CHAPTER 3: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE WITH GENERAL DESIGN CRITERIA

3.1.1 Summary Description

This section contains an evaluation of the design basis of Fermi 2 as measured against the General Design Criteria (GDC) for Nuclear Power Plants, Appendix A of 10 CFR 50, effective May 21, 1971, and subsequently amended July 7, 1971. The General Design Criteria, which are divided into six groups, are intended to establish minimum requirements for the design of nuclear power plants.

The GDC were not written specifically for the BWR; rather, they were intended as a guide to the design of all water-cooled nuclear power plants. As a result, the criteria are generic in nature and subject to a variety of interpretations. For this reason, there are some cases where conformance to a particular criterion is not directly assessable. In these cases, the conformance of plant design to the interpretation of the criterion is discussed. In the discussion of each criterion, the section of the UFSAR where more detailed information is presented to demonstrate compliance with or exception to the criterion is referenced.

Based on the content herein, Edison concludes that the design of Fermi 2 is in accordance with and satisfies the GDC.

3.1.2 Criterion Conformance

3.1.2.1 Group I, Overall Requirements (Criteria 1-5)

3.1.2.1.1 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.

Criterion 1 Conformance - Structures, systems, and components important to safety are listed in Table 3.2-1. The total Quality Assurance Program is described in Chapter 17 and is applied to the items contained in Table 3.2-1. The Quality Assurance Program ensures sound engineering in all phases of design and construction through conformity to regulatory requirements and design bases described in the license application. In addition, the program ensures adherence to specified standards in fabrication and construction. It also includes the observance of proper preoperational and operational testing and maintenance procedures. Documentation of the foregoing is provided by keeping appropriate records. The total

Quality Assurance Program of Edison and its principal contractors is responsive to and satisfies the intent of the quality-related requirements of 10 CFR 50, including Appendix B.

Structures, systems, and components are first classified in Section 3.2 with respect to their location and service, and their relationship to the safety function to be performed. Recognized codes and standards are applied to the equipment in these classifications, as necessary, to ensure a quality product in keeping with the required safety function.

Documents are maintained that demonstrate that all the requirements of the Quality Assurance Program are being satisfied. This documentation shows that appropriate codes, standards, and regulatory requirements are observed, specified materials are used, correct procedures are used, qualified personnel are provided, and the finished parts and components meet the applicable specifications for safe and reliable operation. These Nuclear Quality Assurance Records shall be collected, stored, and maintained in accordance with the requirements of ANSI N45.2.9-1974, Regulatory Guide 1.88 Revision 2 and supplemented by NIRMA 2011 Technical Guidelines, as addressed in Appendix A, subsection A1.88. These records are available so that any desired item of information is retrievable for reference.

The detailed Quality Assurance Program set forth in Chapter 17, and developed by Edison and its contractors, satisfies the requirements of Criterion 1.

For further discussion, see the following:

a. Chapter 3 - Design of Structures, Components, Equipment, and Systems
b. Section 4.2 - Fuel System Design
c. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary
d. Section 5.4 - Reactor Pressure Vessel and Appurtenances
e. Section 5.5 - Component and Subsystem Design
f. Section 6.2 - Containment Systems
g. Section 6.3 - Emergency Core Cooling System
h. Section 7.2 - Reactor Protection System
i. Section 7.3 - Engineered Safety Feature Systems
j. Section 7.6 - Other Systems Required for Safety and Power Generation
k. Chapter 8 - Electrical Power Systems
l. Chapter 12 - Radiation Protection.

3.1.2.1.2 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, tornados, 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.

Criterion 2 Conformance - The design bases enumerated in this criterion are incorporated into the design of structures, systems, and components of Fermi 2. Among the natural phenomena considered are wind and tornado loadings, including static and dynamic water level loadings caused by floods, hurricanes, and other severe storms with wave runup effects; and seismic loadings. In each case the most severe of these phenomena is used as the design basis, together with appropriate combinations of normal and accident conditions. These design bases are developed from detailed analysis of the occurrence and history of these phenomena in the area surrounding the plant location. The method of incorporating these effects is discussed later in Chapter 3. The natural phenomena of the area are discussed in Chapter 2.

A detailed discussion can be found in the following:

a. Section 2.3 - Meteorology
b. Section 2.4 - Hydrological Engineering
c. Section 2.5 - Geology and Seismology
d. Section 3.2 - Classification of Structures, Components, and Systems
e. Section 3.3 - Wind and Tornado Loadings
f. Section 3.4 - Water Level (Flood) Design
g. Section 3.5 - Missile Protection
h. Section 3.7 - Seismic Design
i. Section 3.8 - Design of Category I Structures
j. Section 3.9 - Mechanical Systems and Components
k. Section 3.10 - Seismic Design of Category I Instrumentation and Electrical Equipment.

3.1.2.1.3 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 main 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. Fire-fighting 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.

Criterion 3 Conformance - The design of Fermi 2 is in full compliance with this criterion. The use of noncombustible and heat-resistant materials is maximized. Fire protection and detection measures of appropriate capacities are incorporated in the design, with particular emphasis given to areas containing safety systems, such as the Control Center, and components of engineered safety feature (ESF) systems. The fire protection system (FPS) does not, by rupture or inadvertent operation, prevent the safe shutdown of the plant. The FPS, described in Subsection 9.5.1, provides an adequate supply of water and/or chemicals to fire-fighting stations throughout the plant. The FPS meets the requirements of the applicable laws, codes, and requirements of the State of Michigan, and adheres to the NFPA standards and NEPIA recommendations (Subsection 9.5.1).

A diesel-driven fire pump and a motor-driven fire pump are each independently capable of satisfying plant fire-fighting water requirements. Standby carbon dioxide and Halon systems are provided for fire protection in the diesel generator area and electrical areas in the auxiliary building. The main and auxiliary transformers are protected with deluge fire-fighting equipment. In addition, portable fire extinguishers, hose reels, and hydrants are strategically located throughout the plant area.

Hydrogen, lubrication, and fuel-oil storage facilities are located, designed, and protected to minimize both the probability and effects of fire and explosion. The FPS is discussed in detail in Subsection 9.5.1.

Further discussion of fire protection can be found in the following:

a. Section 6.4 - Habitability Systems

b. Section 8.3 - Onsite Power Systems

c. Section 9.5 - Other Auxiliary Systems

d. Section 14.1 - Test Program.

3.1.2.1.4 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.

Criterion 4 Conformance - Safety-related systems, components, and structures have been designed to accommodate all normal or routine environmental conditions, and conditions associated with postulated accidents including a LOCA. Safety-related systems and components are designed to function properly in the most severe environmental conditions in which their functions are required.

Analyses are performed to determine the effects of missiles, pipe whip, and the jet force of fluid discharge, both inside and outside the primary containment. Where required, restraints, missile shields, additional separation, or additional structural strength are incorporated into the design to ensure proper functioning of safety-related plant features.

Further discussion of environmental and missile design bases can be found in Sections 3.3 through 3.12, and particularly in the following sections.

a. Section 3.5 - Missile Protection

b. Section 3.6 - Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

c. Section 3.11 - Environmental Design of Mechanical and Electrical Equipment

d. Section 10.2 - Turbine Generator.

3.1.2.1.5 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.

Criterion 5 Conformance - There are no safety-related systems or components that are shared with another unit.

3.1.2.2 Group II, Protection by Multiple Fission Product Barriers (Criteria 10-19)

3.1.2.2.1 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.

Criterion 10 Conformance - The reactor core components consist of fuel assemblies, control rods, in-core ion chambers, and related items. The mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The fuel is designed to provide high integrity over a complete range of power levels including transient conditions. The core is sized with sufficient heat-transfer area and coolant flow to ensure that there is no fuel damage under normal conditions or anticipated operational occurrences.

The reactor protection system (RPS) is designed to monitor certain reactor parameters, sense abnormalities, and scram the reactor, thereby preventing fuel damage when trip points are exceeded. Scram trip setpoints are selected on the basis of operating experience and safety design. There is no case in which the scram trip setpoints allow the core to exceed the thermal-hydraulic safety limits. Power for the RPS is supplied by an independent high-inertia ac motor-generator set. Alternative electrical power is available to the RPS buses.

An analysis and evaluation has been made of the effects on core fuel following adverse plant operating conditions. The results of abnormal operational transients are presented in Chapter 15.

The reactor core and associated coolant, control, and protection systems are designed to ensure that the specified fuel design limits are not exceeded during conditions of normal or abnormal operation. Therefore, they meet the requirements of Criterion 10.

For further discussion, see the following:

a. Section 4.2 - Fuel System Design
b. Section 4.3 - Nuclear Design
c. Section 4.4 - Thermal and Hydraulic Design
d. Section 5.5 - Component and Subsystem Design
e. Section 6.3 - Emergency Core Cooling System
f. Section 7.2 - Reactor Protection System
g. Chapter 15 - Accident Analyses.

3.1.2.2.2 Criterion 11 - Reactor Inherent Protection

The reactor core and associated plant 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.

Criterion 11 Conformance - The reactor core is designed to have a reactivity response that regulates or damps changes, both in power level and in spatial distributions of power production, to a level consistent with safe and efficient operation.

The inherent dynamic behavior of the core is characterized in terms of (a) fuel temperature or Doppler coefficient, (b) moderator void coefficient, and (c) moderator temperature coefficient. The combined effects of these coefficients in the power range are termed the power coefficient. Doppler reactivity feedback occurs simultaneously with a change in fuel temperature and opposes the power change that caused it. Moreover, it contributes to system stability. Because the Doppler reactivity opposes load changes, it is desirable to maintain a large ratio between moderator void coefficient and Doppler coefficient for optimum load-following capability. The BWR has an inherently large moderator-to-Doppler coefficients ratio, which permits use of coolant flow rate for load following.

In a BWR, the moderator void coefficient is of importance while the BWR is operating at power. Nuclear design requires the void coefficient inside the fuel channel to be negative. The negative void reactivity coefficients provide an inherent negative feedback during power transients.

The reactor is designed so that the moderator temperature coefficient is small and positive in the cold condition; however, the overall power reactivity coefficient is negative.

The reactor core and associated coolant system are designed so that in the power operating range, prompt inherent dynamic behavior tends to compensate for any rapid increase in reactivity in accordance with Criterion 11.

For further discussion, see the following:

a. Section 4.3 - Nuclear Design
b. Section 4.4 - Thermal and Hydraulic Design.

3.1.2.2.3 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.

Criterion l2 Conformance - The reactor core is designed to ensure that no power oscillation will cause the fuel design limits to be exceeded. The power reactivity coefficient is the composite simultaneous effect of the fuel temperature or Doppler coefficient, moderator void coefficient, and moderator temperature coefficient, on the change in power level. It is negative and well within the range required for adequate damping of power and spatial xenon disturbances. Operating experience has shown that large BWRs are inherently stable against xenon-induced power instability. The large negative operating coefficient provides

a. Good load following with well-damped behavior and little undershoot or overshoot in the heat-transfer response

b. Load following with recirculation flow control

c. Strong damping of spatial power disturbances.

The RPS design provides protection from excessive fuel cladding temperatures and protects the nuclear system process barrier from excessive pressures that threaten the integrity of the system. Local abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through an automatic scram. High integrity of the protection system is achieved through the combination of logic arrangement, trip channel redundancy, power supply redundancy, and physical separation.

The reactor core and associated coolant, control, and protection systems are designed to suppress any power oscillations that could result in exceeding fuel design limits. These systems ensure that Criterion 12 is met.

For further discussion see the following:

a. Section 4.2 - Fuel System Design

b. Section 4.3 - Nuclear Design

c. Section 4.4 - Thermal and Hydraulic Design

d. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

e. Section 7.2 - Reactor Protection System

f. Section 7.7 - Control Systems Not Required for Safety

g. Chapter 15 - Accident Analyses.

3.1.2.2.4 Criterion 13 - Instrumentation and Control

Instrumentation 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.

Criterion 13 Conformance - The fission process is monitored and controlled for all conditions from source range through power operating range. The neutron monitoring system (NMS) detects core conditions that threaten the overall integrity of the fuel barrier caused by excess power generation and provides a signal to the RPS. Fission counters, located in the core, are used for the source range through power operating range. The detectors are located to provide maximum sensitivity to control rod movement during startup, and to provide optimum monitoring in the intermediate and power ranges.

The source range monitor (SRM) subsystem provides neutron flux information during reactor startup and low-flux-level operations. Detectors are inserted into the core for a reactor startup and withdrawn after neutron flux is indicated on the intermediate range monitor (IRM) subsystem. The SRM can provide detection of less than a 20-sec period under the worst possible startup conditions, and is capable of generating a trip signal to block rod withdrawal.

The IRM monitors neutron flux from the upper portion of the SRM to the lower portion of the power range monitor (PRM) subsystems. The IRM is capable of either generating a trip signal to block rod withdrawal or scram the reactor.

The local power range monitor (LPRM) subsystem consists of fission chambers located throughout the core, signal conditioning equipment, and trip functions. The LPRM signals are also used in the average power range monitor (APRM) subsystem, rod block monitor (RBM) subsystem, and Integrated Plant Computer System (IPCS). The RBM is designed to prevent local fuel damage as a result of a single rod withdrawal error under a condition of allowed RBM bypass.

The traversing in-core probe (TIP) subsystem provides a signal proportional to the axial neutron flux distribution of the core. This system provides a means of accurately calibrating the LPRM signal by correlation with the TIP signal.

The reactor protection system (RPS) protects the fuel barriers and the nuclear process barrier by monitoring plant parameters and causing a reactor scram when predetermined setpoints are exceeded.

The reactor manual control system (RMCS) consists of the electrical circuitry, switches, indicators, and alarm devices required for the manipulation of the control rods and surveillance equipment. Separation between the scram function and the normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry.

Reactor pressure vessel (RPV) instrumentation monitors the RPV temperatures, water levels, water flow rate, internal pressure, and water leakage detection from the top head flange. This information is used to assess conditions existing inside the RPV and to assess the physical condition of the RPV. Reactor pressure vessel temperatures are recorded on a multipoint recorder in the control center. Controlled heating and cooling rates allow thermal stress to be appropriately limited. Reactor pressure and vessel water level are also indicated in the control center, in addition to recirculation loop flow, core flow, and the differential pressure between the RPV annulus outside of the core and the core inlet plenum.

To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and nuclear system process barrier, the containment and reactor vessel isolation control system initiates automatic isolation of appropriate pipelines whenever monitored variables exceed pre-selected operational limits.

Nuclear system leakage limits are established so that appropriate action can be taken to ensure the integrity of the nuclear system process barrier. Nuclear system leakage rates classified as identified leakage rates flow to the equipment drain, and those classified as unidentified leakage rates flow to the floor drain sumps. The permissible total leakage rate limit to these sumps is based on NRC requirements. Leakage detection is in accordance with Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems. Flow integrators and recorders are used to determine the leakage by monitoring flow pumped from the drain sumps. The unidentified leakage rate as discussed in Subsection 5.2.7.4 is limited to a value that is less than the value that has been conservatively calculated to be a minimum leakage from a crack large enough to propagate rapidly. The limited leakage volume still allows time for identification and corrective action before integrity of the process barrier is threatened.

The sequence-of-events recorders receive inputs from plant variables, including the primary variables of the RPS. The inputs are scanned and monitored for change of state. The IPCS provides a quick and accurate determination of the core thermal performance. Data reduction, accounting, and logging functions of the IPCS further supplement procedural requirements for control rod manipulation during reactor startup and shutdown.

As previously indicated, adequate instrumentation is provided to monitor system variables in the reactor core, reactor coolant pressure boundary (RCPB), and reactor containment. Appropriate controls are provided to maintain the variables within the operating range and to initiate the necessary corrective action in the event of abnormal operational occurrence or accident. This arrangement of instrumentation and controls meets the requirements of Criterion 13.

For further discussion, see the following:

a. Section 4.2 - Fuel System Design

b. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

c. Section 5.5 - Component and Subsystem Design

d. Section 6.2 - Containment Systems

e. Section 7.2 - Reactor Protection System

f. Section 7.3 - Engineered Safety Feature Systems

g. Section 7.6 - Other Systems Required for Safety and Power Generation

h. Section 7.7 - Control Systems Not Required for Safety.

3.1.2.2.5 Criterion 14 - Reactor Coolant Pressure Boundary

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

Criterion 14 Conformance - The piping and equipment pressure parts, which extend through the outer isolation valve(s) but which are within the RCPB, are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. Systems and components within the RCPB are classified in Section 3.2 as Code Group A. The design requirements, codes, and standards applied to this Code Group ensure a quality product in keeping with the safety functions to be performed.

To minimize the possibility of brittle fracture within the RCPB, the fracture or notch properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness when the system is pressurized to more than 20 percent of the design pressure. Subsection 5.2.4 describes the methods used to control toughness properties. Materials are to be impact tested in accordance with ASME Boiler and Pressure Vessel (B&PV) Code Section III, 1971. The fracture toughness temperature requirements of the RCPB materials also apply for the RCPB piping which penetrates the containment, up to and including the outermost isolation valve.

Piping and equipment pressure parts of the RCPB are assembled and erected by welding unless applicable codes permit flanged or screwed joints. The welding procedures used are designed to produce welds of complete fusion and free of unacceptable defects. All welding procedures, welders, and welding machine operators are qualified in accordance with the requirements of Section IX of the ASME B&PV Code for the materials to be welded.

Qualification records, including the results of procedure and performance qualification tests and identification symbols assigned to each welder, are maintained.

Subsection 5.2.3 contains the detailed material and examination requirements for the piping and equipment of the RCPB prior to and after its assembly and erection. Leakage testing and surveillance are accomplished as described in the evaluation against Criterion 30.

The design, fabrication, erection, and testing of the RCPB ensures an extremely low probability of failure or abnormal leakage, thus satisfying the requirements of Criterion 14.

For further discussion, see the following:

a. Chapter 3 - Design of Structures, Components, and Systems
b. Section 5.4 - Reactor Pressure Vessel and Appurtenances
c. Section 5.5 - Component and Subsystem Design
d. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary
e. Section 15.0 - General (Accident Analyses)
f. Chapter 17 - Quality Assurance.

3.1.2.2.6 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.

<u>Criterion 15 Conformance</u> - The reactor coolant system consists of the RPV and appurtenances, the reactor circulation system, the nuclear system pressure relief system, the main steam lines, the reactor core isolation cooling (RCIC) system, and the residual heat removal (RHR) system. These systems are designed, fabricated, erected, and tested to meet stringent quality requirements and appropriate codes and standards that ensure high integrity of the RCPB throughout the plant lifetime. The reactor coolant system is designed and fabricated to meet the requirements of the ASME B&PV Code Section III, as required by 10 CFR 50.55a, including special waiver provisions.

The auxiliary, control, and protection systems associated with the reactor coolant system act to provide sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. As described in the evaluation of Criterion 13, instrumentation is provided to monitor essential variables to ensure that they are within prescribed operating limits. If the monitored variables exceed their predetermined settings, the auxiliary control and protection systems automatically respond to maintain the variables and systems within allowable design limits.

An example of the integrated protective action scheme, which provides sufficient margin to ensure that the design conditions of the RCPB are not exceeded, is the automatic initiation of the nuclear system pressure relief system on receipt of an overpressure signal. To accomplish overpressure protection, a number of pressure-operated relief valves are provided that can discharge steam from the nuclear system to the suppression pool. The nuclear system pressure relief system also provides automatic depressurization of the nuclear system in the event of a LOCA in which the RPV is not depressurized by the accident. The depressurization of the nuclear system in this situation allows operation of the low-pressure emergency core cooling systems (ECCS) to supply enough cooling water to adequately cool the core. In a similar manner, other auxiliary, control, and protection systems provide assurance that the design conditions of the RCPB are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

The application of appropriate codes, standards, and high quality requirements to the reactor coolant system and the design features of its associated auxiliary, control, and protection systems, ensures that the requirements of Criterion 15 are satisfied.

For further discussion, see the following:

a. Chapter 3 - Design of Structures, Components, Equipment, and Systems

b. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

c. Section 5.4 - Reactor Pressure Vessel and Appurtenances

d. Section 5.5 - Component and Subsystem Design

e. Section 6.3 - Emergency Core Cooling System

f. Section 7.6 - Other Systems Required for Safety and Power Generation

g. Section 15.0 - General (Accident Analyses).

3.1.2.2.7 Criterion 16 - Containment Design

Reactor containment and associated systems shall be provided to establish an essentially leaktight 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.

Criterion 16 Conformance - The primary containment, which includes the drywell and suppression pool, has been designed, fabricated, and erected so as to accommodate, without failure, the pressures and temperatures resulting from the double-ended rupture (or equivalent failure) of any coolant pipe within the primary containment. The primary containment encloses the reactor coolant system and associated instrumentation and controls. During accident conditions, valves which isolate systems that penetrate the primary containment become part of the containment barrier.

The secondary containment, a building that contains the primary containment as well as portions of the reactor process systems and refueling facilities, is maintained at a negative pressure under accident conditions to ensure against leakage. The interior atmosphere is processed to control emissions to the environs so that offsite dose levels are maintained well below the requirements of 10 CFR 100 or 10 CFR 50.67.

Periodic testing and inspection verify the integrity of the reactor containment. Further information on the reactor containment and associated systems can be found in the following:

a. Section 3.8 - Design of Category I Structures

b. Section 6.2 - Containment Systems

c. Section 14.1 - Test Program.

3.1.2.2.8 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) specific acceptable fuel design limits and design conditions of the RCPB 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 supplies, 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 accidents 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 supplies 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 electrical power supplies.

Criterion 17 Conformance - The Fermi 2 onsite power system has four separate emergency diesel generators (EDGs), each of which supplies a separate bus. There are two independent and redundant divisions of ESF, each of which can be powered by a division pair of the EDGs through their associated buses. The diesel generators are of sufficient capacity to provide minimum essential emergency loads, including a single failure, such as the loss of a diesel generator or essential bus. The diesel generators are located in a Category I structure with fire-barrier separation between diesel generators.

Also provided are separate battery power sources to supply power to the separate and redundant ESF dc loads and controls. The battery system consists of two redundant 260/130-V and 24/48-V supplies and chargers. The chargers can be supplied from offsite power or the EDGs, in emergency situations.

The offsite power sources consist of 120-kV and 345-kV independent systems with associated buses and transformers. These supply power to the 4160-V buses. The redundancy of buses within the plant and the division of critical loads between buses yield a system of high reliability and integrity.

The EDGs and batteries have been designed to allow periodic testing and inspection without interruption of normal plant operation. Fault detection and isolation provisions prevent the propagation of faults to alternative systems.

With the above electric system design, Criterion 17 is believed to be satisfied.

Further information on the electric power systems can be found in the following:

a. Section 3.10 - Seismic Design of Category I Instrumentation and Electrical Equipment

b. Chapter 8 - Electric Power.

3.1.2.2.9 Criterion 18 - Inspection and Testing of Electrical Power Systems

Electrical 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 condition of their components. The system 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.

Criterion 18 Conformance - Provisions are made in the design of the offsite and onsite power systems for the inspection and testing of appropriate areas of the system. The EDG system can be tested without interruption of normal operations. The battery system is also designed for periodic testing. The offsite power systems are normally operating; therefore, the status of both the offsite systems and the onsite systems is indicated in the main control room. All systems are designed for periodic inspection. Further information can be found in the following:

a. Chapter 8 - Electric Power

b. Section 14.1 - Test Program.

3.1.2.2.10 Criterion 19 - Main 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.

Holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

Criterion 19 Conformance - The design of the main control room allows continuous occupancy by operating personnel under all operating and accident conditions, including LOCA. All control stations, switches, controllers, and indicators necessary to safely operate and shut down the plant are located in the control center.

Shielding is provided to limit the exposure of control center personnel to a level significantly less than the 5 rem whole-body limit. The control center air conditioning system (CCACS) provides air filtration, recirculation, temperature, and/or humidity control, and has sufficient redundancy to ensure the availability of the system. Recirculation of main control room air is initiated upon a high radiation alarm, with makeup outside air provided to pressurize the control room and selected from the intake with the lower radiation level. Air-operated isolation and recirculation valves can be manually operated. Entrance and exit from the plant (and main control room) in emergency situations are controlled to limit personnel dose to less than 5 rem for the duration of the accident.

Because of the shielding and ventilation systems provided, evacuation of the main control room is a highly improbable event. If, for some reason, evacuation is required, safe shutdown of the reactor can be accomplished from a remote shutdown station. There are sufficient controls and indications at this station to bring the reactor safely to a hot shutdown condition. There is also the capability to bring the reactor to a cold shutdown condition from outside the main control room.

For use of the alternative source term under 10 CFR 50.67, the exposure limit is 5 rem TEDE. Specific accidents that apply the alternative source term per 10 CFR 50.67, and thus utilize the 5 rem TEDE limit, are identified in Section 1.2.1.2.2.3.

Further discussions concerning this criterion are in the following:

a. Section 6.4 - Habitability Systems

b. Section 7.5 - Safety-Related and Power Generation Display Instrumentation

c. Section 9.4 - Air Conditioning, Heating, Cooling, and Ventilation Systems

d. Section 12.1 - Shielding

e. Chapter 15 - Accident Analyses.

3.1.2.3 Group III, Protection and Reactivity Control Systems (Criteria 20 - 29)

3.1.2.3.1 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.

Criterion 20 Conformance - The RPS is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the nuclear system process barrier. Fuel damage is prevented by initiation of an automatic reactor shutdown if monitored nuclear system variables exceed preestablished limits of anticipated operational occurrences. Scram trip settings are selected and verified to be far enough above or below operating levels so as to provide proper protection but not be subject to spurious scrams. The RPS includes the motor-generator power system, sensors, relays, bypass circuitry, and switches that signal the control rod system to scram. The scrams initiated by nuclear system variables, nuclear system high pressure, turbine stop valve closure, turbine control valve fast closure, and RPV low water level will prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a scram in time to prevent the core from exceeding thermal-hydraulic safety limits during abnormal operational transients. Response by the RPS is prompt and the total scram time is short.

A fully withdrawn control rod (withdrawn to 144 in.) will traverse 90 percent of its full stroke in less than 3.5 sec, which is sufficient to ensure that acceptable fuel design limits are not exceeded.

In addition to the RPS, which provides automatic shutdown of the reactor to prevent fuel damage, protection systems are provided to sense accident conditions and initiate automatically the operation of other systems and components important to safety. Systems such as the ECCS are initiated automatically to limit the extent of fuel damage following a LOCA. Other systems automatically isolate the RPV or the containment to prevent the release of significant amounts of radioactive materials from the fuel and the nuclear system process barrier. The controls and instrumentation for the ECCS and the isolation systems are initiated automatically when monitored variables exceed preselected operational limits.

The design of the protection system satisfies the functional requirements as specified in Criterion 20.

For further discussion, see the following:

a. Section 4.2 - Fuel System Design

b. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

c. Section 5.5 - Component and Subsystem Design

d. Section 6.3 - Emergency Core Cooling System

e. Section 7.2 - Reactor Protection System

f. Section 7.3 - Engineered Safety Feature Systems

g. Section 7.6 - Other Systems Required for Safety and Power Generation

h. Chapter 15 - Accident Analyses.

3.1.2.3.2 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.

Criterion 21 Conformance - The RPS design provides assurance that, through redundancy, each channel has sufficient reliability to fulfill the single-failure criterion. No single component failure, intentional bypass, maintenance operation, calibration operation, or test to verify operational availability will impair the ability of the system to perform its intended safety function. Additionally, the system design ensures that when a scram trip point is exceeded there is a high scram probability. However, should a scram not occur, other monitored components will scram the reactor if their trip points are exceeded. There is sufficient electrical and physical separation between channels and between trip logics monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability of the system to respond correctly.

The RPS includes design features that permit inservice testing. This ensures the functional reliability of the system should the reactor variable exceed the corrective action setpoint.

The RPS initiates an automatic reactor shutdown if the monitored plant variables exceed preestablished limits. This system is arranged into two independently powered trip systems. Each trip system has three trip logics, two of which produce an automatic trip signal. The logic scheme is a one-out-of-two twice arrangement. The RPS can be tested during reactor operation. Manual scram testing is performed by operating one of the two manual scram controls. This tests one trip system. The total test verifies the ability to deenergize the scram pilot valve solenoids. Indicating lights verify that the actuator contacts have opened. This capability for a thorough testing program significantly increases reliability.

Control rod drive (CRD) operability can be tested during normal reactor operation. Drive position indicators and in-core neutron detectors are used to verify control rod movement. Each control rod can be withdrawn one notch and then reinserted to the original position without significantly perturbing the nuclear system. One control rod is tested at a time. Control rod mechanism overdrive demonstrates rod-to-drive coupling integrity. Hydraulic supply subsystem pressures can be observed on control center instrumentation. Moreover, the hydraulic control unit scram accumulator and the scram discharge volume level are continuously monitored.

The main steam line isolation valves (MSIVs) may be tested during full reactor operation. They can be closed to 90 percent of full-open position without affecting reactor operation. If reactor power is reduced sufficiently, the isolation valves may be fully closed. Means are provided to evaluate valve stem leakage during reactor shutdown. During refueling operation, valve leakage rates can be determined.

Testing of the RHR system can be performed during normal operation. Main system pumps can be evaluated by taking suction from the suppression pool and discharging through test lines back to the suppression pool. System design and operating procedures also permit the testing of discharge valves into the reactor recirculation loops and into the containment spray headers. The low-pressure coolant injection (LPCI) mode can be tested after reactor shutdown. Each active component of the ECCS provided to operate in a design-basis accident (DBA) is designed to be operable for test purposes during normal operation of the nuclear system, except where such tests directly affect reactor operation.

The high functional reliability, redundancy, and inservice testability of the protection system satisfies the requirements specified in Criterion 21.

For further discussion, see the following:

a. Section 4.2 - Fuel System Design

b. Section 5.5 - Component and Subsystems Design

c. Section 6.2 - Containment Systems

d. Section 6.3 - Emergency Core Cooling System

e. Section 7.2 - Reactor Protection System

f. Section 7.3 - Engineered Safety Feature System

g. Section 14.1 - Test Program

h. Chapter 15 - Accident Analyses.

3.1.2.3.3 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.

Criterion 22 Conformance - The components of protection systems are designed so that the mechanical and thermal environment resulting from any emergency situation in which the components are required to function will not interfere with the operation of that function. Wiring for the RPS outside of the main control room enclosures is run in rigid metallic wireways. The RPS wireways in certain instances are shared by wiring from the corresponding containment and RPV isolation control system channel, but the wires from duplicate sensors on a common process tap are run in separate wireways. The system sensors are electrically and physically separated. Only one trip actuator logic circuit from each trip may be run in the same wireway.

The RPS is designed to permit maintenance and diagnostic work while the reactor is operating. This is accomplished without restricting the plant operation or hindering the output of these safety functions. The flexibility in design afforded the protection system allows operational system testing by the use of an independent trip channel for each trip logic input. When an essential monitored variable exceeds its scram trip point, it is sensed by at least two independent sensors in each trip system. An intentional bypass, maintenance operation, calibration operation, or test will result in a single channel trip. This leaves at least two trip channels per monitored variable capable of initiating a scram. Although each trip system contains two trip channels, only one channel in each trip system must trip to initiate a scram. Thus, the arrangement of two trip channels per trip system ensures that scram will occur as a monitored variable exceeds its scram setting.

The protection system meets the design requirements for functional and physical independence as specified in Criterion 22.

For further discussion, see the following:

a. Section 4.2 - Fuel System Design

b. Section 5.5 - Component and Subsystem Design

c. Section 7.2 - Reactor Protection System

d. Section 7.3 - Engineered Safety Feature Systems

e. Chapter 15 - Accident Analyses.

3.1.2.3.4 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.

Criterion 23 Conformance - The RPS is designed to fail into a safe state. Use of an independent trip channel for each trip logic allows the system to sustain any trip channel failure without preventing the other sensors monitoring the same variable from initiating a scram. A single sensor or trip channel failure will cause a channel trip. Only one trip channel in each trip system must be actuated to initiate a scram. Intentional bypass, maintenance operation, calibration operation, or test will result in a single channel trip. A failure of any one RPS input or subsystem component will produce a trip in one of two channels. This condition is insufficient to produce a reactor scram, but the system is ready to perform its protective function upon another trip.

The environmental conditions in which the instrumentation and equipment of the RPS must operate were considered in establishing the component specifications. Instrumentation specifications are based on the worst expected ambient conditions in which the instruments must operate.

The failure modes of the protection system are such that it will fail into a safe state as required by Criterion 23.

For further discussion, see the following:

a. Section 3.11 - Environmental Design of Mechanical and Electrical Equipment

b. Section 7.2 - Reactor Protection System

c. Section 7.3 - Engineered Safety Feature Systems

d. Chapter 8 - Electric Power.

3.1.2.3.5 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.

Criterion 24 Conformance - There is separation between the RPS and the process control systems. Sensors, trip channels, and trip logics of the RPS are not used directly for automatic control of process systems. Therefore, failure in the controls and instrumentation of process systems cannot induce failure of any portion of the protection system. High scram reliability is designed into the RPS and hydraulic control unit for the CRD. The scram signal and the mode of operation overrides all other signals.

The containment and RPV isolation control systems are designed so that any one failure, maintenance operation, calibration operation, or test to verify operational availability will not impair the functional ability of the isolation control system to respond to essential variables. Corresponding isolation control system channels and RPS channels are not separated from each other, since common power supplies, relay cabinets, primary sensors, and wireways are used for both systems. However, because of the fail-safe design and the one-out-of-two taken twice logic, no single failure in either system can cause failure to scram or failure to isolate.

The protection system is separated from control systems as required in Criterion 24.

For further discussion, see the following:

a. Section 3.12 - Separation Criteria for Safety-Related Mechanical and Electrical Equipment

b. Section 4.2 - Fuel System Design

c. Section 7.2 - Reactor Protection System

d. Section 7.3 - Engineered Safety Feature Systems.

3.1.2.3.6 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.

Criterion 25 Conformance - The RPS provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the nuclear system process barrier. Any monitored variable that exceeds the scram setpoint will initiate an automatic scram and not impair the remaining variables from being monitored. If one channel fails, the remaining portions of the RPS shall function.

The RMCS is designed so that no single failure can negate the effectiveness of a reactor scram. The circuitry for the RMCS is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. Because each control rod is controlled as an individual unit, a failure that results in energizing any of the insert or withdrawal solenoid valves can affect only one control rod. The effectiveness of a reactor scram is not impaired by the malfunctioning of any one control rod.

The most serious rod withdrawal errors occur when the reactor is operating in the power region and the operator withdraws the maximum worth rod. Fuel damage in this event is prevented by the timely action of the rod block monitor, which acts to stop rod movement before safety limits are reached.

The design of the protection system ensures that specified acceptable fuel limits are not exceeded for any single malfunction of the reactivity control systems as specified in Criterion 25.

For further discussion, see the following:

a. Section 4.2 - Fuel System Design

b. Section 4.3 - Nuclear Design

c. Section 4.4 - Thermal and Hydraulic Design

d. Section 7.2 - Reactor Protection System

e. Section 7.6 - Other Systems Required for Safety and Power Generation

f. Section 7.7 - Control Systems Not Required for Safety

g. Chapter 15 - Accident Analyses.

3.1.2.3.7 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 all temperature conditions.

