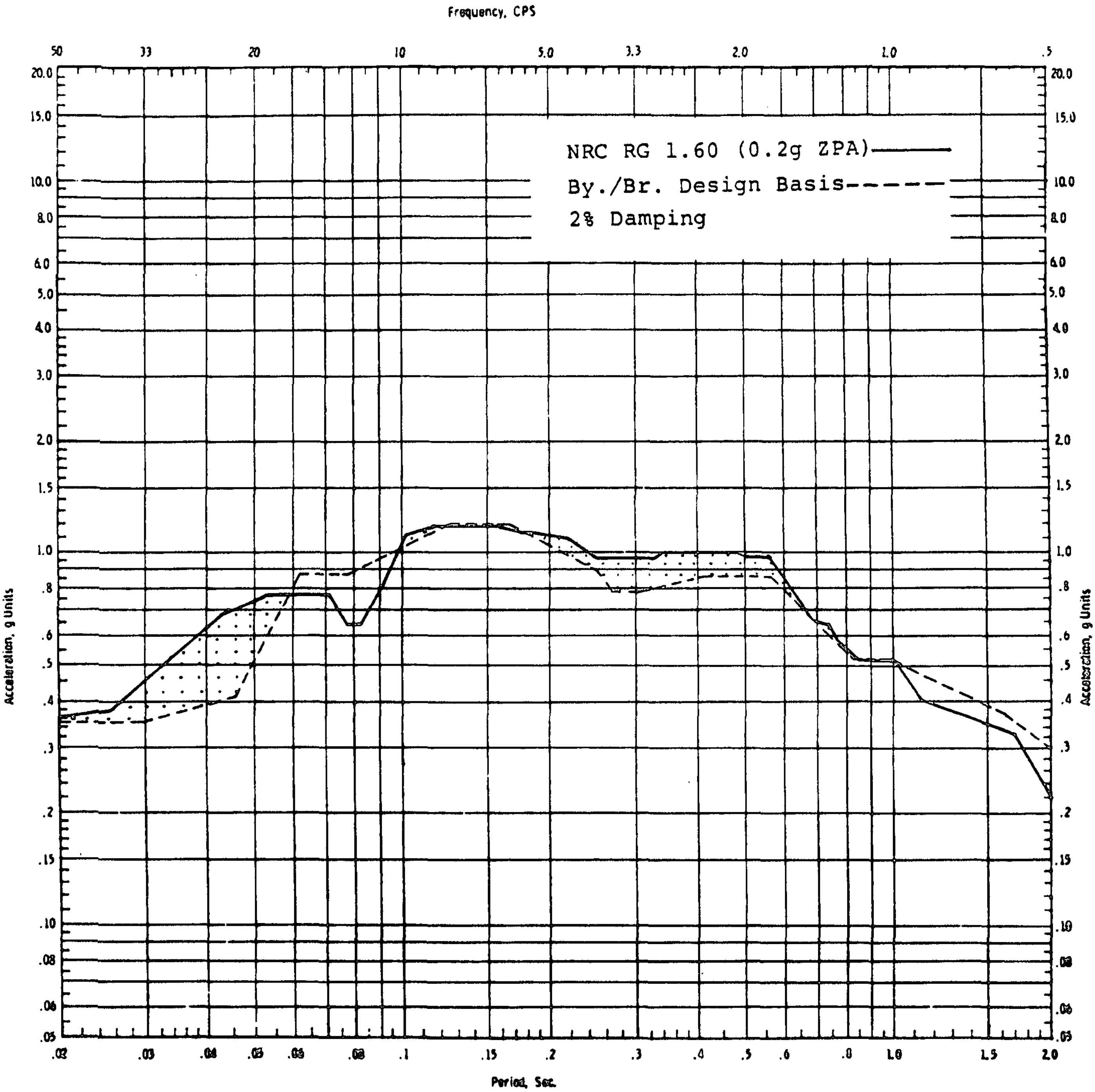

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FIGURE Q130.6-85
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS
LOCATION: AUXILIARY BUILDING
ELEVATION: 401'-0"

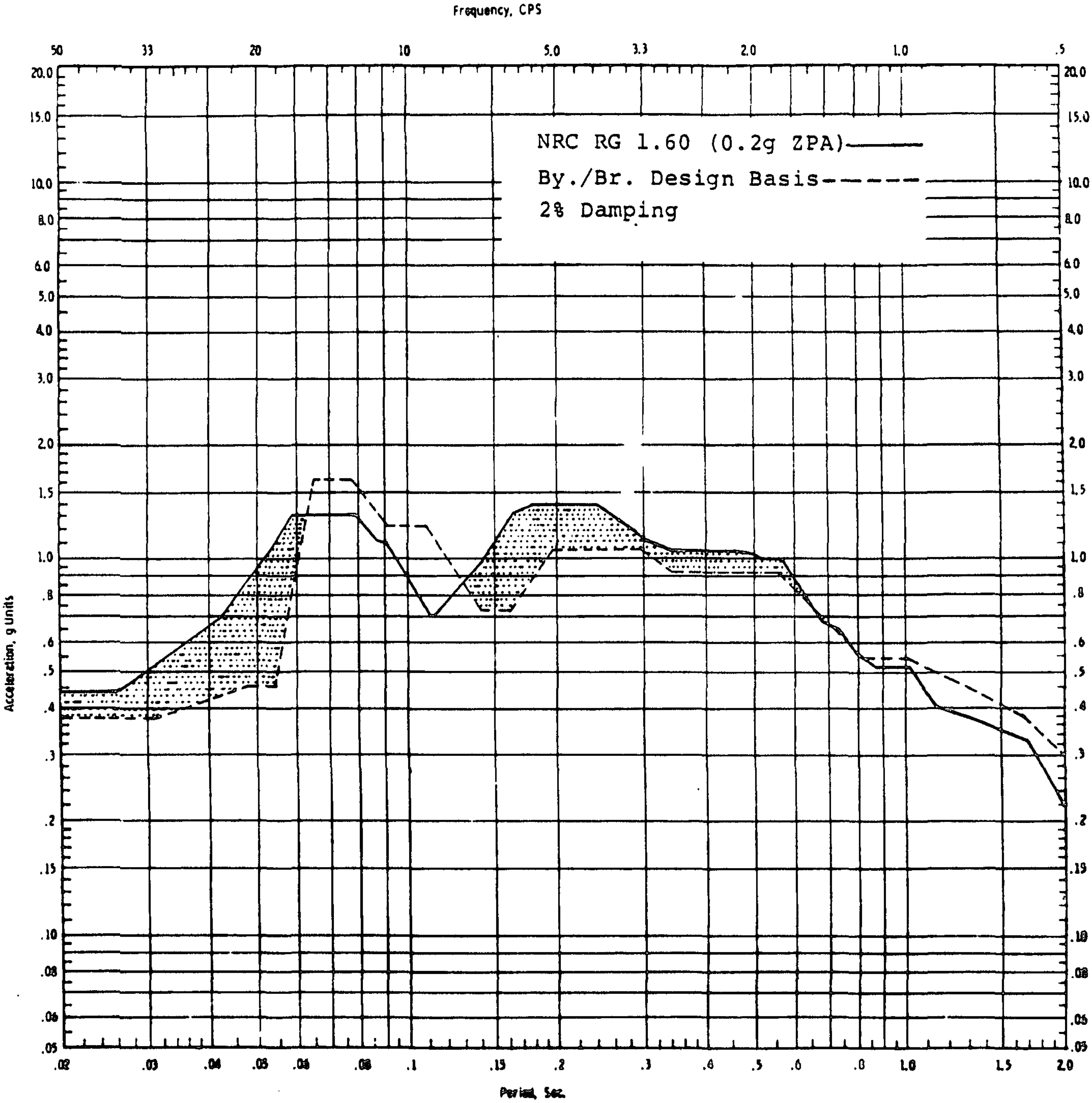

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FIGURE Q130.6-86
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, EW
LOCATION: AUXILIARY BUILDING
ELEVATION: 401'-0"

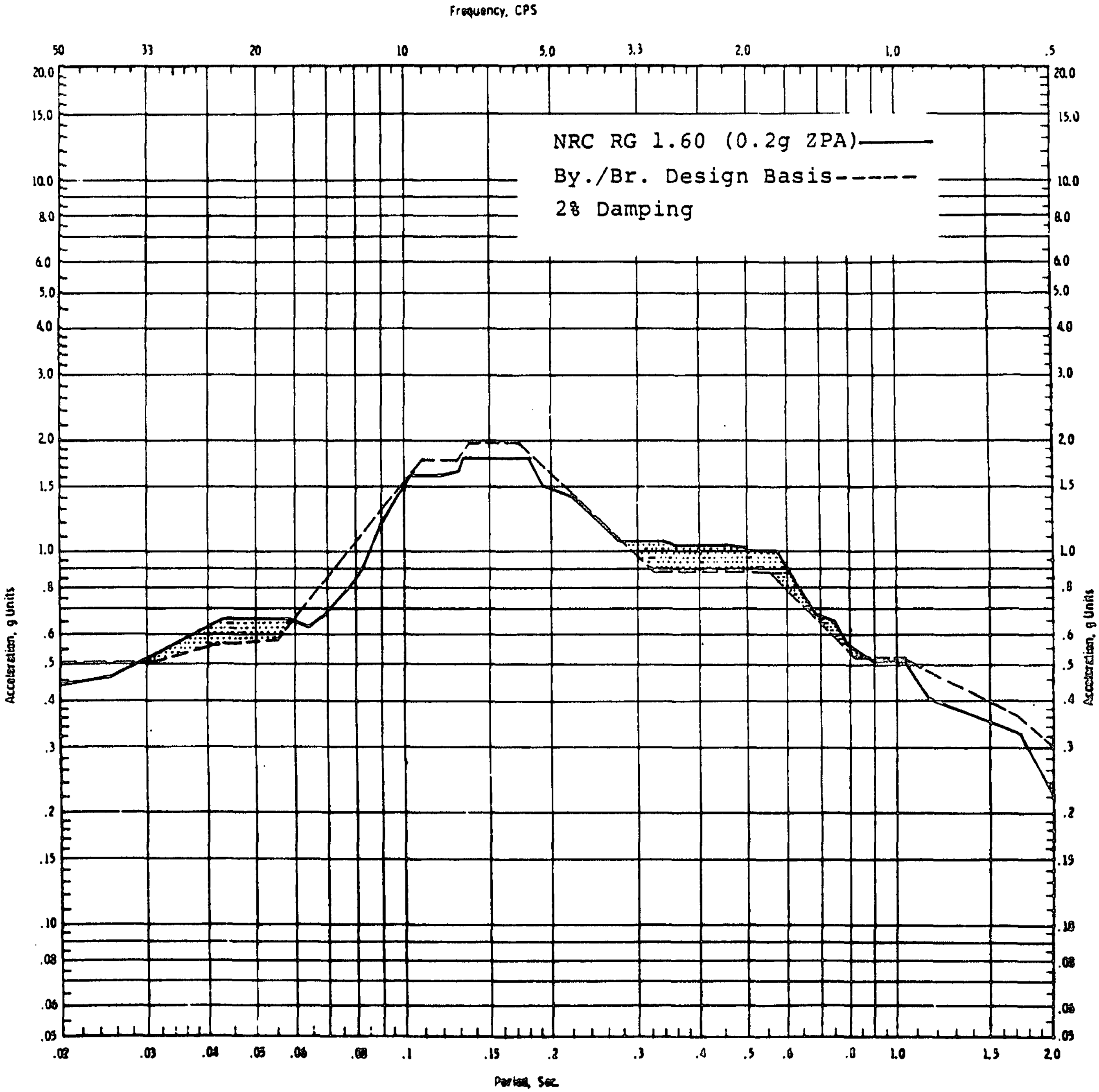

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FIGURE Q130.6-87
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS
LOCATION: AUXILIARY BLDG., TURBINE BLDG.,
HEATER BAY—ELEVATION: 426'-0"

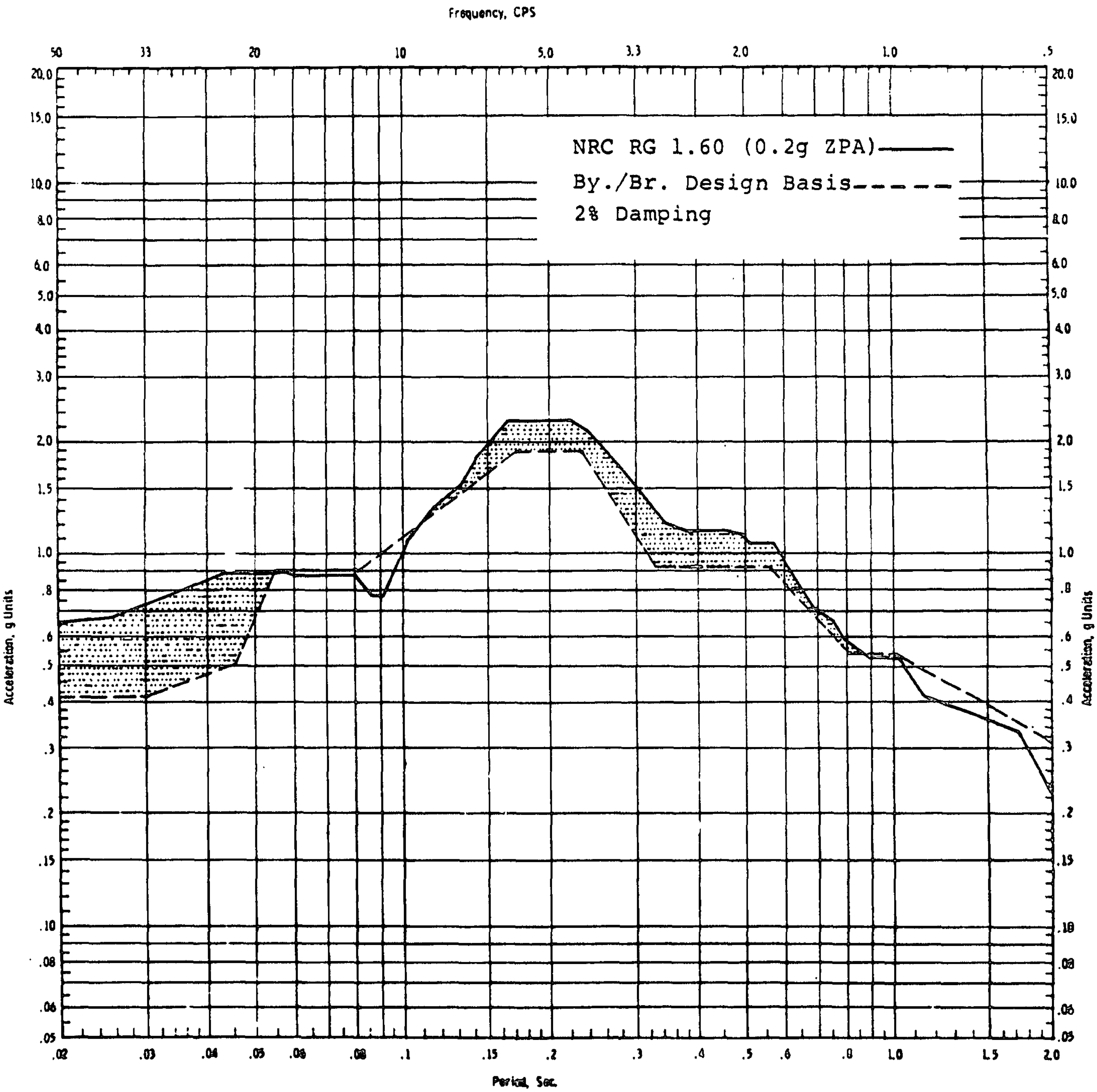

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FIGURE Q130.6-88
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, EW
LOCATION: AUXILIARY BLDG., TURBINE BLDG.,
HEATER BAY—ELEVATION: 426'-0"

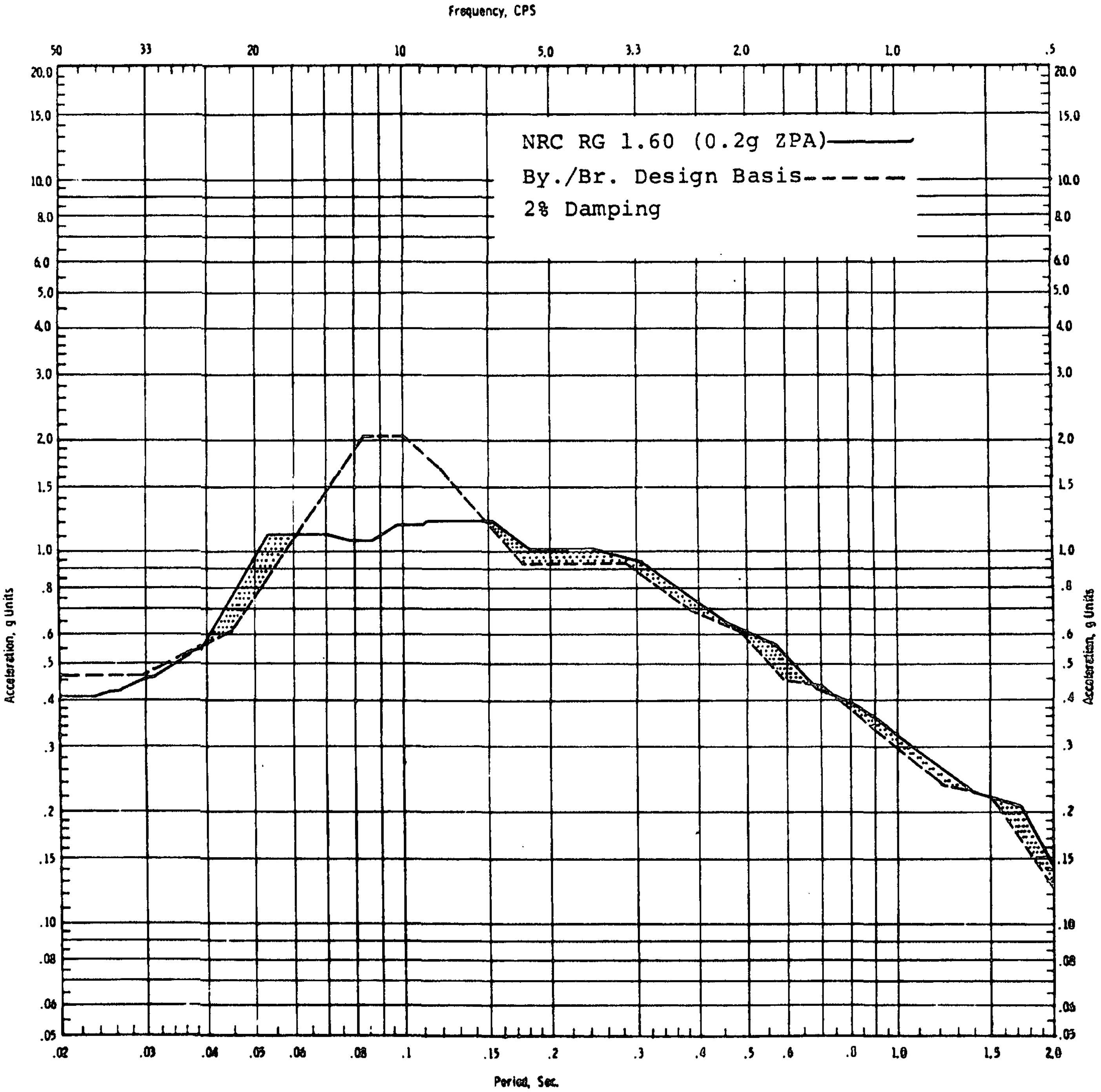

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FIGURE Q130.6-89
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 426'-0"; 439'-0"; 451'-0"

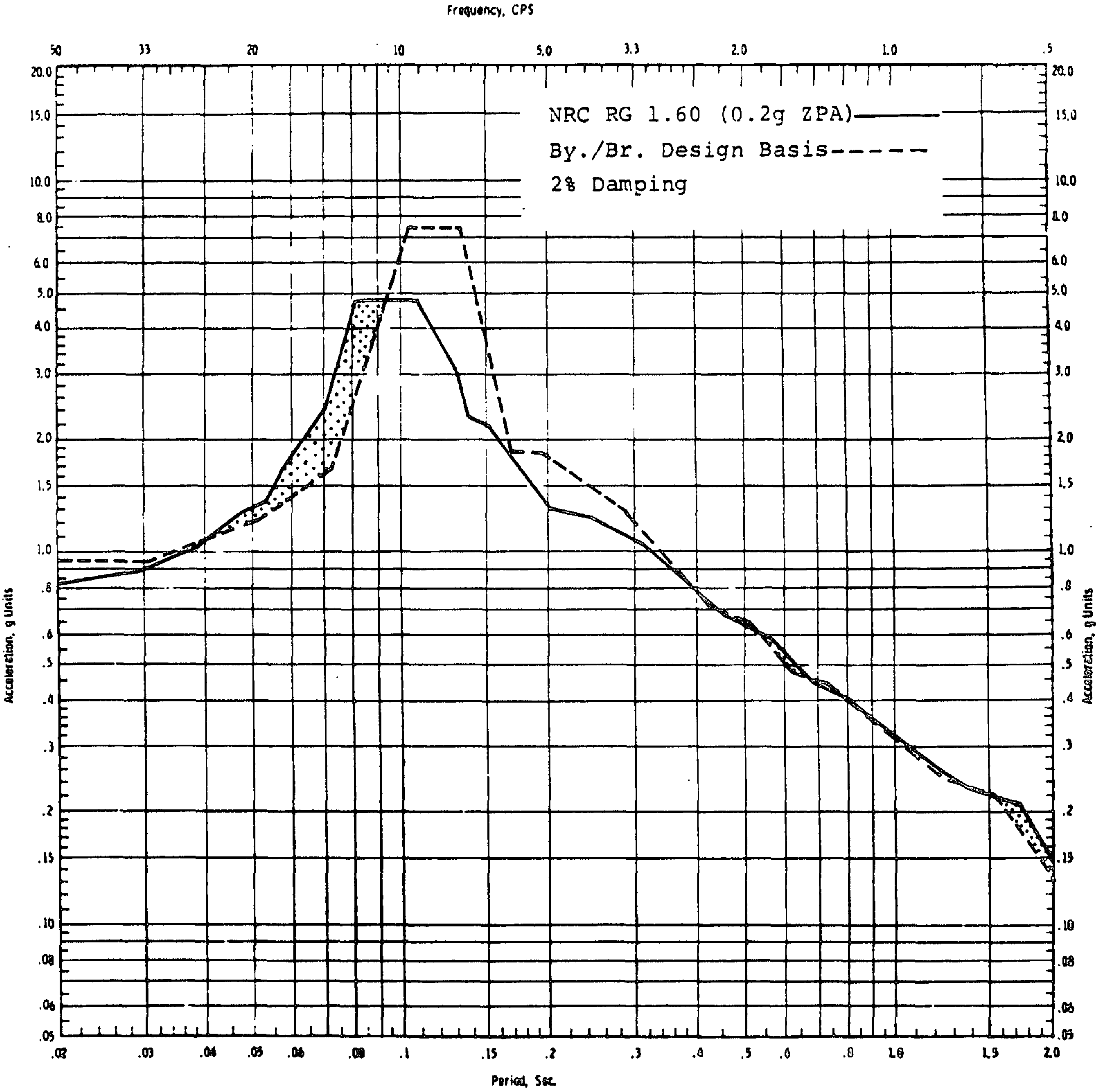

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FIGURE Q130.6-90
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 426'-0"; 439'-0"; 451'-0"

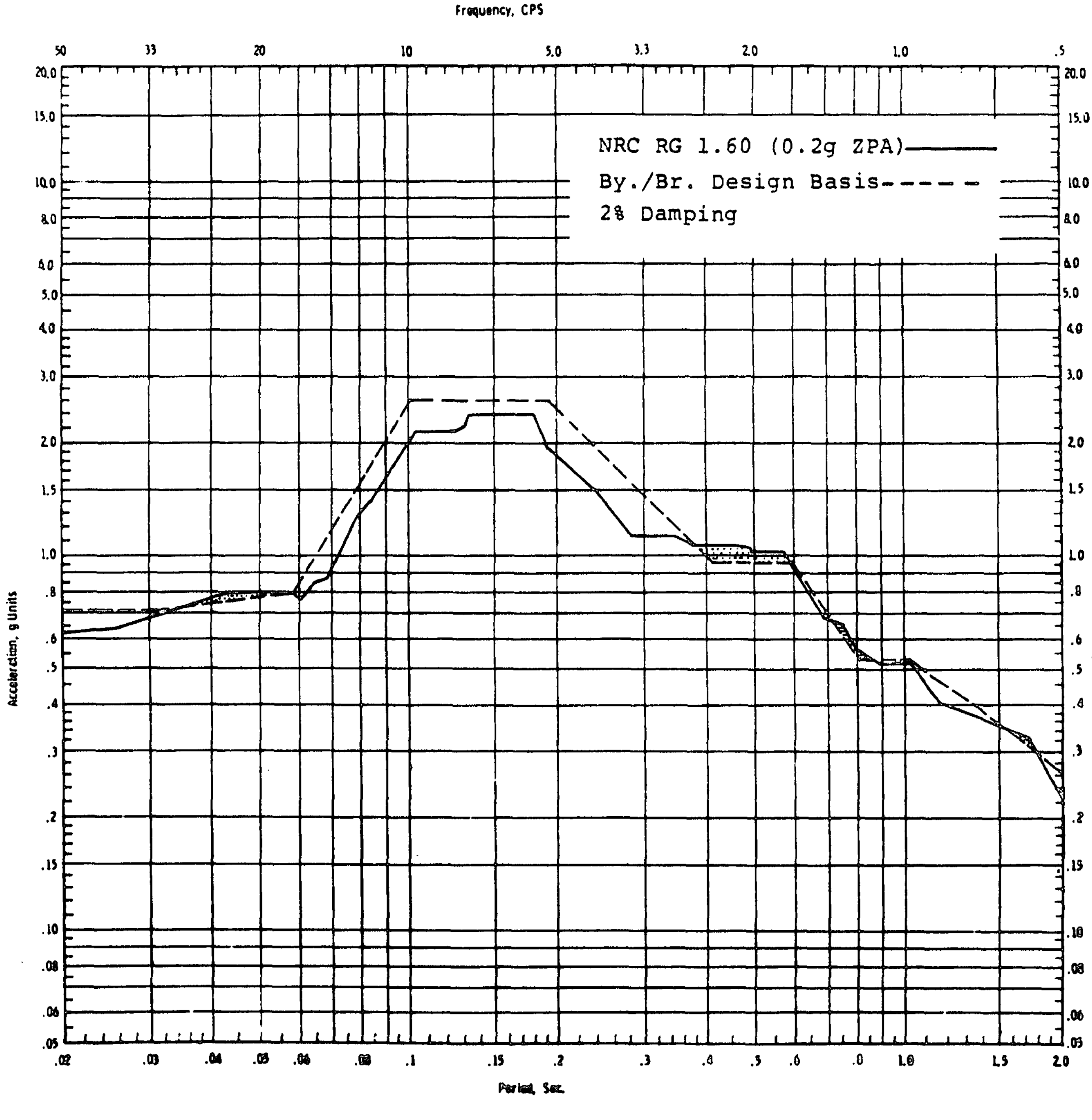

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FIGURE Q130.6-91
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS
LOCATION: AUXILIARY BLDG., TURBINE BLDG.,
HEATER BAY—ELEVATION: 451'-0"

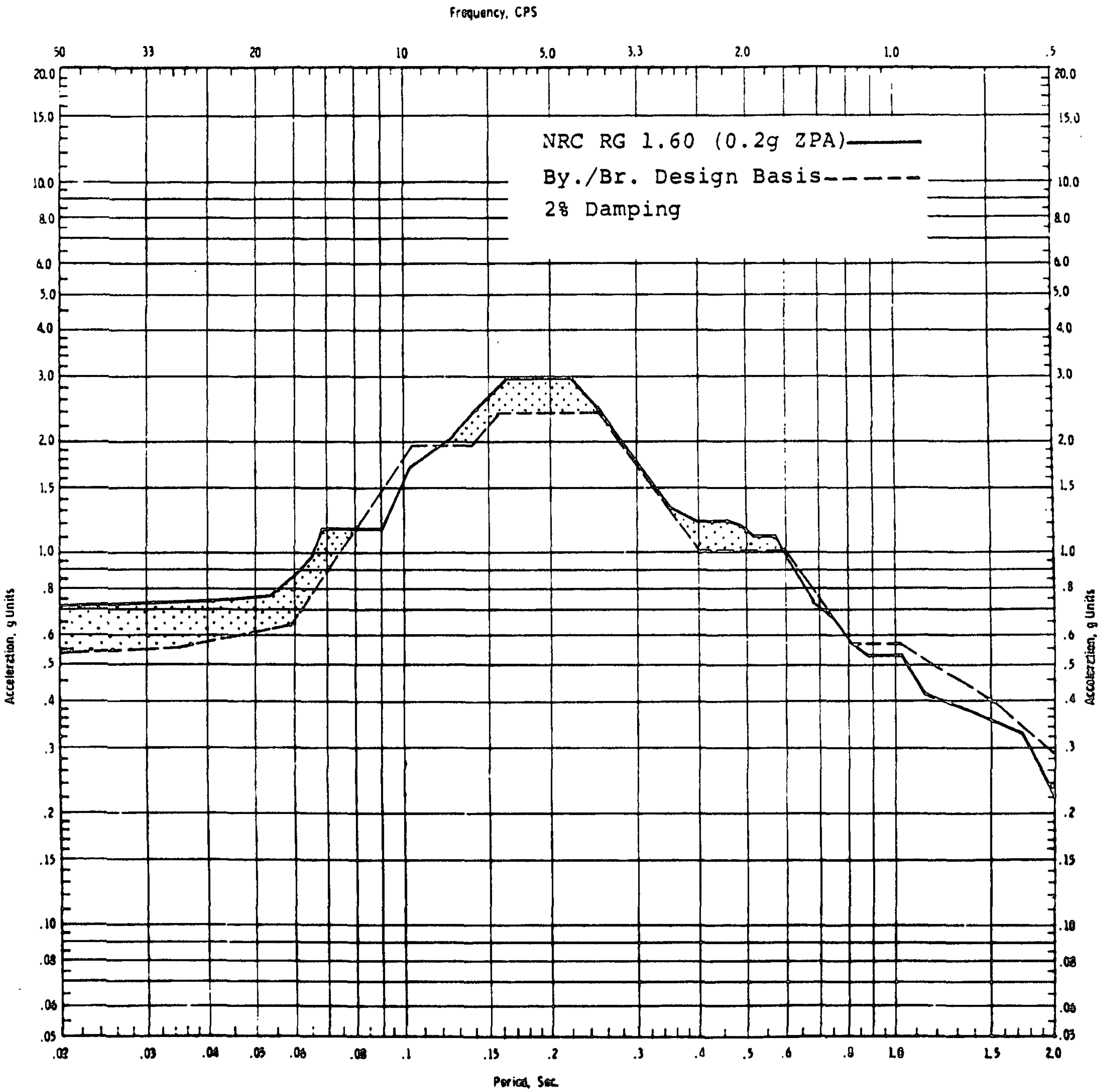

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FIGURE Q130.6-92
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, EW
LOCATION: AUXILIARY BLDG., TURBINE BLDG.,
HEATER BAY—ELEVATION: 451'-0"

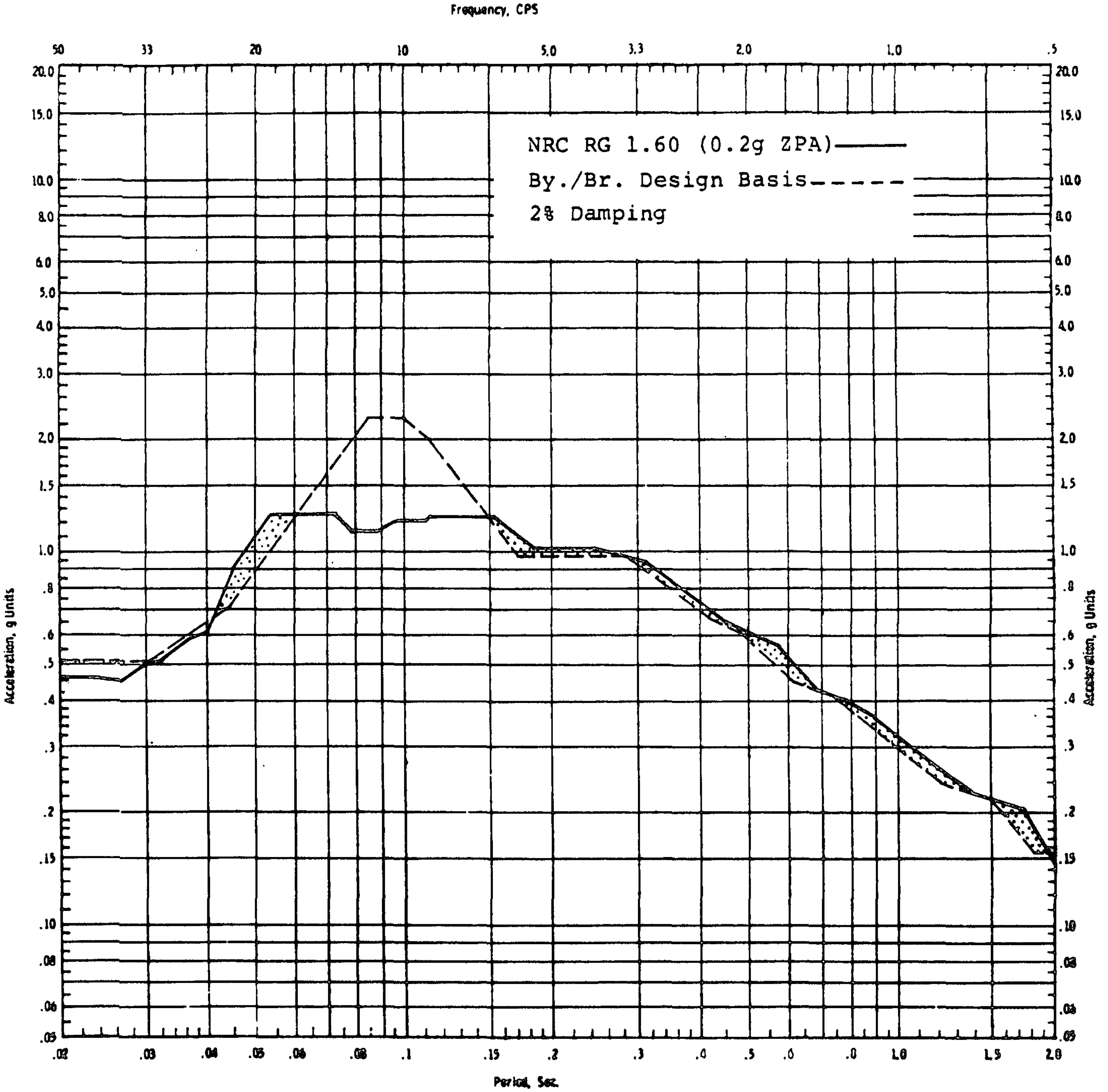

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FIGURE Q130.6-93
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL
LOCATION: AUXILIARY BUILDING WALL
ELEVATION: 467'-0"; 473'-0"; 477'-0"; 485'-0"

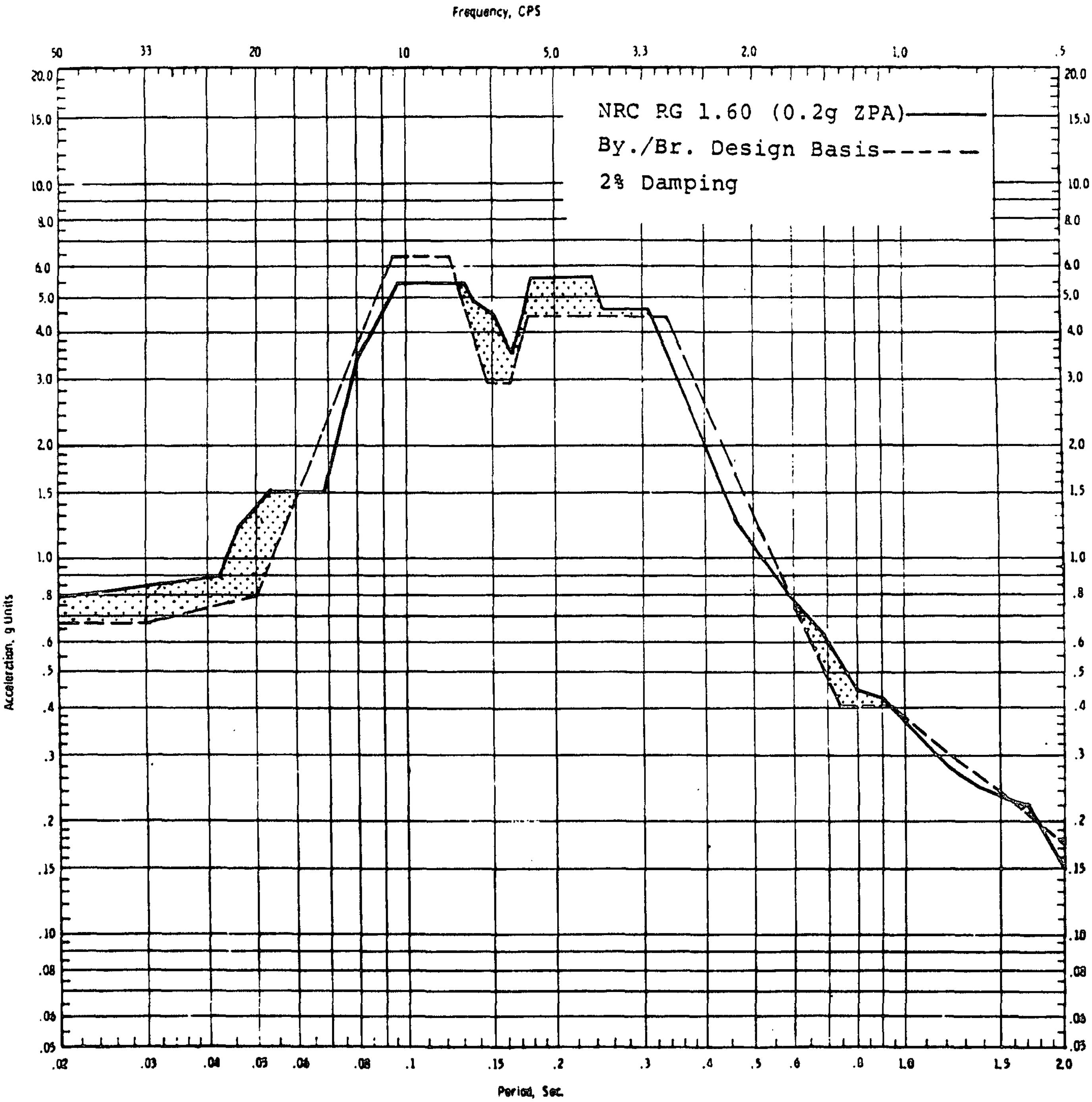

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FIGURE Q130.6-94
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, SLAB
LOCATION: AUXILIARY BUILDING SLAB
ELEVATION: 467'-0"; 477'-0"

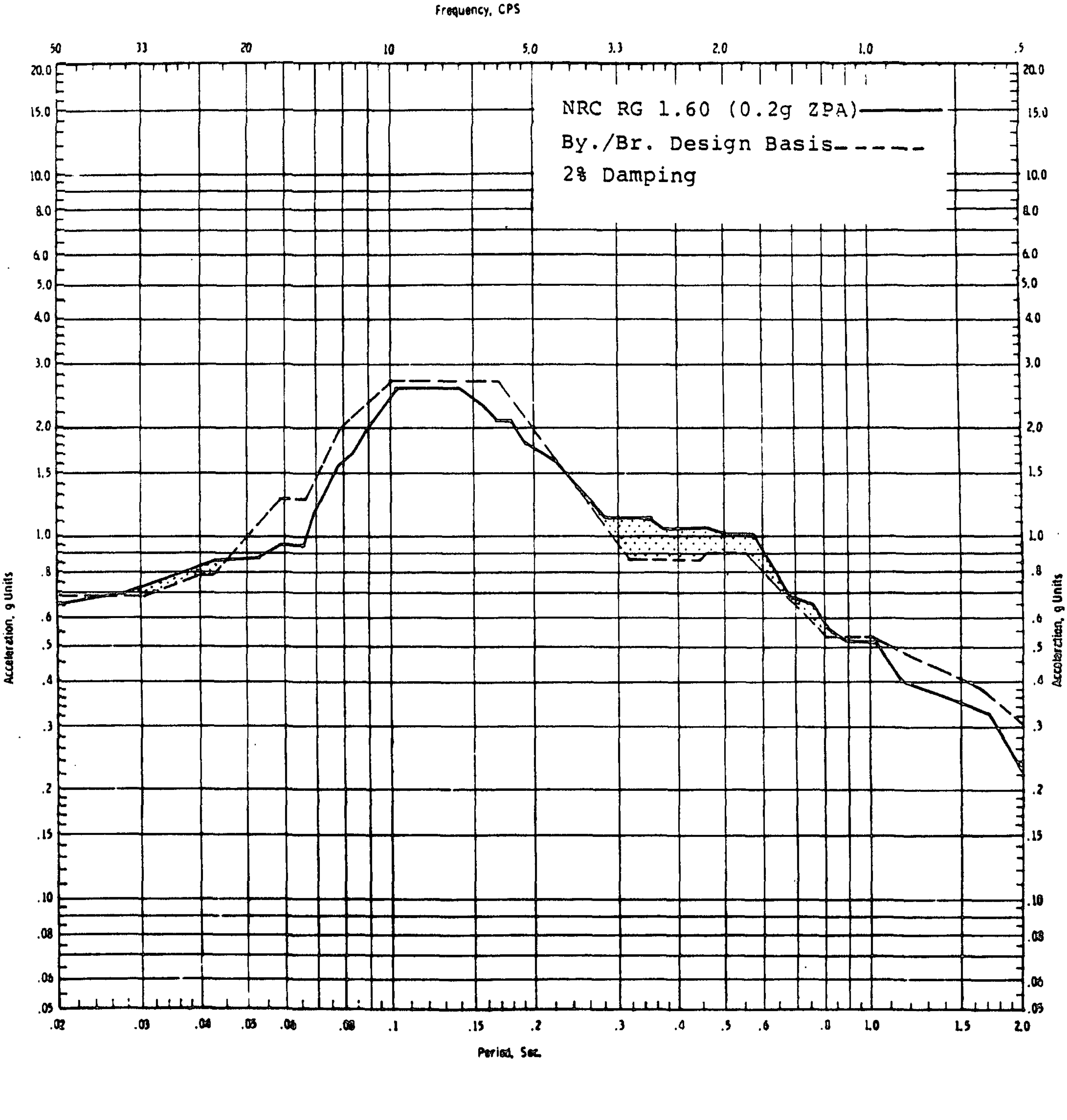

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FIGURE Q130.6-95
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS
LOCATION: AUXILIARY BUILDING
ELEVATION: 477'-0"

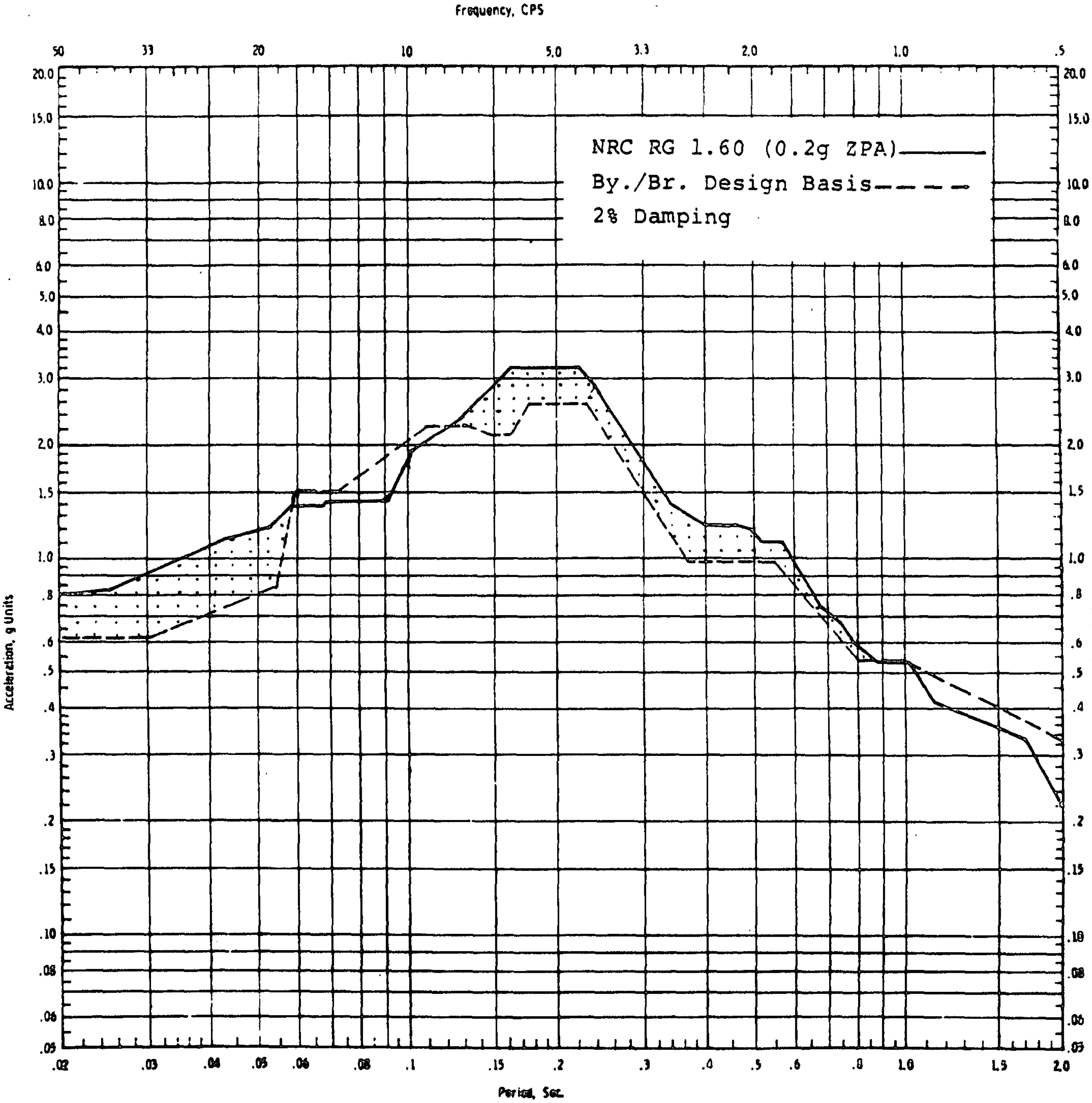

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FIGURE Q130.6-96
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, EW
LOCATION: AUXILIARY BUILDING
ELEVATION: 477'-0"

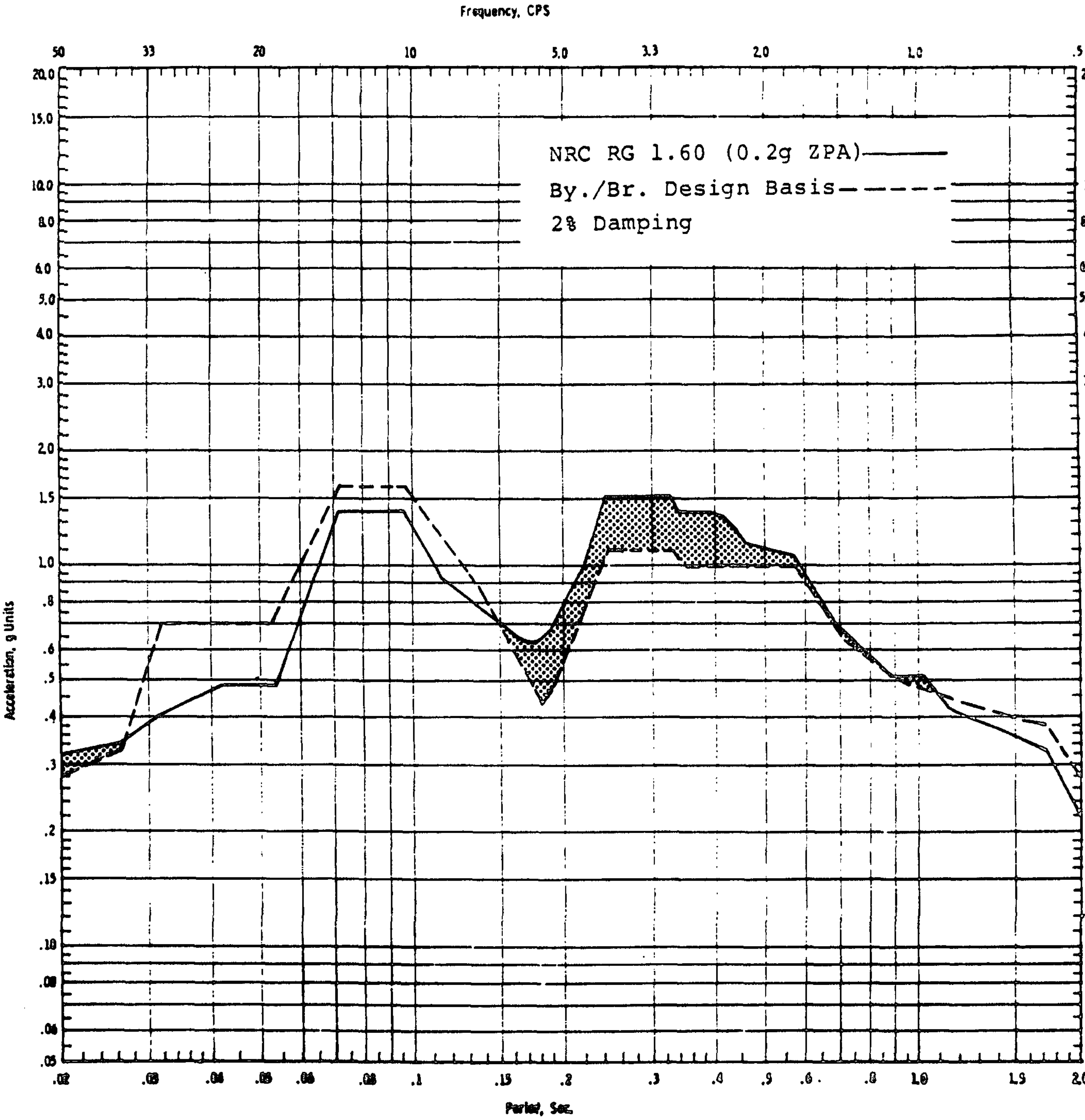

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FIGURE Q130.6-97
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS AND EW
LOCATION: CONTAINMENT BUILDING
ELEVATION: 424'-0"; 436'-0"

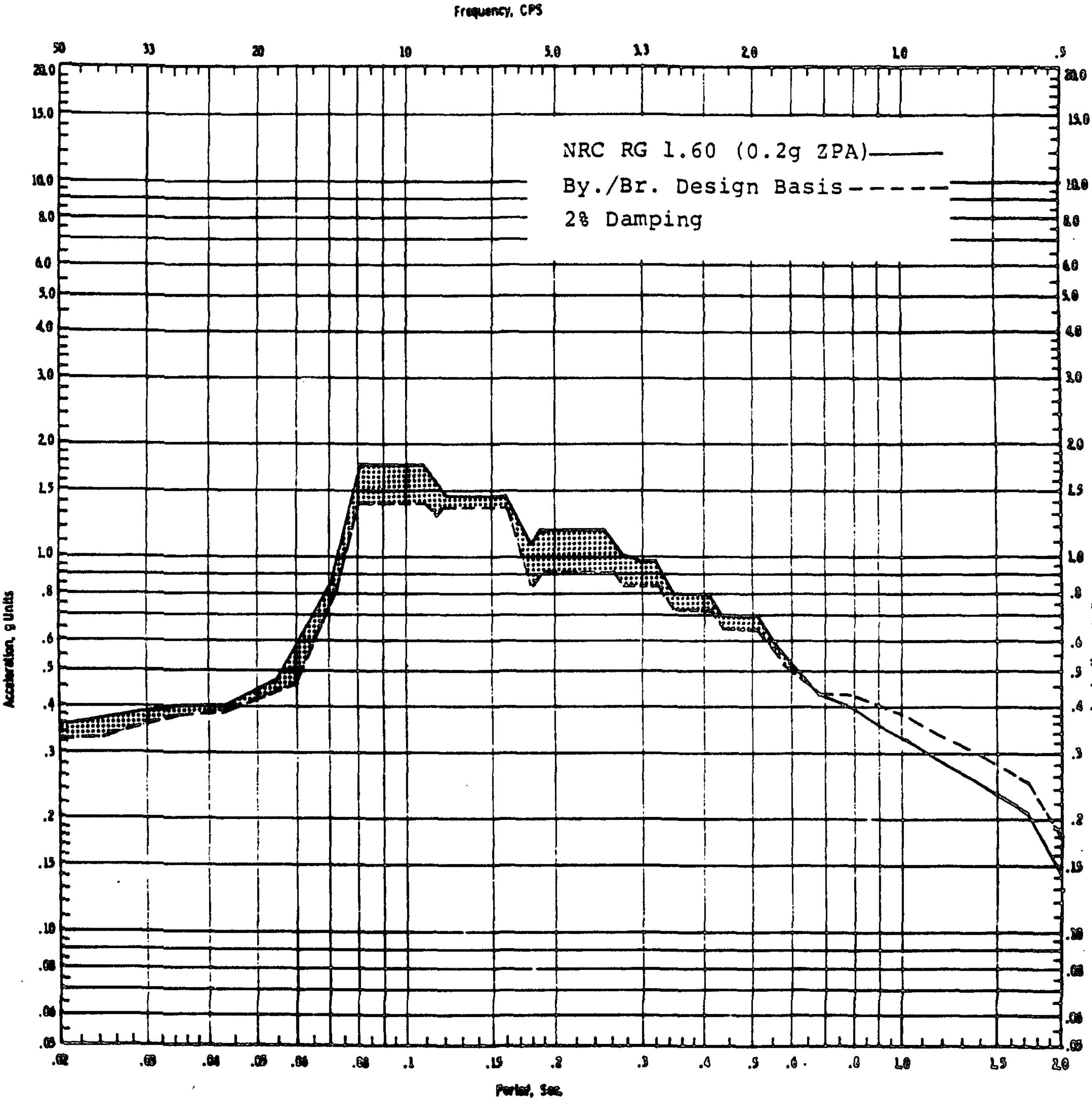

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FIGURE Q130.6-98
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL WALL
LOCATION: CONTAINMENT BUILDING WALL
ELEVATION: 424'-0"; 436'-0"

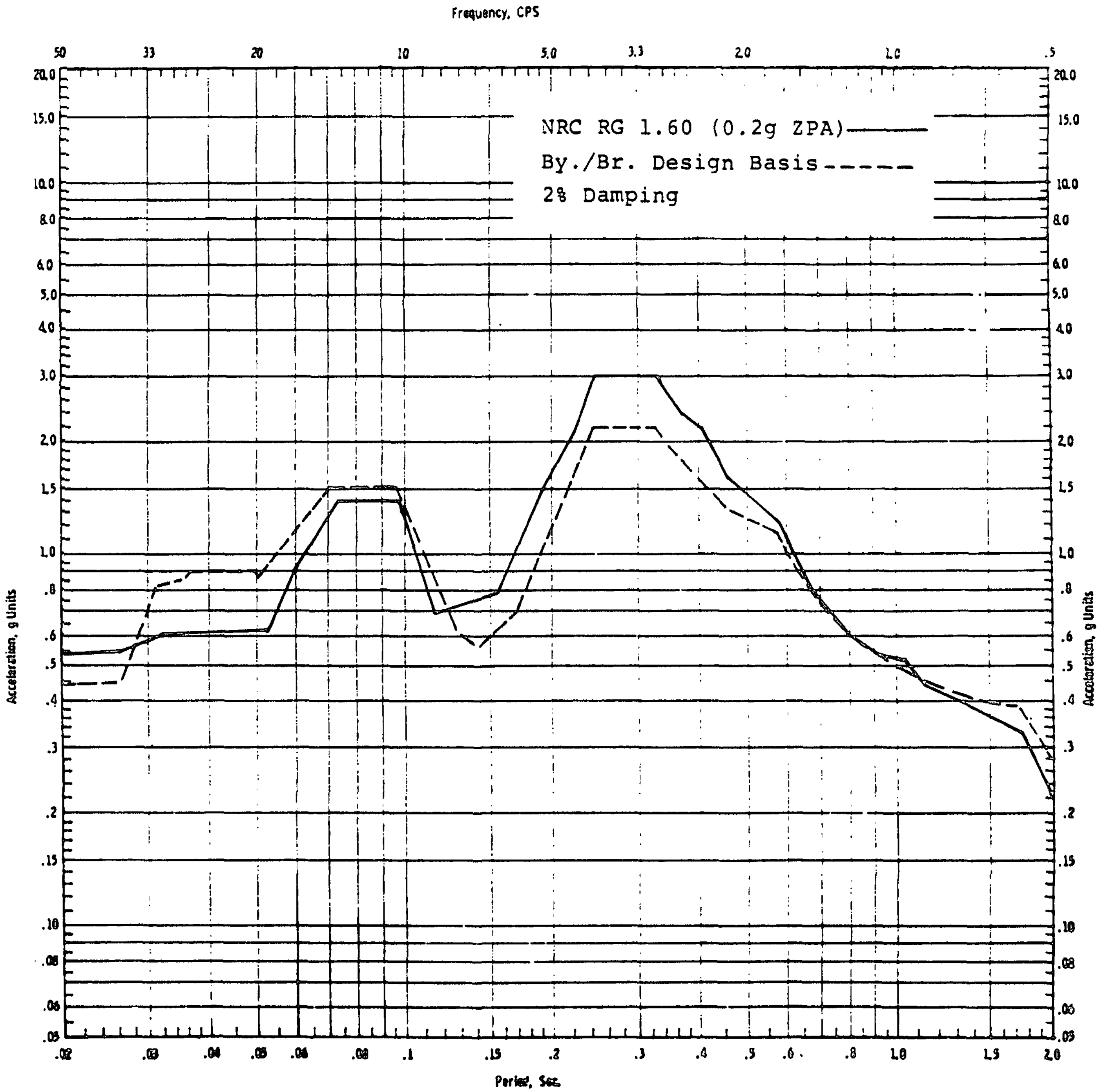

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FIGURE Q130.6-99
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS AND EW
LOCATION: CONTAINMENT BUILDING
ELEVATION: 496'-0"

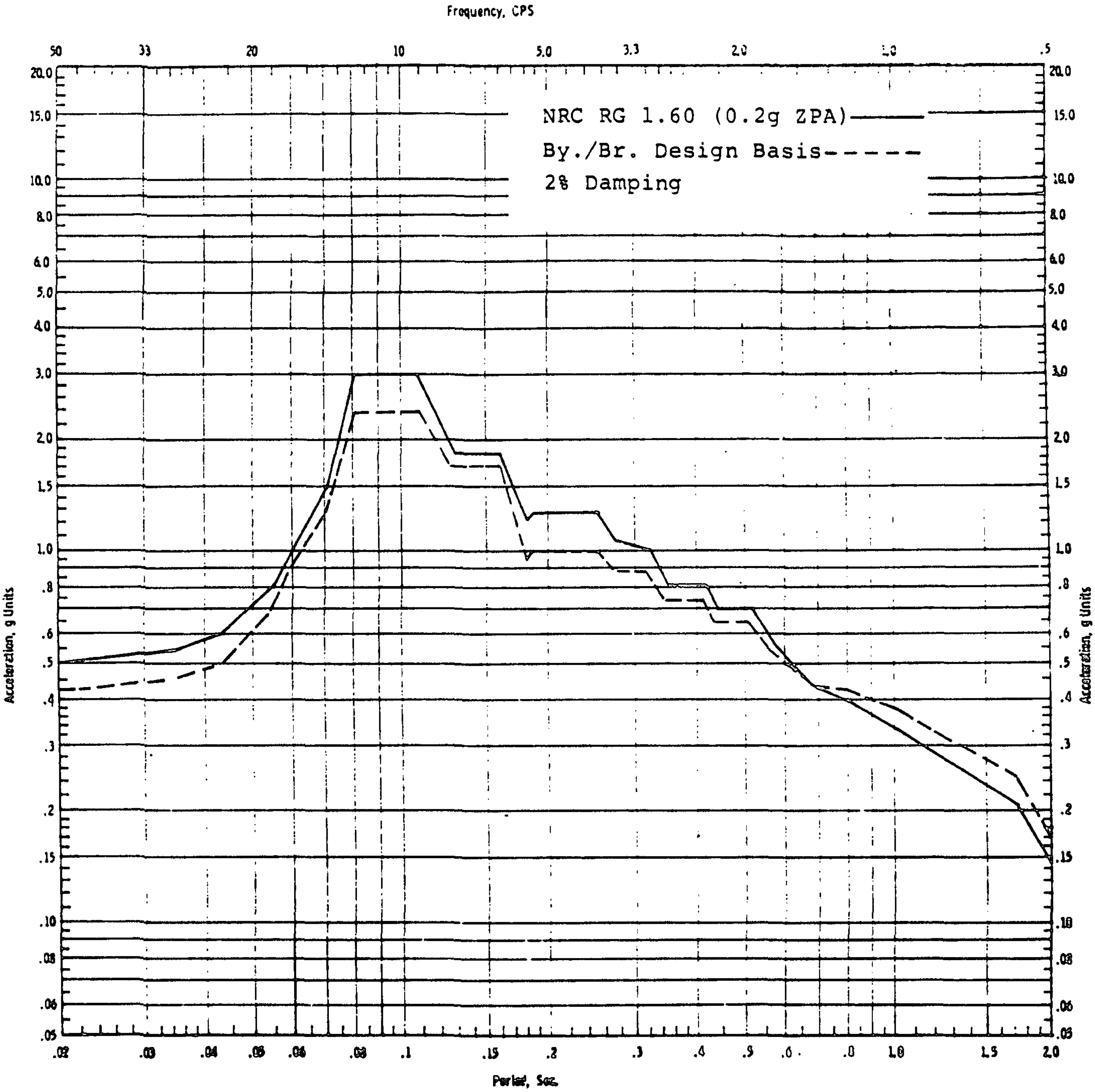

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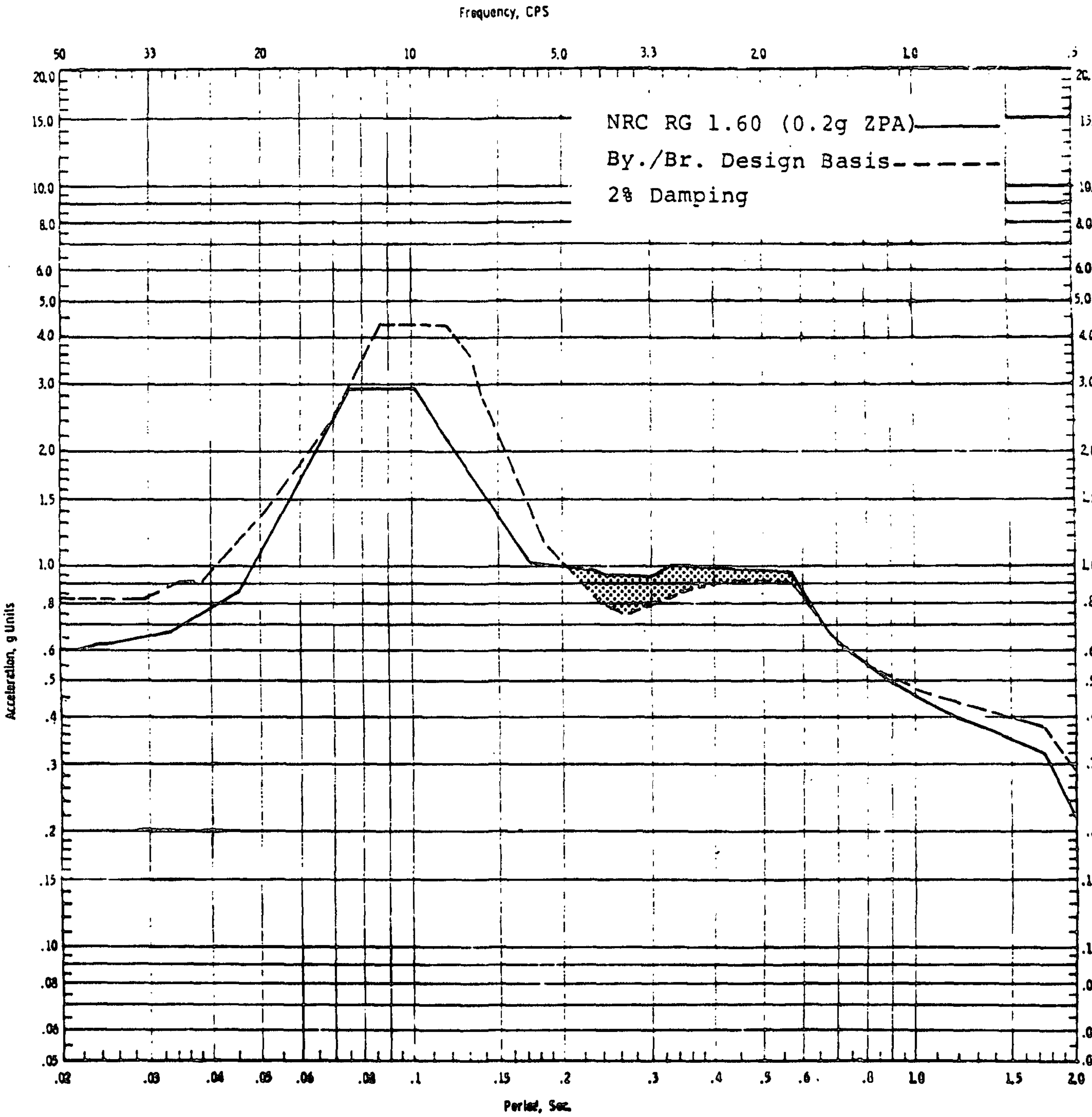
FIGURE Q130.6-100
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL
LOCATION: CONTAINMENT BUILDING WALL
ELEVATION: 496'-0"



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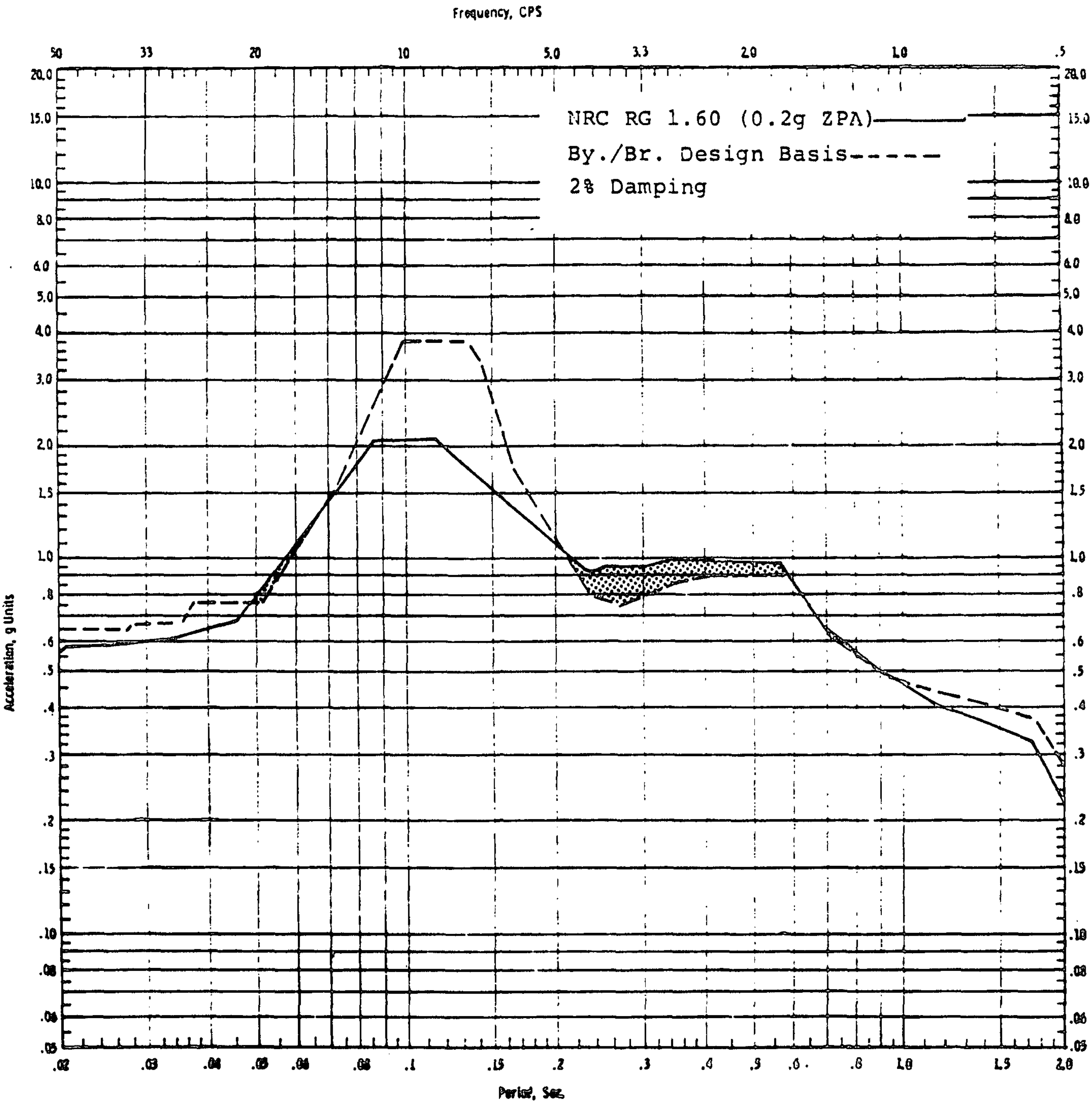
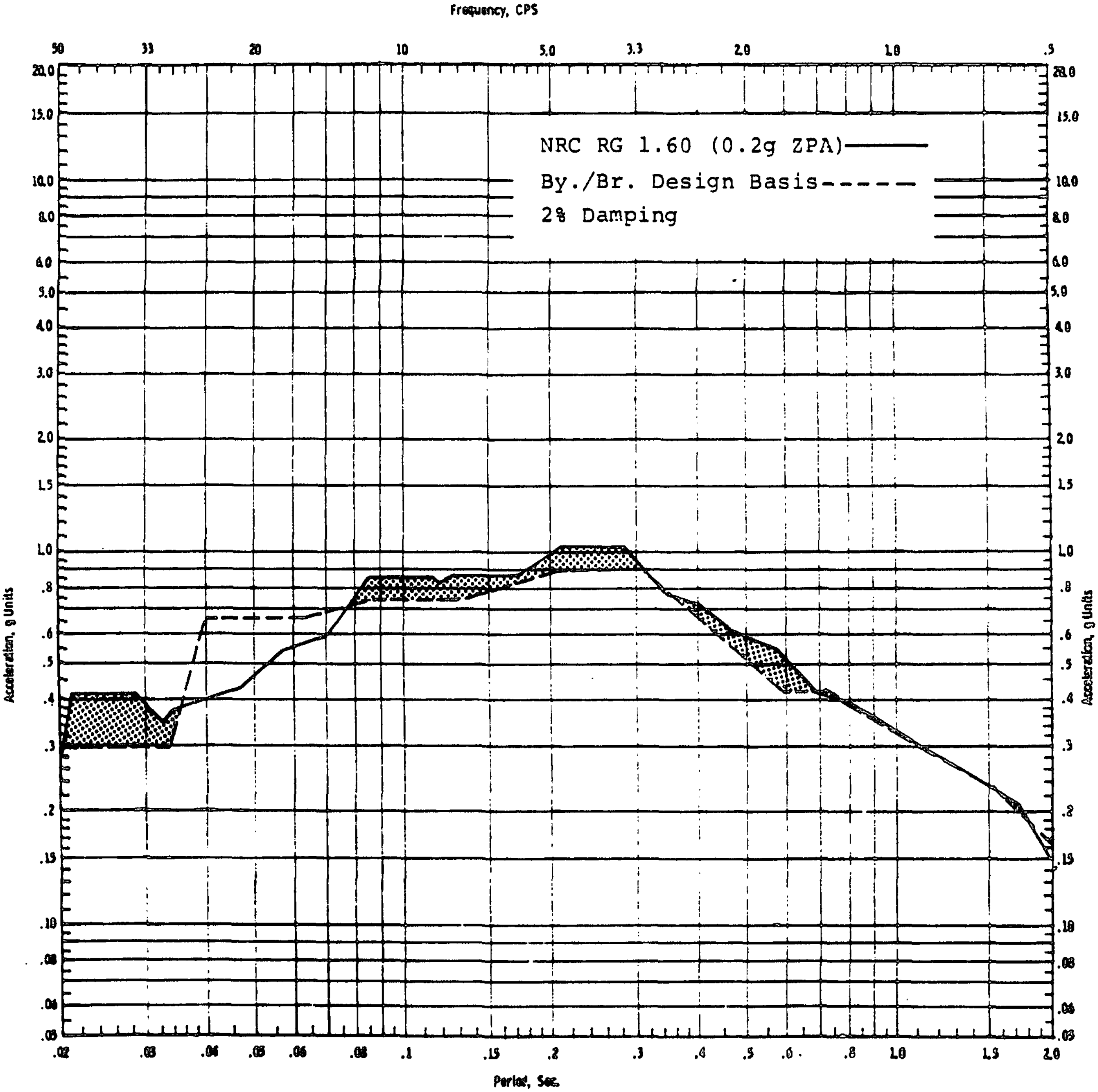
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6-101
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, NS
LOCATION: CONTAINMENT INNER STRUCTURE
ELEVATION: 426'-0"