Criterion 26 Conformance - Two independent reactivity control systems using different design principles are provided. The normal method of reactivity control uses control rod assemblies that contain boron carbide (B_4C) powder in the Ultra-MD control rods, and a combination of B_4C and hafnium in the Duralife 140, Marathon C, and Ultra-HD control rods. Positive insertion of these control rods is provided by means of the control rod drive hydraulic system. The control rods are capable of reliably controlling reactivity changes during normal operation (e.g., power changes, power shaping, xenon burnout, normal startup, and shutdown) via operator-controlled insertions and withdrawals. The control rods are also capable of maintaining the core within acceptable fuel design limits during anticipated operational occurrences via the automatic scram function. The occurrence of a limited number of stuck rods during a scram will not adversely affect the capability to maintain the core within fuel design limits.

The circuitry for manual insertion or withdrawal of control rods is completely independent of the circuitry for reactor scram. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. Two sources of scram energy (accumulator pressure and reactor vessel pressure) provide needed scram performance over the entire range of reactor pressure (i.e., from operating conditions to cold shutdown). The design of the control rod system includes an appropriate margin for malfunctions such as stuck rods. Control rod withdrawal sequences and patterns are selected prior to operation to achieve optimum core performance and, simultaneously, low individual rod worths. Because of the carefully planned and regulated rod withdrawal sequence, prompt shutdown of the reactor can be achieved with the insertion of a small number of the many independent control rods. In the event that a reactor scram is necessary, the occurrence of a limited number of stuck rods will not hinder the capability of the control rod system to render the core subcritical.

The second independent reactivity control system is provided by the reactor coolant recirculation system. By varying reactor flow, it is possible to affect the type of reactivity changes necessary for planned, normal power changes (including xenon burnout). In the event that reactor flow is suddenly increased to its maximum value (pump runout), the core will not exceed fuel design limits because the power flow map defines the allowable initial operating states so that the pump runout will not violate these limits.

The control rod system is capable of holding the reactor core subcritical under all temperature conditions, even when the control rod of highest worth is assumed to be stuck in the fully withdrawn position. This shutdown capability of the control rod system is made possible by designing the fuel with burnable poison (Gd_2O_3) to control the high reactivity of fresh fuel. In addition, the standby liquid control system (SLCS) is available to add soluble boron to the core and render it subcritical.

The redundancy and capabilities of the reactivity control systems for Fermi 2 satisfy the requirements of Criterion 26.

For further discussion, see the following:

a. Section 4.2 - Fuel System Design

b. Section 7.4 - Systems Required for Safe Shutdown

c. Section 7.6 - Other Systems Required for Safety and Power Generation

d. Section 7.7 - Control Systems Not Required for Safety.

3.1.2.3.8 Criterion 27 - Combined Reactivity Control Systems Capability

The reactivity control system 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.

Criterion 27 Conformance - There is no credible event applicable to the BWR which requires combined capability of the control rod system and poison additions by the emergency core cooling network. The primary reactivity control system for the BWR during postulated accident conditions is the control rod system. Abnormalities are sensed and, if protection system limits are reached, corrective action is initiated through an automatic scram. High integrity of the protection system is achieved through the combination of logic arrangement, trip channel redundancy, power supply redundancy, and physical separation. High reliability of reactor scram is further achieved by the separation of scram and manual control circuitry, individual control units for each control rod, and fail-safe design features built into the rod drive system. Response by the RPS is prompt, and the total scram time is short.

In reactor operation, there is a spectrum of possible control rod worths, depending on the reactor state and the control rod pattern chosen for operation. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The RWM prevents rod withdrawal other than by the preselected rod withdrawal pattern. These functions assist the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low-power-level operations. As a result of this carefully planned procedure, prompt shutdown of the reactor can be achieved with scram insertion of less than half of the many independent control rods. If accident conditions require a reactor scram, this can be accomplished rapidly with appropriate margin for the unlikely occurrence of malfunctions such as stuck rods.

The reactor core design assists in maintaining the stability of the core under accident conditions as well as during power operation. Reactivity coefficients in the power range that contribute to system stability are (1) fuel temperature or Doppler coefficient, (2) moderator void coefficient, and (3) moderator temperature coefficient. The overall power reactivity coefficient is negative and provides a strong negative reactivity feedback under severe power transient conditions.

The design of the reactivity control systems ensures reliable control of reactivity under postulated accident conditions with appropriate margin for stuck rods. The capability of cooling the core is maintained under all postulated accident conditions. Thus, Criterion 27 is satisfied.

For further discussion, see the following:

a. Section 4.2 - Fuel System Design
b. Section 4.3 - Nuclear Design
c. Section 4.4 - Thermal and Hydraulic Design
d. Section 7.2 - Reactor Protection System
e. Section 7.6 - Other Systems Required for Safety and Power Generation
f. Section 7.7 - Control Systems Not Required for Safety
g. Chapter 15 - Accident Analyses.

3.1.2.3.9 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 rupture changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 28 Conformance - The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The RWM prevents withdrawal other than by the preselected rod withdrawal pattern. These functions assist the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low-power-level operations control rod procedures.

The control rod mechanical design incorporates a hydraulic velocity limiter that prevents rapid rod ejection. This engineered safeguard protects against a high reactivity insertion rate by limiting the control rod velocity to less than 5 fps. Normal rod movement is limited to 6-in. increments, and the rod withdrawal rate is limited to 3 in./sec by the hydraulic valve.

The plant safety analysis (Chapter 15) provides detailed evaluations of the postulated reactivity accidents as well as abnormal operational transients. Analyses are included for rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition. The initial conditions, assumptions, calculational models, sequences of events, and anticipated results of each postulated occurrence are covered. The results of these analyses indicate that none of the postulated reactivity transients or accidents result in damage to the RCPB. In addition, the integrity of the core, its support structures, or other RPV internals is maintained so that the capability of cooling the core is not impaired for any of the postulated reactivity accidents described in the plant safety analysis.

The design features of the reactivity control system, which limit the potential amount and rate of reactivity increase, ensure that Criterion 28 is satisfied for all postulated reactivity accidents.

For further discussion, see the following:

a. Chapter 3 - Design of Structures, Components, Equipment, and Systems

b. Section 4.2 - Fuel System Design

c. Section 4.3 - Nuclear Design

d. Subsection 4.5.3 - Control Rod Drive Housing Supports

e. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

f. Section 5.4 - Reactor Pressure Vessel and Appurtenances

g. Section 5.5 - Component and Subsystem Design

h. Section 7.6 - Other Systems Required for Safety and Power Generation

i. Chapter 15 - Accident Analyses.

3.1.2.3.10 Criterion 29 - Protection Against Anticipated Operational Occurrences

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

Criterion 29 Conformance - The high functional reliability of the protection and reactivity control systems is achieved through the combination of logic arrangement, redundancy, physical and electrical independence, functional separation, fail-safe design, and inservice testability. These design features are discussed in Criteria 21, 22, 23, 24, and 26.

An extremely high probability of correct protection and reactivity control systems response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Active components can be tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions even in the event of a subsequent single failure. Components important to safety such as the CRD, MSIVs, and RHR pumps are tested during normal reactor operation. Functional testing and calibration schedules are developed using experience. These schedules represent optimized protection and reactivity control system reliability by considering both the failure probabilities of individual components and the reliability effects during individual component testing on the portion of the system not under going testing. The capability for inservice testing ensures the high functional reliability of protection and reactivity control systems should a reactor variable exceed the corrective action setpoint.

The capabilities of the protection and reactivity control systems to perform their safety functions in the event of anticipated operational occurrences meet the requirements of Criterion 29.

For further discussion, see the following:

a. Section 4.2 - Fuel System Design

b. Section 5.5 - Component and Subsystem Design

c. Section 6.2 - Containment Systems

d. Section 6.3 - Emergency Core Cooling System

e. Section 7.2 - Reactor Protection System

f. Section 7.3 - Engineered Safety Feature Systems

g. Chapter 15 - Accident Analyses.

3.1.2.4 Group IV, Fluid Systems (Criteria 30-46)

3.1.2.4.1 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.

Criterion 30 Conformance - By using conservative design practices and detailed quality control procedures, the pressure-retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions. Accordingly, components that comprise the RCPB are designed, fabricated, erected, and tested in accordance with the recognized industry codes and standards listed in Sections 5.2, 5.4, and 5.5. Further, product and process quality planning is provided as described in Chapter 17 to ensure conformance with the applicable codes and standards and to retain appropriate documented evidence verifying compliance. Because the subject matter of this criterion deals with aspects of the RCPB, further discussion on this subject is treated in the response to Criterion 14, Reactor Coolant Pressure Boundary.

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and in some cases, isolate the RCPB from potentially hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased condensate flow from the primary containment cooling system, increased frequency of sump pump operation, and measurement of fission product concentration. In addition to these, large leaks are detected by changes in flow rates in process lines and reactor water level. The allowable leakage rates are based on the predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal ac power associated with a loss of feedwater supply, makeup capabilities are provided by the RCIC system. While the leak detection system provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges. The RCPB and the leak detection system are designed to meet the requirements of Criterion 30.

For further discussion, see the following:

a. Chapter 3 - Design of Structures, Components, Equipment and Systems

b. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

c. Section 5.4 - Reactor Pressure Vessel and Appurtenances

d. Section 5.5 - Component and Subsystem Design

e. Section 7.6 - Other Systems Required for Safety and Power Generation

f. Section 15.0 - Accident Analyses

g. Chapter 17 - Quality Assurance.

3.1.2.4.2 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 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 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.

Criterion 31 Conformance - Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against nonductile fracture. To minimize the possibility of brittle fracture failure of the RPV, it is designed to meet the requirements of ASME B&PV Code Section III, 1968 Edition through Summer 1969 addenda, which considers material properties; steady-state and transient stresses; and the size of flaws, and conforms very closely with Appendix G, which was added in the Summer 1972 Addenda (see Section 5.2 for a discussion of the degree of conformance.)

The nil ductility transition (NDT) temperature is defined as the temperature below which ferritic steel fails in a brittle rather than ductile manner. The RT_{NDT} temperature increases as a function of neutron exposure at integrated neutron exposures greater than 1.0 x 10^{17} n/cm^2 with neutrons of energies in excess of 1 MeV. Since the material RT_{NDT} temperature dictates the minimum operating temperature at which the reactor vessel can be pressurized, it is desirable for the NDT temperature to be low.

The reactor assembly design provides an annular space from the outermost fuel assemblies to the inner surface of the RPV that serves to attenuate the fast neutron flux incident upon the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies, and reactor coolant. Assuming plant operation at rated power, and end-of-life (EOL) cumulative Effective Full Power Years (EFPY) of 52 EFPY (See Appendix B, Section B, for discussion of operation beyond the original design plant life), the maximum fast neutron fluence at the inner surface of the RPV is calculated to be 1.03 x 10^{18} n/cm^2 (fast neutron fluence consists of neutrons having energies greater than 1 MeV) as detailed in Table 4.3-2 (See Appendix B for 60 year projected fluence). EOL RT_{NDT} temperature as calculated from the EOL fluence and chemical composition indicates a substantial margin against the occurrence of brittle fracture. For hydrostatic test, the RPV will not be pressurized until the RPV temperature exceeds the RT_{NDT} by at least 60°F

The RCPB piping, pumps, and valves are designed, maintained, and tested such that adequate assurance is provided that the boundary will behave in a nonbrittle manner throughout the life of the plant.

For further discussion, see the following:

a. Chapter 3 - Design of Structures, Components, Equipment, and Systems

b. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

c. Section 5.4 - Reactor Pressure Vessel and Appurtenances.

3.1.2.4.3 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 leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 32 Conformance - The RPV design and engineering effort includes provisions for inservice inspection. Removable plugs in the sacrificial shield and/or removable panels in the insulation provide access for examination of the vessel and its appurtenances. In addition, all of the remaining portion of the RCPB is provided with removable insulation. Inspection of the RCPB is in accordance with the ASME B&PV Code Section XI. The Inservice Inspection Plan, access provisions, and areas of restricted access are defined in Section 5.2.

Reactor pressure vessel material surveillance samples are located within the RPV to enable periodic monitoring of material properties with exposure. The program includes specimens of the base metal, heat-affected zone metal, and weld material. The samples are placed near the core midplane to obtain maximum exposures. Tests include tensile and impact testing. The test program is in accordance with ASTM El85-73 and the appropriate requirements of 10 CFR 50, Appendixes G and H. Subsequent to developing this surveillance program, the BWRVIP developed an integrated surveillance program (ISP) which replaces the Fermi specific surveillance program. This program is described in section 5.2.4.4.3.

The plant testing and inspection programs ensure that the requirements of Criterion 32 will be met.

For further discussion, see the following:

a. Chapter 3 - Design of Structures, Components, Equipment, and Systems

b. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

c. Section 5.4 - Reactor Pressure Vessel and Appurtenances

d. Section 5.5 - Component and Subsystem Design

e. Section 14.1 - Test Program.

3.1.2.4.4 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.

Criterion 33 Conformance - Means provided for detecting reactor coolant leakage are discussed in the conformance to Criterion 30. As stated, the RCIC system provides makeup for small leaks, and the ECCS provides core cooling for the complete range of discharges from ruptured pipes. Protection is provided for the full spectrum of possible discharges to the extent that fuel clad temperature limits are not exceeded utilizing either onsite or offsite redundant power sources.

The plant is designed to provide ample reactor coolant makeup for protection against small leaks in the RCPB for anticipated operational occurrences and postulated accident conditions. The design of these systems meets the requirements of Criterion 33.

For further discussion, see the following:

a. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

b. Section 5.6 - Instrumentation Requirements

c. Section 6.3 - Emergency Core Cooling System

d. Section 7.6 - Other Systems Required for Safety and Power Generation.

3.1.2.4.5 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.

Criterion 34 Conformance - The RHR system provides the means to

a. Remove decay heat and residual heat from the nuclear system so that refueling and nuclear system servicing can be performed

b. Supplement the fuel pool cooling and cleanup system capacity during shutdown to provide additional cooling capacity.

The RHR system is designed for three modes of operation:

a. Shutdown cooling

b. Containment cooling

c. LPCI.

The LPCI mode of operation, part of the ECCS, does not apply to Criterion 34 since its purpose is to reflood the core rather than remove decay heat.

The major equipment of the RHR system consists of two heat exchangers, four main system pumps, and four service water pumps. The equipment is connected by associated valves and piping, and the instrumentation and controls are provided for proper system operation. The main system pumps are sized on the basis of the flow required during the LPCI mode of operation, which is the mode requiring the maximum flow rate. The heat exchangers were sized on the basis of the required duty for the steam condensing function, which is the mode requiring the maximum heat exchanger capacity. However, Edison has decided to delete the steam condensing mode of the RHR system and has disconnected the equipment that would be necessary to use this mode of RHR.

One loop, consisting of a heat exchanger, two main system pumps in parallel, and associated piping, is located in one area of the reactor building. The other heat exchanger, pumps, and piping (forming a second loop) are located in another area of the reactor building to minimize the possibility of a single physical event causing the loss of the entire system. The two loops of the RHR system are cross connected by a single header, making it possible to supply either loop from the pumps in the other loop. Either of these redundant loops can fully meet the most limiting of the three modes of operation.

The division and redundancy in the RHR system apply to the electric power system also. As discussed in Section 8.3, the electric power system is divided into two separate, redundant divisions, each of which is independently capable of supplying power to one group of the redundant safety equipment and components required for safe shutdown at the plant. Each division is supplied by electrically and physically separate offsite power sources. Four 2850-kW standby diesel generators, two in each division, supply adequate power to their respective division in the event that offsite power is not available. The diesel generators, buses, and switchgear of Division I are electrically and physically separated such that no single failure could interrupt both divisions of electric power. Also, all of the above onsite emergency ac power equipment is housed in Category I structures that also provide protection against missiles and natural phenomena. The batteries, buses, and other equipment of the dc power systems are likewise divided into two redundant, separate, full-capacity divisions with the same equipment protection as provided for the ac power systems. Thus, the power from onsite and offsite power systems conforms to Criterion 34.

The RHR system is adequate to remove residual heat from the reactor core and ensure that fuel and RCPB design limits are not exceeded. Redundant offsite and onsite electric power systems are provided. The design of the RHR system, including its power supply, meets the requirements of Criterion 34.

For further discussion, see the following:

a. Section 5.5 - Component and Subsystem Design

b. Section 6.3 - Emergency Core Cooling Systems

c. Section 7.3 - Engineered Safety Feature Systems

d. Section 8.3 - Onsite Power Systems

e. Section 9.2 - Water Systems

f. Chapter 15 - Accident Analyses.

3.1.2.4.6 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 (l) 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.

Criterion 35 Conformance - The ECCS consists of the following: (1) high pressure coolant injection system (HPCI), (2) automatic depressurization system (ADS), (3) core spray system, and (4) LPCI (an operating mode of the RHR system). The ECCS is designed to limit fuel cladding temperature over the complete spectrum of possible break sizes in the nuclear system process barrier, including a complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel.

The HPCI system consists of a steam turbine, a constant-flow pump, system piping, valves, controls, and instrumentation. The HPCI system ensures that the reactor core is adequately cooled to prevent excessive fuel clad temperatures for breaks in the nuclear system that do not result in rapid depressurization of the RPV. The HPCI continues to operate until RPV pressure is below the pressure at which LPCI operation or core spray system operation maintains core cooling. Water to supply the HPCI and core spray systems is available from either the condensate storage tank or the suppression pool. The supply of water for LPCI operation is available from the suppression pool only.

In case the capability of the feedwater pumps, CRD pumps, RCIC, and HPCI is not sufficient to maintain the reactor water level, the ADS functions to reduce the reactor pressure so that flow from LPCI and the core spray system enters the RPV in time to cool the core and prevent excessive fuel clad temperature. The ADS uses five of the nuclear system pressure relief valves to relieve the high-pressure steam to the suppression pool.

Two independent loops are provided as a part of the core spray system. Each loop consists of a pair of centrifugal water pumps driven by electric motors, a spray sparger in the RPV above the core, piping and valves to convey water from the suppression pool to the sparger, and the associated instrumentation and controls instrumentation. In cases of low water level in the RPV or high pressure in the drywell, the core spray system automatically sprays water onto the top of the fuel assemblies in time, and at a sufficient flow rate, to cool the core and prevent excessive fuel temperature. The LPCI system starts from the same signals that initiate the core spray and operates independently to achieve the same objective by flooding the RPV.

In cases of low water level in the reactor or high pressure in the containment drywell, the LPCI mode of the RHR system pumps water into the RPV in time to flood the core and prevent excessive fuel temperature. Low-pressure coolant injection operation provides protection to the core in case of a large break in the nuclear system when the feedwater pumps and the HPCI system are unable to maintain RPV water level. Protection provided by LPCI also extends to a small break where the ADS has operated to lower the RPV pressure which would result in the LPCI and the core spray system starting to provide core cooling.

Results of the performance of the ECCS for the entire spectrum of liquid line breaks are discussed in Section 6.3.

3.1.2.4.7 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 the integrity and capability of the system.

Criterion 36 Conformance - The ECCS is designed as discussed in Criterion 35. The engineering and design efforts for the ECCS include inservice inspection considerations. The spray rings within the vessel are accessible for inspection during each refueling outage. Removable plugs in the sacrificial shield and/or panels in the insulation provide access for examination of nozzles. Removable insulation is provided on the ECCS piping out to and including the first isolation valve outside containment. Inspection of the ECCS is in accordance with the intent of Section XI of the ASME Code. The Inservice Inspection Plan, access provisions, and areas of restricted access are defined in Section 5.2.

During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the primary containment can be visually inspected at any time. Components inside the primary containment can be inspected when the drywell is open for access. When the RPV is open, for refueling or other purposes, the spargers and other internals can be inspected. Portions of the ECCS that are part of the RCPB are designed to specifications for inservice inspection to detect defects. Particular attention is given to the reactor nozzles and the core spray and feedwater spargers. The design of the RPV and internals for inservice inspection and the plant testing and inspection program ensure that the requirements of Criterion 36 are met.

For further discussion, see the following:

a. Section 4.2 - Fuel System Design

b. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

c. Section 5.4 - Reactor Pressure Vessel and Appurtenances

d. Section 7.3 - Engineered Safety Feature Systems.

3.1.2.4.8 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.

Criterion 37 Conformance - The ECCS consists of the HPCI system, ADS, LPCI mode of the RHR system, and core spray system. Each of these systems is provided with sufficient test connections and isolation valves to permit appropriate periodic functional testing that ensures the structural and leaktight integrity of its components.

The HPCI, LPCI, and core spray systems are designed to permit periodic testing to ensure the operability and performance of the active components of each system.

The pumps and valves of these systems will be tested periodically to verify operability. Flow-rate tests will be conducted on the core spray, LPCI, and HPCI systems. The ADS logic will be tested on a routine basis. Operability of the safety/relief valves will be tested when they are removed on a periodic schedule for valve testing and overhaul.

The complete ECCS will be subjected to tests in order to verify the performance of the full operational (Section 14.1) sequence that brings each component system into operation. The operation of the associated cooling water systems is discussed in the evaluation of Criterion 46. It is concluded that the requirements of Criterion 37 are met.

For further discussion, see the following:

a. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

b. Section 6.3 - Emergency Core Cooling System

c. Section 7.3 - Engineered Safety Feature Systems

d. Chapter 8 - Electric Power

e. Section 14.1 - Test Program.

3.1.2.4.9 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 electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 38 Conformance - In the event of a LOCA within the reactor containment, the pressure suppression system will rapidly condense the steam to prevent containment overpressure. The containment feature of pressure suppression employs two separate compartmented sections of the primary containment: the drywell that houses the nuclear system and the suppression chamber containing a large volume of water. Any increase in pressure in the drywell from a leak in the nuclear system is relieved below the surface of the suppression chamber water pool by connecting vent lines, thereby condensing steam being released to the drywell. Any pressure buildup in the suppression chamber is equalized with the drywell by a vent line and vacuum breaker arrangement. Cooling systems remove heat from the reactor core, the drywell, and water in the suppression chamber during accident conditions. Thus, continuous cooling of the primary containment is provided.

The ECCS is actuated to provide core cooling in the event of a LOCA. Low water level in the RPV or high pressure in the drywell will initiate the ECCS to prevent excessive fuel temperature. Sufficient water is provided in the suppression pool to accommodate the initial energy that can transiently be released into the drywell from the postulated pipe failure.

The suppression chamber is sized to contain this water, in addition to the water displaced from the reactor primary system, together with the free air initially contained in the drywell.

Either or both RHR system loops, which include the heat exchangers, can be manually activated to remove energy from the containment in the containment cooling mode. The redundancy and capability of the offsite and onsite electric power systems to provide power for the RHR system are presented in the Criterion 34 Conformance Evaluation.

The pressure suppression system is capable of rapid containment pressure and temperature reduction following a LOCA so that design limits are not exceeded. Redundant offsite and onsite electric power systems provide assurances that system safety functions can be accomplished. The design of the containment heat removal system meets the requirements of Criterion 38.

For further discussion, see the following:

a. Section 5.5 - Component and Subsystem Design

b. Section 6.2 - Containment Systems

c. Section 6.3 - Emergency Core Cooling System

d. Section 7.3 - Engineered Safety Feature Systems

e. Chapter 8 - Electric Power

f. Chapter 9 - Auxiliary Systems

g. Chapter 15 - Accident Analyses.

3.1.2.4.10 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.

Criterion 39 Conformance - Provisions are made to facilitate periodic inspection of active components and other important equipment of the containment pressure-reducing systems. During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the primary containment can be visually inspected at any time, and will be inspected periodically. Components inside the primary containment can be inspected when the drywell is open for access. The testing frequencies of most components will be correlated with the component inspection.

The pressure suppression chamber is designed to permit appropriate periodic inspection. Space is provided outside the chamber for inspection and maintenance. There are two hatches that permit access to the suppression chamber for inspection.

The containment heat removal system is designed to permit periodic inspection of major components both outside and inside the primary containment as discussed in Section 14.1. This design meets the requirements of Criterion 39.

For further discussion, see the following:

a. Section 5.5 - Component and Subsystem Design

b. Section 6.2 - Containment Systems

c. Section 6.3 - Emergency Core Cooling System

d. Section 7.3 - Engineered Safety Feature Systems

e. Section 9.2 - Water Systems

f. Section 14.1 - Test Program.

3.1.2.4.11 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 systems, 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.

Criterion 40 Conformance - The containment heat removal function is accomplished by the containment cooling mode of the RHR system. This mode consists of the suppression pool cooling subsystem and containment spray subsystem.

The RHR system is provided with sufficient test connections and isolation valves to permit periodic pressure testing. The containment spray mode is subjected to a periodic air test.

The pumps and valves of the RHR system will be operated periodically to verify operability. The containment spray mode is not fully testable, but the operation of the initiation signal and components can be verified. The suppression pool cooling mode is not automatically initiated, but operation of the components is periodically verified. The operation of associated cooling water systems is discussed in the response to Design Criterion 46. It is concluded that the requirements of Criterion 40 are met.

For further discussion, see the following:

a. Section 5.5 - Component and Subsystem Design

b. Section 7.3 - Engineered Safety Feature Systems

c. Chapter 8 - Electric Power

d. Section 14.1 - Test Program.

3.1.2.4.12 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 quantity 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.

Criterion 41 Conformance - Fission products or other materials that leak into the drywell following postulated accidents are mostly contained in the drywell. Those that leak to the reactor building are processed by the standby gas treatment system (SGTS). The SGTS draws air from the reactor building and discharges it through a high-efficiency particulate air (HEPA) filter and deep-bed charcoal filters to reduce the levels of radiation before exhausting the air to the environment. The SGTS is designed to meet Category I requirements and can be powered from either the onsite or offsite power sources. An on-line continuous gas monitoring system allows operating personnel to evaluate the drywell atmospheric conditions, including hydrogen and oxygen concentration. To counteract the buildup of combustible gases to unacceptable limits, the drywell is rendered inert with nitrogen gas. For further details, see Sections 6.2 and 9.3.6.

3.1.2.4.13 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.

Criterion 42 Conformance - All parts of the systems described for Criterion 41 can be inspected periodically (except small lengths of piping or ducting passing through concrete shielding) for visible indications of damage or potential failure. Access is provided to all active components for inspection and maintenance. Section 6.2, Containment Systems, includes a description of the preoperational and inservice performance inspection programs to ensure the integrity and capability of the containment atmosphere cleanup systems. For further details, see Section 6.2.

3.1.2.4.14 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.

Criterion 43 Conformance - The integrity of the containment atmosphere cleanup systems is verified by preoperational and inservice testing. Testing (including filter dioctyl phthalate penetration testing [DOP] and freon testing) for the SGTS is discussed in Section 6.2. Inspection and testing of the containment are also discussed in Sections 6.2 and 14.1. Testability of the power sources is described in Chapter 8. For further discussion, see the following:

a. Section 6.2 - Containment Systems

b. Chapter 8 - Electric Power

c. Section 14.1 - Test Program.

3.1.2.4.15 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 interconnection, 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.

Criterion 44 Conformance - The RHR service water (RHRSW) system, the emergency equipment service water (EESW) system, and the EDG service water system are designed in accordance with Criterion 44 to transfer heat from structures, systems, and components important to safety, to the ultimate heat sink under normal operating and accident conditions. The systems have suitable redundancy to accommodate a single failure without hindering the safety function of the systems. Appropriate leak-detection capability is provided. The RHRSW system is provided to remove heat from the RHR system during plant shutdown and post-accident conditions. The EDG service water system removes heat rejected by the EDG when operating and the EESW provides cooling water for equipment required to operate during and following an accident, as needed.

Electric power for the operation of each system may be supplied from offsite or onsite power sources, with distribution arranged such that a single failure will not prevent the system from performing its safety function.

For discussion of the above systems, see the following:

a. Subsection 5.5.7 - Residual Heat Removal System

b. Chapter 8 - Electric Power

c. Section 9.2 - Water Systems.

3.1.2.4.16 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.

Criterion 45 Conformance - The systems discussed under Criterion 44 Conformance are designed to permit periodic inspection and/or monitor system integrity. Where physical inspection is not possible (e.g., buried pipes) periodic integrity testing, such as hydrostatic testing, is performed. Periodic inspection requirements are also established. For further details, see Section 9.2, Water Systems.

3.1.2.4.17 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.

Criterion 46 Conformance - The service water systems discussed in the conformance to Criterion 44 are designed to conform to the requirements of Criterion 46. Provisions are made for testing the actuation of the systems from both normal and emergency power sources, and for monitoring the integrity of components. Initial and periodic testing of these systems is described in Section 14.1. For further details, see the following:

a. Section 9.2 - Water Systems

b. Section 14.1 - Test Program.

3.1.2.5 Group V, Reactor Containment (Criteria 50-57)

3.1.2.5.1 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 conservatism of the calculational model and input parameters.

Criterion 50 Conformance - The reactor containment structures, including access openings, penetrations, and the containment heat removal system are designed with sufficient margin to meet the intent of Criterion 50. The design includes consideration of metal/water reactions and other chemical reactions subsequent to the postulated LOCA. The primary reactor containment consists of the drywell, pressure suppression chamber, and interconnecting vent pipes and vent header.

The containment was initially designed for 56 psig at 281°F. Subsequently, the containment has been analyzed for the envelope of conditions representing the spectrum of LOCAs by Chicago Bridge and Iron Company (CBI), the design fabricator, and is considered adequate without exception. Metal temperatures are not expected to reach the maximum temperature of 340°F, except for localized impingement areas. Continued integrity of the primary containment is ensured by initial and periodic testing and inspection.

Further discussion of containment design may be found in the following:

a. Section 3.8 - Design of Category I Structures

b. Section 6.2 - Containment Systems

c. Section 14.1 - Test Program.

3.1.2.5.2 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) ferritic materials behave 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 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.

Criterion 51 Conformance - Operational, test, and postulated accident temperatures are combined with appropriate pressures and other loads in the load-combination equations of Section 3.8. The resulting loads are used in determining the required material properties and construction methods according to the ASME B&PV Code, and AISC requirements, as well as specific material requirements imposed by other codes, standards, and special considerations. All of these codes, standards, special requirements, and analytical techniques used in determining the adequacy of containment material fracture toughness, are given in Section 3.8. Methods of ensuring compliance with these codes are covered by the Quality Assurance Program discussed in Chapter 17.

For further discussion of containment design, refer to the following:

a. Section 3.8 - Design of Category I Structures

b. Section 6.2 - Containment Systems.

3.1.2.5.3 Criterion 52 - Capability for Containment Leakage 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.

Criterion 52 Conformance - Provisions for containment leakage rate testing conform to Criterion 52. Section 6.2 discusses the provisions for containment leakage rate testing which conform to this criterion as well as to 10 CFR 50, Appendix J, Option B. Containment leak rate testing is discussed in the following:

a. Section 3.8 - Design of Category I Structures

b. Section 6.2 - Containment Systems.

3.1.2.5.4 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 leak-tightness of penetrations which have resilient seals and expansion bellows.

Criterion 53 Conformance - The reactor containment design permits access to penetrations and other important areas for implementation of the surveillance program described in the Technical Specifications. Penetrations and resilient seals and bellows are inspected visually, and leaktightness is verified by periodic containment pressure tests. The frequency of inspection will be consistent with the leakage rate for the individual units. Initial leak rate tests of the containment vessel and necessary action were performed to ensure that the actual leak rate was below the design values. Provisions in containment design for the performance of the tests are described in Section 6.2, Containment Systems.

3.1.2.5.5 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.

Criterion 54 Conformance - Piping systems penetrating the containment are designed to withstand a pressure at least equal to the containment maximum internal pressure. All piping systems penetrating the containment are provided with isolation valves.

Proper valve closing time is achieved by appropriate selection of valve, operating type, and operator size. Isolation valve closing time was verified during the functional performance tests prior to reactor startup. The design of piping systems penetrating reactor containment includes provisions for appropriate testing of isolation valves and valve leakage.

Major leaks in the pipe are located by increased temperature, radiation, and/or drain sump flow. Provisions are made to permit leakage testing of the isolation valves. For further discussion, see Section 6.2, Containment Systems.

3.1.2.5.6 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:

a. one locked closed isolation valve inside and one locked closed isolation valve outside containment, or

b. one automatic isolation valve inside and one locked closed isolation valve outside containment, or

c. 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

d. 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.

Criterion 55 Conformance - Conformance to this criterion is discussed on a line-by-line basis in Subsection 6.2.4.2.2.2. It is shown that Fermi 2 conforms to this criterion to the extent that it is consistent with the safety requirements of the various systems. Several lines required to be open for injecting liquids following accidents use testable check valves for isolation (feedwater, SLCS, and ECCS discharge lines).

3.1.2.5.7 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:

a. one locked closed isolation valve inside and one locked closed isolation valve outside containment, or

b. one automatic isolation valve inside and one locked closed isolation valve outside containment, or

c. 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

d. 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.

Criterion 56 Conformance - Conformance to this criterion is discussed in Subsection 6.2.4.2.2.3. This criterion requires one isolation valve inside the containment and one outside. Fermi 2 is based on the design basis that placing isolation valves inside the suppression chamber would reduce the reliability of the connecting systems. Justification for this configuration is included in Subsection 6.2.4.2.2.3.1.

3.1.2.5.8 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 the containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Criterion 57 Conformance - Piping forming a closed loop within the containment is provided with isolation valves in accordance with Criterion 57. Each line that penetrates the primary reactor containment and is neither part of the RCPB nor connected directly to the containment atmosphere has at least one containment isolation valve that is either automatic or locked closed, or capable of remote manual operation. This valve is located outside the containment but as close to the containment as practicable.

Containment isolation valves and the associated tables and figures are discussed in Section 6.2, Containment Systems.

3.1.2.6 Group VI, Fuel and Radioactivity Control (Criteria 60-64)

3.1.2.6.1 Criterion 60 - Control of Releases of 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.

Criterion 60 Conformance - An extensive system, including filtration, evaporation, and demineralization, has been designed for liquid waste treatment. Offgas from the steam-jet air ejector is processed by appropriate holdup in charcoal delay beds. Liquid wastes are normally processed (dewatered, solidified, etc.) and packaged in suitable containers for eventual disposition in licensed burial grounds. Should any condition exist that could prevent safe release of liquid waste, the liquid radwaste system has ample tankage to permit deferring the release. This system is designed to be able to receive anticipated surges in liquid waste volumes. The offgas system is capable of safely processing, for release, considerably more radioactive gas than would be expected during normal plant conditions and anticipated operational occurrences. For additional information, refer to Chapter 11.

Fermi 2 potable water was originally supplied from the onsite Fermi 1 water treatment plant and pumped through the Fermi 2 distribution system. Under this condition, the system was not subject to the requirements of Design Criterion 60. In 1995, the Fermi 2 water supply was connected to the Frenchtown Township Water Treatment Plant (FTWTP) system, and the Fermi 1 plant was abandoned.

The Fermi 2 potable water supplies the makeup demineralizer system and the sanitary, drinking, kitchen, and safety shower systems. The makeup demineralizer system is the only interconnection between the potable water and systems having the potential for containing radioactive material. At this interconnection, the potable water system is protected by an air gap, an NRC accepted design provision to prevent the inadvertent contamination of the FTWTP system with radioactive material.

3.1.2.6.2 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 reflect 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.

Criterion 61 Conformance

New-Fuel Storage - New fuel may be placed in the fuel storage pool or placed in dry storage in the new-fuel storage vault located inside the secondary containment (reactor building). The geometric design of the storage racks precludes accidental criticality (see Criterion 62 Conformance Evaluation). Use of the new fuel storage vault is subject to the restrictions discussed in Section 9.1.1.2.1.

Spent-Fuel Handling and Storage - The handling of new- and spent-fuel assemblies for reactor refueling is within the reactor building. Fuel storage pool water is allowed to flood the reactor well to provide shielding above the reactor and spent fuel. Fuel pool water is circulated through the fuel pool cooling and cleanup (FPCC) system to maintain fuel pool temperature, purity, clarity, and level. Storage racks preclude accidental criticality (see Criterion 62 Conformance Evaluation).

Reliable decay heat removal is provided by the closed-loop FPCC system. It consists of two circulating pumps, two heat exchangers, two filter-demineralizers, two skimmer surge tanks, and the required piping, valves, and instrumentation. The pool water is circulated through the system; suction is taken from surge tanks, flow passes through the heat exchanger and filters, and is discharged through diffusers at the bottom of the fuel pool and reactor well. Pool water temperature is maintained below 125°F when removing the maximum normal heat load from the pool with the reactor building closed cooling water temperature at its maximum. If it appears that the pool temperature will exceed 150°F, the FPCC system can be connected to the RHR system.

This increases the cooling capacity of the FPCC system and ensures that the temperature will not exceed 150°F.

There are no connections to the fuel storage pool that could allow the fuel pool to be drained below the pool gate between the reactor well and fuel pool. The high and low level switches indicate pool-water-level changes in the main control room and pump room. Pool-water-level indication is painted on the pool walls. Fission product concentration in the pool water is minimized by use of the filter-demineralizer. This minimizes the release of fission products from the pool to the reactor building environment.

No testing is planned because at least one pump, heat exchanger, and filter-demineralizer are continuously in operation while fuel is stored in the pool. Duplicate units are operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms are adequate to verify system operability.

<u>Dry Spent Fuel Storage</u> - Storage of spent fuel at the Fermi Independent Spent Fuel Storage Installation (ISFSI) is governed by the regulations in 10 CFR 72 that are applicable to Part 72 general licensees, and the Certificate of Compliance (CoC) for the spent fuel storage cask. Furthermore, in accordance with the provisions of 10 CFR 50.68(c), while a spent fuel transportation package approved under 10 CFR 71 or a spent fuel storage cask approved under 10 CFR 72 is in the spent fuel pool:

1. The requirements of 10 CFR 50.68(b) do not apply to the fuel located within that package or cask, and
2. The requirements of 10 CFR 71 or 10 CFR 72, as applicable, and the requirements of the package or cask CoC apply to the fuel within that package or cask.

<u>Radioactive Waste System</u> - The radioactive waste systems provide all equipment or connections for portable systems necessary to collect, process, and prepare for disposal all radioactive liquid, gaseous, and solid waste produced as a result of reactor operation. Liquid radwastes are classified, contained, and treated as high or low conductivity, chemical, sludges, or concentrated wastes. Processing includes filtration, ion exchange, analysis, and dilution. Liquid wastes are also decanted, and sludge is accumulated for disposal as solid

radwaste. Wet solid wastes are packaged in approved disposal containers. Dry solid radwastes are packaged in strong, tight containers. Gaseous radwastes are monitored, processed, recorded, and controlled so that radiation doses to persons outside the controlled area are below those allowed by applicable regulations.

Accessible portions of the reactor and radwaste buildings have sufficient shielding to maintain dose rates within the limits set forth in 10 CFR 20. The radwaste building is designed to preclude accidental release of radioactive materials to the environs.

The radwaste systems are used on a routine basis and do not require specific testing to ensure operability. Performance is observed by radiation monitors during operation.

The fuel storage and handling, and radioactive waste systems are designed to ensure adequate safety under normal and postulated accident conditions. The design of these systems meets the requirements of Criterion 61.

For further discussion, see the following:

a. Section 5.5 - Component and Subsystem Design

b. Section 6.2 - Containment Systems

c. Section 9.1 - Fuel Storage and Handling

d. Section 9.3 - Process Auxiliaries

e. Chapter 11 - Radioactive Waste Management

f. Chapter 12 - Radiation Protection

g. Section 14.1 - Test Program.

3.1.2.6.3 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.

Criterion 62 Conformance - Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in fuel storage is prevented by the geometrically safe configuration of the storage rack. There is sufficient spacing between the assemblies to ensure that the array, when fully loaded, is substantially subcritical. Fuel elements are limited by rack design to only toploading and fuel assembly positions. The fuel racks are Category I structures.

New fuel may be stored underwater in the spent fuel pool or placed in dry storage in the top-loaded new-fuel storage vault subject to the restrictions discussed in Section 9.1.1.2.1. This vault contains a drain to prevent the accumulation of water. The new-fuel storage vault racks (located inside the secondary containment) are designed to prevent an accidental critical array, even in the event that the vault becomes flooded or subjected to seismic loadings. The 6.625 in. by 11.5 in. center-to-center new-fuel assembly spacing limits the effective multiplication factor of the array to not more than 0.90 for new dry fuel. The effective neutron multiplication factor of the reactor (k_{eff}) will not exceed 0.95 if the new fuel is flooded.

Spent fuel is stored under water in the spent fuel pool. The high-density spent-fuel racks are constructed of stainless steel and include sheets of Boraflex or Boral, which are neutron attenuators. Sheets of Boraflex are used in all walls of the racks that contain Boraflex. For the racks that contain Boral, Boral panels are not needed on the exterior walls of modules facing non-fuel regions. In addition, Boral panels are used on only one exterior surface of the modules that face each other across the small water gap between the modules. The remaining conventional (low-density) spent-fuel racks are constructed of aluminum.

The spent-fuel storage racks are Category I structures designed to ensure that a k_{eff} not greater than 0.95 is maintained when the racks are fully loaded with fuel of the highest anticipated reactivity and flooded with unborated water at room temperature (68°F). The calculated k_{eff} includes a margin for uncertainty in k_{eff} calculations and in mechanical tolerances, statistically combined, so that the true k_{eff} will be less than 0.95 with a 95 percent probability at a 95 percent confidence level.

Refueling interlocks include circuitry that senses conditions of the refueling equipment and the control rods. These interlocks reinforce operational procedures that prohibit making the reactor critical. The fuel handling system is designed to provide a safe, effective means of transporting and handling fuel and is designed to minimize the possibility of mishandling or improper operation.

The use of conventional and of geometrically safe configurations for new-fuel storage and conventional and high-density storage racks for spent-fuel storage and the design of fuel handling systems preclude accidental criticality in accordance with Criterion 62.

For further discussion, see the following:

a. Section 7.6 - All Other Systems Required for Safety and Power Generation

b. Section 9.1 - Fuel Storage and Handling.

Dry Spent Fuel Storage - Storage of spent fuel at the Fermi Independent Spent Fuel Storage Installation (ISFSI) is governed by the regulations in 10 CFR 72 that are applicable to Part 72 general licensees, and the Certificate of Compliance (CoC) for the spent fuel storage cask. Furthermore, in accordance with the provisions of 10 CFR 50.68(c), while a spent fuel transportation package approved under 10 CFR 71 or a spent fuel storage cask approved under 10 CFR 72 is in the spent fuel pool:

1. The requirements of 10 CFR 50.68(b) do not apply to the fuel located within that package or cask, and

2. The requirements of 10 CFR 71 or 10 CFR 72, as applicable, and the requirements of the package or cask CoC apply to the fuel within that package or cask.

3.1.2.6.4 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.

Criterion 63 Conformance - Appropriate systems have been provided to meet the requirements of this criterion. A malfunction of the FPCC system, which could result in loss of RHR capability and excessive radiation levels, is alarmed in the main control room. Alarmed conditions include low fuel pool cooling water pump discharge pressure, high/low level in the fuel storage pool and skimmer surge tanks, and flow in the drain lines between fuel pool gates between fuel pool and reactor well. System temperature is also continuously monitored and alarmed in the main control room. The reactor building ventilation radiation monitoring system detects abnormal amounts of radioactivity and initiates appropriate action to control the release of radioactive material to the environs.

Area radiation and tank and sump levels are monitored and alarmed to give indication of conditions that may result in excessive radiation levels in radioactive waste system areas. These systems satisfy the requirements of Criterion 63.

For further discussion, see the following:

a. Section 7.6 - Other Systems Required for Safety and Power Generation

b. Section 9.1 - Fuel Storage and Handling

c. Section 11.2 - Liquid Radwaste System

d. Section 11.3 - Gaseous Radwaste System

e. Section 11.5 - Solid Radwaste System

f. Section 11.7 - Onsite Storage Facility.

3.1.2.6.5 Criterion 64 - Radioactivity-Release Monitoring

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.

Criterion 64 Conformance - Means have been provided for monitoring radioactivity releases resulting from normal and anticipated operational occurrences.

The following station releases are monitored:

a. Gaseous releases from the offgas system and the gland seal exhaust delay pipe

b. Liquid discharge to the circulating water pond decant line

c. Reactor building ventilation

d. Radwaste building ventilation

e. Turbine building ventilation

f. Deleted

g. Onsite storage building ventilation.

In addition, the drywell containment atmosphere is monitored by onsite monitors.

For further discussion of the means and equipment used for monitoring radioactivity releases, see the following:

a. Section 5.2 - Integrity of Reactor Coolant Pressure Boundary

b. Section 7.6 - Other Systems Required for Safety and Power Generation

c. Section 11.4 - Process and Effluent Radiation Monitor Systems.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components, and systems of the nuclear plant are considered important to safety because they perform safety actions required to avoid or mitigate the consequences of abnormal operational transients or accidents. The purpose of this section is to classify structures, components, and systems according to the importance of the safety functions they perform. In addition, design requirements are placed on such equipment to ensure the proper performance of safety actions, when required.

3.2.1 Seismic Classification

Plant structures, systems, and components important to safety are designed to withstand the effects of a safe-shutdown earthquake (SSE) and remain functional if they are necessary to ensure

a. Integrity of the reactor coolant pressure boundary (RCPB)

b. Capability to shut down the reactor and maintain it in a safe condition

c. Capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures set forth in 10 CFR 50.67 or 10 CFR 100.

Plant structures, systems, and components, including their foundations and supports, designed to remain functional in the event of an SSE and an operating-basis earthquake (OBE), are designated as Category I, as generically indicated in Table 3.2-1. A detailed tabulation of all Fermi 2 facility Category I equipment items is provided in the Central Component System (CECO). In this tabulation, each equipment item is described, facility installation locations are noted, the aseismic qualification basis is summarized, and the representative qualification documentation is identified. The CECO list is updated to reflect the facility item's aseismic status on a continual basis. The method of seismic qualification of some items is indicated in Table 3.2-2.

The Fermi 2 design fully conforms to the requirements of Regulatory Guide 1.29, Revision 3, Seismic Design Classification.

The radwaste system for the Fermi 2 plant is excluded from Category I criteria since the conservatively calculated offsite whole-body dose from radwaste system failure does not exceed 0.5 rem as specified in Regulatory Guide 1.29. The dose-rate considerations and analyses are discussed in Chapter 11, particularly in Subsections 11.2.3 and 11.3.3.

The recirculation pumps of a BWR plant are not considered essential for safe plant shutdown under either normal or abnormal conditions, even though Paragraph (h) of the Regulatory Position of this guide implies that reactor coolant pumps are required for safety. Thus, the pump seal purge system is not designed to meet Category I requirements with the exception of the components required for containment isolation. However, the pump seal and motor cooling water system are Category I, consistent with the structural design of the pumps and the recirculation system.

All Category I structures, systems, and components have been analyzed under the loading conditions of the SSE and OBE. Since the two earthquakes vary in intensity, the design of

Category I structures, components, and systems to resist each earthquake and other loads is based on levels of material stress or load factors, whichever is applicable, and yield margins of safety appropriate for each earthquake. The margin of safety provided for Category I structures, components, and systems for the SSE is sufficiently large to ensure that their safety functions are not jeopardized. For further details of specific seismic design criteria, refer to

a. Sections 3.7 and 3.9 for mechanical design criteria
b. Sections 3.7 and 3.8 for structural design criteria
c. Sections 3.7 and 3.10 for electrical design criteria
d. Sections 3.7 and 3.10 for instrumentation and control design criteria.

3.2.2 System Quality Group Classification

System Quality Group classifications as defined in Regulatory Guide 1.26 have been determined for each water-, steam-, or radioactive waste-containing component of those applicable fluid systems relied upon to

a. Prevent or mitigate the consequences of accidents and malfunctions originating within the RCPB
b. Permit shutdown of the reactor and maintain it in the safe-shutdown condition
c. Contain radioactive material.

A tabulation of Quality Group classification for each component so defined is shown in Table 3.2-1. Figures 3.2-1 and 3.2-2 depict the relative locations of these components along with their Quality Group classification.

Regulatory Guide 1.26 was still under development at the completion of the AEC staff review of the Fermi 2 construction permit application. Thus, the minimum code requirements for each Quality Group classification were those proposed by Edison and accepted by the AEC staff as evidenced in Subsection 3.3.3 and Table 3.3-3 of the AEC Safety Evaluation Report (Reference 1) resulting from their review. The substance of the table is shown in Table 3.2-3. Subsequent to issuance of the construction permit, Edison requested waiver from the requirements of 10 CFR 50.55a, which became effective July 12, 1971. The differences between the code requirements of Section 50.55a and those actually used, which were those required at the time of procurement of the component, are shown in Table 3.2-4. These differences were the subject of a waiver requested by Edison and approved by the AEC (Reference 2) except for Valve B31-F023. Reference 2 listed the purchase order date as October 1970 and code applied as NPVC, 70, for Valve B31-F023 in error. The correct data are November 1969 and NPVC, 68, as listed in Table 3.2-4.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

REFERENCES

1. "Safety Evaluation by the Division of Reactor Licensing, USAEC in the Matter of the Detroit Edison Company, Enrico Fermi Atomic Power Plant Unit 2, Docket 50-341," dated May 17, 1971.

2. Edison Letter EF2-17172, dated May 31, 1973, and AEC letter to Edison dated July 12, 1973. Re: Waiver of the code requirements of 10 CFR Part 50.55a.

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
I. Reactor system							
1. Reactor pressure vessel	GE	C	I	A	S	III-A	
2. Reactor Vessel support	GE	C	I	N/A	S	None	
3. Reactor Vessel appurtenances pressure retaining portions	GE	C	I	A	S	III-A	
4. CRD housing support	GE	C	I	N/A	S	None	i
5. Reactor internal structures, engineered safety features	GE	C	I	N/A	S	None	
6. Control rods	GE	C	I	N/A	B	None	
7. Control rod drives	GE	C	I	N/A	S	III-A	
8. Core support	GE	C	I	N/A	S	None	
9. Power range detector hardware	GE	C	I	N/A	S	III-A	j
10. Fuel assemblies	GE	C	I	N/A	B	None	
11. Reactor vessel stabilizer truss	GE	C	I	N/A	S	None	
II. Nuclear boiler system							
1. Vessels, level instrumentation chambers	GE	C	I	A	S	III-A	
2. Piping, relief valve discharge	E	C	I	B	B	III-2	
3. Piping, relief valve discharge inside vent line	E	C	I	D+	B	B31.1.0	
4. Relief valve discharge T-quenchers	E	C	I	C	B	III-3	
5a. Piping, main steam, from reactor inboard drywell penetration process pipe connectors	GE	C	I	A	S	B31.7-1	
5b. Piping, main steam, drywell penetration process pipe and piping to outboard MSIVs	E	C, R	I	A	B	III-1	

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
6. Pipe supports, main steam	GE	C	I	N/A	S	B31.7-1	
7. Pipe restraints, main steam	E	C, R	I	N/A	B	None	
8. Piping, other within outer-most isolation valves	E	C, R	I	A	B	III-1	j
9. Piping, instrumentation beyond outermost isolation valves	E	R, T	N/A	D	S	B31.1.0	j
10. Relief valves	GE	C	I	A	S	NPVC-1	
11. Valves, main steam isolation valves	GE	C, R	I	A	S	NPVC-1	
12. Valves, other, isolation valves and within	E	C, R	I	A	B	III-1	j
13. Valves, instrumentation beyond outermost isolation valves	E	R, T	N/A	D	S	B16.5	j
III. Reactor recirculation system							
1. Piping	GE	C	I	A	S	B31.7-1	j
2. Pipe suspension recirculation line	GE	C	I	N/A	S	B31.7-1	
3. Pipe restraints recirculation line	GE	C	I	N/A	S	None	
4. Pumps	GE	C	I	A	S	NPVC-1	z
5. Valves	GE	C	I	A	S	NPVC-1	j
6. Motor, pump	GE	C	I	N/A	S	None	
IV. CRD hydraulic system							
1. Valves	GE, E	R	I	B	S	III-2	j
2. Valves, other	GE, E	R	N/A	D	S	B16.5	j
3. Piping, scram discharge volume lines	E	R	I	B	B	III-2	
4. Piping, insert and withdraw lines	E	C. R	I	B	B	III-2	
5. Piping, other	E	R	N/A	D	S	B31.1.0	j
6. Hydraulic control unit	GE	R	I	N/A	S	None	l

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
V. Standby Liquid control system							
1. Standby liquid control tank	GE	R	I	D	BM	API 650	m
2. Pump	GE	R	I	C	BM	NPVC-3	
3. Pump motor	GE	R	I	N/A	BM	None	
4. Valves, explosive	GE	R	I	C	BM	NPVC-3	
5. Valves, isolation and within	E	C, R	I	A	B	III-1	j
6. Valves, beyond isolation valves	E	R	I	C	B	III-3	j
7. Piping, within isolation valves	E	C,R	I	A	B	III-1	j
8. Piping, beyond isolation valves	E	R	I	C	B	III-3	j
VI. Neutron monitoring system							
1. Piping, TIP	GE	R	I	N/A	S	None	
2. Valves, isolation, TIP subsystem	GE	R	I	N/A	S	None	
3. Instrumentation and control rod block monitoring	GE	R	II/I	N/A	S	None	
4. APRM	GE	R	I	N/A	S	IEEE 344, IEEE 323	
VII. Reactor protection system							
1. Electrical trip	GE	R, T	I	N/A	B	IEEE 344, IEEE 323	
VIII. Process radiation monitoring system							
1. Main steam line radiation monitors, fuel pool ventilation exhaust radiation monitors	GE	R	I	N/A	B	IEEE 344, IEEE 323	
2. Control center emergency air inlet radiation monitors	E	A	I	N/A	B	IEEE 323	

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
3. Control center normal make-up air radiation monitors	E	A	II/I	N/A	B	IEEE 323	
1. Torus Hardened Vent Radiation Monitor System	E	A	II/I	N/A	S	IEEE 344	
IX. Residual heat removal system							
1. Heat exchangers, primary side	GE	R	I	B	S	III-C	
2. Heat exchangers, secondary side	GE	R	I	C	S	VIII & TEAM-C	
3. Piping, within outer most isolation valves	E	C, R	I	A	B	III-1	j
4. Piping, beyond outer most isolation valves	E	R	I	B	B	III-2	j
5. Pumps	GE	R	I	B	S	NPVC-2	
6. Pump motors	GE	R	I	N/A	S	None	
7. Valves, isolation, LPCI line and SDC suction	E	C, R	I	A	B	III-1	
8. Valves, isolation, torus suction, containment spray, head spray and test lines	E	C, R	I	B	B	III-2	j, x
9. Valves, beyond isolation valves	E	R	I	B	B	III-2	
X. Core spray system							
1. Piping, within outermost isolation valves	E	C, R	I	A	B	III-1	j
2. Piping, beyond outermost isolation valves	E	R	I	B	B	III-2	j
3. Pumps	GE	R	I	B	S	NPVC-2	
4. Pump motors	GE	R	I	N/A	S	None	
5. Valves, isolation and within	E	C, R	I	A	B	III-1	j

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
6. Valves, beyond outermost isolation valves	E	R	I	B	B	III-2	j
XI. High-pressure coolant injection system							
1. Steam turbine	GE	R	I	N/A	S	None	n
2. Piping, suction line from condensate storage tank isolation valve	E	R,O	I	B	B	III-2	j
3. Piping, turbine steam supply and discharge	E	R	I	B	B	III-2	
4. Piping, return test line to condensate storage tank downstream of second isolation valve	E	R,O	N/A	D	S	B31.1.0	
5. Piping, within outermost isolation valve	E	C,R	I	A	B	III-1	
6. Piping, suppression pool suction and pump discharge	E	R	I	B	B	III-2	j
7. Main pump	GE	R	I	B	S	NPVC-2	
8. Booster pump	GE	R	I	B	S	NPVC-2	
9. Valves, beyond outermost isolation valves	E	R	I	B	B	III-2	
10. Valves, outer isolation and within	E	C,R	I	A	B	III-1	j
11. Valves, beyond isolation valves, motor operated	E	R	I	B	B	III-2	j
XII. Reactor core isolation cooling system							
1. Piping, within outermost isolation valves	E	C,R	I	A	B	III-1	j
2. Piping, beyond outermost isolation valves	E	R	I	B	B	III-2	j

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
3. Piping, return test line to condensate storage tank downstream of second isolation valve and vacuum pump discharge line to containment isolation valves	E	R,O,A	I,N/A	D	S	B31.1	j
4. Pumps	GE	A	I	B	S	NPVC-2	
5. Valves, isolation and within	E	C,R	I	A	B	III-1	j
6. Valves, other	E	R	I	B	B	III-2	j
7. Turbine	GE	A	I	N/A	S	None	n
8. Piping, suction line from condensate storage tank	E	R,O	I	B	B	III-2	j
XIII. Fuel service equipment							
1. Fuel preparation machine	GE	C,R	II/I[y]	N/A	S	None	
2. General-purpose grapple	GE	C,R	II/I	N/A	S	None	
XIV. Reactor pressure vessel service equipment							
1. Steam line plugs	E	C	I	N/A	S	None	
2. Dryer and separator sling and head strongback	GE	C	I	N/A	S	None	
3. Head Strongback Carousel	GE	C	I	N/A	S	None	
XV. In-vessel service equipment							
1. Control rod grapple	GE	C	II/I	N/A	S	None	
2. Reactor Cavity Work Platform	E	R	II/I	N/A	B	None	
XVI. Refueling equipment							
1. Refueling equipment platform assembly	GE	C	II/I	N/A	S	None	
2. Refueling bellows	E	C	I	B	S	III-2	aa
XVII. Storage equipment							

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
1. Defective- fuel storage container	GE	R	I	N/A	S	None	
2. High-density fuel storage racks	E	R	I	N/A	B	III-NF	
XVIII. Radwaste System							
1. Tanks, atmospheric vessels	E	W	N/A	D	S	API-620 & 650 VIII	o
2. Heat exchangers and evaporators	E	W	N/A	C,D	S	VIII & TEMA-C	p
3. Piping and valves	E	C,R,W	N/A	D	S	B31.1.0	
4. Pumps	E	W	N/A	C,D	S	III-3, B31.1.0	j,p
5. Piping and valves, containment isolation	E	C,R	I	B	B	III-2	p
6. Valves, flow control and filter system	E	W	N/A	C,D	S	III-3, B16.5	p
7. Valves, other	E	W	N/A	D	S	B16.5	
XIX. Reactor water cleanup							
1. Vessels: filter demineralizer	GE	R	N/A	C	S	III-C	
2. Heat exchangers, regenerating	GE	R	N/A	D	S	III-C, TEMA-R	
nonregenerating: tube side,	GE	R	N/A	D	S	III-C, TEMA-R	
Nonregenerating: shell side	GE	R	N/A	D	S	VIII, TEMA-R	
3. Piping, within outermost isolation valves	E	C,R	I	A	B	III-1	
4. Piping, beyond outermost isolation valves	E	R,T,W	N/A	C,D	S	III-3 B31.1.0	j,k
5. Pumps (recirculation, precoat, and holding)	GE	R	N/A	D	S	NPVC-3	
6. Valves, isolation valves and within	E	C,R	I	A	B	III-1	j,q,r
7. Valves, beyond reactor isolation valves	GE	R	N/A	C	S	NPVC-3	j
	E	R,T,W	N/A	D	S	B16.5	j

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
XX. Fuel pool cooling and cleanup system							
1. Vessels, filter-demineralizers	GE	W	II/I	C	S	VIII	
2. Vessels, other	E	W	II/I	N/A	S	None	
3. Heat exchangers	GE	R	II/I	C,	S	VIII, TEMA-R	
4. Piping	E	W,R	N/A, I/I, I	C,B,D	S	III-3, B31.1.0	j
5. Pumps	GE	R	II/I	C	S	NPVC-3	
6. Valves	E	R	N/A, II/I, I	C,B,D	B	III-3, B16.5	J
XXI. Control center panels	GE, E	A	I	N/A	S,B	IEEE	t
XXII. Local panels and racks	GE, E	R,A,H	I	N/A	S,B	IEEE	t
XXIII. Offgas system							
1. Tanks, drains and condensate receiver	E	T	N/A	D	S	VIII	
2. Heat exchangers	E	T	N/A	D	S	AEG-VIII, TEMA-C	
3. Piping	E	T	N/A	D	S	B31.1.0,	
4. Pumps, ring water vacuum	E	T	N/A	D	S	MANF. STD	
5. Valves, flow control	E	T	N/A	D	S	B31.1.0	
6. Valves, other	E	T	N/A	D	S	B31.1.0	
7. Pressure vessels, ring water buffer tanks	E	T	N/A	D	S	AEG-VIII	
XXIV. RHR service water system							
1. Piping	E	H,O,R	I	C	B	III-3	
2. Pumps	E	H	I	C	B	III-3	
3. Pumps motors	E	H	I	N/A	B	None	
4. Valves	E	H,R	I	C	B	III-3	
5. Mechanical draft cooling towers, including structure fans, and related hardware	E	H	I	N/A	B	None	

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
XXV. Plant service and cooling water systems							
1. Piping and valves forming part of primary containment boundary	E	C,R	I	B	B	III-2	
XXVI. Noninterruptable air and pneumatic supply systems							
1. Vessels, accumulators supporting safety-related systems	E	C,R	I	C	B	III-3	
2. Piping and valves	E	C,R	I	C	B	III-3	
3. Control air compressors	E	A	I	D	B	VIII, B31.1.0	
4. Control air dryers	E	A	I	D	B	VIII, B31.1.0	
5. Receiver tanks	E	A	I	C	B	III-3	
6. Control air aftercooler	E	A	I	D	B	VIII, B31.1.0	
7. Isolation valves	E	A,R	I	C	B	III-3	
8. Pressure regulating valves	E	A,R	I	C	B	III-3	
XXVII. Diesel generator systems							
1. Day tanks, fuel oil storage and day tanks	E	H	I	C	B	III-3	
2. Piping and valves, fuel oil system	E	H	I	C	B	III-3 (see Fig. 9.5-4, 5 and 6)	
3. Pumps, fuel oil system	E	H	I	N/A	B	None	
4. Pumps, piping, valves and heat exchangers, diesel service water system	E	H	I	C	B	III-3	
5. Jacket and air coolant piping, valves, and heat exchangers	E	H	I	C	B	III-3 (see Fig. 9.5-7)	
6. Pump motors, diesel service water system	E	H	I	N/A	B	None	
7. Diesel generators	E	H	I	N/A	B	None	
8. Starting air receivers piping and valves, combustion air intake piping	E	H	I	C	B	III-3	

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
9. Lube oil cooler	E	H	I	C	B	III-3	
10. Exhaust piping	E	H	I	D	S	B31.1.0	
11. Skid-mounted lube oil system	E	H	I	N/A	S		
12. Starting Air Receivers Safety Relief Valves	E	H	I	D	B	VIII, B31.1.0	
XXVIII. Primary containment atmosphere control system							
1. Piping and valves from primary containment through outer isolation valve	E	R	I	B	B	III-2	
XXIX. Standby gas treatment system							
1. Containment pressure boundary piping and valves	E	R	I	B	B	III-2	
2. Piping, downstream to secondary containment suction valves	E	R,A	NA	D	S	B31.1.0	
3. Piping and valves, secondary containment suction valves to filter unit ductwork	E	R,A	I	D	B	B31.1.0	
4. Cooling and exhaust fan	E	A	I	N/A	B		
5. Filter unit and associated duct and valves	E	A	I	N/A	B		
6. Exhaust vent stack	E	A,O	I	N/A	B	AISC	
7. Piping and valves, inlet header to torus vent stack	E	R,O	I	D	B	B31.1.0	
XXX. Emergency equipment cooling water system							
1. All components with safety functions, except as listed in XXV	E	R	I	C	B	III-3	
XXXI. Emergency equipment area cooling system							

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
1. All components with emergency equipment cooling coils safety function	E	R,A	I	C	B	III-3	
2. RHR complex HVAC system components with safety fuction	E	H	I	N/A	B		
XXXII. Power conversion system							
1. Main steam piping from outboard MSIVs to third MSIVs	E	R,T	II/I	D	S	B31.1.0	
2. Main steam branch-line piping and valves downstream of outboard MSIVs (for branch lines between outboard and third MSIVs)	E	R,T	II/I	D	S	B31.1.0	
3. Feedwater piping, beyond outermost isolation valves	E	R,T	N/A	D	S	B31.1.0	
4. Feedwater piping, within outermost isolation valve	E	C,R	I	A	B	III-1	
5. Valves, isolation valves and within, feedwater	E	C,R	I	A	B	III-1	
6. Valves, beyond outermost isolation valves, feedwater	E	R,T	N/A	D	S	B16.5	
							r
XXXIII. Condensate storage and transfer system							u, s
1. Condensate storage tank	E	O	N/A	D	S	USAS B96.1	s
2. Piping and valves, except HPCI/RCIC suction	E	M	N/A	D	S	B31.1.0, B16.5	s
3. Other components	E	M	N/A	D	S	(see Table 3.2-2)	
XXXIV. Auxiliary ac power system							
1. All components with safety function	E	A,R,H	I	B, N/A	B	IEEE 308/IEEE 344	

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
2. Primary electrical penetrations	E	R,C	I	B	B	IEEE 336 III-NE IEEE 317	v
3. Diesel generator packages including auxiliaries (e.g., governor, voltage regulator, excitation system, and control and relay protection equipment) not listed in XXVII	E	H	I	N/A	B		
4. 4160V switchgear	E	A,H	K	N/A	S		
5. 480V load centers	E	A,H	K	N/A	S		
6. 480V motor control centers	E	A,R,H	I	N/A	S		
7. Conduit and tray supports (installation containing class 1E cables and other installations whose failure may damage other safety-related items)	E	A11	I	N/A	S		
8. Transformers	E	A,H	K	N/A	S		
9. Valve operators	E	A11	I	N/A	S		
10. Protective relays and control panels	E	H,R	I	N/A	S		
11. 120V ac instrument power supply and distribution equipment	E	A	K	N/A	S		
12. Fire-rated penetrations	E	A11	I	N/A	S		
XXXV. DC power systems							
1. All components with safety function	E	A,R,H	I	N/A	B	IEEE 308	
a. 260/130V batteries, battery racks, battery chargers, and dc distribution equipment	E	A,R,H	K	N/A	S		

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
b. Conduit and tray supports (installations containing class 1E cables and other installations whose failure may damage other safety-related items)	E	A11	I	N/A	S		
XXXVI. Civil structures							
1. Primary containment	E	R	I	B	B	III-B	
2. Reactor auxiliary building	E	R,A	I	N/A	B	ACI-318, AISC	
3. Auxiliary building steel framed penthouse components that are not required to support the crane or the secondary containment	E	A	II/I	N/A	S	AISC	
4. Steam tunnel	E	T	I	N/A	B	ACI 318, AISC	
5. Radwaste building	E	W	N/A	N/A	S	ACI 318, AISC	
6. Circulating water pump house	E	P	N/A	N/A	S	ACI 318, AISC	
7. Control center complex (including cable spreading room)	E	A	I	N/A	B	ACI 318, AISC	
8. RHR complex	E	H	I	N/A	B	ACI 318, AISC	
9. Radiation shielding	E	R,A,C	I	N/A	B	ACI 318, AISC	
Sacrificial shielding wall							
Reactor building							
Auxiliary building							
Control center complex							
Masonry wall, safety related							
10. Support truss (pipe break)	E	C	I	N/A	S		
11. ISFSI Equipment Storage Building	E	I	N/A	N/A	B	ACI 318, AISC	
12. ISFSI Storage Pad	E	J	I	N/A	APP 17.2A	ACI 349	
13. ISFSI Fabrication Pad	E	K	N/A	N/A	APP 17.2A	ACI 318	

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
14. ISFSI Transfer Pad	E	L	N/A	N/A	APP 17.2A	ACI 318	
15. Original Cat. I 4160-V Ductbanks between RHR Complex & Auxiliary Bldg.	E	O	I	N/A	B	ACI 318	
16. Second Set of Cat. I 4160-V Ductbanks between RHR Complex & Auxiliary Bldg.	E	O	I	N/A	B	ACI 349-01 & ACI 318-05	
17. ISFSI Cask Transfer Facility	E	N	N/A	N/A	APP 17.2A	ACI 318	
18. FLEX Storage Facility #1 & #2	E	O	N/A	N/A	N/A	ACI 318 & AISC	
XXXVII. Post-LOCA hydrogen control system							
1. All components with safety function	E	R	I	B	B	III-2	
XXXVIII. Reactor building crane	E	R	I	N/A	S	EOCI	
XXXIX. Control center air conditioning system							
1. Condenser coil and associated piping	E	A	I	C	B	III-3	
2. Chilled water piping	E	A	I	D	B	B31.1.0	
3. Piping, chilled water makeup	E	R,A	N/A	D	S	B31.1.0	
4. Isolation dampers	E	R,A	I	N/A	B		
5. Cooling units for equipment room	E	A	I	N/A	B		
6. Chillers	E	A	I	D	B	VIII	
7. Multizone units	E	A	I	N/A	B		
8. Supply fans	E	A	I	N/A	B		
9. Recirculation, emergency makeup air filter units	E	A	I	N/A	B		
10. Recirculation air filter units and fans	E	A	I	N/A	B		

TABLE 3.2-1 STRUCTURES, SYSTEMS, AND COMPONENTS CLASSIFICATION[a]

Principle Component[b]	Scope of Supply[c]	Location[d]	Category[e]	Quality Group Classification[f]	Quality Assurance Requirements[g]	Principal Construction Code[h]	Remarks
11. Chilled water pumps	E	A	I	D	B		
12. Return fans	E	A	I	N/A	B		
13. Associated ductwork	E	A	I	N/A	B		
14. Associated motors	E	R,A	I	N/A	B	None	
XL. Shore barrier	E	O	I	N/A	B	None	
XLI. MSIV leakage control system	(Not Required per License Amendment No. 160)						
XLII. Postaccident sampling							
1. Sample isolation valves and piping	E	R	I	A,B	B	III-1,2	
2. Sampling station and tubing downstream of isolation valves	GE, E	A	N/A	C,D	S	III-3, B31.1	j
XLIII. Cable and associated hardware with safety function	GE, E	All	N/A	N/A	B	IEEE/ICC/ WG-12-32 I333 323	t
XLIV. Locally mounted instrumentation with safety function (not rack or panel mounted)	GE, E	R,A,H	I	N/A	S,B	IEEE	
XLV. Fire detection, suppression, and extinguishing systems, emergency lighting, and breathing apparatus	E	All	N/A	N/A	N/A	N/A	s

Note a

Safety-related instrumentation and control systems and components are identified in Chapter 7 and will be subject to the operational QA Program requirements.

Note aa

The reactor refueling bellows was designed, fabricated, and installed as ASME Class 2 but was not N-stamped.

Note b

A module is an assembly of interconnected components that constitutes an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies, and signal processors; mechanical modules include turbines, strainers, and orifices.

Note c

GE = Supplied by General Electric

E = Supplied by the Detroit Edison Company.