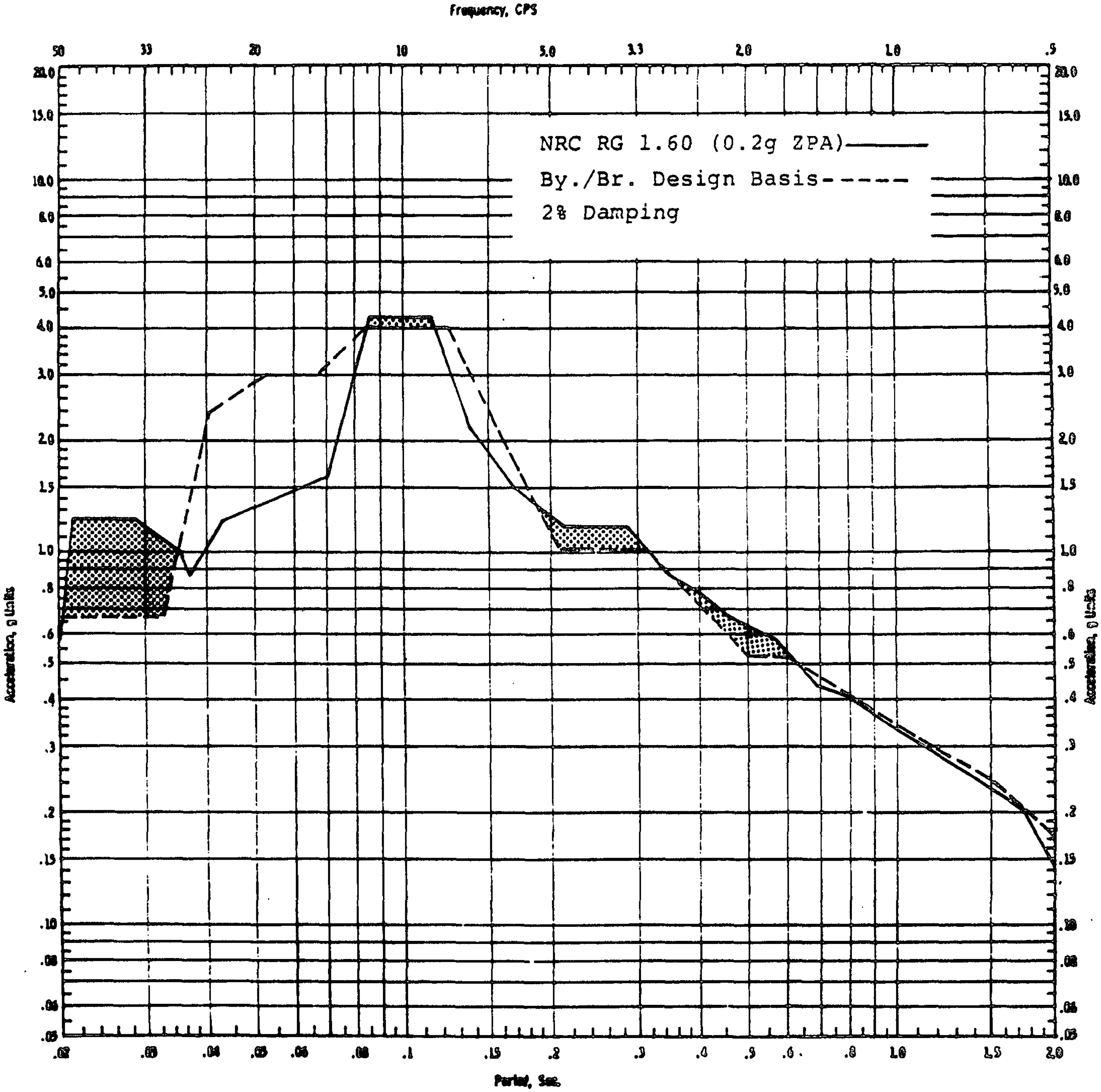
BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6-102
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, HORIZONTAL, EW
LOCATION: CONTAINMENT INNER STRUCTURE
ELEVATION: 426'-0"



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6-103
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, WALL
LOCATION: CONTAINMENT INNER STRUCTURE WALL
ELEVATION: 412'-0"; 426'-0"



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6-104
COMPARISON OF B/B AND RG 1.60 SPECTRA
EXCITATION: SSE, VERTICAL, SLAB
LOCATION: CONTAINMENT INNER STRUCTURE SLAB
ELEVATION: 390'-0"; 401'-0"; 412'-0"; 426'-0"

Security - Related Information Figure Withheld Under 10 CFR 2.390

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-1

CONTAINMENT BUILDING

Security - Related Information Figure Withheld Under 10 CFR 2.390

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-2

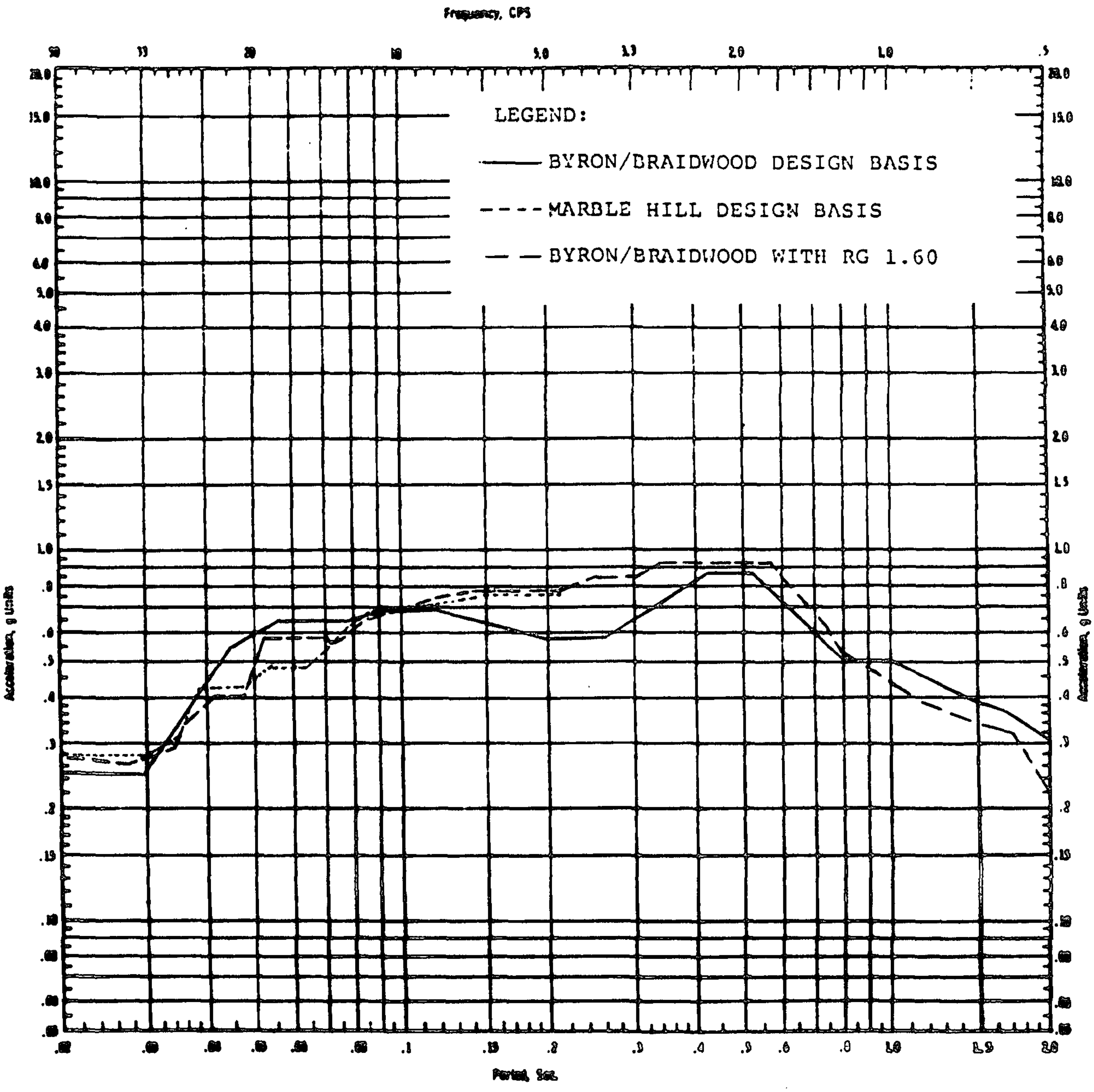
AUXILIARY AND FUEL HANDLING
BUILDINGS

Security - Related Information Figure Withheld Under 10 CFR 2.390

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-3

MAT PLAN AUXILIARY AND FUEL HANDLING BUILDING COMPLEX



EXCITATION HORIZ : 2% DAMPING

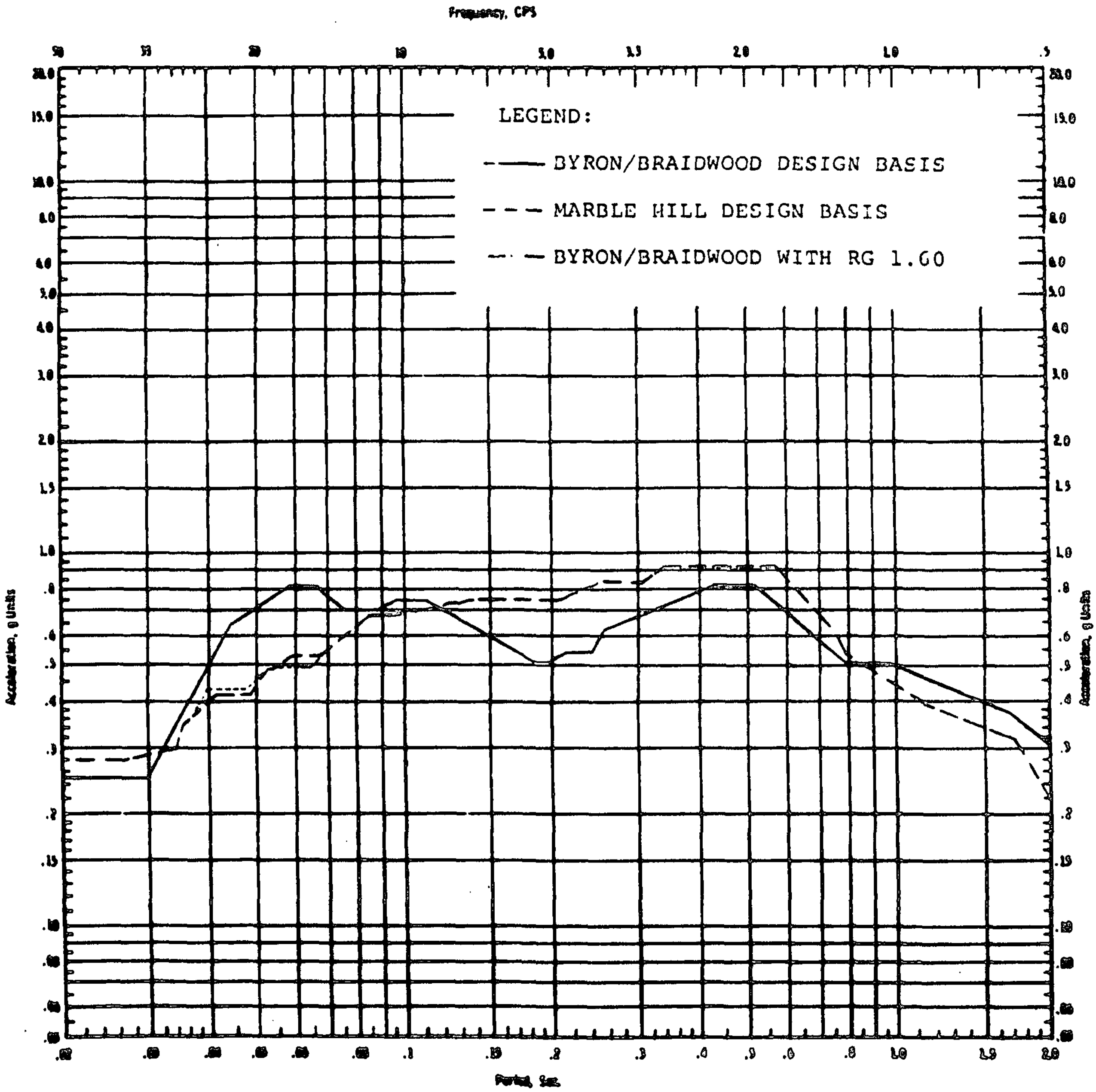
LOCATION: AUX

SPECTRA NO. 101-SS-NS

ELEVATION: 346'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-4
SPECTRA 101-SS-NS COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 346'-0"



EXCITATION HORIZ : 2% DAMPING

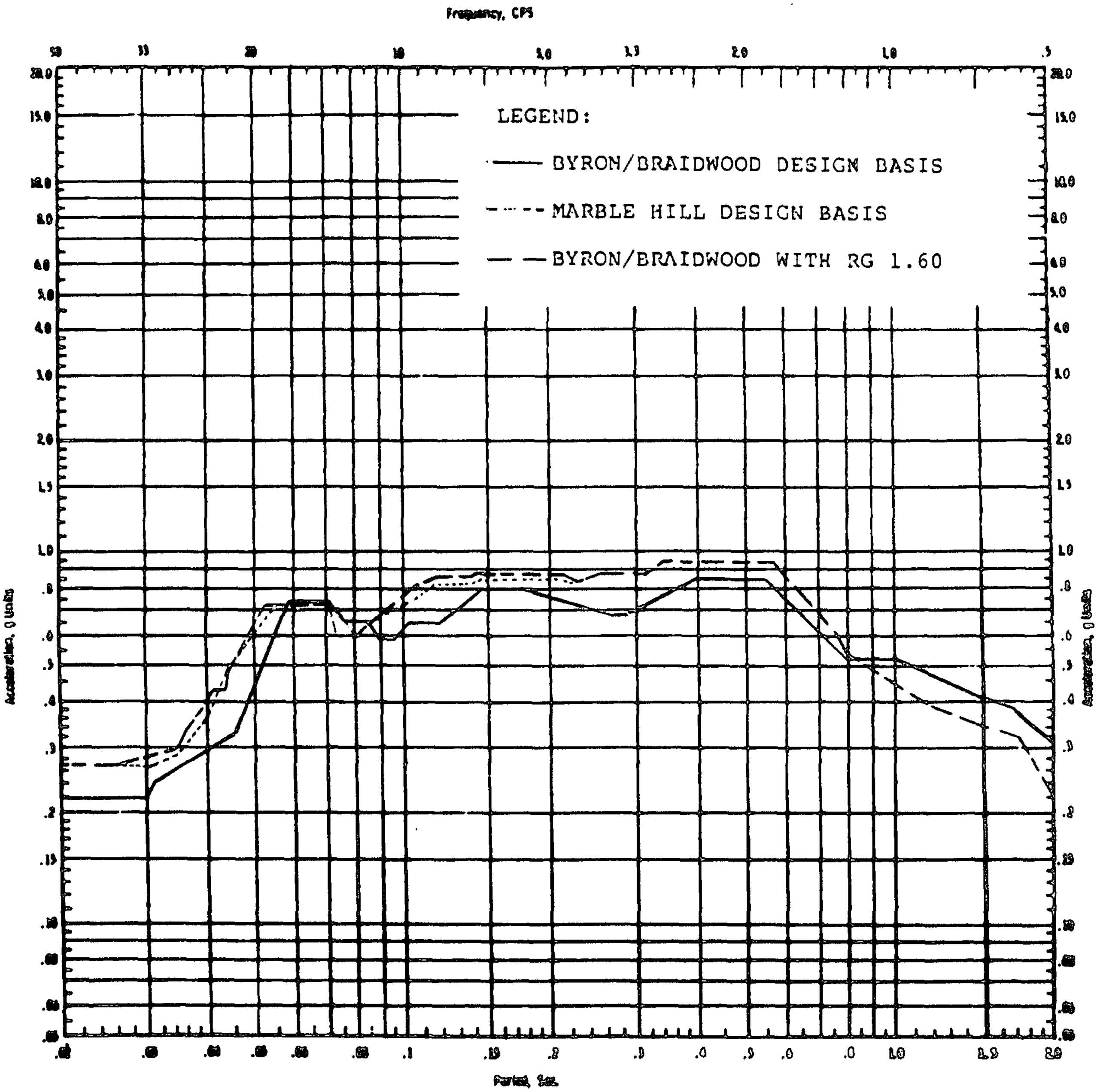
LOCATION: AUX

SPECTRA NO 101-SS-EW

ELEVATION: 346'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-5
SPECTRA 101-SS-EW COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 346'-0"



EXCITATION HORIZ : 2% DAMPING

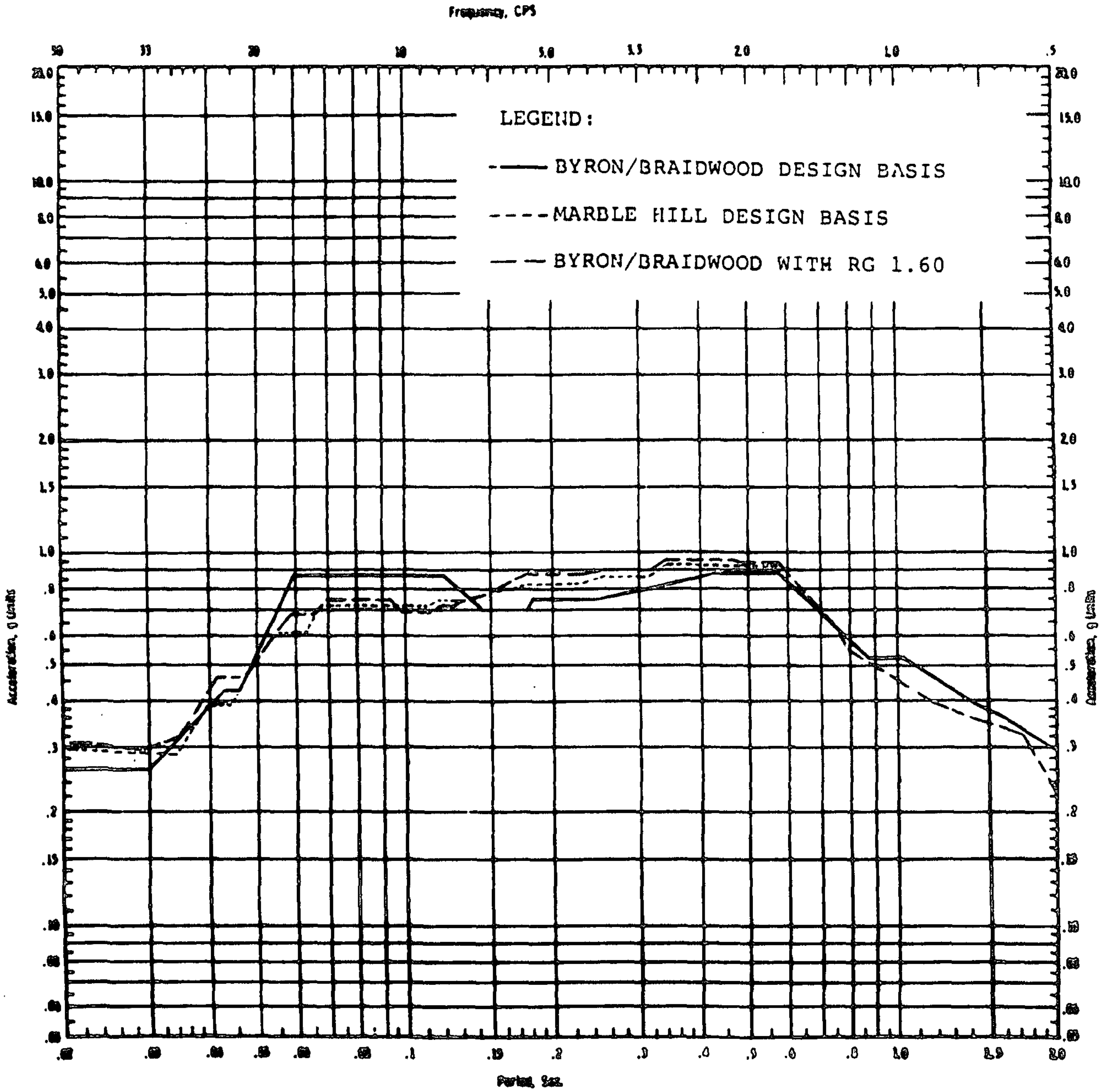
LOCATION: AUX

SPECTRA NO 102-SS-NS

ELEVATION: 364'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-6
SPECTRA 102-SS-NS COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 364'-0"



EXCITATION HORIZ : 2% DAMPING

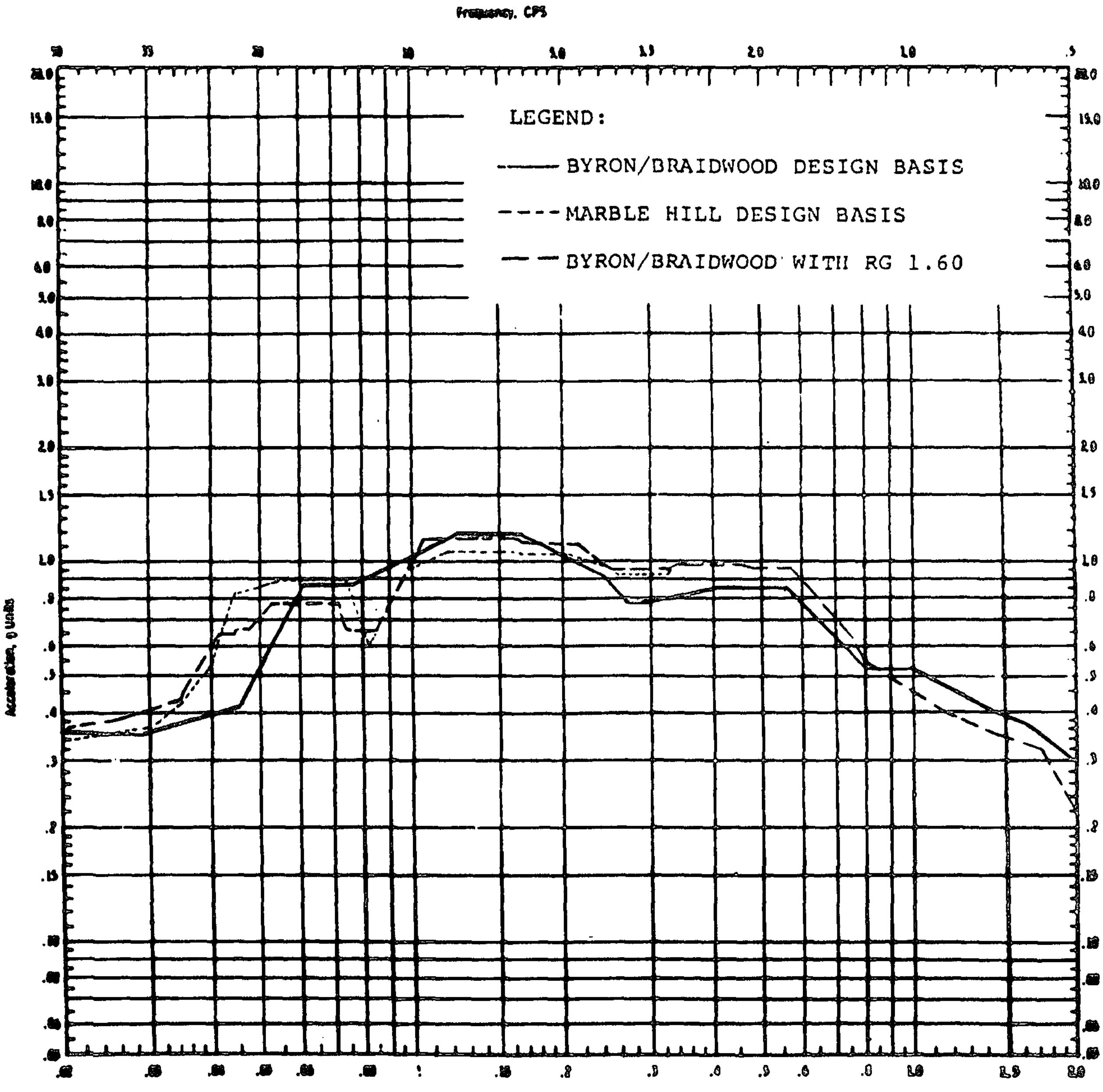
LOCATION: AUX

SPECTRA NO 102-SS-EW

ELEVATION: 364'0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-7
SPECTRA 102-SS-EW COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 364'-0"



EXCITATION HORIZ: 2% DAMPING

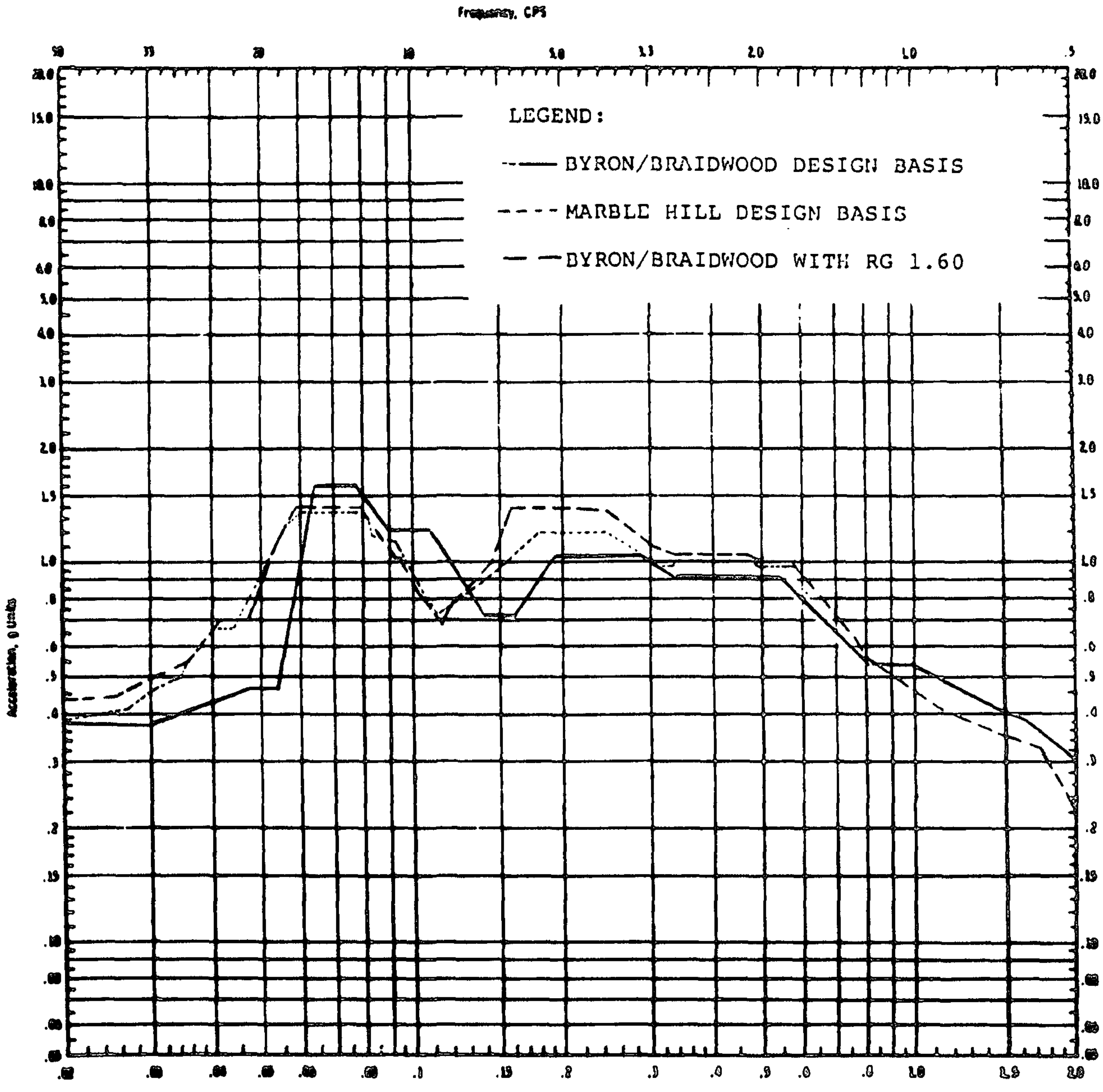
LOCATION: AUX-TUB-RAD

SPECTRA NO. 107-SS-NS

ELEVATION: 401'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-8
SPECTRA 107-SS-NS COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY, TURBINE, AND RADWASTE BUILDINGS—ELEVATION: 401'-0"



EXCITATION HORIZ : 2% DAMPING

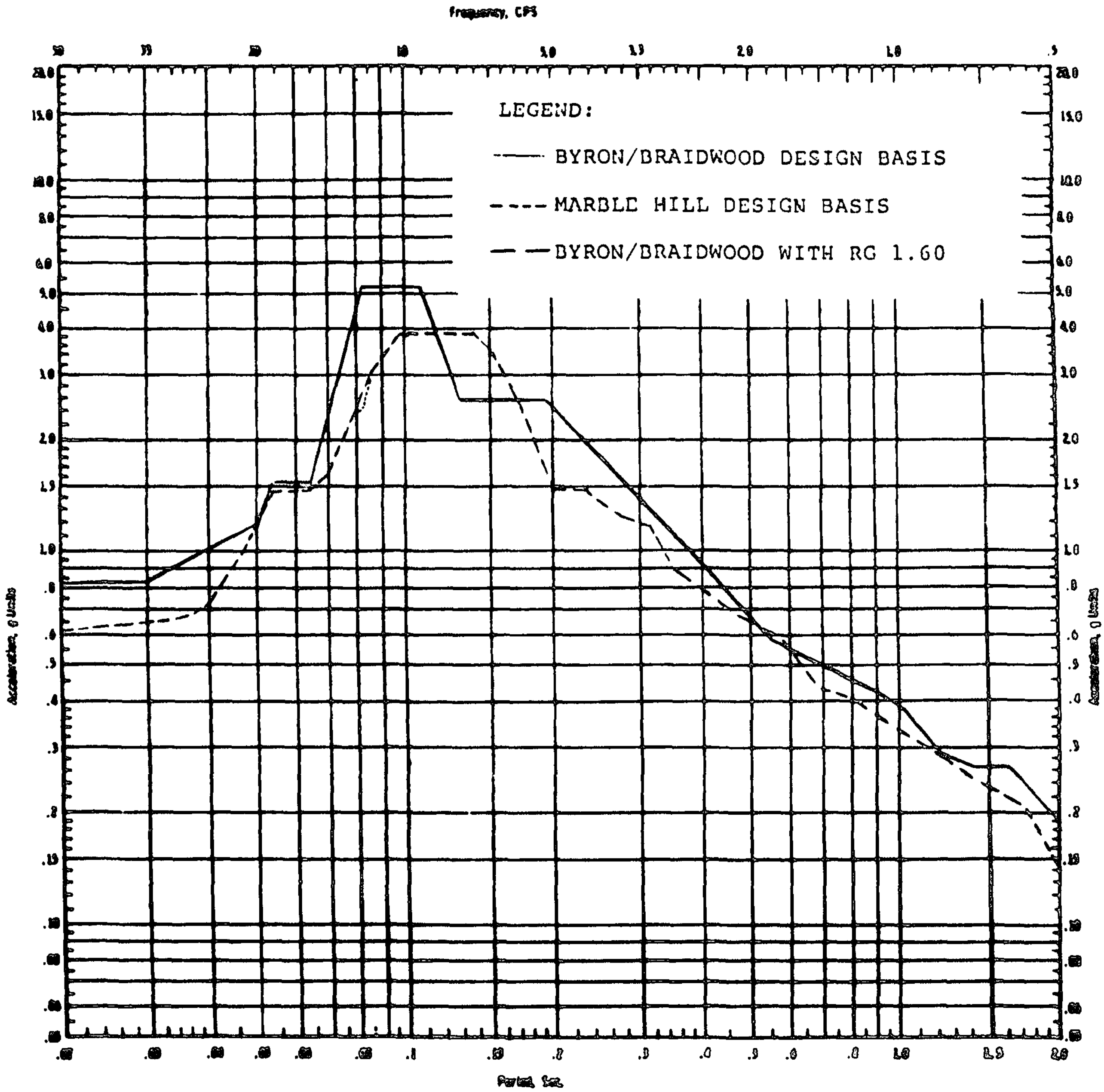
LOCATION: AUX - TUR - RAD

SPECTRA NO 107-SS-EW

ELEVATION: 401'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-9
SPECTRA 107-SS-EW COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY, TURBINE, AND RADWASTE
BUILDINGS—ELEVATION: 401'-0"



EXCITATION: VERT : 2% DAMPING

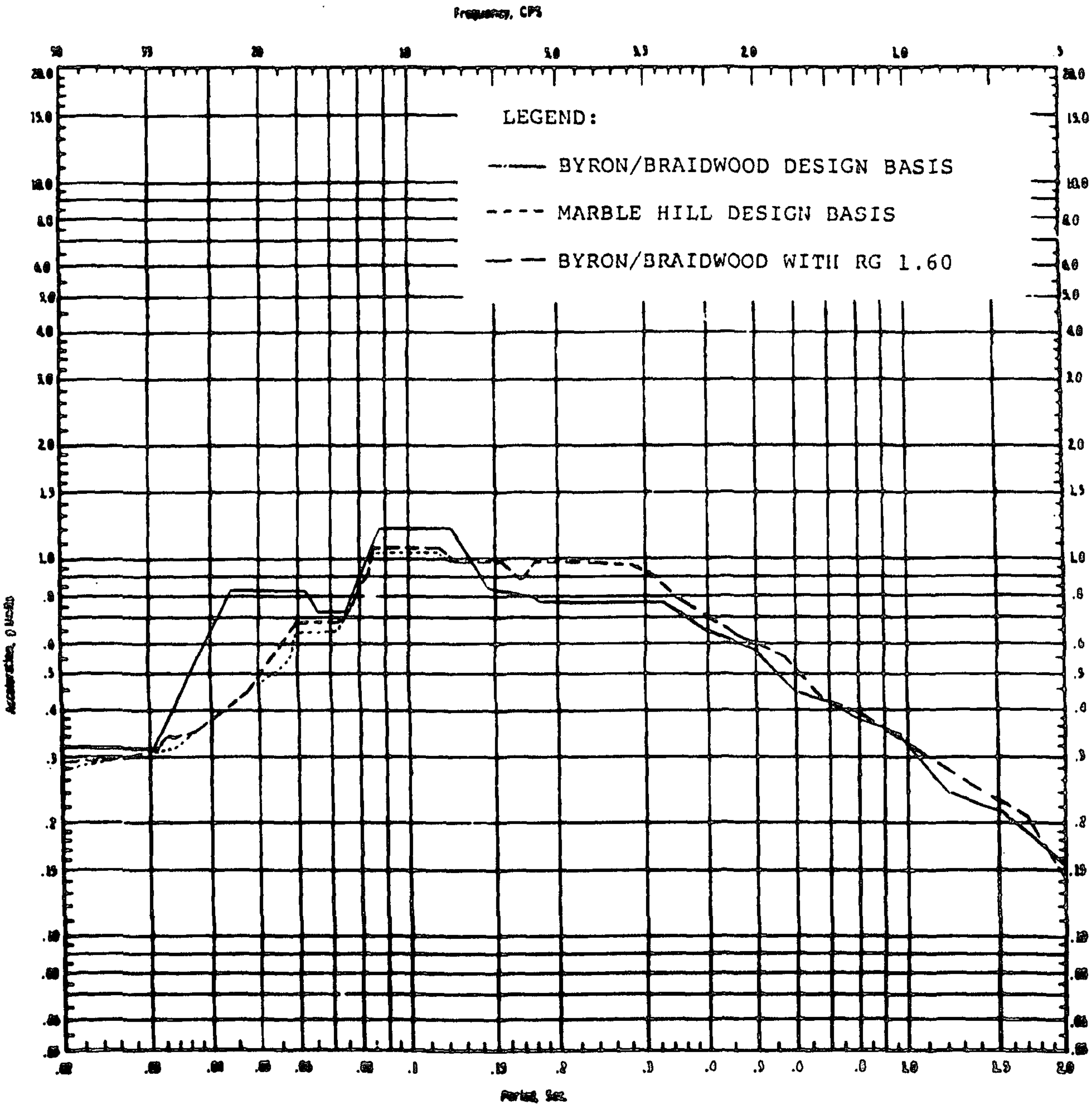
LOCATION: AUX - FHB SLAB

SPECTRA NO 107-SS-VS

ELEVATION: 401'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-10
SPECTRA 107-SS-VS COMPARISON
EXCITATION: VERTICAL, 2% DAMPING
LOCATION: AUXILIARY—FUEL HANDLING BUILDING
SLAB—ELEVATION: 401'-0"



EXCITATION VERT : 2% DAMPING

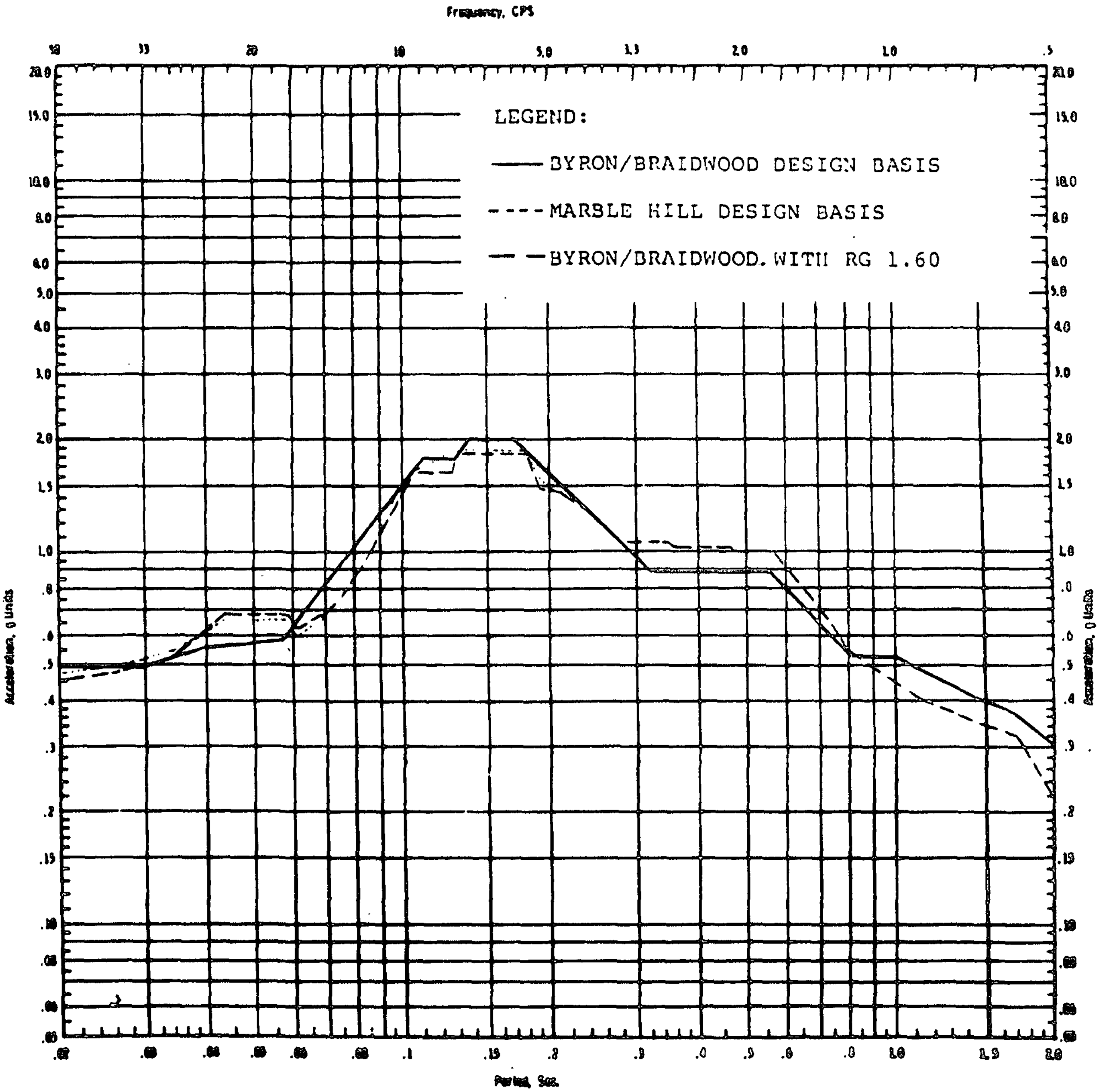
LOCATION: AUX - FHB WALL

SPECTRA NO. 107-SS-VW

ELEVATION: 401'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-11
SPECTRA 107-SS-VW COMPARISON
EXCITATION: VERTICAL, 2% DAMPING
LOCATION: AUXILIARY—FUEL HANDLING BUILDING
WALL—ELEVATION: 401'-0"



EXCITATION HORIZ : 2% DAMPING

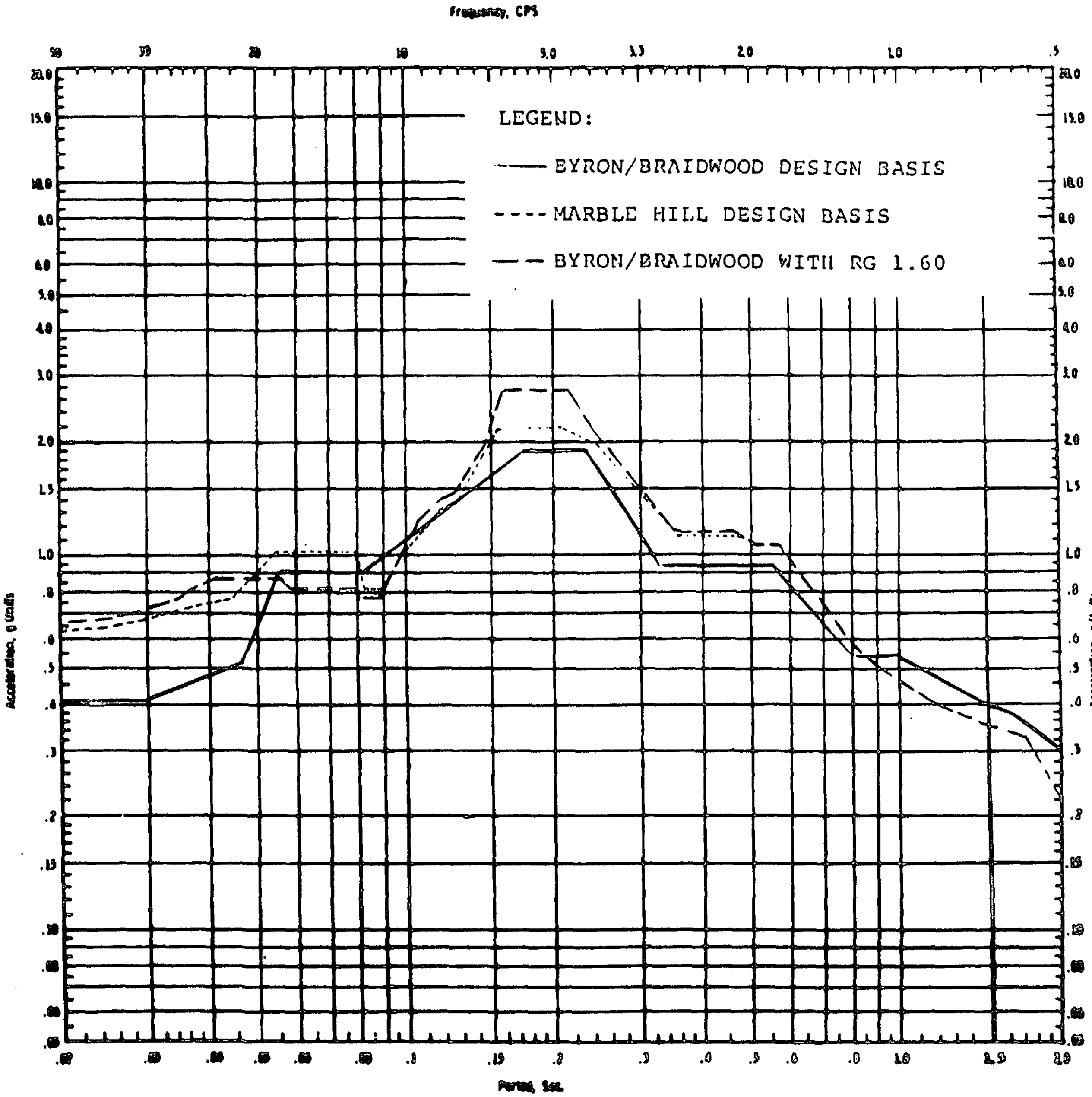
LOCATION: AUX-FHB-TUR

SPECTRA NO 108-SS-NS

ELEVATION: 426'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-12
SPECTRA 108-SS-NS COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY, FUEL HANDLING, AND
TURBINE BUILDINGS—ELEVATION: 426'-0"



EXCITATION HORIZ : 2% DAMPING

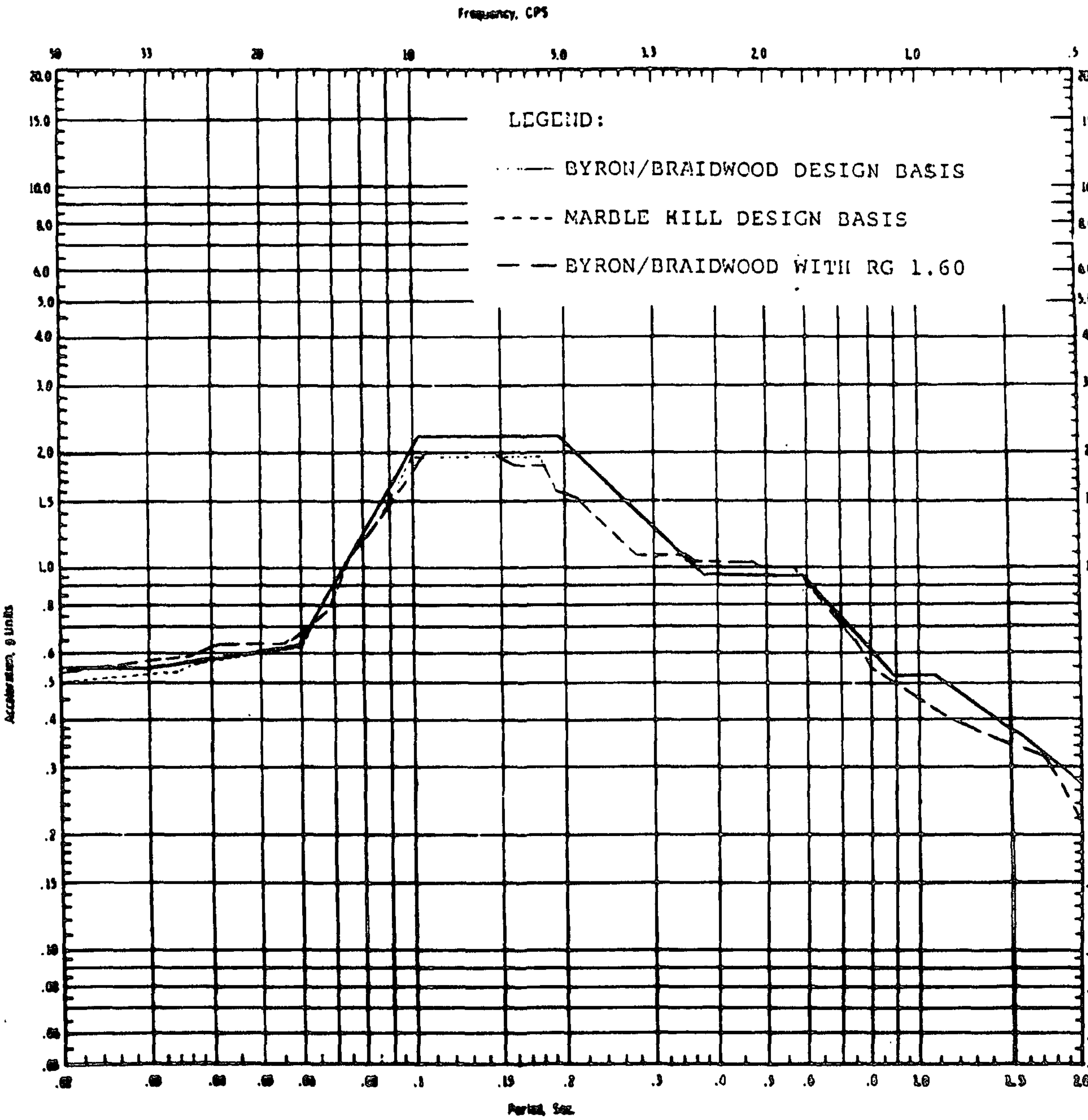
LOCATION: AUX - FHB - TUR

SPECTRA NO 108-SS-EW

ELEVATION: 426'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-13
SPECTRA 108-SS-EW COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY, FUEL HANDLING, AND
TURBINE BUILDINGS—ELEVATION: 426'-0"



EXCITATION HORIZ : 2% DAMPING

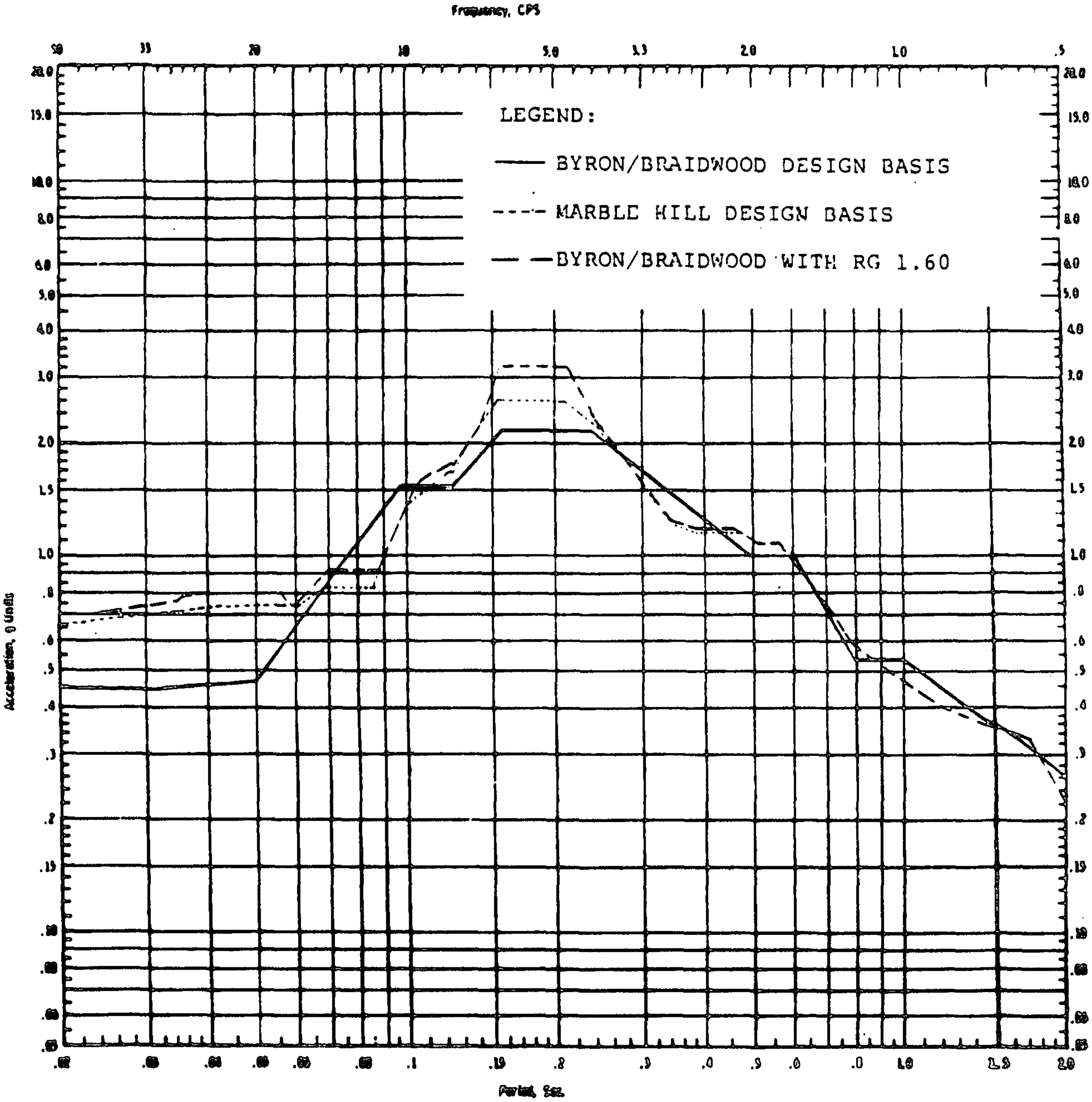
LOCATION: AUX

SPECTRA NO 109-SS-NS

ELEVATION: 439'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-14
SPECTRA 109-SS-NS COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 439'-0"



EXCITATION HORIZ: 2% DAMPING

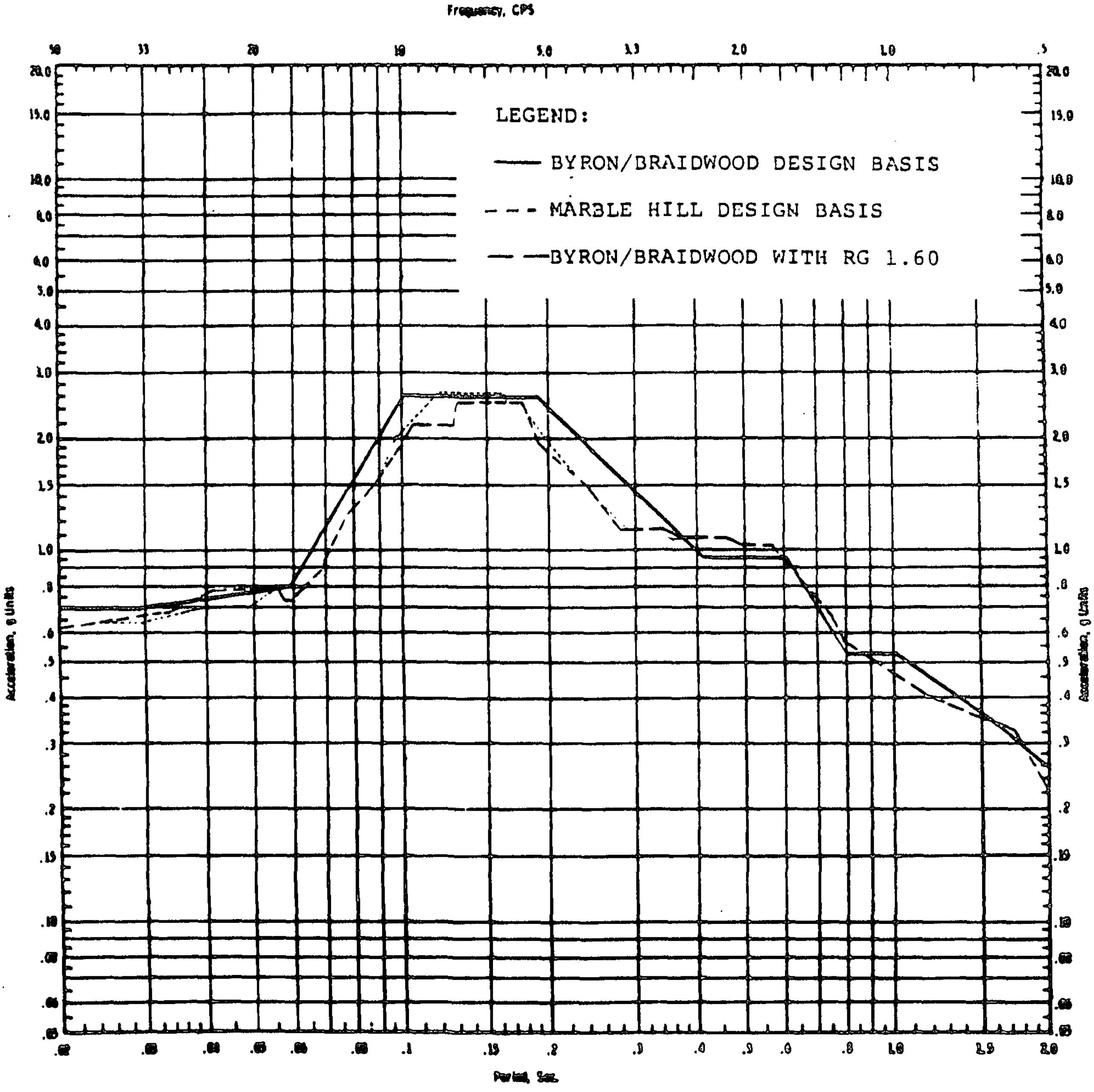
LOCATION: AUX

SPECTRA NO 109-SS-EW

ELEVATION: 439'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-15
SPECTRA 109-SS-EW COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 439'-0"



EXCITATION HORIZ : 2% DAMPING

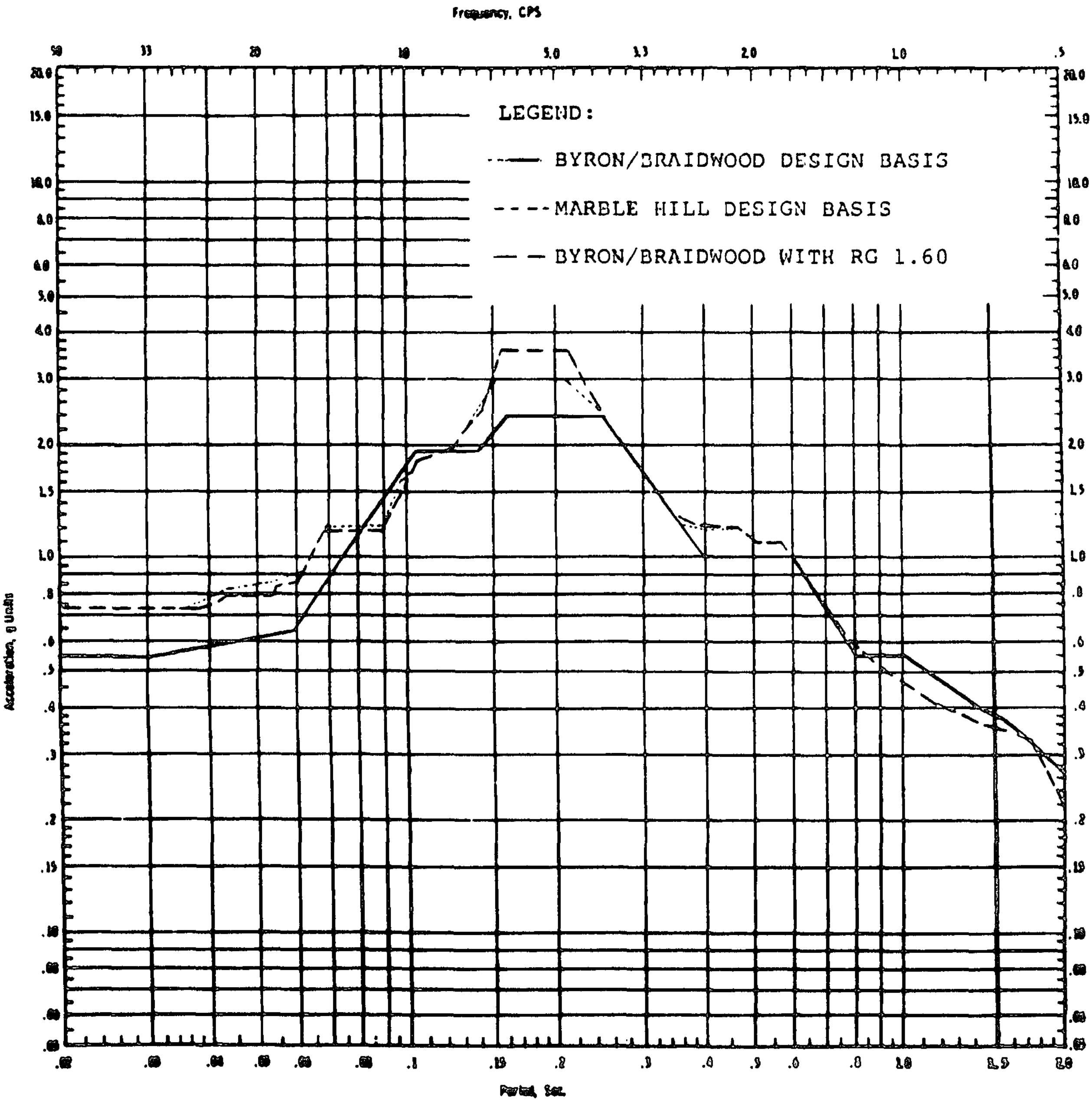
LOCATION: AUX- TUR- HB- RAD

SPECTRA NO 110-SS-NS

ELEVATION: 451'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-16
SPECTRA 110-SS-NS COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY, TURBINE, HEATER BAY, AND
RADWASTE BUILDINGS—ELEVATION: 451'-0"



EXCITATION HORIZ : 2% DAMPING

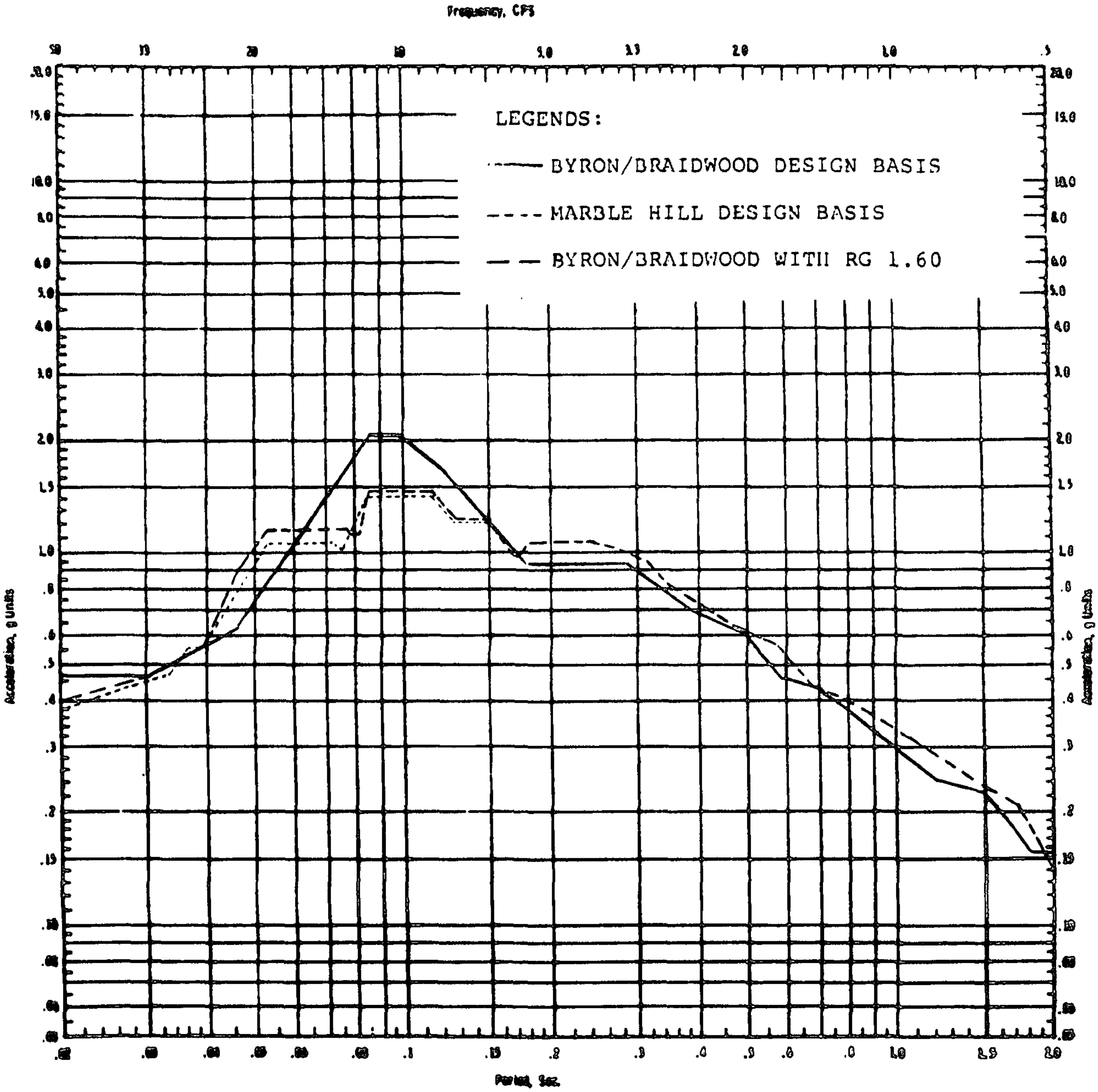
LOCATION: AUX-TUR-HB-RAD

SPECTRA NO 110-SS-EW

ELEVATION: 451'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-17
SPECTRA 110-SS-EW COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY, TURBINE, HEATER BAY, AND RADWASTE BUILDINGS—ELEVATION: 451'-0"



EXCITATION VERT : 2% DAMPING

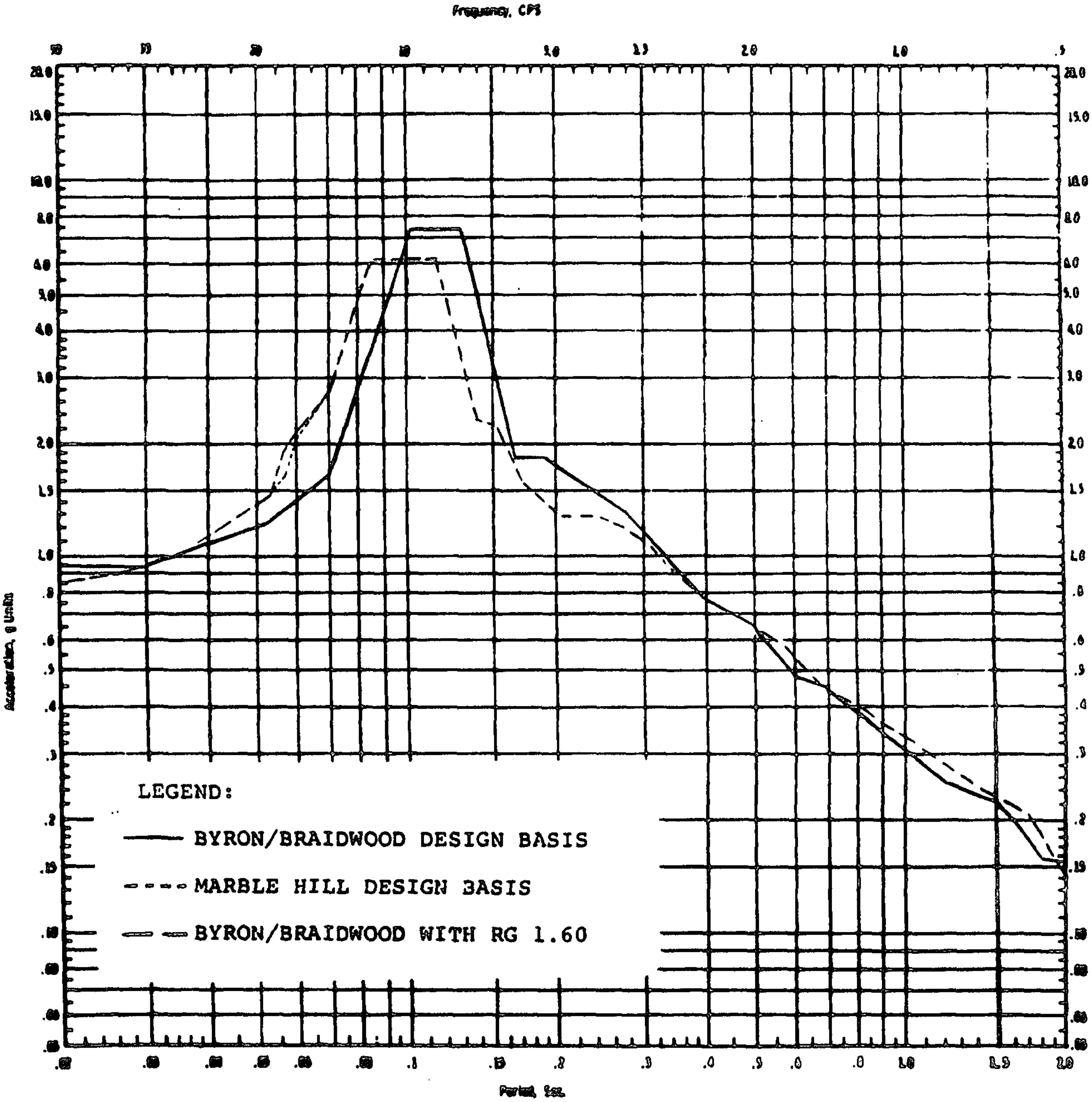
LOCATION: AUX-FHB WALL

SPECTRA NO 110-SS-VW

ELEVATION: 451'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-18
SPECTRA 110-SS-VW COMPARISON
EXCITATION: VERTICAL, 2% DAMPING
LOCATION: AUXILIARY AND FUEL HANDLING
BUILDINGS WALL—ELEVATION: 451'-0"