Note d

Location abbreviations are:

A = Auxiliary building

C = Part of, or within, primary containment

H = RHR complex

I = Independent Spent Fuel Storage Installation (ISFSI) Equipment Storage Building

J = Independent Spent Fuel Storage Installation (ISFSI) Storage Pad

K = ISFSI Fabrication Pad

L = ISFSI Transfer Pad

M = Any other location

N = ISFSI Cask Transfer Facility

O = Outdoors onsite

P = Circulating water pump house

R = Reactor building

T = Turbine building

W = Radwaste building

Note e

I = The equipment is constructed in accordance with the seismic requirements for the SSE and OBE as described in Section 3.7.

K = The equipment is constructed in accordance with the seismic requirements as described in Section 3.10.

NA = The seismic requirements for the SSE are not applicable to the equipment.

II/I = The equipment is constructed in accordance with the seismic requirements of Category II/I described in Section 3.7.

Note f

The structure, system, or component is constructed in accordance with the codes listed in Table 3.2-3.

Note g

B = The structure, system, or component meets the QA requirements of 10 CFR 50, Appendix B, in accordance with the QA Program described in Chapter 17.

S = Items ordered with specific QA requirements identified in the purchase documents. This includes items purchased prior to the issuance of 10 CFR 50, Appendix B (35 FR 10499, June 27, 1970). For example, this would include items purchased under the contract with General Electric (the NSSS supplier), which was effective August 15, 1968.

BM = The system or component will be maintained according to the requirements of 10 CFR 50, Appendix B, but was not originally procured according to Appendix B.

App 17.2A = ISFSI Storage Pad and ISFSI Cask Transfer Facility are ITS-C; See UFSAR Appendix 17.2A

Note h

Notation for principal construction codes is:

III-A,B,C,1,2,3 - ASME Boiler and Pressure Vessel Code Section III, Class A,B,C,1,2, or 3 or Subsection NE, Class NE. (Pre-1971 versions of the code used the Class A,B,C, designation while 1971 and later versions used the Class 1,2,3 designation. Equipment was ordered throughout a period requiring use of both designations)

VIII - ASME Boiler and Pressure Vessel Code Section VIII, Pressure Vessels, Division I

B31.7-1,2,3 - ANSI Nuclear Power Piping Code Class 1, II, III

B31.1.0 - ANSI B31.1.0 Standard Code for Pressure Piping, Power Piping

NPVC - 1,2,3 Draft ASME Code for Pumps and Valves for Nuclear Power, Class I,II,III

IEEE 308-1971 - IEEE Criteria for Class 1E Electric System, for Nuclear Power Generating Station

IEEE 317-1971 - IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations

IEEE 344-1971 - Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations.

IEEE/ICC/WG-12-32 - Proposed Guide for Type Tests of Class I Cables and Connections Installed Inside the Containment of Nuclear Generating Stations

TEMA-C,R - Tubular Exchanger Manufacturer Association, Class C,R

ACI 318 - Building Code Requirements for Reinforced Concrete 1963 and 1971. Note: Code Year 2005 used for ISFSI structures and the second set of Category I 4160-V underground ducts, manholes and cable vault structures only.

ACI 349-01 – Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary

AEG-VIII - Manufactured in West Germany in accordance with ASME Boiler and Pressure Vessel Code Section VIII, Division I, but not code stamped. Code compliance certified by third-party inspectors

AISC - Specification for the Design Fabrication and Erection of Structural Steel for Buildings

API 650 - Welded steel tanks for oil storage

API 620 - Specifications for Welded Steel Storage Tanks

B96.1 - USAS B96.1 - Welded aluminum alloy field-erected storage tanks

B16.5 - ANSI B16.5 - Steel pipe flanges and flanged fittings

EOCI - Electric Overhead Crane Institute.

(Other Civil and Structural Codes are given in Section 3.8.)

Note i

Maintenance on all components within the reactor internal structures will be performed in accordance with 10 CFR 50, Appendix B.

Note j

1. All instrument lines that are connected to the RCPB and are not utilized to actuate safety systems are Quality Group D from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation

2. All other instrument lines:

    - Through the root valve; shall be of the same classification as the system to which they are attached

    - Beyond the root valve, if used to actuate a safety system; shall be of the same classification as the system to which they are attached

    - Beyond the root valve; if not used to actuate a safety system, are Quality Group D.

3. All sample lines from the outer isolation valve or the process root valve through the remainder of the sampling system are Quality Group D

4. Portions of instrument lines (regardless of the originating quality group) passing through primary containment are part of a penetration assembly that is part of containment. As such, these lines are Quality Group B, consistent with the Containment Quality Group. This is in accordance with Regulatory Guide 1.11 and Note 2(a) referenced from 10 CFR 50.55a(d)(2).

Note k

The recirculation pumps of a BWR plant are not considered essential for safe plant shutdown under either normal or abnormal conditions, even though Paragraph (h) of the Regulatory Position of Regulatory Guide 1.29 implies that reactor coolant pumps are required for safety. Thus, the pump seal purge system is not designed to meet Category I requirements with the exception of the components required for containment isolation. However, the pump seal and motor cooling water system are Category I, consistent with the structural design of the pumps and the recirculation system.

Note l

The hydraulic control unit (HCU) is a GE factory-assembled engineered module of valves, tubing, piping, and stored water which controls a single control rod drive (CRD) by the application of precisely timed sequences of pressures and flows. Control is accomplished by slow insertion or withdrawal of the control rods for power control, and rapid insertion for reactor scram

Although the HCU, as a unit, is field installed and connected by process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide. Thus, although the codes and standards invoked by the Group A, B, C, D pressure integrity quality levels clearly apply at all levels to the interfaces between the HCU and the connecting conventional piping components (e.g., pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components, and instruments)

The design and construction specification for the HCU invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example: (1) all welds are LP inspected, (2) all socket welds are inspected for gap between pipe and socket bottom, (3) all welding is performed by qualified welders, and (4) all work is done per written procedures

Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses that permit the use of manufacturer's standards and proven design techniques which are not explicitly defined within the codes of Quality Groups A, B, or C. This is supplemented by the QC techniques described above.

Note m

The standby liquid control system storage tank is Group D plus the following additional QC:

a. Spot radiographic inspection was performed on all vertical and horizontal shell butt welds and on all bottom butt welds. Methods, techniques, and acceptance standards were in accordance with the requirements of API 650

b. Liquid-penetrant inspection was performed on all tank nozzle welds below and including the overflow nozzle both internal and external to the tank. All fillet and socket welds received a random liquid penetrant examination. Methods, technique, and acceptance standards were in accordance with the ASME B&PV Code Section VIII, Division I.

Note n

The RCIC and HPCI turbines do not fall within the applicable design codes. To ensure that the turbine is fabricated to the standards commensurate with their safety and performance requirements, GE has established specific design requirements for this component. These requirements are given in the appropriate GE internal documents.

Note o

The radwaste system for Fermi 2 is excluded from Category I criteria because the conservatively calculated offsite whole-body dose from radwaste system failure does not exceed 0.5 rem as specified in Regulatory Guide 1.29. The dose-rate considerations and analyses are discussed in Chapter 11, particularly Subsections 11.2.3 and 11.3.3.

Note p

Section VIII of ASME B&PV Code and ANSI B31.1.0 apply downstream of the outermost isolation valves.

Note q

Three valves, one inside and two outside the containment, are placed in the RWCU influent line. The RWCU effluent line has two valves, one inside and one outside containment. The RWCU system beyond the third isolation valve G3352F119 on the influent line up to the outside containment isolation valve G3352F220 on the effluent line is constructed in accordance with the applicable codes of Code Group D.

Note r

The first valve capable of timely actuation in branch lines connected to the main steam lines between the outermost containment isolation valve and the third isolation valve, meets all of the pressure integrity requirements of Group D plus the following additional requirements:

1. Pressure-retaining components of all cast parts of valves are subject to volumetric examination or surface examination methods. Ultrasonic examination to equivalent standards is used as an alternate to radiographic methods. If size or configuration does not permit effective volumetric examination, magnetic-particle or liquid-penetrant methods are substituted

2. All inspection records are retained for the life of the plant (See Appendix B, Section B, for discussion of operation beyond the original design plant life). These records include data pertaining to the qualification of inspection personnel, examination procedures, and examination results. A certification has been obtained from the vendors of the turbine stop valves and turbine bypass valves stating that all cast pressure-retaining parts of a size and configuration for which volumetric examination methods are effective have been examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards may be used as an alternative to radiographic methods.

Note s

The spring-loaded piston operator of the valve is held open by air pressure during normal operation. Fail-open solenoid valves are used to release air pressure and to permit the check valve piston operators to close. The valves are remote manually operated from the main control room using signals that indicate loss of feedwater flow.

The classification of the feedwater line from the reactor pressure vessel through the third isolation valve is Quality Group A. The remainder of these systems is Quality Code Group D.

Note t

The specific IEEE construction codes used for a particular component may be found in the purchase document referenced in the Master Instrument List.

Note u

The outermost valve of the three isolation valves in the feedwater lines is similar to a boiler feed pump check valve.

Note v

The condensate storage tank is designed, fabricated, and tested to meet the intent of API 650. In addition, the specifications for the tank require that

1. All shell joints are full penetration and fusion welds
2. All shell joints are radiographed 100 percent
3. Shell to bottom joint is 100 percent liquid penetrant examined.

Note w

Fire detection, suppression, and extinguishing systems, emergency lighting, and breathing apparatus impacting safety-related areas of the plant are periodically inspected, maintained, and tested for proper operation per the Operational QA Program Requirements.

Note x

Residual heat removal (RHR) head spray line between reactor pressure vessel and bulkhead penetration is removed. Therefore, head spray portion of RHR shutdown cooling is no longer part of the reactor coolant pressure boundary. Also, an in-line blank orifice plate isolates the head spray piping from the RHR System. The head spray piping and its associated components have been downgraded to Quality Group B (piping and components between and including isolation valves E1150F022 and E1150F023) or Quality Group D (all other head spray piping and components that are not part of the RHR System pressure boundary).

Note y

The fuel preparation machines are used for removing and replacing channels on fuel assemblies and fuel bundle inspection. They are not required to prevent or mitigate the consequences of postulated accidents. Therefore, they are classified QA level non-Q and seismic category II/I. They were originally supplied by GE as passive, safety-related components, seismically qualified to the Fermi 2 design basis OBE and SSE seismic events.

Note z

The reactor recirculation pumps are upgraded to the 4th generation design. The modified RCPB components were designed and manufactured to ASME III, Class 1, 1989, No addenda.

TABLE 3.2-2 CATEGORY I MECHANICAL COMPONENTS: METHOD OF SEISMIC QUALIFICATION

Components	Testing	Analysis	Remarks
General			
Category I piping		X	
NSSS valves (by GE)	X	X	See note a
BOP valves (by Edison)	X	X	See note b
Penetration assemblies		X	
Specific			
Reactor vessel and internals		X	
Control rods		X	
Control rod drives and housings		X	
Fuel assemblies		X	
Safety/relief valves		X	
Air accumulators		X	
Main steam isolation valves	X	X	See note a
Recirculation pumps and motors		X	Nonessential; see note c
Recirculation valves		X	
CRD hydraulic control units	X		
Standby liquid control tank		X	
SLCS pump and motor		X	
RHR heat exchangers		X	
RHR pumps		X	
RHR pump motors		X	
Core spray		X	
Core spray pump motors		X	
HPCI steam turbine		X	
HPCI pumps		X	
RCIC steam turbine		X	
RCIC pumps		X	
Refueling platform		X	See note e
Refueling bellows		X	Nonessential; see note c

TABLE 3.2-2 CATEGORY I MECHANICAL COMPONENTS: METHOD OF SEISMIC QUALIFICATION

Components	Testing	Analysis	Remarks
Fuel storage racks		X	
RHR service water pumps		X	
RHR service water pump motors		X	
RHR cooling towers		X	
Control air compressors		X	
Control air dryers	X		
Control air aftercoolers	X		
Control air receiver tanks		X	
Control air afterfilter	X		
Diesel generator day tanks		X	
Fuel oil tanks		X	
Fuel oil pumps		X	
Diesel generator service water pump		X	
Diesel generator pump motors		X	
Diesel generators		X	
Standby gas treatment filter units		X	
EECW heat exchangers		X	
EECW pumps and motors		X	
EECW makeup pumps and motors		X	
EECW service water pump and motors		X	
ECCS equipment area cooling units		X	
EECW makeup tanks		X	
Primary containment		X	
Reactor building crane		X	
Post-LOCA hydrogen control system	X	X	See note d
Drywell coolers		X	
Drywell cooler fans		X	
Floor and equipment drain sumps		X	
Floor and equipment drain sump pumps		X	

TABLE 3.2-2 CATEGORY I MECHANICAL COMPONENTS: METHOD OF SEISMIC QUALIFICATION

Components	Testing	Analysis	Remarks
Reactor building HVAC isolation dampers		X	
Control center multizone units		X	
Return air fans		X	
Chillers		X	
Chilled water pumps and motors		X	
Emergency makeup air filter		X	
Recirculation air filter		X	
Recirculation air filter fans		X	
Fan-coil units		X	
Battery room fans		X	

---

[a] Prototype test was conducted for the main steam isolation valves (Atwood and Morrill type, furnished by GE).

[b] Prototype tests were conducted for Limitorque motor operators, including operability tests.

[c] Components that are listed as nonessential are not required to operate during or after a safe-shutdown earthquake but have to retain their integrity for pressure-retaining functions.

[d] Prototype tests were conducted for the hydrogen control and recombiner system, and seismic analysis was conducted as part of the stress analysis of pressure-retaining components and piping.

[e] The refueling platform has been reclassified as Seismic Category II/I.

TABLE 3.2-3 MINIMUM CODE REQUIREMENTS FOR QUALITY GROUP CLASSIFICATION

Component	Group A	Group B	Group C	Group D
Pressure vessels	ASME B&PV Code Section III, Class A	ASME B&PV Code Section III, Class C	ASME B&PV Code Section VIII, Division I	ASME B&PV Code Section VIII, Division I or equivalent
0-15 psig storage tanks	None	API-620	API-620	API-620 or equivalent
Atmospheric storage tanks	None	API-650, ANSI B96.1	API-650, ANSI B96.1	API-650, ANSI B96.1 or equivalent
Piping	ANSI B31.7, Class I	ANSI B31.7, Class II	ANSI B31.7, Class III	ANSI B31.1.0 or equivalent
Pumps and valves	ASME Code for Pumps and Valves Class I	ASME Code for Pumps and Valves Class II	ASME Code for Pumps and Valves Class III	Valves-ANSI B31.1.0 or Equivalent Pumps-ASME Code for pumps. Valves Class III or equivalent

---

These code requirements were established and agreed to by the AEC during the Construction Permit Review (AEC Staff Safety Evaluation Report, Table 3.3.3) and do not, in all cases, conform to the codes indicated in Regulatory Guide 1.26. However, as noted under Principal Construction Code, Table 3.2-1, many of the construction codes actually used exceed the above and meet the Regulatory Guide 1.26 requirements. For example, the primary electrical penetrations conform to ASME B&PV Code Section III, Subsection NE, Class NE.

These requirements were supplemented and modified as shown in Table 3.2-4 and explained in Subsection 3.2.2.

For code definitions, see Note h of Table 3.2-1.

TABLE 3.2-4 CODE STATUS OF CLASS I (A) PRIMARY PRESSURE BOUNDARY COMPONENTS

Component Description	Quantity	Plant Identification System Number	Purchase Order Date	Code Applied	Code Required per 10 CFR 50, 55a
Reactor pressure vessel[a]	1	B11-A001	Jan. 67	ASME III[b] 69S[c]	ASME III, 70S
RPV head nozzle	1	B11-D072[i]	May 71	ASME III, 70S	ASME III, 70S
CRD housinga	185	B11-D141, 142, 143, 144	Aug. 70	ASME III, 69W[d]	ASME III, 70S
CRD[e]	185	B11-D146	July 70	ASME III, 69W	ASME III, 70S
In-core housing	55	B11-D190, 198	Sept. 70	ASME III, 69W	ASME III, 70S
Jet pump instrument penetration	2	B11-D235	Jan. 71	ASME III, 70S	ASME III, 70S
Safety/relieve valve	15	B21-F013	Jan. 71[f]	NPVC, 70	ASME III, 71
MSIV inboard	4	B21-F022	Oct. 69	NPVC, 68[g]	ASME III, 71
MSIV outboard	4	B21-F028	Oct. 69	NPVC, 68	ASME III, 71
Primary steam piping	1	B21-G001	Sept. 70	B31.7, [h]69	ASME III, 71S
Main steam flow element	2	B21-N005	Jan. 71	B31.7, 69	ASME III, 71S
Recirc. pump[j]	2	B31-C001	Dec. 69	NPVC, 68	ASME III, 71
Recirc. gate valve	2	B31-F023	Nov. 69	NPVC, 68	ASME III, 71
Recirc. gate valve	4	B31-F031	Nov. 69	NPVC, 68	ASME III, 71
Recirc. piping	2	B31-G001	June 70	B31.7, 69	ASME III, 71S
Recirc. flow element	2	B31-N013	Jan. 71	B31.7, 69	ASME III, 71S

[a] Upgraded from 1965 ASME Code, 1969 Summer Addendum edition except for specific nozzle and attachment magnetic-particle tests (refer to AEC Question 2.5.1 and Edison PSAR Amendment 11 dated September 15, 1970).

[b] ASME III = ASME Boiler and Pressure Vessel Code Section III.

[c] S = Summer Addendum to the Code.

[d] W = Winter Addendum to the Code

[e] Pressure boundary components only

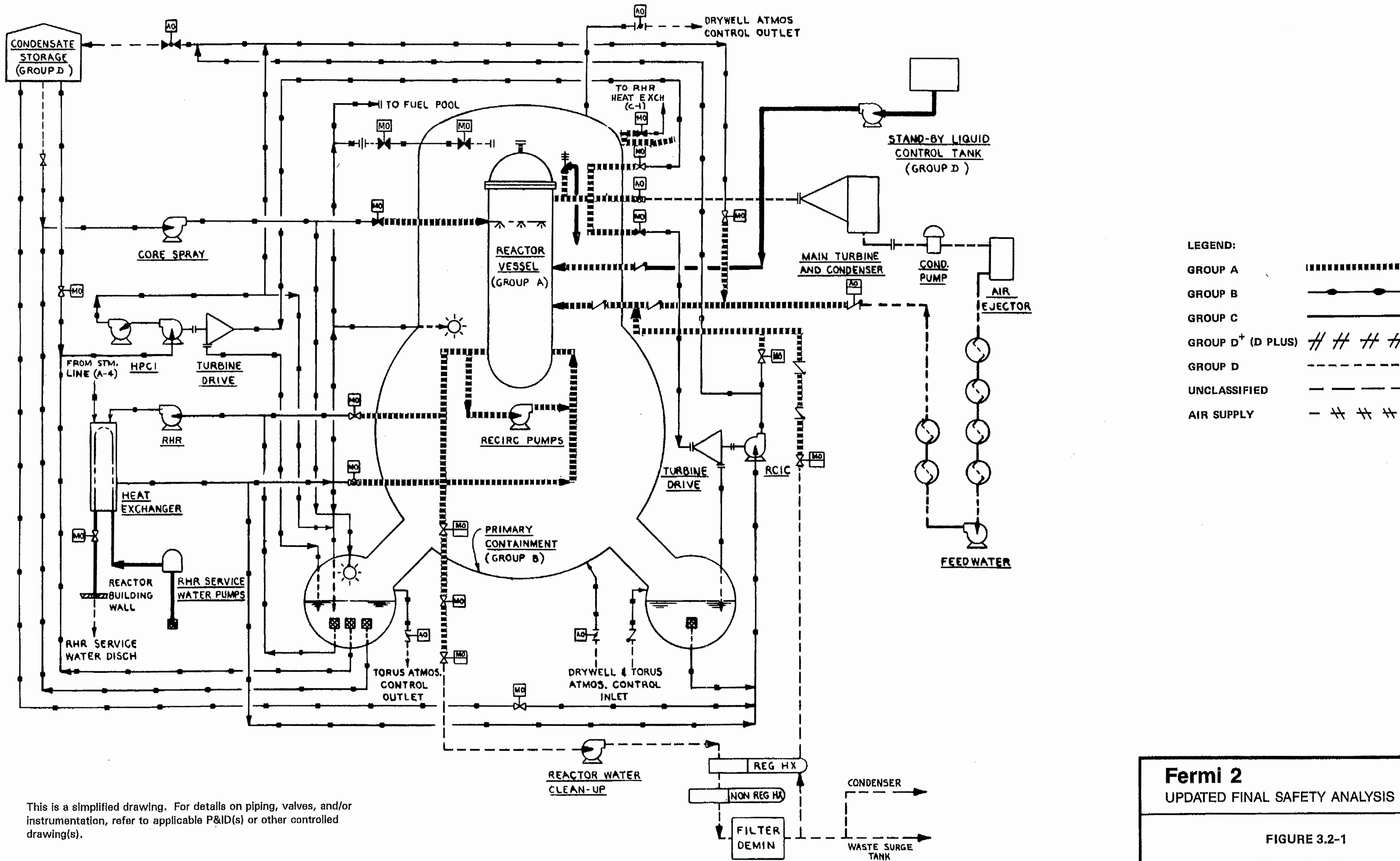
[f] The purchase order was revised on 4/18/77 to delete Dresser as the vendor and replace it with Target Rock. The original procurement requirements and codes remained applicable.

[g] NPVC = ASME Draft Code for Pumps and Valves for Nuclear Power.

[h] B31.7 = ANSI B31.7 Code for Nuclear Power Piping.

[i] GE master part number B11-D072 is deleted

[j] Upgraded to 4th Generation Design. Cover Assembly Per ASME III, 1989.



This is a simplified drawing. For details on piping, valves, and/or instrumentation, refer to applicable P&ID(s) or other controlled drawing(s).

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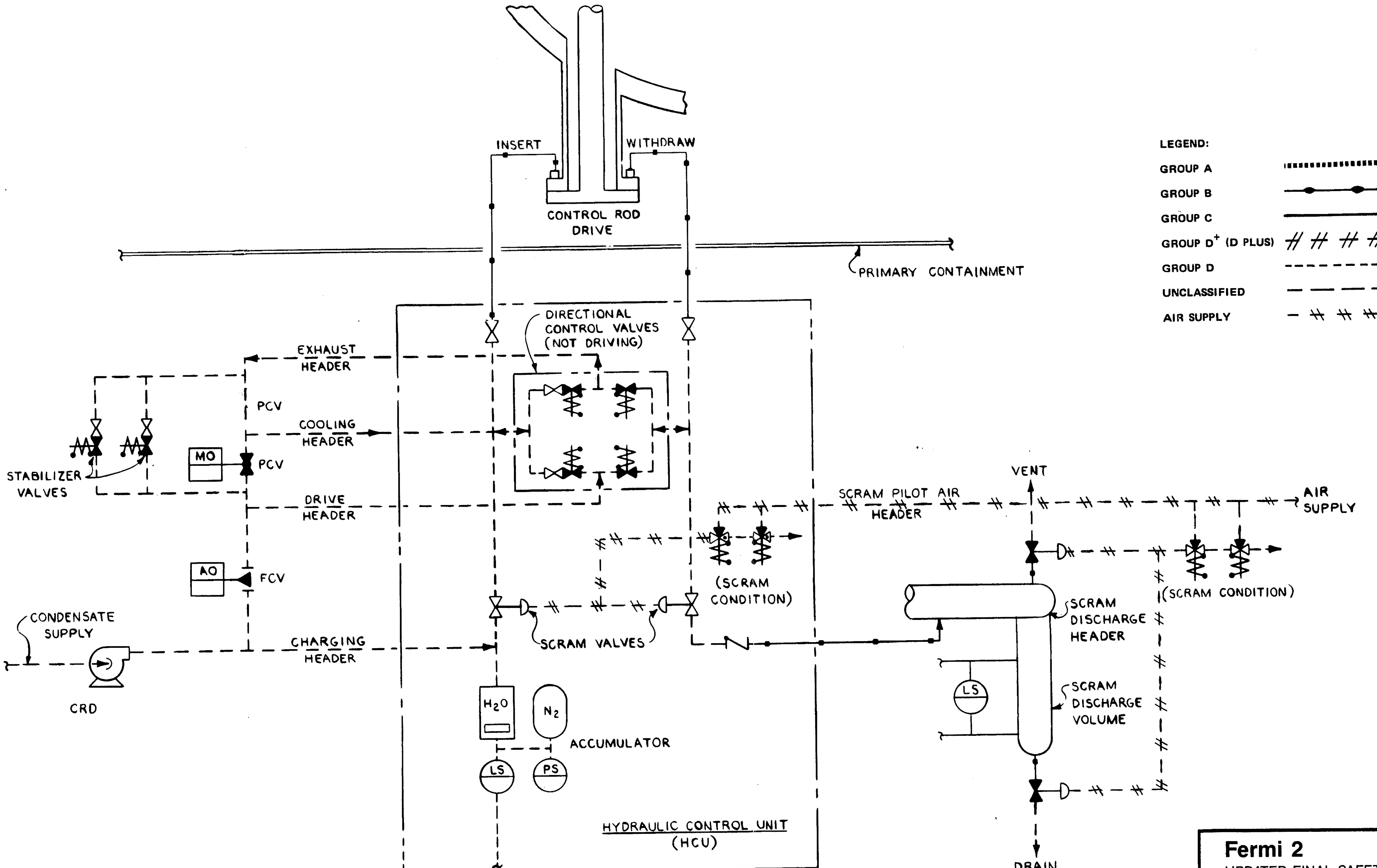
FIGURE 3.2-1

GROUP CLASSIFICATION DIAGRAM NUCLEAR BOILER SYSTEM



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.2-2

GROUP CLASSIFICATION DIAGRAM
CONTROL ROD DRIVE SYSTEM

3.3 WIND AND TORNADO LOADINGS

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

For service load conditions, the Category I structures of Fermi 2 are designed to withstand a 90 mph fastest-mile sustained wind velocity, 30 ft above ground level. This wind velocity has a l00-year recurrence interval.

3.3.1.2 Basis for Wind Velocity Selection

The wind velocity and recurrence interval is based on ASCE Paper No. 6038 by H. C. S. Thom (Reference 1). The 90-mph velocity for the Fermi site was read from Figure 5 of this paper. Figure 3.3-1 is a reproduction of Figure 5 of ASCE Paper No. 6038. This paper is referenced in the ANSI A58.1-1972 Code (Reference 2) for selecting basic wind speeds for locations in the United States.

The design of 90 mph is conservative for the Fermi 2 site when compared to measured values recorded at Detroit City Airport and Toledo, Ohio. As discussed in Subsection 2.3.1, the fastest-mile wind recorded was 77 mph at Detroit City Airport.

3.3.1.3 Vertical Velocity Distribution and Gust Factor

The relationships to determine the vertical velocity distribution of the wind are obtained from page 1139 of ASCE Paper No. 3269 (Reference 3) for coastal areas and are as follows:

for $V_{30} \leq 60$ mph

$$V_z = V_{30} \left(\frac{z}{30}\right)^{0.3} \tag{3.3-1}$$

for $V_{30} > 60$ mph

$$V_z = V_{30} \left(\frac{z}{30}\right)^{x} \tag{3.3-2}$$

where

V_{30} = basic wind velocity (mph) at height 30 ft above ground level (grade)

x = factor which varies from 0.3 when V_{30} = 60 mph to 0.143 when V_{30} = 130 mph (Reference 3)

V_z = wind velocity (mph) at height (z) above grade

z = distance above grade in feet

Thus, at heights between 100 and 150 ft above grade, the height of the upper portion of the reactor building, the wind velocity is calculated to be 123.5 mph. Gust factors have also been determined by the methods given on pages 1124 through 1198 in ASCE Paper No. 3269 (Reference 3). For all Category I structures, the gust factor varies linearly from 1.1 at grade level to 1.0 at 400 ft. However, a gust factor of 1.1 was used for the full height of both the reactor/auxiliary building and the residual heat removal (RHR) complex except for the blow-away siding design during the design tornado, where a factor of 1.0 was used.

3.3.1.4 Determination of Applied Forces

The design wind velocity specified in Subsection 3.3.1.1 is translated into an equivalent static pressure according to the provisions outlined on pages 1150-1151 in ASCE Paper No. 3269 (Reference 3). The dynamic pressure is the product of one-half the air density and the square of the resultant design velocity, and represents the kinetic energy per unit volume of moving air. For standard air and velocity, V_z, in mph, pressure in pounds per square foot is given by

$$q = 0.002558\, V_z^2 \qquad (3.3\text{-}3)$$

The equivalent static pressure to be applied to the structure is given by

$$p = q \times C_D \qquad (3.3\text{-}4)$$

where

p = average pressure, pounds per square foot

C_D = average pressure coefficient

q = dynamic pressure, pounds per square foot

Positive and negative average pressure coefficients of 0.9 and 0.5, respectively, which include the appropriate shape and drag coefficients, are applied to the walls of rectangular flat-topped structures. An average pressure coefficient of 0.8 is used for roof suction. Table 3.3-1 lists the equivalent static pressure as a function of height above grade for rectangular flat-topped Category I structures.

3.3.2 Tornado Loadings

If tornadic winds traverse the site, the reactor is capable of being shut down and secured in a safe-shutdown mode. Some minor superstructure damage could be incurred by the reactor/auxiliary building. Damage could occur to other nonseismic structures suchas the turbine building, condensate storage tanks, and incoming power lines, without affecting the ability to shut down the reactor and maintain integrity of containment and essential heat removal systems during and following a tornado that might traverse the site. Simultaneous damage to all of these items is not expected. However, as a design objective, the reactor is capable of being safely shut down and maintained in a safe-shutdown condition with the loss of all such nonseismic structures. Components that directly affect the ultimate safe shutdown of the plant are located either under the protection of reinforced concrete or underground.

Where structural failure could affect the operation and functions of the primary containment and reactor primary system, and for structures affecting equipment necessary for safe shutdown of the reactor, tornado effects are considered in the design of these structures.

3.3.2.1 Applicable Design Parameters

For extreme environmental load conditions, the Category I structures housing the systems required for a safe shutdown of the plant in the event of a tornado are designed to withstand the effects of a tornado by providing either sufficiently strong structures or appropriate venting. With the exception of the 4160-V RHR cable vaults, manholes, and ductbanks, the design parameters of the Fermi 2 design-basis tornado are

a. A rotational wind velocity of 300 mph

b. A translational wind velocity of 60 mph

c. An external pressure drop of 3 psi at the rate of 1 psi/sec.

Although the Fermi 2 design was established before the issuance of Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants (April 1974), the above parameters compare well with this guide. The rotational and translational wind velocities given in the guide are slightly different (290 mph and 70 mph, respectively). However, the total maximum velocity is the same. Likewise, although the rate of pressure drop given in the guide is faster (2 psi/sec), the magnitude of the pressure drop is the same.

The tornado missile design of the 4160-V RHR cable vaults and the manholes and ductbanks between these cable vaults and the Reactor/Auxiliary building cable vaults is based on criteria established in Regulatory Guide 1.76, Revision 1 (March 2007) and tornado missile analysis methods specified in NUREG-0800, Standard Review Plan 3.5.3 Revision 3, dated March 2007.

The Design Basis Tornado wind characteristics in accordance with Regulatory Guide 1.76 Revision 1 (March 2007) are as follows:

a. A maximum wind velocity of 230 mph

b. A maximum rotational wind velocity of 184 mph

c. A translational wind velocity of 46 mph

d. An external pressure drop of 1.2 psi at the rate of 0.5 psi/sec

Tornado wind velocity and pressure drop corresponding to the tornado generated missiles is used to evaluate the adequacy of the 4160-V RHR cable vaults and the manholes and ductbanks between these cable vaults and the Reactor/Auxiliary building cable vaults.

3.3.2.2 Determination of Forces on Structures

All tornado wind pressure and differential pressure effects are considered static in application since the natural period of the building structures and their exposed elements are short compared with the rise in time of the applied design pressures.

The tornado wind rotational velocity varies linearly with radius (r) from zero at the center to a maximum at a distance R_c from the center and inversely with r as r increases beyond R_c. That is

$$v = \frac{c}{r} \text{ for } r > R_c \text{ (Reference 4)} \qquad (3.3\text{-}5)$$

where

V = velocity, fps

r = radial distance from center of tornado, ft

c = a constant for tornadoes empirically established at 10^5 ft^2/sec (Reference 4)

At the design rotation velocity of 300 mph (440 fps), r = R_c and is 227 ft, as determined from the above equation. The resulting rotational velocity is shown in Figure 3.3-2. The total velocity profile is obtained by algebraically adding 60 mph translational velocity to the rotational velocity profile. This is also shown in Figure 3.3-2. This results in a maximum velocity of 360 mph on the strong side of the tornado and a maximum velocity of 240 mph on the weak side. The rotational velocity distribution also varies according to the elevation above ground level reaching a maximum 300 mph at a point approximately 225 ft above ground (Reference 5), which is approximately 75 ft above the top of the reactor building. However, no reduction in rotational wind velocities is used, and therefore the analysis is conservative.

The maximum differential pressure of 3 psi occurs as a result of vortex action at the center of the tornado. Differential pressure as a function of cyclonic radius for the model (Reference 4) is given by the expressions:

$$p(r) = \rho V_c^2 \left[1.0 - 0.5\left({}^{r}/_{R_c}\right)^2\right] \text{ for } r < R_c \qquad (3.3\text{-}6)$$

$$p(r) = 0.5\, \rho V_c^2 \left({}^{R_c}/_{r}\right)^2 \text{ for } r \geq R_c \qquad (3.3\text{-}7)$$

where

p(r) = pressure drop, lb/ft^2

ρ = mass density of air = 0.002376 $lb\text{-}sec^2/ft^4$

V_c = maximum rotational velocity = 440 fps

The standard value of air density is assumed because, although the air density is expected to be reduced, its effect may be offset by the presence of dust. The pressure diagram resulting from the evaluation of the above equations is presented in Figure 3.3-3.

The tornado velocity is converted to an equivalent static pressure according to the procedures given in ASCE Paper No. 3269 (Reference 3), conservatively considering no variation with height and a gust factor of 1.0. This pressure is then combined with the barometric pressure.

When a flat object is placed in a tornado wind, the load on it is equal to the sum of the windward pressure and leeward pressure as the barometric pressure drop on both faces cancels out. However, when an unvented, enclosed object is placed in a tornado wind, the total windward pressure equals the leeward velocity pressure plus the barometric pressure drop. The total pressure diagrams for the vented and unvented cases are shown in Figure 3.3-4.

Most structures are unvented. However, the reactor/auxiliary building above the fifth floor is designed to vent as discussed in Subsection 3.3.2.3.2.

Category I structures have been designed to withstand the effects due to simultaneous action of tornado wind velocity pressures, atmospheric pressure drop, and a single tornado-generated missile.

The 4160-V RHR cable vaults, manholes, and ductbanks have been designed to simultaneous action of tornado wind velocity pressure, pressure drop, and a single tornado-generated missile in accordance with Regulatory Guide 1.76 Revision 1 (March 2007) and NUREG-0800, Standard Review Plan 3.5.3 Revision 3, dated March 2007.

Design tornado loads were found by making use of the following expression:

$$W_T = (p + C_p q)A + M_T \quad (3.3\text{-}8)$$

where

W_T	=	design tornado load
p	=	atmospheric pressure drop
q	=	wind velocity pressure
C_p	=	pressure coefficient described in Subsection 3.3.1.4
A	=	exposed area under consideration
M_T	=	effects of single tornado-generated missile

Both overall (overturning, sliding) and local effects of tornado-generated loads W_T have been investigated. Structure under consideration was placed at various locations in the tornado wind field, and the governing combination of p and q was selected for each particular effect.