EXCITATION VERT : 2% DAMPING

LOCATION: AUX - FHB SLAB

SPECTRA NO 110-SS-VS

ELEVATION: 451'-0

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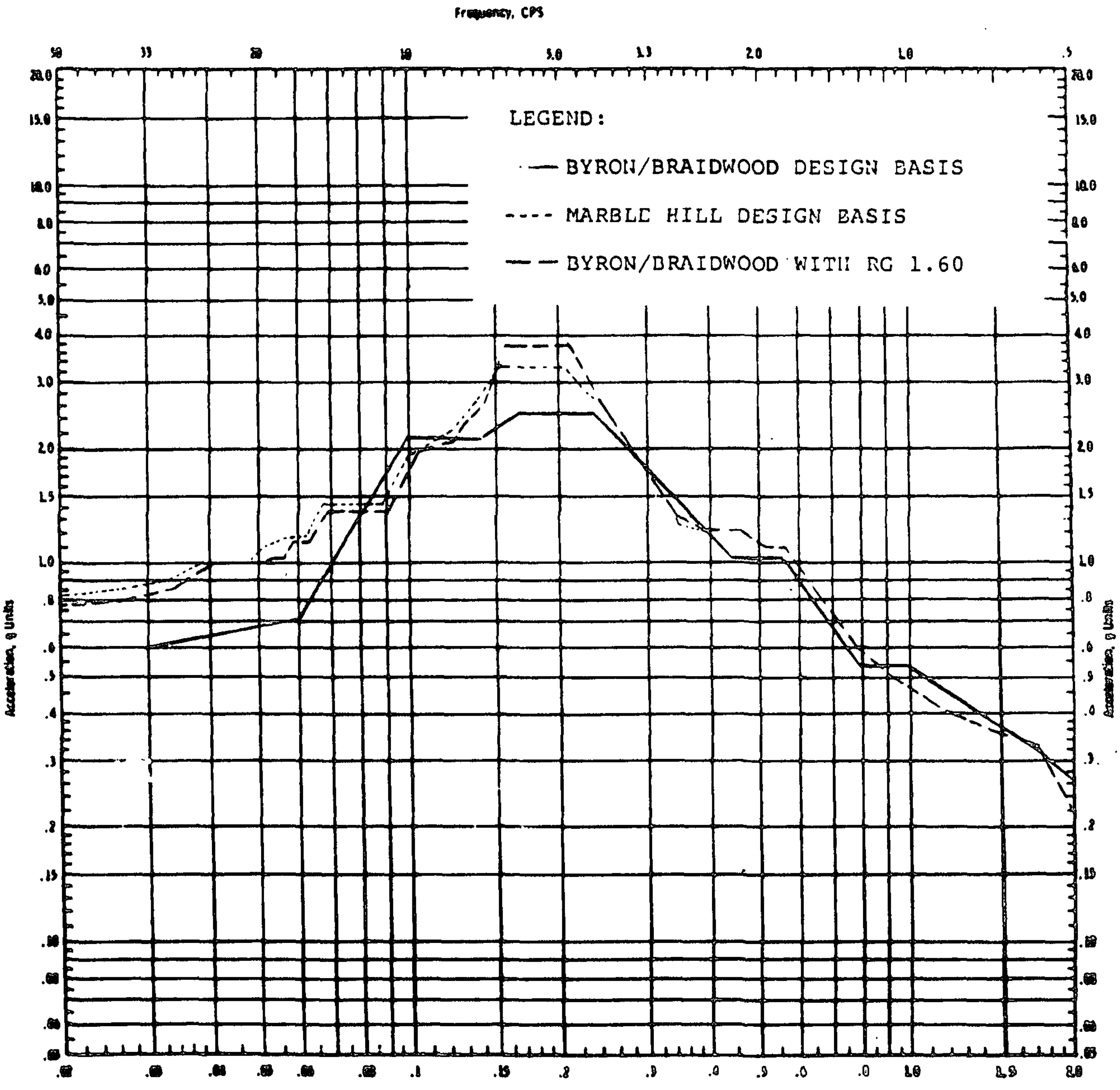
FIGURE Q130.6a-19
SPECTRA 110-SS-VS COMPARISON
EXCITATION: VERTICAL, 2% DAMPING
LOCATION: AUXILIARY AND FUEL HANDLING
BUILDINGS SLAB—ELEVATION: 451'-0"

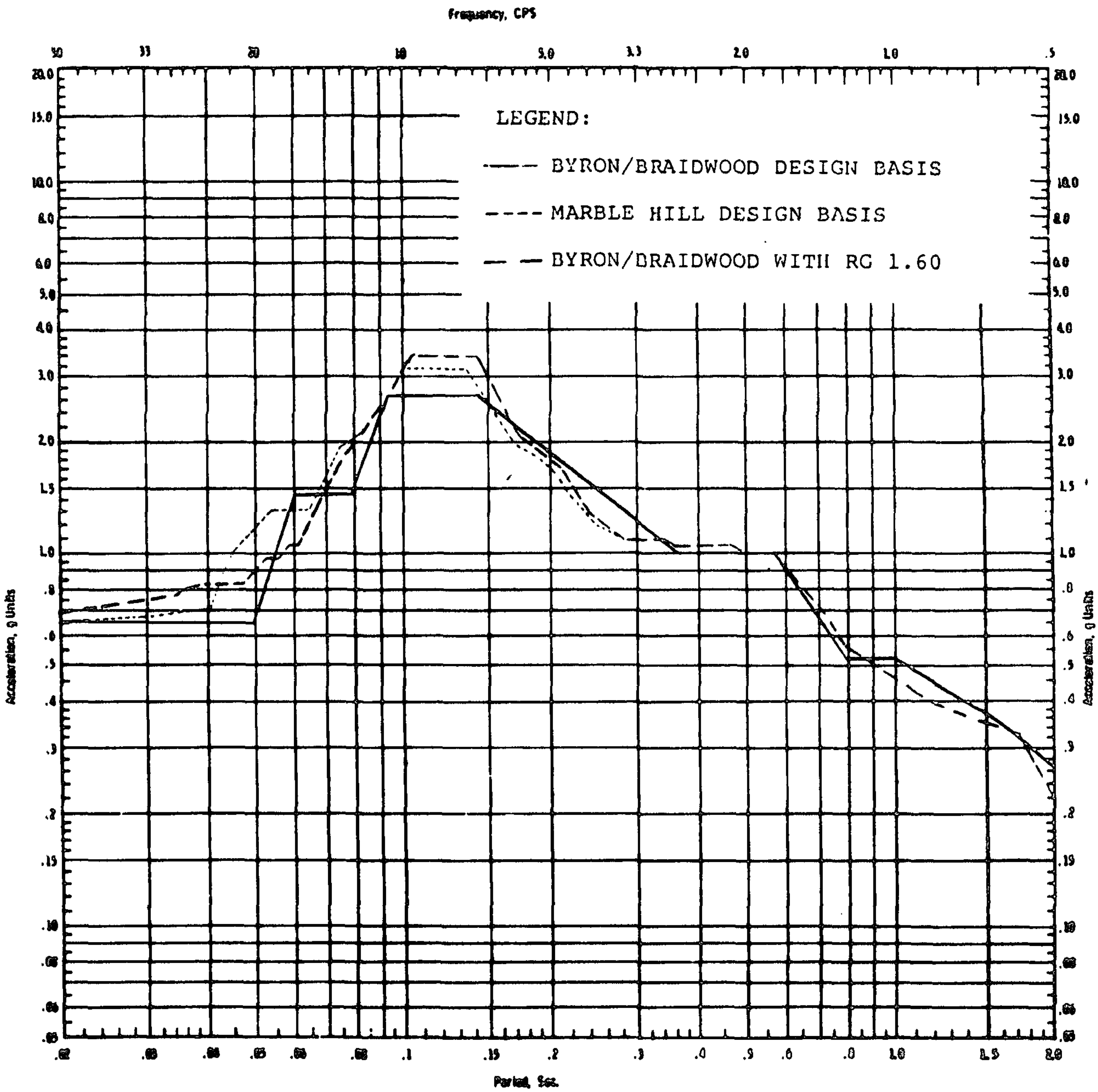


EXCITATION HORIZ : 2% DAMPING LOCATION: AUX

SPECTRA NO 111-SS-EW ELEVATION: 467'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-20
SPECTRA 111-SS-EW COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 467'-0"



EXCITATION HORIZ : 2% DAMPING

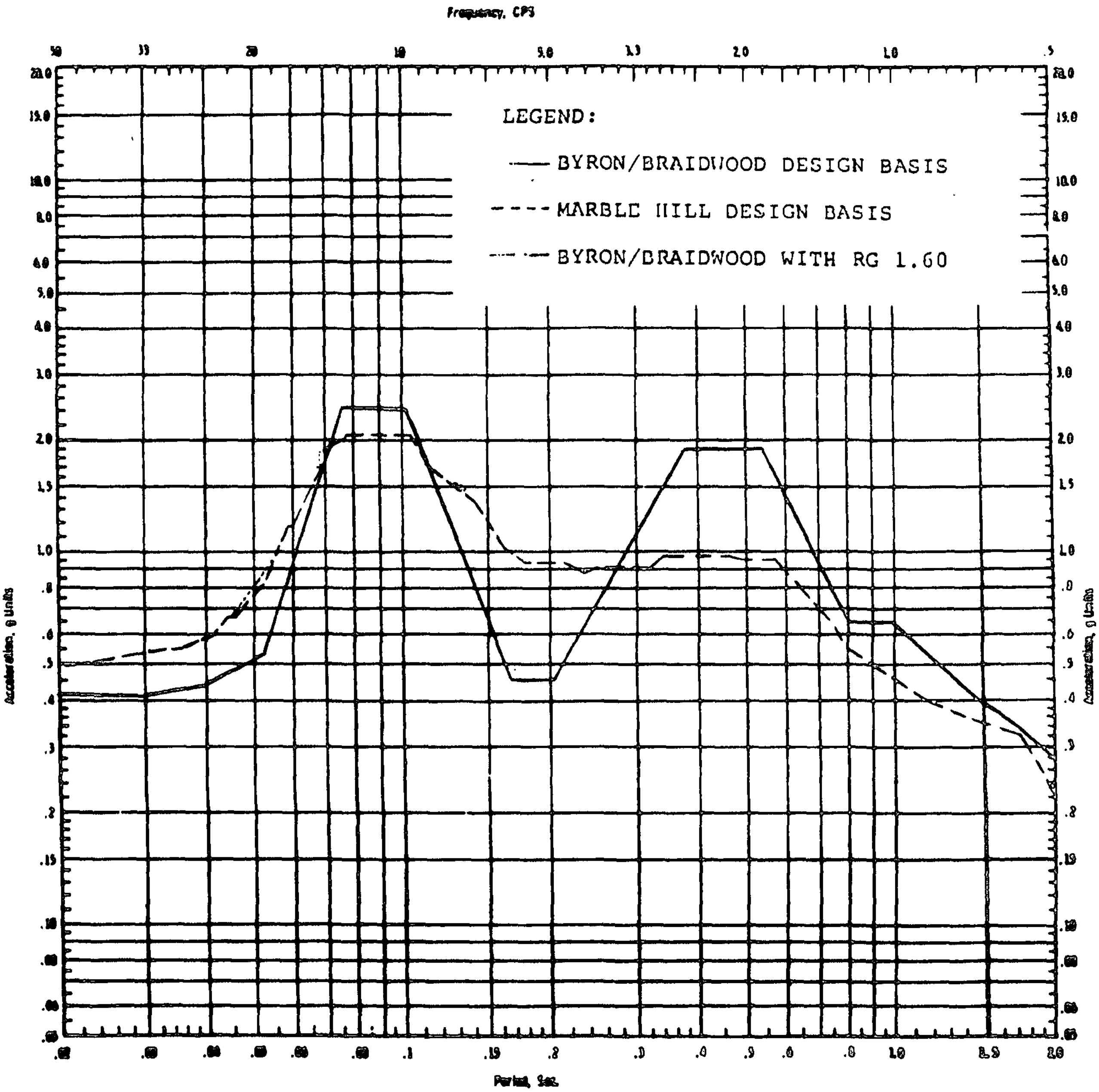
LOCATION: AUX

SPECTRA NO 111-SS-NS

ELEVATION: 467'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-21
SPECTRA 111-SS-NS COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 467'-0"



EXCITATION HORIZ : 2% DAMPING

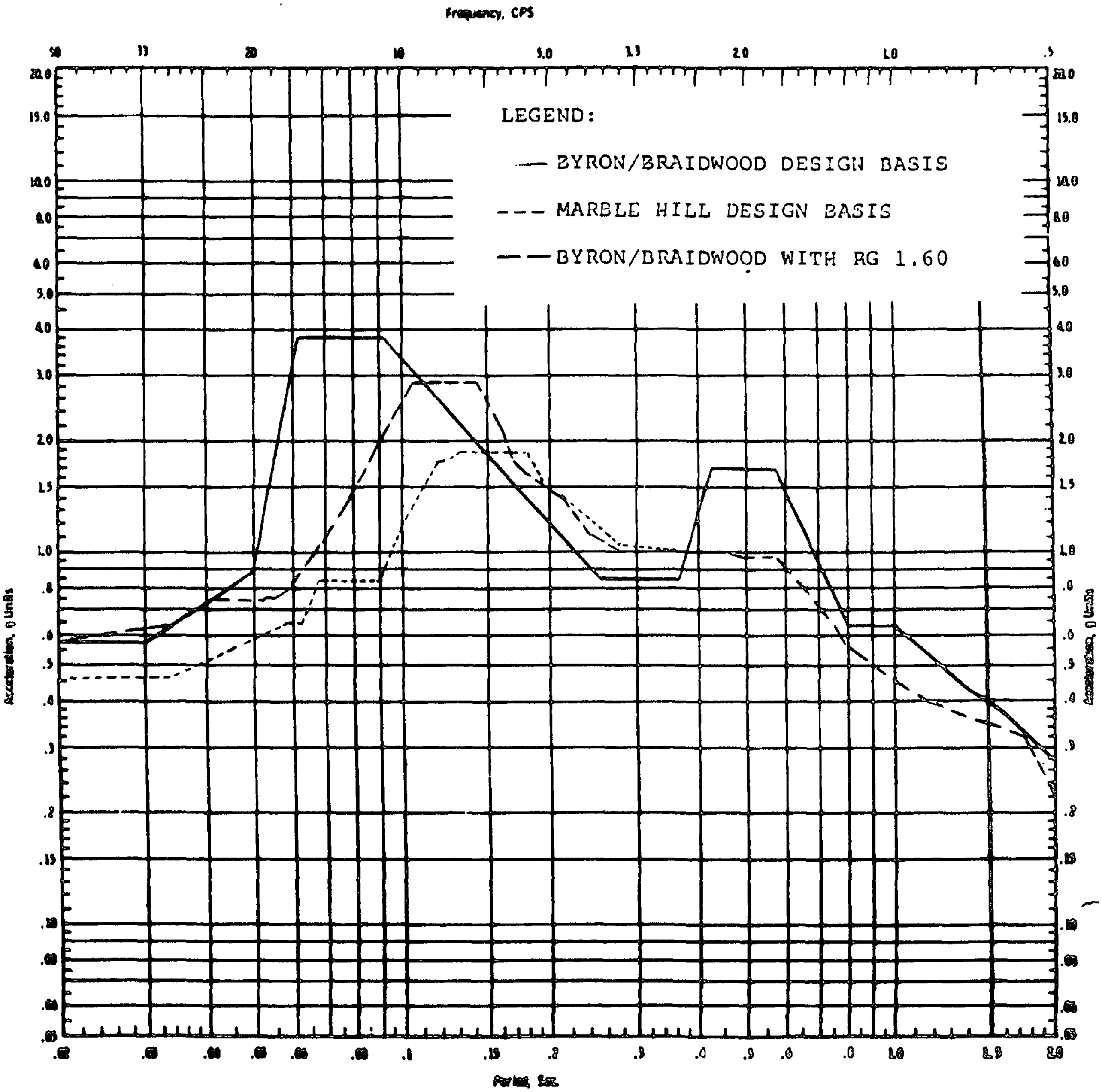
LOCATION: FHB

SPECTRA NO 112-SS-EW

ELEVATION: 473'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-22
SPECTRA 112-SS-EW COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: FUEL HANDLING BUILDING
ELEVATION: 473'-0"



EXCITATION HORIZ : 2% DAMPING

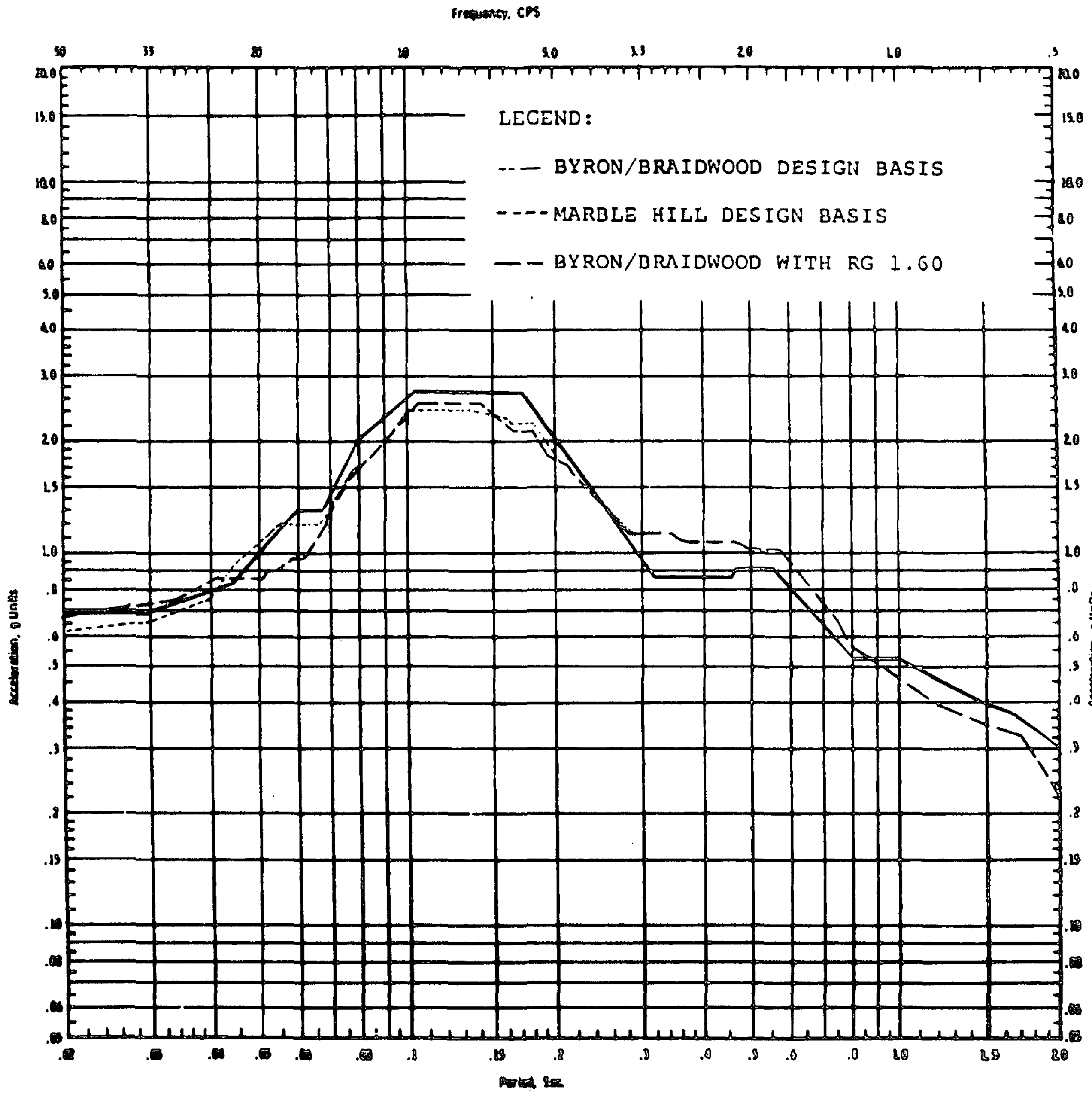
LOCATION: FHB

SPECTRA NO 112-SS-NS

ELEVATION: 473'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-23
SPECTRA 112-SS-NS COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: FUEL HANDLING BUILDING
ELEVATION: 473'-0"



EXCITATION HORIZ : 2% DAMPING

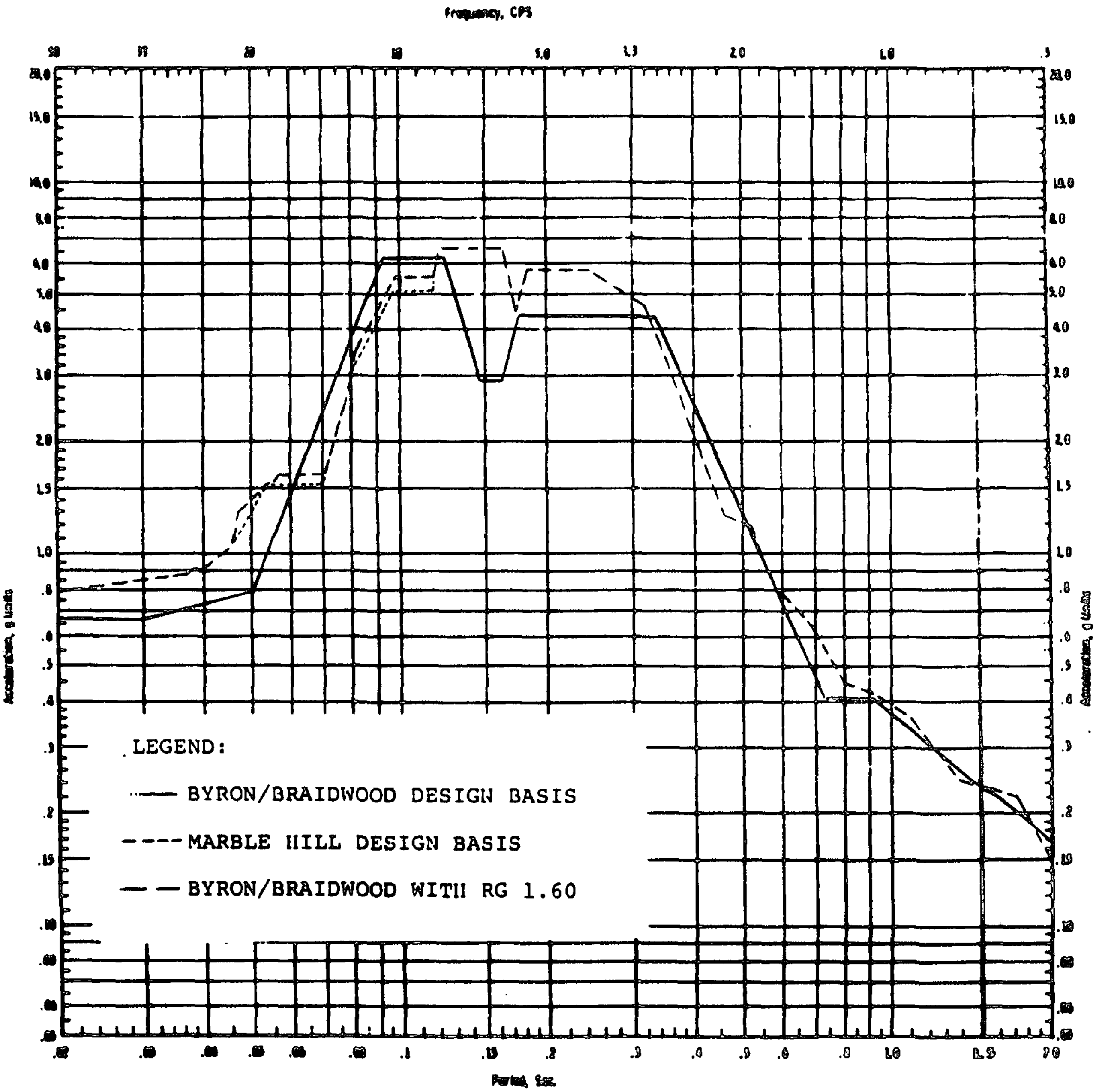
LOCATION: AUX

SPECTRA NO 113-SS-NS

ELEVATION: 477'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-24
SPECTRA 113-SS-NS COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 477'-0"



EXCITATION VERT : 2% DAMPING

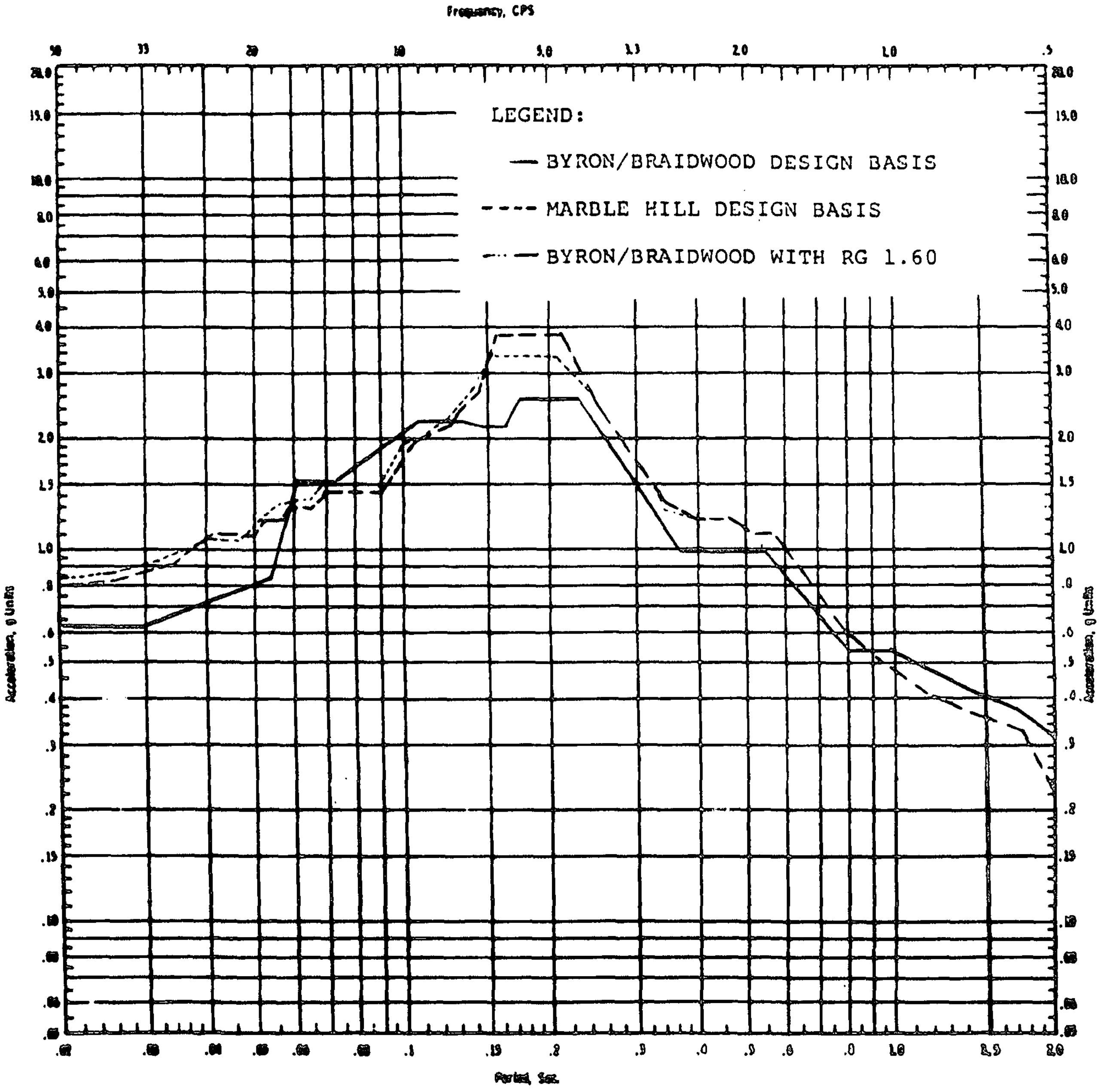
LOCATION: AUX - FHB SLAB

SPECTRA NO 113-SS-VS

ELEVATION: 477'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-25
SPECTRA 113-SS-VS COMPARISON
EXCITATION: VERTICAL, 2% DAMPING
LOCATION: AUXILIARY AND FUEL HANDLING
BUILDINGS SLAB—ELEVATION: 477'-0"



EXCITATION HORIZ : 2% DAMPING

LOCATION: AUX

SPECTRA NO 113-SS-EW

ELEVATION: 477'-0

BYRON/BRAIDWOOD STATIONS
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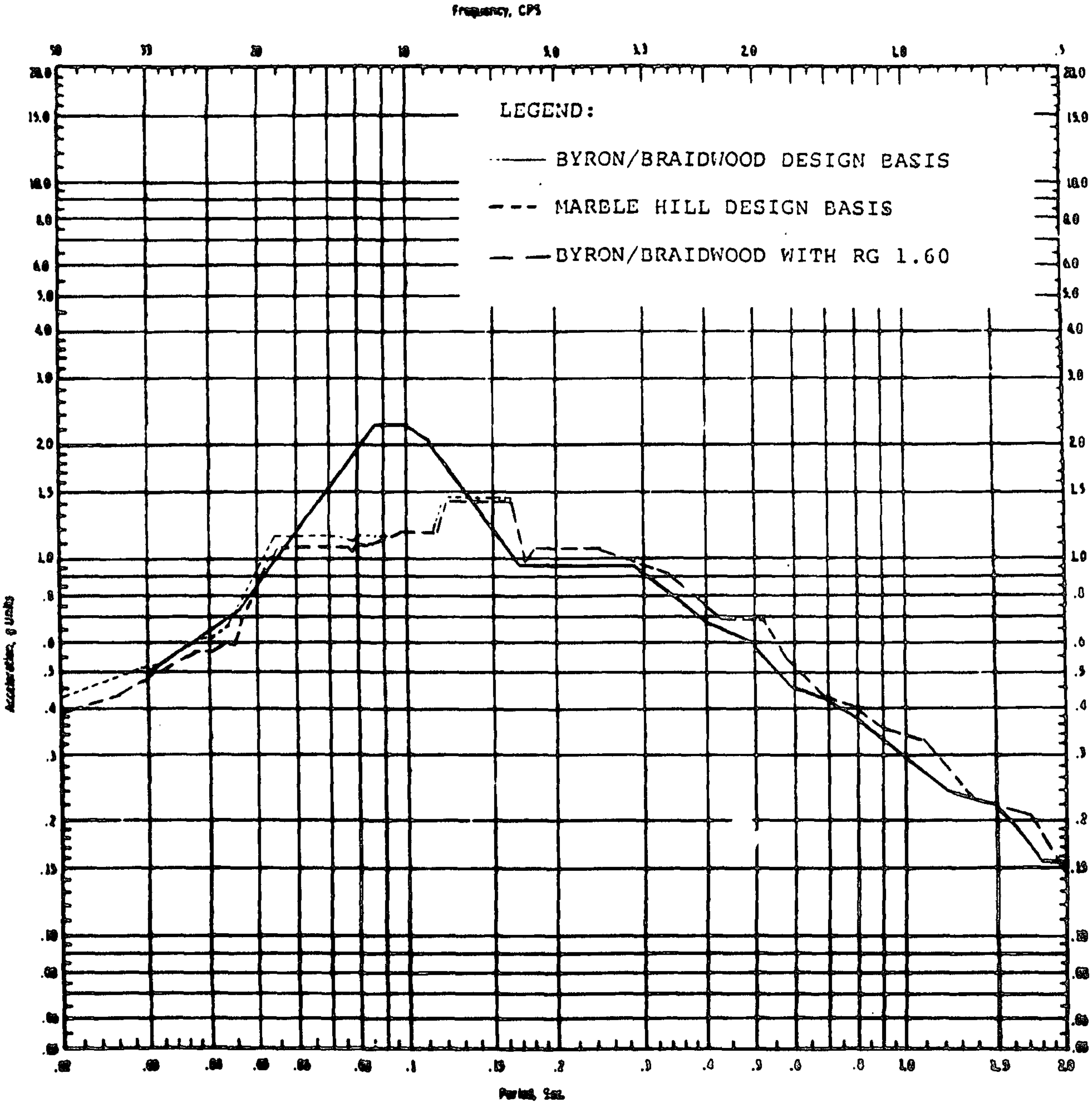
FIGURE Q130.6a-26
SPECTRA 113-SS-EW COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 477'-0"



EXCITATION VERT : 2% DAMPING

LOCATION: AUX-FHB WALL

SPECTRA NO 113-SS-VW

ELEVATION: 477'-0, 485'-0

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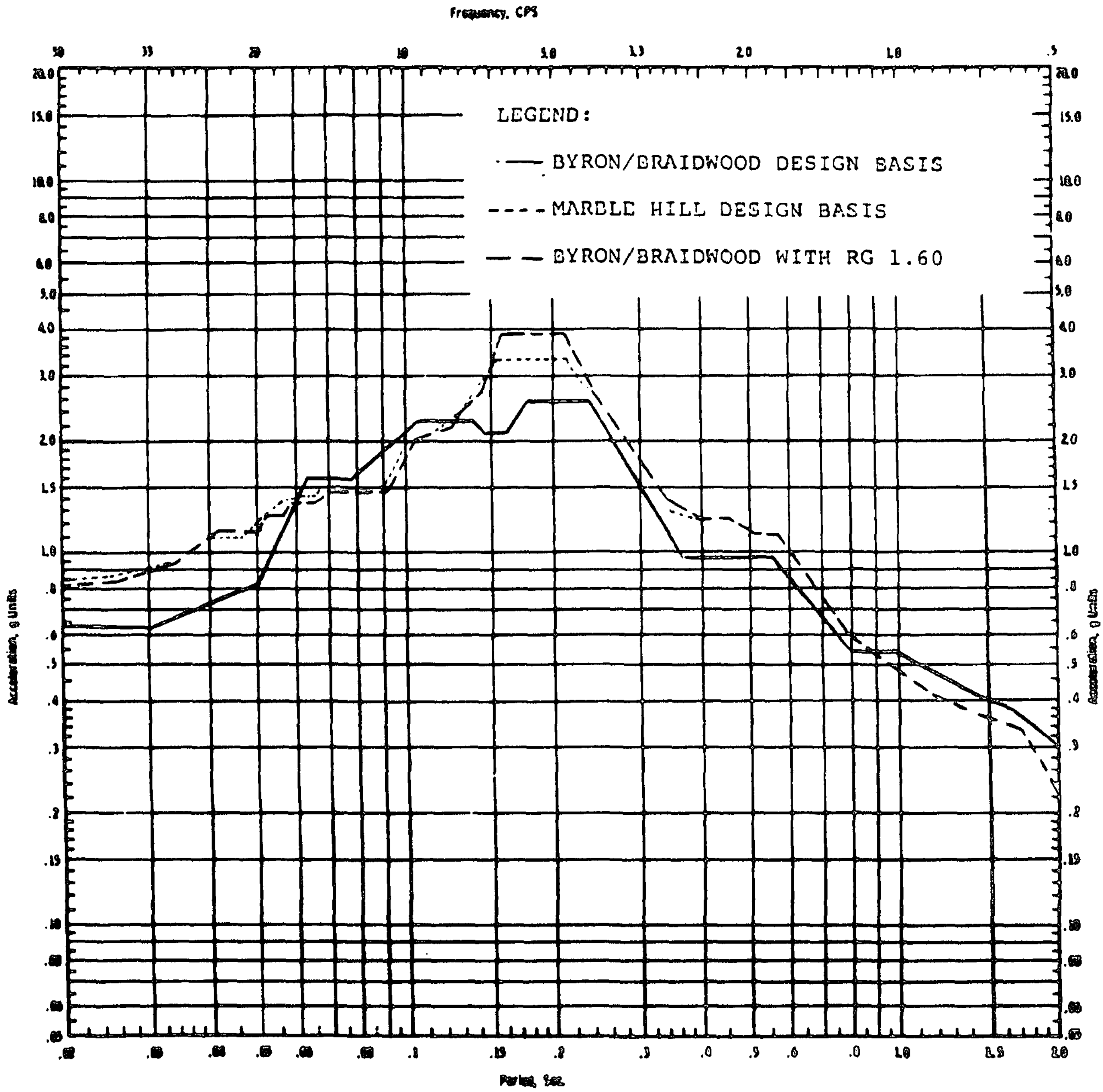
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-27
SPECTRA 113-SS-VW COMPARISON
EXCITATION: VERTICAL, 2% DAMPING
LOCATION: AUXILIARY AND FUEL HANDLING
BUILDINGS WALL—ELEVATION: 477'-0"; 485'-0"



EXCITATION HORIZ : 2% DAMPING

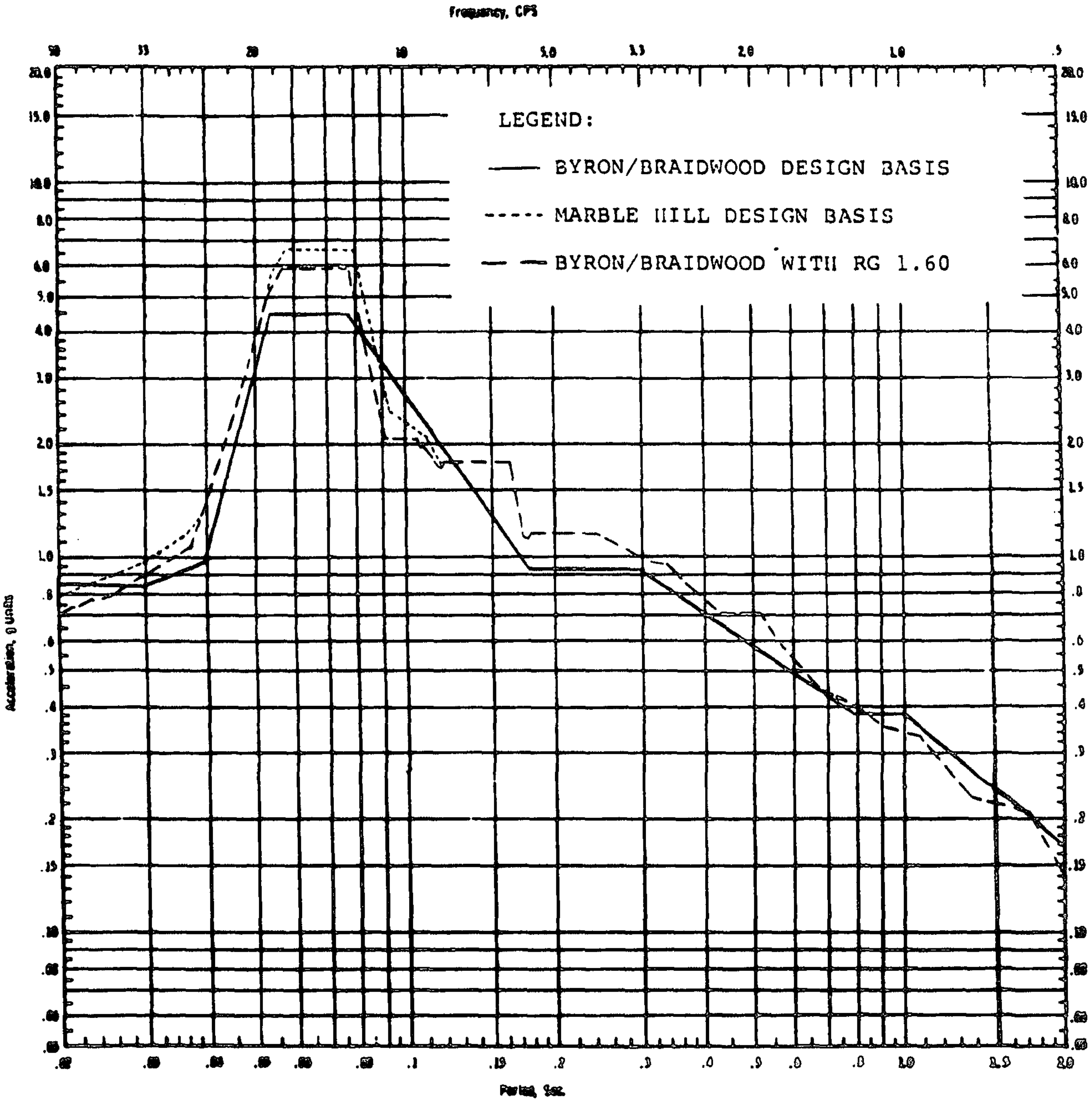
LOCATION: AUX

SPECTRA NO 114-SS-EW

ELEVATION: 485'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-28
SPECTRA 114-SS-EW COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 485'-0"



EXCITATION VERT : 2% DAMPING

LOCATION: AUX - FHB SLAB

SPECTRA NO 114 - SS - VS

ELEVATION: 485'-0

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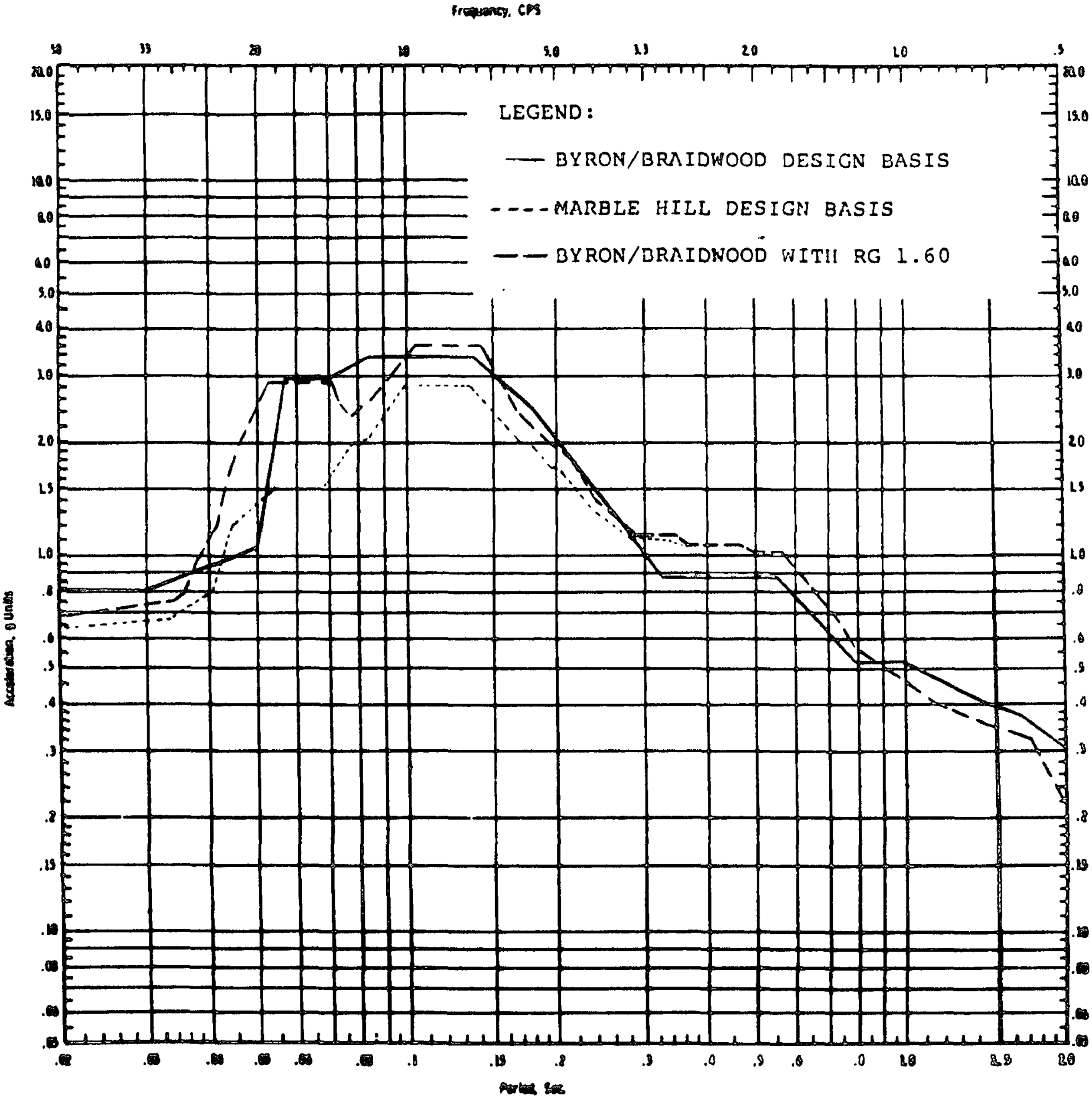
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.6a-29
SPECTRA 114-SS-VS COMPARISON
EXCITATION: VERTICAL, 2% DAMPING
LOCATION: AUXILIARY AND FUEL HANDLING
BUILDINGS SLAB—ELEVATION: 485'-0"



EXCITATION HORIZ : 2% DAMPING

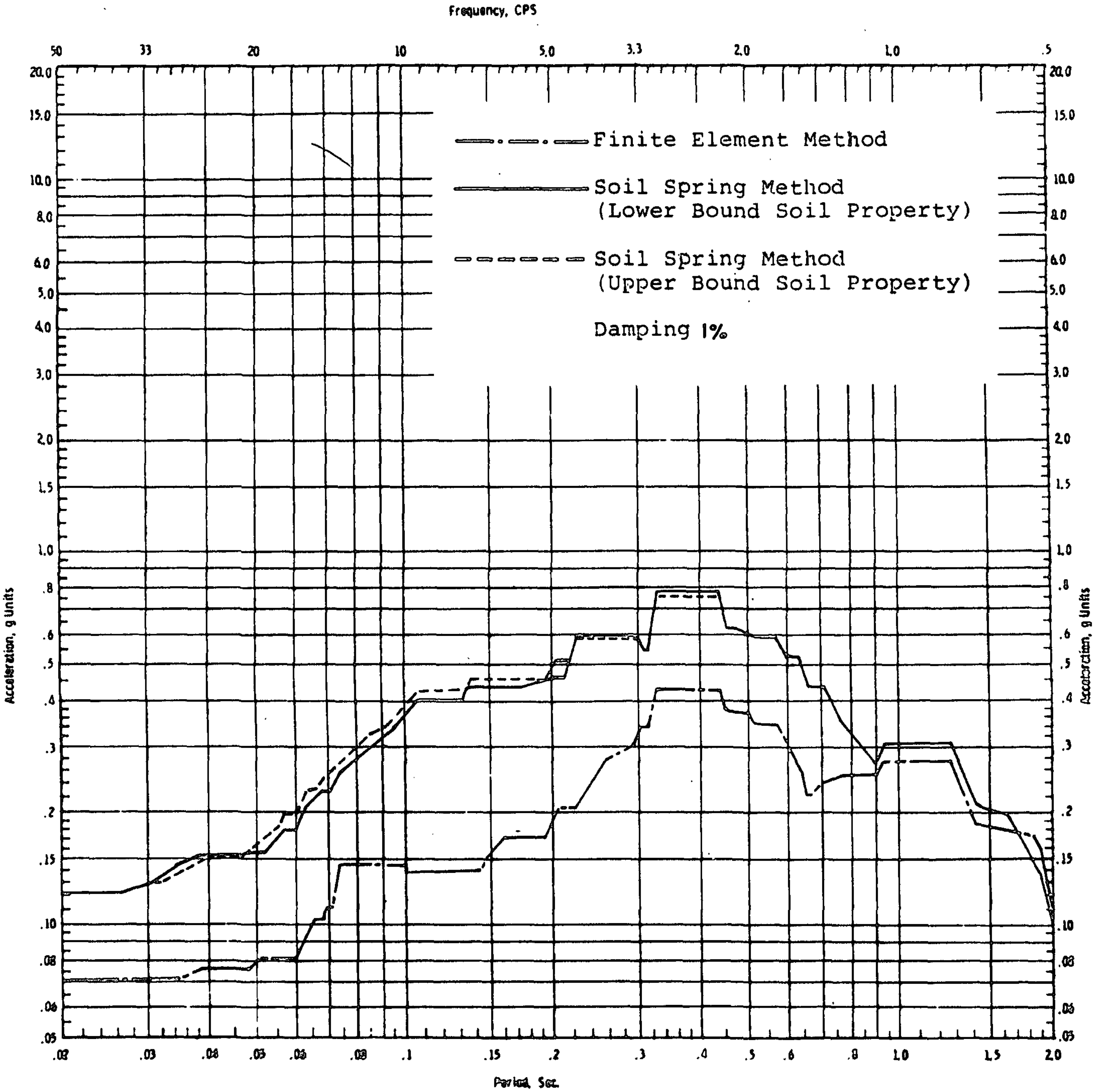
LOCATION: AUX

SPECTRA NO 114-SS-NS

ELEVATION: 485'-0

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

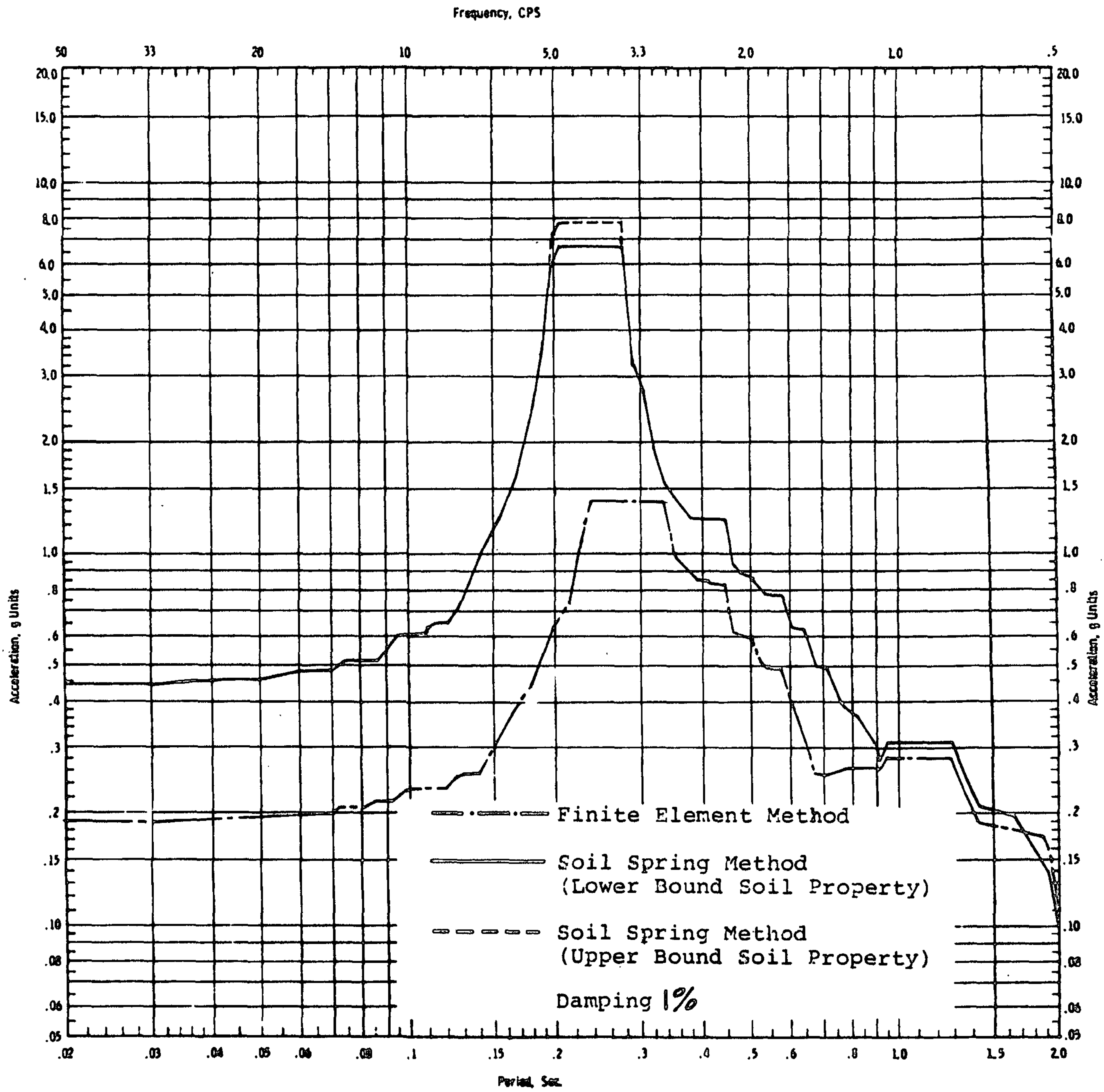
FIGURE Q130.6a-30
SPECTRA 114-SS-NS COMPARISON
EXCITATION: HORIZONTAL, 2% DAMPING
LOCATION: AUXILIARY BUILDING
ELEVATION: 485'-0"



BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.9-1

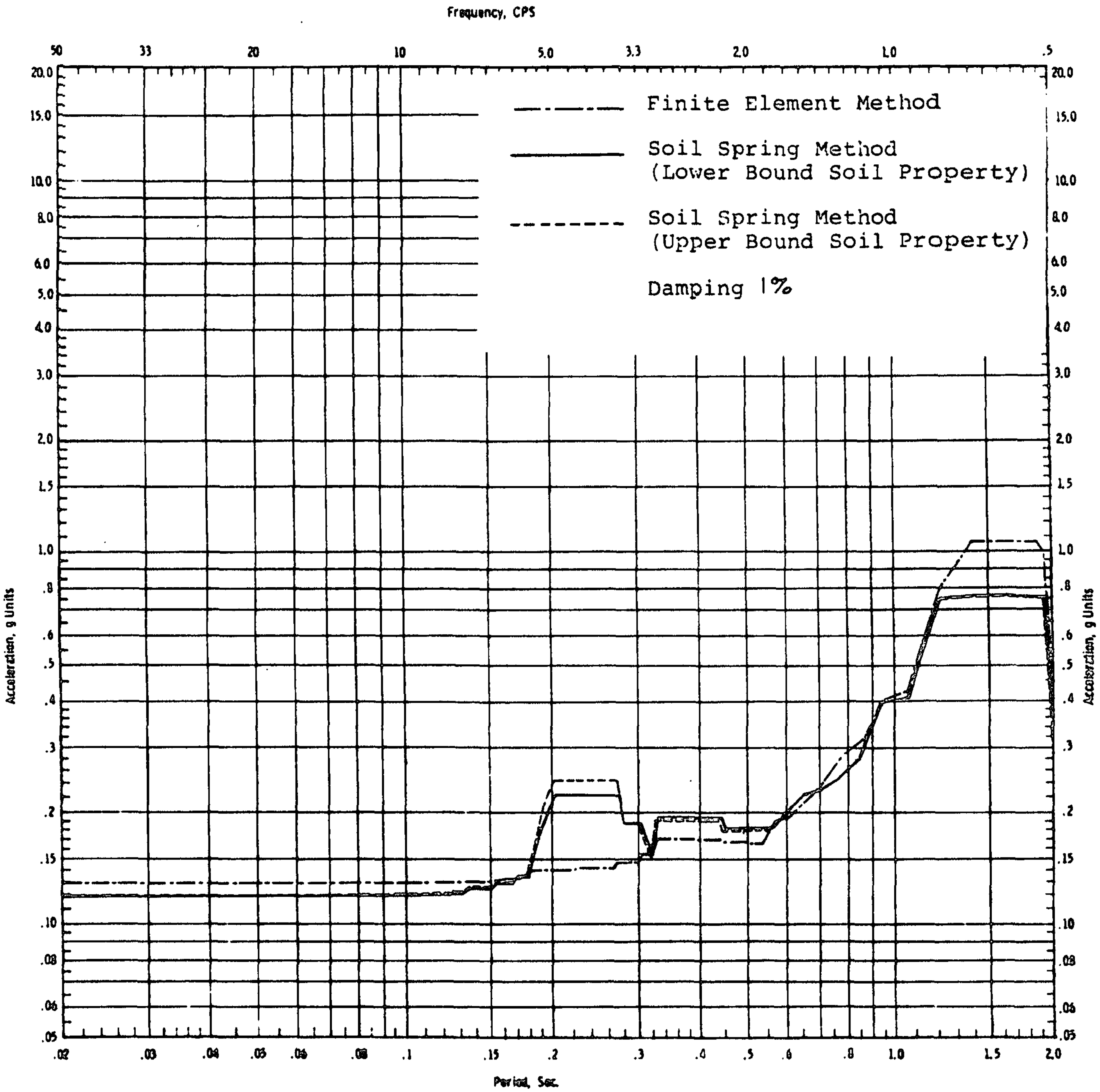
RIVER SCREEN HOUSE HORIZONTAL EW
OBE SPECTRA AT EL. 664'-0"



BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.9-2

RIVER SCREEN HOUSE HORIZONTAL EW
OBE SPECTRA AT EL. 702'-0"



BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE Q130.9-3

RIVER SCREEN HOUSE HORIZONTAL EW
OBE SPECTRA AT EL. 744'-0"

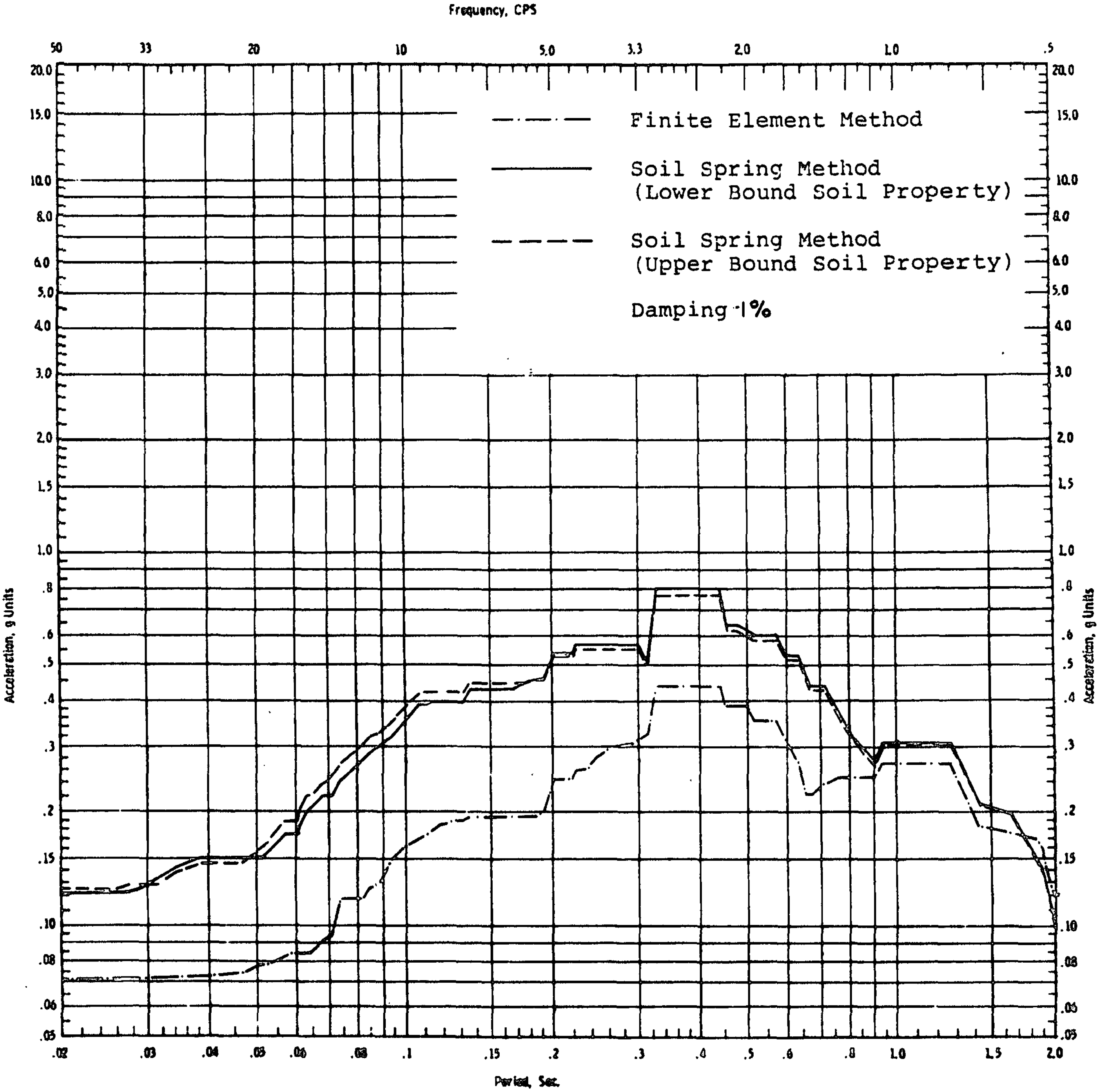


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FIGURE Q130.9-4

RIVER SCREEN HOUSE HORIZONTAL NS
OBE SPECTRA AT EL. 664'-0"

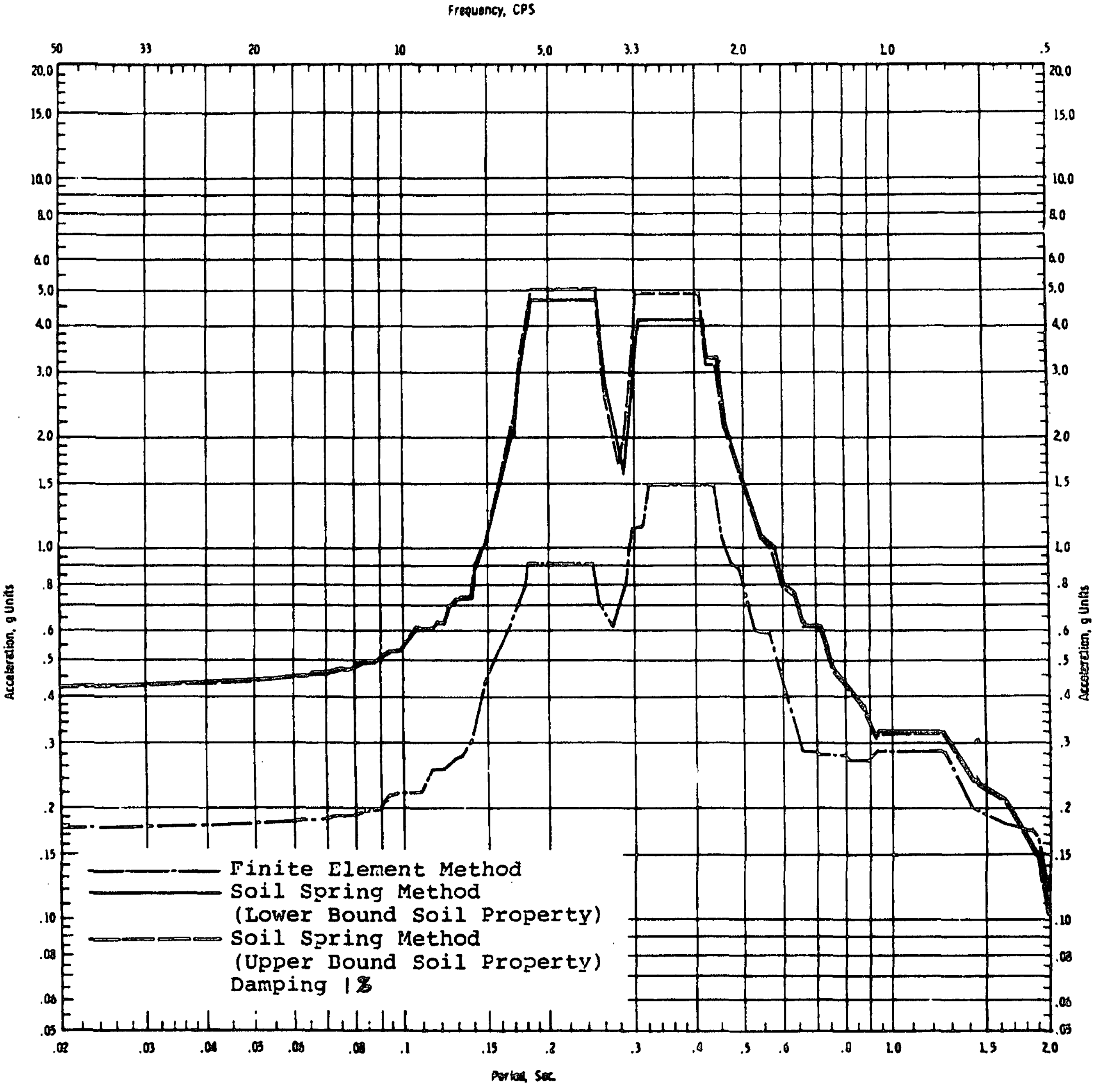

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FIGURE Q130.9-5

RIVER SCREEN HOUSE HORIZONTAL NS
OBE SPECTRA AT EL. 702'-0"

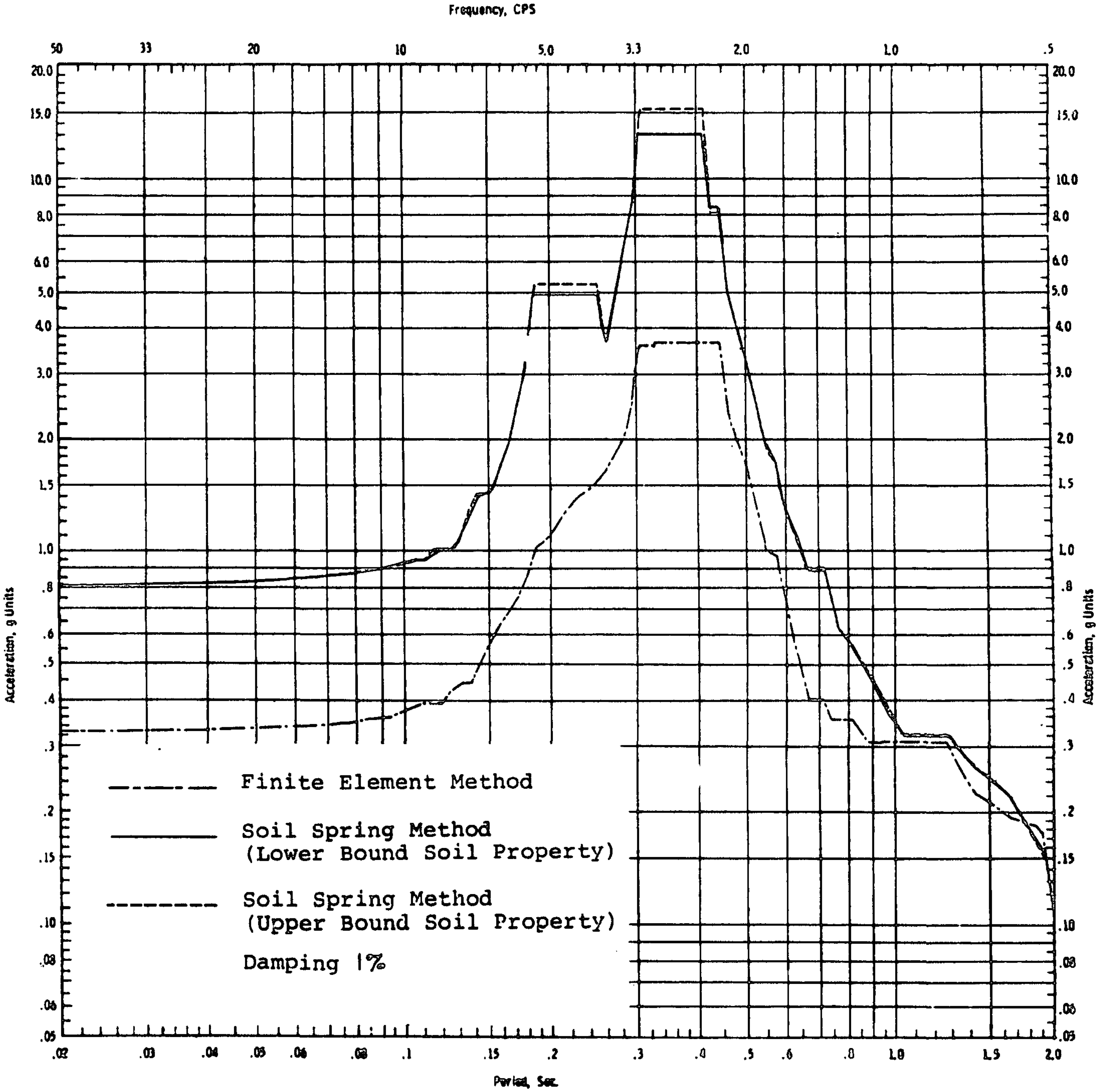

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FIGURE Q130.9-6

RIVER SCREEN HOUSE HORIZONTAL NS
OBE SPECTRA AT EL. 744'-0"

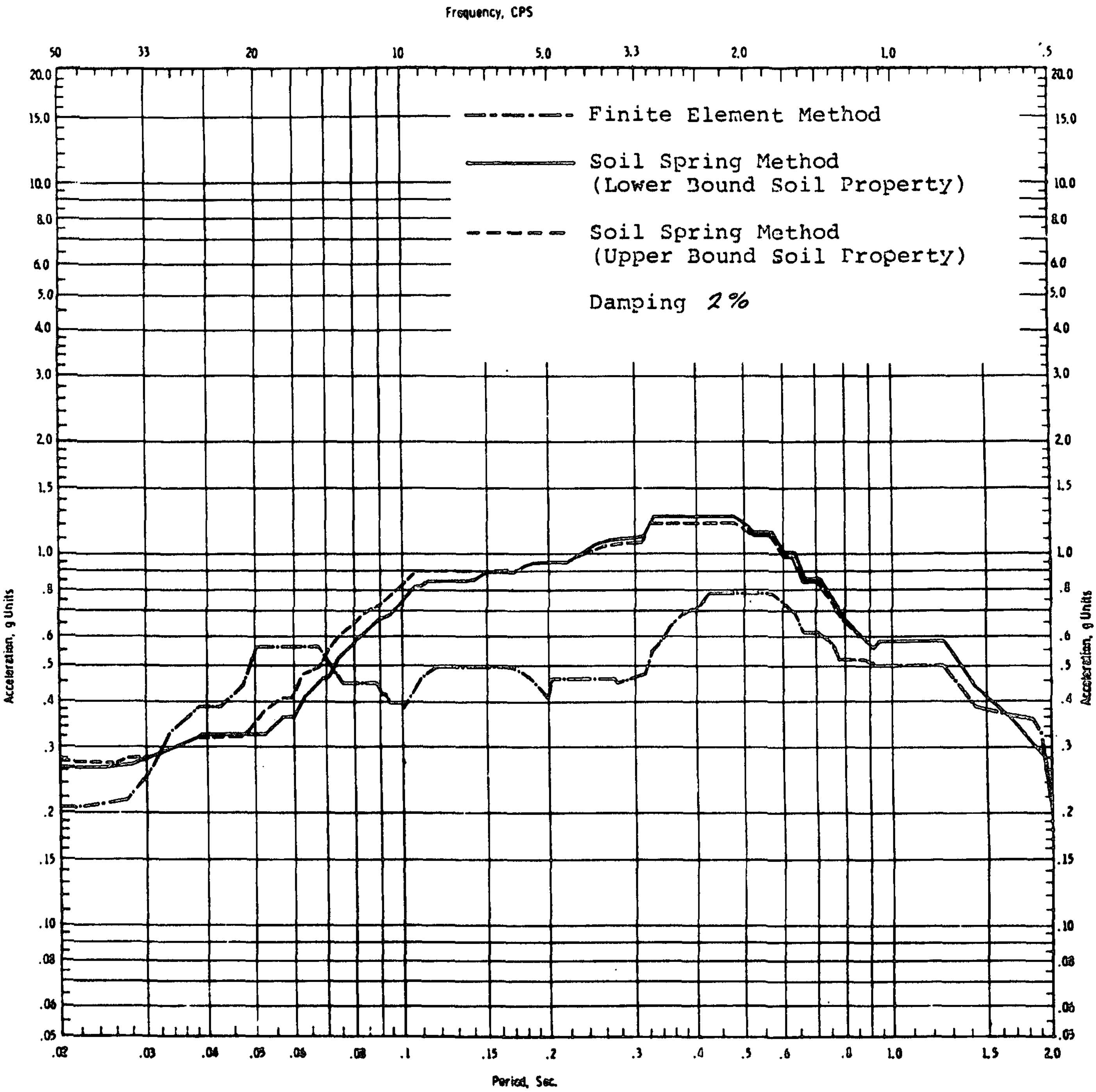

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FIGURE Q130.9-7

RIVER SCREEN HOUSE HORIZONTAL EW
SSE SPECTRA AT EL. 664'-0"

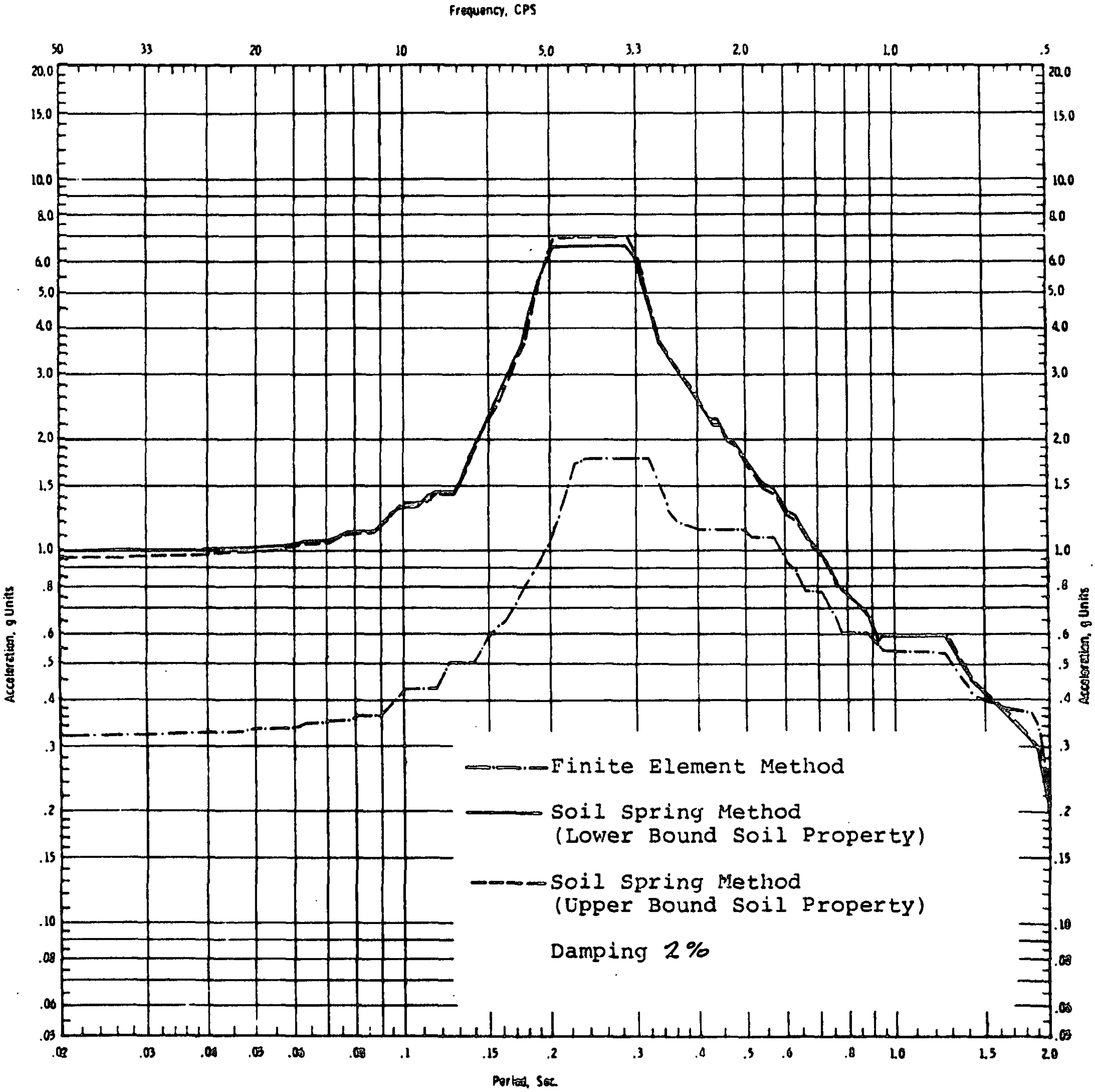

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FIGURE Q130.9-8

RIVER SCREEN HOUSE HORIZONTAL EW
SSE SPECTRA AT EL. 702'-0"

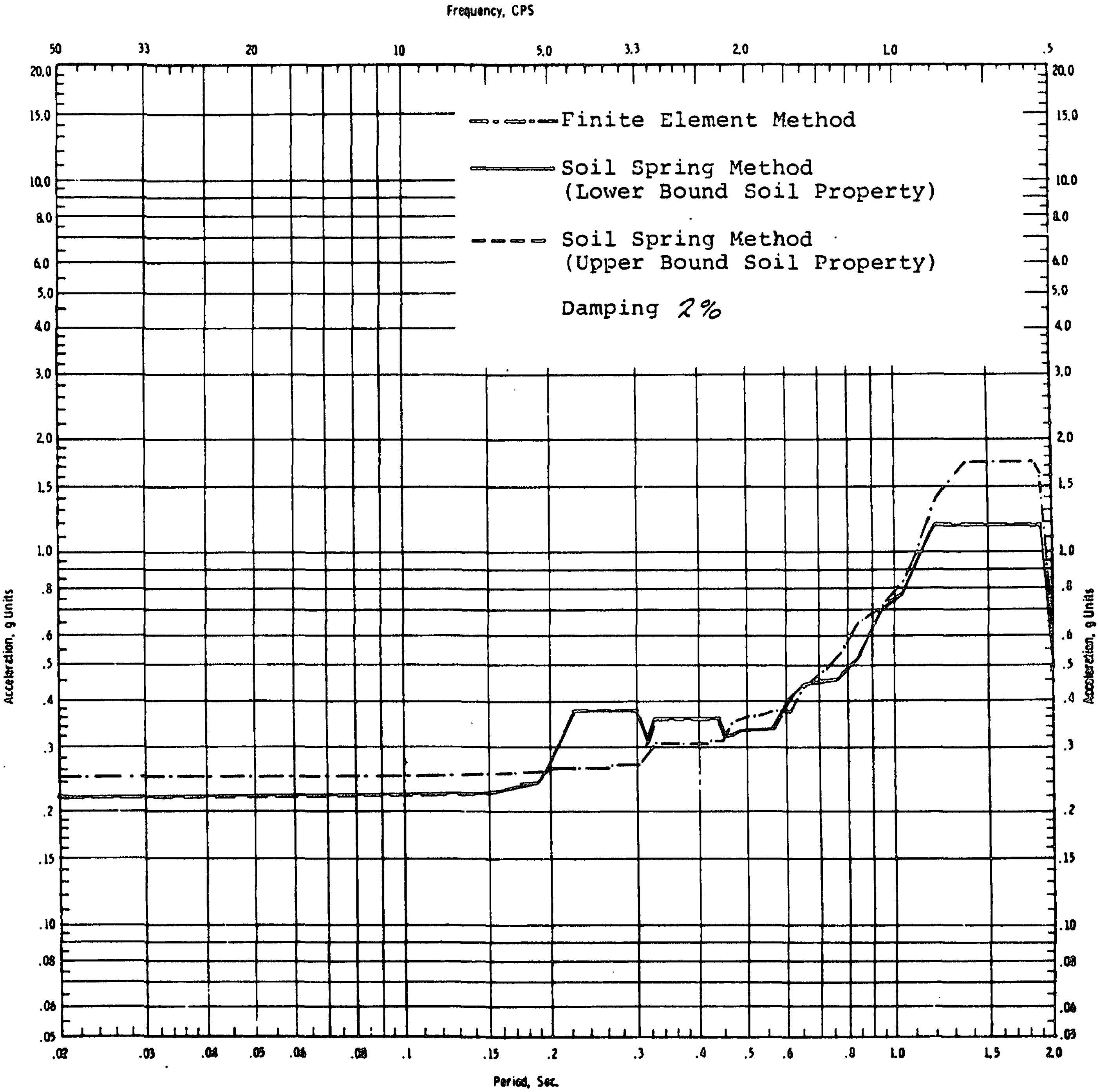

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FIGURE Q130.9-9

RIVER SCREEN HOUSE HORIZONTAL EW
SSE SPECTRA AT EL. 744'-0"

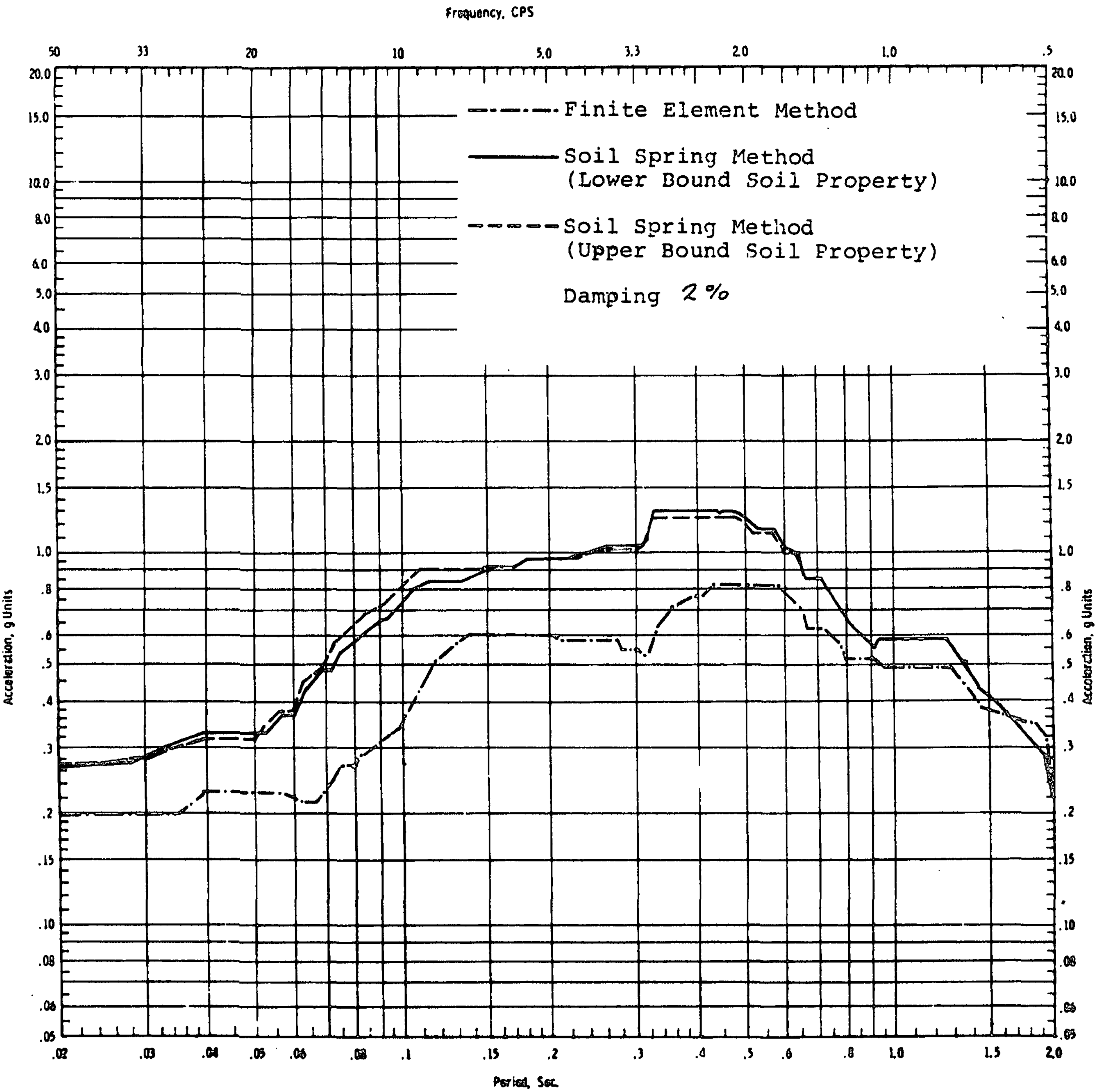

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FIGURE Q130.9-10

RIVER SCREEN HOUSE HORIZONTAL NS
SSE SPECTRA AT EL. 664'-0"

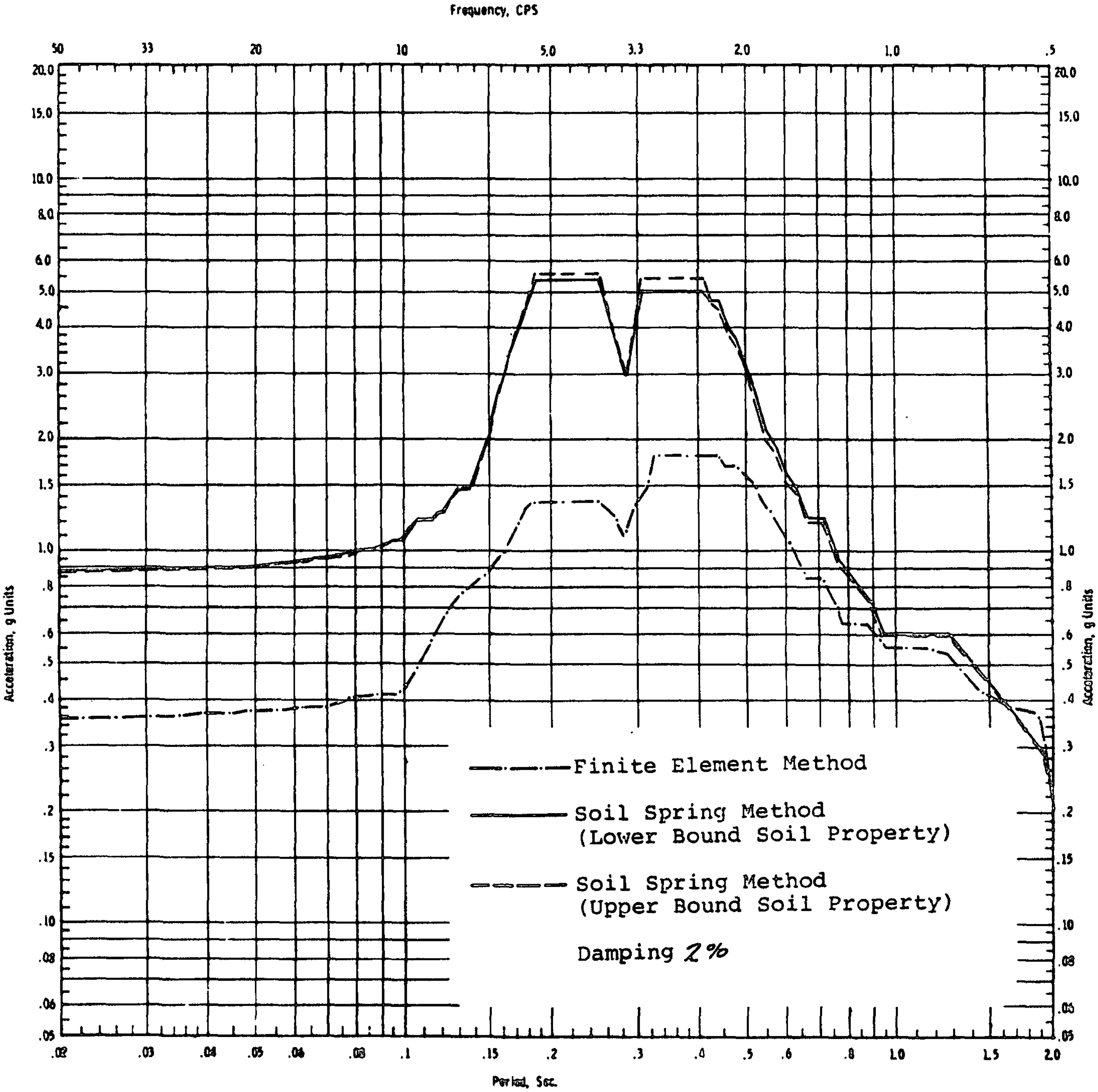

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FIGURE Q130.9-11

RIVER SCREEN HOUSE HORIZONTAL NS
SSE SPECTRA AT EL. 702'-0"

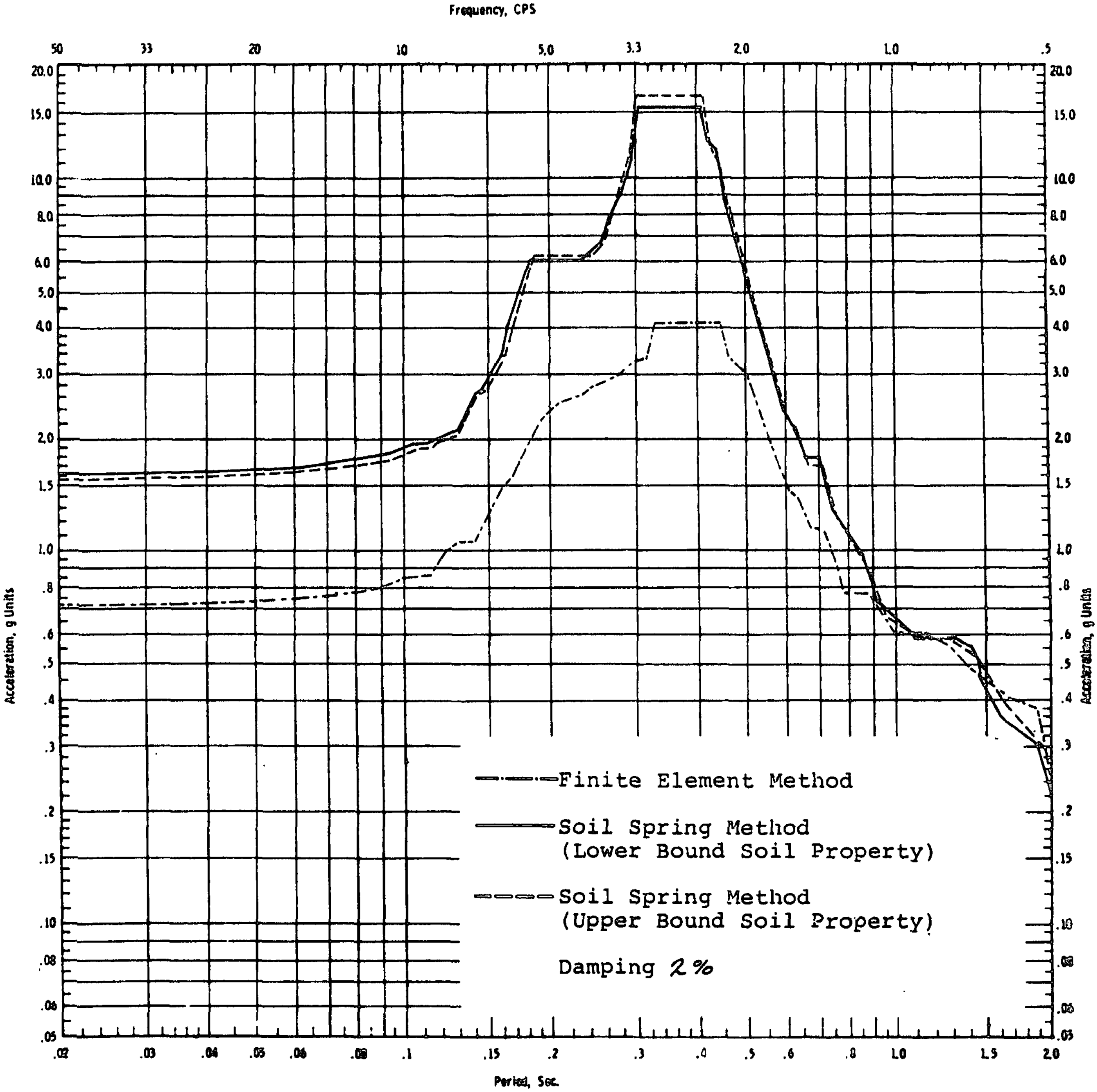

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FIGURE Q130.9-12

RIVER SCREEN HOUSE HORIZONTAL NS
SSE SPECTRA AT EL. 744'-0"

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FIGURE Q130.9-13 CROSS SECTION OF RIVER SCREEN HOUSE

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FIGURE Q130.9-14

SHEAR WALL ARRANGEMENT AT
ELEVATION 664'-0"

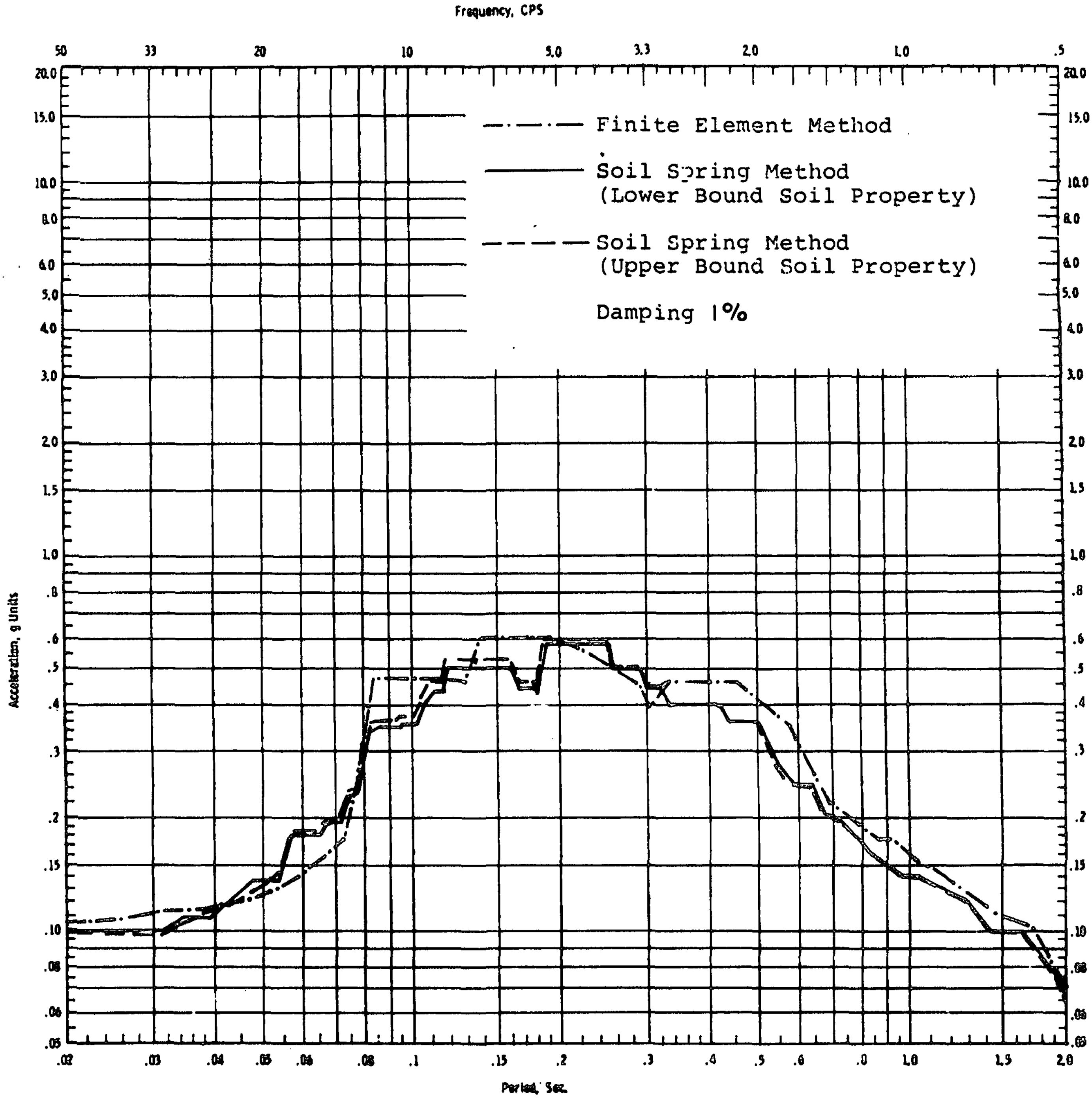

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FIGURE Q130.9-15

RIVER SCREEN HOUSE VERTICAL
OBE SPECTRA AT EL. 664'-0"

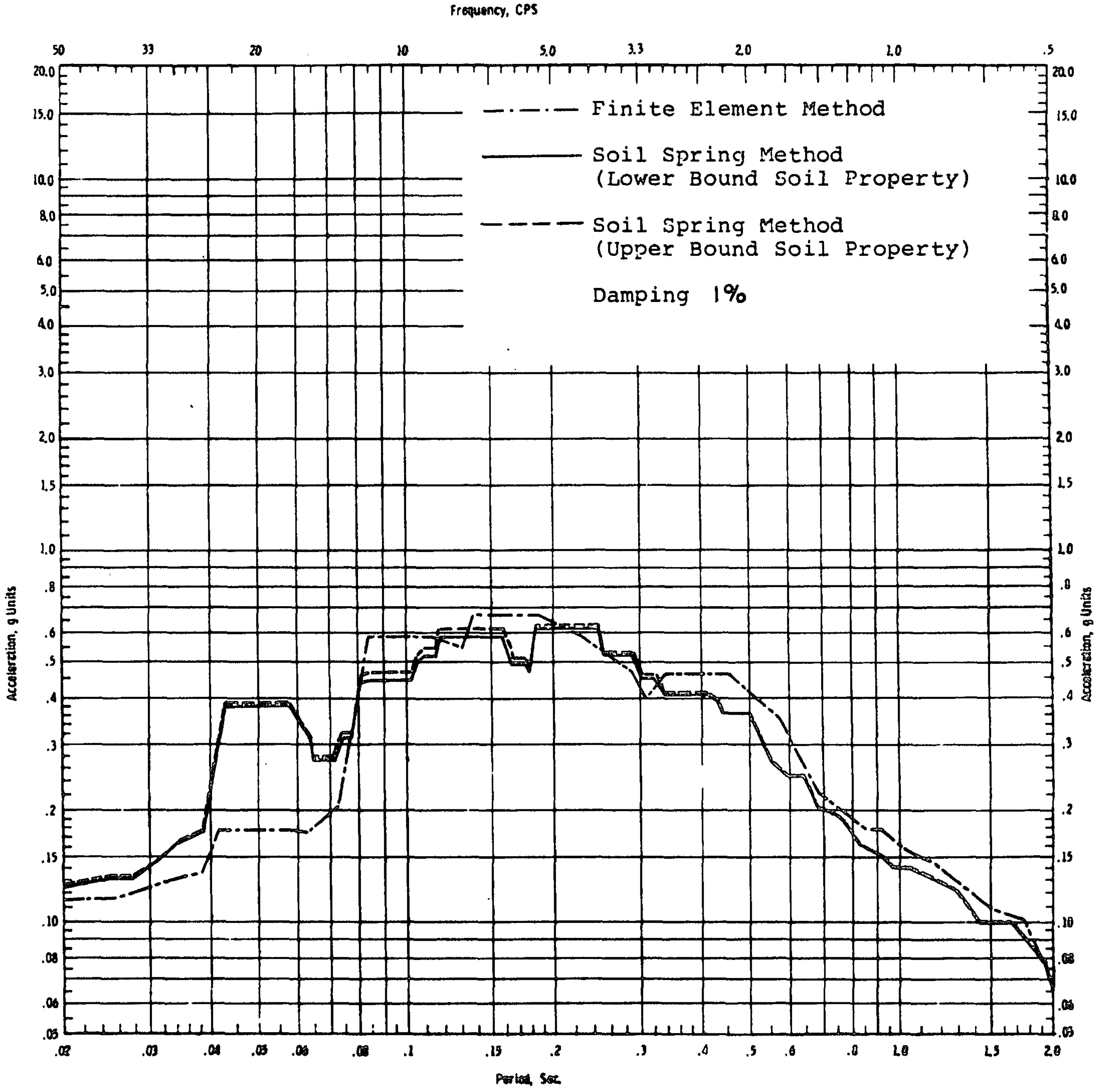

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FIGURE Q130.9-16

RIVER SCREEN HOUSE VERTICAL
OBE SPECTRA AT EL. 702'-0"

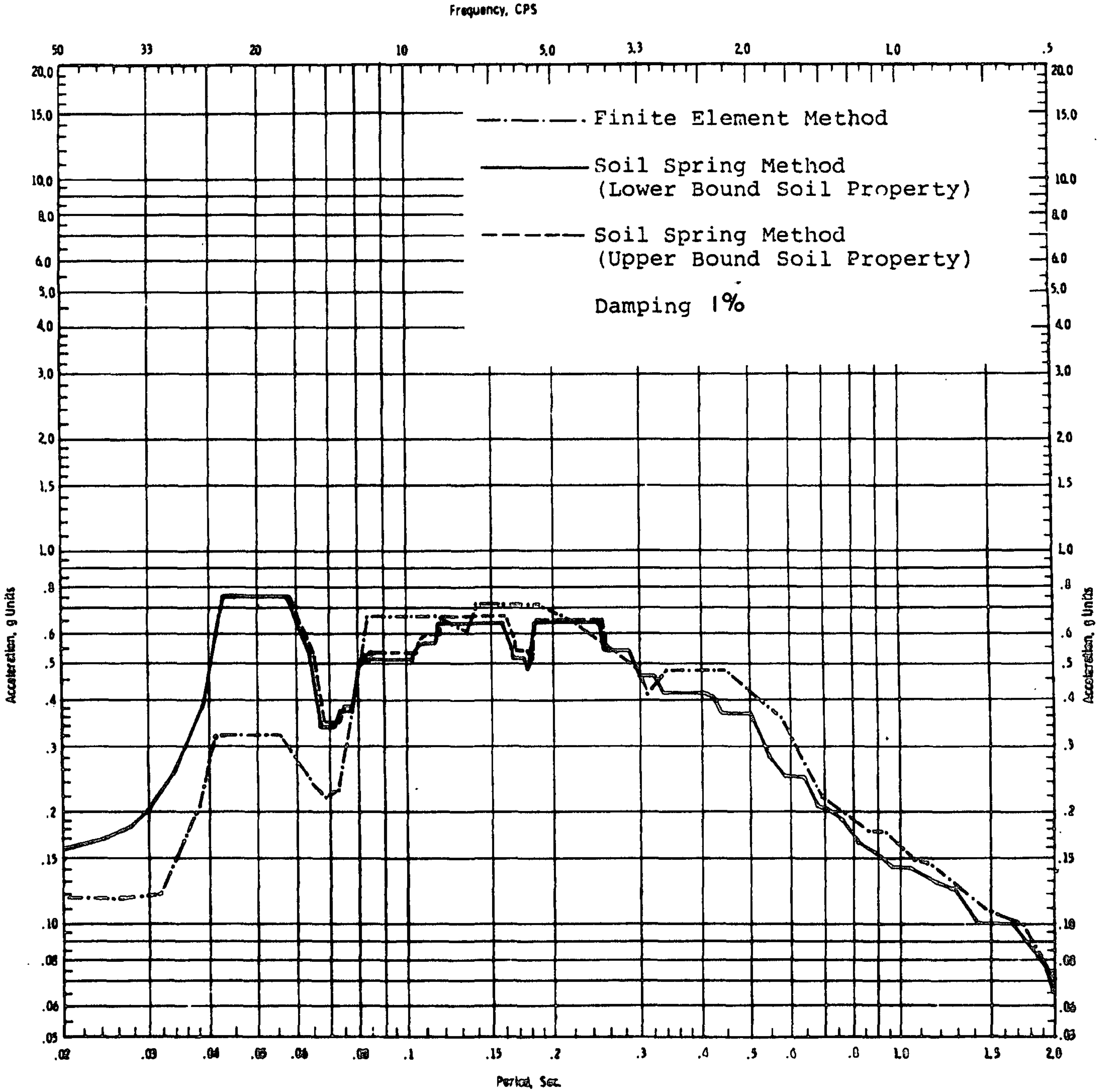

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FIGURE Q130.9-17

RIVER SCREEN HOUSE VERTICAL
OBE SPECTRA AT EL. 744'-0"

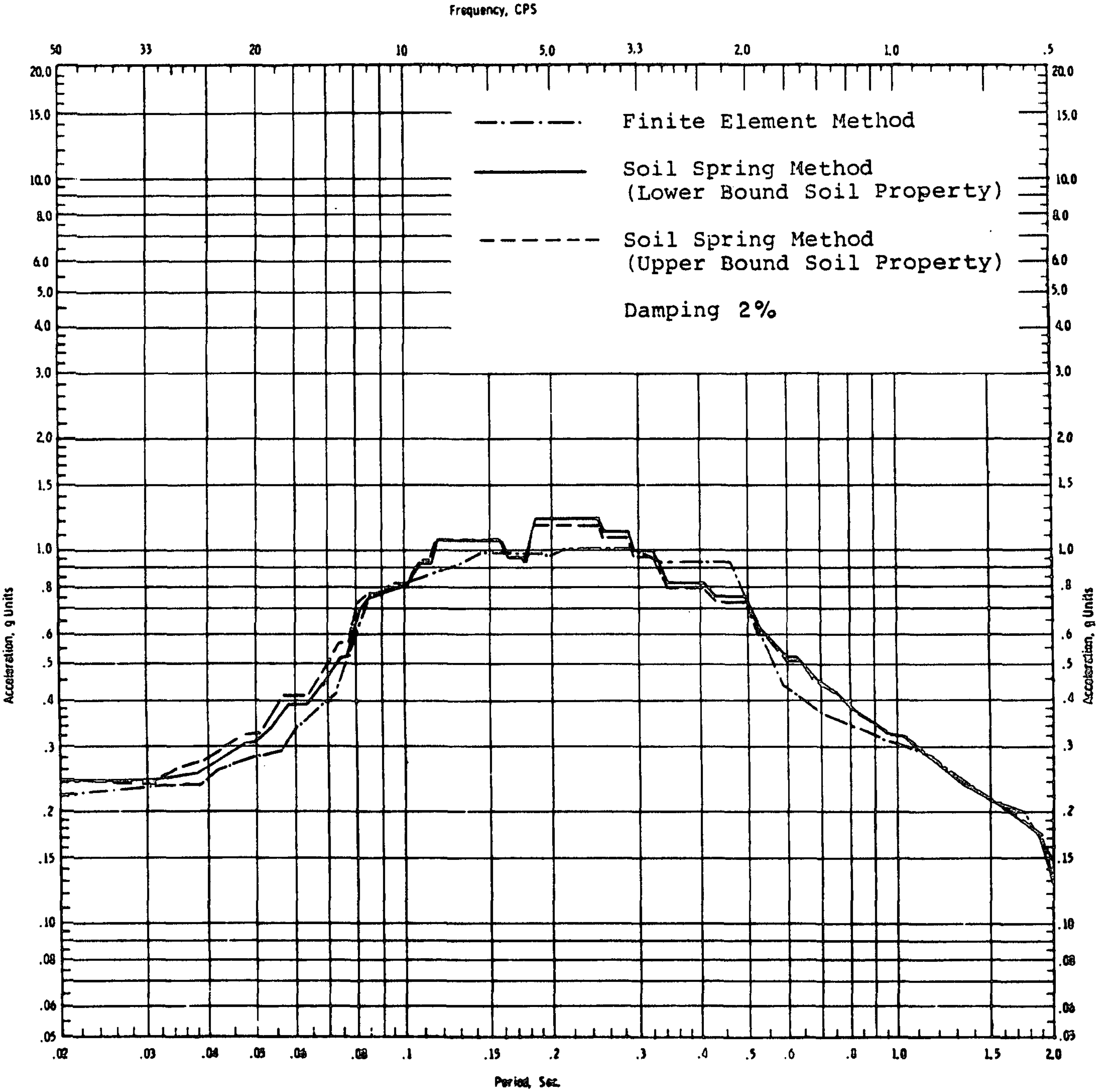

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FIGURE Q130.9-18

RIVER SCREEN HOUSE VERTICAL
SSE SPECTRA AT EL. 664'-0"

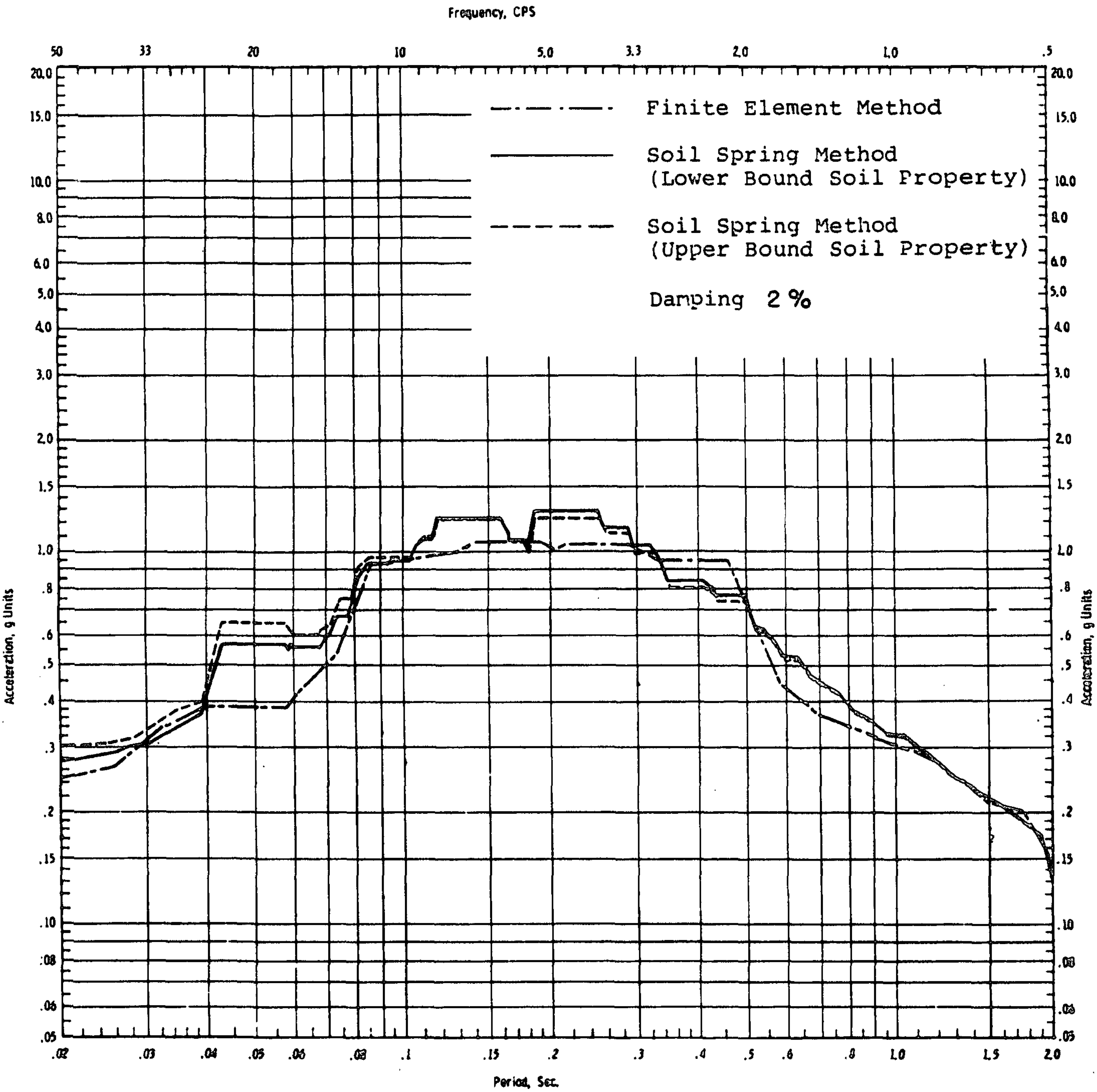

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FIGURE Q130.9-19

RIVER SCREEN HOUSE VERTICAL
SSE SPECTRA AT EL. 702'-0"

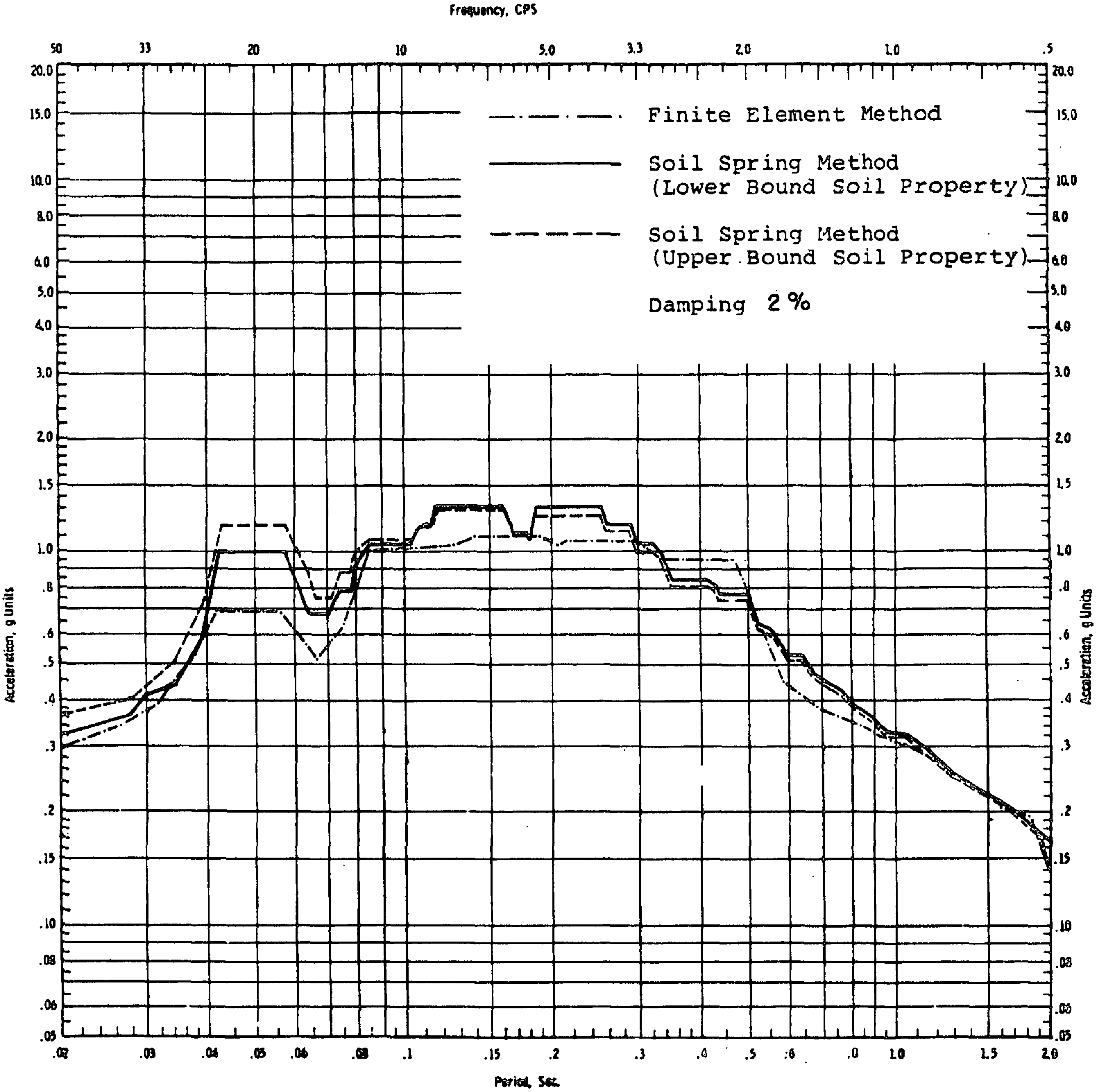

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FIGURE Q130.9-20

RIVER SCREEN HOUSE VERTICAL
SSE SPECTRA AT EL. 744'-0"

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FIGURE 3.8-1

CONTAINMENT STRUCTURE CROSS SECTION

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FIGURE 3.8-2 REINFORCING DETAILS IN CONTAINMENT BASE SLAB AND WALL-BASE SLAB JUNCTION

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FIGURE 3.8-3

REINFORCING DETAILS IN CONTAINMENT
DOME AND DOME RING

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FIGURE 3.8-4

TYPICAL BUTTRESS SECTION REINFORCING DETAILS AND TENDON ANCHORAGES

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FIGURE 3.8-5

PARTIAL SECTION OF CONTAINMENT WALL
REINFORCING DETAILS AROUND
EQUIPMENT HATCH

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FIGURE 3.8-6

PARTIAL PLAN OF CONTAINMENT WALL
REINFORCING DETAILS AROUND
EQUIPMENT HATCH

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FIGURE 3.8-7

CONTAINMENT BASE SLAB BOTTOM
REINFORCING DETAILS

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FIGURE 3.8-8

CONTAINMENT BASE SLAB TOP
REINFORCING DETAILS

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FIGURE 3.8-9

ANCHORAGE DETAILS OF CONTAINMENT
INTERIOR STRUCTURES THROUGH
BASE SLAB LINER PLATE

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FIGURE 3.8-10 CONTAINMENT WALL REINFORCING DETAILS AT CRANE BRACKET

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FIGURE 3.8-11

ARRANGEMENT OF VERTICAL, HOOP, AND DOME TENDONS

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FIGURE 3.8-12

ARRANGEMENT OF CONTAINMENT WALL
HOOP TENDONS

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FIGURE 3.8-13

TYPICAL DETAIL OF VERTICAL AND HORIZONTAL
LINER PLATE STIFFENERS

REVISION 7
DECEMBER 1998

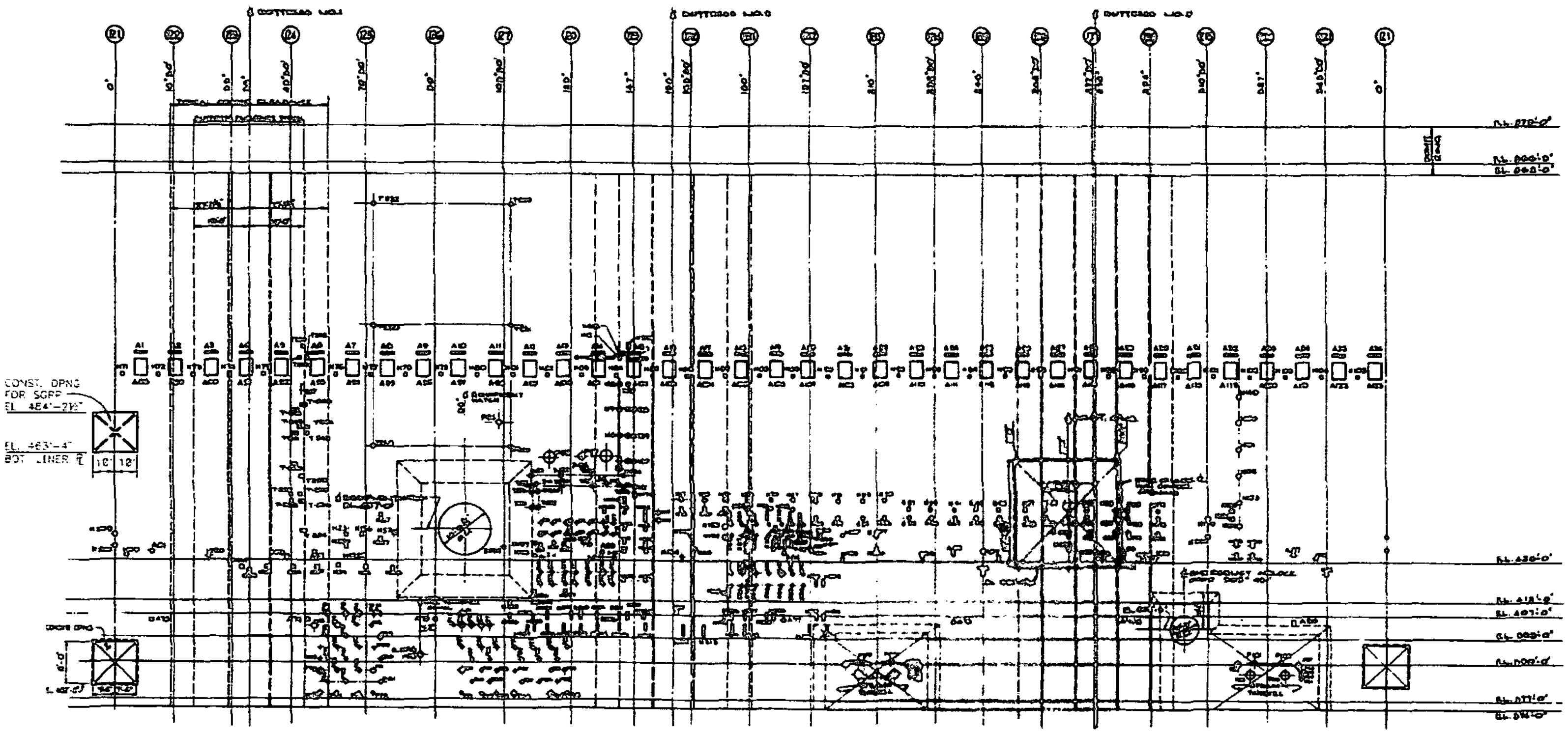

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FIGURE 3.8-14

ARRANGEMENT OF PENETRATION AND EMBEDMENTS IN CONTAINMENT WALL

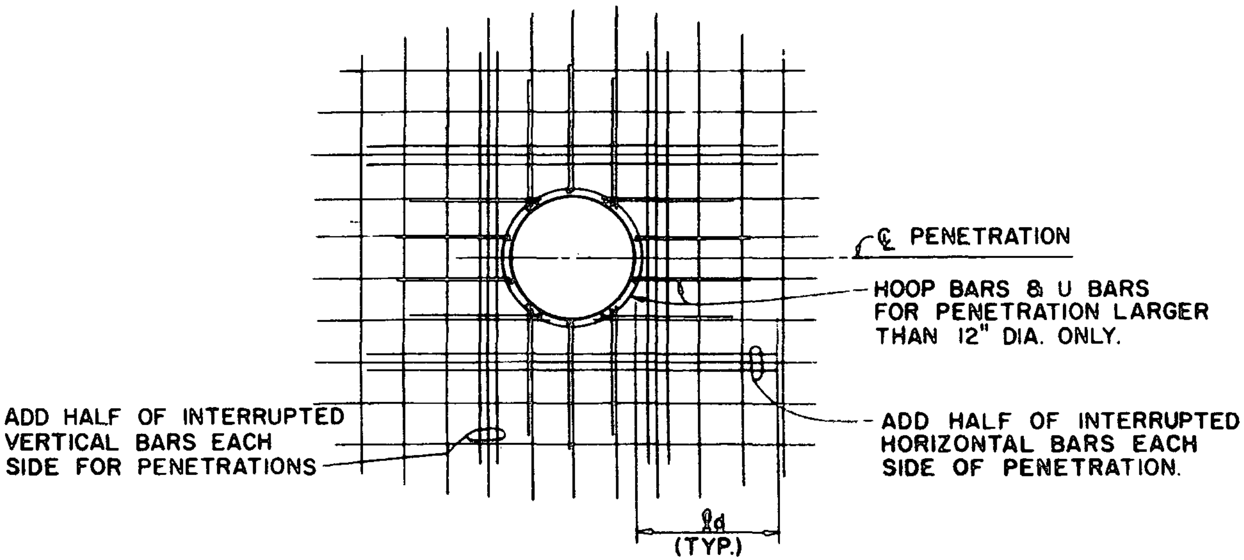

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FIGURE 3.8-15

TYPICAL REINFORCING DETAIL AT PENETRATION

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FIGURE 3.8-16

PARTIAL SECTION CONTAINMENT WALL
REINFORCING DETAILS AROUND MAIN STEAM
PENETRATION

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FIGURE 3.8-17 PARTIAL PLAN CONTAINMENT WALL REINFORCING DETAILS AROUND MAIN STEAM PENETRATION

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FIGURE 3.8-18 CONTAINMENT WALL REINFORCING AT PERSONNEL AIR LOCK

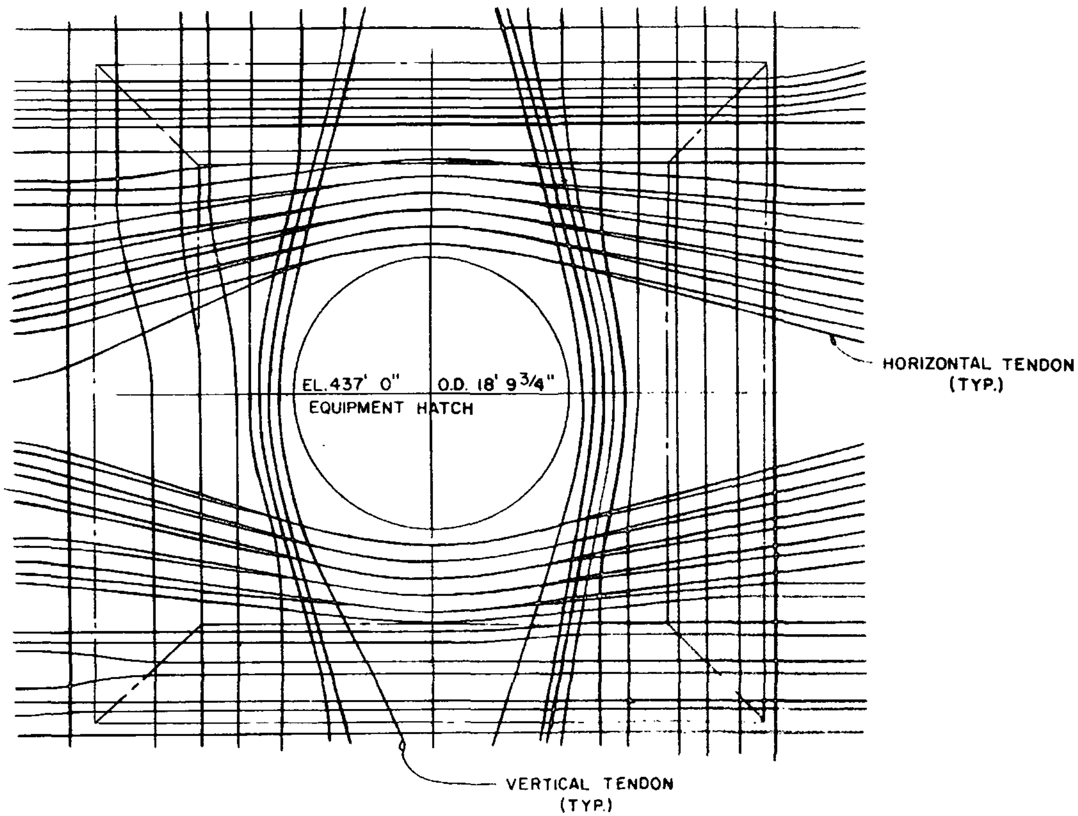

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FIGURE 3.8-19

ARRANGEMENT OF TENDONS AROUND EQUIPMENT HATCH

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FIGURE 3.8-20

CONTAINMENT ROOF PLAN—
DOME REINFORCING DETAILS

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FIGURE 3.8-21

CONTAINMENT ROOF PLAN—
ARRANGEMENT OF DOME TENDONS

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FIGURE 3.8-22

PARTIAL PLAN VIEW SHOWING DOME LINER STIFFENERS

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FIGURE 3.8-23

DOME LINER STIFFENERS

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FIGURE 3.8-24 SOR III THIN SHELL MODEL