The effects of tornado missiles are local in nature. Accordingly, they have been taken into account in the design of structural elements and disregarded in case of overturning and sliding.

3.3.2.3 Ability of Category I Structures To Perform Despite Failure of Structures Not Designed for Tornado Loads

3.3.2.3.1 General

Superficial structural damage can be tolerated by the reactor/ auxiliary building and the RHR complex. Nonseismic structures such as the turbine building, condensate storage tanks, and incoming power distribution system can withstand some structural damage without affecting the safe-shutdown capabilities of Category I structures and equipment. As indicated in Table 3.3-2, systems required for a safe shutdown of the reactor are housed in well-protected structures.

3.3.2.3.2 Reactor/Auxiliary Building Above the Fifth Floor (Blow-Away Siding)

The panels and roof above the refueling floor are designed to release (blow away) during the design-basis tornado, as described in Section 3.8, while the remainder of the exposed frame is designed for the full tornado load. The design and analyses of these panels under tornado loadings have been presented to and accepted by the AEC by Reference 6. Further design requirements imposed on this portion of the reactor building are as follows:

a. For the design-basis tornado, and assuming that panels equivalent to 10 percent of the surface area of the panels are caught and do not release, the stress levels of the structural steel frame of this portion of the reactor building must not exceed 95 percent of the yield stress

b. With all siding in place, the reactor building will be capable of withstanding a 200-mph tornado wind at stress levels limited to 95 percent of the yield strength of the steel.

This additional load limitation provides a range of pressure within which the siding is designed to blow off.

For the reactor building above the refueling floor, the maximum load on the projected area of the exposed steel supporting frame with 10 percent of the siding is 464 lb/ft^2 during the design- basis tornado, while the maximum load on the structure for the 200-mph tornado with the siding all intact is -98 lb/ft^2 (suction) on the leeward side and 46 lb/ft^2 on the windward side. Surface pressure for the remainder of the reactor and auxiliary buildings and the RHR complex is included in the loading combinations considered in Section 3.8.

A postulated explosion of the 20,000-gallon liquid hydrogen tank at the HWC gas facility may also cause some damage to the roof and siding of the reactor building above the refueling floor. The hydrogen tank has been located sufficiently far from the reactor building to assure that blast forces from an explosion would be less than the pressure forces from a design basis tornado. Therefore, the tornado analysis bounds the effects of a hydrogen-tank explosion at the roof and siding above the refueling floor.

3.3.2.3.3 Fuel Pool Exposure

In the unlikely event of a tornado of sufficient severity to cause the panels above the refueling floor to release, the spent-fuel pool would be exposed. This concern was identified by the AEC as Post Construction Permit Open Item No. 9. The AEC had requested additional spent-fuel protection, but agreed later that no additional protection was required (Reference 7).

With the siding blown off during the design-basis tornado, the refueling floor would be exposed. However, based on the GE publication, "Tornado Protection for the Spent Fuel Storage Pool," APED-5696, Class-I (November 1968), there is no credible mechanism by which a significant amount of water could be sucked from the fuel pool by a tornado. The fuel stored in the spent-fuel storage pool would be protected by approximately 22 ft. 6 in. of water covering the tops of the fuel storage racks and by the racks themselves.

3.3.2.3.4 Crane and Crane Support Structures

The reactor building superstructure steel frame and anchor bolts are designed for the design-basis tornado described in Subsection 3.3.2.2 at a stress level of 95 percent of yield. Therefore, there would be no danger of failure of the columns supporting the crane bridge and trolley. Moreover, the crane and trolley are restrained from motion in the horizontal direction when not in use by "dead-man" safety pins.

The crane is provided with electrically operated locking bars effectively connecting the unloaded crane to the runway when not in use and capable of withstanding a design-basis tornado wind force of 410 lb/ft^2 due to a 360-mph resultant wind velocity. Restraints are provided on the crane bridge and trolley to prevent either from leaving their respective rails due to horizontal and vertical displacement in the event of a design-basis earthquake. For further details, see Subsection 3.7.3.18. Vent holes are provided in girders and other enclosed structures such as trolley frame, trucks, and electrical cabinets of such size to withstand an atmospheric pressure reduction of 1.0 psi/sec, maximum reduction of 3 psi, due to the design-basis pressure transient.

3.3.2.3.5 Other Venting

Because of the depressurization that can occur when the very low- pressure area within the funnel of a tornado engulfs a structure, structures housing equipment necessary for safe shutdown must either be designed for the depressurization, or be vented. In the Fermi 2 design, all such structures, with the exception of the steam tunnel, are designed for the depressurization. Venting of the steam tunnel is accomplished by blowout panels that are designed to release in the event of external depressurization.

3.3.2.3.6 Residual Heat Removal Complex

The RHR complex cooling towers are exposed to the flight of potential missiles. However, as discussed in Section 3.5, the probability of damage is negligible.

All systems contained in the RHR complex are divided into two separate and redundant groups, Division I and Division II, with a thick wall between the two divisions that serves as a missile barrier. This further reduces the probability of safety-related systems not being able to perform their functions. The RHR complex is described in Section 9.2.

3.3.2.3.7 Tornado Failure of Nonseismic Structures

Protection against the possibility of failure of Category I structures due to the tornado-induced failure of nonseismic structures is provided by the inherent structural integrity of the Category I structures to mitigate other postulated, equally severe events. Further, the site building arrangement (see Figure 1.2-5), as well as the history and probability of tornadoes likely to occur at the site, minimizes the probability of a tornado engulfing a nonseismic structure.

3.3.2.3.7.1 Probability of Occurrence

The probability of a design-basis tornado occurring at the 1000-acre Fermi site is 4.075×10^{-5}, or a recurrence interval of 24,500 years (Subsection 2.3.1.3.2). This probability is significantly further diminished by factoring in the horizontal surface area occupied by the Fermi 2 Category I structures - approximately 1 acre.

3.3.2.3.7.2 Category I - Nonseismic Structure Arrangement

Category I structures are located with respect to nonseismic structures in a manner that minimizes, if not eliminates, the probability of failure of a Category I structure due to the tornado-induced failure of the nonseismic structure. The impingement of a nonseismic structure upon a Category I structure, or the generation of missiles from a nonseismic structure, are the only unlikely events that could be postulated to occur.

Some temporary trailers and miscellaneous construction material may be stored near Category I structures to support plant outages.

There is a refuel outage building adjacent to the south wall of the reactor building and a small prefabricated metal building housing nitrogen inerting equipment immediately west of the reactor building. In addition, approximately 30 ft west of the RHR complex, there is a 345-kV switchyard. To the north is a reinforced-concrete cooling tower.

To the east is the turbine house-radwaste building, which consists of a reinforced-concrete structure and steel superstructure. The failure of other nonseismic structures further eastward of the turbine house-radwaste building would not affect the Category I structures, as missiles or impingement caused by their failure would affect only the reinforced-concrete turbine house. The turbine house can absorb energy resulting from either another nonseismic structure failure, or its own partial failure.

3.3.2.3.7.3 Turbine Building

The improbable direct strike of a tornado on the turbine building could result in a worst-case event where portions of the metal siding and support columns and girders deform and impinge against the thick, heavily reinforced concrete wall of the adjacent reactor/auxiliary building (see Figure 1.2-20). This impingement could result in superficial structural damage, but would not prevent the reactor from being brought into a safe-shutdown mode.

The collapse of the turbine building roof would not affect the operation of any safety-related equipment. There is no safety- related equipment in the turbine building that would be required to operate if the roof were to collapse.

A postulated explosion of the 20,000-gallon liquid hydrogen tank at the HWC gas facility may also cause some damage to the roof and siding of the turbine building above the operating floor. The hydrogen tank has been located sufficiently far from the turbine building to assure that blast forces from an explosion would be minimized, and that stop and control valve closure inputs to the reactor protection system would remain functional. However, even if trip function (direct scram) is postulated to fail, other diverse signals, such as reactor pressure and high neutron flux, will scram the reactor. Therefore the consequences of a turbine trip with a postulated failure of direct scram are bounded by the design basis earthquake event.

3.3.2.3.7.4 Category I Buildings

The Fermi 2 Category I buildings are designed for the postulated severe loading conditions in appropriate loading combinations (Section 3.8). Their construction generally consists of thick, heavily reinforced concrete walls. A spectrum of missiles was selected, approved by the NRC, and used as a design basis for these buildings. As discussed in 3.3.2.3.7.2, the arrangement of nonseismic structures with respect to Category I buildings minimizes the effect of a nonseismic structure failure on Category I buildings.

3.3 WIND AND TORNADO LOADINGS

REFERENCES

1. H. C. S. Thom, "New Distributions of Extreme Winds in the United States," Journal of the Structural Division, ASCE, Vol. 94, (ST 7), pp. 1787-1801, July 1968.
2. ANSI A581-1972, Building Code Requirements for Minimum Design Loads in Buildings and Other Structures.
3. Task Committee on Wind Forces, Committee on Loads and Stresses, Wind Forces on Structures, Final Report, Paper No. 3269, Vol. 26 Transactions, ASCE, 1961.
4. J. M. McLaughlin, Sargent & Lundy, Design of Nuclear Power Plants for Tornados, Tornado Phenomenology and Related Protective Design Measures, Conference at the University of Wisconsin, April 26-28, 1970, page 5.
5. J. A. Dunlop, and K. Wiedner, "Nuclear Power Plant Tornado Design Considerations," Journal of the Powers Division, Proceedings of the ASCE, pp. 407-417, March 1971.
6. Detroit Edison Company, Technical Report, PSAR Open Item No. 7, "Tornado Winds-Refueling Floor Siding and Superstructure," May 8, 1973. Submitted to and approved by the AEC, Docket No. 50-341, per letter of August 2, 1973 from AEC to Edison.
7. Letter from AEC to Edison, June 11, 1974 responding to Edison letters EF 2-l8679, August 9, 1973, and EF 2-19171, August 14, 1973.

TABLE 3.3-1 EQUIVALENT STATIC WIND PRESSURE

Used in the Design of Category I Structures

Height Above Grade (ft)	Positive Pressure ($1b/ft^2$)	Negative Pressure ($1b/ft^2$)	Total Pressure ($1b/ft^2$)
0 to 50	22.8	12.7	35.5
50 to 100	34.9	19.4	54.3
100 to 150	42.5	23.6	66.1
150 to 200	47.9	26.6	74.5

TABLE 3.3-2 SYSTEMS REQUIRED TO ATTAIN SAFE SHUTDOWN IN THE EVENT OF A TORNADO

Systems	Location
Emergency equipment cooling water system	Reactor/auxiliary building
Emergency equipment service water system	RHR complex
Reactor core isolation cooling system	Reactor/auxiliary building
Emergency diesel generator system	RHR complex
Residual heat removal system (shutdown cooling)	Reactor/auxiliary building
RHR service water system	RHR complex, reactor/auxiliary building
Reactor protection system	Reactor/auxiliary building

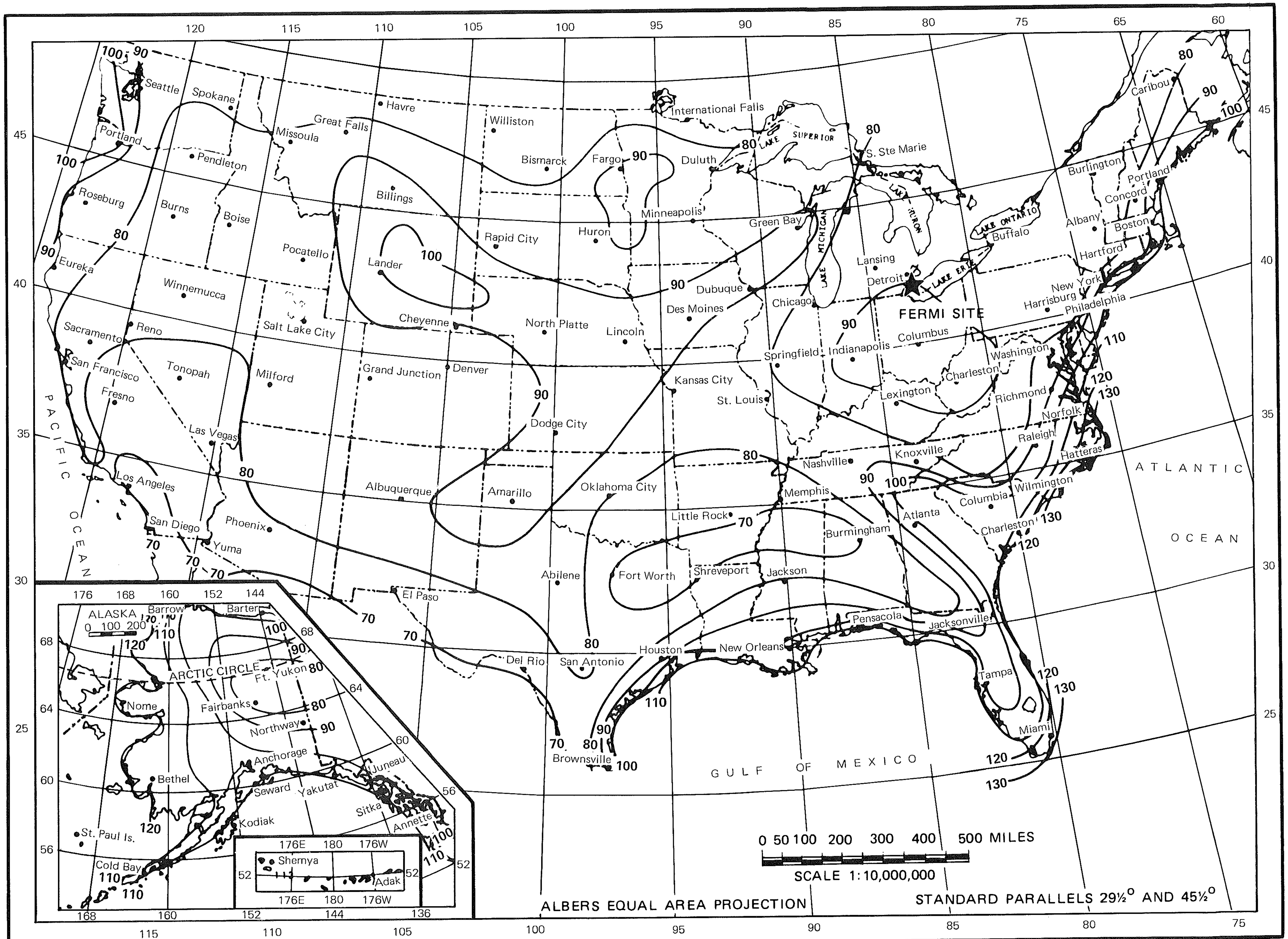

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FIGURE 3.3-1

ISOTACH 0.01 QUANTILES IN MILES PER HOUR ANNUAL EXTREME – 30 FT ABOVE GROUND 100-YEAR MEAN RECURRENCE INTERVAL

REFERENCE: ASCE PAPER NO. 6038, FIGURE 5, JULY, 1968 (REFERENCE 1)

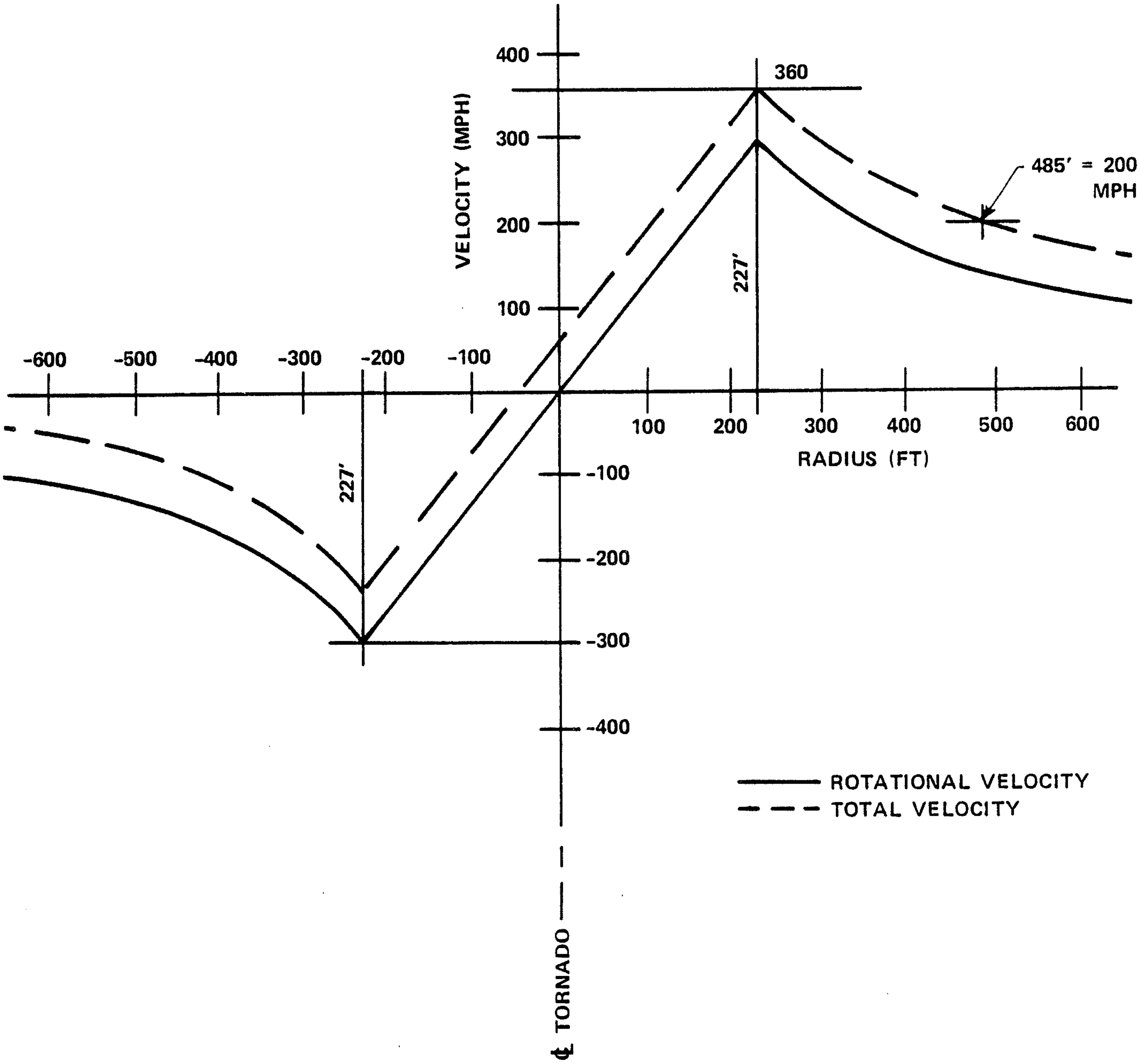

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FIGURE 3.3-2

TORNADO VELOCITY VERSUS DISTANCE

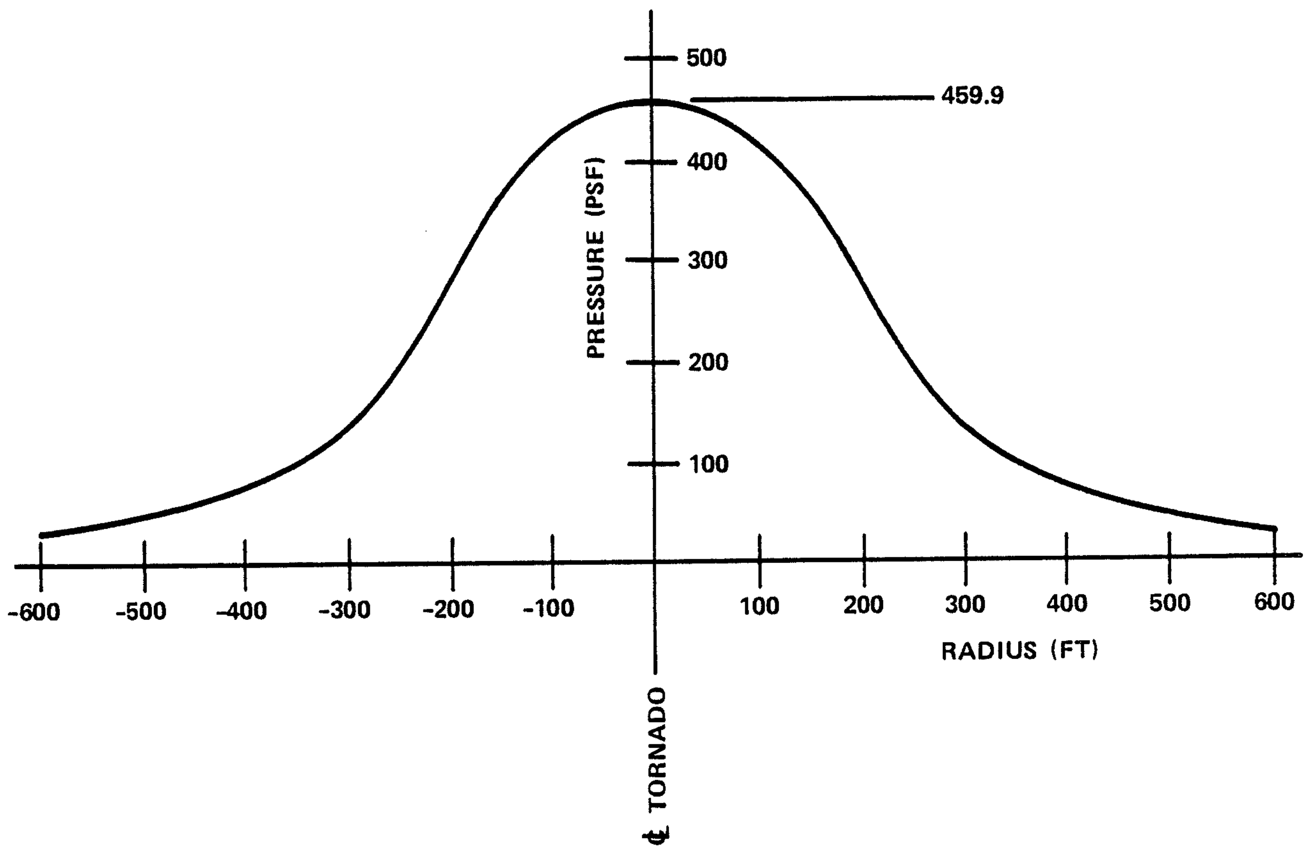

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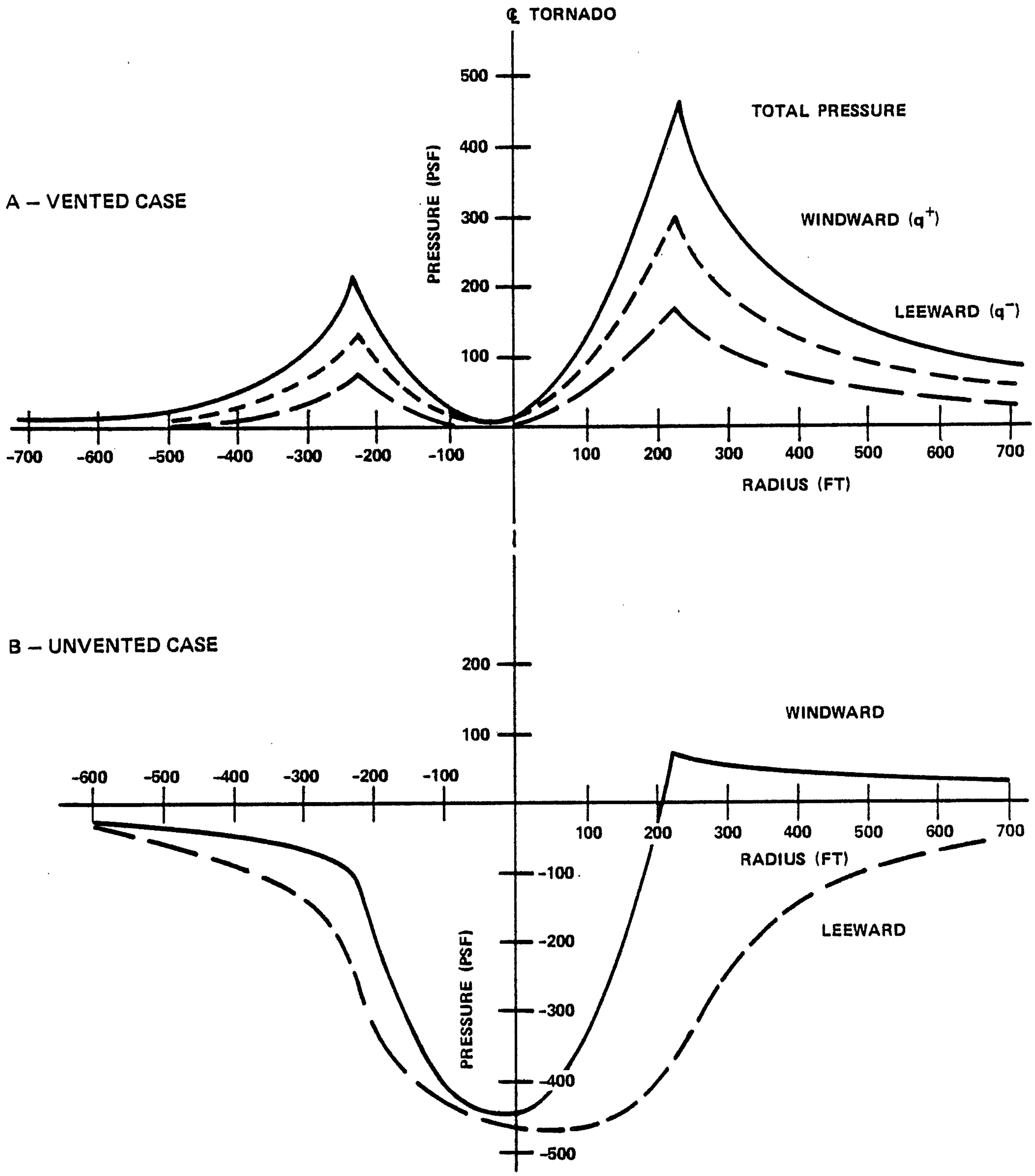
FIGURE 3.3-3

TORNADO PRESSURE DROP VERSUS DISTANCE



REFERENCE: TASK COMMITTEE ON WIND FORCES, COMMITTEE ON LOADS AND STRESSES, WIND FORCES ON STRUCTURES, FINAL REPORT, PAPER NO. 3269, VOL. 26 TRANSACTIONS, ASCE 1961.

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FIGURE 3.3-4

TORNADO WIND PRESSURE – VENTED AND UNVENTED

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 Flood Elevations

From consideration of several types of hypothetical flooding, it was found that the maximum stillwater elevation that could occur at the site is 586.9 ft (New York Mean Tide, 1935), and results from the probable maximum meteorological event (PMME), with a storm path along the axis of Lake Erie (N 67.5° E). Flood design is discussed in Subsection 2.4.2 and the maximum elevation determination in Subsection 2.4.5.

All Category I components are protected from the adverse effects of the maximum flood elevation by their location within reinforced-concrete Category I structures, as described in this section and in Chapter 2.

As stated in Subsection 2.4.2, Fermi 2 Category I structures and components are designed against flooding up to a minimum elevation of 588 ft, or 1.1 ft above the maximum stillwater elevation.

3.4.2 Phenomena Considered in Design Load Calculations

Category I structures and components are designed for the static and hydrodynamic forces associated with wind-generated waves as specified in Subsection 2.4.5. The effects of a tsunami are not considered because the site is located in an area designated as having potentially minor seismic activity. Any seismic disturbance would be local and would result in only minor excitations in Lake Erie. Tsunami considerations are discussed in Subsection 2.4.6.

3.4.3 Flood Force Application

The pressure induced by the maximum stillwater elevation is considered to be hydrostatic. A lateral pressure distribution below the flood line on the walls of the Category I structures is determined. From this, the uplift pressure on the Category I structure basement slabs and flotation potential is then calculated. This pressure is included in the load combinations considered in the design of the slabs. Pressures induced by wave action are discussed in Subsection 2.4.5.

3.4.4 Flood Protection

Flood protection for Fermi 2 Category I structures and components includes waterproofing the structure, designing the structure to withstand the hydrostatic and hydrodynamic forces associated with flooding, maximum usage of watertight seals and penetrations below the maximum flood elevation, and locating the Category I components within the reinforced-concrete Category I structures.

3.4.4.1 Reactor Building Structure

The Category I reactor/auxiliary building, which houses safety-related systems and components, is designed against flooding to Elevation 588.0 ft, or 1.1 ft above the PMME stillwater flood elevation of 586.9 ft. All doors and penetrations through the outside walls below the design flood elevation are of watertight design.

As stated in Subsection 2.4.2.2.2, there are only a few essential penetrations in the exterior walls of the reactor/auxiliary building. All of these penetrations below Elevation 588.0 ft are watertight.

The presence of the turbine building prevents waves and wave runup above the sill elevations on the east wall of the reactor/ auxiliary building, thereby preventing flooding of the building. The south wall of the reactor/auxiliary building has two large openings and several waterproofed pipe-sleeved openings. The large openings are an air-locked rail car door and an air-locked personnel door. Both of these air-locked doors are completely waterproofed to preclude wave runup flooding.

In addition, all watertight doors have signs on both sides stating that the door is to be secured closed except for immediate use.

The several watertight sleeve openings, the walls of the building, and the watertight doors are designed to withstand the hydrostatic head of the maximum flood level. Maximum wave effects and forces are discussed in Subsection 2.4.5.4.

Leakage is not expected through the several watertight access openings and the waterproofed sleeved openings in the reactor/ auxiliary building.

The walls of the reactor/auxiliary building are waterproofed below the finished grade elevation of 583.0 ft.

Waterstops on all construction joints and water seal rings on all penetrations are provided on all openings below the maximum flood level. The waterstops are joined to form a continuous watertight seal. Joint preparation and joint sealants are in conformance with the recommendations and the guidelines of American Concrete Institute (ACI) standards. All work is inspected by qualified personnel to ensure that leakage is kept to a minimum.

All interior floor drain systems inside the reactor/auxiliary building are independent of the yard storm drainage system, and therefore no potential water backflow into the structure is anticipated during the design flood condition. Shore protection is not required to preclude flooding of this structure.

3.4.4.2 Residual Heat Removal Complex Structure

The residual heat removal (RHR) complex is watertight to Elevation 590.0 ft. There are no openings on the north, south, and west walls. All pipe and electrical penetrations on the east wall below Elevation 590.0 ft are waterproofed. However, if any amount of leakage should occur, it would go directly into an RHR Complex compartment. Then, it is pumped to the Circulating Water Reservoir.

The remaining openings to be considered would be the access doors on the east wall. These doors are normally closed and locked, and have their thresholds at Elevation 590.0 ft. They are of steel construction and are shielded behind reinforced-concrete missile walls. The insignificant amount of runup above the flooded elevation of 586.9 ft may find its way through the door threshold and door jambs, at Elevation 590.0 ft, and be diverted into the floor drain system in the building. The leakage through the gaps of the doors could never exceed the drain capacity of the Elevation 590.0-ft floor drain system. The structure is also designed to withstand the wave action associated with this flooding. (Refer to Subsection 2.4.2.2.3, Residual Heat Removal Complex Flood Criteria.)

The RHR complex is described in Subsection 9.2.5.

The RHR complex reservoir is floodproof. The reservoir overflow is a nonsiphon floodproof post. All active equipment that could be damaged by water (pump motors, switchgear, diesel generators) is located above the maximum water flood level.

Moreover, all interior floor drains are independent of the yard storm drain system. Thus, there is no potential for backflow flooding. Walls of the RHR complex below grade level are watertight.

3.4.4.3 Category I Yard Structures

The Category I piping and electrical ducts between the RHR complex and the reactor building are below the site flood elevation of 586.9 ft during the PMME. The RHR supply, RHR return, and emergency equipment service water pipelines to both divisions will continue to function during the flood.

There are two sets of Category I ductbanks between the RHR complex and the Reactor/Auxiliary building, with a Division I and Division II ductbank in each set. In each case, the buried cable ducts between the RHR complex and the Reactor/Auxiliary building provide adequate cable separation to maintain independence of redundant circuits.

The first set of ductbanks was installed during plant construction. The physical separation of the two redundant, below-grade circuits is 30 ft at the point the cable ducts leave the southeast corner of the reactor building. The ducts make a sweeping bend, with a minimum separation of 20 ft between them. After the bend, the ducts parallel the reactor building in a westerly direction, with 24 ft of separation. This separation is constant until the ducts pass under the rail car air lock, where the separation widens until the ducts enter (still below grade) the RHR complex.

Each circuit is separately housed in a cast-in-place, rectangular reinforced-concrete duct. The duct is covered by successive layers of compacted-rock fill placed up to the finished site nominal grade of 583 ft. The duct runs vary in elevation from 573 ft minimum to 580 ft maximum. Since the maximum ground-water elevation is 576 ft, the cables are not specifically designed for continuous underwater service. For low voltage power, control and instrumentation cables, there is no long term mechanism for water related insulation degradation due to lack of voltage stressor or a credible common mode failure mechanism. Therefore, low voltage cables perform their design functions while their external surface remains continuously wetted due to surrounding water. 4160-V essential power circuits are not routed within these ductbanks.

The second set of 4160-V RHR cable vaults, ductbanks and associated manholes is installed above the maximum ground water elevation of 576.0 ft with ducts sloped to the manholes, such that circuits contained are not subject to continuous wetting. These are also cast-in-place, rectangular reinforced concrete ductbanks, but are located with the ductbank top approximately six inches below the surface and manhole covers at grade level. 4160-V essential power circuits are routed within these ductbanks. Although the manholes and cable vaults may be subject to flooding during the duration of the PMME, the 4160-V essential bus tie cables are qualified for wet conditions in excess of six months, which is greater than this duration.

The minimum elevation for cable termination in either the RHR complex or the reactor building is 588.7 ft, which is above the site probable stillwater elevation of 586.9 ft.

3.4.4.4 Internal Flood Protection

3.4.4.4.1 General

Category I and nonseismic structures are provided with equipment and floor drainage systems designed to collect and remove all waste liquids from their points of origin to a suitable disposal area in a controlled and safe manner. All collected liquid waste is routed to sumps in each of the respective buildings, where it is allowed to accumulate for periodic discharge to the radwaste building for treatment. Abnormal sump water levels (high-high or low-low) are annunciated in the control room. The locations of the sumps and sump pumps, the capacity of each sump and sump pump, and the design bases for equipment and floor drainage systems of each building are described in Subsection 9.3.3.

To prevent backflow flooding through the equipment and floor drainage systems, the following considerations were included in the system design:

a. Independence of building systems to negate the possibility of abnormal occurrences in one building from affecting normal operation in other buildings

b. Check valves and manual isolation valves in each sump pump discharge line to prevent backflow to the sump.

c. Redundant check valves and a manual isolation valve located in both the equipment drain and floor drain 6-inch transfer lines near the secondary containment boundary to prevent backflow into secondary containment.