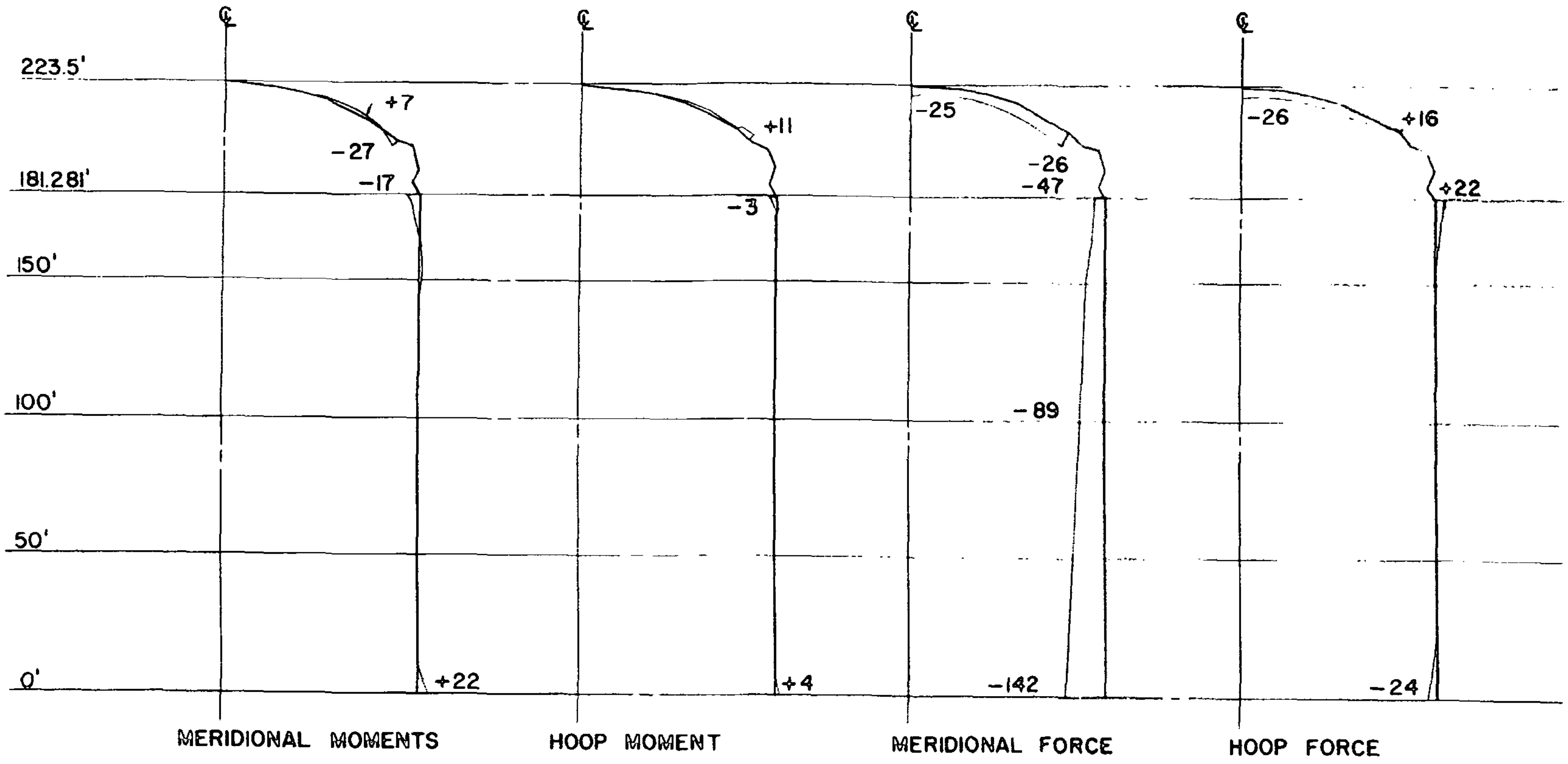

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FIGURE 3.8-25

CONTAINMENT MOMENTS AND FORCES—
DEAD LOAD

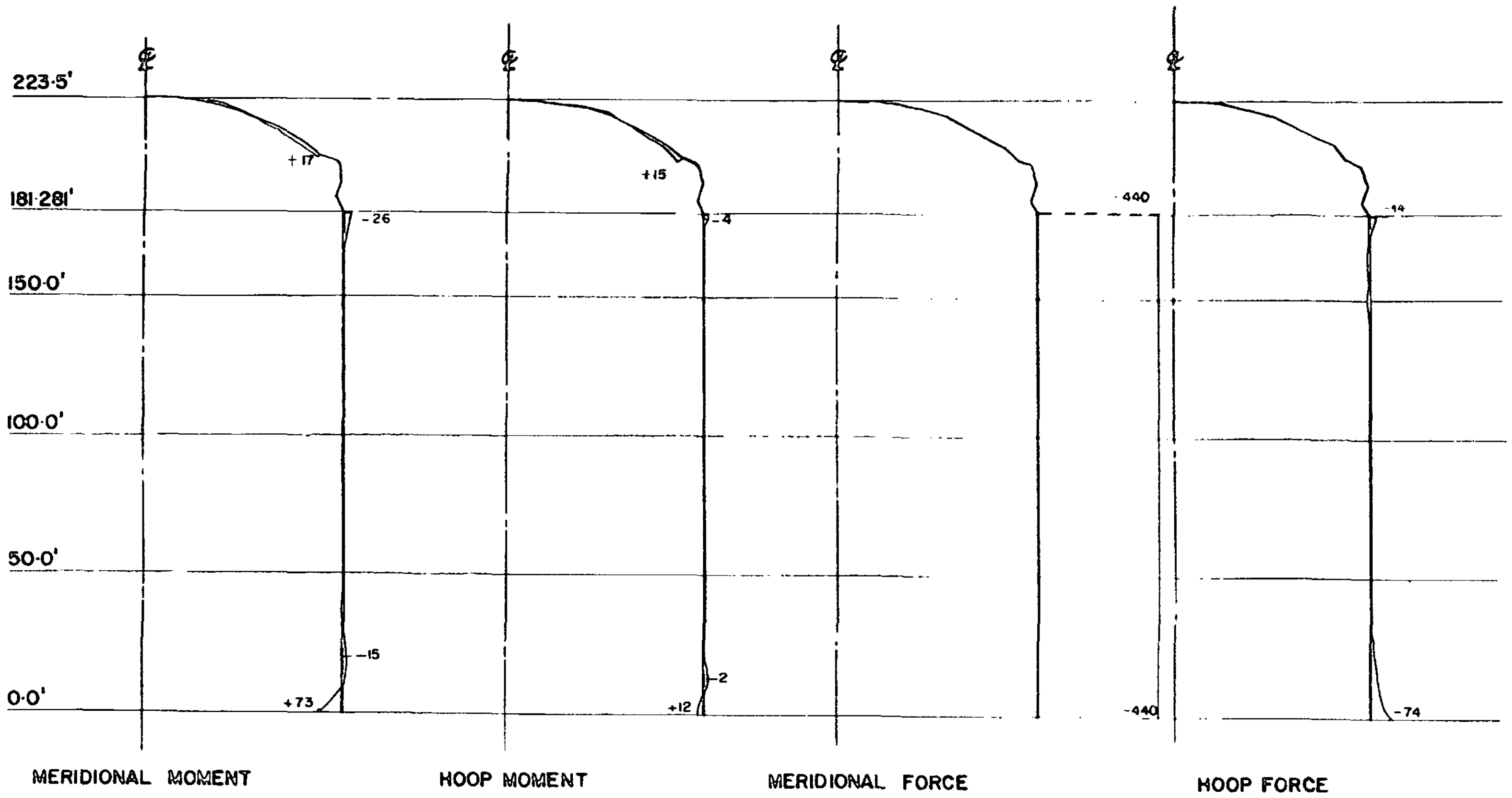

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FIGURE 3.8-26

CONTAINMENT MOMENTS AND FORCES—
VERTICAL PRESTRESSING

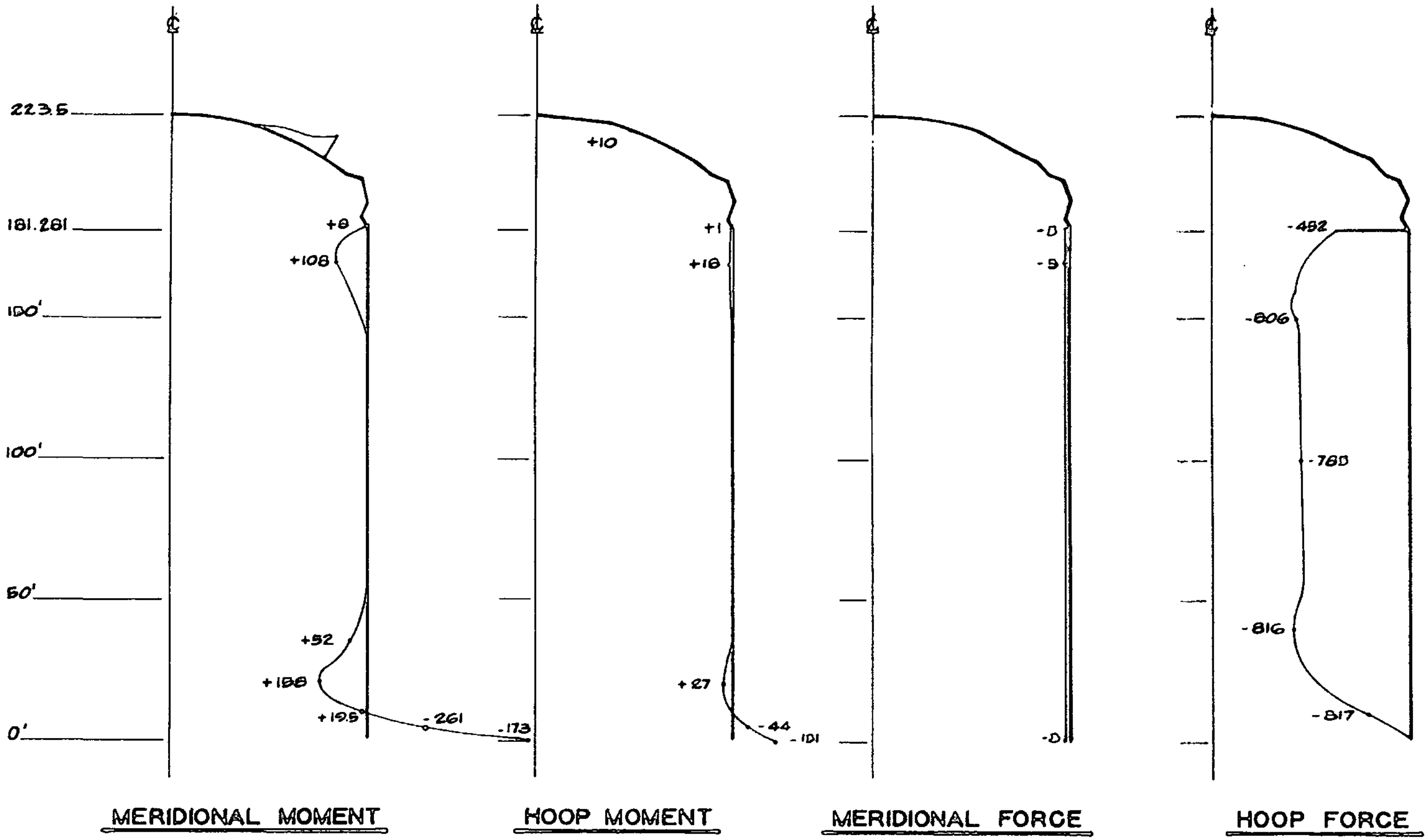

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FIGURE 3.8-27

CONTAINMENT MOMENTS AND FORCES—
HOOP PRESTRESSING

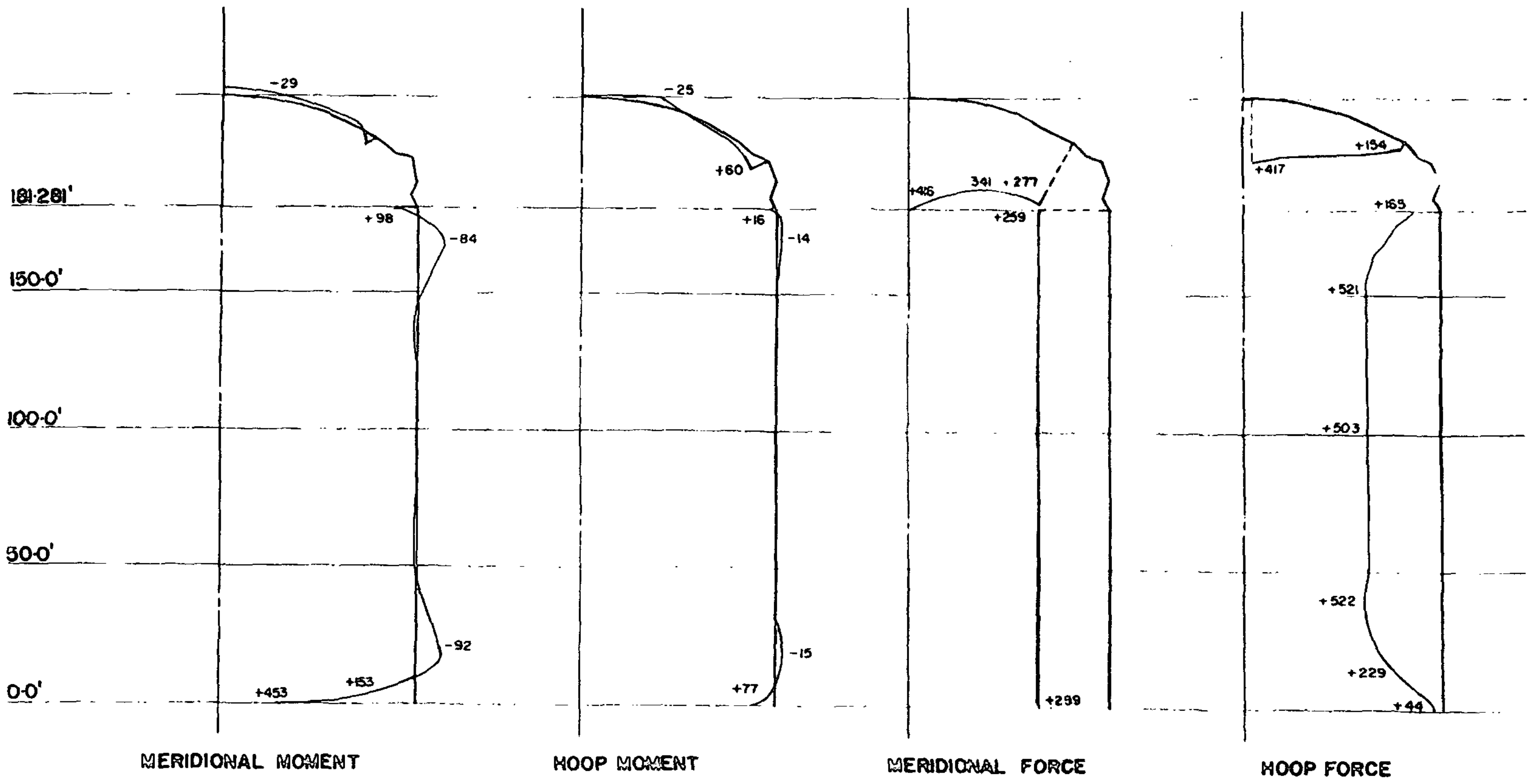

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FIGURE 3.8-2b

CONTAINMENT MOMENTS AND FORCES—
INTERNAL PRESSURE

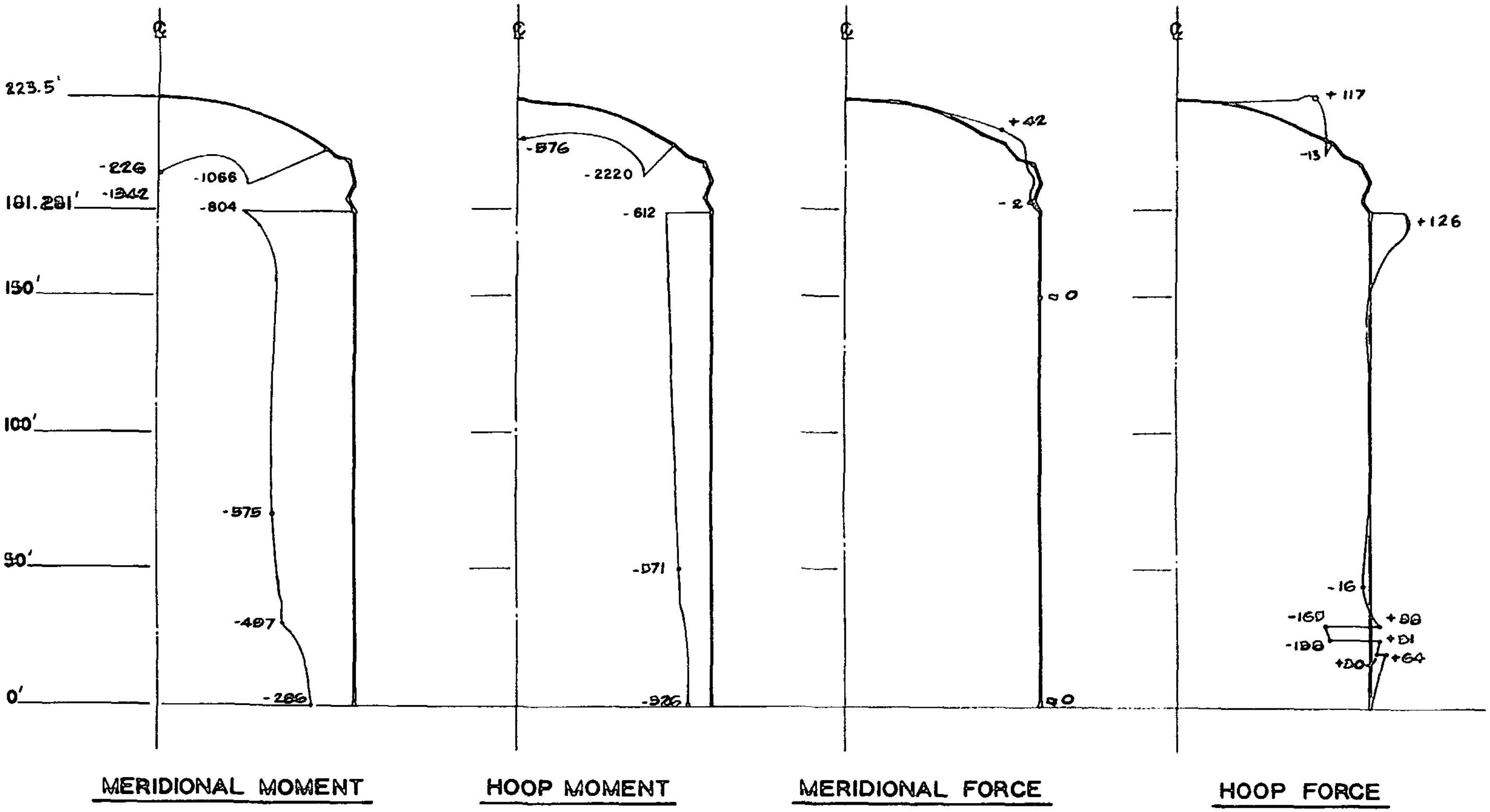

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FIGURE 3.8-29

CONTAINMENT MOMENTS AND FORCES—
WINTER GRADIENT TEMPERATURE

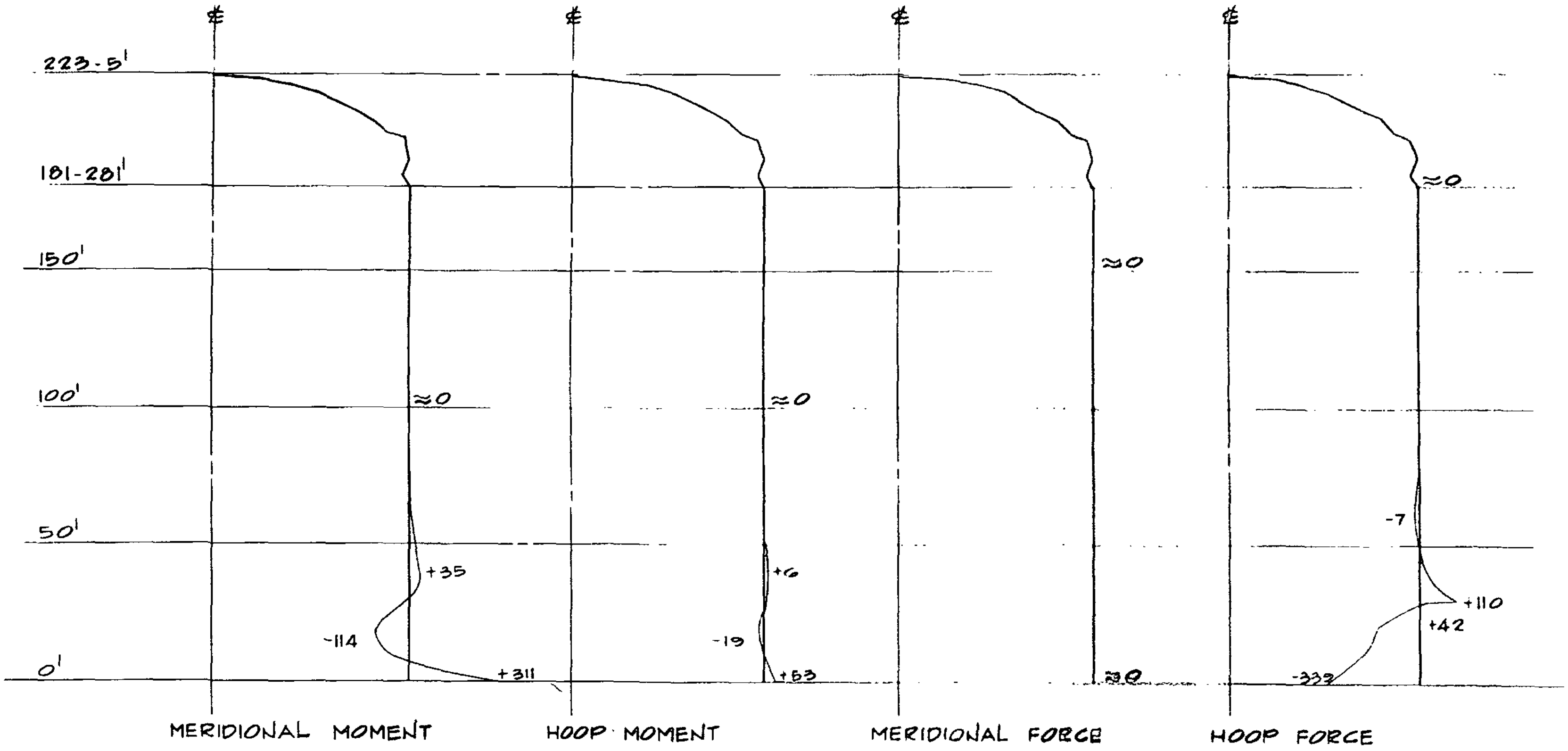

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FIGURE 3.8-30

CONTAINMENT MOMENTS AND FORCES—
WINTER AXIAL TEMPERATURE

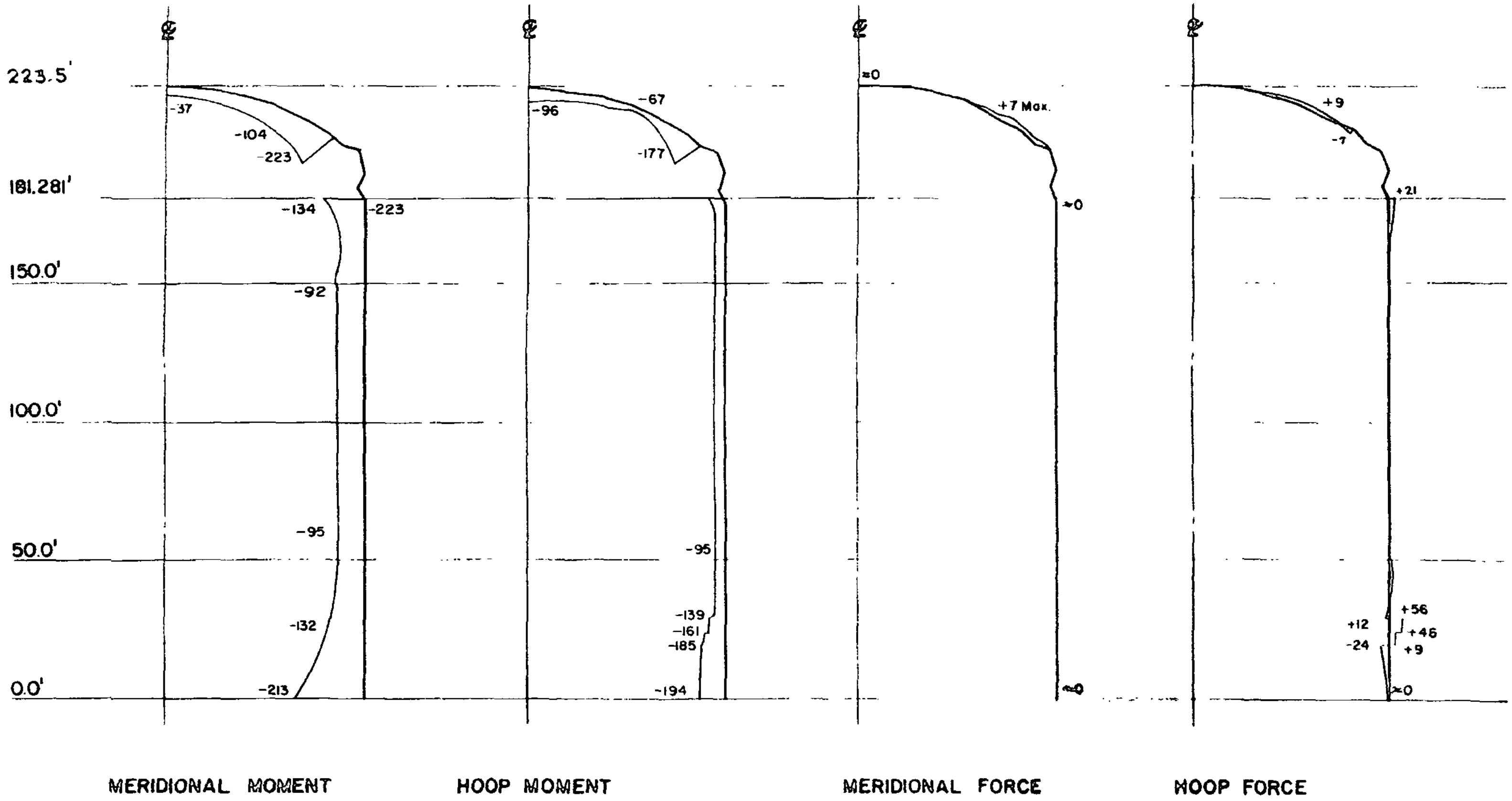

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FIGURE 3.8-31

CONTAINMENT MOMENTS AND FORCES—
SUMMER GRADIENT TEMPERATURE

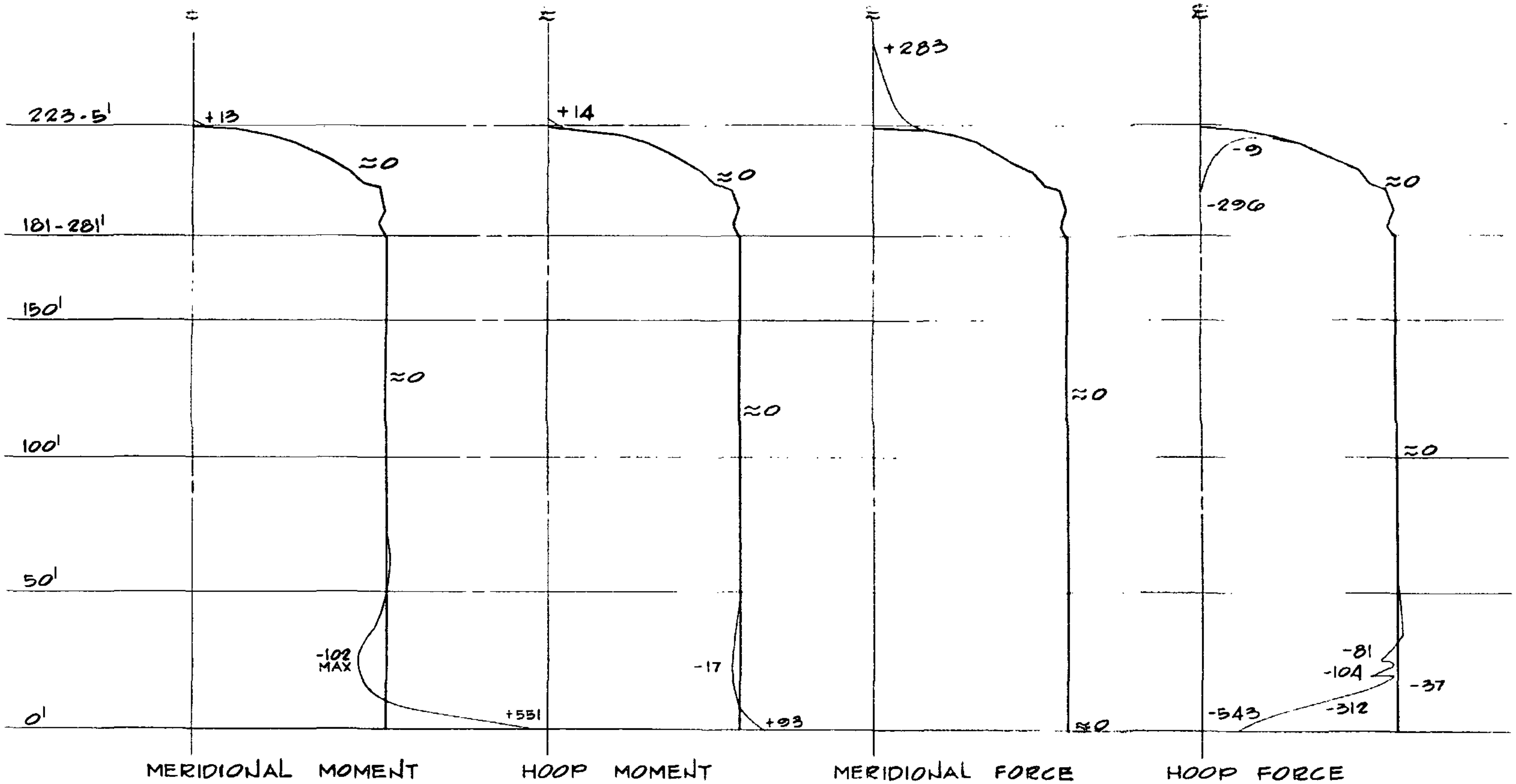

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FIGURE 3.8-32

CONTAINMENT MOMENTS AND FORCES—
SUMMER AXIAL TEMPERATURE

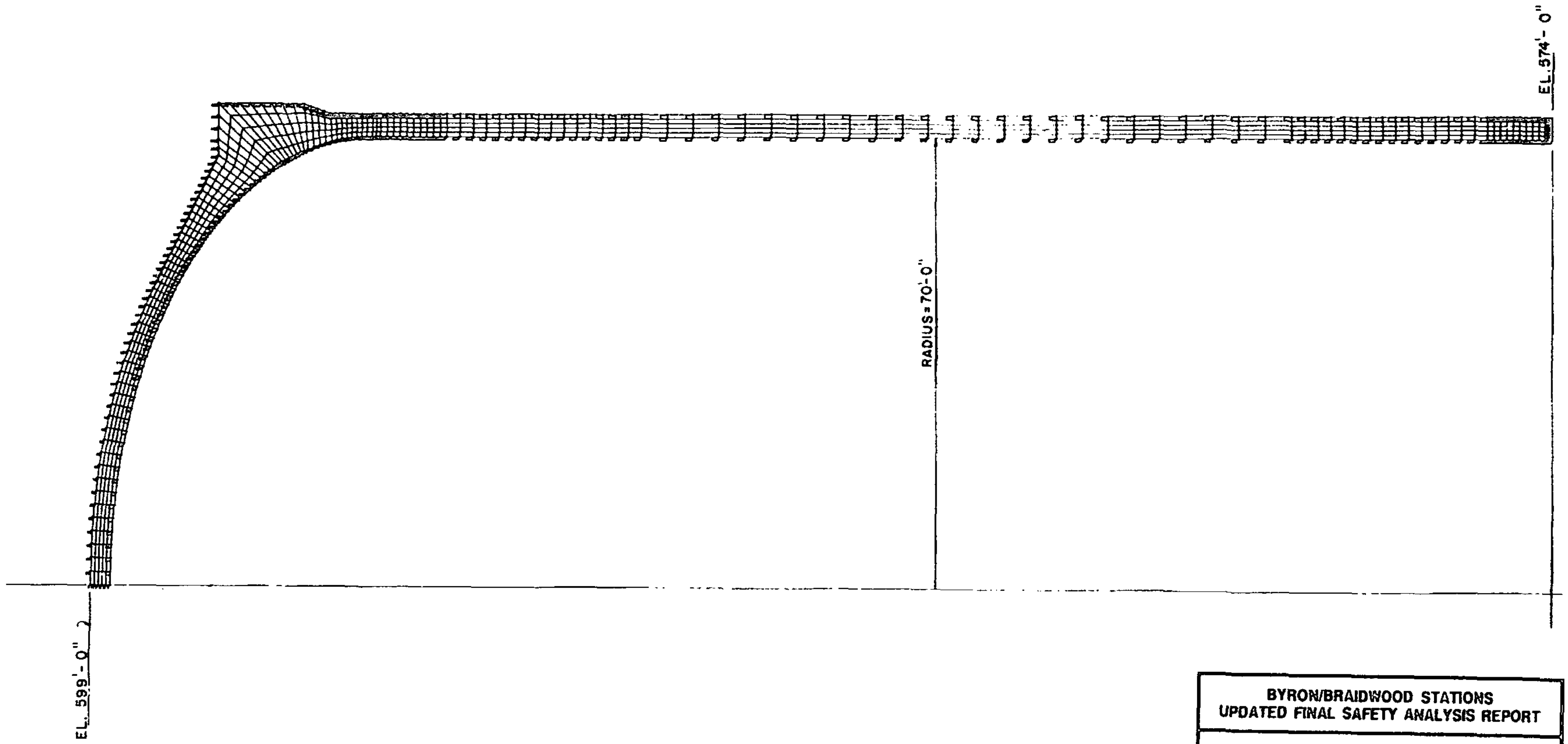

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FIGURE 3.8-33

CONTAINMENT THIN SHELL MODEL

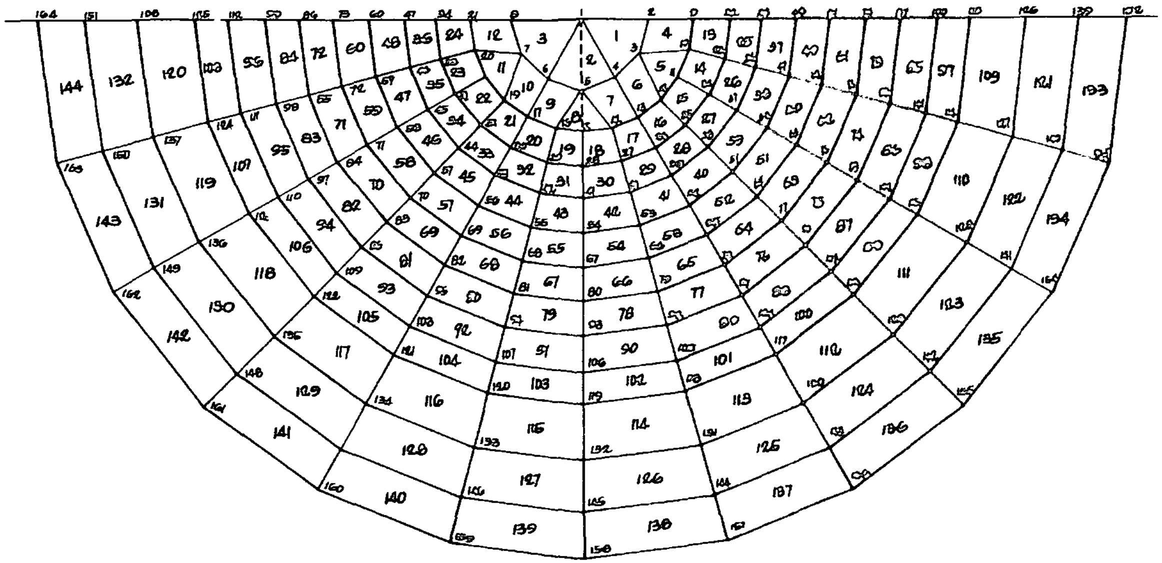

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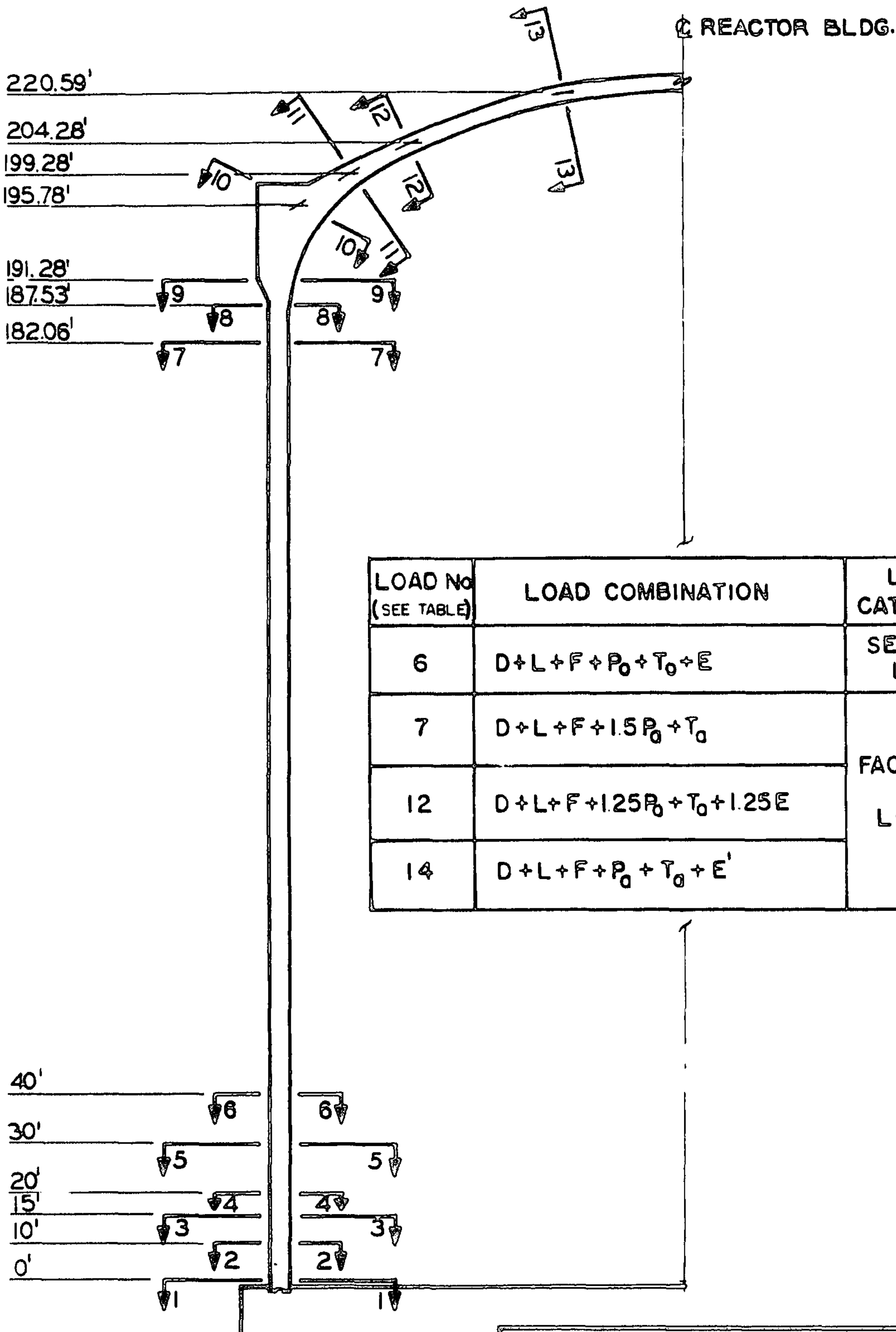
FIGURE 3.8-34

CONTAINMENT BASE SLAB FINITE ELEMENT MODEL



LOAD No (SEE TABLE)	LOAD COMBINATION	LOAD CATEGORY
6	$D + L + F + P_o + T_o + E$	SERVICE LOADS
7	$D + L + F + 1.5 P_a + T_a$	FACTORED LOADS
12	$D + L + F + 1.25 P_a + T_a + 1.25 E$	
14	$D + L + F + P_a + T_a + E'$	

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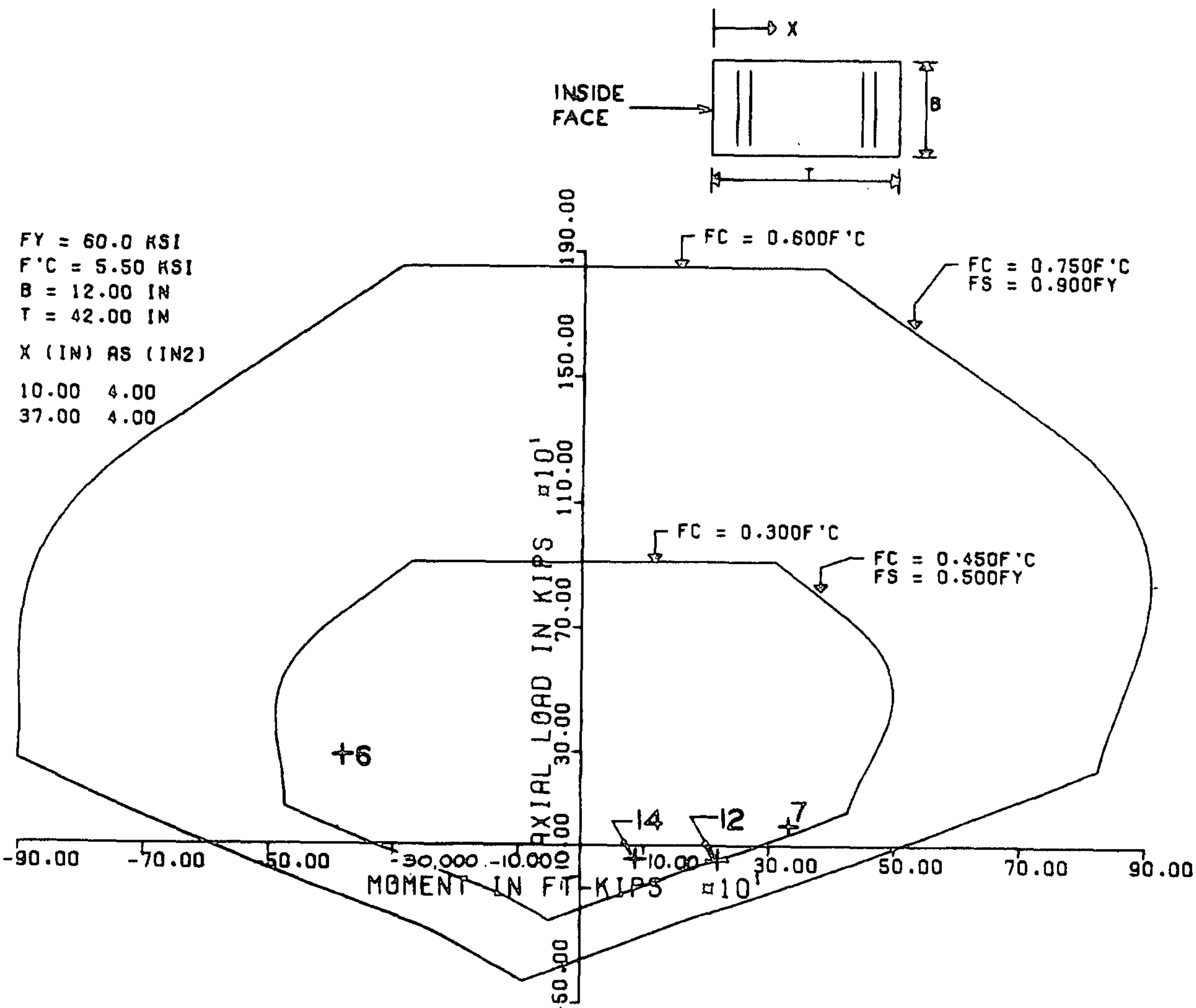
FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 1 OF 27)



SECTION 1 MERIDIONAL REINFORCEMENT

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FIGURE 3.8-35 LOCATION OF SECTIONS FOR INTERACTION DIAGRAMS (SHEET 2 OF 27)

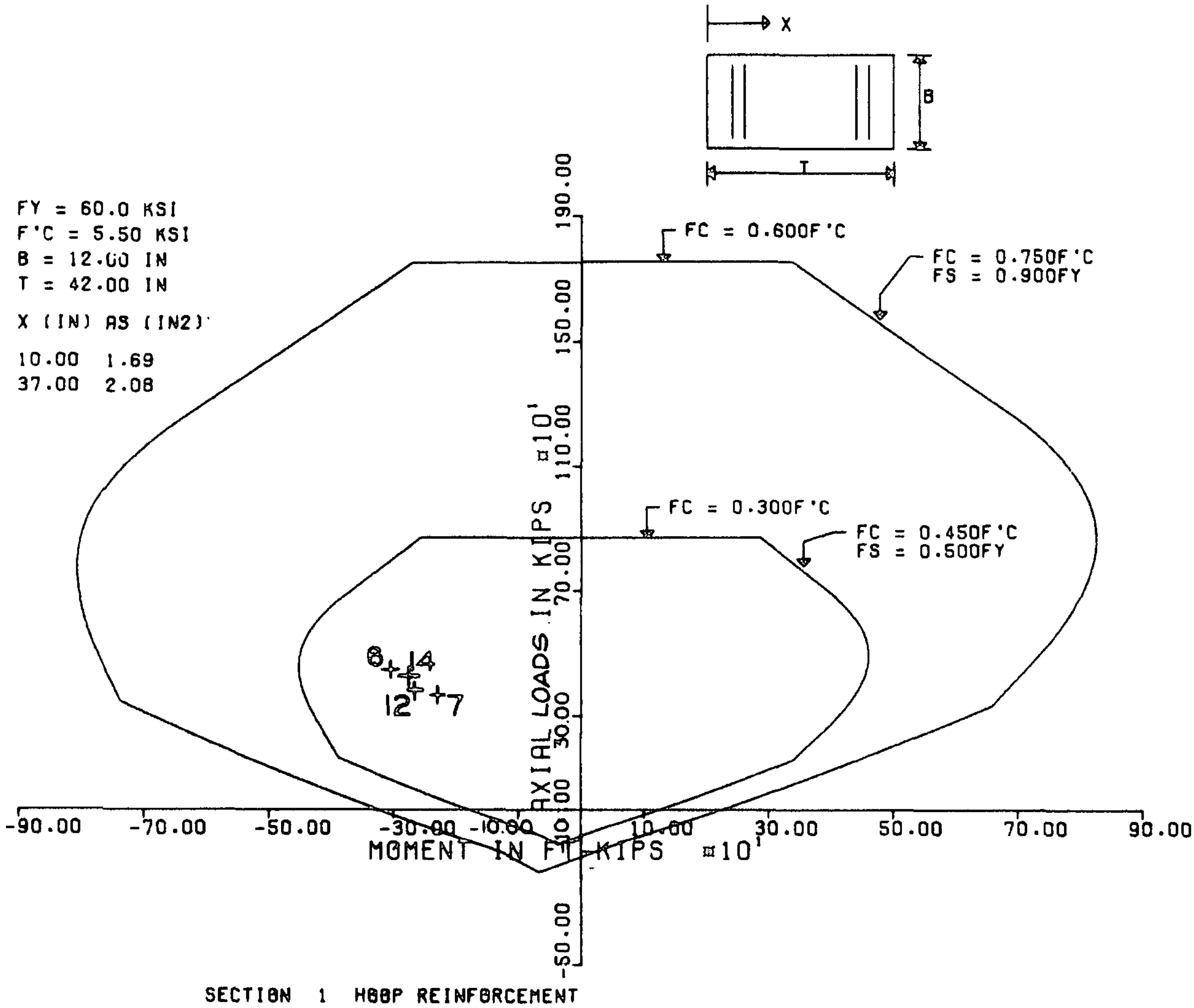

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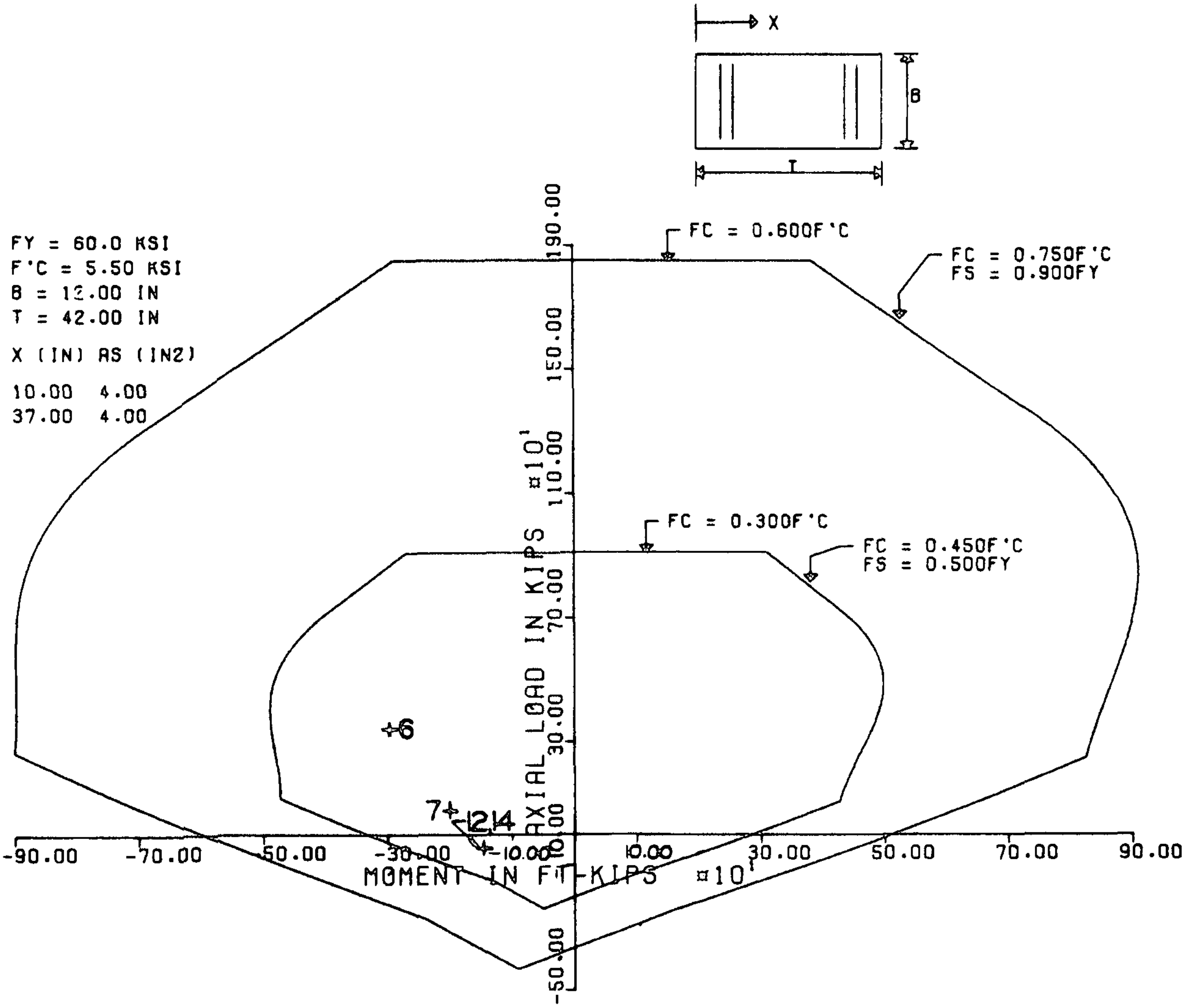
FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 3 OF 27)



SECTION 2 MERIDIONAL REINFORCEMENT

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 4 OF 27)

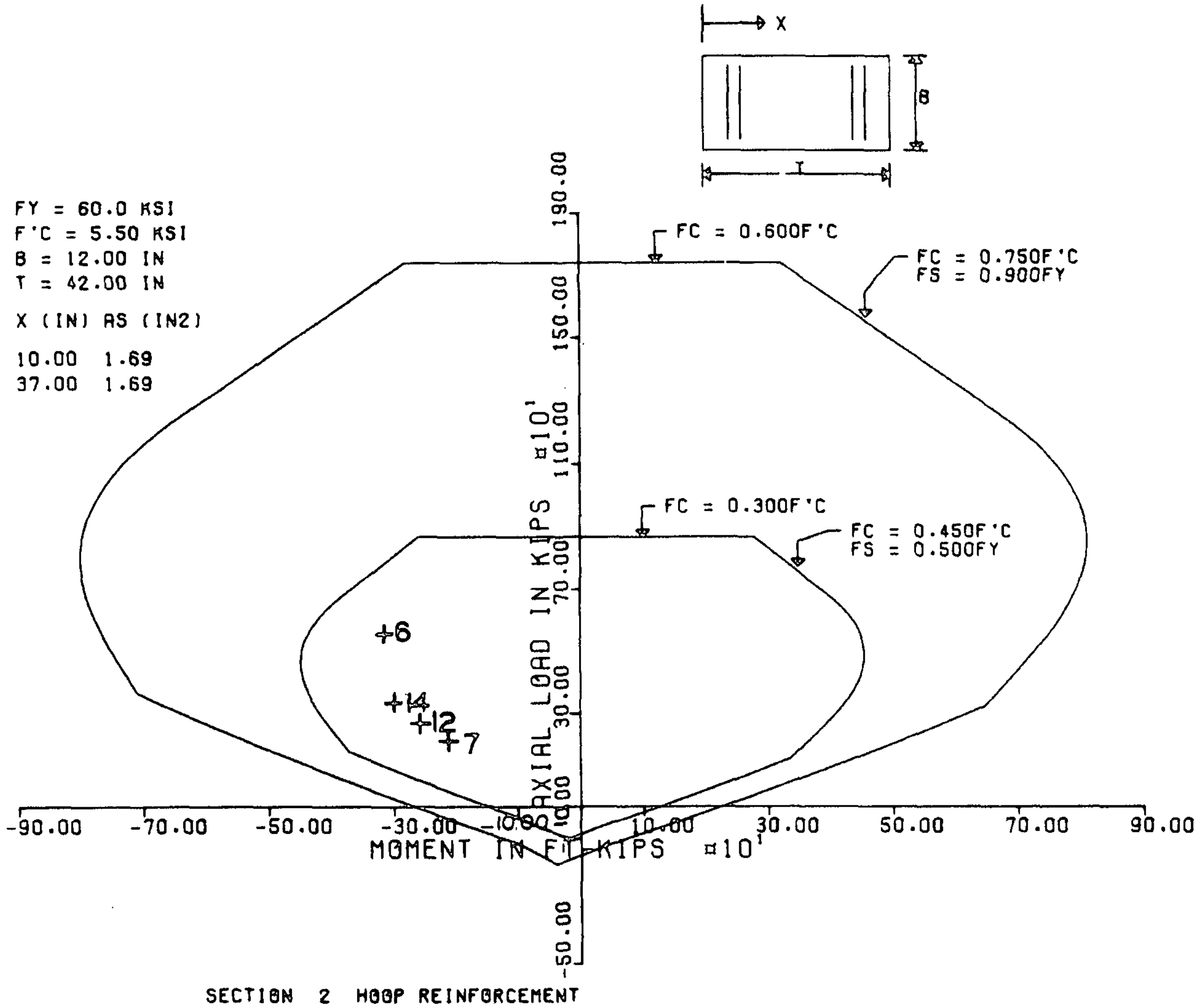

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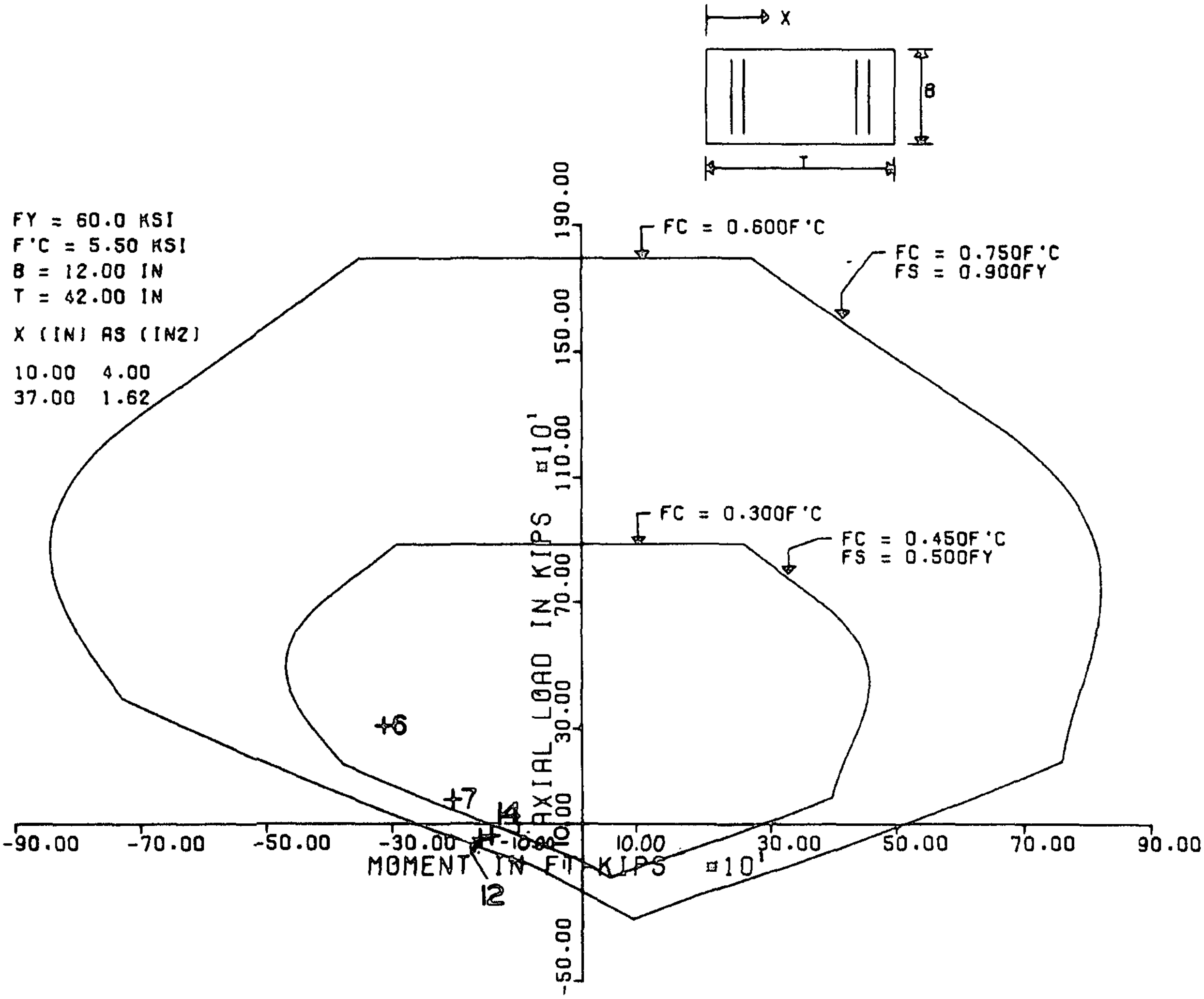
FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 5 OF 27)



SECTION 3 MERIDIONAL REINFORCEMENT

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FIGURE 3.8-35

LOCATION OF SECTIONS FOR INTERACTION DIAGRAMS (SHEET 6 OF 27)

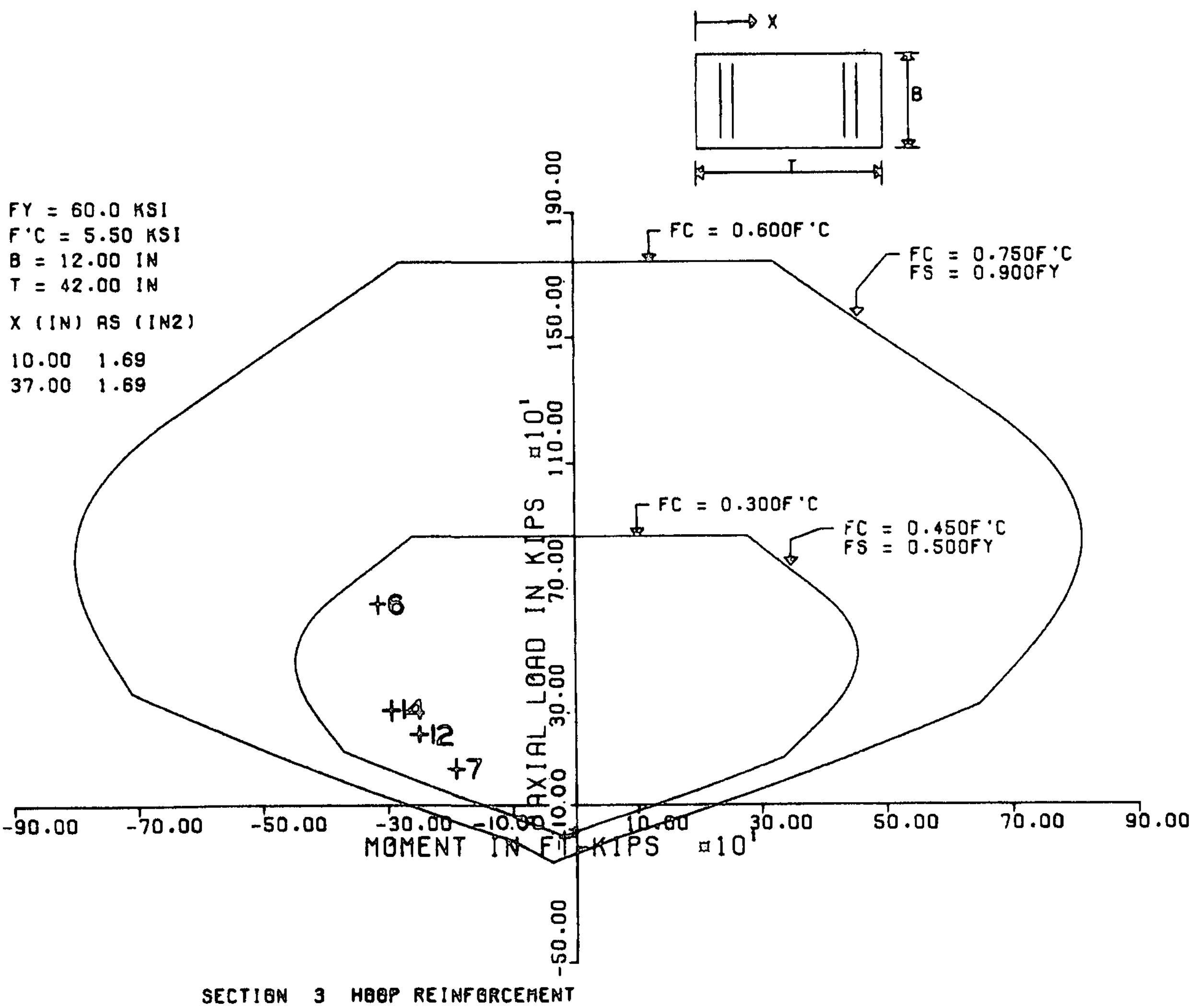

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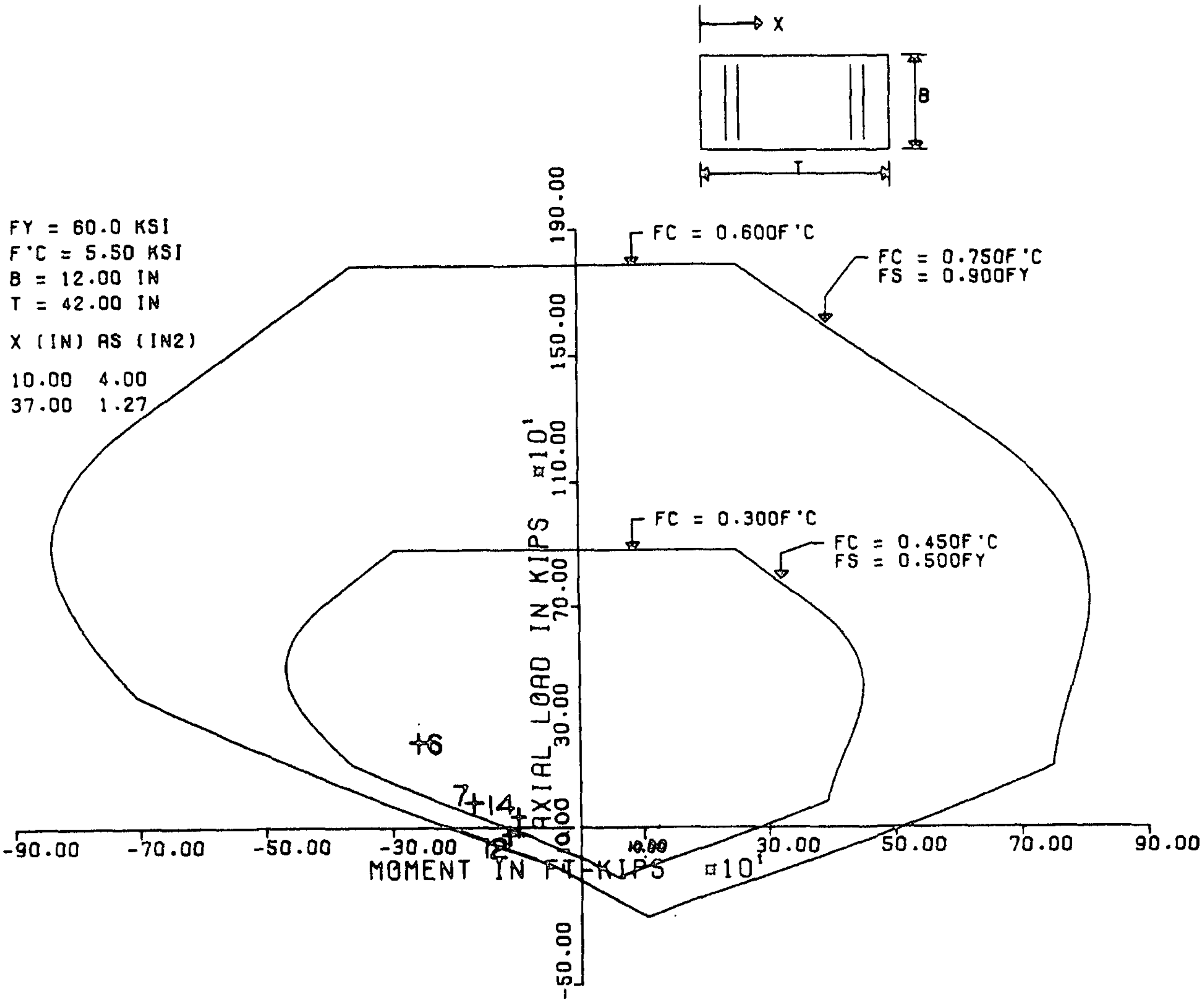
FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 7 OF 27)



SECTION 4 MERIDIONAL REINFORCEMENT

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FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 8 OF 27)

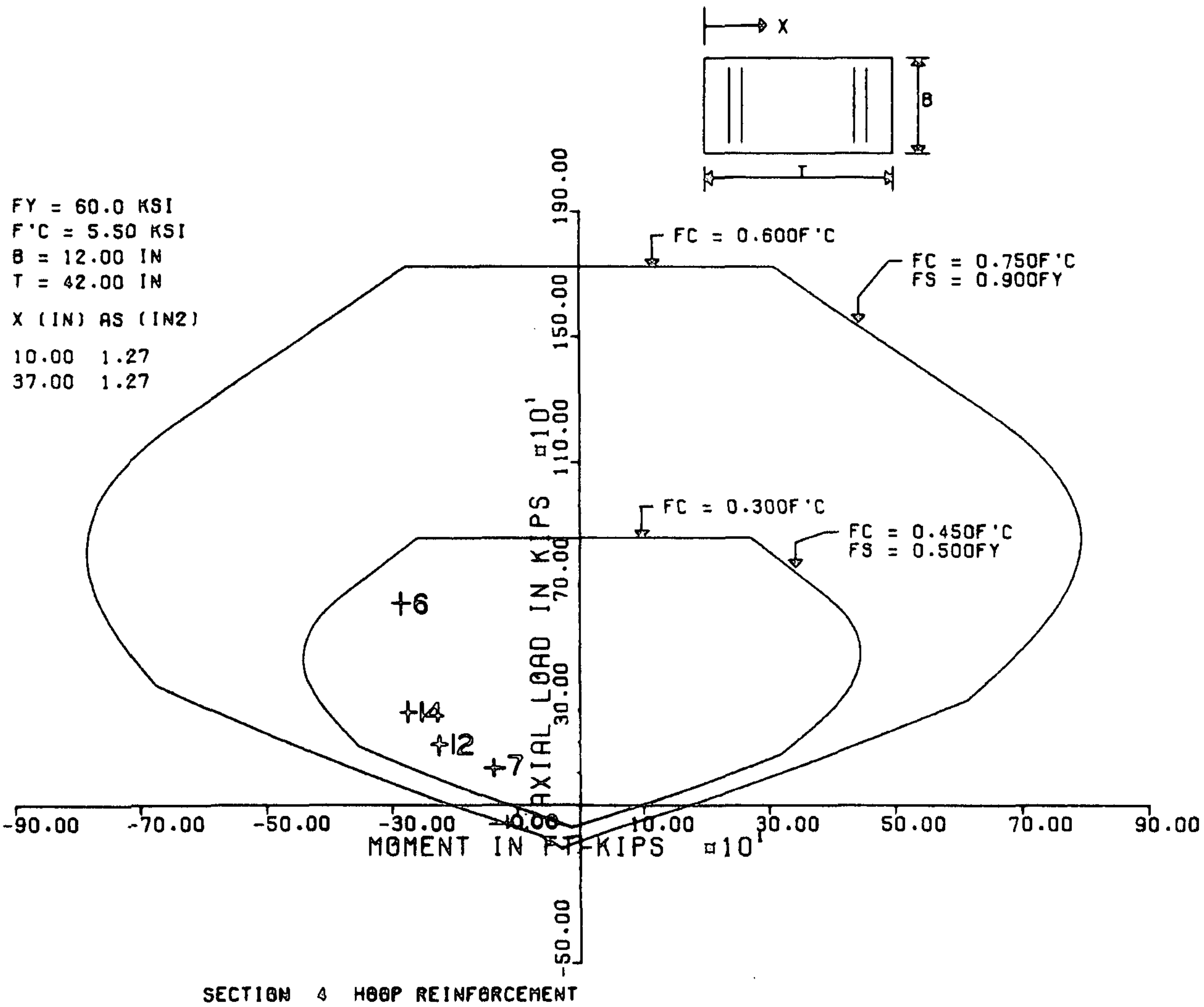

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FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 9 OF 27)

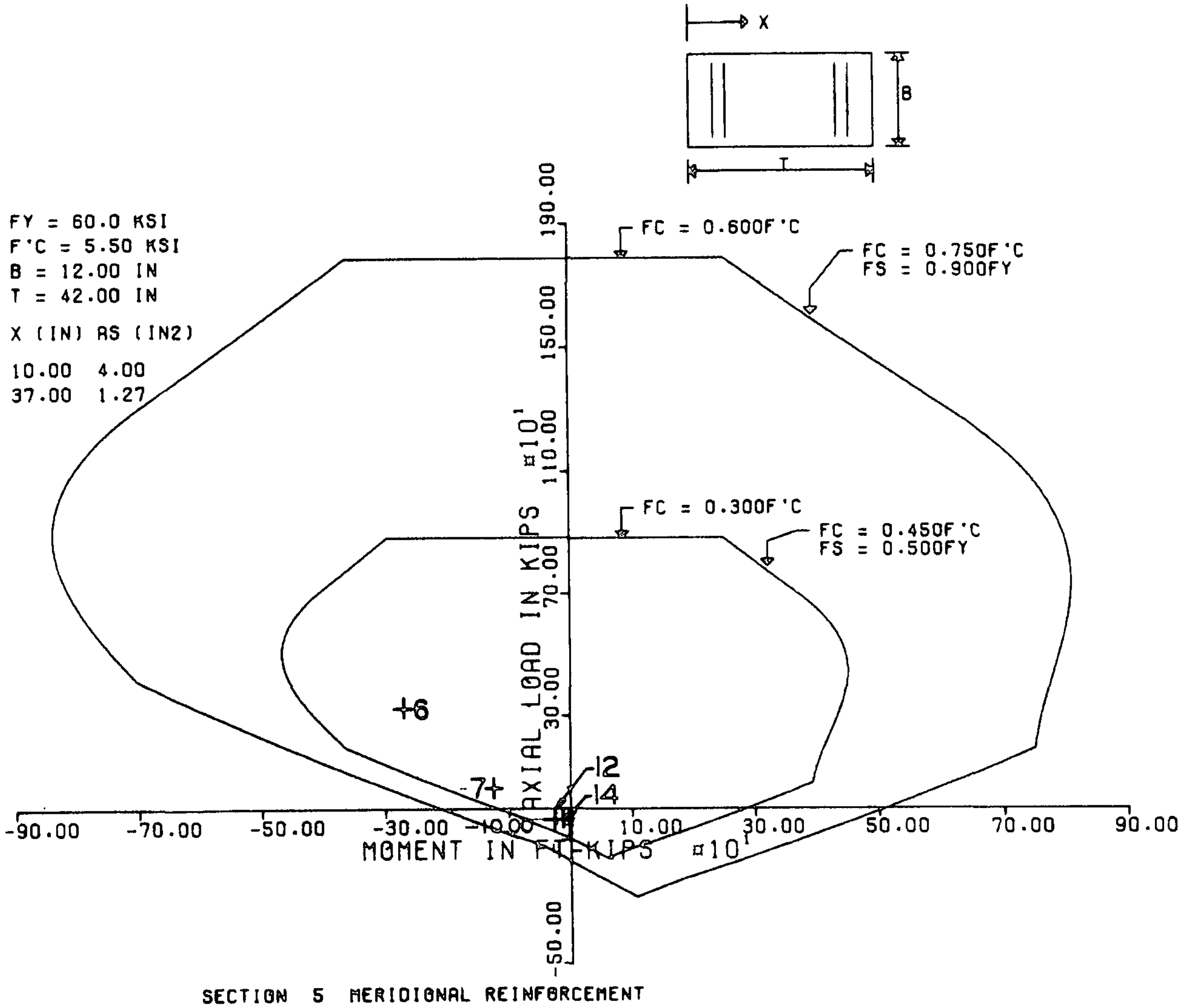

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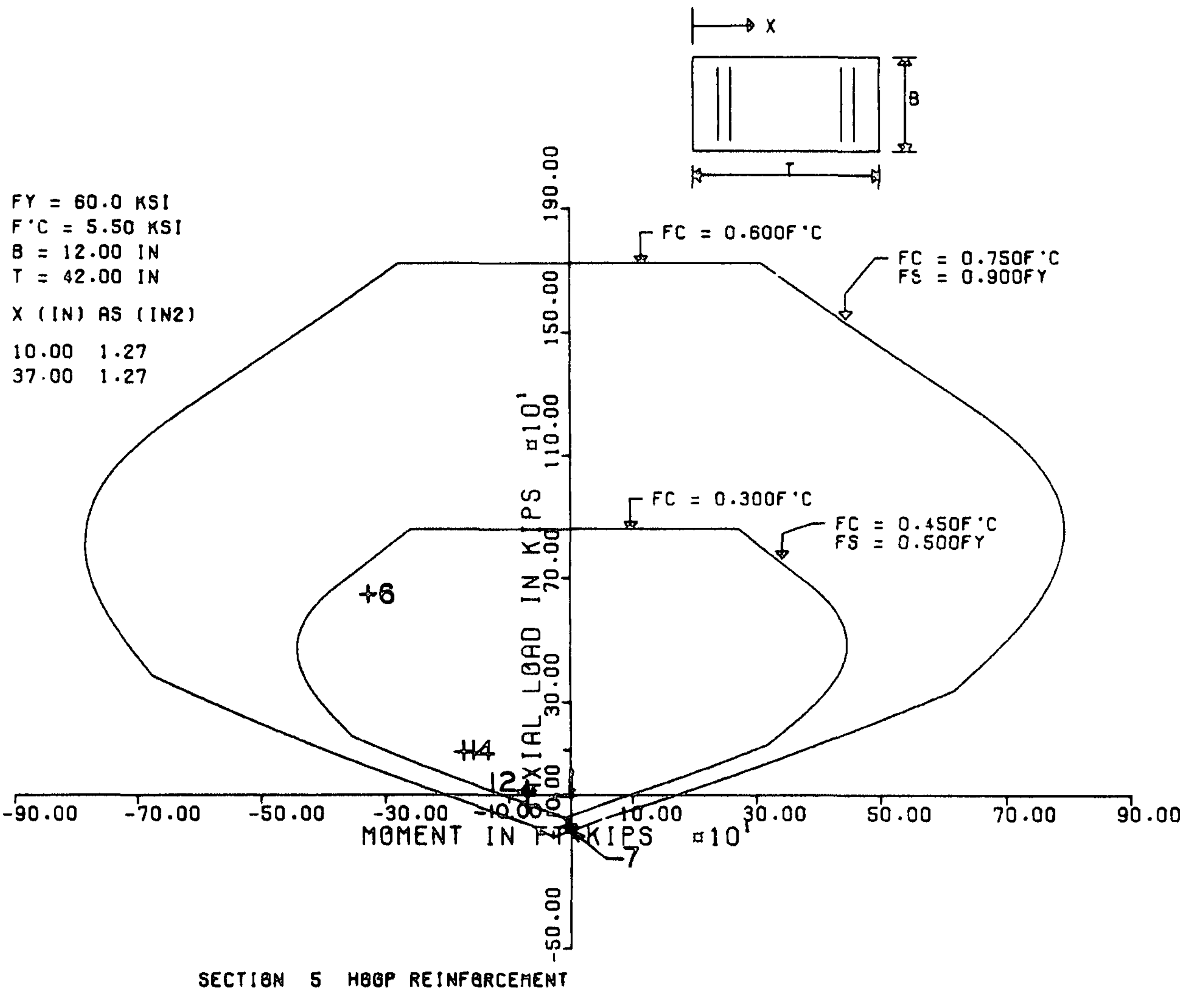
FIGURE 3.8-35

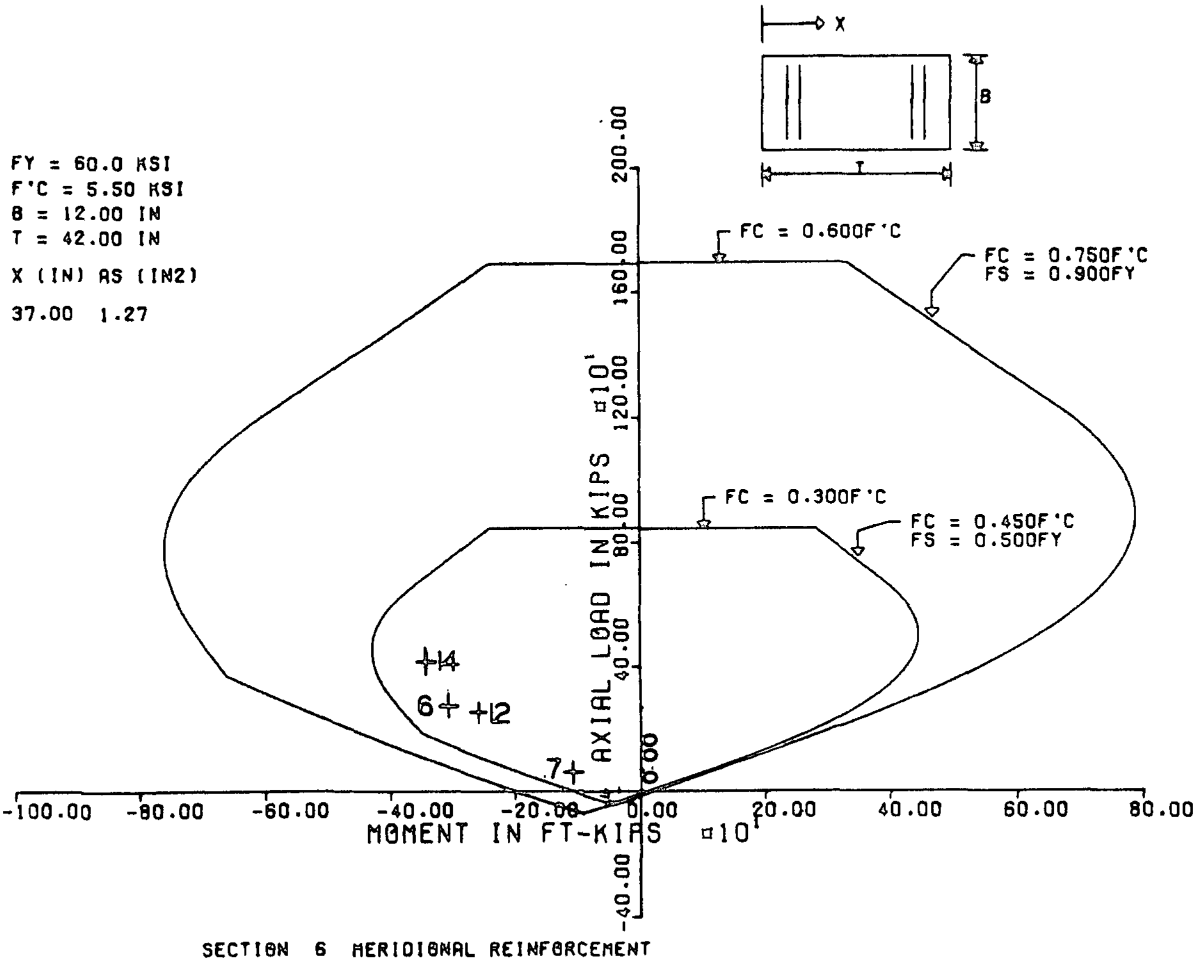
LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 10 OF 27)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

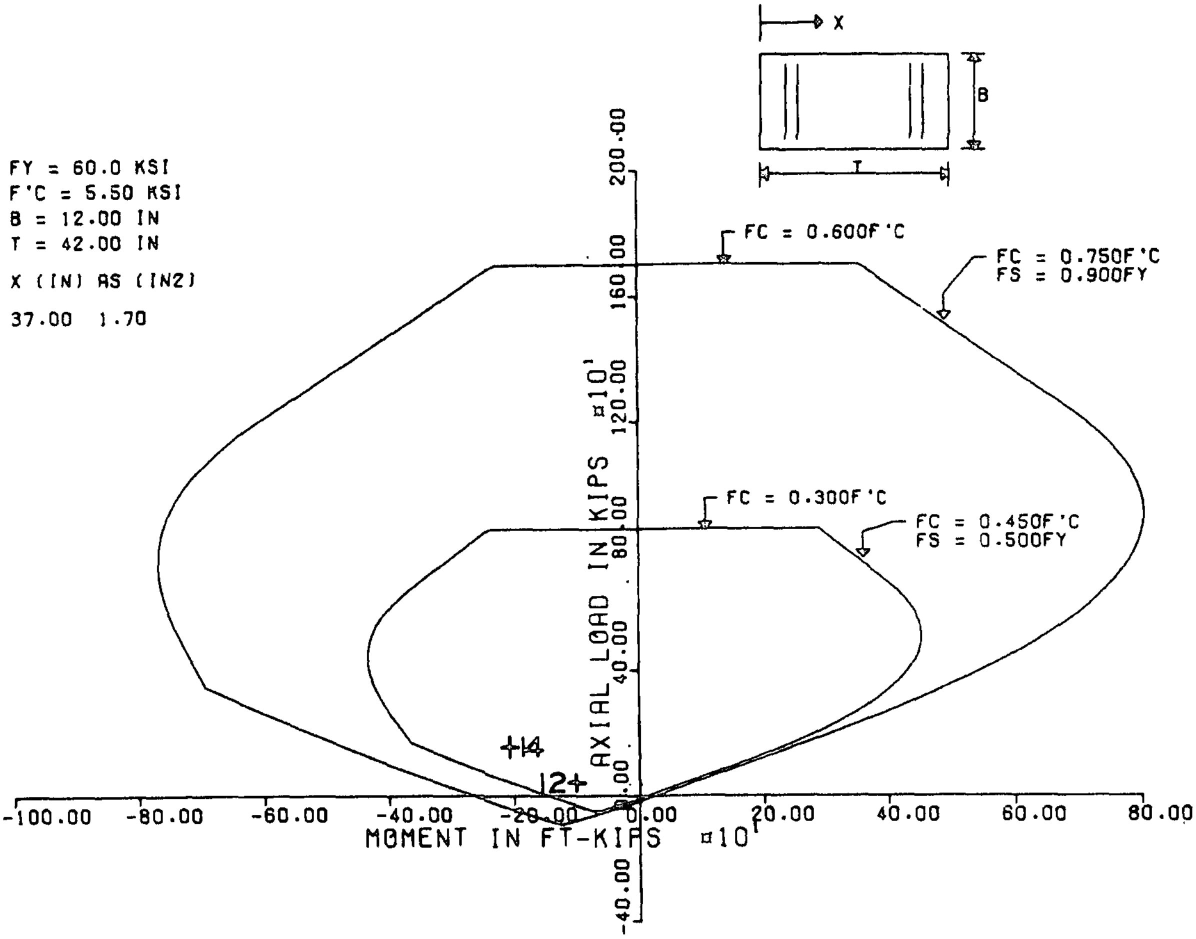
LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 11 OF 27)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

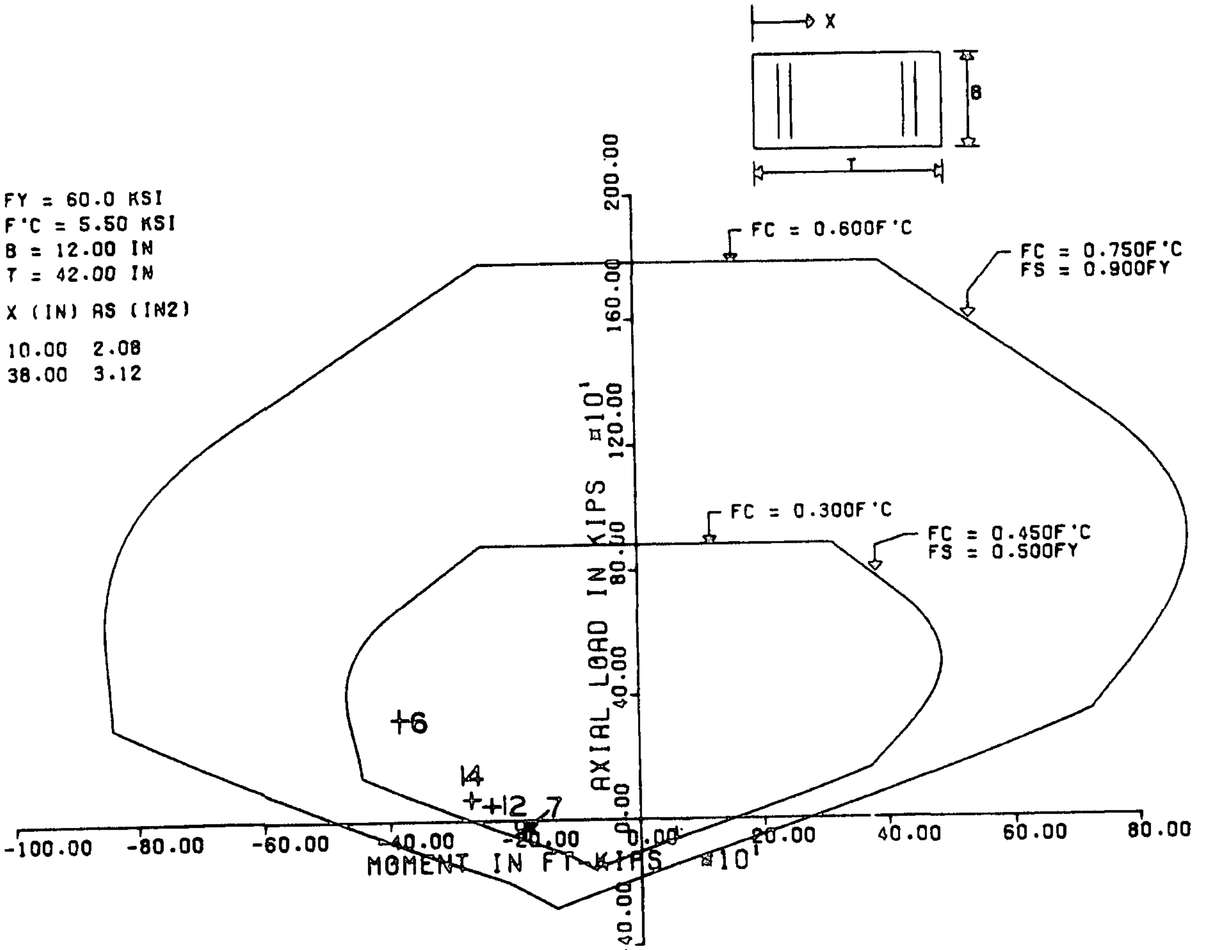
FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 12 OF 27)



SECTION 6 HOOP REINFORCEMENT

BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.8-35 LOCATION OF SECTIONS FOR INTERACTION DIAGRAMS (SHEET 13 OF 27)

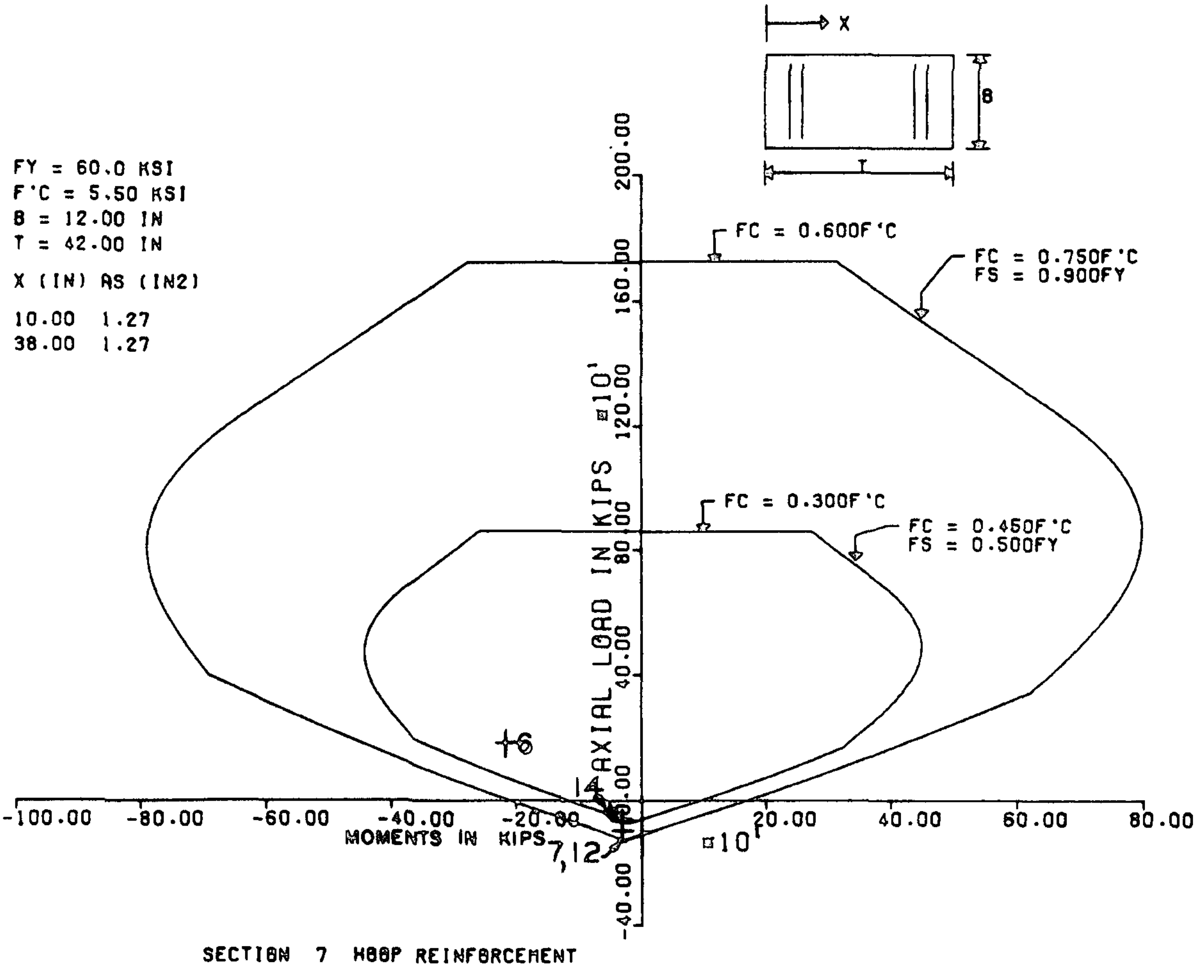


SECTION 7 MERIDIONAL REINFORCEMENT

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

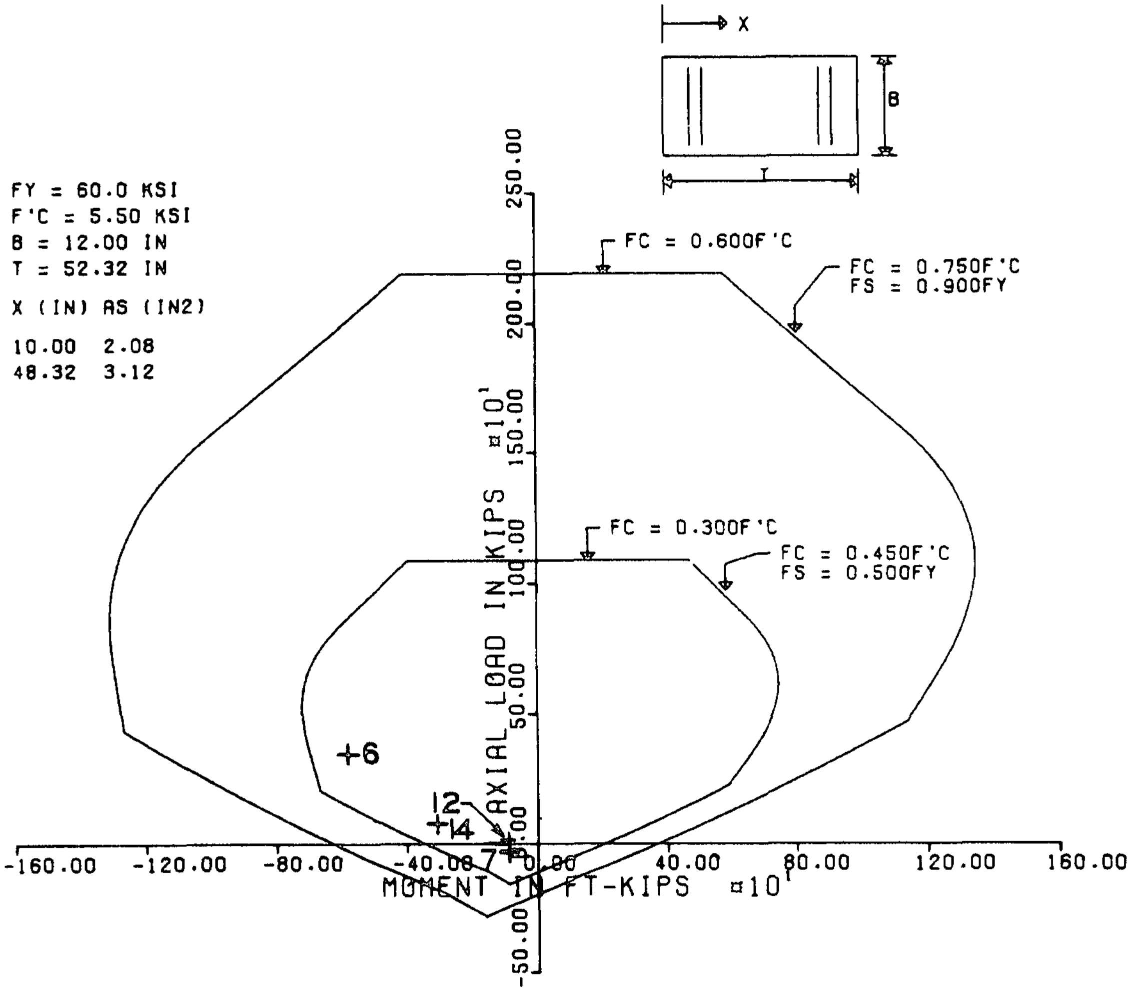
LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 14 OF 27)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

LOCATION OF SECTIONS FOR INTERACTION DIAGRAMS (SHEET 15 OF 27)

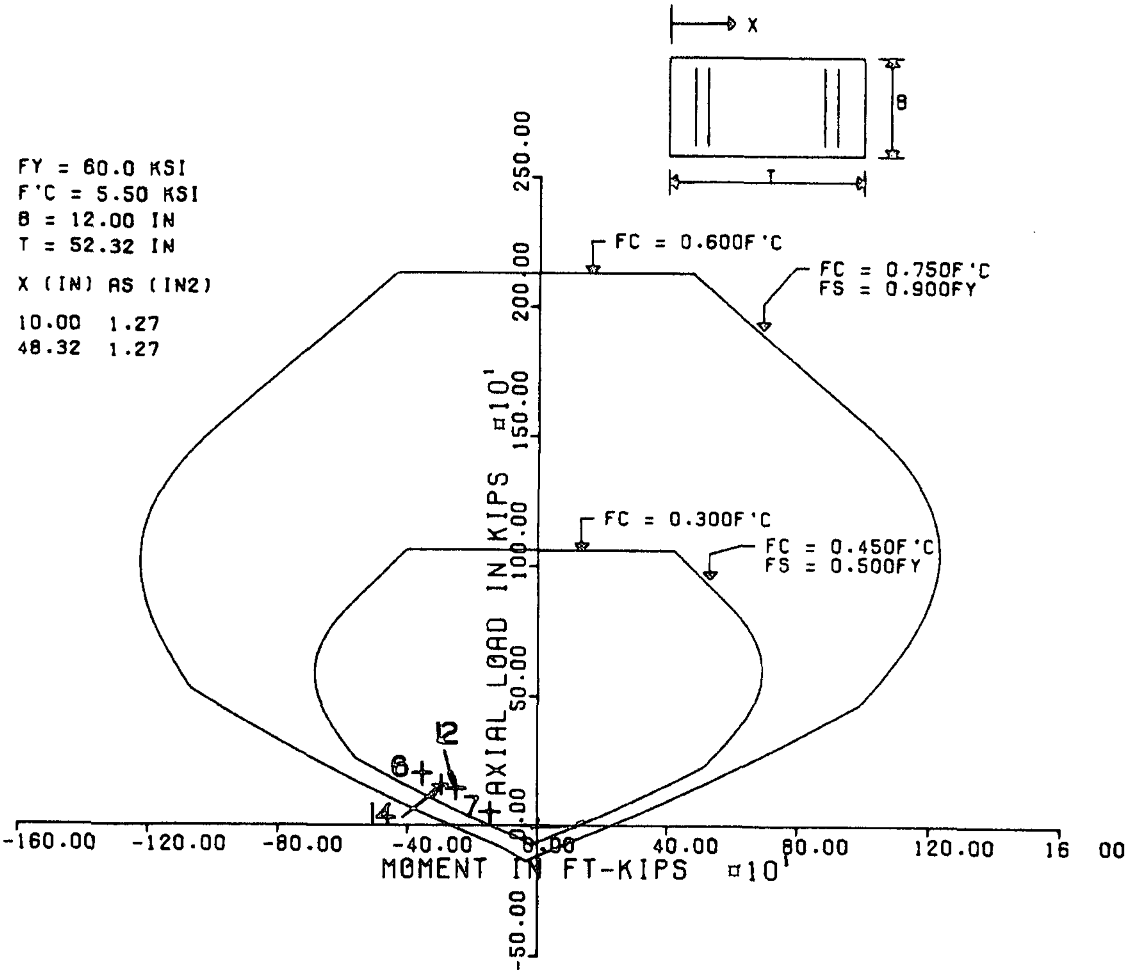


SECTION 8 MERIDIONAL REINFORCEMENT

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 16 OF 27)

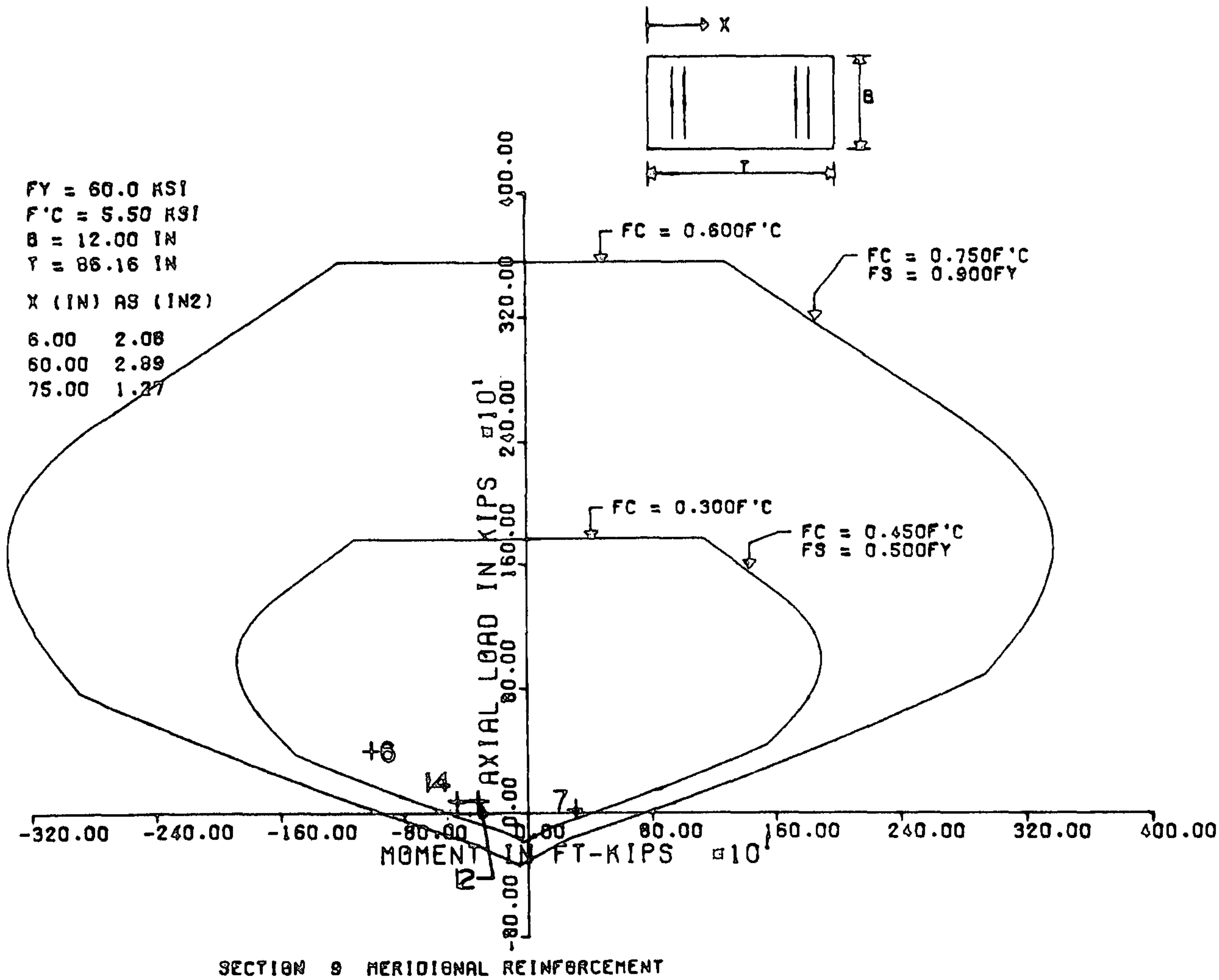


SECTION 8 HOOP REINFORCEMENT

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

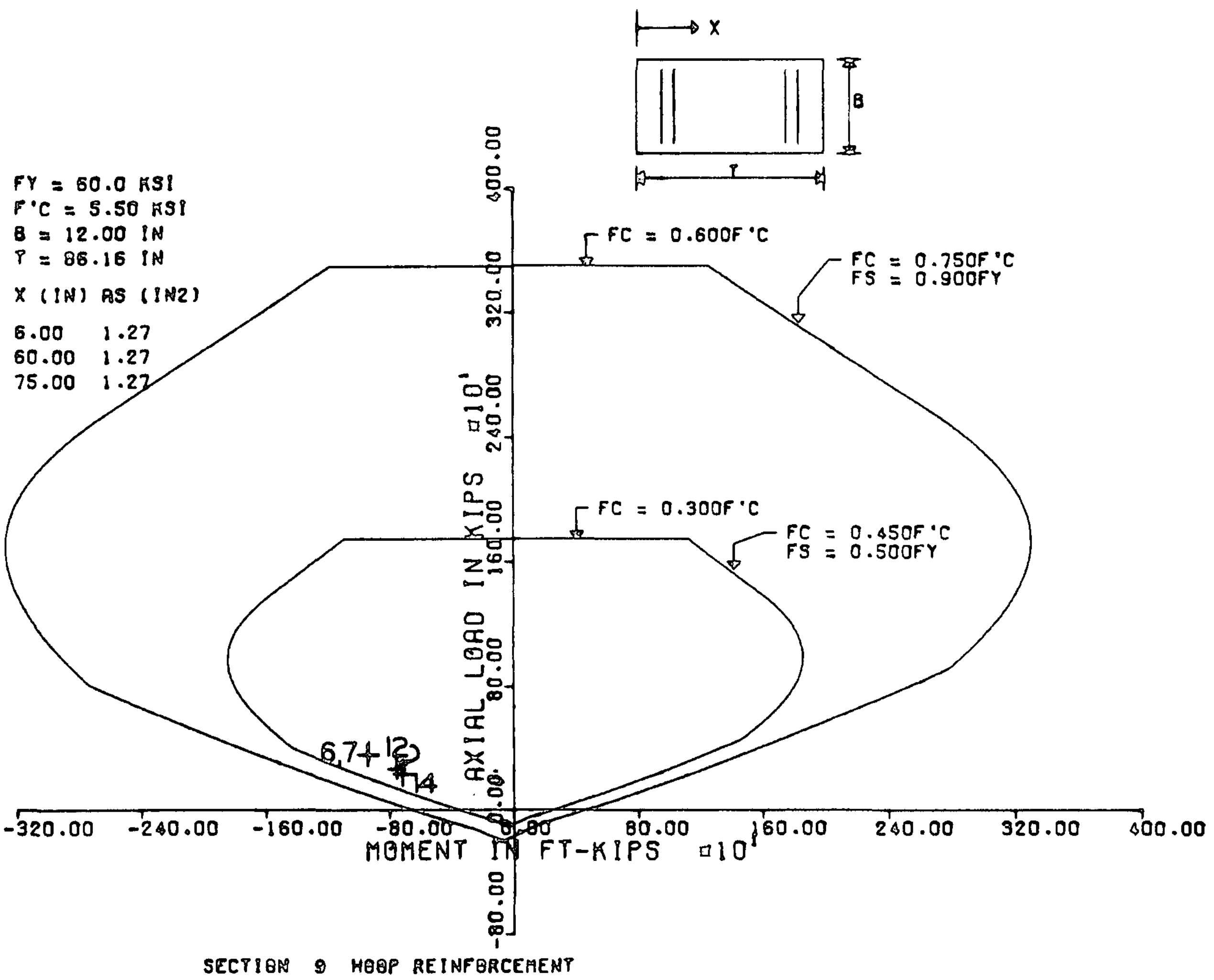
LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 17 OF 27)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 18 OF 27)



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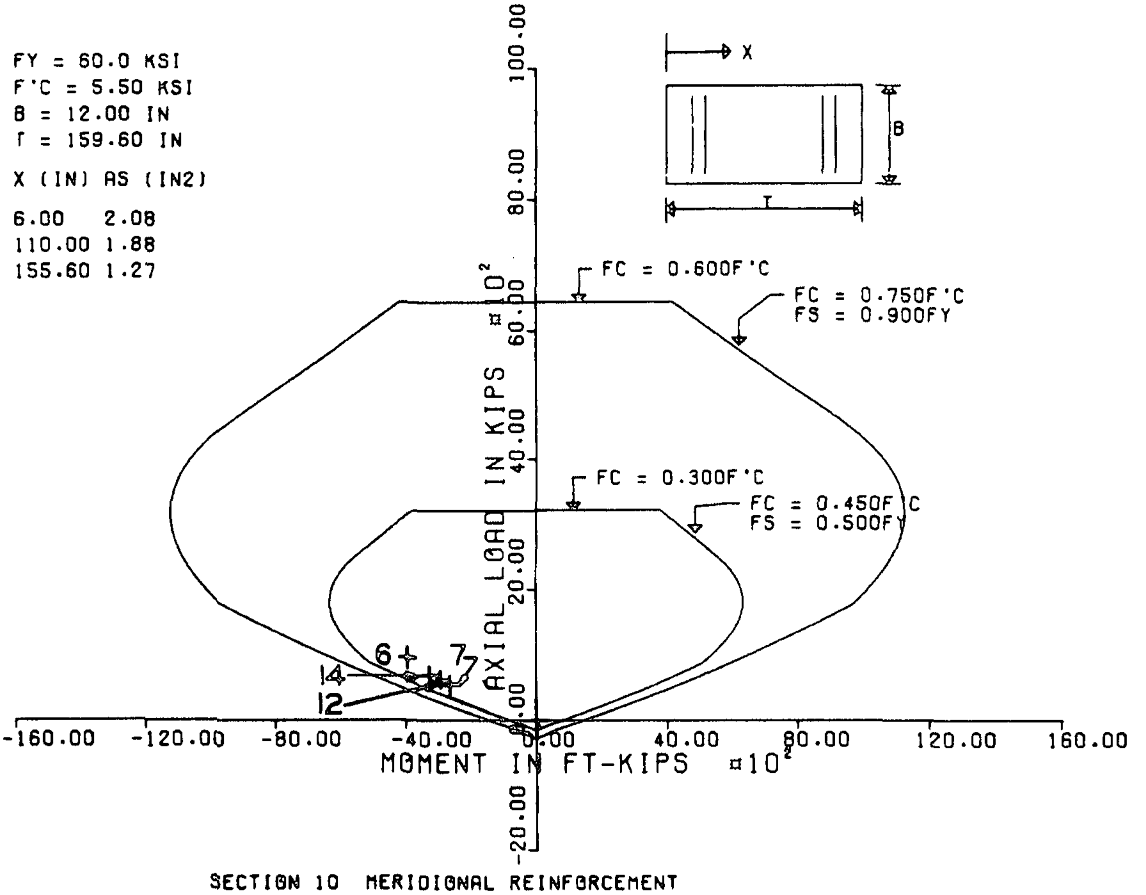
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 19 OF 27)

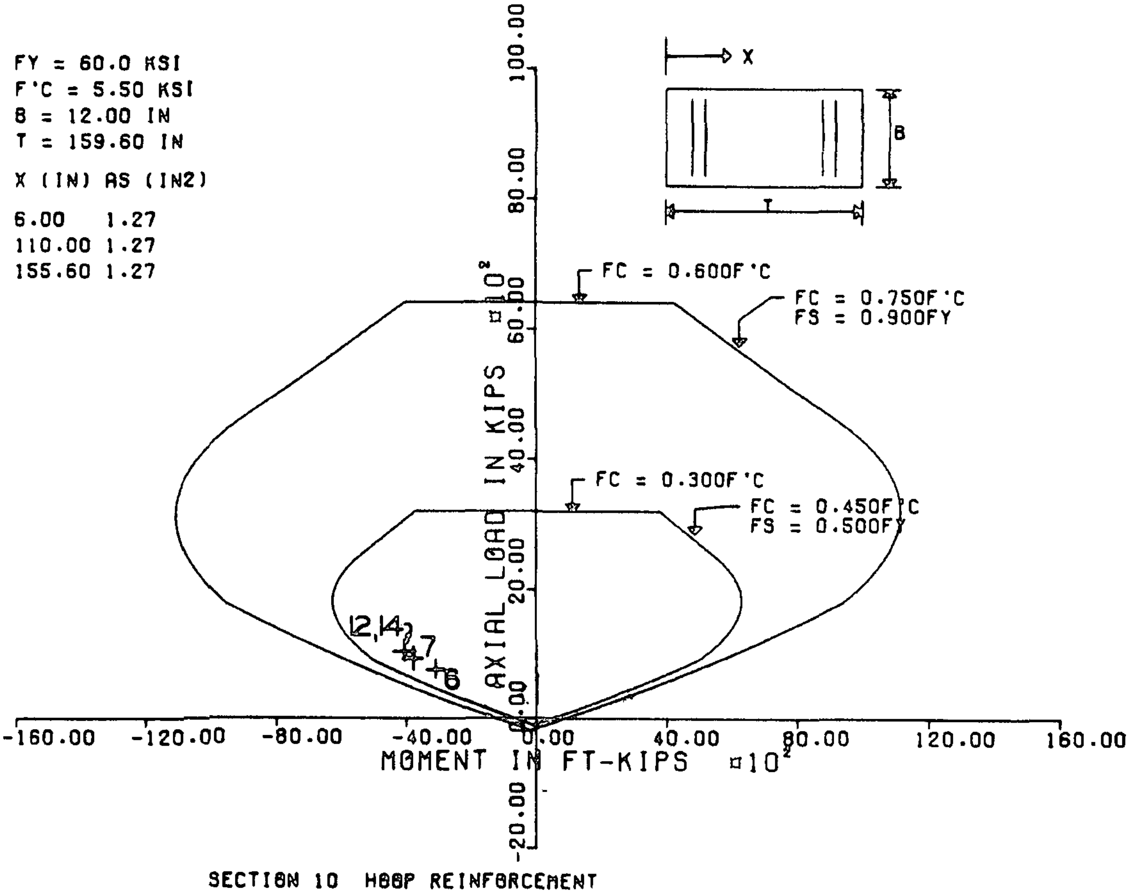


SECTION 10 MERIDIONAL REINFORCEMENT

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

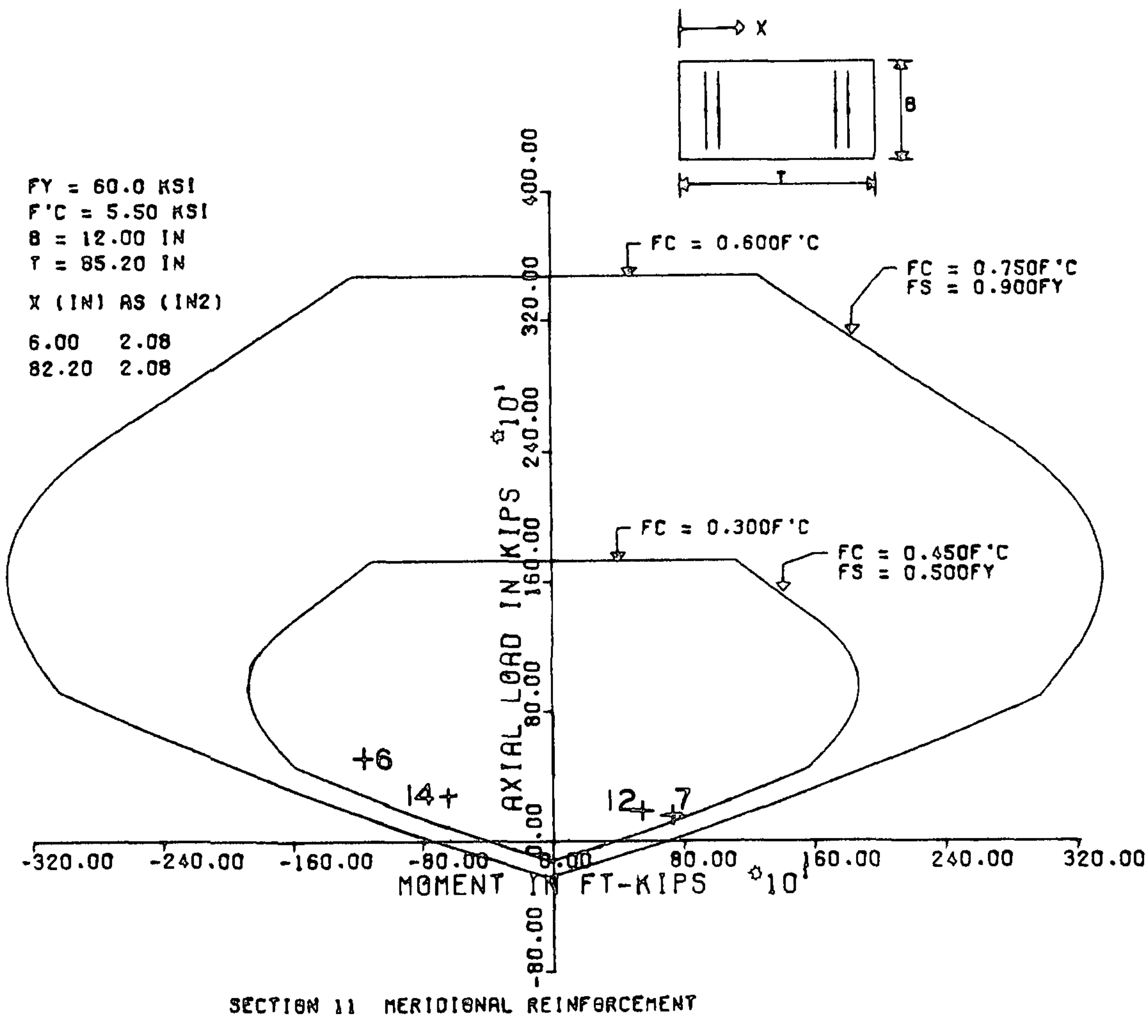
LOCATION OF SECTIONS FOR INTERACTION DIAGRAMS
(SHEET 20 OF 27)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

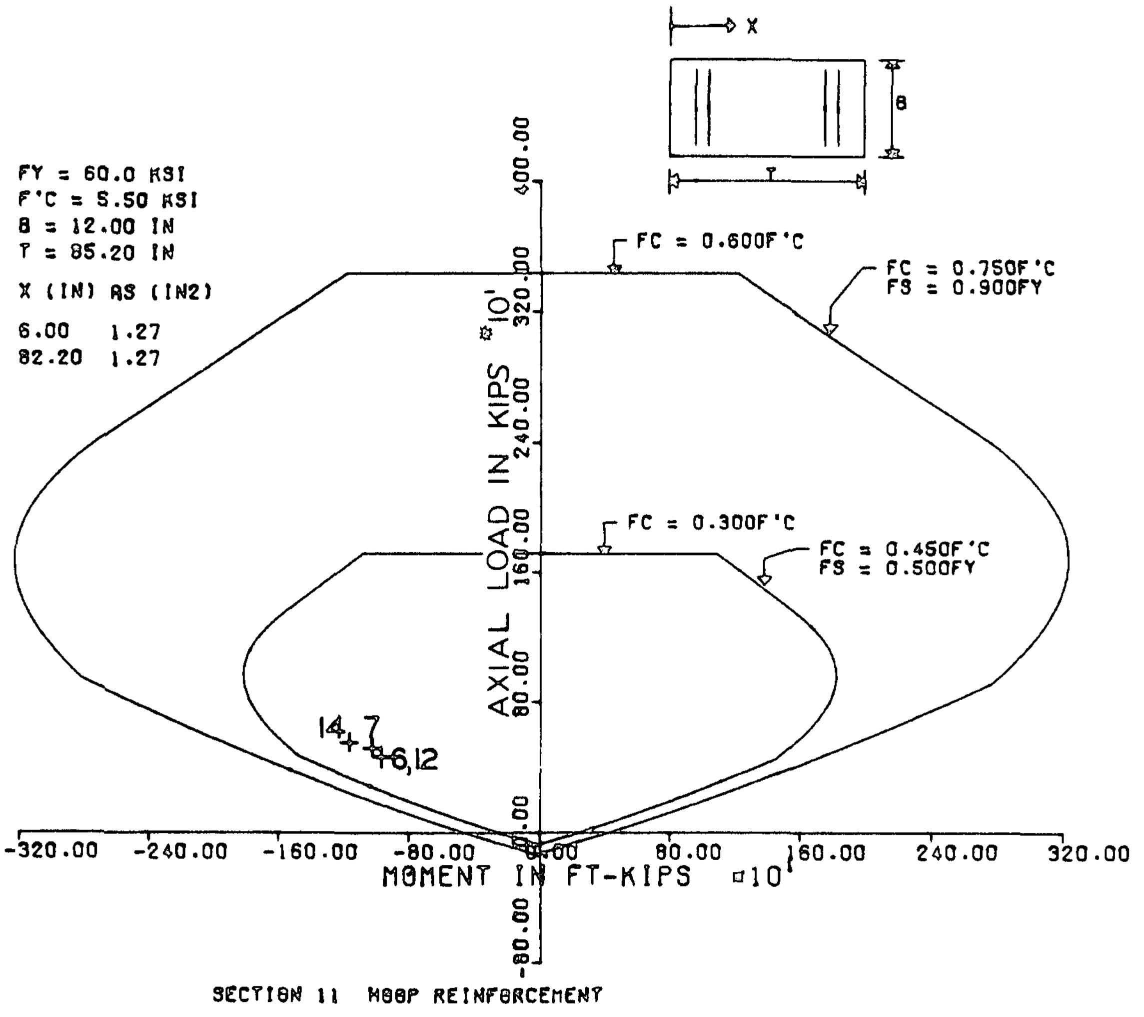
LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 21 OF 27)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

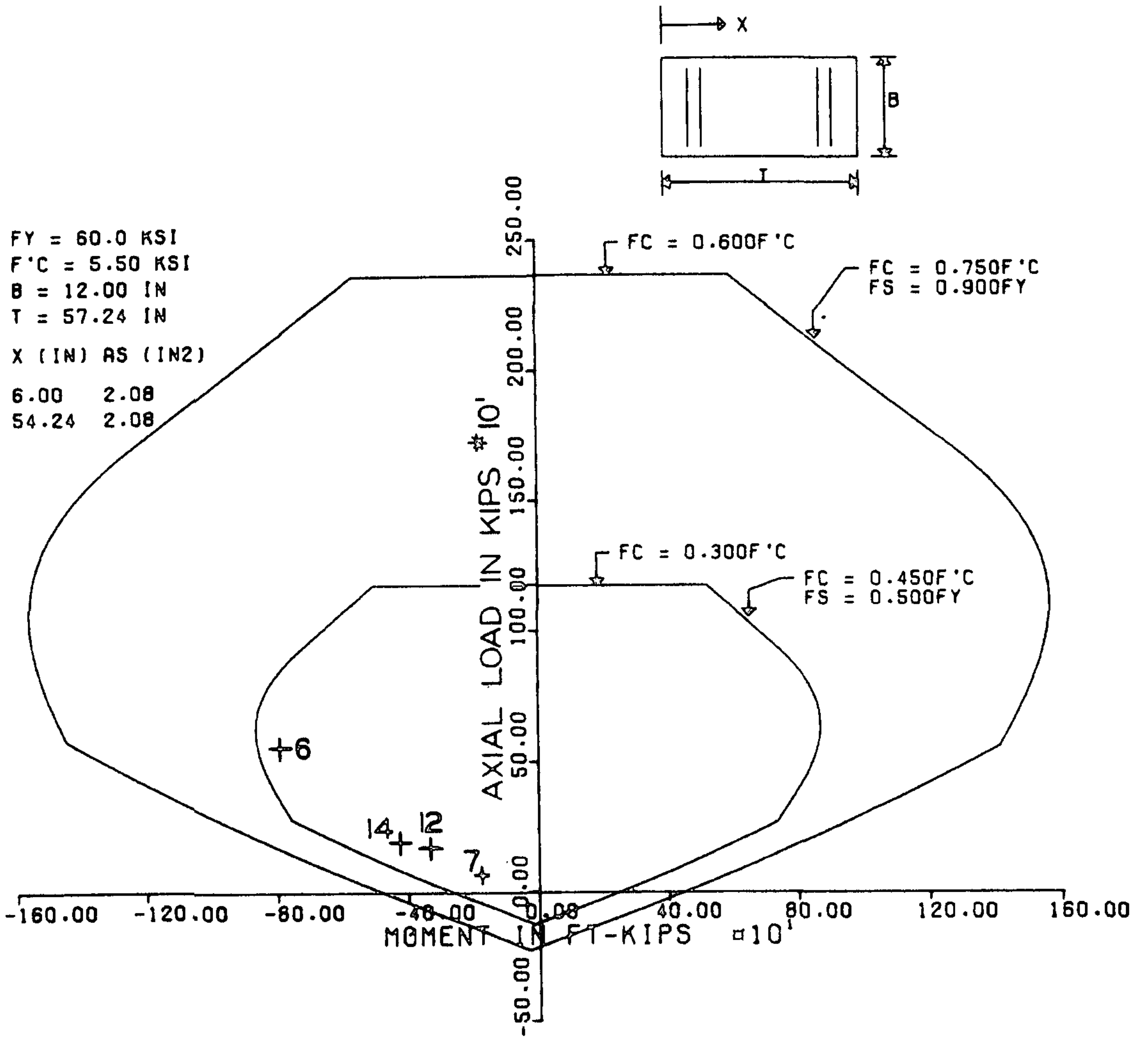
LOCATION OF SECTIONS FOR INTERACTION DIAGRAMS (SHEET 22 OF 27)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

LOCATION OF SECTIONS FOR INTERACTION DIAGRAMS (SHEET 23 OF 27)

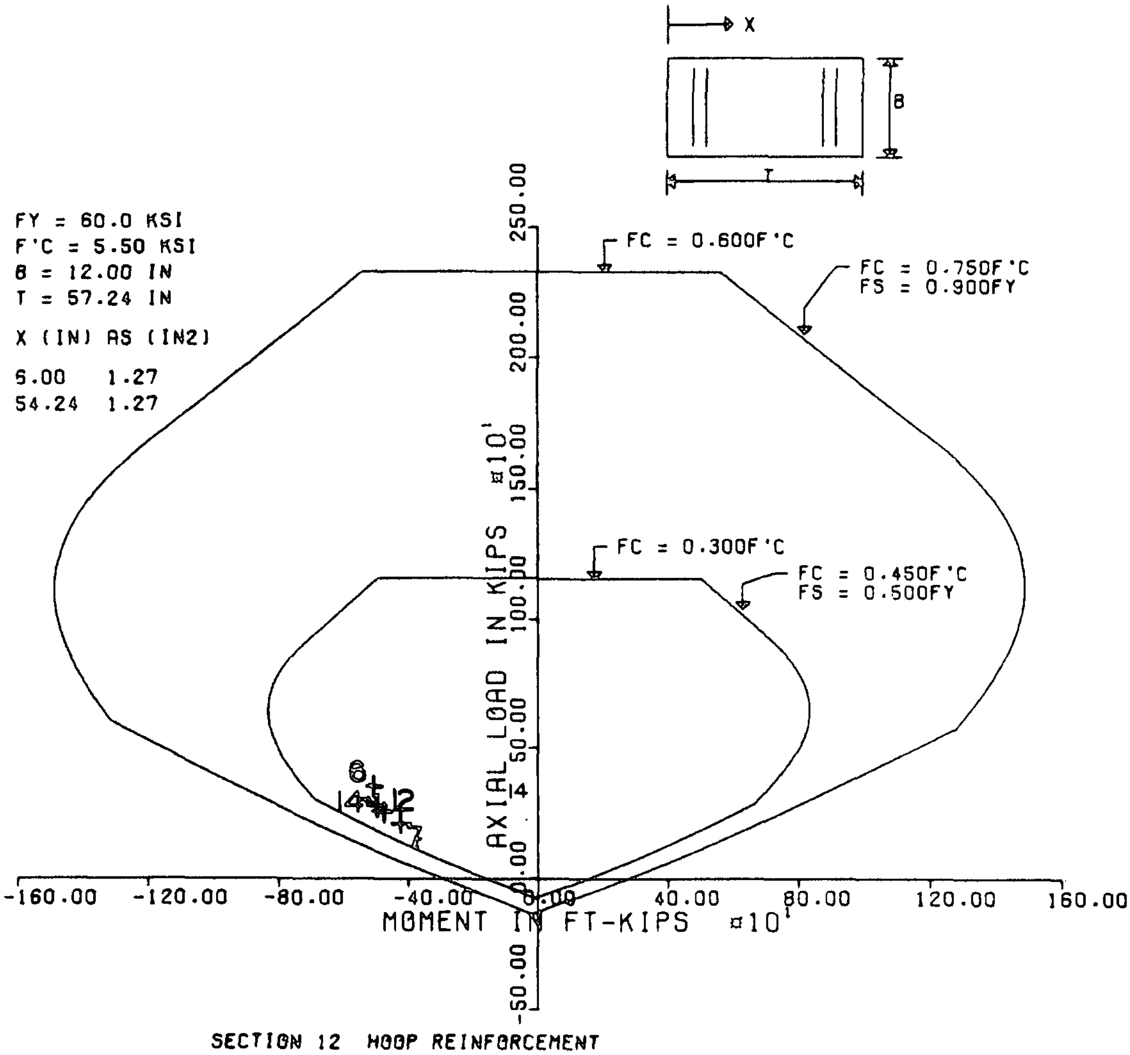


SECTION 12 MERIDIONAL REINFORCEMENT

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

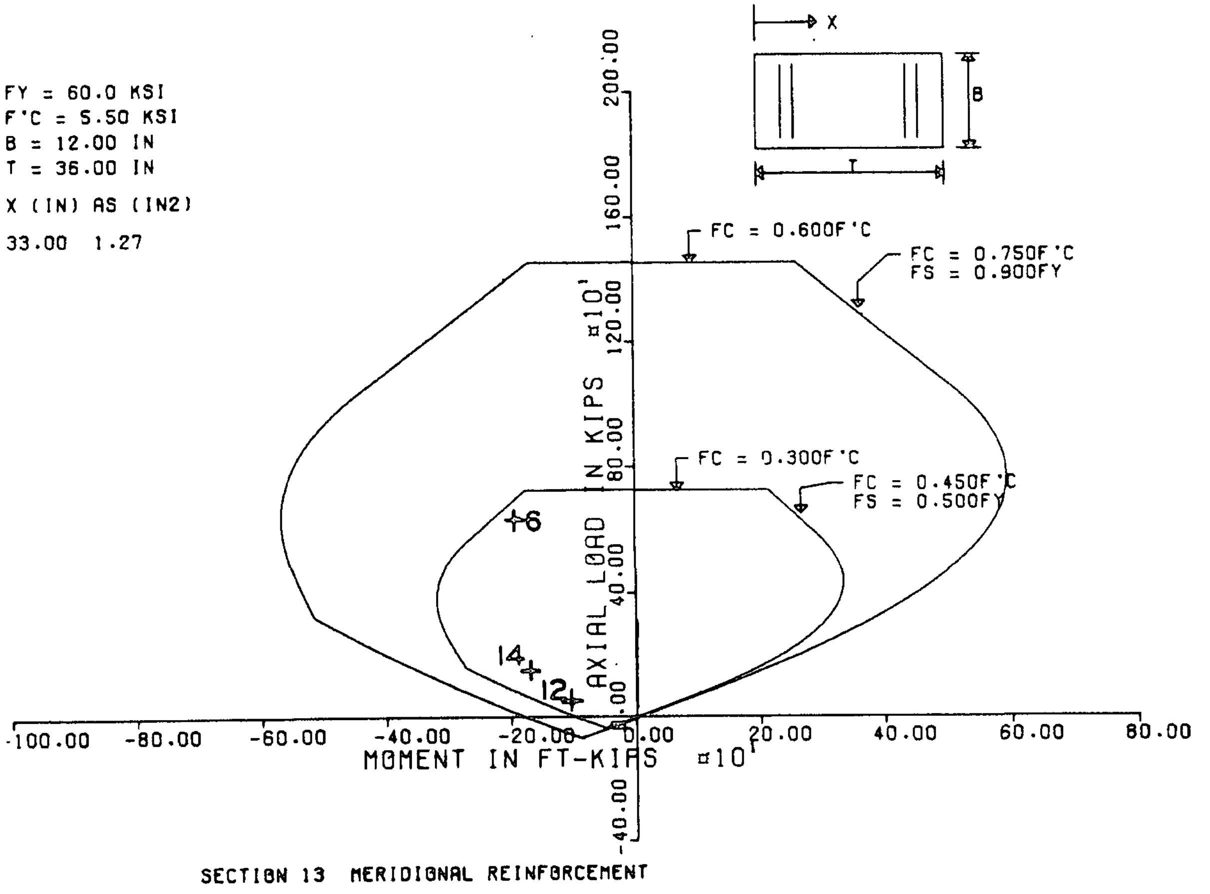
LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 24 OF 27)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 25 OF 27)



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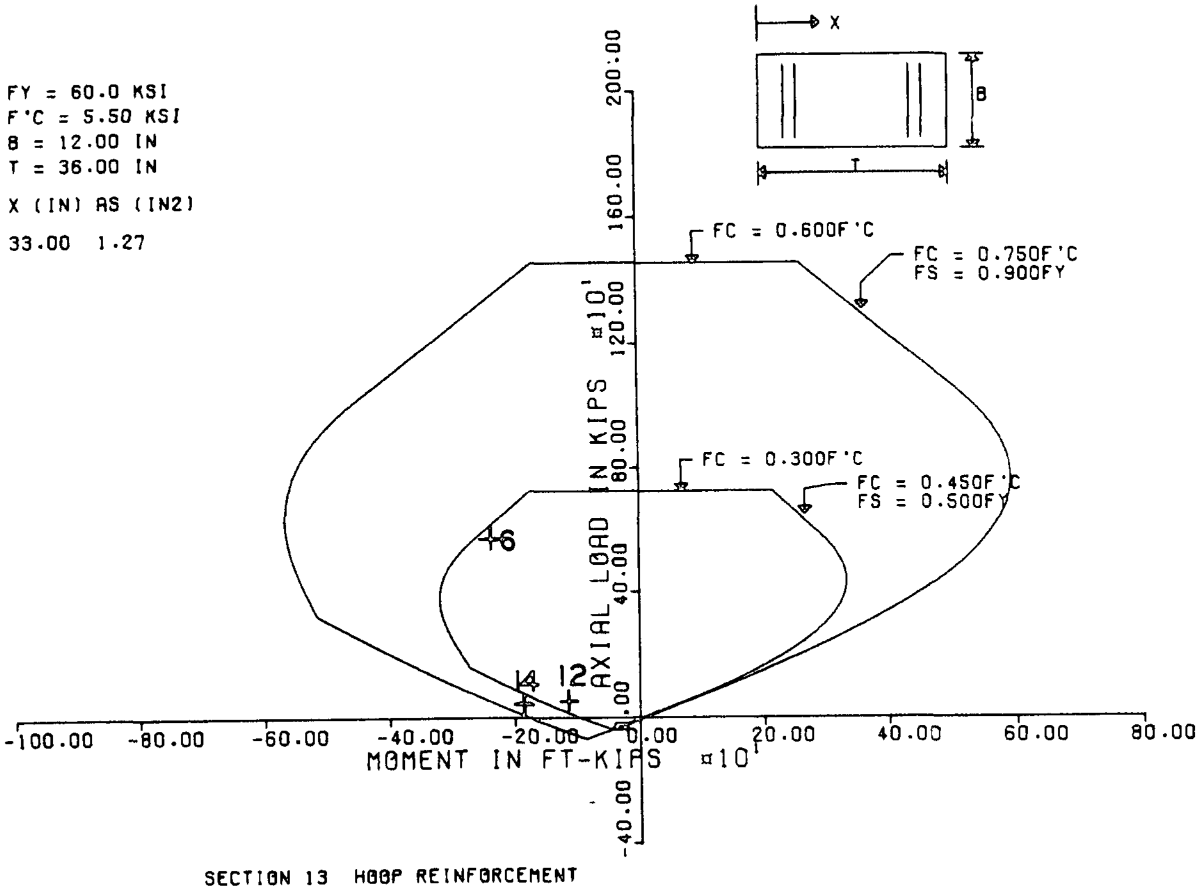
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 26 OF 27)

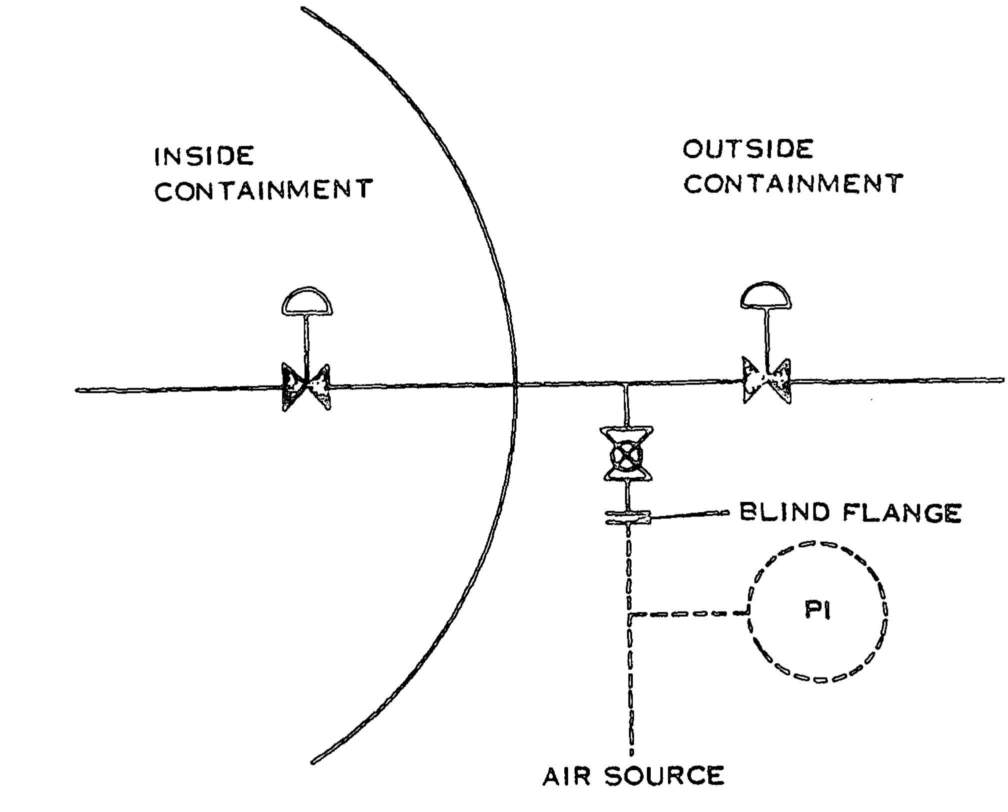


BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-35

LOCATION OF SECTIONS FOR
INTERACTION DIAGRAMS
(SHEET 27 OF 27)

Figure 3.8-36 has been deleted intentionally.



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-37

ISOLATION VALVE TESTING PROVISIONS

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-38

EQUIPMENT HATCH AIR LOCK

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Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 3.8-39

EMERGENCY PERSONNEL ACCESS LOCK

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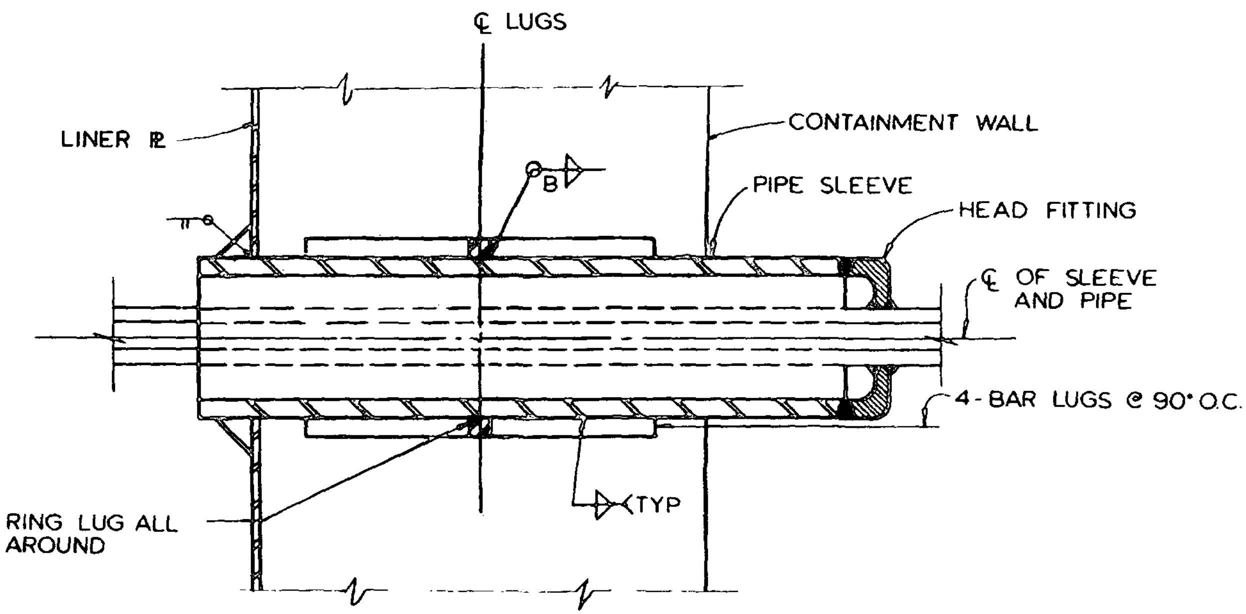
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-40

TYPE I HEAD FITTING



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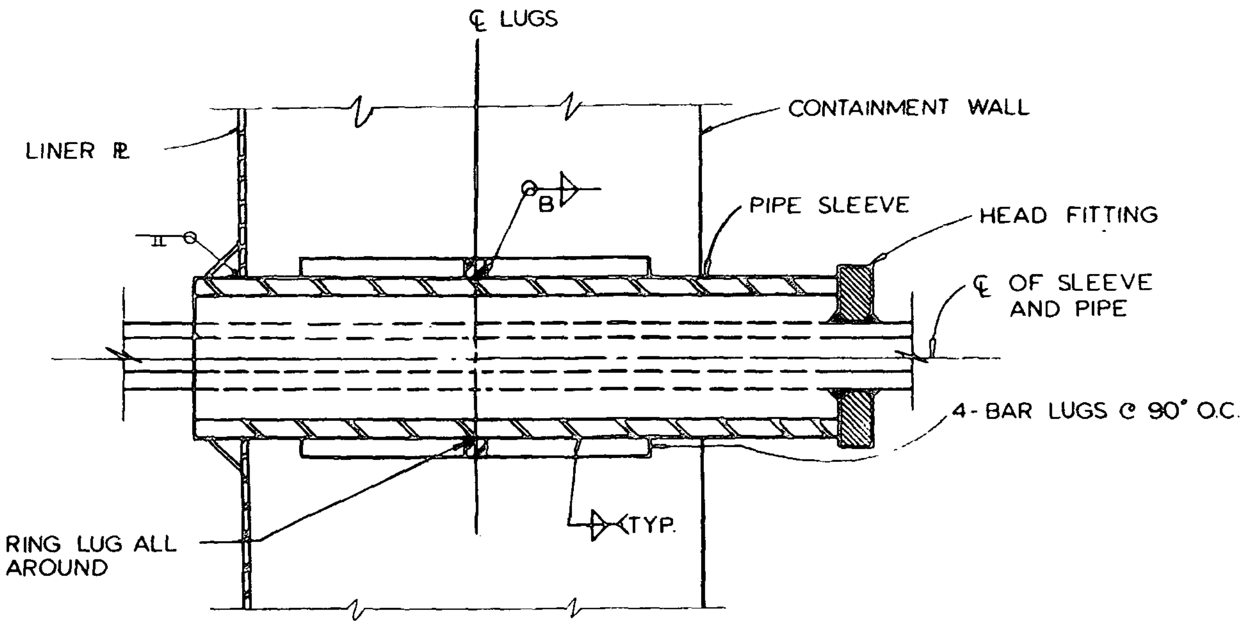
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-41

TYPE II HEAD FITTING



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-42

TYPE III HEAD FITTING

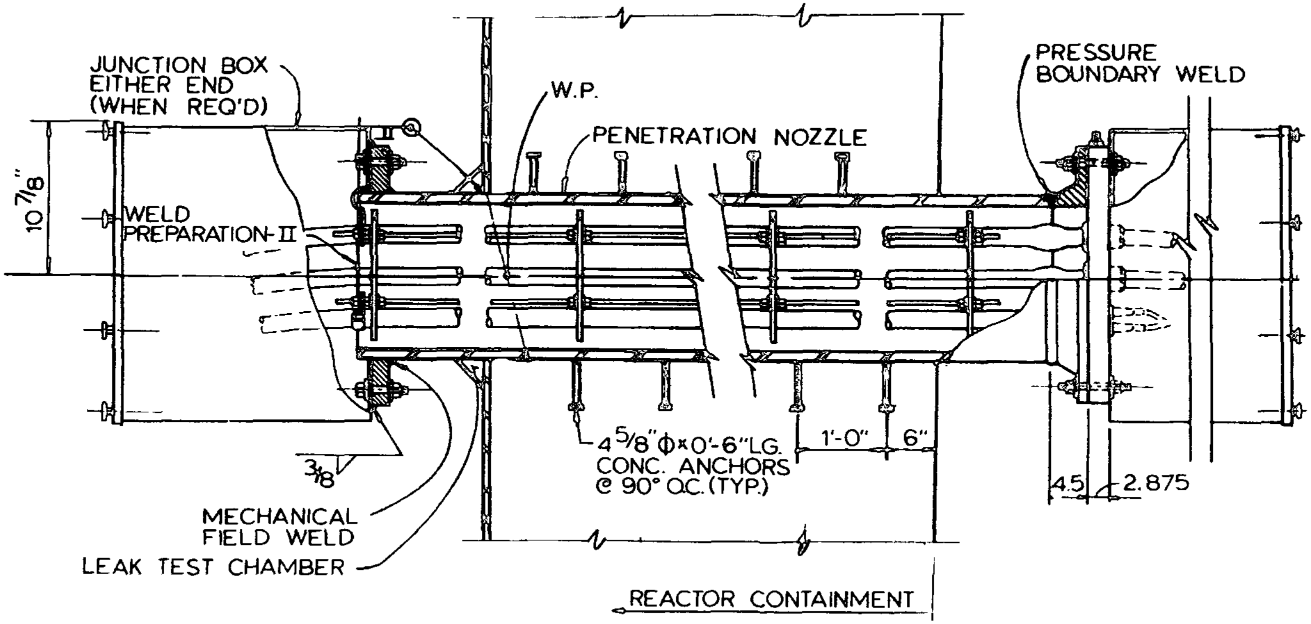


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FIGURE 3.8-43

ELECTRICAL PENETRATION

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BYRON STATION
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FIGURE 3.8-44

BYRON MAIN BUILDING COMPLEX MAT
FOUNDATION PLAN

Security - Related Information Figure Withheld Under 10 CFR 2.390

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-45

BRAIDWOOD MAIN BUILDING COMPLEX MAT FOUNDATION PLAN

Security - Related Information Figure Withheld Under 10 CFR 2.390

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-46

CONTAINMENT BUILDING
EAST-WEST SECTION

Security - Related Information Figure Withheld Under 10 CFR 2.390

BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.8-47 CONTAINMENT BUILDING NORTH-SOUTH SECTION

Security - Related Information Figure Withheld Under 10 CFR 2.390

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-48

CONTAINMENT BUILDING PRIMARY
SHIELD WALL AT INLET NOZZLE

Security - Related Information Figure Withheld Under 10 CFR 2.390

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-49

CONTAINMENT BUILDING PRIMARY
SHIELD WALL AT OUTLET NOZZLE

Security - Related Information Figure Withheld Under 10 CFR 2.390

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-50

CONTAINMENT BUILDING PRIMARY
SHIELD WALL AT
NEUTRON DETECTOR CAVITY

Security - Related Information Figure Withheld Under 10 CFR 2.390

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-51

CONTAINMENT BUILDING NSSS
COMPONENT ENCLOSURES
(SHEET 1 OF 2)

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 3.8-51 CONTAINMENT BUILDING NSSS COMPONENT ENCLOSURES (SHEET 2 OF 2)

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Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 3.8-52 AUXILIARY-FUEL HANDLING BUILDING COMPLEX SHEAR WALLS AT ELEVATION 401'-0"

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FIGURE 3.8-53

AUXILIARY BUILDING SECTION

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FIGURE 3.8-54

SHEAR WALL—SLAB DIAPHRAGM
ABOVE GRADE
DETAIL 1

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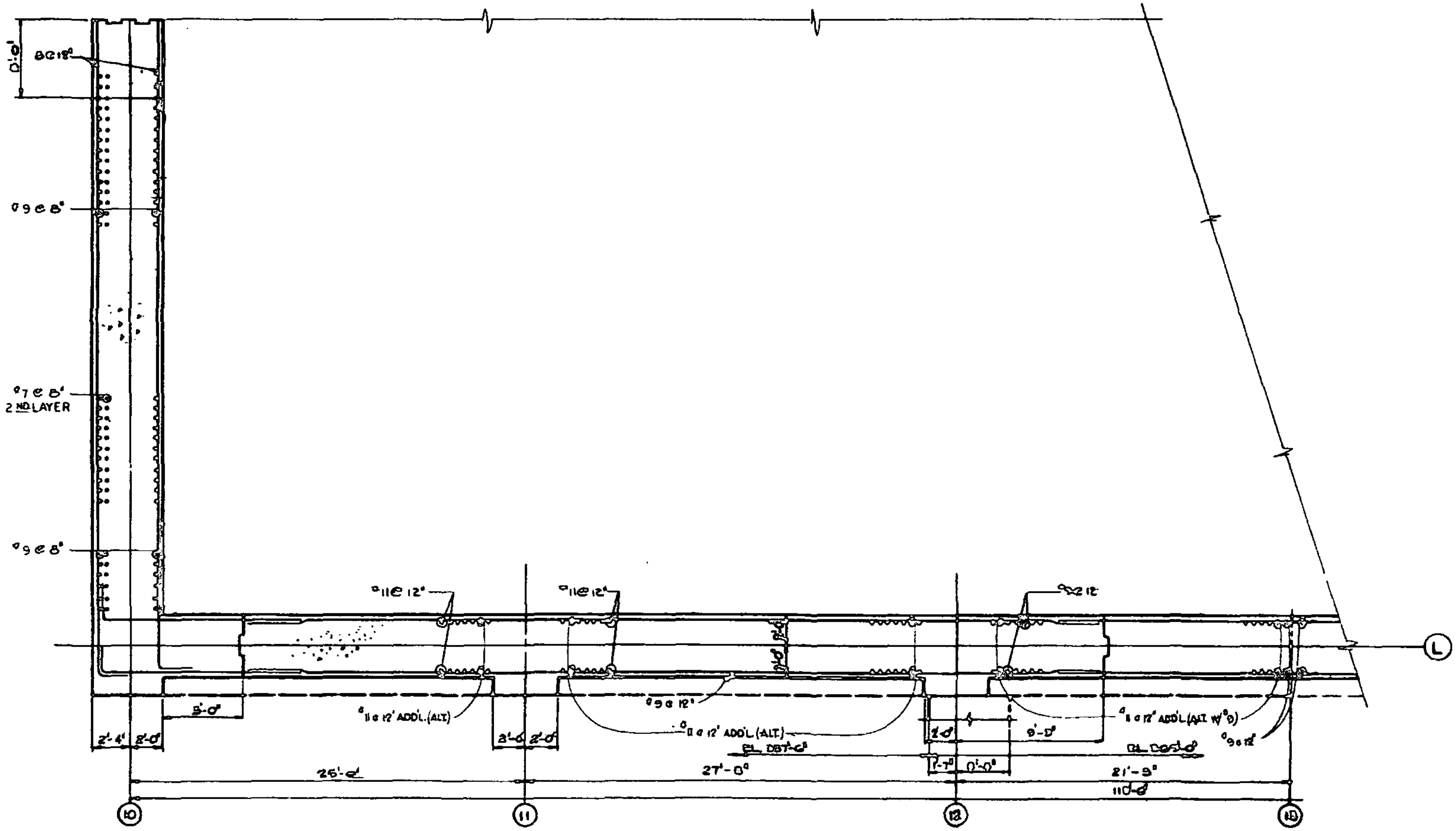
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-55

SHEAR WALL—SLAB DIAPHRAGM
BELOW GRADE
DETAIL 2



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-56

TYPICAL WALL CORNER REINFORCING PLAN

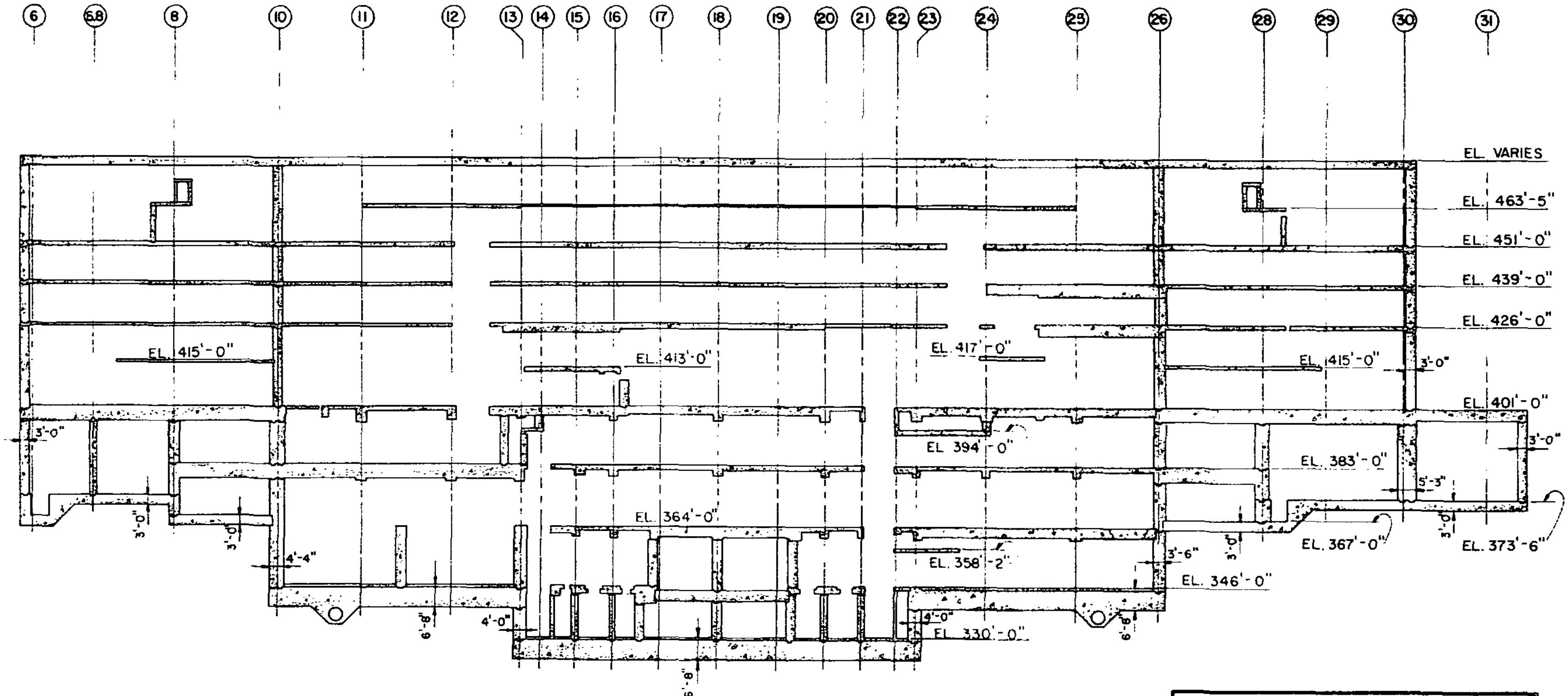


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FIGURE 3.8-57

AUXILIARY BUILDING SECTION B-B

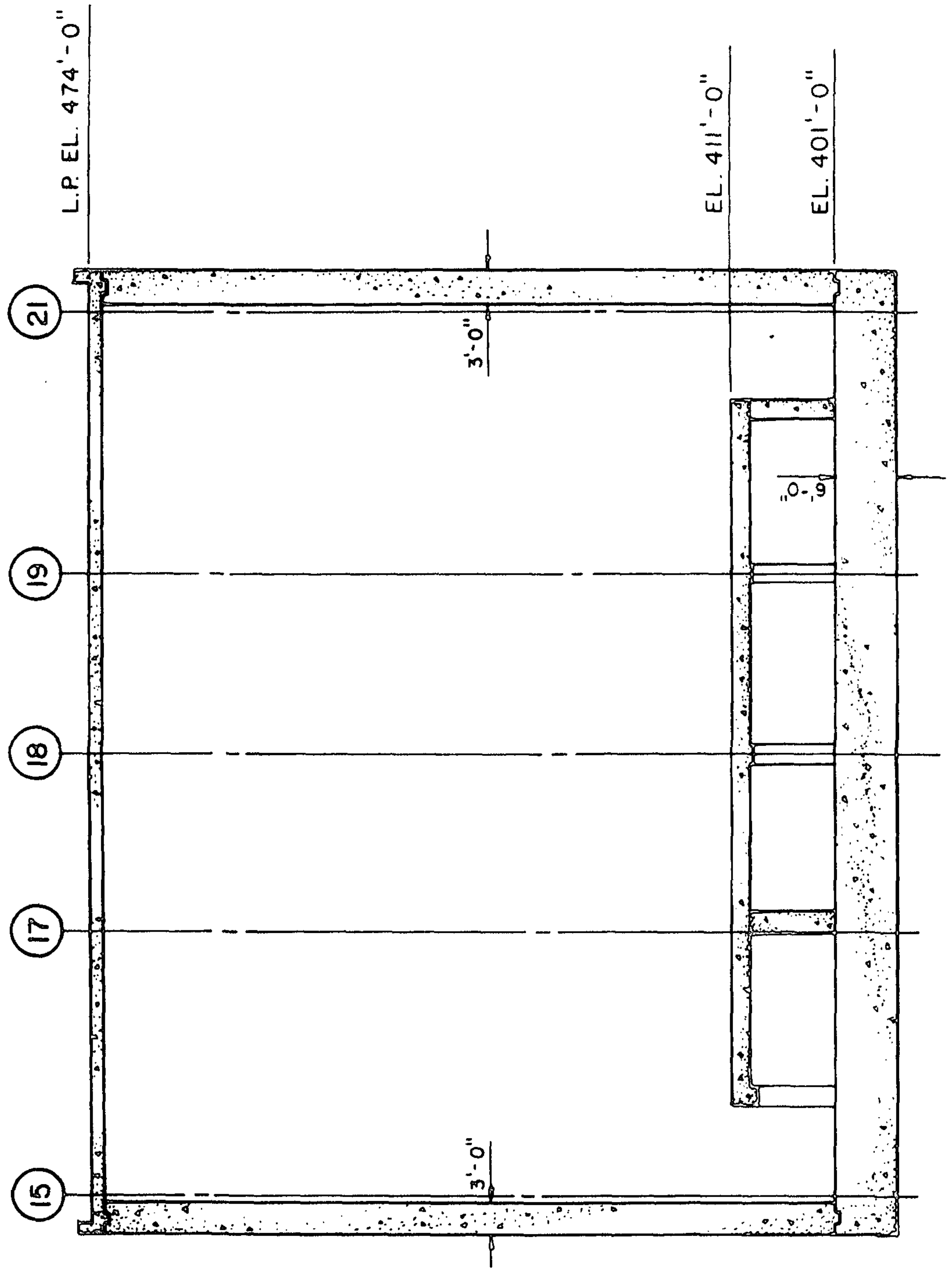

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FIGURE 3.8-58

FUEL HANDLING BUILDING SECTION C-C

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FIGURE 3.8-59

RIVER SCREEN HOUSE FOUNDATION PLAN

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BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-60

RIVER SCREEN HOUSE FOUNDATION PLAN
(CONTINUED)

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FIGURE 3.8-61

RIVER SCREEN HOUSE SECTION

Security - Related Information Figure Withheld Under 10 CFR 2.390

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-62

RIVER SCREEN HOUSE SECTION 9-9

SecuritySecurity - Related Information Figure Withheld Under 10 CFR 2.390 R 2.390

BYRON STATION
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FIGURE 3.8-63

RIVER SCREEN HOUSE FLOOR FRAMING PLAN

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BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-64

RIVER SCREEN HOUSE ROOF FRAMING PLAN

Figures 3.8-65 through 3.8-73 have been deleted.