3.4.4.4.2 Design Analyses

The potential for backflow flooding through the equipment and floor drainage systems due to the PMME flood is evaluated as follows:

a. The PMME is postulated to have occurred and the associated flooding in the turbine building will be consistent with the site water-level accumulation during the incident

b. Flood water would enter the equipment or floor drainage piping system through the collector tanks and their overflow lines in the radwaste building. These overflow lines are provided for routing the collection tank overflow to the radwaste building sumps. The collection tanks are in the basement (approximate Elevation 557 ft) of the radwaste building. As the floodwaters rise, the collection tanks would be filled through the overflow line and the system piping would be backfilled to the check valves in the 6-inch transfer lines.

c. Redundant check valves and a manual isolation valve in both the floor and equipment drain transfer lines are located near the secondary containment boundary just before the pipe exits into the turbine building. The design configuration allows for periodic leak testing of the check valves and this combined with redundancy of the check valves, and the presence of a manual isolation valve ensures that no single active failure will result in backflow flooding into the reactor building such that the safe shutdown capability of the reactor through the emergency core cooling system (ECCS) would be affected.

3.4.4.5 Shore Barrier

Neither the reactor/auxiliary building nor the RHR complex depends on a shore barrier to preclude flooding of the structures.

Although the Category I structures do not require protection against flooding from wave runup, a shore barrier is included in the Fermi 2 plant design to protect other portions of the plant from wave effects. The design of the shore barrier was approved by the AEC by Reference 1. The shore barrier was designed by Dames & Moore, specialists in applied earth sciences.

The shore barrier is a rubble mound revetment with a cover of armor stone, which fronts the Fermi 2 unit as shown in Figure 2.4-22. It has a toe elevation of 572 ft, a crest elevation of 583 ft, and a lakeward-side slope of 2:1 (horizontal to vertical). The design allows for the possibility of a 6 percent to 8 percent displacement of stone during the PMME. The design of the barrier is further discussed in Subsection 2.4.5 and shown in cross section in Figure 2.4-22. The barrier preserves the integrity of the plant site fill placed to Elevation 583 ft as well as protecting the main plant portion of the site against wave forces. The purpose and design of the barrier are also discussed in Subsection 2.4.5.7.

The surveillance requirements and limiting conditions for operation of the shore barrier are contained in the Technical Requirements Manual.

3.4.4.6 Condensate Storage Tanks

The condensate storage tanks are not seismic structures. However, a seismic analysis was performed for the condensate storage tanks using the Fermi 2 design-basis earthquake with the tank in the fully loaded condition. The maximum shell stresses were found to be well within the allowable limits. Tank rupture is not anticipated. For added conservatism, a containing barrier has been built around the tanks, and modifications to the site grade have been made in the immediate vicinity of the tanks. This will prevent any of the condensate liquid from reaching the distant yard drainage system should leakage occur. The Category I structures are located approximately 600 ft west of the condensate storage tanks.

3.4 WATER LEVEL (FLOOD) DESIGN

REFERENCE

1. Letter from W. R. Butler, AEC, to C. M. Heidel, Detroit Edison, Subject: Beach Barrier Design, dated April 16, 1974

3.5 MISSILE PROTECTION

Protection against the hypothetical effects of missiles is provided in accordance with the following damage limit criteria:

a. The integrity of the containment system is maintained

b. The capability for shutdown of the reactor and maintenance of core cooling capability is maintained

c. A missile accident that is not a LOCA does not initiate a LOCA.

Where possible, missile protection is achieved through basic plant component arrangement such that, if a missile-generating failure should occur, the direction of the flight of the missile would be away from Category I structures or other critical system components. Examples of such arrangements are shown in Figure 3.5-1, Sheets 1 through 6, which show the general arrangement of piping, pumps, motor, valves, and other equipment in the drywell indicating component missile protection by separation. Where it is impossible to provide protection through selective plant layout and where the structures available do not provide sufficient missile protection, barriers are provided to prevent potential missiles from damaging critical systems and structures.

An analysis of potential missiles and the missile protection provided follows. Although it is not given in the order specified in Regulatory Guide 1.70, the information requested in the guide is presented. The reason for the change in order is to present a more comprehensive discussion of the missile protection included in the Fermi 2 design.

3.5.1 Missile Selection (Sources)

3.5.1.1 Missiles From Pressurized Equipment

3.5.1.1.1 Missiles Considered

Potential missiles from pressurized equipment that were investigated include the following:

a. Valve bonnets (large and small)

b. Valve stems

c. Thermowells

d. Vessel head bolts

e. Pieces of pipe

f. High-pressure gas cylinders.

3.5.1.1.2 Design Evaluation

Using conservative assumptions, it has been determined that the potential missiles from items a. through e. above, originating from fluid lines, cannot achieve sufficient energy to penetrate the drywell, critical system components, or missile shields to the extent that safe reactor shutdown would be impaired. An added conservatism exists because of the separation criteria and barriers described in Subsections 3.5.3 and 3.5.4. The probability of incapacitating more than one of the redundant reactor protection system (RPS) safe-shutdown and engineered safety feature (ESF) system components by a single missile is negligible. The driving force for these potential missiles is assumed to come from the kinetic energy of the water or steam.

In the event of a break in a fluid-carrying component, the velocity of the exiting fluid is determined. The drag force of the fluid that propels a missile is proportional to the product of the fluid mass density and velocity squared. By applying this drag force to each potential missile, the missile attaining the most kinetic energy is determined. Damage resulting from impact of this missile is then analyzed. Small missiles are assumed to achieve maximum fluid velocity instantly, which is conservative because a missile requires a finite time to accelerate to this velocity after being dislodged. In addition, missiles in a horizontal trajectory tend to fall out of the fluid jet. Therefore, the driving force acts for a shorter time and the missile probably achieves a velocity lower than its maximum.

High-pressure gas cylinders on the Fermi 2 site that are capable of generating potentially high-energy missiles are as follows:

a. Hydrogen gas storage cylinders

b. Service gas storage cylinders (welding gases, nitrogen, and spare breathing air)

c. Emergency breathing air cylinders

d. Oxygen and hydrogen reagent cylinders

e. Hydrogen and oxygen storage vessels at the HWC Gas Supply Facility

The hydrogen and service gas storage cylinders are located more than 300 ft from the reactor building. Any potential missiles must first pass through the first floor of the turbine building and through several concrete walls (with a combined thickness of more than 5 ft) before reaching the reactor building wall. There is insufficient energy stored in these cylinders for any potential missile to penetrate these walls.

Emergency breathing air cylinders are stored in seismically qualified storage racks located along the north wall of the reactor building ventilation room. The concrete walls of this room are sufficient to prevent any potential missiles from reaching critical locations outside of this room. Equipment inside this room can be damaged by potential missiles, but this will not prevent a safe reactor shutdown. A design-basis earthquake (DBE) will not initiate emergency breathing air cylinder damage because the cylinders are secured in seismically qualified storage racks.

The primary containment hydrogen monitors require supplies of hydrogen and oxygen to act as reagent gases. These cylinders are located adjacent to each monitor, thereby minimizing the tubing run to each instrument. The cylinders, regulators, piping, and racks are seismically designed and installed. The racks are also designed to restrain the cylinders to prevent them from becoming missiles if punctured.

Using the barrier and procedures of Subsections 3.5.3 and 3.5.4, respectively, results of the investigation showed that additional missile barriers for potential missiles from pressurized equipment are not required. With the assumption of maximum missile velocity and minimum missile energy required for perforation, the results are conservative.

The HWC gas supply facility is located approximately 1100 feet northwest of the nearest safety-related structure (the RHR Complex). The hydrogen and oxygen storage tanks and the gaseous hydrogen tube bank are designed to remain in position during the design basis earthquake. Since the site for the HWC Gas Storage Facility was chosen to provide the required separation from safety-related structures, a release from this location would not affect plant safety. Potential blast effects from tank ruptures are enveloped by the existing analyses of the design basis tornado and design basis earthquake.

3.5.1.2 Missiles From Rotating Equipment

3.5.1.2.1 Missiles Considered

Potential missiles from rotating equipment, which could require a missile barrier, include

a. High-pressure turbine rotor segment

b. Low-pressure turbine rotor segment

c. Recirculation pump or motor segment

d. Emergency diesel generator (EDG) segment.

All probable paths of flight of these potential missiles have been investigated.

3.5.1.2.2 Design Evaluation

As stated in Subsection 10.2.3, after the low pressure (LP) turbine rotor replacement during RF05, there is no design basis turbine missile at Fermi 2. The HP turbine rotor was replaced in RF07. The new HP turbine rotor, which was reviewed for overspeed capability, was found to be higher in overspeed than the maximum theoretical overspeed of the unit (LP rotors and generator). Moreover, the seventh stage blades of the HP turbine rotor are smaller in length and lighter in weight that the eighth stage blades of the LP turbine rotors. Based on this, it is concluded that the HP turbine rotor missile analysis is bounded by the LP turbine missile analysis. The HP turbine rotor and generator rotor missiles cannot completely breach their respective outer casings. The new HP and LP turbine rotors are of monoblock construction. The monoblock rotors have higher speed capability than the maximum attainable speed of the turbine generator units. Per General Electric, the supplier of the new rotors, the probability of missiles being generated is well below 10 to the -8 power.

The most substantial piece of nuclear steam supply system (NSSS) rotating equipment is the reactor recirculation system (RRS) pump and motor. This potential missile source is addressed in detail in References 3 and 4.

It is concluded in Reference 3 that destructive pump overspeed can result in certain types of missiles. A careful examination of shaft and coupling failures shows that the fragments will not result in damage to the containment or to vital equipment.

a. Low-Energy Missiles (Kinetic Energy Less Than 1000 ft-lb)

   Low-energy-level missiles may be created at motor speeds of 300 percent of rated as a result of failure of the end structure of the rotor. The structure consists of the retaining ring, the end ring, and the fans. Missiles potentially generated in this manner will strike the overhanging ends of the stator coils, the stator coil bracing, support structures, and two walls of 1/2-in.-thick steel plate. Because of the ability of these structures to absorb energy, it is concluded that missiles would not escape this structure. It is at this point that frictional forces would tend to bring the overspeed sequence to a stop

b. Medium-Energy Missiles (Kinetic Energy Less Than 20,000 ft-lb)

In the postulated event that the body of the rotor were to burst, medium-energy missiles could be created. The likelihood that these missiles would escape the motor is considered less than the likelihood of escape for the low-energy missiles described above, because of the additional amount of material constraining missile escape, such as the stator coil, field coils, and stator frame directly adjacent to the rotor

c. The Motor As a Potential Missile

Since bolting is capable of carrying greater torque loads than the pump shaft, pump bolt failure is precluded. Since pump shaft failure decouples the rotor from the overspeed driving blowdown force, only those cases with peak torques less than those required for pump shaft failure (five times rated) will have the capability of driving the motor to overspeed. When missile-generation probabilities are considered along with a discussion of the actual load-bearing capabilities of the system, it is evident that these considerations support the conclusion that it is unrealistic that the motor would become a missile.

It is concluded in Reference 4 that destructive overspeed of the pump and motor could occur as a result of a full double-ended pipe break LOCA in the recirculation pump suction line. In the event of motor failure, the motor stator and frame structure would prevent the release of any missiles as indicated above. In the event of pump destructive overspeed, impeller missiles could be produced. However, they will not penetrate the pump case. They could be ejected from the open end of the broken pipe. However, pipe restraints have been installed to prevent potential missile points in the pipe from developing. (See Subsection 5.5.1.4.)

Potential missiles from an EDG would be small auxiliary items knocked loose from the engine exterior by blows from within. Analysis has shown that the maximum velocity of these missiles would be 40 fps, with a maximum mass of 5 lb each. These missiles are of lower energy than potential tornado-generated missiles. As the external walls of the EDG rooms are constructed to withstand the tornado-generated missiles, missiles ejected from an EDG will be contained within that EDG room and therefore cannot incapacitate another EDG in the other division.

3.5.1.3 Tornado-Generated Missiles

3.5.1.3.1 General

Tornado forces and the design-basis tornado are discussed in Section 3.3. Objects lying in the path of tornadoes may be picked up by the tornado due to aerodynamic lift force or due to the rapid pressure reduction that may have injected the object into the tornado wind field. The objects that are potential missiles vary in size, shape, and number. The design-basis missiles selected for consideration in the Fermi 2 design are a 4-in. x 12-in. x 12-ft plank with a density of 40 lb/ft^3, and a 4000-lb passenger car traveling at 50 mph at a maximum of 25 ft above grade elevation. The design-basis missiles are given in Subsection 12.2.1.7.1 of the PSAR.

For the Category I 4160-V electrical ductbanks between the RHR cable vaults and the Reactor/Auxiliary building cable vaults, the top of the ductbanks is located approximately six inches below grade, the top of the manholes is located at grade level, and RHR cable vaults are located above grade. The design for this ductbank system is based on Regulatory Guide 1.76 Revision 1 (March 2007) (Reference 17) and, as such, the design is evaluated for the design-basis tornado missiles described in Regulatory Guide 1.76 Revision 1.

3.5.1.3.2 Additional Analyses

The missile barriers listed in Subsection 3.5.3 provide protection against tornado generated missiles; however, three areas received additional analysis to ensure resistance to tornado generated missiles. They are the spent fuel pool, the fan blades of the cooling towers in the Residual Heat Removal (RHR) complex, and the miscellaneous penetrations and openings in the exterior walls of the Reactor/Auxiliary Building and RHR Complex.

3.5.1.3.2.1 Spent Fuel Pool - Reactor Building

As the siding above the refueling floor is designed to release in the event of a design-basis tornado, potential damage to fuel in the spent fuel pool from tornado-generated missiles is of concern. The AEC noted this concern in its Safety Evaluation Report on the Construction Permit (Reference 2). The concern was identified as Post Construction Permit Open Item No. 9. This concern has also been the subject of analyses submitted to the AEC by GE (Reference 5). The Edison position on this open item was submitted to the AEC in August 1973 (Reference 6). The Edison position was based on the GE report (Reference 5) and a study of the probability of a tornado striking the site and showed that the probability of damage to fuel in the spent fuel pool by a tornado-borne missile is extremely small (7×10^{-10} per year) and that no additional protection is required. The AEC waived the requirement to provide tornado protection of the spent fuel pool in June 1974 (Reference 7) based on its own independent assessment. The AEC cited the low probability of a tornado, the lower likelihood that objects could be lifted to the elevation of the fuel pool and become missiles, and the expectation that where spent fuel damage were to occur, the associated offsite exposure radiological consequences would likely be within 10CFR100 limits.

3.5.1.3.2.2 Residual Heat Removal Complex Mechanical Draft Cooling Towers

A study was performed to determine the probability that both cooling tower divisions can be rendered out-of-service by tornado- generated missiles entering the fan discharge stack (Reference 8). The result of this study, as determined below, is that this probability is very small and is conservatively estimated between 10^{-9} and 10^{-10} per year. The RHR cooling towers and their missile protection features are described in Subsection 9.2.5.

In the cooling tower study, several potential design-basis tornado missiles are considered. These represent the complete range of all possible missiles that may be potential threats to the safety of the cooling towers:

a. A 4-in. x 1-ft x 12-ft wood plank

b. A 13.5-in.-diameter x 35-ft-long utility pole

c. A 1-in.-diameter x 3-ft-long steel rod

d. A 6-in.-diameter x 15-ft-long schedule 40 steel pipe

e. A 12-in.-diameter x 15-ft-long schedule 40 steel pipe.

Other missiles cited in the literature, such as a 2-in. x 4-in. x 1-ft wood piece, a 9-in. brick, a 6-in. x 12-in. x 2-in.-thick concrete slab, a 1-ft block concrete, and a "standard" automobile are not able to reach the level of the cooling towers if they are injected at ground level or at elevations of 200 ft or less (Reference 9).

Each design-basis missile was then analyzed for its ability to impact the cooling tower fan blades.

Using the three-dimensional wind flow field proposed by Bates and Swanson (Reference 10), the vertical impact velocities of the design-basis missiles at different roof elevations have been calculated assuming the objects are injected into the tornado wind field at different elevations. The results are shown in Table 3.5-1.

None of the missiles except the wood plank picked up at ground level or injected at 50-ft or 100-ft elevations, is able to reach the level of the cooling tower. The steel rod injected at 50 ft and other objects injected into the tornado wind field at higher elevations (250 ft) may be hurled into the cooling towers, but only a few missiles could be of this type.

Even if a missile lands in the cooling tower, it will not damage the cooling tower fan blades. The Marley Company, the manufacturer of the Fermi 2 RHR complex mechanical draft cooling towers, has calculated that the fan blades would safely withstand the impact from an object weighing 17 lb falling freely from an elevation of 250 ft. This is equivalent to a kinetic energy of about 8.5×10^4 ft-lb. Therefore, the fan blades are able to withstand the impact from smaller missiles; e.g., design-basis missile c. listed above (1-in.-diameter x 3-ft-long steel rod).

The number of missiles assumed to impact a cooling tower is then determined. The number of missiles that are injected into the tornado field depends on factors such as the number of "loose" objects lying in an area of a 3000-ft radius circle around the RHR complex, which contains the cooling towers. Therefore, the number of missiles injected into the tornado funnel cannot be decided with any degree of certainty. It is assumed that of all the potentially damaging objects available, two of them will be picked up by the design-basis tornado at just the right time and location to become a missile.

The cooling tower system is designed such that it can function even if one tower division is damaged and rendered out of operation. Therefore, for the cooling tower system to be out of service, both tower divisions must be damaged simultaneously by tornado missiles. For this to happen, the following sequence of events must occur:

a. A tornado strikes a point in the plant site. Based on the meteorological data and on Thom's model, the probability of this event is calculated as 7×10^{-4} per year

b. An object which is accelerated horizontally does not bounce and is ejected into the tornado at a 45° angle. This probability is conservatively estimated at 10^{-1}

c. The object maintains the orientation inside the tornado and exposes its maximum cross-sectional area to the full wind force. Since objects will tend to tumble, the probability of this event is conservatively estimated at 10^{-1}

d. The object is thrown into a cooling tower division. Objects of the type being considered here could land anywhere within 100 ft of the tornado funnel. This is a circular area of 500 ft diameter. The area of the cooling tower fan discharges in the RHR complex is about 850 ft^2. Therefore, the probability of a missile landing in a cooling tower division is approximately 4.3 x 10^{-3}. This is multiplied by two because it was assumed earlier that the two objects would be injected into the tornado wind field

e. The missiles land simultaneously in both tower divisions. The probability of this joint occurrence is calculated as the product of the probability of one missile landing in one tower division and the probability of the second missile landing in the other tower division simultaneously. Using the concept of statistical independence of these events, the probability of the joint event is conservatively estimated to be between 10^{-9} and 10^{-10} per year.

The draft ANSI standard on Plant Design Against Missiles (Reference 11) recommends that no protective measures be required if the combined probability of missile ejection and subsequent unacceptable damage is less than 10^{-7} per year. As the probability of tornado damage to the cooling tower unit calculated above is considerably lower than the acceptable limit, and because certain components and portions of the tower structure are hardened against tornado missiles and the fan blades can be replaced after a tornado (as described in subsection 9.2.5.2.2), it is concluded that no missile protective covers are required for the cooling towers. It may be noted that the probability evaluated herein is very conservative because most tornadoes have velocities lower than 300 mph. Some missiles, even though hurled into the towers, may lose part of their kinetic energy if they strike the walls. Such missiles are not effective in damaging the fan blades.

The 8-lb steel-rod missile could damage the fan blades if the velocity were high enough (i.e., slightly higher than listed in Table 3.5-1). The latest probability study on damage to the towers indicated a probability of 5 x 10^{-18} per year for all four cooling tower fans to be damaged by 20 steel-rod (rebar) missiles.

3.5.1.3.2.3 Exterior Walls/Roofs - Reactor/Auxiliary Building/RHR Complex

The exterior walls/roofs of the Reactor, Auxiliary, and Residual Heat Removal Complex buildings have been designed to resist the impact of tornado-generated missiles such that the safety related systems and components required for safe shutdown as identified in Tables 3.3-2 and 3.5-2 are generally protected. A limited number of these Seismic Category I systems and components located outside of (or otherwise not protected by these) Seismic Category I structures are evaluated based on a probabilistic missile damage analysis (Reference 19). The specific targets for which no tornado missile protection was required based on the risk analysis are listed in Table 3.5-3. The specific acceptance criterion for tornado damage for the unprotected systems and components required for safe-shutdown following a tornado event is that the cumulative sum of the mean damage probabilities for these systems and components be less than 10^{-6} per year as established in References 27 and 28. The aggregate mean damage probability corresponding to the scope of equipment identified in Table 3.5-3 is less than 10^{-6} per yr, which satisfies the regulatory acceptance criterion.

The manner in which these targets were identified and selected for evaluation is described under the "Scope" section below. The use of TORMIS as an appropriate tool for evaluating tornado missile risk was generically accepted by the NRC in Reference 23 subject to site-specific approval of the first application. The "Analysis" section below describes the manner and degree to which the Fermi 2 analysis meets the constraints of the original NRC SER (Reference 23) or was otherwise found to be acceptable in the site-specific SER approving its use (Reference 25).

3.5.1.3.2.3.1 Scope

The exterior walls/roofs of the Reactor, Auxiliary, and Residual Heat Removal Complex buildings have been designed to resist the impact of tornado-generated missiles such that the safety related systems and components required for safe shutdown identified in Tables 3.3-2 and 3.5-2 are generally protected. A limited number of these Seismic Category I systems and components located outside of (or otherwise not protected by these) Seismic Category I structures are evaluated as not requiring unique tornado missile protection by burial or barriers on the basis of a probabilistic missile damage analysis.

Table 3.5-3 identifies the specific features evaluated in the probabilistic tornado missile analysis. The specific targets included in this table represent wall penetrations and doors in the exterior surfaces of these structures. Generally, specific safety-related targets are not associated with any particular penetration; hence, the tornado missile hazard associated with these penetrations and openings is limited to and characterized by the probability of missile penetration of the target itself. However, specific safety-related targets can be associated with missiles penetrating the reactor building railroad air lock doors, the first floor auxiliary building south wall entrance, and the EDG removable wall panels.

Unprotected safety-related equipment not identified in UFSAR Table 3.3-2 as being required for safe reactor shutdown following a tornado was not included as targets. Examples include Control Room Emergency Filtration system south emergency makeup intake, the south portion of the Auxiliary Building rooftop and the Standby Gas Treatment equipment located on the refuel floor. In addition, the RHR Mechanical Draft Cooling Towers which are specifically licensed for post-tornado repair and restoration (See UFSAR Section 3.5.1.3.2.2) and the Spent Fuel Pool which was evaluated on the basis of an alternative risk analysis (See Section 3.5.1.3.2.1) were both excluded from the scope of analysis.

Other features that were excluded for this risk analysis are the buried underground cable vaults between the RHR complex and the auxiliary building, the EDG fuel oil tank vents and the EDG exhaust stacks, which are located on the roof of the RHR complex. Both of these rooftop features are provided with tornado missile shield protection specifically designed to prevent vertically travelling missiles from entering the RHR complex and damaging the EDG fuel oil tanks and diesel engines.

3.5.1.3.2.3.2 Analysis

The mean cumulative damage probability for the targets identified in Table 3.5-3 was evaluated using TORMIS, a Monte Carlo based program for simulating tornados that was developed from the NRC approved EPRI version of this program (References 20, 21, 22). Major inputs to the analysis include:

a. the regional probabilities of the occurrence of tornados
b. the location and size of eligible targets
c. location and number of potential missile sources

Given these inputs, TORMIS computes the hit and damage probabilities associated with each target. These probabilities are post-processed to generate the aggregate risk associated with all targets. The term “target damage” is used in a general sense to mean any damage (or “loss of function”) criteria caused by a tornado missile hitting the target. Target damage is not necessarily the same as target hit, but hit can equal damage for fragile equipment. The “damage” probabilities included in this analysis consisted of using the built-in TORMIS penetration, spall, and perforation equations for selected steel and concrete targets. In addition, the missile size, impact orientation, and velocity vector orientation were used to compute the probabilities of missiles entering “pipe-penetration” type openings. The TORMIS feature for overall structural response damage modeling capability was not used for this analysis.

In Reference 23, the NRC approved use of the (EPRI) TORMIS methodology subject to the following constraints:

1. Data on tornado characteristics should be employed for both broad regions and small areas around the site. The most conservative values should be used in the risk analysis or justification provided for those values selected.
2. The EPRI study proposes a modified tornado classification, Modified F (F')-scale for which the velocity ranges are lower by as much as 25% than the velocity ranges originally proposed in the Fujita (F)-scale. Insufficient documentation was provided in the studies in support of the reduced F'-scale. The F-scale tornado classification should therefore be used in order to obtain conservative results.
3. Reductions in tornado wind speed near the ground due to surface friction effects are not sufficiently documented in the EPRI study. Such reductions were not consistently accounted for when estimating tornado wind speeds at 33 feet above grade on the basis of observed damage at lower elevations. Therefore, users should calculate the effect of assuming velocity profiles with ratios Vo (speed at ground level) ÷ V33 (speed at 33 feet elevation) higher than that in the EPRI study. Discussion of sensitivity of the results to changes in the modeling of the tornado wind speed profile near the ground should be provided.
4. The assumptions concerning the locations and numbers of potential missiles presented at a specific site are not well established in the EPRI studies. However, The EPRI methodology allows site specific information on tornado missile availability to be incorporated in the risk calculation. Therefore, users should provide sufficient

information to justify the assumed missile density based on site specific missile sources and dominant tornado paths of travel.

5. Once the EPRI methodology has been chosen, justification should be provided for any deviations from the calculation approach.

Also, as generically approved in Reference 23 (and clarified through Reference 26), the TORMIS methodology is not approved for proposing:

a. elimination of existing tornado barriers
b. technical specification (TS) changes, or
c. plant modifications

The description of the Fermi 2 site-specific TORMIS analysis was reviewed against the criteria established in References 23 and 26 and was approved in Reference 25 based on the following characteristics:

1. Definition of the Fermi 2 TORMIS Tornado Sub-Region

A site-specific analysis was performed to generate a tornado hazard curve data set for the TORMIS analysis. The tornado data retained in the National Climatic Data Center Storm Events Data Base (NCDC, 2006) files for the years 1950-2005 were used to analyze both broad and small regions around Fermi 2 in order to identify a suitable representative sub-region for the site. Tornado occurrences were mapped for the large region, a 15º longitude x 15º latitude area centered on the Fermi 2 site, and statistical tests were performed using 1º x 1º and 3º x 3º blocks to identify a suitably homogeneous sub-region. The historical records of tornado occurrences within the sub-region tornado were used to establish the tornado occurrence rate, (Enhanced-Fujita) EF-scale intensities, path length, width, and direction variables to be specified as input for use in the TORMIS analysis.

The statistical analysis of the sub-region data established a mean occurrence rate of 3.1E-4 per year over the 56-year period. In accordance with the TORMIS methodology, backwards averaging was used to estimate a detrended occurrence rate to correct for changes in the annual reporting trends. The adjusted mean occurrence rate was determined to be 4.002E-4/year based on the 30-year backwards average.

2. Tornado Windspeed Intensity

The analysis utilizes the original Enhanced Fujita (EF) scale windspeeds as per Reference 24. Though the 1983 NRC SER called for the use of the F-scale of tornado intensity for assigning tornado windspeeds to each intensity category (F1-F5), the EF-scale was subsequently adopted in the positions of NRC Reg. Guide 1.76 Revision 1 that are based on Reference 24.

3. Characterization of Tornado Windspeed as a Function of Height Above Ground Elevation

The Fermi 2 TORMIS simulations were performed with the TORMIS rotational velocity Profile 3, which has increased near ground windspeeds over Profile 5; the profile used in the 1981 EPRI TORMIS reports. Hence, the Fermi 2 runs were made with higher near ground windspeeds than in the EPRI study. A sensitivity study was conducted by running the original EPRI profiles and comparing the results. The most conservative profile with highest near ground windspeeds was conservatively used.

4. Missile Characterization and Site-Structure Models

Walkdowns of the Fermi 2 site were performed to characterize the missile sources and plant configuration. This information was developed into the plant modeling inputs for the TORMIS analysis that describe the facility by specifying the geometry, location, and material properties of the structures/components and the location of potential missile sources. Missile sources (buildings, houses, storage areas, vehicles, etc.) were catalogued and modeled to a distance of approximately 2,500 feet. This is done by specifying missile origin zones around the facility and a statistical description of missile types, based on the facility survey. The site surveys were conducted just prior to refueling outages to maximize the estimated population of available missiles and missiles sources. The Fermi 2 site missiles include the 20 standard TORMIS missiles in Reference 21, including structural sections, pipes, wood members, other construction materials, and an automobile category. In addition to the 20 standard TORMIS missile types, three Fermi 2 specific missiles were created for the analysis, one to represent scaffold clamps of which there were a large number present during the site walkdown, one to represent the sections of metal siding that enclose the upper portions of the Reactor and Turbine Buildings, and the third to represent the large number of concrete block also identified during the site walkdowns. The TORMIS analysis used over 200,000 missiles in the simulations of EF5 tornadoes striking Fermi 2.

5. Deviations from the Original EPRI Methodology

The Fermi 2 analysis is performed using an update of TORMIS developed from the original EPRI NP-2005 source code. With some exceptions, this version of TORMIS implements the original NRC SER approved methodology. Revisions of the original NRC-approved version of the code generally implement changes necessary to enable continued use of the program on modern computing platforms and to enable analysis of larger problems. Specifically, the original main frame based random number generator has been replaced with a new machine independent algorithm and the code was re-dimensioned to allow larger numbers of missiles and surfaces.

The updated TORMIS program implements an algorithm for evaluating the risk of damage to piping penetrations credited in the Fermi 2 analysis that was not present in the original NRC approved methodology. The method consists of identifying the minimum required missile size, angle of orientation and angle of incidence at impact necessary for a missile to be capable of passing through a pipe penetration target. Missiles that are too large, not oriented correctly, or that impinge obliquely on a target are screened out based on these criteria. This method eliminates from the calculated cumulative risk those impacts which would not realistically have resulted in missile penetration of a pipe penetration target.

3.5.1.3.3 Conclusion

As a result of these studies, the tornado-generated missiles to be considered in barrier design are the wood plank and the automobile, previously described.

3.5.1.4 Site-Related Missiles

3.5.1.4.1 Airplanes

Airports in the vicinity of the Fermi 2 site are listed in Table 2.2-2 and shown in Figure 2.2-1. Table 2.2-2 also lists the proximity to the site, number of and type of aircraft, and other physical and operations data. As discussed in Section 2.2, the nearest airport (2 miles away) cannot accommodate aircraft large enough to be a hazard to Fermi 2 and the nearest major airport is too far away (19 miles north-northwest of the site) to be considered a potential hazard with regard to large-aircraft takeoff and landing. In addition, there are no nearby military airports that could be expected to accommodate aircraft with bomb or explosive loads.

3.5.1.4.2 Military Activities

There are no military facilities within 10 miles of the plant. There are two restricted areas in Lake Erie, 20 and 27 miles from the plant, which are used as impact areas for small arms, ground artillery, and anti-aircraft artillery from Camp Perry and from the test-firing range at Erie Industrial Park. However, restriction to weapon horizontal-firing range and direction, as well as the nature of the projectiles, preclude a threat to the plant.

3.5.1.5 Primary Containment Internal Missiles

The potential for missiles inside the containment due to gravitational effects from unrestrained equipment is possible only during maintenance situations. All equipment and components located inside the containment and associated with reactor operation and safety are restrained. Equipment moved into the containment for maintenance operations (including hoists) is controlled by administrative procedures and is removed when personnel leave the maintenance site or prior to returning to reactor operation. Where possible and practical, maintenance equipment used inside the containment is temporarily restrained. In view of the above, any missiles due to gravitational effects are expected to be relatively small and any resulting damage is anticipated to be minor.

3.5.2 Selected Missiles

As a result of the investigations described in Subsection 3.5.l, the missiles to be considered in barrier design are the tornado generated missiles. These missiles are those considered as a design basis in the PSAR and approved by the AEC as documented in the AEC Safety Evaluation Report (Reference 2). For the Category I 4160-V electrical ductbanks between the RHR cable vaults at the RHR complex and the Reactor/Auxiliary building, the tornado missiles identified in Regulatory Guide 1.76 Revision 1 (Reference 17) are considered.

3.5.2.1 Tornado-Generated Missiles

The tornado-generated missiles are a 4-in. x 12-in. x 12-ft wood plank with a density of 40 lb/ft^3, traveling end-on at a velocity of 255 mph with a contact area of 48 in.^{2}; and a 4000 lb passenger car traveling through the air at 50 mph at a maximum 25 ft above grade elevation. The car has a contact area of 20 ft^2. In the case of tornado-generated missiles, it is assumed that only walls and other vertical exposed surfaces are subject to impacts. Roof structures would be subject only to free-falling ballistic-type projectiles (e.g., wood or stone debris) without high tornadic wind force components. If penetration of the roof structures should occur, such penetration would not constitute a hazard, since the projectile would have very low energy, and the concrete floors and walls protect safety-related equipment for safe shutdown.

The following Design Basis Tornado missiles from Table 2 of Regulatory Guide 1.76 Revision 1 (March 2007) (Reference 17) are considered for the Category I 4160-V RHR cable vaults and the manholes and ductbanks between these cable vaults and the Reactor/Auxiliary building cable vaults:

a. 6.625" diameter x 15 ft long Schedule 40 steel pipe weighing 287 lbs and traveling horizontally at 135 fps

b. 4,000 lb, 16.4 ft x 6.6 ft x 4.3 passenger car traveling horizontally through the air at 135 fps at a maximum height of 30 ft above ground

c. 1" diameter solid steel sphere, weighing 0.147 lb and traveling horizontally at 26 fps

Vertical missiles are all missiles listed above with a vertical velocity equal to 67% of their horizontal speed.

In addition, the following missiles addressed in the Safety Evaluation Report are also evaluated for penetration resistance and regeneration of secondary missiles:

a. 1" diameter x 3 ft long steel rod weighing 8 lbs, traveling horizontally at 250 fps

b. 13.5" diameter x 35 ft long utility pole weighing 1490 lbs, traveling at 247 fps

Vertical missiles are all missiles listed above with a vertical velocity equal to 67% of their horizontal speed.

3.5.3 Missile Barriers and Loadings

Structures, shields, and barriers designed to withstand missile effects are given in Table 3.5-2 according to the equipment protected. In addition to these barriers, the steel plate primary containment vessel is completely enclosed in and surrounded by a reinforced-concrete structure as described in Subsection 3.8.4. This concrete structure, in addition to serving as a radiation shield for personnel in the reactor building, provides a major structural barrier for the protection of the containment and reactor system against missiles that may be generated external to the primary containment.

The suppression chamber has no source of internal or external missile generation. The vent pipes connecting the suppression chamber to the drywell are protected by jet deflectors. The vent discharge headers and piping are designed to withstand the jet reaction force caused by flow discharge into the suppression pool. The control rod drive (CRD) mechanisms are located in a concrete vault below the reactor pressure vessel.

3.5.4 Barrier Design Procedures

3.5.4.1 Overall Structural Response

To determine the capability of the missile barriers provided, the impact and penetration of potential missiles must be determined. Since the missile mass is small compared with the mass of any Category I structure, the only meaningful overall structural response is that of the structural element impacted by the missile. The overall response of the structural element is investigated by designing the element for the forces transmitted to it by the missile.