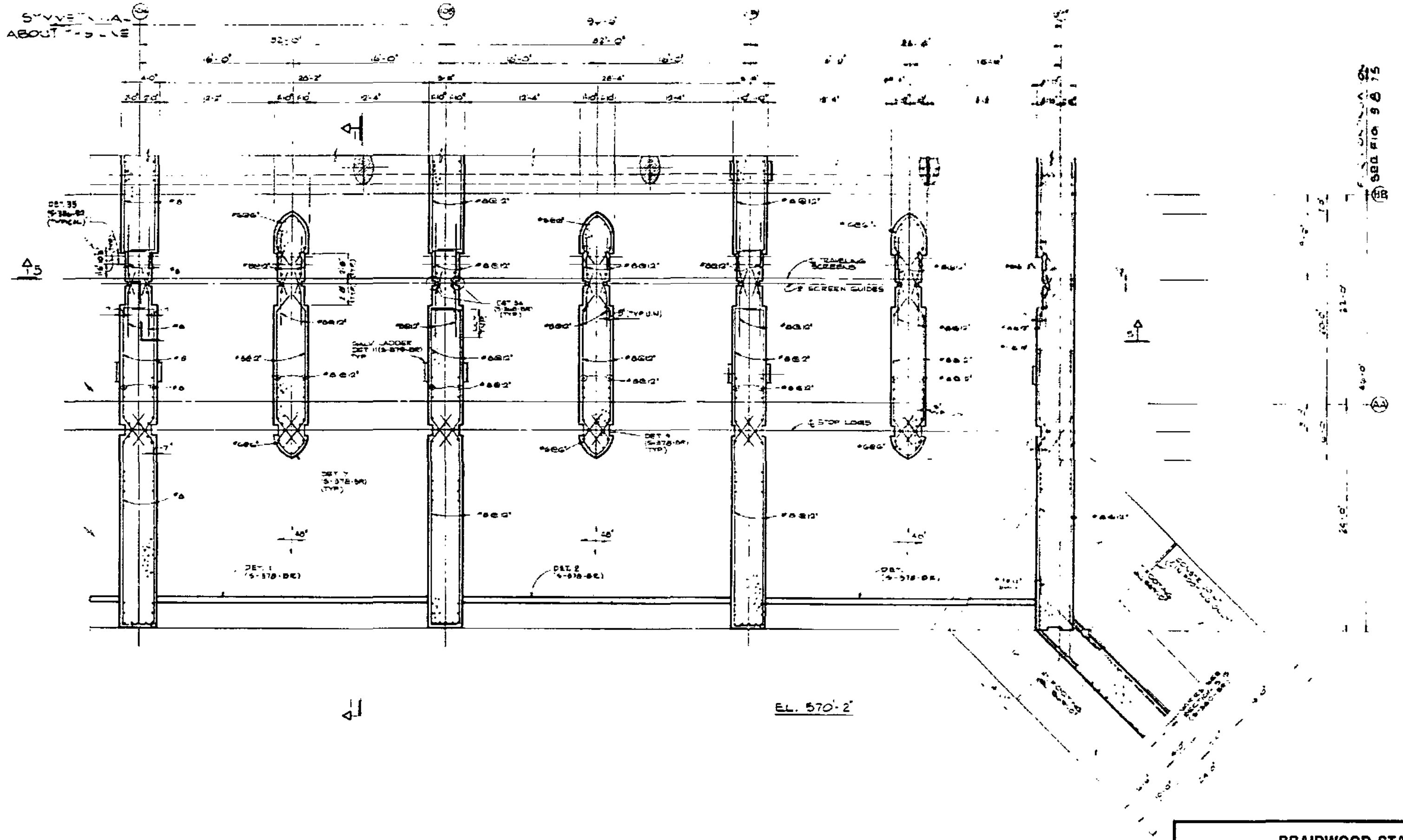

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FIGURE 3.8-74

LAKE SCREEN HOUSE FOUNDATION PLAN

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FIGURE 3.8-75

LAKE SCREEN HOUSE FLOOR FRAMING PLAN
ELEVATION 588'-0"

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BRAIDWOOD STATION UPDATED FINAL SAFETY ANALYSIS REPORT
3.8-76 LAKE SCREEN HOUSE FLOOR FRAMING PLAN EL. 602'-10"

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BRAIDWOOD STATION
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FIGURE 3.8-77

LAKE SCREEN HOUSE SECTION 1-1

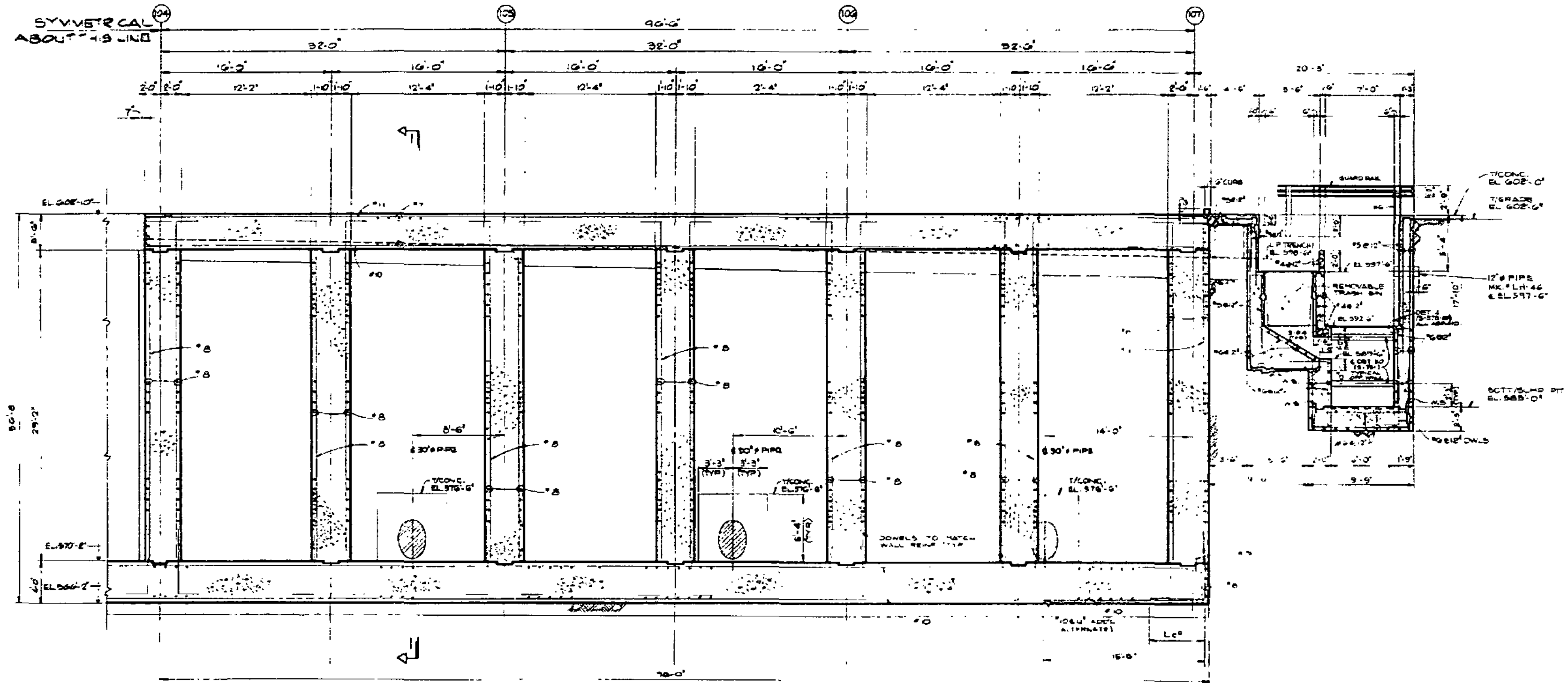

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FIGURE 3.8-78

LAKE SCREEN HOUSE SECTION 5-5

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FIGURE 3.8-79 DEEP WELL ENCLOSURES

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FIGURE 3.8-80

SAFETY VALVE ROOM FLOOR
ELEVATION 377'-0"

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FIGURE 3.8-81

SAFETY VALVE ROOM SECTION 1-1

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FIGURE 3.8-82

SAFETY VALVE ROOM SECTION 4-4

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DECEMBER 2002

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FIGURE 3.8-83 REFUELING WATER STORAGE TANK

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FIGURE 3.8-84

REFUELING WATER STORAGE TANK
TYPICAL DOME SECTION

FSAR \ Div. 2	D	L	F	P_t	P_a	T_t	T_o	T_a	E_o	E_{ss}	W	W_t	R_o	R_a	R_r	P_r	*H_a
D	X																
L		X															
F			X														
P'				X													
P_a					X												
P_o																X	
T_o						X	X										
T_a								X									
E									X								
E'										X							
W											X						
W'												X					
R_o													X				
R_a														X			
(R_{rr}) Y_r															X		
(R_{rm}) Y_m															X		
(R_{rj}) Y_j															X		
H'																	
M	Missiles that are not mentioned in Div. 2																

*H_a - Summer, 1977 post LOCA internal flooding.

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FIGURE 3.8-85

LOAD DEFINITIONS

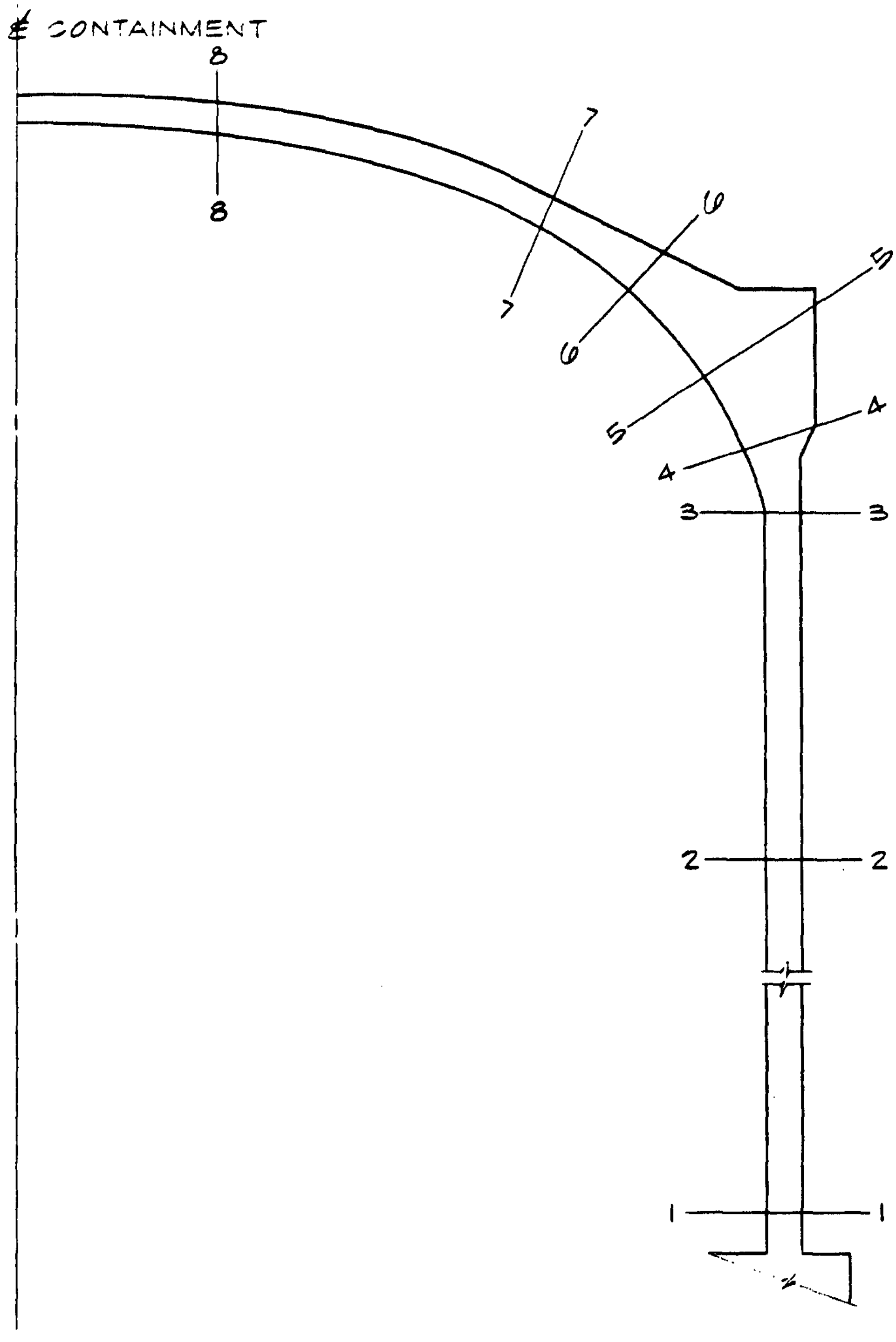

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FIGURE 3.8-86

SECTIONS FOR THIN SHELL VS. THICK SHELL CONTAINMENT ANALYSIS

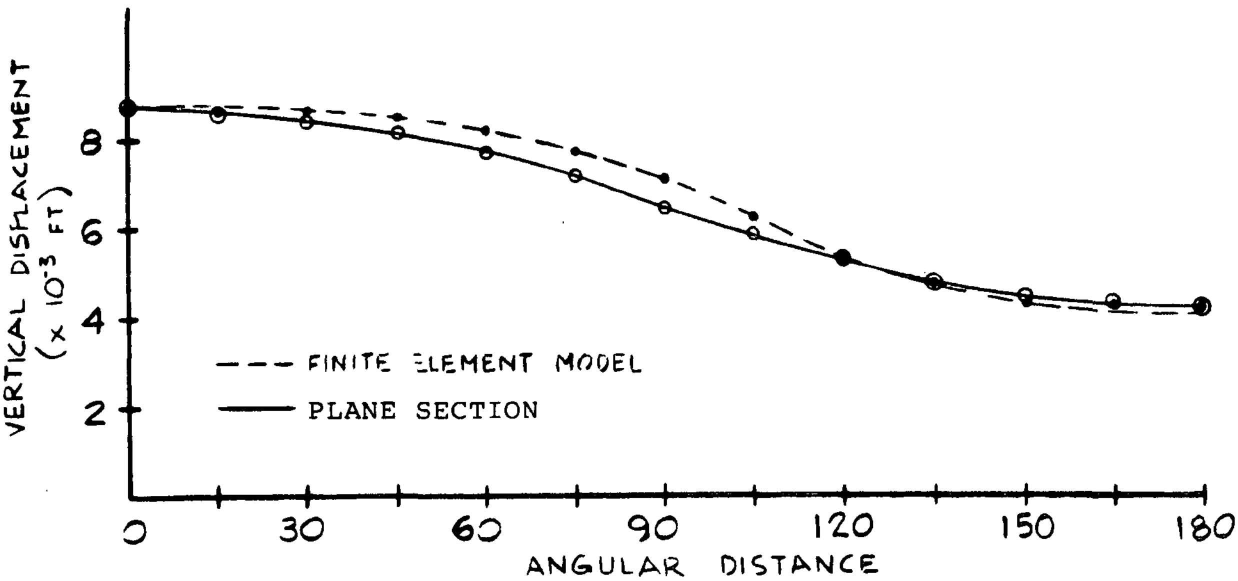

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FIGURE 3.8-87

VERTICAL DISPLACEMENT ALONG HALF THE CIRCUMFERENCE OF THE CONTAINMENT WALL AT 50 FT ABOVE THE BASE MAT

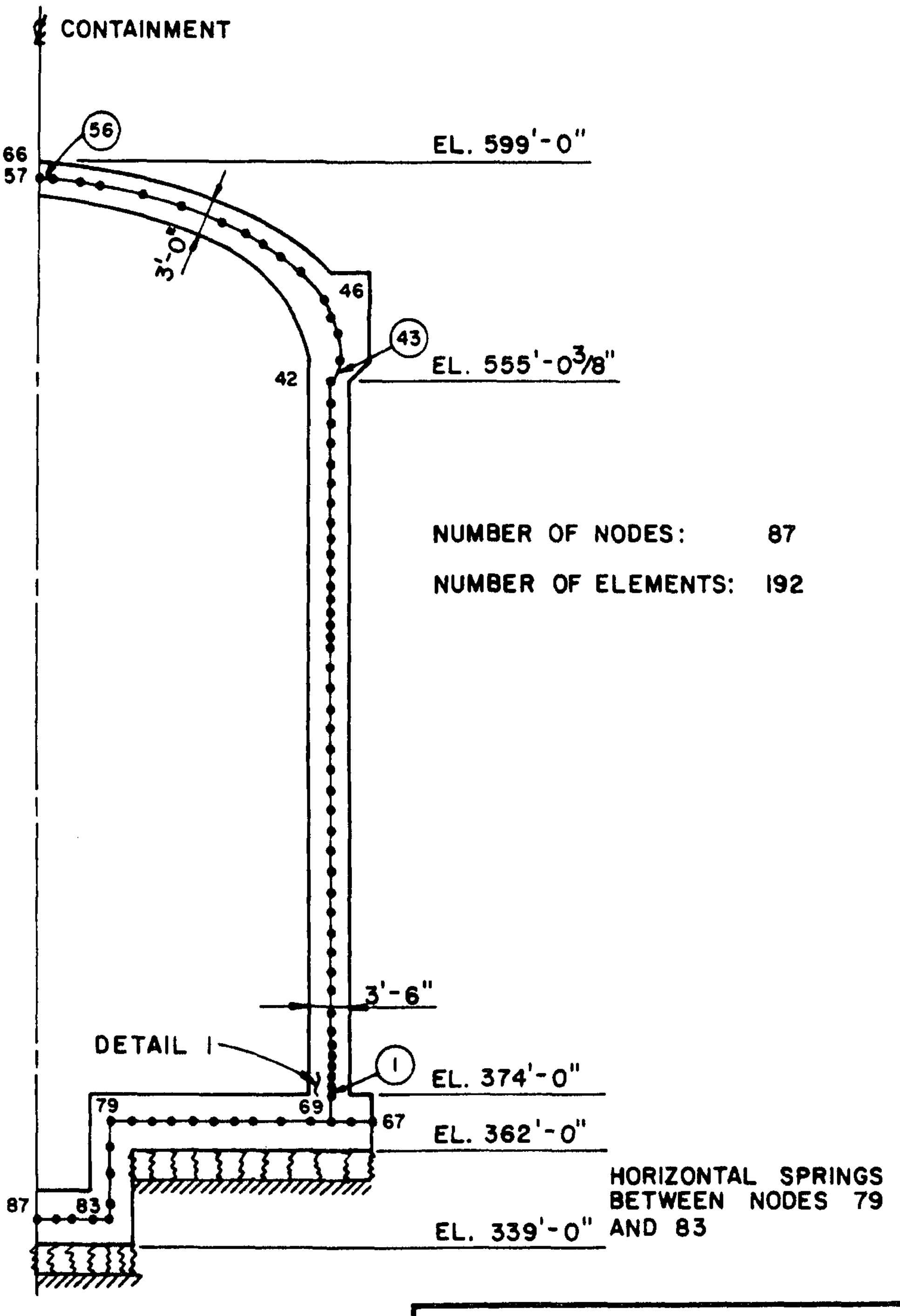

FIGURE 1

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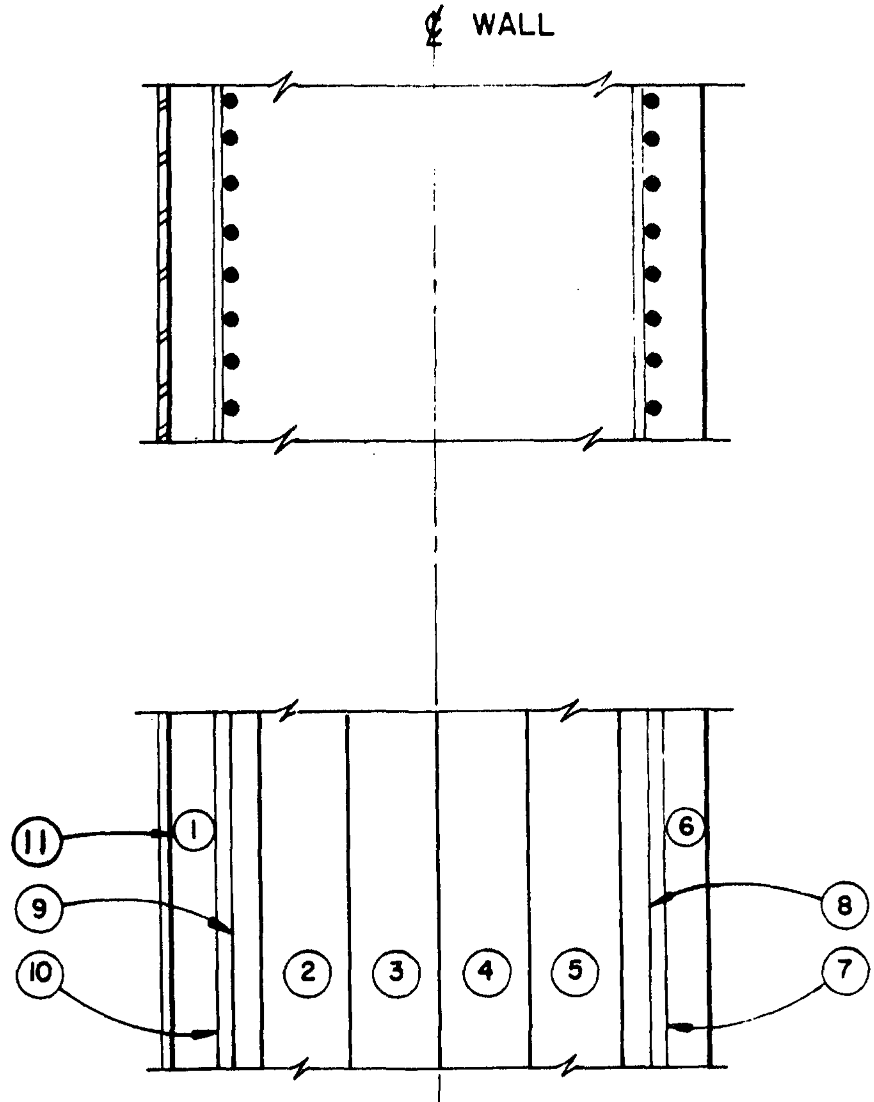
FIGURE 3.8-88

CONTAINMENT FINITE ELEMENT MODEL



CONCRETE LAYERS 1-6
HOOP STEEL 7 & 9
MERIDIONAL STEEL 8 & 10
LINER 11
(STRENGTH CONTRIBUTION OF LINER WAS NEGLECTED)

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FIGURE 3.8-89

LAYERED ELEMENT FOR NON-LINEAR FINITE ELEMENT ANALYSIS

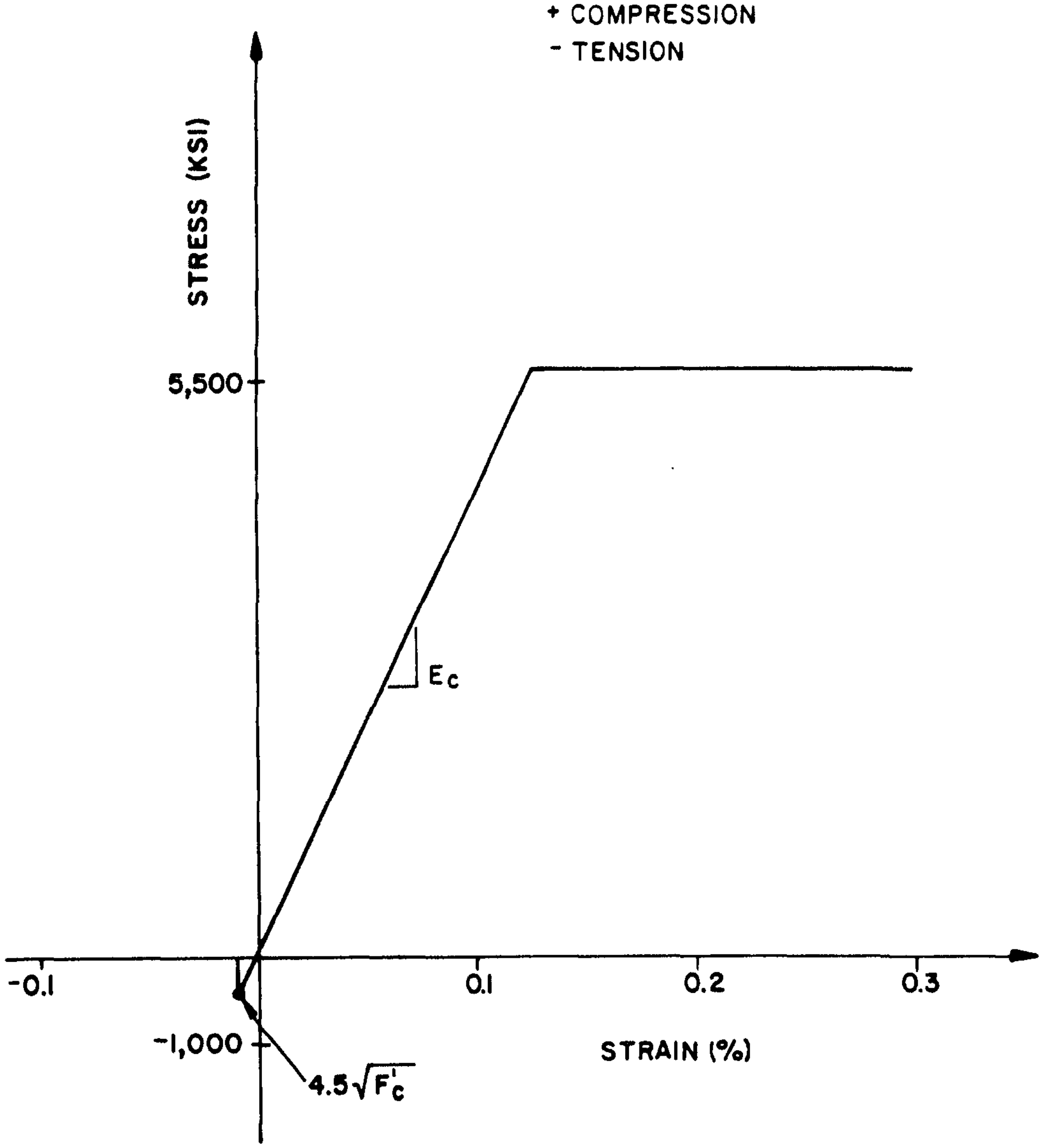

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FIGURE 3.8-90

CONTAINMENT CONCRETE IDEALIZED
STRESS-STRAIN CURVE

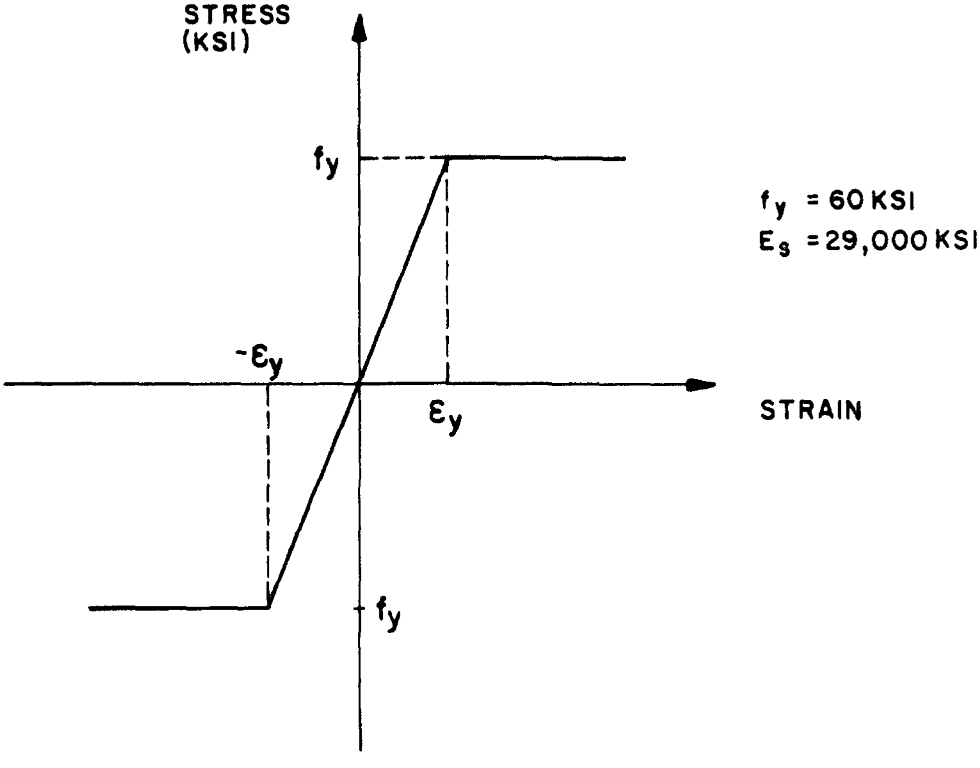

FIGURE 4

NON - LINEAR STRESS - STRAIN RELATION FOR REINFORCING STEEL

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8-91

NON-LINEAR STRESS-STRAIN RELATION FOR REINFORCING STEEL

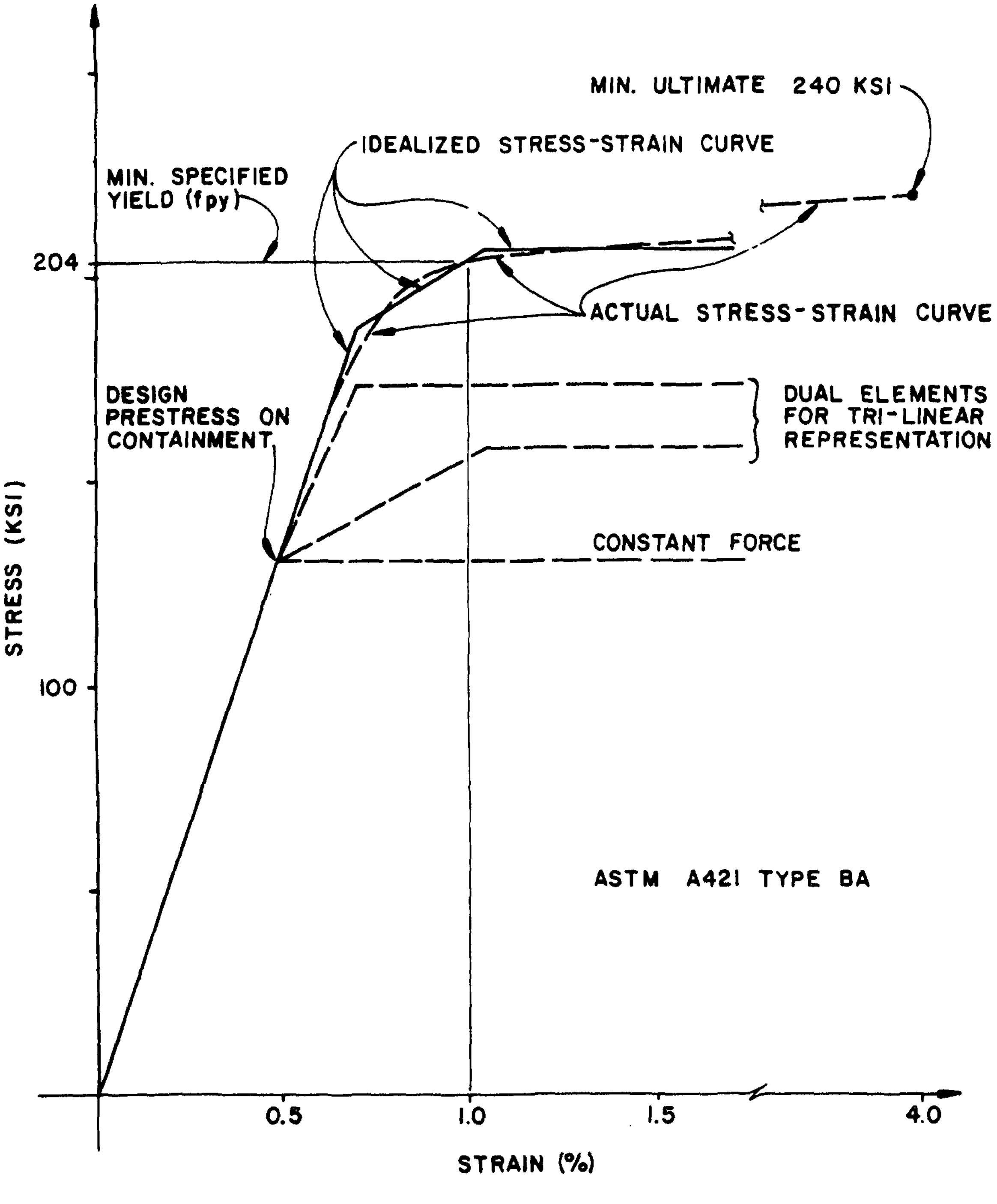

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FIGURE 3.8-92

TENDON IDEALIZED STRESS-STRAIN CURVE

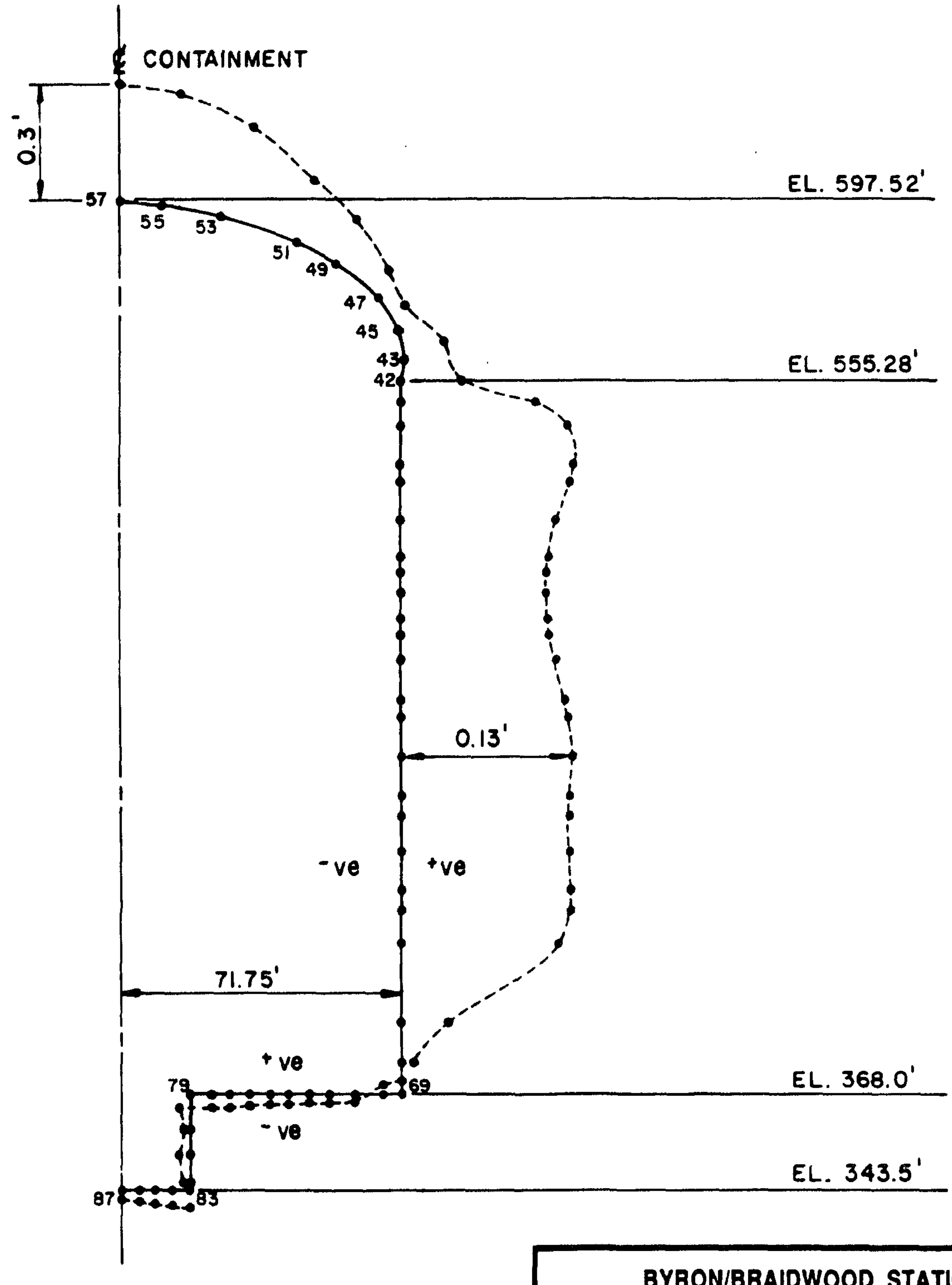

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FIGURE 3.8-93

DISPLACEMENT (FT)

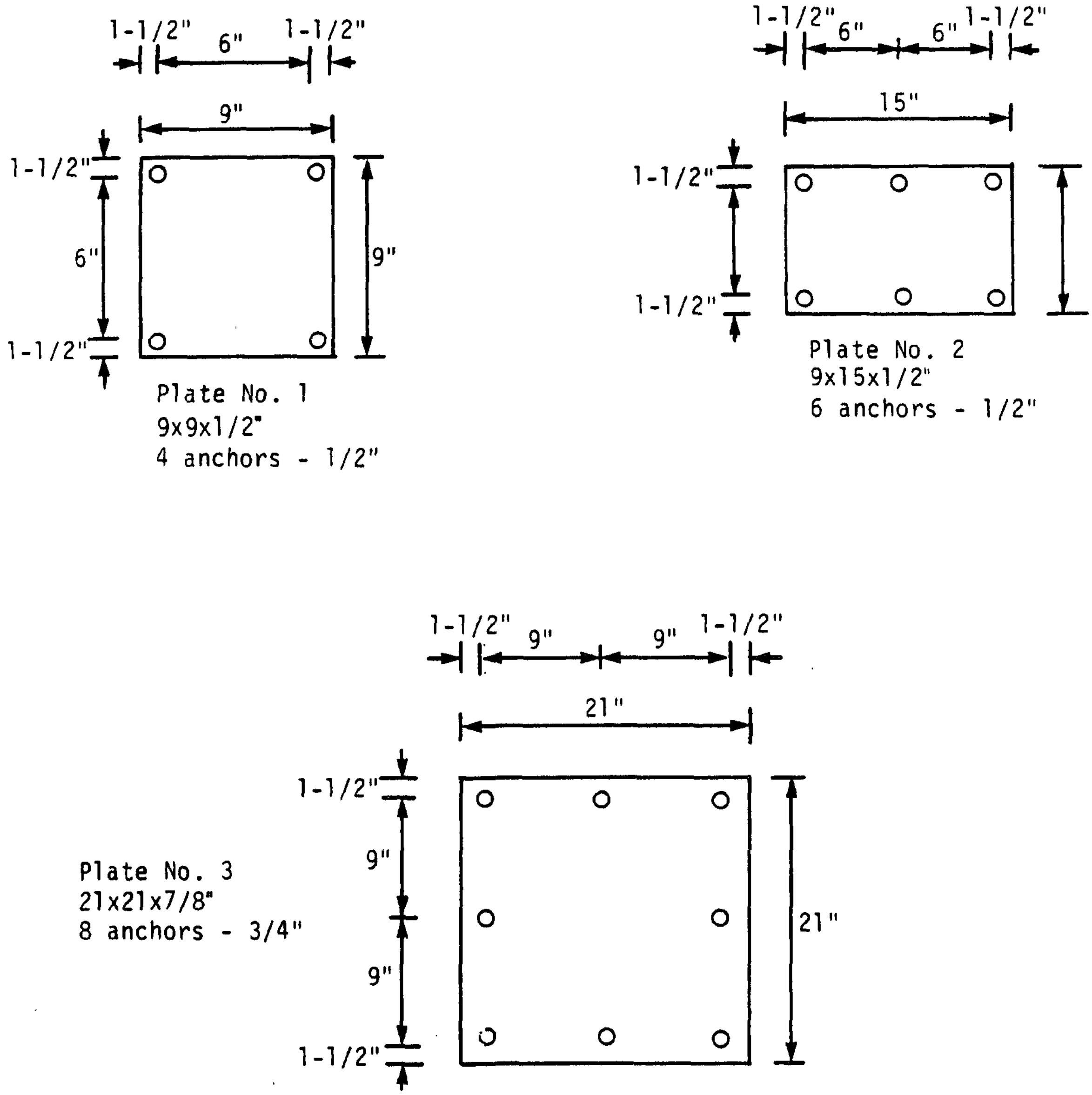

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FIGURE 3.8-94

EXPANSION ANCHOR PLATES

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FIGURE 3.8-95

ESSENTIAL SERVICE WATER
DISCHARGE STRUCTURE

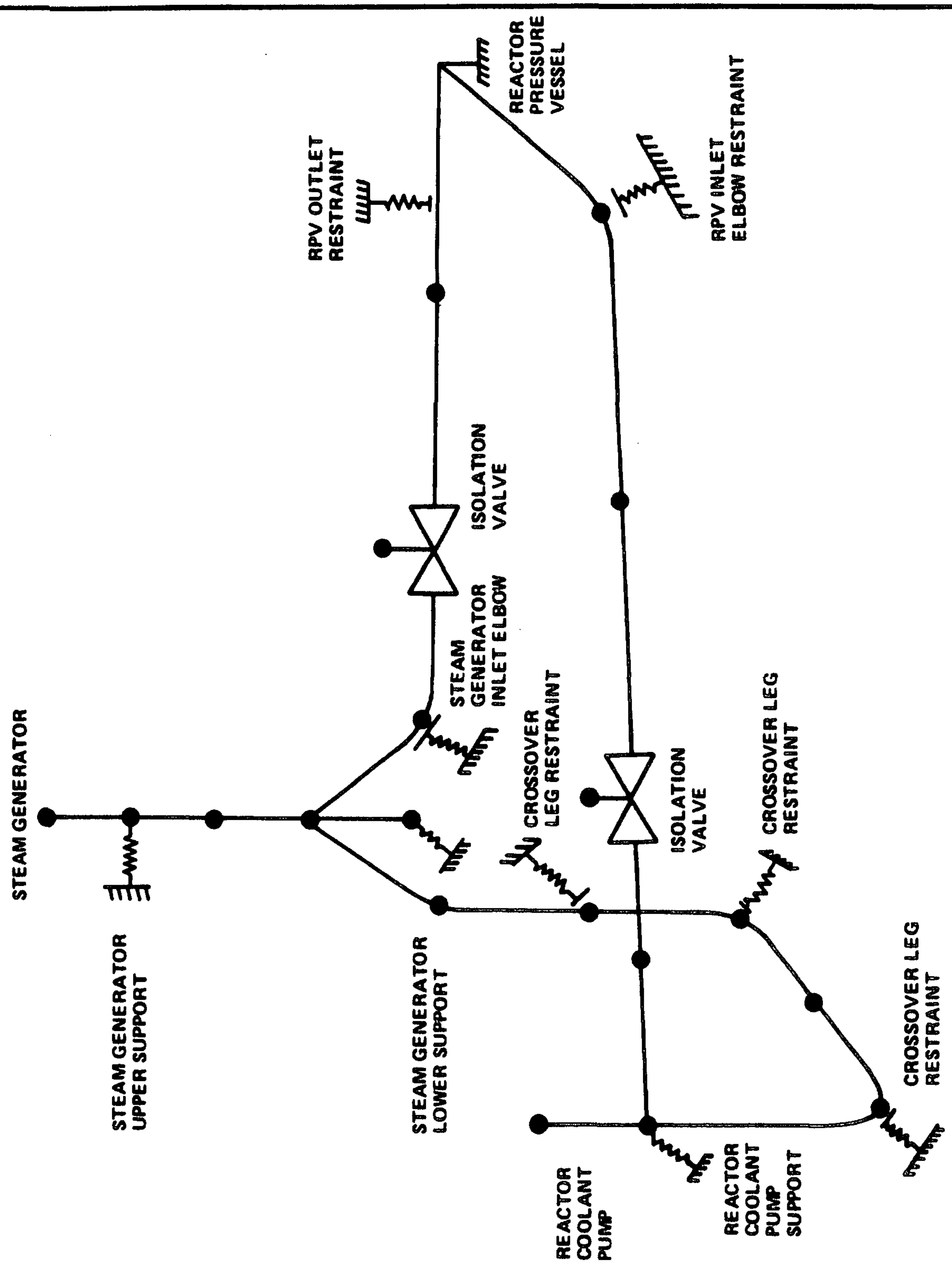

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FIGURE 3.9-1

REACTOR COOLANT LOOP SUPPORTS SYSTEM, DYNAMIC STRUCTURAL MODEL

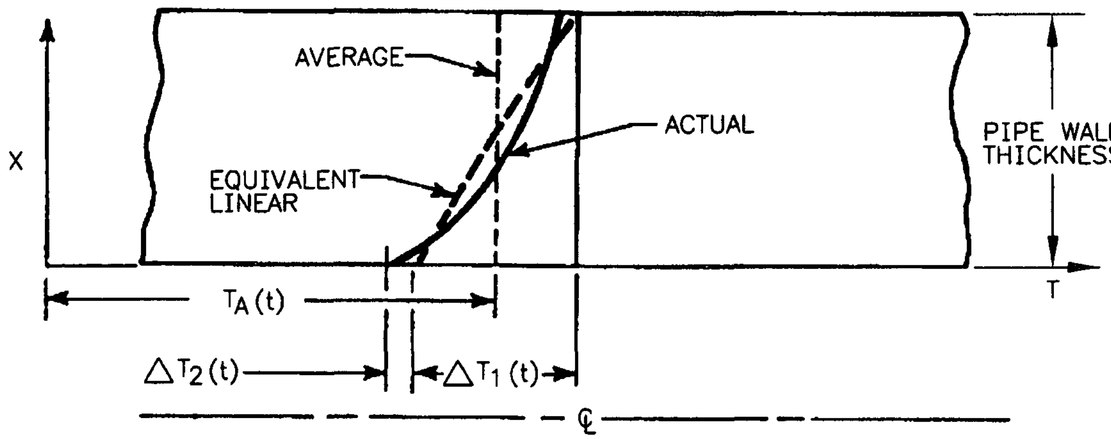

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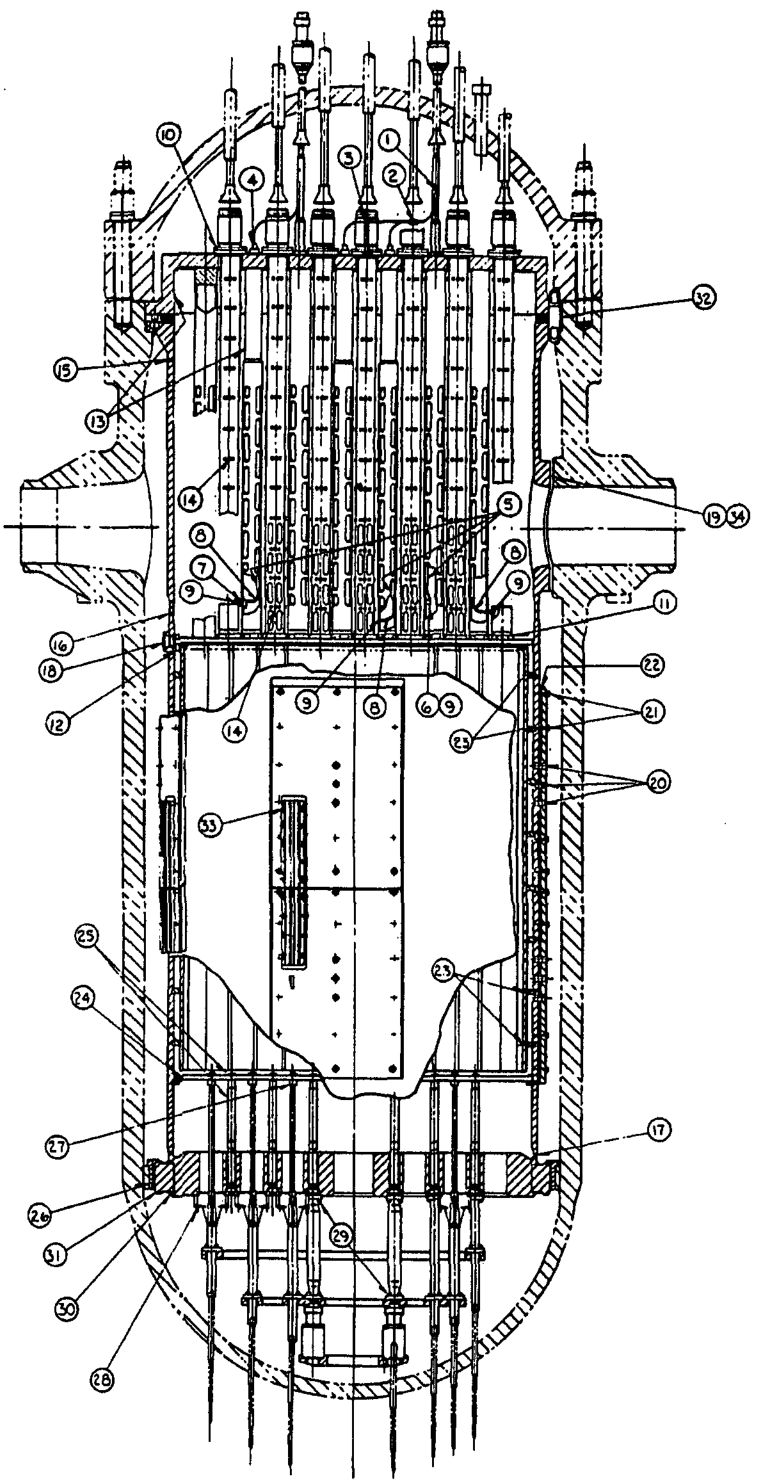
FIGURE 3.9-2
THROUGH-WALL THERMAL GRADIENTS



SEE NOTES	STEP	FEATURES TO BE EXAMINED
1	1	THERMOCOUPLE CONDUIT CLAMPS INSIDE THE THERMOCOUPLE COLUMN.
1, 8	2	CONDUIT CONNECTOR FITTINGS AND CROSS RUN CLAMP ARRANGEMENTS.
1	3	CLAMP ARRANGEMENTS AT THE MOUNTING BRACKET LOCATIONS.
1	4	PLUG TO CONDUIT WELD AT THE FOUR SUPPORT COLUMNS ADJACENT TO THE THERMOCOUPLE COLUMNS.
1, 2	5	ACCESSIBLE ANGLE CONDUIT CLAMPS INSIDE THE UPPER SUPPORT COLUMNS.
1	6	ACCESSIBLE WELD JOINTS AT THE THERMOCOUPLE STOP FOR THE SELF INSTRUMENTED COLUMNS.
1	7	WELD JOINTS ON ACCESSIBLE SUPPORT COLUMN AND MIXING DEVICE GUSSETS (THERMOCOUPLE SUPPORT HARDWARE.)
3	8	RIGIDITY OF EXPOSED PORTION OF THERMOCOUPLE CONDUIT RUNS, AT ACCESSIBLE LOCATIONS. (INSIDE SUPPORT COLUMNS LOWER END.)
4	9	RIGIDNESS OF THE ACCESSIBLE PROTRUDING THERMOCOUPLE TIPS.
1, 5	10	THERMOCOUPLE COLUMN AND GUIDE TUBE SCREW LOCKING DEVICES.
1, 5	11	ACCESSIBLE SUPPORT COLUMN, MIXING DEVICE, ORIFICE PLATE, AND CORE PLATE INSERT SCREW LOCKING DEVICES.
6, 7	12	UPPER CORE PLATE INSERTS.
1	13	DEEP BEAM WELDS AT THE SKIRT AND AT THE OUTER HOLLOW ROUNDS.
1	14	ACCESSIBLE GUIDE TUBE WELDS.
1	15	UPPER BARREL TO FLANGE GIRTH WELD.
1	16	UPPER BARREL TO LOWER BARREL GIRTH WELD.
1	17	LOWER BARREL TO CORE SUPPORT GIRTH WELD.
1, 7	18	UPPER CORE PLATE ALIGNING PIN WELDS AND BEARING SURFACES.
9	19	OUTLET NOZZLE INTERFACE SURFACE CONDITION.
1	20	NEUTRON SHIELD PANEL DOWEL PIN COVER PLATE WELDS.
1, 5	21	NEUTRON SHIELD PANEL SCREW LOCKING DEVICES.
13	22	INTERFACE SURFACES AT THE SPACER PADS ALONG THE TOP AND BOTTOM ENDS OF THE NEUTRON PANELS.
1	23	BAFFLE ASSEMBLY SCREW LOCKING ARRANGEMENTS AT THE TWO TOP AND THE TWO BOTTOM FORMER ELEVATIONS.
5	24	LOWER CORE PLATE TO CORE BARREL FLANGE SCREW LOCKING DEVICES ACCESSIBLE AT THE 0°, 90°, 180°, AND 270° AXES.
1, 5, 10	25	CORE SUPPORT COLUMNS AND THEIR SCREW LOCKING DEVICES.
1, 7, 13	26	VESSEL CLEVIS LOCKING ARRANGEMENTS AND BEARING SURFACES.
11	27	ACCESSIBLE (3) INSTRUMENTATION GUIDE COLUMN LOCKING COLLARS NEAREST THE RUNWAY.
1, 5, 12	28	LOCKING DEVICES AND CONTACT OF THE CRUCIFORM SHAPED BOTTOM INSTRUMENTATION GUIDE COLUMNS WHERE ATTACHED TO THE CORE SUPPORT AND TIE PLATES.
1, 5	29	LOCKING DEVICES OF THE SECONDARY CORE SUPPORT BUTT COLUMNS AT THE CORE SUPPORT, TIE PLATE AND BASE PLATE.
1	30	RADIAL SUPPORT KEY WELDS
1, 7	31	RADIAL SUPPORT KEY LOCKING ARRANGEMENTS AND BEARING SURFACES.
1, 7	32	HEAD AND VESSEL ALIGNING PIN SCREW LOCKING DEVICES AND BEARING SURFACES.
1, 5	33	IRRADIATION SPECIMEN GUIDE SCREW LOCKING DEVICES AND DOWEL PINS
9	34	VESSEL NOZZLE INTERFACE SURFACE CONDITION.

NOTES:

1. VISUALLY EXAMINE WELDS USING 5-10X MAGNIFICATION. NO CRACKS ALLOWED.
2. VERIFY THAT CLAMPS ARE NOT DAMAGED.
3. GRASP EXPOSED CONDUIT & VERIFY THAT IT IS RIGID & TIGHTLY FASTENED.
4. GRASP EXPOSED PORTION OF THERMOCOUPLE TIP. VERIFY THAT TIP DOES NOT MOVE FREELY.
5. VERIFY THAT LOCKING DEVICES ARE CRIMPED & UNDAMAGED.
6. VERIFY THAT INSERTS ARE SEATED (.0015 FEELER MUST NOT PASS THRU INTERFACE) 50% CONTACT REQUIRED.
7. VISUALLY EXAMINE FACES FOR DAMAGE USING 5-10X MAGNIFICATION.
8. VERIFY THAT FITTINGS ARE TIGHT.
9. VISUALLY EXAMINE NOZZLE SURFACES FOR ANY EVIDENCE OF DAMAGE.
10. VERIFY THAT ACCESSIBLE COLUMNS ARE SEATED ON LOWER CORE PLATE. A .0015 FEELER GAGE MUST NOT PASS THRU 90°.
11. VERIFY THAT LOCKING COLLARS ARE TIGHT. NO MOVEMENT ALLOWED.
12. VERIFY SEATING USING A .0015 FEELER GAGE. FEELER MUST NOT PASS THRU 90°.
13. VERIFY SEATING USING A .0015 FEELER GAGE. NO GAPS ALLOWED.

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FIGURE 3.9-3

PRE- AND POST-HOT FUNCTIONAL EXAMINATION POINTS FOR INTERNALS INTEGRITY

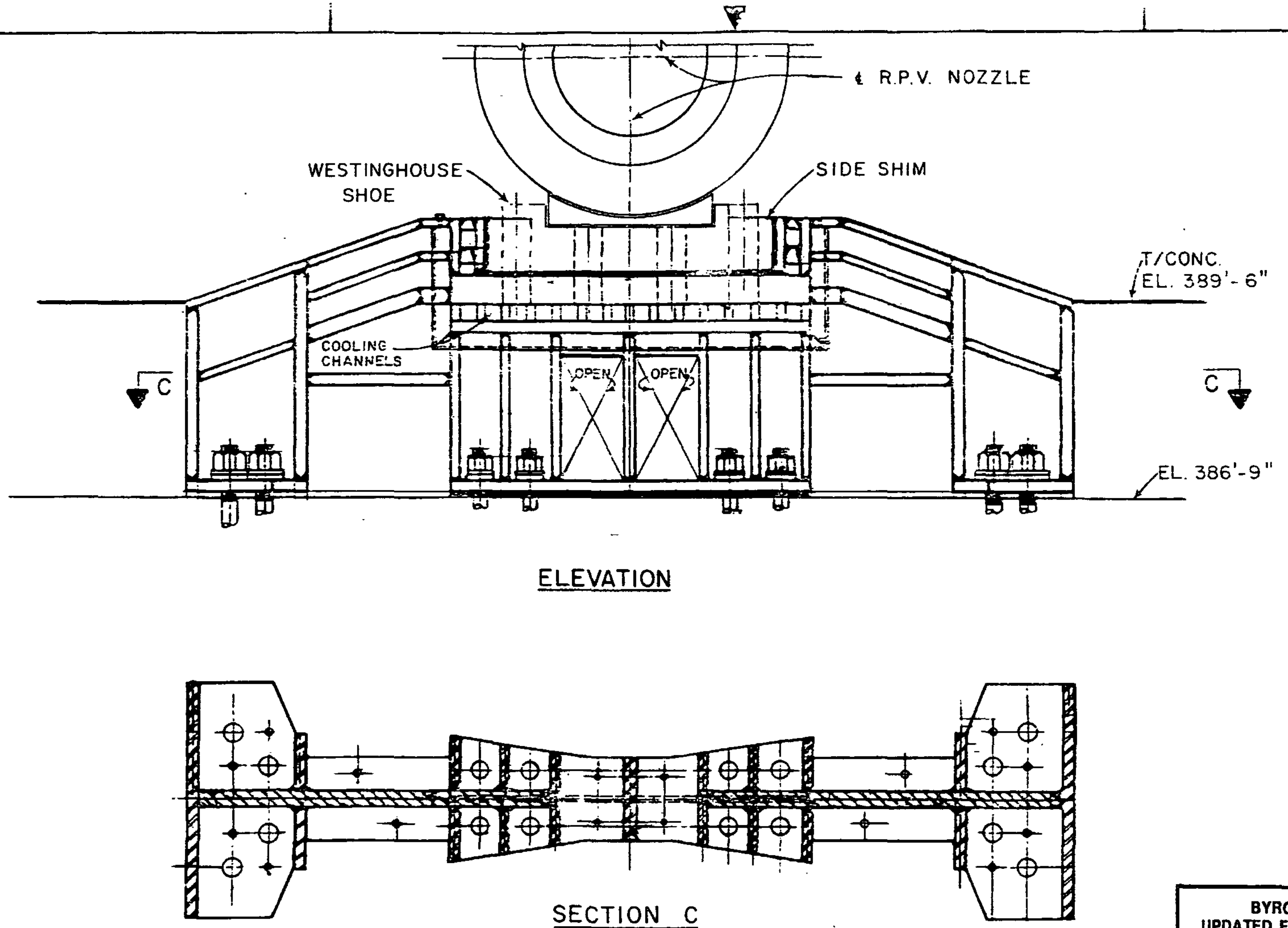

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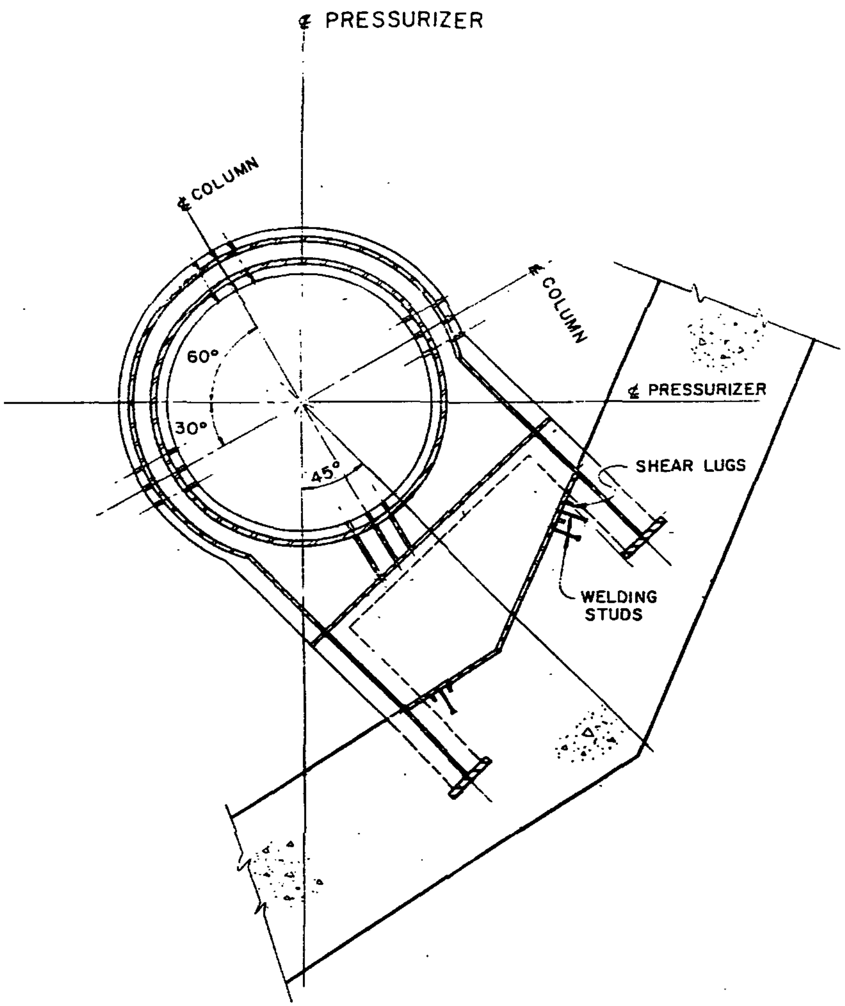
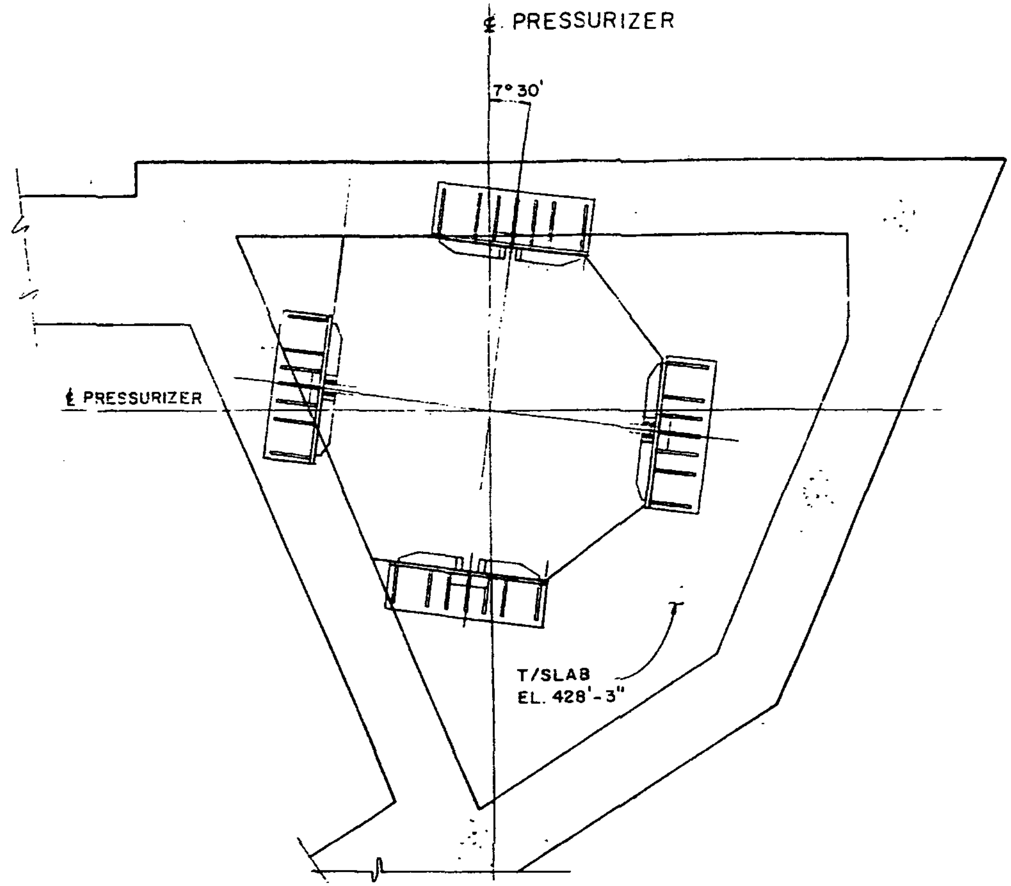
FIGURE 3.9-4

REACTOR VESSEL SUPPORT



LOWER LATERAL SUPPORT



UPPER LATERAL SUPPORT

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FIGURE 3.9-5

PRESSURIZER SUPPORTS

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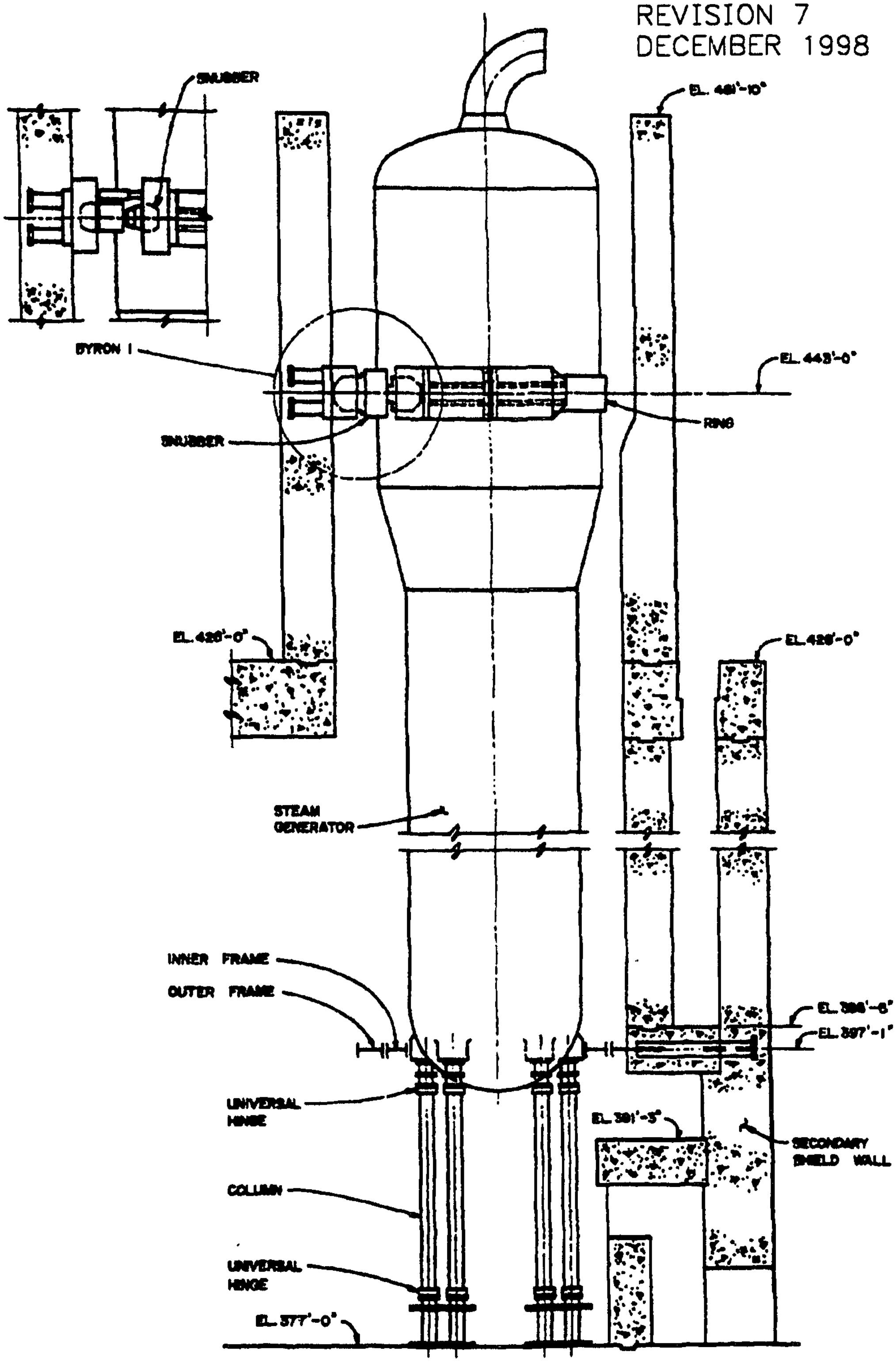