3.5.4.2 Edge Impact

For edge impact, punching shear stress was checked after obtaining the maximum force impacted to the element by the missile. The punching shear stress is given by the following expressions:

$$Qs = \frac{mV_o}{t_d s} \quad \text{(for rigid missles)} \tag{3.5-2}$$

$$Qs = \frac{F_1}{s} \quad \text{(for nonrigid missles)} \tag{3.5-3}$$

where

F_1 = maximum contact force = $1.14WV_o$

and

t_d = impact time = $\frac{2D}{V_o}$

D' = penetration depth calculated by modified Petry Formula (Subsection 3.5.4.7)

V_o = initial velocity of missile

m = mass of missile

s = perimeter of area enclosed by a border extending one-half of the panel thickness beyond contact area

W = weight of missile

3.5.4.3 Central Impact

For central impact in the case of rigid missiles, the maximum force impacted to a structural element is calculated by the following expression:

$$F = \frac{mV_o 2}{2D'} \tag{3.5-4}$$

and

$$t_d = \text{duration of force} = \frac{2D'}{V_o} \tag{3.5-5}$$

After the force F and its duration t_d are obtained, the element is designed for this dynamic load. For central impact in the case of nonrigid missiles, the panel is modeled as a single degree of freedom system with equivalent mass and equivalent stiffness. The equation of motion for impact is solved to get maximum deflection of the element. This deflection is compared with allowable (or ductility ratio) to arrive at a satisfactory design.

3.5.4.4 Impact Analytical Procedures

The impact of the missile is considered plastic because of the local unrecoverable deformations of either the missile or the target or of both. The velocity of the missile and the target (concrete panel) after the impact, V_a, is determined from the consideration of conservation of linear momentum and is expressed by the following equation:

$$M_m V_i = M_m V_a + M_e V_a \quad (3.5\text{-}6)$$

where

M_m = mass of missile

V_i = velocity of impact

M_e = effective mass of target

For the Category I 4160-V RHR cable vaults and the manholes and ductbanks between these cable vaults and the Reactor/Auxiliary building cable vaults, overall structural response is based on the dynamic response of the structures and impulse-load time history. A simplified method based on idealization of the structure to an equivalent single-degree-of freedom system is utilized.

The procedure used in determining impactive force and time duration of the impact follows the guidance in Reference 16.

The impactive force and time duration of a hard missile, such as the 6” diameter schedule 40 steel pipe, is determined by the expression shown in Section 3.5.4.3. The impactive force and time duration for soft missiles, such as the automobile and wood plank, is determined by the Riera formula, as outlined in Reference 16.

3.5.4.5 Punching Shear Analytical Procedure

Reinforced-concrete panels are checked for the punching shear failure and the flexural yielding failures. The effective mass, M_e, of the panel for the case of punching shear failure is obtained as follows:

$$M_e = (A + d)(B + d)dw \quad (3.5\text{-}7)$$

where

A, B = dimensions of missile

d = thickness of panel

w = density of target material

3.5.4.6 Flexural Failure Analytical Procedure

The effective mass for the case of flexural failure of a panel is defined as that mass which must be concentrated at the point of impact on an equivalent weightless slab so that it will have the same kinetic energy as the actual slab when the point of impact is subjected to unit velocity.

For a flexural failure, the energy transferred to the slab is compared with its energy capacity at an appropriate ductility ratio. For a punching shear failure, the shear capacity at the critical section is compared with the shear force transferred to the slab.

3.5.4.7 Depth of Penetration Analytical Procedure

The depth of penetration into concrete walls is calculated using the Modified Petry Formula (Reference 12). The concrete barrier thickness was selected to prevent secondary missiles formed by scabbing from damaging both divisions of protected systems safe shutdown equipment.

Concrete wall/slab thickness provided for the Category I 4160-V RHR cable vaults, manholes, manhole covers, and ductbanks between these cable vaults and the Reactor/Auxiliary building cable vaults are more than the minimum acceptable barrier thickness required as shown in Table 1 of NUREG-0800, Standard Review Plan 3.5.3 Revision 3, dated March 2007 (Reference 18).

Modified Petry Formula (Reference 12) is used to determine the concrete protective cover thickness to prevent penetration and regeneration of secondary missiles for the two additional tornado missiles identified in the Safety Evaluation Report.

The method of calculation used to determine the energy required to penetrate a steel plate is based on extensive tests conducted by the Stanford Research Institute (Reference 13). During these tests, rod-shaped missiles were impacted against square steel plates having clamped edges. The results of the tests are described by the following expression for minimum energy per unit diameter of missile required for perforation of a steel plate:

$$\frac{E}{D} = U\left(0.344T^2 + \frac{W}{W_s}0.032T\right) \qquad (3.5\text{-}8)$$

where

- E = critical energy required for penetration, ft-lb
- D = diameter of missile, in.
- U = ultimate tensile strength of steel plate, lb/in.^{2}
- T = plate thickness, in.
- W = length of side of square window in the target frame between the rigid supports, in.
- W_s = test constant = 4 in.

No composite section (concrete with steel plate backing or the like) has been used for missile-resistant structural elements.

The impact of a turbine-generator missile on the reactor building or auxiliary building is discussed and references are cited in Subsection l0.2.3. The impact of a turbine missile on the RHR complex has also been evaluated.

3.5.5 Missile Barrier Features

The missile barriers listed in Table 3.5-2 provide adequate protection against potential tornado-generated missiles. In addition, it has been shown that the probability of missile damage to either fuel in the spent-fuel pool or the RHR cooling tower fans, both of which could be exposed to such damage, is extremely small. Together with the redundancy and separation provided, the missile protection provided for Fermi 2 is adequate.

The general arrangement of piping and equipment in the drywell showing the separation of redundant systems is given in Figure 3.5-1, Sheets 1 through 6.

For assumed failures of the high pressure coolant injection (HPCI) system, the automatic depressurization system (ADS) functions to reduce the reactor pressure to a value low enough to allow the low pressure coolant injection (LPCI) and core spray systems to pump water to the reactor pressure vessel (RPV) in time to cool the core consistent with the design basis. (See Subsection 6.3.2.2.2.) The ADS uses five of the 15 safety/relief valves (SRVs) of the nuclear boiler pressure-relief system to achieve the automatic blowdown to the suppression pool. Protection from simultaneous damage to the HPCI steam line inside the containment and to the SRVs designated for ADS function due to pipe whip or fragments of pipes is provided by physical separation. The HPCI steam source is provided from main steam line A, while only the SRVs on main steam lines C and D are considered available for performance of the ADS function.

3.5 MISSILE PROTECTION

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3.5 MISSILE PROTECTION

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TABLE 3.5-1 MISSILE TRAJECTORY DATA FOR TORNADO MISSILES NEAR THE RESIDUAL HEAT REMOVAL COMPLEX COOLING TOWERS

Missile	Initial Elevation (ft)	Peak Elevation (ft)	Vertical Velocity at Impact (fps)
a. 4-in. x 1-ft x 12-ft-long wood plank	0	734	97
	50	739	97
	100	732	97
	250	702	96
b. 13.5-in. diameter x 35-ft-long utility pole	0	0	-
	60	60	-
	100	100	-
c. 1-in. diameter x 3-ft-long steel rod	0	2	-
	50	662	133
	100	664	132
	250	604	128
d. 6-in. diameter x 15-ft-long Schedule 40 steel pipe	0	-	-
	50	50	-
	100	100	-
	250	268	96
e. 12-in. diameter x 15-ft-long Schedule 40 steel pipe	0	-	-
	50	50	-
	100	100	-
	250	250	77

TABLE 3.5-2 EQUIPMENT PROTECTED FROM MISSILES AND ASSOCIATED MISSILE BARRIERS

A. REACTOR AND AUXILIARY BUILDINGS

Equipment Protected	Missile Barriers
1. All items whose failure could affect the operation and functions of the primary reactor containment and those that are necessary for safe shutdown of the reactor	1. a. All exterior concrete walls b. Reactor building fifth floor concrete slab c. Auxiliary building concrete roof slab d. Auxiliary building fifth floor concrete slab e. Reactor building fifth floor equipment hatch cover
2. Air conditioning equipment for the control center	2. a. Auxiliary building concrete roof slab b. Walls between auxiliary and turbine building c. Shield barrier at the Auxiliary Building / Turbine Building third floor portal. (see Note 1)
3. Reactor pressure vessel	3. Shield plug over reactor pressure vessel
4. Main control room, battery room ESF switch-gear room, emergency closed cooling water system, residual heat removal system, relay room, control rod drive units	4. Combined thickness of walls and/or floors of the reactor and auxiliary buildings above and including the fourth floor. Removable exterior precast panel in Division I Switchgear Room South Wall is protected by a 1-inch steel plate.

Note 1: There are two EECW lines in the Auxiliary Building which are potentially susceptible to tornadic induced missiles coming from the Turbine Building through the connecting portal on the third floor.

B. RHR COMPLEX BUILDING

Equipment Protected	Missile Barriers
All items whose failure could affect the operation and functions of the primary containment and those that are necessary for safe shutdown of the reactor (including the EDGs)	a. All exterior concrete walls b. All concrete roof slabs except the RHR complex cooling tower discharges c. Isolation walls between redundant systems

C. Category I 4160-V RHR cable vaults and the manholes and ductbanks between these cable vaults and the Reactor/Auxiliary building cable vaults

Equipment Protected	Missile Barriers
All items whose failure could affect the operation and functions of the primary containment and those that are necessary for safe shutdown of the reactor (including the EDGs)	a. All ductbanks b. All concrete walls c. All concrete roof slabs d. Access covers at RHR cable vaults e. Manholes

TABLE 3.5-3 List of Unprotected Plant Targets Accepted Based on TORMIS Analysis

DESCRIPTION	BUILDING / SITE LOCATION
Pipe penetration P-150	AB
Pipe penetration P-151	AB
Pipe penetration P-152	AB
Pipe penetration P-153	AB
Electrical penetration E-11117	AB
Electrical penetration E-11116	AB
Instrumentation penetration I-5504	AB
Instrumentation penetration I-5505	AB
Ventilation penetration V-521	AB
Electrical Penetration E-5654	AB
Pipe penetration P-139	AB
Pipe penetration P-140	AB
Pipe penetration P-141	AB
Pipe penetration P-142	AB
Pipe penetration P-143	AB
Electrical penetration E-11153	AB
Electrical penetration E-11154	AB
Pipe penetration P-136	AB
Pipe penetration P-137	AB
Pipe penetration P-138	AB
Electrical penetration E-1270	AB
Electrical penetration E-1271	AB
Electrical penetration E-1272	AB
Electrical penetration E-1273	AB
Pipe penetration P-10765	AB
Electrical penetration E-15132	AB
Electrical penetration E-11054	AB
Pipe penetration P-10766	AB
Class 1E Electrical Cables East of Door R1-15 (Safety related electrical cables East of R1-15)	AB
Electrical penetration E-5757	RB
Pipe penetration P-5609	RB
Pipe penetration P-5624	RB
Pipe penetration P-5625	RB
Pipe penetration P-17305	RB
Pipe penetration P-17319	RB
Outer Railroad Air Lock Door R1-1	RB
Electrical penetration E-5543	RB

TABLE 3.5-3 List of Unprotected Plant Targets Accepted Based on TORMIS Analysis

DESCRIPTION	BUILDING / SITE LOCATION
Electrical penetration E-10764	RB
Pipe penetration P-156 (Area around pipe protected by flange)	RB
Pipe penetration P-156 (Pipe in opening)	RB
Electrical penetration E-5521	RB
Pipe penetration P-158	RB
Pipe penetration P-157	RB
Pipe penetration P-161	RB
Pipe penetration P-162	RB
Instrumentation penetration I-5657	RB
Pipe penetration P-160	RB
Pipe penetration P-12343	RB
Pipe penetration P-159	RB
Removable Panel (EDG-11)	RHR
Removable Panel (EDG-12)	RHR
Removable Panel (EDG-13)	RHR
Removable Panel (EDG-14)	RHR
Door to Motor Drive for Cooling Tower Fan (North End, East Tower, Top Door)	RHR
Door to Motor Drive for Cooling Tower Fan (North End, East Tower, Bottom Door)	RHR
Door to Motor Drive for Cooling Tower Fan (North End, West Tower, Top Door)	RHR
Door to Motor Drive for Cooling Tower Fan (North End, West Tower, Bottom Door)	RHR
Door to Motor Drive for Cooling Tower Fan (South End, East Tower, Top Door)	RHR
Door to Motor Drive for Cooling Tower Fan (South End, East Tower, Bottom Door)	RHR
Door to Motor Drive for Cooling Tower Fan (South End, West Tower, Top Door)	RHR
Door to Motor Drive for Cooling Tower Fan (South End, West Tower, Bottom Door)	RHR
Roof Penetration MK-142	RHR
Roof Penetration MK-144	RHR
West Wall Penetration MK-219	RHR
West Wall Penetration MK-220	RHR
West Wall Penetration MK-221	RHR
West Wall Penetration MK-222	RHR
West Wall Penetration MK-344	RHR

TABLE 3.5-3 List of Unprotected Plant Targets Accepted Based on TORMIS Analysis

DESCRIPTION	BUILDING / SITE LOCATION
West Wall Penetration MK-345	RHR
West Wall Penetration MK-346	RHR
West Wall Penetration MK-347	RHR
Doors R3-13 (Security Door RBD17) & R3-28	AB
Door R3-12 (Security Door RBD21)	AB
Concrete Block Wall #215	AB
Refuel Floor Equipment Hatch Cover (A/B – 10/11)	RB
Inner Railroad Air Lock Door R1-2 (effectively modeled as intersection with targets 57, 58, 59, and 60)	RB
Class 1E Equipment West of Interior Access Door R1-12	AB
Safety-related piping behind Railroad Air Lock Doors (Div. 2 EESW supply & return & RHR Containment Spray)	RB
Safety-related piping behind Railroad Air Lock Doors (Div. 1 EESW supply & FPCCU supply & return)	RB
Safety-related piping behind Railroad Air Lock Doors (RHR Containment Spray – vertical)	RB
Safety-related piping behind Railroad Air Lock Doors (RHR Containment Spray – horizontal)	RB

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Refer to Plant Drawing DW GEN ARRANGE

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FIGURE 3.5-1, SHEET 1

GENERAL ARRANGEMENT OF PIPING AND EQUIPMENT IN THE DRYWELL

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FIGURE 3.5-1, SHEET 2

GENERAL ARRANGEMENT OF PIPING AND EQUIPMENT IN THE DRYWELL

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FIGURE 3.5-1, SHEET 3

GENERAL ARRANGEMENT OF PIPING AND EQUIPMENT IN THE DRYWELL

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Refer to Plant Drawing DW GEN ARRANGE

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FIGURE 3.5-1, SHEET 5

GENERAL ARRANGEMENT OF PIPING AND EQUIPMENT IN THE DRYWELL

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FIGURE 3.5-1, SHEET 6

GENERAL ARRANGEMENT OF PIPING AND EQUIPMENT IN THE DRYWELL

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

Fermi 2 is designed with appropriate protection against the consequences of a LOCA. Specifically included are an emergency core cooling system (ECCS) to protect the core from the thermal-hydraulic consequences of a LOCA; a containment system to protect the public from the radiological consequences of a LOCA; and a system of restraints, equipment, piping arrangements with physical separation of redundant components, and protective shields to limit damage escalation from the dynamic effects (i.e., blowdown jet forces and pipe whip) associated with a LOCA.

The design provisions and corresponding criteria for the emergency core cooling and containment systems are covered in Chapter 6. Subsection 3.6.1 describes the measures that have been used to ensure that the containment vessel and all essential equipment within the containment, including components of the reactor coolant pressure boundary (RCPB), engineered safety feature (ESF) systems, and equipment supports, are adequately protected against the postulated LOCA dynamic effects.

The measures taken for protection against dynamic effects associated with the postulated rupture of high- and moderate-energy fluid piping outside the containment are described in Subsection 3.6.2.

Detailed analytical methods and computer codes are discussed in Subsection 3.6.3.

3.6.1 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping Inside the Containment

3.6.1.1 Systems in Which Design-Basis Pipe Breaks Occur

3.6.1.1.1 Break Location Criteria

All piping that is part of the RCPB and that is subject to reactor pressure continuously during normal plant operation, is considered as a potential initiator of a pipe break, and is analyzed for its dynamic effects damage potential. Piping that is never or only infrequently (i.e., during test operations) subject to reactor pressure is not considered as an initiator of a pipe break. Initial pipe-break events are not assumed to occur in pump and valve bodies because of their greater wall thickness and their location in the low-stress portions of the piping systems.

3.6.1.1.2 Longitudinal and Circumferential Breaks

The following types of breaks were postulated in the RCPB piping systems: (1) circumferential breaks were postulated in piping having a nominal diameter greater than 1 in. and (2) longitudinal breaks were postulated in piping having a nominal diameter greater than 4 in.

Except where limited by structural design features, a circumferential break results in pipe severance with full separation. The break was assumed perpendicular to the longitudinal axis of the pipe at the break location. The fluid discharge coefficient at the break was determined from analytical or experimental work.

A longitudinal break results in an axial split without severance. For design purposes, the longitudinal break was assumed to be rectangular in shape, with an area equal to the largest piping cross-sectional flow area at the point of break.

3.6.1.1.3 Major Piping Systems Considered for Dynamic Effects of Postulated Pipe Breaks

The major piping systems inside the containment considered for protection against dynamic effects of the postulated ruptures of piping are the piping associated with the following systems:

a. Main steam system-inside and outside the containment

b. Recirculation system

c. Feedwater system

d. High-pressure coolant injection (HPCI) system

e. Reactor core isolation cooling (RCIC) system

f. Core spray (CS) systems

g. Residual heat removal (RHR) supply and return lines.

These and other minor non-safety class system (see Subsection 3.6.1.1.4) pipe-break analyses have been submitted to the AEC in References 1 through 11.

References 1 through 11 describe the Fermi 2 conservative design against the dynamic effects of postulated pipe ruptures inside the containment and they show that the requirements of 10 CFR 50, Appendix A, and 10 CFR 100, as well as the intent of Regulatory Guide 1.46, are in fact met. Supplemental analyses have also been completed to establish as-built compliance with these criteria.

In addition, a detailed analysis of a postulated line break in the region of a reactor vessel nozzle safe-end and its effects on the sacrificial shield wall was performed in response to ACRS concerns and was submitted to the AEC (References 12 and 13).

There are no ASME Boiler and Pressure Vessel (B&PV) Code Section III, Class 2 and 3, high-energy piping systems located inside the primary containment.

3.6.1.1.4 Consideration of Other Systems (Non-Safety Class Systems)

Certain "other" systems (see Section 3.2) and components are not required for the safe shutdown of the reactor nor are they required for the limitation of the offsite release in the event of a LOCA. However, while none of this equipment is needed during or following a LOCA, some dynamic effects must be considered where a non-safety class system or component failure could initiate or escalate a LOCA in one of the following systems or components:

a. Reactor water cleanup (RWCU) system

b. RPV vent line

c. Main steam drains

d. Standby liquid control system.

3.6.1.2 Design-Basis Pipe-Break Criteria

The following definitions are used for piping run terminology.

Main Run - Piping interconnecting terminal ends. All branch lines from the main run are considered branch runs, with the exception of the following:

a. Free-ended branch lines throughout which there is no significant restraint to thermal expansion are considered part of the main run

b. All ASME B&PV Code Section III, Class 1, branch lines that are included with the main run piping in the code stress analysis computer mathematical model are considered part of the main run.

Piping Run - A main or branch run.

Terminal End - Piping originating at the structure or components (such as vessel and equipment nozzles and structural piping anchors) that acts as a rigid constraint to the thermal expansion. Typically, the anchors assumed for the piping code stress analysis are considered terminal ends. In-line fittings, such as valves, not assumed to be anchored in the piping code stress analysis, are not terminal ends. The branch connection to the main run is one of the terminal ends of a branch run, except where the branch run was classified as part of a main run as defined above.

Break Location in ASME B&PV Section III, Class 1 Piping Runs

Postulated pipe-break locations are selected in accordance with the intent of Regulatory Guide 1.46; NRC Branch Technical Position (BTP) APCSB 3.1, Appendix B; and as expanded in NRC BTP MEB 3-1. For ASME Section III, Class 1 piping systems, the postulated break locations are as follows:

a. The terminal ends of the pressurized portions of the run

b. At intermediate locations between the terminal ends where the maximum stress range between any two load sets (including zero load set), according to Subarticle NB- 3600 of ASME Section III for upset plant conditions and an independent operating-basis earthquake (OBE) event transient, exceeds the following:

1. If the stress range calculated using Equation 10 of the Code exceeds 2.4 S_m but is not greater than 3 S_m, no breaks will be postulated unless the cumulative usage factor exceeds 0.1

2. If the stress ranges, as calculated by Equation 12 or 13 of the Code, exceed 2.4 S_m, or if the cumulative usage factor exceeds 0.1 when Equation 10 exceeds 3 S_m.

c. Arbitrary intermediate pipe breaks no longer need to be postulated, per Generic Letter 87-11

3.6.1.2.1 Core Cooling Requirements

The designed emergency core cooling system (ECCS) capability can be maintained provided that dynamic-effect consequences do not exceed the following break area, break combination, and maintenance of minimum core cooling requirements.

3.6.1.2.1.1 Maximum Allowable Break Areas

The maximum allowable break areas are as follows:

a. For breaks involving recirculation piping, the total effective area of all broken pipes, including the effective area of the recirculation line break, shall not exceed the total effective area of the design-basis double-ended recirculation line break (see Subsection 6.2.1.3). By limiting the total area of all broken pipes involving recirculation loops to an area less than or equal to that of the design-basis accident (DBA) (circumferential break of recirculation loop), no accident could be more severe than the DBA

b. For breaks not involving recirculation piping, the effects are much less severe than recirculation line breaks. Hence, the total break area can be allowed to be larger than the recirculation breaks. Therefore, the total break area shall not exceed the sum of one feedwater header pipe area, one steam line (upstream of flow limiter) pipe area, and one core spray pipe area.

3.6.1.2.1.2 Break Combinations

In addition to the pipe-break-area restrictions, breaks involving one recirculation loop shall not result in loss of function or damage to the other recirculation loop or loss of coolant from the other loop in excess of that which would result from a break of the attached cleanup connection on the suction side of the loop.

3.6.1.2.1.3 Required Cooling Systems

To ensure compliance with Appendix A of 10 CFR 50, General Design Criteria for Nuclear Power Plants, the following cooling system requirements, including all required support systems, must be met after an additional single active safety system failure:

a. For breaks not involving recirculation piping, at least two low-pressure coolant injection (LPCI) pumps or one core spray system shall be available for core cooling

b. For breaks involving recirculation piping, at least one core spray line and two LPCI pumps or two core spray lines shall be available for core cooling

c. For a steam line break with a total effective break area of less than 0.4 ft^2, either the HPCI or automatic depressurization system (ADS) shall be available for reactor depressurization. At least (n-l) ADS valves must be available (n = total number of ADS valves)

d. For liquid breaks such as cleanup suction or combination of liquid and steam breaks whose total break area is less than 1.0 ft^2 and in which the ADS system is required for depressurization, at least (n-l) ADS valves must be available

e. For breaks smaller than the equivalent flow area of one open ADS valve, at least (n-l) ADS valves must be available. However, the required number of ADS valves will be one less for each additional steam break area equivalent to the area of one open ADS valve.

3.6.1.2.2 Containment System Integrity

The following shall be considered in addressing the LOCA dynamic effects with respect to containment system integrity:

a. Leaktightness of the primary containment fission product barrier shall be ensured throughout any LOCA, unless analyses show that offsite dose consequences are within 10 CFR 50.67 guidelines or 10 CFR 100 guidelines

b. For lines that penetrate the drywell and are normally closed during operation, the inboard isolation valve shall be as close as practical to the reactor pressure vessel. This arrangement reduces the length of pipe subject to a pipe break

c. For lines that penetrate the primary containment and are open during normal operation, the outboard isolation valve shall be as close as practical to the primary containment.

3.6.1.2.3 Control Rod Insertion Capability

To maintain the ability to insert the control rods in the event of a pipe break, the control rod drive (CRD) withdrawal lines shall be protected from the dynamic effects so that no more than one in any nine-rod array is allowed to be completely crimped (totally blocked). Complete severance of withdrawal lines will not affect the rod-insert function. Protection for the CRD insertion lines is not required since a reactor pressure of 600 psig or higher can adequately insert the control rods.

3.6.1.3 Design Loading Combinations

Design criteria, design stress limits, and various loading combinations for safety class system components and equipment, including the RCPB system components, are described in detail in Subsection 3.9.2. Design criteria, design stress limits, and loading combinations for various types of pipe-whip restraints and support systems for Fermi 2 are described in Subsection 3.6.1.5 (see also References 1 through 11).

A description of analytical methods and computer codes used is given in Subsections 3.6.1.4 and 3.6.1.7, respectively.

3.6.1.4 Dynamic Analyses

3.6.1.4.1 Analytical Methods

3.6.1.4.1.1 General Description of Analytical Methods

The prediction of time-dependent and steady-thrust reaction loads caused by blowdown of subcooled, saturated, and two-phase fluid from a ruptured pipe is used in design and evaluation of dynamic effects of pipe breaks. Unsteady loads result from depressurization wave propagation that causes the various sections of pipe to be loaded with time-dependent forces. Steady blowdown thrust loads are all equivalent to a corresponding thrust applied normal to the plane of the break and opposite to fluid blowdown velocity. These loads can be computed for each section of the piping system, and corresponding external restraints can be provided if it is necessary to limit the movement of the piping system. A detailed description of the analytical assumptions and methods used to compute these blowdown loads is given in Section A of Reference 1 and in References 2 and 5.

A schematic diagram representing modeling of physical systems in pipe-whip analysis is given in Figure 3.6-1.

3.6.1.4.1.2 Blowdown Types and Associated Thrust Loads

The blowdown types and associated thrust loads considered in the analyses are summarized in this subsection.

The two components of the thrust reaction load considered are

a. Blowdown thrust - This thrust is caused by fluid acceleration from the break and static pressure in the break itself

b. Wave thrust - This thrust is caused by momentum transfer associated with decompression and compression waves (sonic waves) propagating in the various pipe sections. It is assumed that simple pipe bends and turns (without flow-area change) do not attenuate the traveling pressure waves or cause reflections.

Only the wave thrust produces reaction loads on bound pipe segments, whereas blowdown thrust applies only to the broken pipe segment. In the initial phase of a blowdown caused by a pipe rupture, both the wave and blowdown thrusts are present and they are additive. However, when the steady blowdown phase is reached, the wave thrust becomes zero and all bound pipe segment reaction loads disappear.

In designing protective devices to minimize the effects of pipe rupture, the jet impingement loads on surrounding mechanical system components, equipment, and structures were also considered to ensure that the effects of pipe rupture would not propagate to other vital plant systems.

3.6.1.4.1.3 Circumferential Breaks and Associated Thrust Loads

When analyzing a case where a single straight segment of broken pipe is attached to a pressure vessel, the magnitudes of both blowdown and wave thrust loads are computed. Depending on the state of the fluid in the piping system, nonflashing liquid or vapor phase, the resulting thrust loads will be different, as shown in Figures 3.6-2 and 3.6-3, respectively.

However, bends, friction, and flashing of near-saturated water all affect the blowdown characteristics. Therefore, for actual analyses, these factors were taken into account and the resulting time-dependent thrust force diagrams are modified as shown in Figure 3.6-4 for steam lines. Figure 3.6-4 shows a typical timethrust diagram for a line containing steam. After the initial wave thrust has died down, the blowdown thrust approaches the steady-state value.

a. Friction effects - Thrust reaction forces are attenuated by pipe friction that exerts its most direct effect on the blowdown rate. Figure 3.6-5 shows the steady-state thrust as a function of the friction coefficient (K = FL/D) for steam and saturated water. Reference 2, Section II-F, summarizes the methods used for including friction effects

b. Flashing effects - The effects of phase change are much less important for vapor flows than for low-quality saturated liquid/vapor mixtures. Therefore, methods for predicting time-dependent and steady blowdown properties of vapors are relatively straightforward. However, methods for predicting time-dependent blowdown of saturated mixtures must provide somewhat higher than expected loads for design purposes. Refer to Reference 14, Paragraph 4.2, for analytical development

c. Traveling speed of wave thrust - Flow disturbances propagate at sonic speed relative to the fluid. The sonic speed is important in predicting time-dependent flow properties before steady blowdown rates are reached. For development of sonic velocities used in the Fermi 2 design, refer to Reference 1 and Reference 2, Section II-C.

3.6.1.4.1.4 Longitudinal Breaks and Associated Thrust Loads

In the case of a longitudinal break of a pipe, the blowdown flow will come from both the upstream and the downstream directions except for lines with a dead end. For longitudinal breaks in dead-end lines, the analysis is similar to the analysis of circumferential breaks. If the longitudinal break area is sufficiently small, flow rate will be limited by the break itself; however, if the break is large, flow rate will be limited by the sum of upstream and downstream pipe areas or any applicable restriction area. The geometric character of a longitudinal pipe fracture is still relatively uncertain. Therefore, it is reasonable to consider an ideal, short nozzle-type break rather than a sharp-edged orifice-type break that would reduce the computed reaction thrust. A longitudinal break is shown in Figure 3.6-6. Figures 3.6-7 and 3.6-8 show thrusts for longitudinal breaks. In Fermi 2 piping system analyses, it is postulated that a longitudinal break area is equal to the pipe flow cross-sectional area. Refer to Reference 1, Section I of Reference 2, and Reference 14 for details on the analysis of thrust loads for longitudinal breaks.

3.6.1.4.1.5 Jet Impingement Loads

Jet impingement loads result from blowdown flow that forms a jet of fluid and imparts impact forces to pipes or other mechanical and structural target objects in its path. Analysis for components subject to jet impingement loads is described in Section D of Reference 1.

3.6.1.4.2 Modeling of Physical Systems

3.6.1.4.2.1 Circumferential Break Model

The circumferential break pipe/restraint system is modeled in the analyses such that the pipe immediately upstream of the elbow is loaded as a beam whose point of fixity is usually taken at a fitting or at the nonpiping component element such as a pump, vessel, or containment penetration. The weights of these pipes are small compared to the blowdown thrust loads; therefore, gravitational forces are neglected in the model. However, the mass of all piping, fittings, valves, or any other concentrated weight is considered in the dynamic analysis to account for the inertial effects of these masses. A schematic diagram of pipe/restraint is shown in Figure 3.6-9 for the circumferential break case.

The weight of the beam section, L, shown in Figure 3.6-9 is treated as a distributed mass. If a concentrated weight exists in the beam between the restraint and the break, it is treated in the model as an additional point mass transferred to the beam end at the break location line of action. The restraint closest to the broken end is assumed to carry the total dynamic load. No credit is taken for additional restraints, if any, along the pipe that would reduce the loading on the primary restraint.

3.6.1.4.2.2 Longitudinal Break Model

Figure 3.6-l0 shows a model in which a longitudinal break occurs along the bend of an elbow. The model elements are generally similar to those of the circumferential break. However, an additional element, the equivalent beam restraint, L_3, is present as shown in the figure. This element shares the applied load with the beam element from the instant the break occurs. The applied load in the longitudinal break case has two components. The first component, F_{BA}, acts parallel to the axis of the equivalent beam restraint as a compression force if the equivalent beam restraint ends in a true point of fixity; that is, a vessel, containment penetration, etc. If the equivalent beam restraint does not end at a point of fixity, the force F_{BA} will load some other combination of beams and equivalent beam restraint. The second component, F_{BB}, acts perpendicular to the equivalent beam restraint, L_3, and the beam, L. The equivalent beam restraint is treated in the model as a beam spring whose force is directly opposite to the thrust load. The mass, however, is treated as an additional equivalent point mass along with any other concentrated loads it may contain, applied to the end of the beam section.

3.6.1.4.2.3 Pipe Response Modes of the Model

The five pipe response modes of the model are as follows:

a. <u>First response mode</u> - The first mode of response is the free movement of the piping system before it contacts the restraint. In this mode, the energy that is not dissipated as deformation energy of the beam in the circumferential break and of the beam and equivalent beam restraint in the longitudinal break, becomes kinetic energy of the beam system

b. <u>Second response mode</u> - This response mode is initiated the instant the pipe hits the restraint. Analysis of this response mode requires a complex mathematical model because of the multilink response of the system involved. In this mode the thrust force, restraint force, and pipe-bending resistance moments all have to be considered to compute the accelerations, velocities, and displacements at the broken end of the pipe

c. Third response mode - In this mode the restraint and the bound end of the pipe have ceased to move, but the free end of the pipe is still in motion. During this period, the forces and moments of the various load elements, the energy balance, and the kinetic energy are computed as a function of the displacement. If the kinetic energy is computed to be zero or negative, the free end of the pipe is assumed to be stationary

d. Fourth response mode - In this mode the movement at the free end of the beam relative to the bound end is zero, and the computation process continues as in the third mode

e. Fifth response mode - In this mode the steady-state response of the piping system is computed. The computed steady-state load is compared to the maximum allowable restraint load.

The five modes of response listed above describe the computational process used in the dynamic analysis of pipe rupture thrusts and corresponding effects on the pipe-whip restraints. Details of the computer code used in this analysis are given in Reference 5. The results of the analyses are reported in Reference 1.

3.6.1.5 Protective Measures

Protection against the dynamic effects of a pipe rupture is provided in the form of pipe-whip restraints, equipment shields, and physical separation of piping, equipment, and instrumentation.

Detailed analyses of pipe restraints and restraint support systems, and test results of the pipe restraints installed in Fermi 2 are described in References 1 and 9, respectively, which were submitted to the AEC as topical reports. Supplemental analyses were also performed for the as-built configuration.

3.6.1.5.1 Pipe Restraint Design

3.6.1.5.1.1 Design Criteria

Pipe restraints, as differentiated from piping supports, are designed to function and carry loads for an extremely low probability of gross failures in the RCPB and other vital safety system piping. The RCPB piping integrity does not depend on the piping restraints during normal, upset, emergency, or faulted conditions as defined in paragraph NB-3113, Section III, of the ASME B&PV Code, but relies on piping supports to maintain the piping design stress values and/or piping integrity.

The pipe restraints (that is, those devices that serve only to control the movement of a ruptured pipe following gross failure) are subjected to once-in-a-lifetime loading. Local pipe and restraint deformations that occur upon impact do not further affect the integrity of the RCPB. For the purpose of design, the pipe-break event is considered to be a faulted condition and the pipe, its restraints, and structure to which the restraint is attached, are analyzed accordingly.

Piping within the broken loop shall no longer be considered part of the RCPB. Plastic deformation in the pipe is considered as a potential energy absorber. Limits of strain can be imposed that are similar to strain levels allowed in restraint plastic members. Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads.

Therefore, the design objectives governing the extent of permissible damage resulting from postulated dynamic effects of pipe whip are as follows:

a. The integrity of the primary containment system must be maintained

b. Safe shutdown and maintenance of core cooling integrity must be ensured.

To ensure the previous general design criteria of pipe restraints, the following specific design requirements must be met.

a. The restraints shall in no way increase the RCPB stresses by their presence during any mode of reactor operation or condition

b. The restraint system shall function to stop the movement of a pipe failure (gross loss of piping integrity) without allowing damage to critical components or missile generation.

3.6.1.5.1.2 Types of Pipe Restraint Components

To establish a design basis relating to material selection, fabrication, inspection, installation, quality assurance, and applicable design limits, three types of restraint hardware are defined. In addition, the structural and civil components are considered as a separate type.

a. Type I - Restraint energy absorption members - Those members that, under the influence of impacting pipes (pipe whip), will absorb energy by significant plastic deformation (e.g., U-bolts, rods, cables)

b. Type II - Restraint connecting members - Those components that form a direct link between the restraint plastic members and the structure (e.g., clevises, brackets, pins)

c. Type III - Restraint connecting member structural attachments - Those fasteners that provide the method of securing the restraint connecting members to the structure (e.g., weld attachments, bolts)

d. Type IV - Structural and civil components - Those steel and concrete structures that ultimately must carry the restraint load (e.g., sacrificial shield, trusses).