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FIGURE 3.9-6

STEAM GENERATOR
TYPICAL ELEVATION AND SUPPORTS

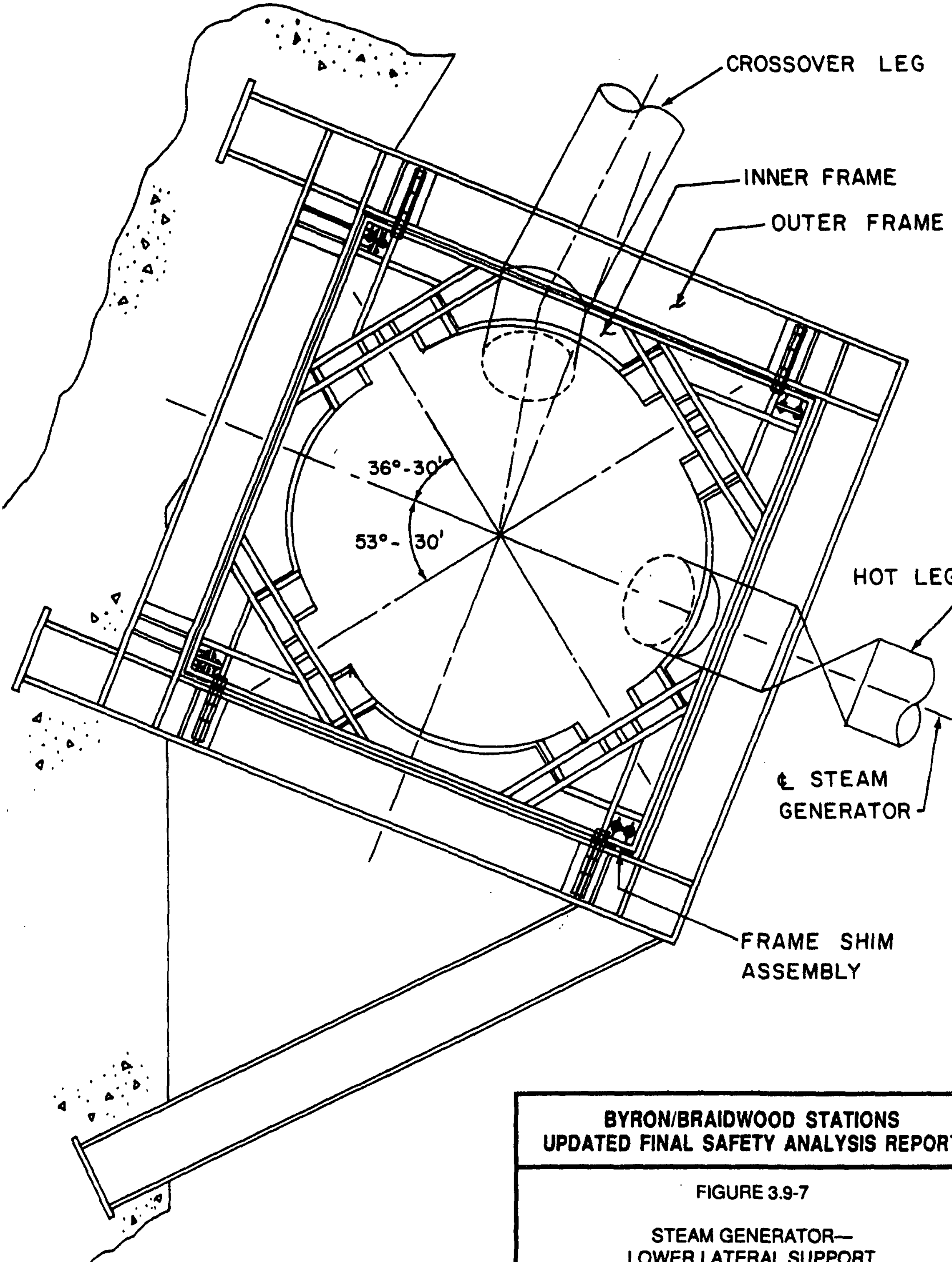

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FIGURE 3.9-7

STEAM GENERATOR—
LOWER LATERAL SUPPORT
(BYRON UNIT 1 ONLY)

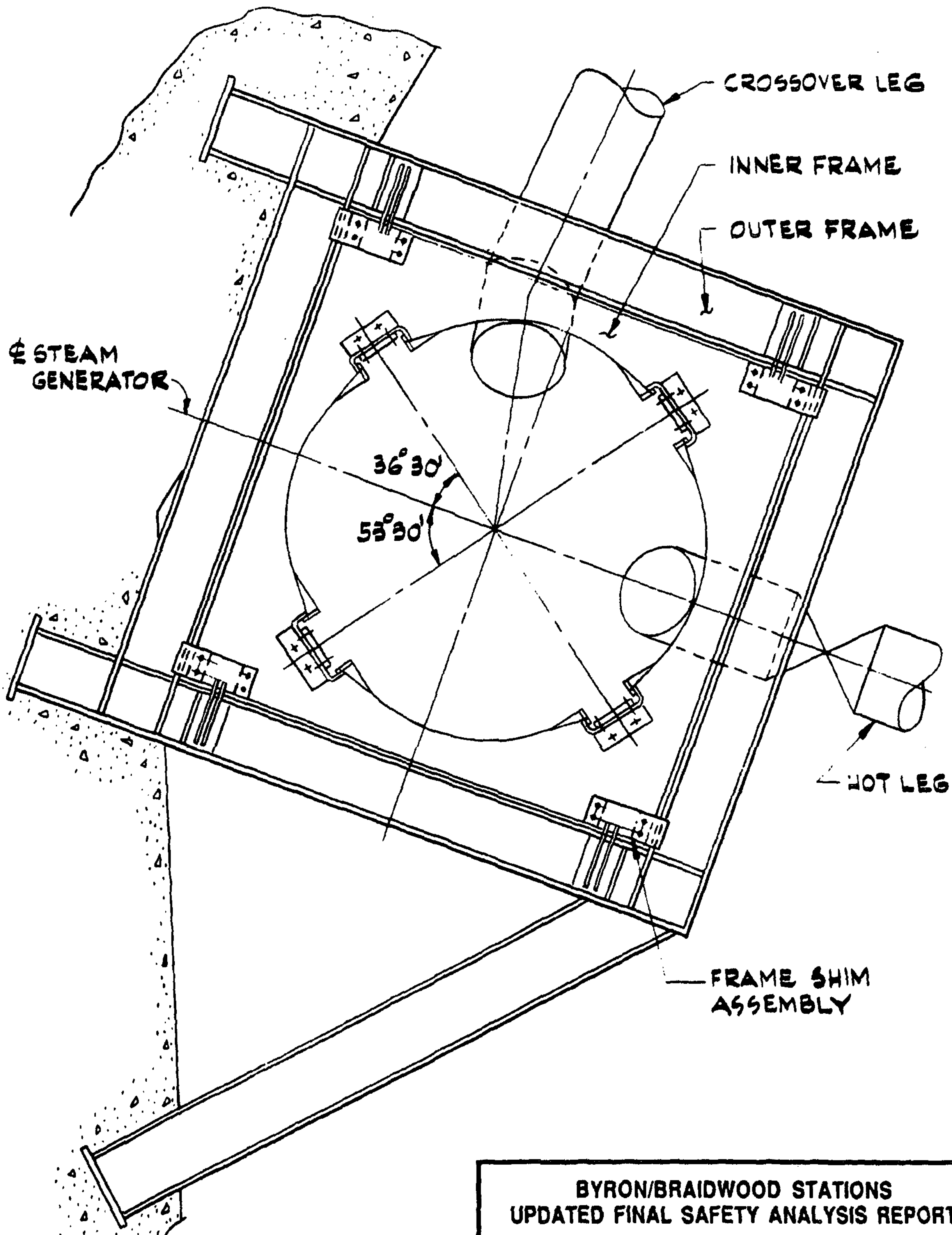

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FIGURE 3.9-7a

STEAM GENERATOR—
LOWER LATERAL SUPPORT
(BYRON UNIT 2 AND BRAIDWOOD UNITS 1 AND 2)

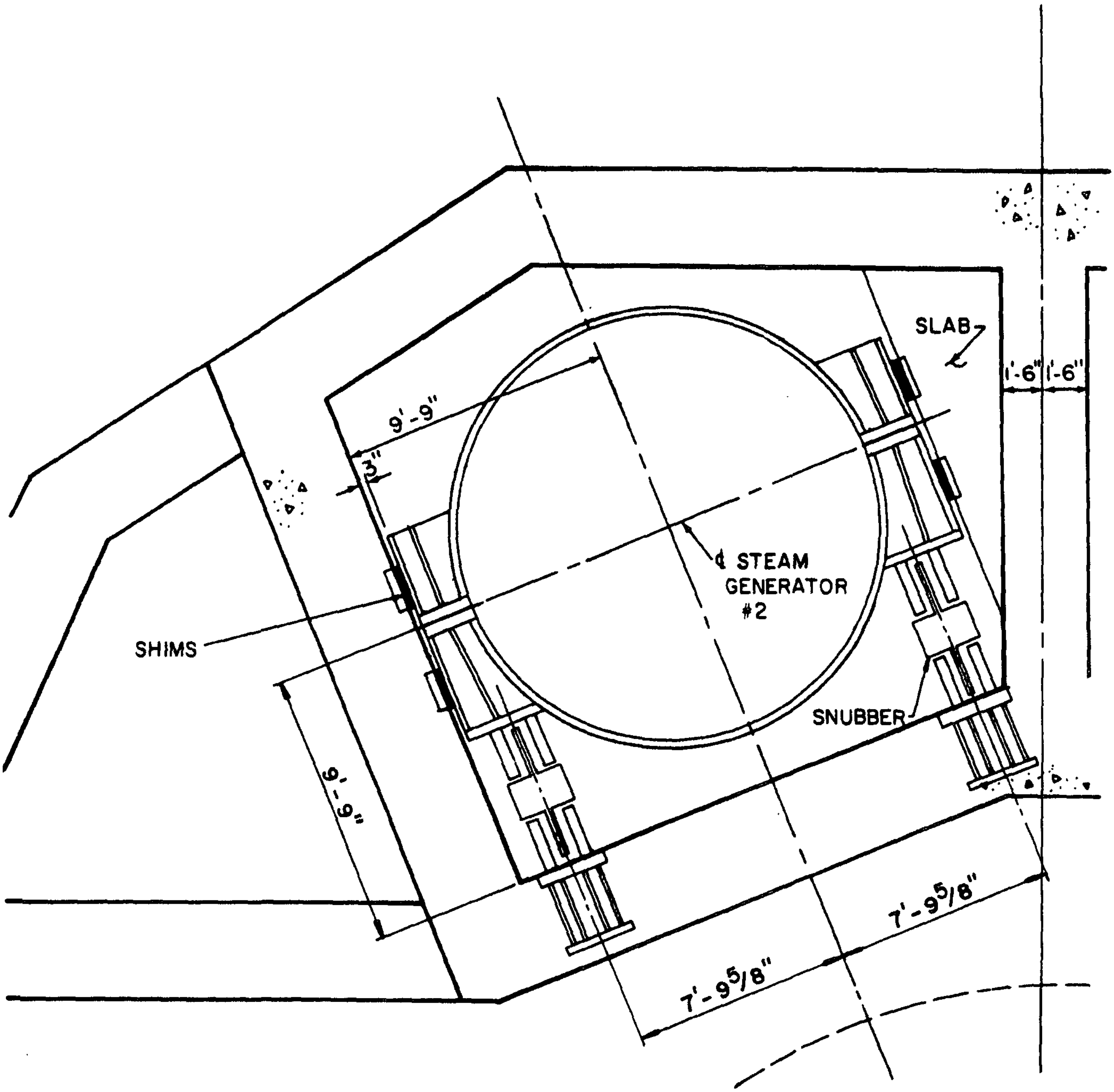

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FIGURE 3.9-8

STEAM GENERATOR—
UPPER LATERAL SUPPORT

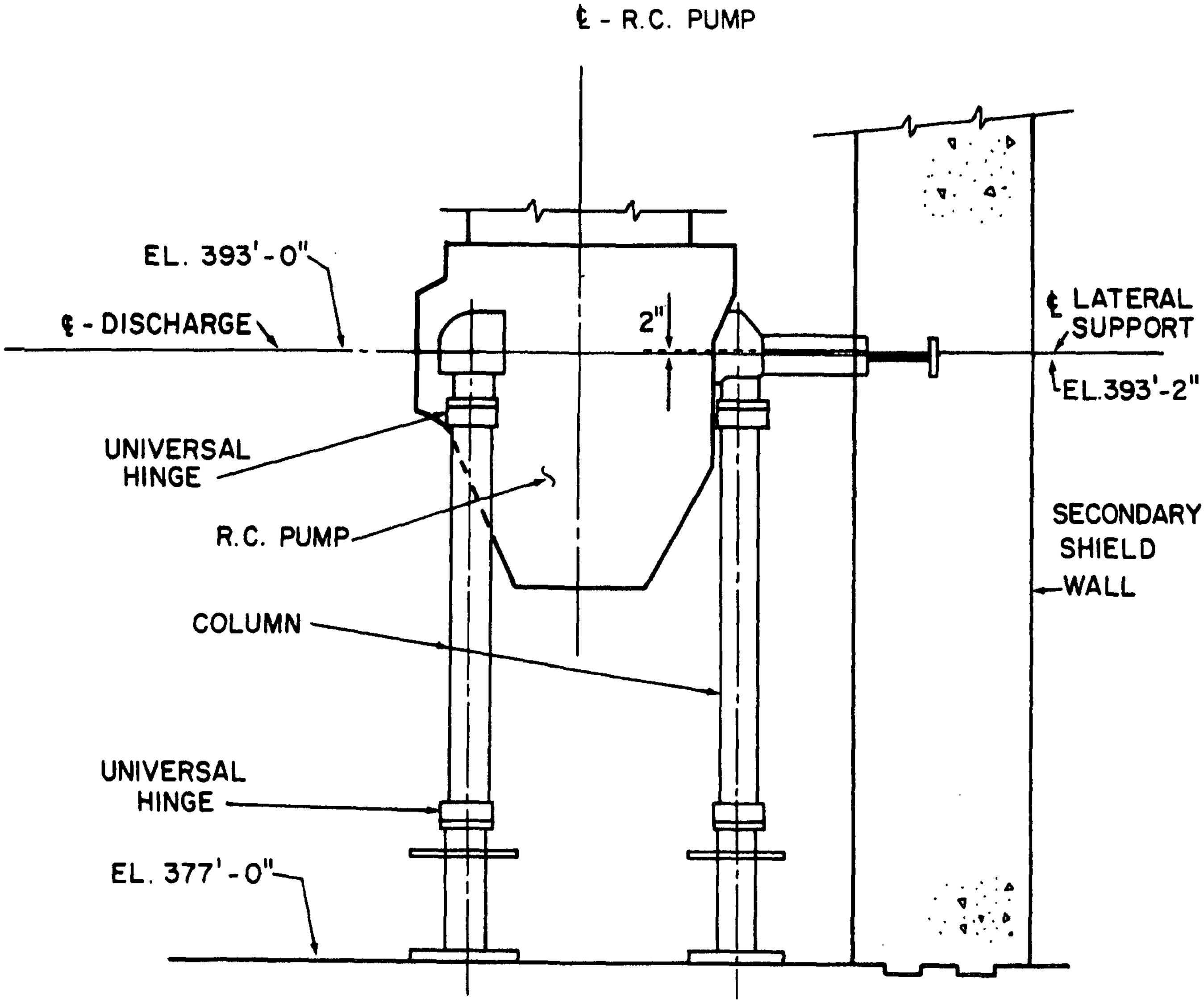

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FIGURE 3.9-9

REACTOR COOLANT PUMP—
TYPICAL ELEVATIONS AND SUPPORT

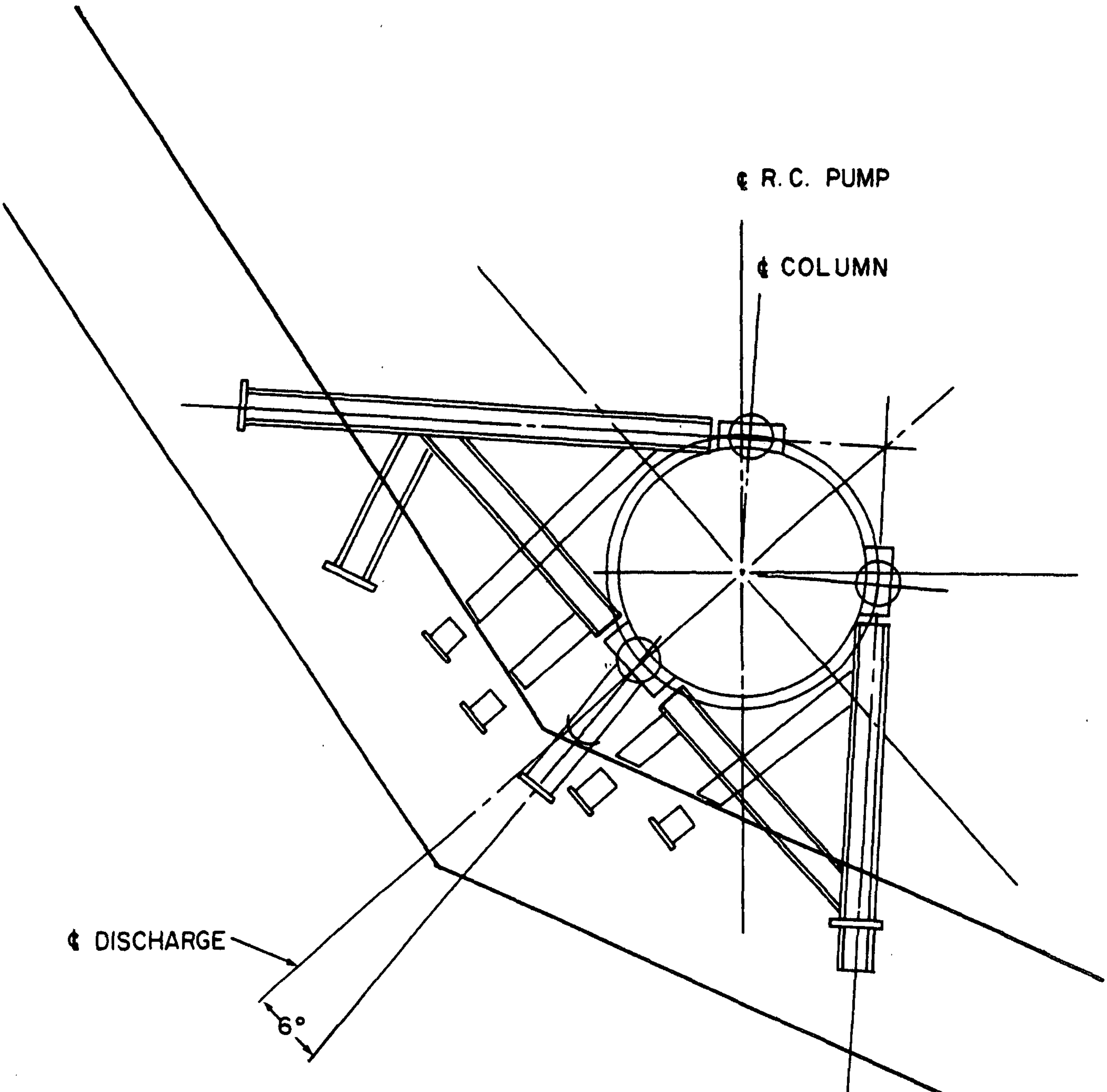

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FIGURE 3.9-10

REACTOR COOLANT PUMP—
LATERAL SUPPORT

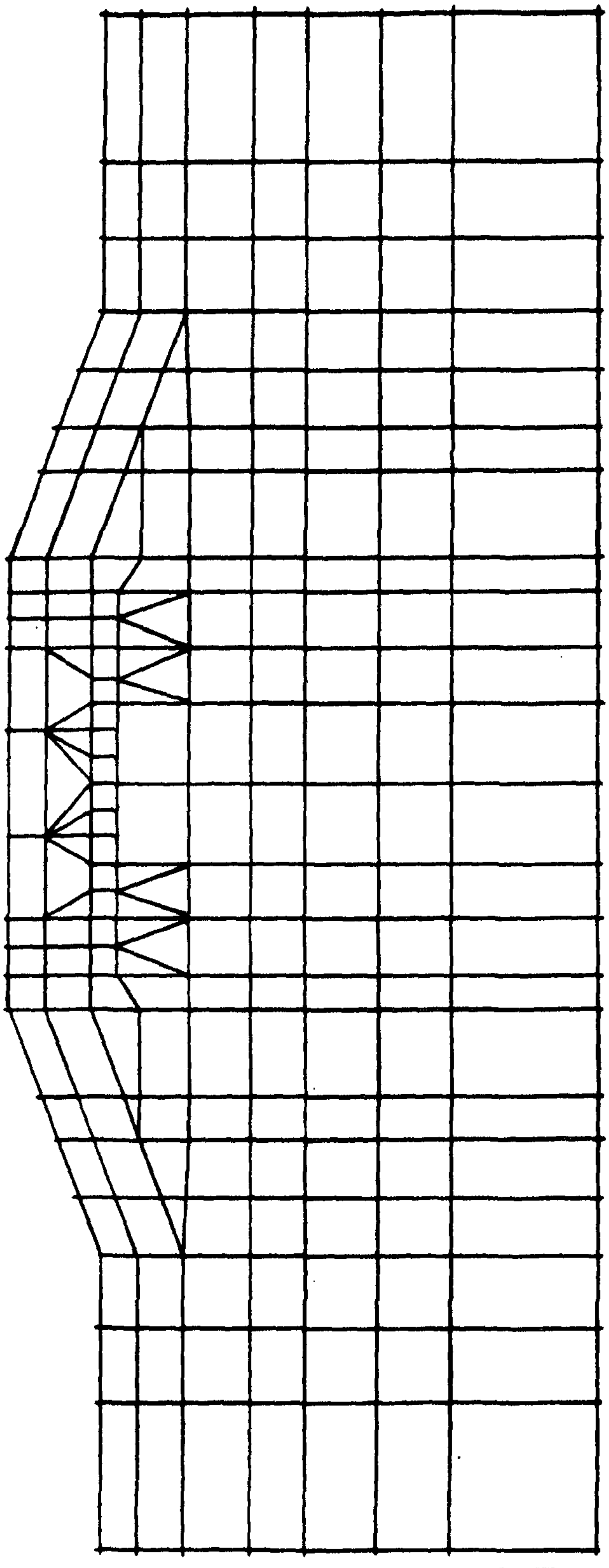

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FIGURE 3.9-11

FINITE-ELEMENT MODEL—
REACTOR PRESSURE VESSEL SUPPORT

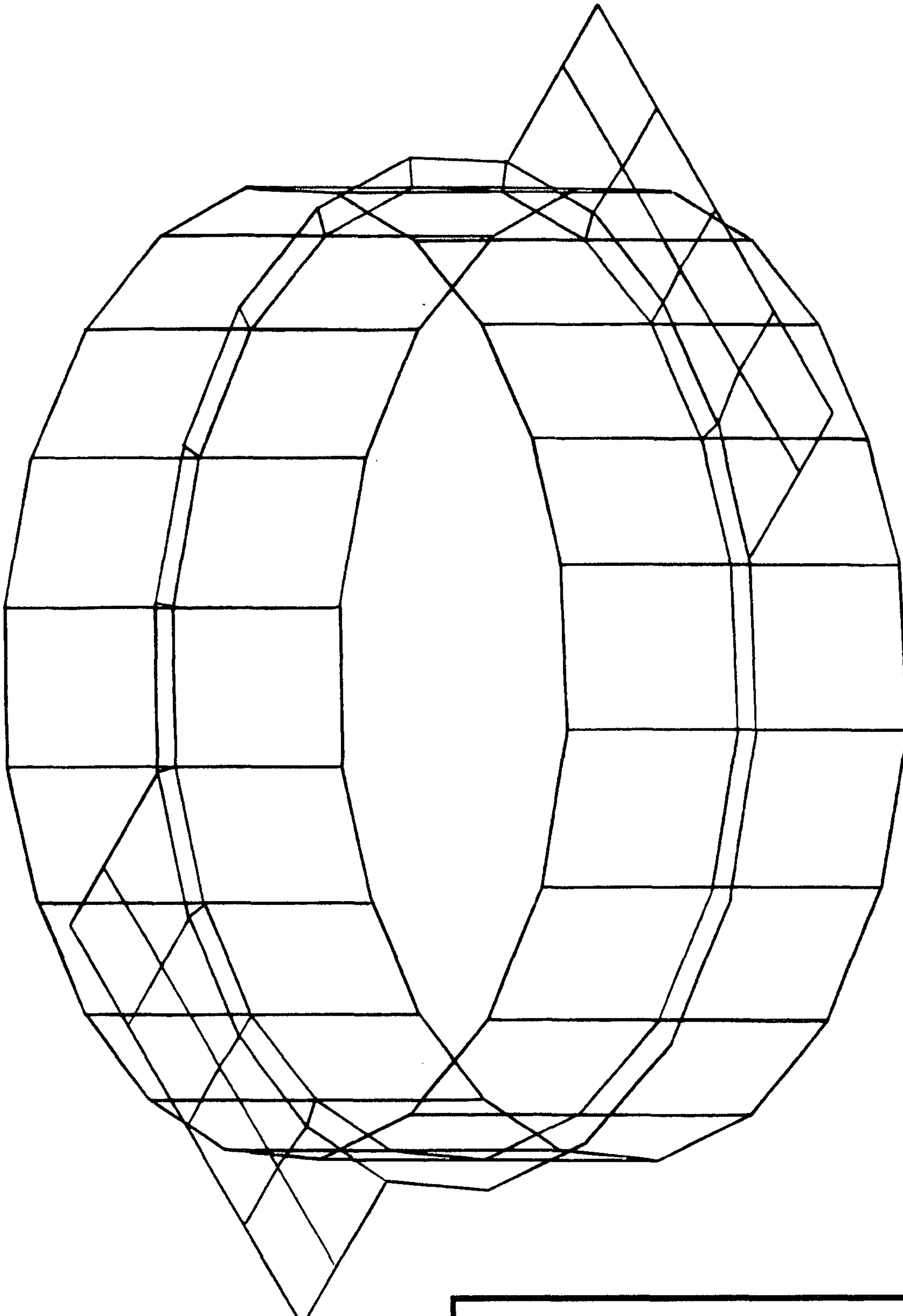

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FIGURE 3.9-12

FINITE-ELEMENT MODEL—
STEAM GENERATOR UPPER LATERAL SUPPORT

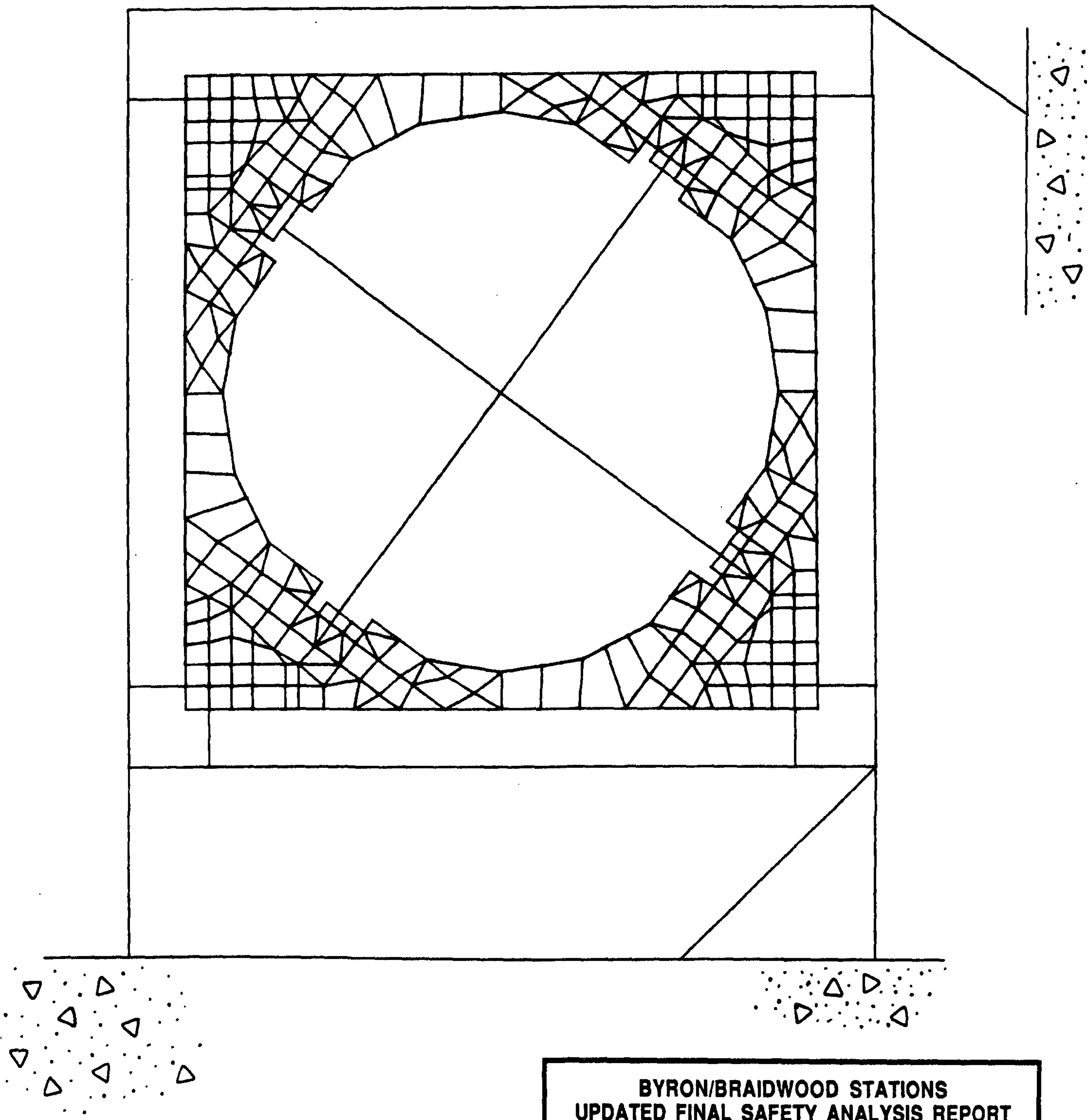

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FIGURE 3.9-13

FINITE-ELEMENT MODEL—
STEAM GENERATOR LOWER LATERAL SUPPORT

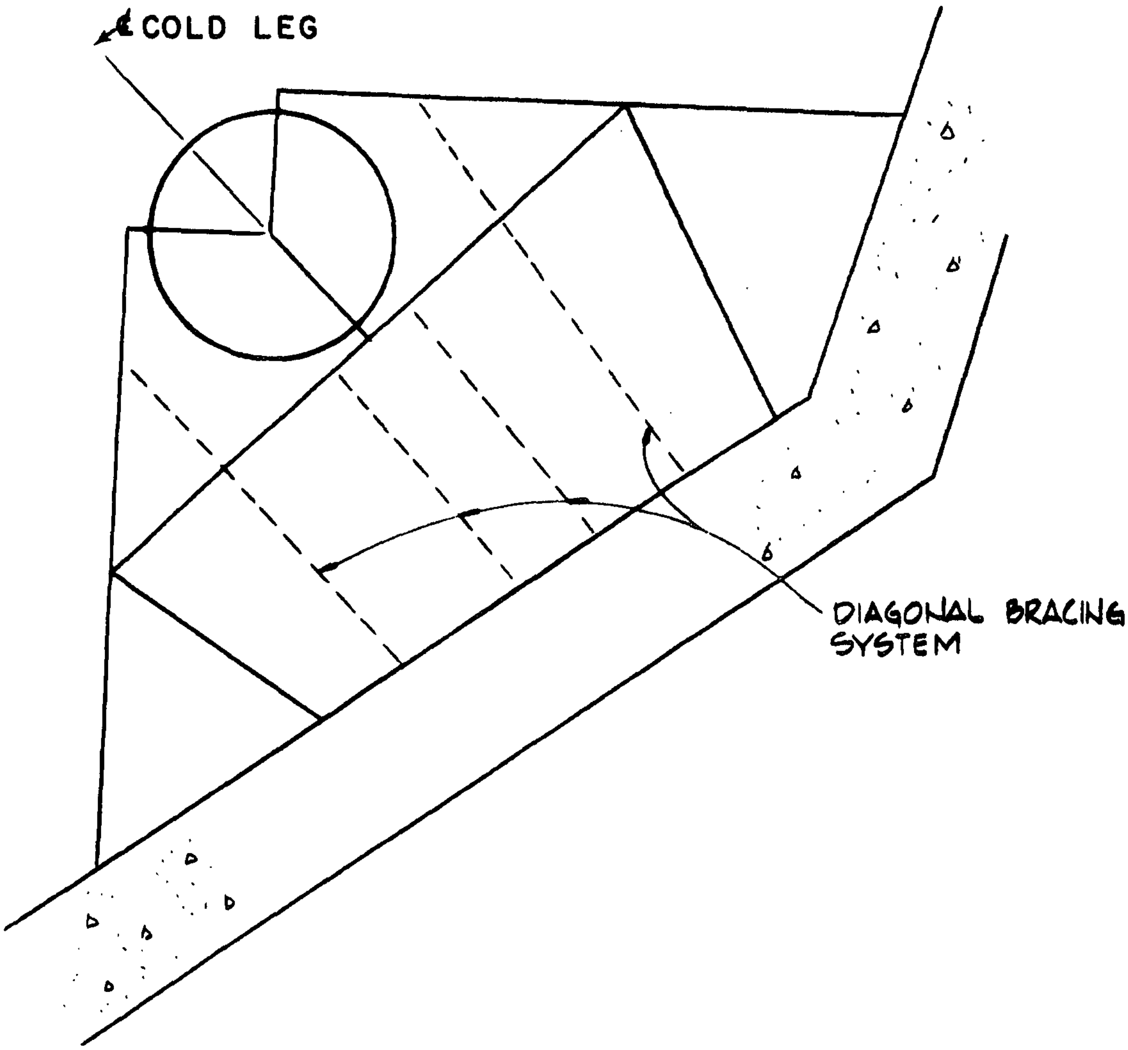

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FIGURE 3.9-14

FINITE-ELEMENT MODEL—
REACTOR COOLANT SUPPORT

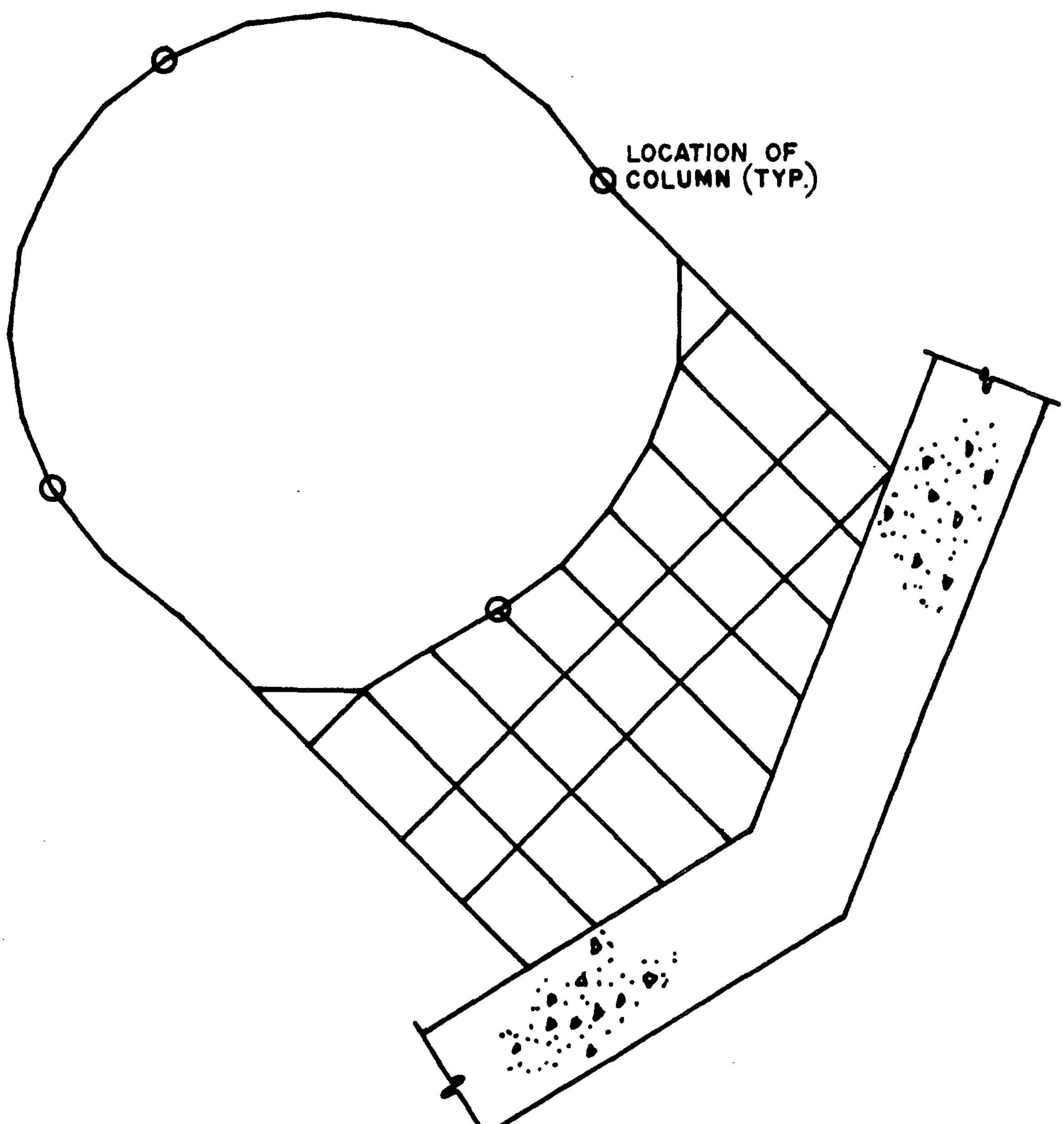

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FIGURE 3.9-15

FINITE-ELEMENT MODEL—
PRESSURIZER LOWER LATERAL SUPPORT

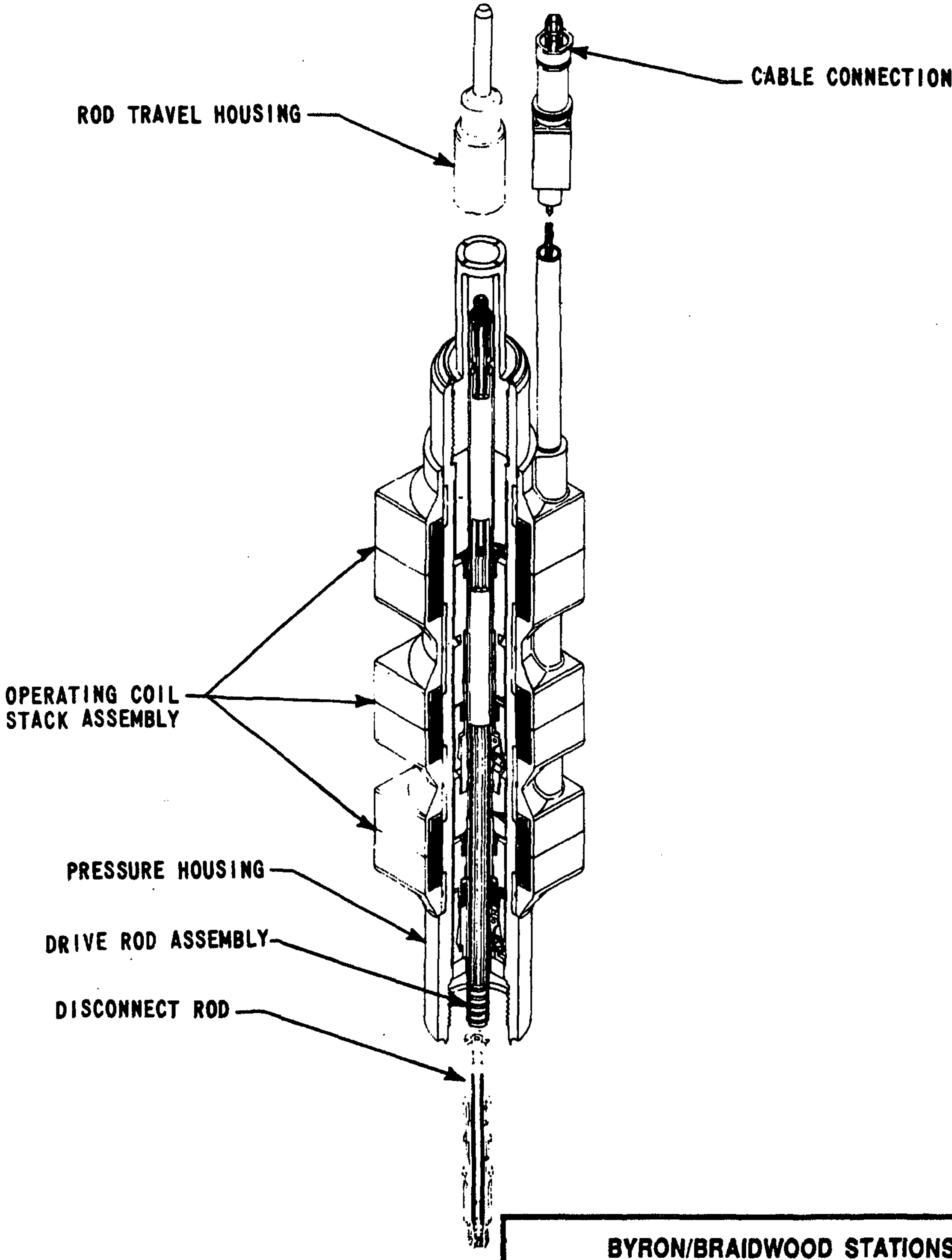

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FIGURE 3.9-16

FULL-LENGTH CONTROL ROD
DRIVE MECHANISM

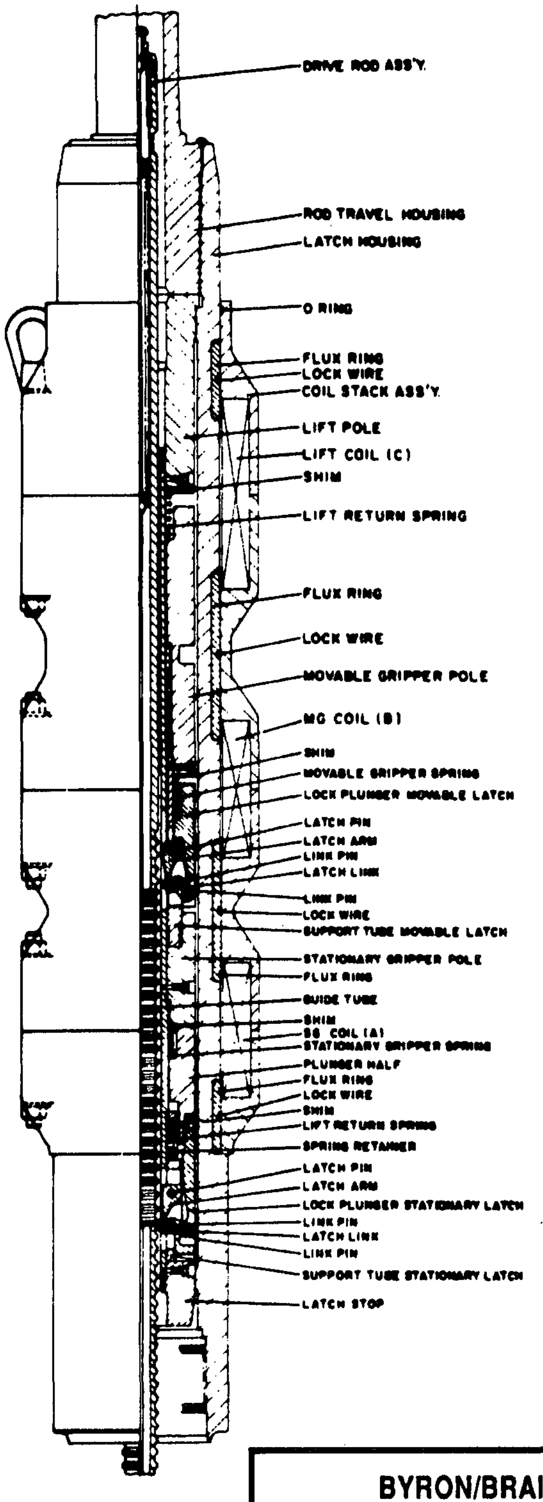

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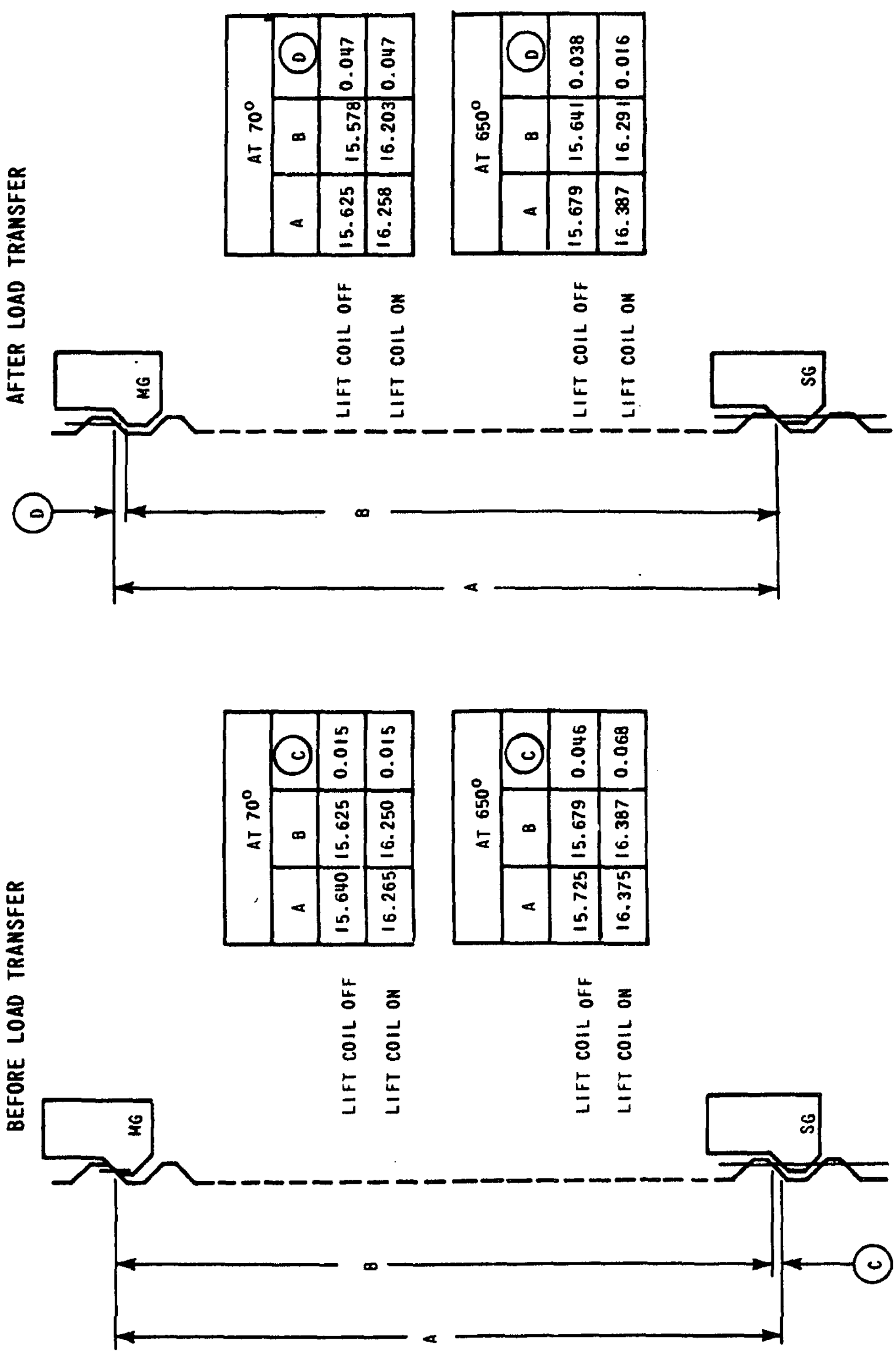
FIGURE 3.9-17

FULL-LENGTH CONTROL ROD
DRIVE MECHANISM SCHEMATIC



AFTER LOAD TRANSFER

AT 70°	A	B	D
LIFT COIL OFF	15.625	15.578	0.047
LIFT COIL ON	16.258	16.203	0.047

AT 650°	A	B	D
LIFT COIL OFF	15.679	15.641	0.038
LIFT COIL ON	16.387	16.291	0.016

BEFORE LOAD TRANSFER

AT 70°	A	B	C
LIFT COIL OFF	15.640	15.625	0.015
LIFT COIL ON	16.265	16.250	0.015

AT 650°	A	B	C
LIFT COIL OFF	15.725	15.679	0.046
LIFT COIL ON	16.375	16.387	0.068

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FIGURE 3.9-18

NOMINAL LATCH CLEARANCE AT MINIMUM AND MAXIMUM TEMPERATURE

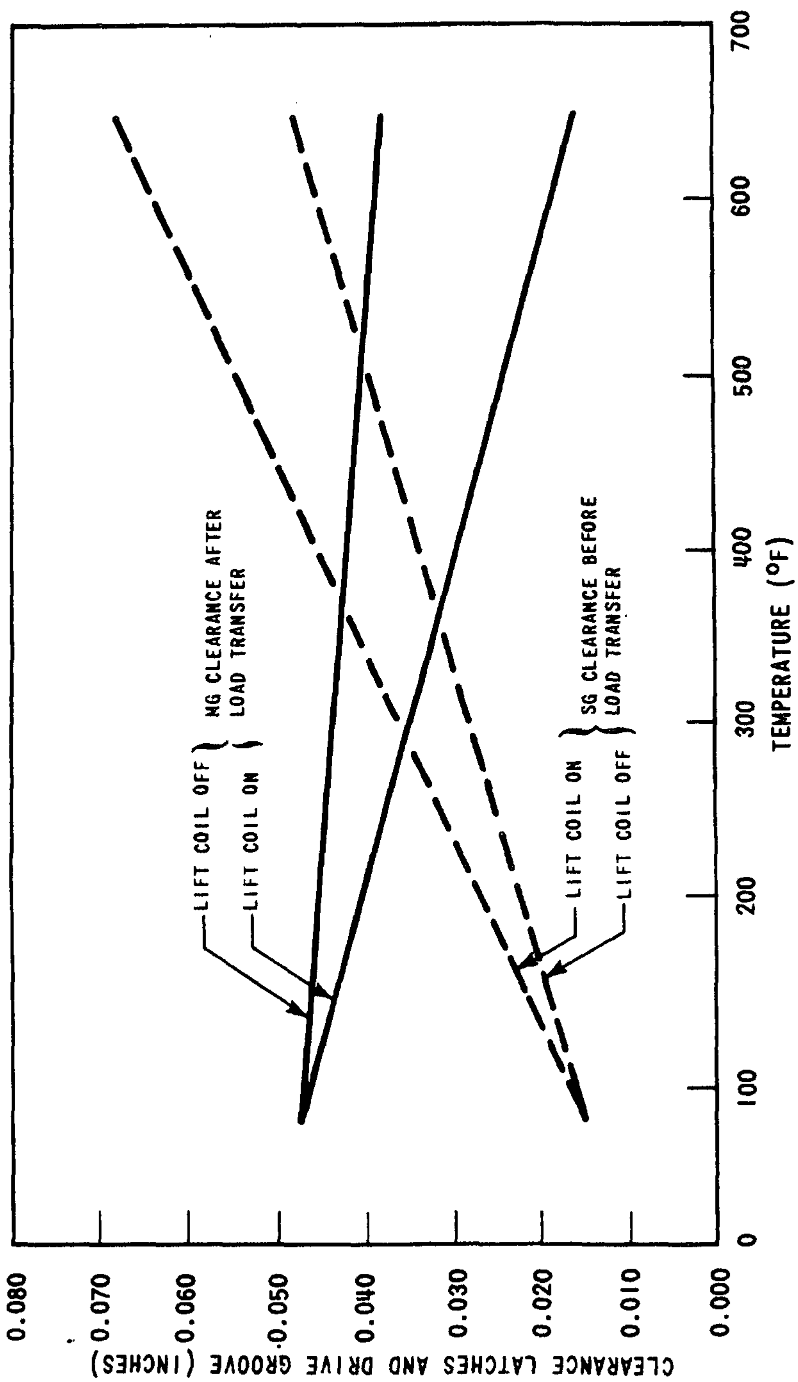

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FIGURE 3.9-19

CONTROL ROD DRIVE MECHANISM LATCH
CLEARANCE THERMAL EFFECTS

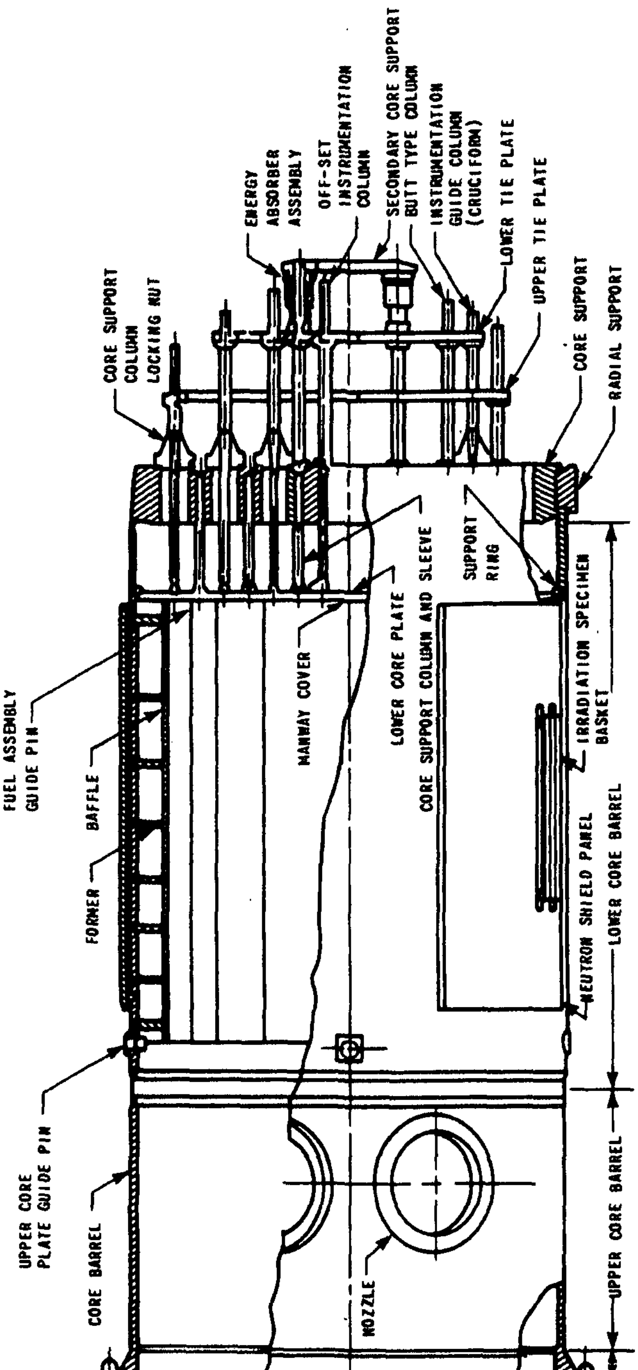

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FIGURE 3.9-20

LOWER CORE SUPPORT ASSEMBLY
(CORE BARREL ASSEMBLY)

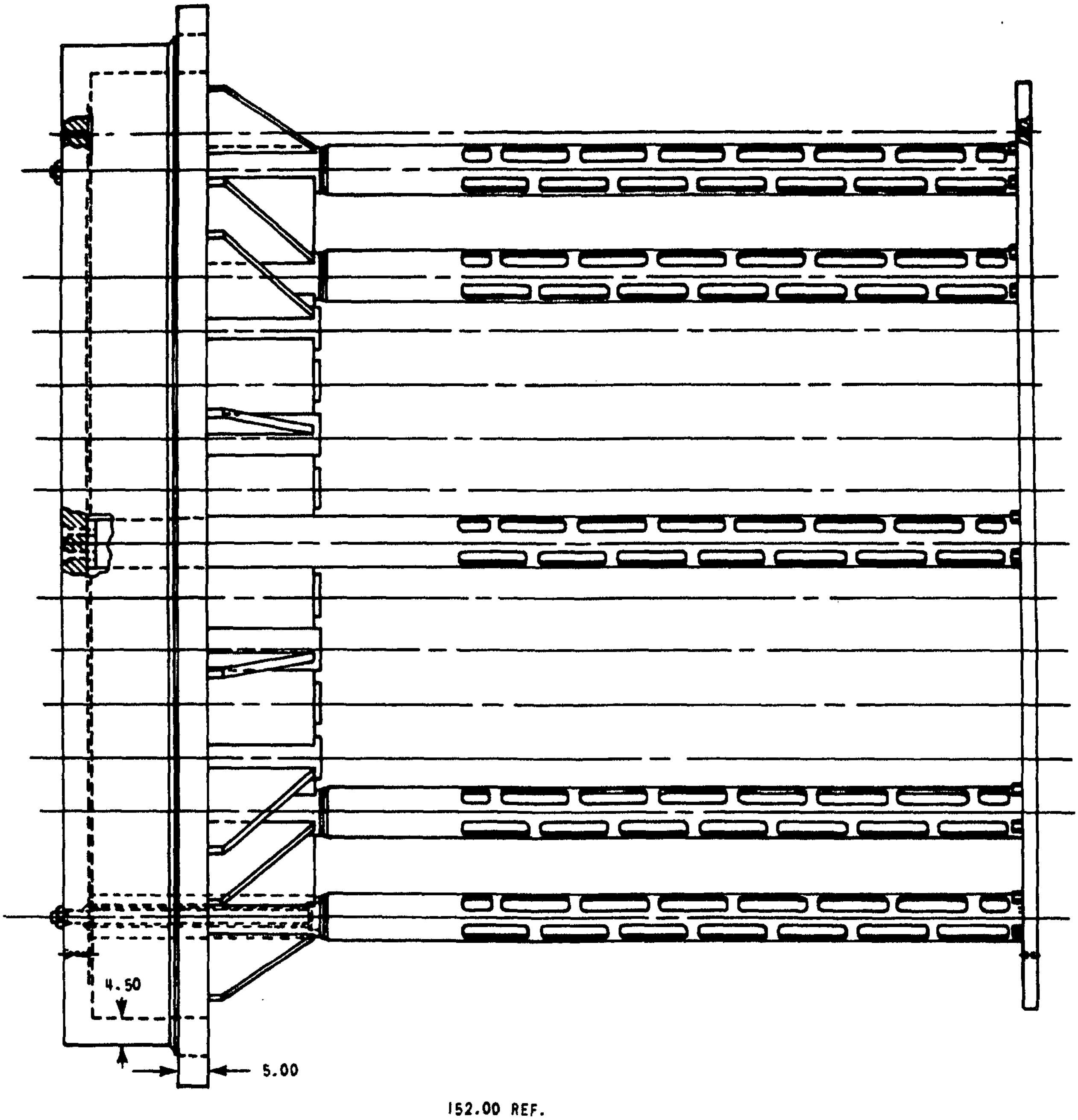

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FIGURE 3.9-21

UPPER CORE SUPPORT STRUCTURE

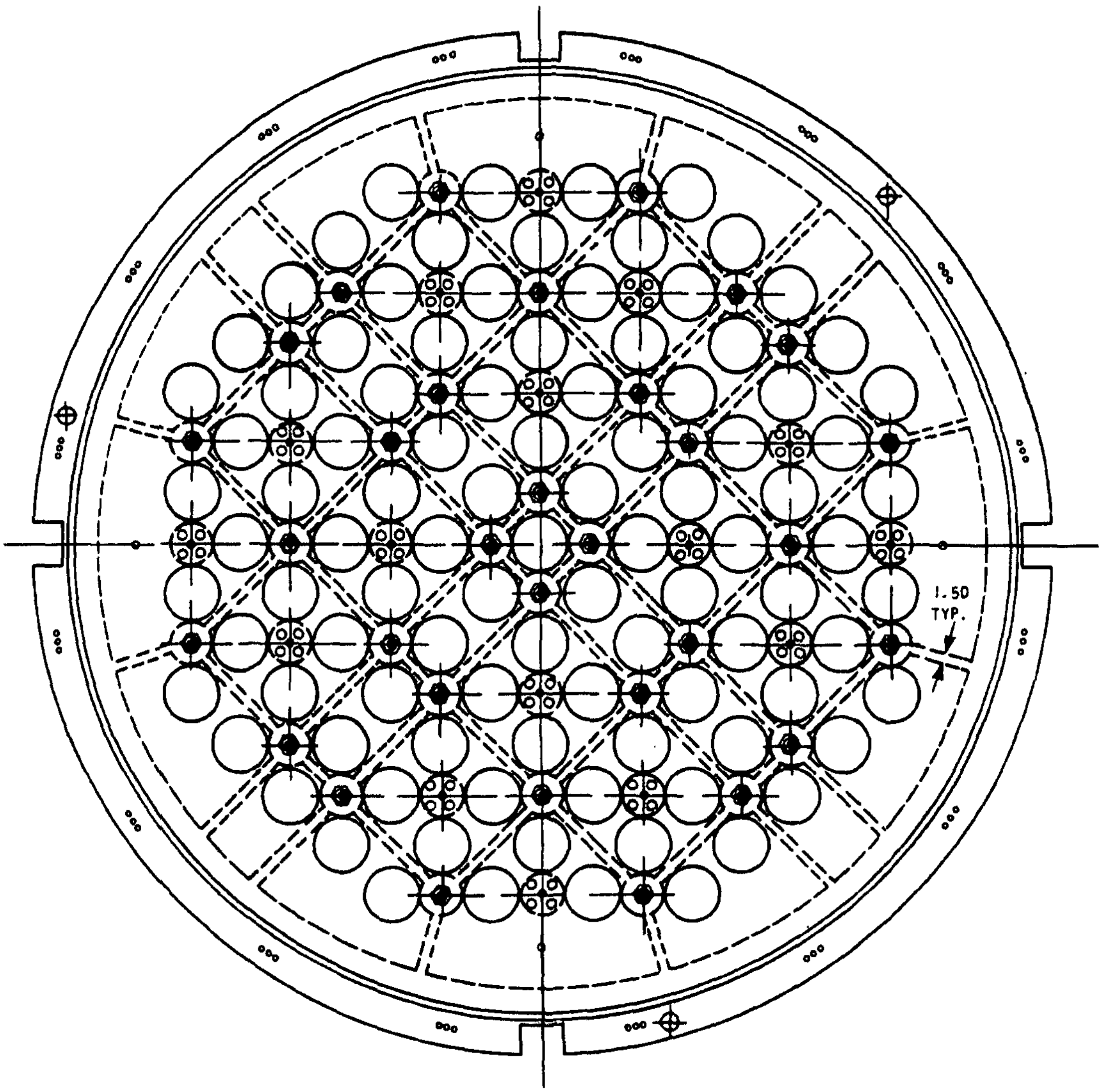

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FIGURE 3.9-22

PLAN VIEW OF UPPER CORE SUPPORT STRUCTURE

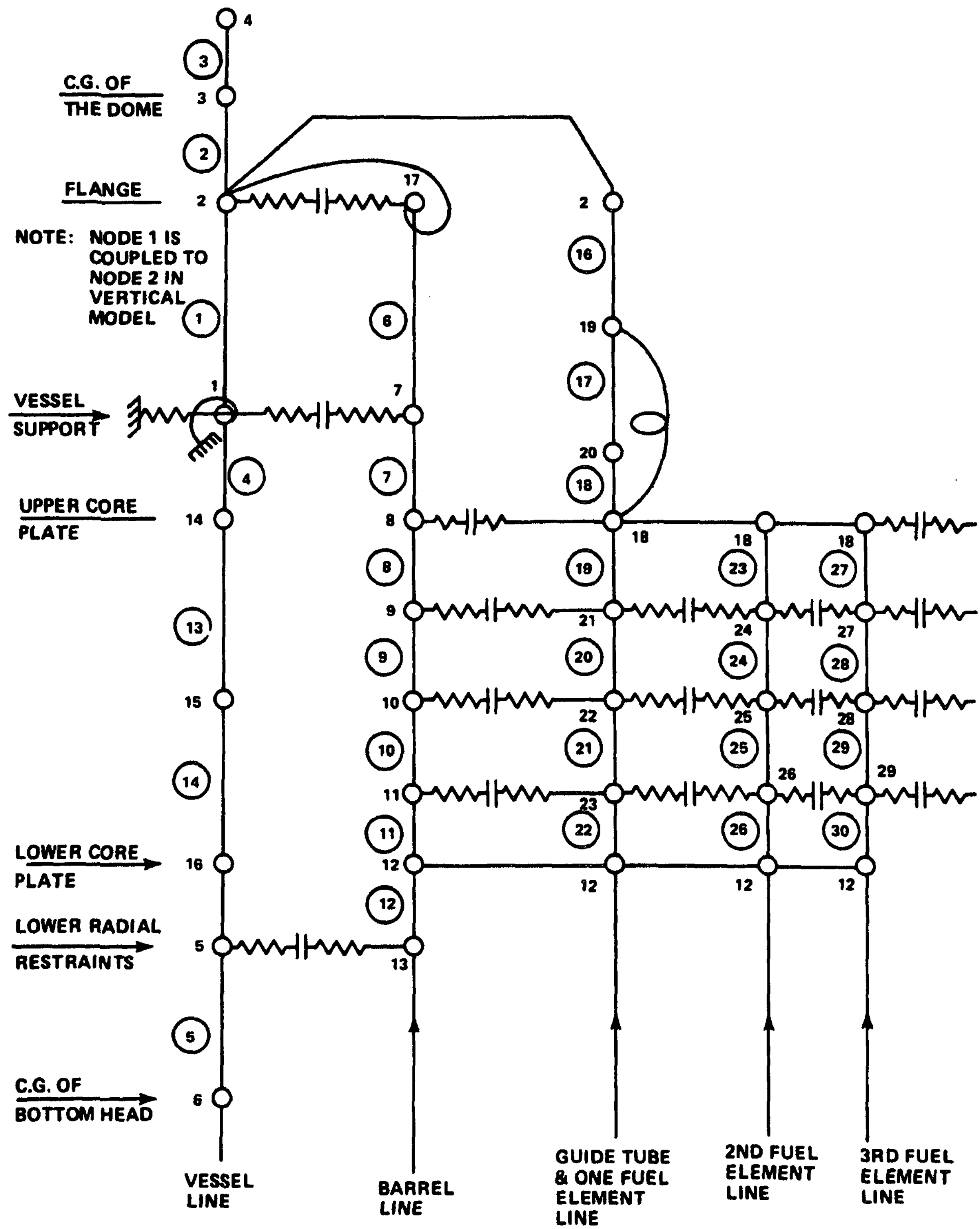

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FIGURE 3.9-23

REACTOR INTERNALS MODEL FOR DAR12 VARIABLES

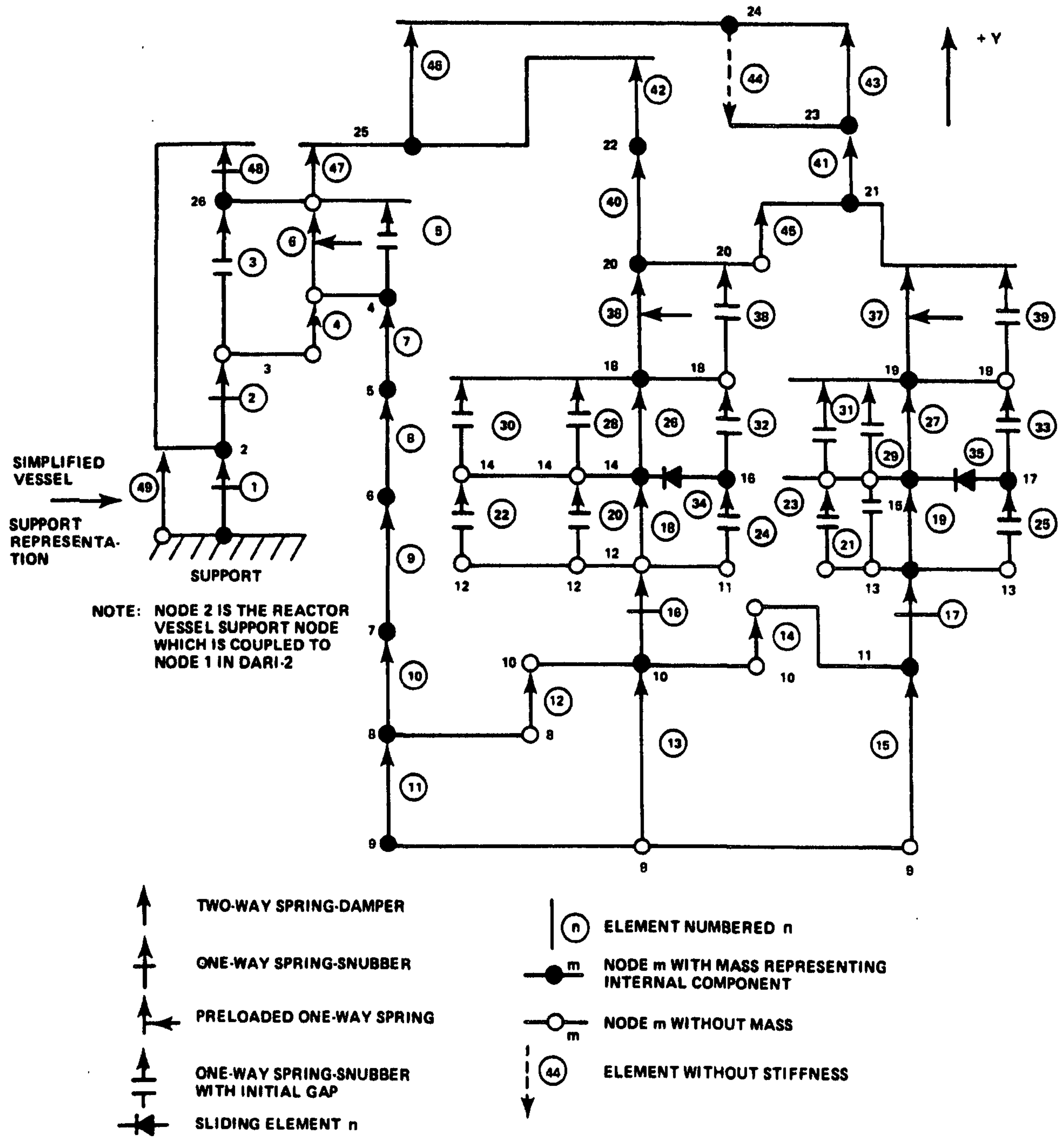

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FIGURE 3.9-24

REACTOR INTERNALS MATHEMATICAL MODEL
FOR WOSTAS VARIABLES