3.6.1.5.1.3 Loading Basis for Pipe Restraints

For the purpose of designing the pipe restraints as defined in Subsection 3.6.1.5.1.2, the following faulted loading combinations are used:

a. Dynamic Loading

   1. Blowdown thrust of the pipe section that impacts the restraint

2. Dynamic inertia loads of the moving pipe section that is accelerated by the blowdown thrust and impacts the restraint.

b. Static Loading

1. Maximum steady-state blowdown thrust following initial dynamic loading when pipe movement ceases

2. Effective piping weight on the restraint, if significant.

3.6.1.5.1.4 Design Basis for Pipe Restraints

The four types of pipe restraints are

Type I

a. Materials - All materials that are used to absorb energy through significant plastic deformation shall conform to

1. ASME B&PV Code Section III, Subsection NB, Class 1 Components, or

2. ASTM specifications with consideration for brittle fracture control.

b. Inspection - Inspection and identification of material shall conform to

1. ASME B&PV Code Section III, Subsection NB, Class 1 Components (Section V, Non-Destructive Examination Methods), or

2. ASTM specification procedures, including volumetric and surface inspection.

c. Design limits

1. Design local strain - The permanent deformation in metallic ductile materials shall be limited to 50 percent of the minimum actual uniform elongation based on restraint material tests for stainless steel restraint bars

2. Design steady-state load - The maximum restraining load will be limited to:

(a) 80 percent of the minimum calculated static ultimate restraint strength at the drywell design temperature for bar-type restraints

(b) 75 percent of certified minimum breaking strength for cables determined on the basis of tests (Reference 14).

3. Dynamic material mechanical properties - The material selected must exhibit tensile impact properties that are not less than

(a) 70 percent of the static percent elongation

(b) 80 percent of the statistically determined minimum total energy absorption.

Type II

a. Materials - Material selection shall conform to ASTM Specifications, including considerations for brittle fracture control

b. Inspection - Inspection shall conform to ASME/ASTM requirements or process qualification and finished parts surface inspection per ASTM methods

c. Design limits - Design limits shall be based on the following stress limits:

1. Primary stresses shall be limited to the higher of

(a) 70 percent of S_u , where S_u = minimum ultimate strength by tests or ASTM Specification

(b) $S_y + 1/3 (S_u - S_y)$, where S_y = minimum yield strength by tests or ASTM Specification.

Type III

a. Fasteners

1. Materials - Fastener materials shall conform to ASTM and ASME requirements

2. Inspection - All fasteners shall be inspected or certified per applicable ASTM and ASME specifications

3. Design limits - Same as Type II.

b. Welds

1. Materials - Weld material for attachment to carbon steel structures shall conform to AWS/ASME specification per:

(a) AWS A5.1, A5.5, or A5.17, low hydrogen electrode for metal arc welding, or

(b) AWS A5.18 or A5.20 filler metal for MIG or TIG welding.

2. Procedures - Procedures and welders shall be qualified per AWS Code D1.0 - latest edition for welding in building structures

3. Design limits - Design limits shall be based on the following stress limits:

The maximum primary weld stress intensity (two times maximum shear stress) will be limited to three times AWS or AISC allowable weld shear stress.

Type IV

Design requirements for structural equipment are not codified to the same extent as for the mechanical and electrical equipment. The industry recognizes this inadequacy and is actively working in this area. For example, standards for concrete containment are being developed by both the ASME and the American Concrete Institute (ACI). It is also impractical to "qualify" a structure as is done for electrical or instrument components. It is therefore current practice within the industry for structural requirements to be developed and specified by a qualified structural engineer, and for those requirements and methods of implementing the design to be reviewed by the NRC. Structures are designed to respond to conditions associated with the specific structure including operational and accident loadings, seismic loadings, wind loadings, and tornado loadings.

The design-basis approach of categorizing components is consistent in allowing less stringent inspection requirements for those components subject to lower stresses. Considerable strength margins exist in Types II through IV even to the limit of load capacity (fracture) of a Category I component. It is recognized that impact properties in all components must be considered since brittle-type failures could reduce the restraint system effectiveness. For details of load combinations, design limits, stress criteria, and materials specifications, see Section 3.8.

3.6.1.5.1.5 Design Basis for Seismic Guide

The normal function of a seismic guide is to support a piping system and limit deflection under seismic loading. Because of the limited space in the area of the inboard main steam isolation valve (MSIV), it was necessary in this particular case to combine the function of a seismic guide with the function of a pipe-whip restraint. Details are described in Reference 1.

The seismic guide is designed with a low clearance to maintain small deflections during seismic events, so that the containment penetration and the inboard MSIV will not be subjected to high stresses. To limit the pipe motion within the confines of available space in the event of a pipe rupture, the existing seismic guide was redesigned to include the function of a low-clearance pipe-whip restraint.

The seismic guide contains 40 crushable energy-absorbing stainless steel tubes with a 1-in. outside diameter, a 0.156-in.-thick wall, and a 10-in. length. These tubes will not be in contact with the pipes under normal and/or seismic events. However, in the event of a pipe rupture requiring the pipe-whip restraint to function, the pipe is free to work on the crushable tubes, thus dissipating its kinetic energy to the tubes.

3.6.1.5.1.6 Verification Tests for Pipe-Whip Restraints

The dynamic test program conducted by Edison with the assistance of GE verifies the adequacy of the Fermi 2 pipe-whip restraint designs. The concept of large-clearance design with plastic deformation of restraint material to absorb the kinetic energy of a whipping pipe has been proven in these tests, which are described in Reference 9.

The overall conclusion can be drawn from the results of the actual tests that sufficient conservatism exists in the analysis methods used to initially predict the effectiveness of the design concept in restraining vital piping without hindering their normal expansion and contraction in the course of plant operation. Therefore, it is concluded that the pipe-whip restraint designs are effective, with sufficient margins to meet all of the safety design requirements.

3.6.1.5.1.7 Design Basis for Recirculation System Restraints

Restraints for the recirculation system piping are the GE cable-type restraints. These restraints are discussed in Reference 14.

3.6.1.5.2 Separation and Protective Provisions for Safety- Related Systems and Equipment

3.6.1.5.2.1 Separation Criteria

Separation of safety-related mechanical and electrical systems and equipment is provided such that the General Design Criteria of 10 CFR 50 are fulfilled by providing the protection against the single-failure criterion. That is, all safety-related systems and equipment are arranged such that a single failure of any active component in a redundant system does not result in a loss of capability of the system to perform its safety function (see Section 3.12).

3.6.1.5.2.2 System Separation

The mechanical and electrical systems and equipment separation are as follows:

a. Mechanical systems and equipment - Piping for a redundant safety system is run independent of its counterpart. Supports, restraints, and mechanical components of redundant piping of the same system are not shared in common, unless it can be shown that such sharing does not significantly impair their ability to perform their safety functions.

   The systems and equipment that meet the separation criteria are as follows:

   1. LPCI
   2. CS
   3. HPCI
   4. ADS
   5. RCIC.

b. Electrical systems and equipment - The electrical portions of the following systems are affected by the separation criteria:

   1. Reactor protection
   2. HPCI
   3. CS
   4. RHR
   5. Emergency closed cooling water (ECCW)

6. RCIC.

The corresponding electrical equipment includes

1. Instrument channels
2. Trip systems
3. Trip actuators
4. Standby power sources
5. Average power range monitor
6. Intermediate range monitor.

These systems and equipment have also been designed and fabricated in accordance with the intent of IEEE Standard 279-1971 and IEEE Standard 308-1971, as applicable.

3.6.1.5.2.3 Physical Separation

The physical separation for mechanical and electrical systems and equipment is as follows:

a. Mechanical systems and equipment

   l. Mechanical equipment and piping, including control safety conduit and tubing and containment penetrations for safety-related systems, are physically separated to meet the single-failure criterion

   2. The ADS is physically separated from the HPCI system such that no portion of the HPCI influent line or HPCI steam supply line is located within the jet impingement damage distance or pipe-whip damage distance of any component considered essential to the ADS operation

   3. Provisions are made to ensure that no single failure could incapacitate both the HPCI and RCIC systems.

b. Electrical systems and equipment

   Electrical equipment and wiring for the reactor protection system (RPS) and the ECCS subsystems are physically separated under separate divisions, designated as Divisions I and II, to conform to the requirements of the single-failure criterion by arrangement and/or protective barriers.

3.6.1.5.3 Protective Shields and Jet Deflectors

Jet deflectors are provided in the drywell at the inlet of each vent pipe to prevent possible damage from jet forces, which might accompany a pipe break in the drywell. In addition, piping and electrical penetrations in the primary containment are either designed to withstand or are shielded from the jet impingement forces arising from the rupture of the largest local pipe or connection. Details of the piping penetration jet deflectors are discussed in Subsection 3.8.2.1.3.1. The sacrificial shield and the containment floor also act as shields for the pipe-whip and jet impingement forces arising from a break in the unrestrained portions of the pipe inside the drywell.

3.6.1.6 Pipe-Whip Restraint Support System

Pipe-whip restraints are provided at required locations along the length of pipes under pressure to withstand forces arising from whipping of the pipes in the event of a postulated pipe rupture. These restraints are designed so that the energy dissipated during whipping of the pipe after rupture is absorbed by plastic yielding of the restraints; this provision of absorption of energy by plastic yielding results in further reduction of the reactive force due to whipping. Depending upon the location of the restraints and configuration of the piping network, the restraints are attached directly to the sacrificial shield, through trusses to the sacrificial shield, directly to the reactor pressure vessel (RPV) pedestal, or through trusses to the RPV pedestal and drywell floor. The design of the pipe-whip restraint support system (PWRSS) is described in Reference 1.

3.6.1.6.1 Design Criteria

3.6.1.6.1.1 Design Basis

The structural analysis of the components of the PWRSS is performed using linear elastic methods. The stresses resulting from such an analysis for the load combinations involving pipe rupture forces are limited to Φfy, where f_y is the maximum stress resulting in first yielding of the structure as specified in AISC and ACI specifications, and Φ is the reserve strength factor, which depends on the type of structure and ranges in value from 0.85 to 0.95. The reserve strength factors are used to ascertain that the structure does not reach first yield under the specified load combinations. This factor also includes the effect of strength variation in materials and workmanship.

An underlying assumption in the design of the PWRSS is that only one pipe-rupture event can take place in any given instant.

The design forces for the PWRSS are derived from the results of the dynamic pipe-whip analyses, described in the preceding subsections, and expressed as equivalent static loads. In deriving these equivalent static loads, consideration is given to the following parameters:

a. Time dependence of pipe-rupture loads

b. Flexibility and damping of the components of the PWRSS

c. Second-order effects in the restraints, such as strain hardening and variation of material properties with rate of strain.

3.6.1.6.1.2 Load Combinations and Allowable Stresses

The load combinations described here involve only the loads due to pipe-whip forces and corresponding allowable stresses. The values of allowable stresses are the maximum possible values. The actual limiting values for design are dependent on the type, function, and method of construction of the particular structure; and hence, if necessary, the values of allowable stresses are suitably reduced. The load combinations and allowable stresses given are applicable to all components of the PWRSS.

a. Load Combinations for Design of PWRSS Components

Load Combination category	D	L	T_a	P_a	R	E	E'	M
Abnormal/severe environment	1.0	1.0	1.0	1.0	1.0	1.0		
Abnormal/extreme environmental	1.0	1.0	1.0	1.0	1.0		1.0	1.0

NOTES:

1. Loads not applicable to a particular structure under consideration may be deleted.

2. If for any load combination the effect of any load other than D reduces the total load, it shall be deleted from the combination.

NOTATION

D = Dead load of structure plus any other permanent loads

L = Conventional floor live loads and movable equipment loads

T_a = Thermal effects that may occur during an accident

P_a = Pressure loads that may occur during an accident

R = Statically equivalent forces arising out of effects that include jet impingement, dynamic rupture load associated with whipping pipe, and accidental thermal pipe reaction

E = Operating-basis earthquake (OBE) effects

E' = Safe-shutdown (formerly design-basis) earthquake (SSE) effects

M = Effects of missile impact

b. Allowable Stresses

1. Concrete

(a) Compression

Membrane	$0.60f_c'$
Membrane plus flexural	$0.75\ f_c'$
Local compression	$0.90\ f_c'$

(b) Shear

Permissible nominal shear stress and design of necessary shear reinforcement are as per the provisions in Chapter 11 of ACI 318-71, Building Code Requirements for Reinforced Concrete

(c) Membrane Shear

The principal stresses resulting from membrane shear and normal stresses are computed for all combinations. If the principal tension is greater than $3\sqrt{f_c'}$ in localized areas, then reinforcing steel is provided to carry the total tensile force

In addition to the specific requirements stated above, all the other provisions of ACI 318-71 apply.

2. Reinforcing Steel

   Tension $0.9\ f_y$

   Compression (load carrying) $0.9\ f_y$

3. Structural Steel

   The allowable stresses are 1.6 times those given in AISC specifications. The following requirements are also satisfied:

   (a) Allowable shear stress = $\frac{0.95\ F_y}{\sqrt{3}}$

   (b) Allowable shear stress in fillet welds = 1.6 times those given in AISC specifications.

   (c) Allowable tensile stress in a plane perpendicular to the plate thickness = $\frac{2}{3} F_y$

NOTATION

f_c' = Specified compressive strength of concrete, psi

f_y = Specified yield strength of reinforcement

F_y = Specified yield strength of structural steel

3.6.1.6.1.3 Components of Pipe-Whip Restraint Support System

The primary components in the PWRSS are

a. Pipe-whip restraints
b. Sacrificial shield
c. Trusses
d. Reactor support pedestal
e. Drywell floor.

A schematic representation of interactions among various components of the PWRSS is given in Figure 3.6-11; the relative locations of the components are shown in Figure 3.6-12.

Some descriptions of structures, analytical methods, loads, and stresses for the design of PWRSS are given in Reference 15.

3.6.1.7 Computer Programs Used in Analysis and Design of Pipe- Whip Restraints and Restraint Support Systems

The computer programs used in the analysis and design of pipe-whip restraints and restraint support systems are:

a. PDA - PDA (Pipe Dynamic Analysis Program for Pipe Rupture Movement) was developed by GE to solve nonlinear, two-dimensional dynamic equations of pipe-whip and the restraining device motions. This program is used to generate the time-dependent forcing functions for the design of pipe-whip restraint devices. For a detailed program description refer to Reference 5

b. INDIA - INDIA (Interaction Diagram for Reinforced Concrete Members) was developed and is maintained by Sargent & Lundy (S&L). It has been designed to plot the bending moment-axial load interaction diagram for reinforced-concrete members

   Interaction diagrams can be obtained for any of the criteria of design, ultimate strength, yield strength, or working stress. Both compression and tension axial loads are considered, as well as positive and negative moments

   The program output includes a listing of the results for the specified design criterion. The interaction diagram is plotted, if so desired

c. KALSHEL - KALSHEL (Kalnins' Shell of Revolution) was developed by A. Kalnins of Lehigh University and is maintained by S&L. The program analyzes thin axisymmetric shells of revolution for arbitrary load conditions. It is based on a computation scheme set forth in the publication by A. Kalnins, "Analysis of Shells of Revolution Subjected to Symmetrical and Nonsymmetrical Loads," Journal of Applied Mechanics, ASME, Vol. 31, September, 1969, pp. 467-476. For the solution, the general boundary value problem of a rotationally symmetric shell is transformed into a new system of first-order ordinary differential equations. An Adams method of numerical integration is used as a basis for the solution of transformed equations

   The shell wall may vary in thickness along the meridian and may consist of up to four layers of different isotropic or orthotropic materials. Branch shells may be connected to the main shell. Surface loads and live loads in the radial, tangential, and/or meridional directions and meridional moments may be considered in the analysis. Temperature distributions that may be considered to vary linearly across the thickness may also be considered. All loads may be asymmetric

   The program output includes the shell displacements in the radial, tangential, and meridional directions, meridional rotations, meridional moment, hoop moment, meridional force, hoop force, transverse shear force, and twist shear force. Outer fiber stresses calculated from the stress resultants may also be obtained. Sargent & Lundy has modified the program to sum the displacement and stress resultants of the individual Fourier harmonics along meridians at specified angles

d. SLSAP - SLSAP (Sargent & Lundy Structural Analysis Program) was developed by E. L. Wilson of the University of California, Berkeley, and is maintained by S&L. The program uses the stiffness matrix method to analyze two-and three-dimensional frames, trusses, and grids, three-dimensional elastic axially symmetrical solids, plates, and shells for arbitrary static loads. Dynamic analyses for frequencies and mode shapes, spectral analysis, and numerical integration analyses are also possible

   The program allows materials with arbitrary elastic constants, combined loading, rigid members, elastic supports, and a combination of different element types

   The program output includes displacement and rotations of all joints or nodes, forces or stresses in members or elements, frequencies and mode shapes, and dynamic response in terms of displacements and forces

e. SOR-III - SOR-III (Shell of Revolution) was developed by Knolls Atomic Power Laboratory for the AEC. It is maintained by S&L. The program analyzes thin shells of revolution subjected to axisymmetric loading by numerically integrating the governing differential equations using a generalized Adams-Moulton method

   Arbitrary distribution of normal, tangential, and moment surface loadings, as well as edge forces and deflections, may be considered in the axisymmetric loadings. Input of boundary conditions allows for the consideration of elastic support conditions. The effect of temperature variations along the meridian or across the thickness is also considered

   The program output includes shell displacements, outer fiber stresses and strains, and stress resultants

f. STRESS-II - STRESS-II (Structural Engineering Systems Solver) was developed by Massachusetts Institute of Technology and is maintained by University Computing Company. It uses the stiffness matrix method to analyze plane and space trusses and frames and plane grids

   The structure can be analyzed for arbitrary joint loads, member loads, temperature changes, and joint displacements. A plotting feature is available with the program

   The output includes joint displacements, equilibrium check, and reactions and member forces

g. TEMCO-III - TEMCO-III (Reinforced Concrete Sections Under Eccentric Loads and Thermal Gradients) was developed and is maintained by S&L. It analyzes reinforced-concrete sections subjected to combined external loads and thermal gradients. The analysis may be done assuming either a cracked or an uncracked section. Temperature effects are induced in the section by reactions created by translational or rotational restraints

   The analysis may be done for separate or combined action of tensile or compressive axial force, shear force, bending moment, and thermal gradients

The program output includes the location of the neutral axis, stresses in the steel and concrete, and an equilibrium check

h. Additional computer program descriptions are given in Section 3.13.

3.6.2 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping Outside Containment

The evaluation of pipe breaks of high-energy systems outside the containment includes the main steam and feedwater lines, the HPCI system, the RCIC system, the RWCU system, reactor building heating steam lines, and the CRD system.

Evaluation of the effects of through-wall leakage cracks in moderate-energy piping systems is also reported in this section. The evaluation takes into account the potential damaging effects of either water flooding or spraying from the pipe crack, and considers the overall capability of achieving reactor shutdown and maintaining a cold-shutdown condition.

The plot plan with the relative sizes of major structures is shown in Figure 1.2-5. The reactor portion of the reactor/ auxiliary building, including the primary containment, contains most of the Level I systems and components. Areas within the auxiliary portion of the reactor/auxiliary building that contain Level I systems and components are

a. Main control room and associated heating, ventilating and air conditioning (HVAC) room

b. Switchgear room

c. DC power supply rooms

d. Relay rooms

e. Cable spreading rooms

f. Standby gas treatment compartments

g. HVAC rooms.

The standby ac power system, residual heat removal service water (RHRSW) pumps and emergency equipment service water (EESW) pumps are housed in the RHR complex, a physically separate structure.

The reactor/auxiliary building is mounted on a common foundation in which the reactor portion is separated from the auxiliary portion by a sealed wall. The sealed wall between the reactor building and auxiliary building is for secondary containment purposes. Piping systems whose operating pressure and temperature conditions are consistent with a high-energy classification, as defined in Subsection 3.6.2.1.2.1, are listed in Subsection 3.6.2.1.3.

In all cases investigated, conservatism was exercised in determining the consequences resulting from the postulated pipe break. These results were in turn used to develop design provisions that would provide the necessary protection to mitigate any adverse effects on the ability to achieve a safe reactor shutdown. These design provisions are detailed in the sections describing the adverse conditions they were intended to mitigate, and are summarized in Section 3.6.2.4. The modifications to plant design called out herein provide assurance against unacceptable consequences of the postulated pipe breaks.

The reactor/auxiliary building floor plans are shown in Figures 3.6-13 through 3.6-19. The floor plans include a room, area, and compartment number system that is referred to throughout this section, particularly in the discussion on the environmental effects of high-energy pipe breaks and the moderate-energy pipe breaks. By cross reference, an easy method is provided for following the discussion.

3.6.2.1 Design-Basis Pipe Break Evaluation

The design-basis pipe break was postulated to occur in all high-energy piping systems defined in Subsection 3.6.2.1.2.1. Throughwall leakage cracks were postulated to occur in all moderate-energy piping systems defined in Subsection 3.6.2.1.2.l.

3.6.2.1.1 Approach To Evaluation

The approach to the evaluation of postulated breaks in high-energy fluid systems and through-wall leakage cracks in moderate energy fluid systems is described in this section. The approach takes into consideration the rules and guidance provided in the following:

a. AEC letter dated December 15, 1972, and the errata sheet dated January 12, 1973 (Reference 16)

b. AEC letter dated July 12, 1973 (Reference 17)

c. Branch Technical Position ASB 3-1 (formerly BTP APCSB 3-1), "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," July 1981 (Reference 18)

d. Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," July 1981 (Reference 19)

In instances where the provisions of the above documents differ, to the extent practical based on the stage of design and construction of the plant, the evaluation methods are based on the guidance provided in the Items c. and d. Branch Technical Positions.

3.6.2.1.1.1 High-Energy Fluid Systems

A summary of the basic approach used in the evaluation of the consequences of high-energy pipe breaks is as follows:

a. Pipe-break locations are as given in Subsection 3.6.2.1.2.2

b. Evaluation of the direct consequences of the break on systems and components required for a safe cold shutdown, taking into consideration the effects of pipe whip, jet impingement, flooding, and environmental conditions (temperature, pressure, and humidity)

c. Evaluation of the ability of Category I structures to withstand the effects of the pipe break, taking into consideration the effects of pipe whip, jet impingement, flooding, pressure, and temperature in combination with the specified seismic event loads and normal plant loads

d. A determination of the remaining systems and components available to ensure and maintain a safe cold shutdown. This determination is made in accordance with Subsection 3.6.2.1.4 and Table 3.6-1. In making this determination, the following assumptions were made with regard to the operability of these systems and components:

1. If the pipe break directly results in a turbine-generator trip or an RPS trip (scram), offsite power is assumed to be unavailable.
2. A single component failure is assumed to occur in addition to the postulated pipe break and any other system or component failures resulting as a direct consequence of this pipe break.
3. Operator action is assumed l0 minutes after pipe break.

e. Assurance that the escape of steam, water, and heat from structures enclosing the ruptured pipe does not prevent occupation of the main control room, nor does it impair the ability of instrumentation, electric power supplies, components, and controls to initiate, actuate, and complete a safety action. In this regard, a loss of redundancy, but not the loss of a function, is permissible.

3.6.2.1.1.2 <u>Moderate-Energy Fluid Systems</u>

A summary of the basic approach used in the evaluation of the consequences of moderate-energy through-wall leakage cracks is given below.

a. For piping systems located in areas containing systems and components important to safety, through-wall leakage cracks were postulated at the most adverse locations to determine the effects from both water spray and flooding. In areas where safety systems and components are not located, the effects of flooding in other areas were considered

b. Evaluation of direct consequences of leakage cracks on systems and components, taking into account the effects of resulting water spray and flooding

c. A determination was made of systems and components available to ensure and maintain a safe cold shutdown

   This determination was made in accordance with Subsection 3.6.2.1.4 and Table 3.6-1. In making this determination, the following assumptions were made with regard to the operability of these systems and components:

   1. If water spray or flooding from the pipe crack directly results in a turbine-generator trip or an RPS trip (scram), offsite power was assumed to be unavailable
   2. A single component failure was assumed to occur in addition to the system or component failures resulting from water spray or flooding. In the event the pipe crack is assumed to occur in one of two or more redundant trains of a dual-purpose essential system, failure of components in the other train or trains of that system was not assumed

3. Operator action was assumed l0 minutes after the pipe crack.

d. It is established that water spray or flooding does not prevent occupation of the main control room, nor does it impair the ability of instrumentation, electric power supplies, components, and controls to initiate, actuate, and complete a safety action

e. In the event a safe shutdown cannot be ensured considering those systems failed or assumed to have failed as a consequence of the leakage cracks, plant modifications were instituted or protection was provided to those systems or components.

3.6.2.1.2 Design-Basis Pipe Break Criteria

3.6.2.1.2.1 Definition of High-Energy Fluid Systems

High-energy fluid systems include those systems that under normal or upset plant conditions are pressurized during operation and one of the following conditions exists:

a. The maximum operating temperature exceeds 200°F

b. The maximum operating pressure exceeds 275 psig.

A fluid system meeting the above definition less than 2 percent of the time is not considered a high-energy fluid system.

Moderate-energy fluid systems include those that during normal plant conditions are either in operation or maintained pressurized under conditions where both of the following conditions exist:

a. The maximum operating temperature is 200°F or less

b. The maximum operating pressure is 275 psig or less.

3.6.2.1.2.2 Design-Basis High-Energy Break/Crack Locations

Break locations are postulated outside the containment in accordance with the following criteria:

a. ASME B&PV Code Section III, Class 1, pipe breaks are postulated to occur at the following locations in each piping run or branch run:

1. The terminal ends

2. Any intermediate locations between terminal ends, where the maximum stress range as calculated by Equation 10, and either Equation 12 or 13, of NB-3653, derived on an elastically calculated basis under the loadings associated with normal and upset plant conditions, exceeds 2.4 S_m

3. Any intermediate location between terminal ends where the cumulative usage factor derived from the pipe fatigue analysis and based on all normal, upset, and testing plant conditions exceeds 0.1

4. Arbitrary intermediate pipe breaks no longer need to be postulated, per Generic Letter 87-11

b. ASME B&PV Code Section III, Class 2 and 3, pipe breaks are postulated to occur at the following locations in each piping run or branch run:

   1. The terminal ends

   2. Any intermediate location between terminal ends, where the stresses as calculated by Equations 9 and 10 of NC/ND-3652, derived on an elastically calculated basis under the loadings associated with normal and upset plant conditions, exceed 0.8 ($1.2\ S_h + S_a$)

   3. Arbitrary intermediate pipe breaks no loner need to be postulated, per Generic Letter 87-11

c. In situations where detailed stress analyses of ASME B&PV Code Section III, Class 2 and 3, piping systems are not used to select postulated break locations, and, in the case of those high-energy systems outside the containment that are not analyzed to ASME III Code requirements, break locations are conservatively assumed to occur at all fitting welds where a break has the potential of causing unacceptable damage to systems and/or components necessary to effect and/or maintain a safe shutdown.

d. The break analysis of seismically analyzed non-ASME class piping is postulated according to the requirements for ASME Class 2 and 3 piping.

For those portions of the piping passing through the primary containment penetrations and extending to the first outboard isolation valve, pipe breaks were not postulated since the piping was conservatively designed and restrained beyond the valve such that, in the event of a postulated pipe break outside the containment, the transmitted pipe loads will neither impair the operability of the valve nor affect the integrity of the piping of the containment penetration.

Design criteria for piping between the primary containment and outboard isolation valves provide for maximum stresses considering all normal and upset conditions as calculated by the equations in Paragraph NB-3653 of ASME B&PV Code Section III, which may not exceed the following limits:

a. If Equation 10 results in $S < 2.4\ S_m$, no other requirement need be met

b. If Equation 10 results in $S > 2.4\ S_m$, then Equations 12 and 13 must result in $S < 2.4\ S_m$ and Equation 14 must yield a value of $U < 0.1$.

3.6.2.1.2.3 Design-Basis Break Types and Orientation

The following high-energy breaks are postulated at the locations described in Subsection 3.6.2.1.2.2:

a. Circumferential breaks in piping runs and branch runs exceeding 1 in. nominal pipe size

b. Longitudinal breaks in piping runs and branch runs 4 in. nominal pipe size and larger

c. Longitudinal breaks are not postulated at terminal ends.

Longitudinal breaks are considered parallel to the axis of the pipe and oriented at any point around the pipe circumference.

Circumferential breaks are considered to be perpendicular to the axis of the pipe.

The break area is equal to the internal cross-sectional area of the ruptured pipe in the case of circumferential breaks and longitudinal breaks. Longitudinal breaks extend a distance of one diameter on each side of the break location.

3.6.2.1.2.4 Design-Basis Through-Wall Leakage Cracks

Through-wall leakage cracks in piping exceeding 1 in. nominal pipe size were generally postulated at the most adverse locations in moderate-energy piping systems located in areas that contain systems and components important to safety, but in which no high-energy systems are present. However, through-wall leakage cracks need not be postulated in portions of piping where the calculated stresses satisfy BTP MEB 3-1 (reference 19) exclusion criteria. These through-wall leakage cracks were assumed to be half the pipe diameter in length and half the pipe wall thickness in width.

3.6.2.1.3 Identification of Energy Systems

The high- and moderate-energy systems included in this evaluation are identified below.

3.6.2.1.3.1 High-Energy Piping Systems

The piping systems located inside the reactor/auxiliary building but outside the primary containment and meeting the definition of high-energy systems, defined in Subsection 3.6.2.1.2.1, are

a. Main steam

b. Feedwater

c. High-pressure coolant injection system steam supply

d. Reactor core isolation cooling system steam supply

e. Reactor water cleanup

f. Control rod drive insert and withdrawal lines and charging line

g. Reactor building heating steam lines.

The piping systems listed below have normal/upset pressure and/or temperature conditions that fall into the high energy category; however, since these systems are operated less than 2 percent of the time, they are not considered high-energy systems. This is consistent with the definitions presented in Subsection (3.6.2.1.2.1).

These systems are

a. Residual heat removal system

b. Core spray system

c. Reactor core isolation cooling system discharge

d. High pressure coolant injection system discharge.

3.6.2.1.3.2 Moderate-Energy Systems

Appropriate portions of the following piping systems meeting the definition of moderate-energy systems as defined in Subsection (3.6.2.1.2.1) were evaluated:

a. Residual heat removal system

b. Core spray system

c. High-pressure coolant injection system

d. Fire protection system

e. Reactor core isolation cooling system

f. Fuel pool cleanup system

g. Reactor building closed cooling water system

h. Service water

i. Emergency equipment cooling water

j. Reactor water cleanup system

k. Control rod drive system

l. Torus water management system

m. Chilled water system

n. Reactor building heating steam system

o. Supplemental cooling chilled water system.

3.6.2.1.4 Identification of Systems and Components Required for Safe Shutdown

The systems and components that contribute to attaining and maintaining a safe shutdown are listed in Table 3.6-1. The listing is broken down into two categories. The first, General Requirements, indicates those systems or components that are required regardless of the piping break being evaluated. The second, Specific Requirements, indicates the additional systems and components required for specified pipe breaks (i.e., main steam line, feedwater line, etc.).

These systems and components were evaluated with respect to the effects of a postulated break of a high- or moderate-energy fluid system.

3.6.2.1.5 Assessment of Acceptability

3.6.2.1.5.1 Components and Equipment

From the approach to evaluation defined in Subsection 3.6.2.1.1, the component and/or equipment was evaluated and an assessment made of its acceptability to the hypothetical accident by the following:

a. The loss of function of a component is acceptable if an analysis can show that a redundant component or backup system is available to perform the component's safety function and to ensure safe reactor shutdown

b. An evaluation of a component's capability to perform is based on its ability to function in the environmental conditions of flooding present after the postulated pipe break

c. An evaluation of a component's capability to perform is based on its ability to withstand the impact forces of an impacting pipe. Throughout the evaluation, the impacted component is conservatively assumed incapable of performing its function. An exception to this assumption is the case where one pipe impacts another pipe of equal or greater size and equal or greater wall thickness

d. An evaluation of a component's capability to perform is based on its ability to withstand the environmental conditions of pressure, temperature, and humidity during blowdown compared to the allowable pressure, temperature and humidity conditions of the equipment design specifications and/or test qualifications

e. An evaluation of a component's capability to perform is based on its ability to function in the conditions of high-energy jet impingement.

3.6.2.1.5.2 Structures

In accordance with the approach to evaluation (Subsection (3.6.2.1.1), plant structures and structural components have been analyzed to demonstrate ability to withstand the pipe whip impact, jet impingement, temperature, pressurization, and flooding hydrostatic loads resulting from postulated ruptures. Plant structures include those located within the reactor/auxiliary building. The overall criteria governing acceptability of structural loads resulting from postulated ruptures are as follows.

a. Damage to any structure caused by consequences of a postulated rupture, either directly or indirectly through failure of an adjacent structure, may not impair the function of any systems or equipment required to place and maintain the reactor in a cold-shutdown condition

b. The design leaktightness of the primary containment shall be preserved in the event of a postulated rupture

c. The structural integrity of the main control room to achieve a safe cold shutdown shall be preserved in the event of a postulated rupture.

The criteria for acceptability of loads on structural components resulting from postulated ruptures are given in Subsection (3.8.4.5.1).

A dynamic response amplification factor of 2.0 was used to account for the dynamic effects of impact loading when used in conjunction with a static evaluation. In those cases where a dynamic evaluation was performed, the amplification factor was explicitly determined through the dynamic analysis techniques.

In the analysis of structural components such as single beams or slabs whose failure would not jeopardize the overall structural integrity, the effects of direct stress, flexure, shear, buckling, and of the reversal of normal design loads due to pipe rupture were considered. These analyses were generally performed using limit analysis techniques, such as collapse load analysis for beams and frames and yield line theory for concrete slabs, which account for resistance of structural elements into their plastic range. The allowable loads were determined on the basis of the maximum ductility factors in Table 3.6-2, derived from Reference 20. The maximum deflections under the applied loads did not exceed the applicable ductility factor times the deflection at first yield in the structure.

Maximum section strength of concrete structures was computed using the ultimate strength design method. Maximum section strength of steel members was based on the assumption of elastic-perfectly-plastic material properties and the plastic design criteria in References 21 and 22. Material yield strength was multiplied by the dynamic increase factors specified in Table 3.6-3, derived from Reference 23 for analyses under rapidly applied pipe rupture loads. Statistical variation in material properties and elevated temperature effects was accounted for in a conservative manner.

The methods used in the structural analyses are presented in Subsection 3.6.3.

3.6.2.1.6 Identification of Analysis Requirements

Not all postulated pipe-break locations in the main steam or feedwater lines were subjected to a dynamic pipe whip analysis since, under certain conditions, the loss of a component is acceptable (Subsection 3.6.2.1.5.1).

The pipe ruptures are assumed at locations where the consequence of the pipe whip, either due to the longitudinal or the circumferential pipe ruptures, has the worst potential effect with respect to a particular system or component required for safe shutdown (Table 3.6-1). All areas of postulated pipe break were conservatively examined. For areas where the damage levels were acceptable, a further evaluation was not required. For areas where a damage potential existed and this damage would preclude the safe shutdown of the reactor, these areas were so listed and further evaluated.

One or more of the exclusion criteria defined in Table 3.6-4 and listed below were used to locate areas where no damage potential exists from pipe whip or jet impingement, or where damage would not preclude the safe reactor shutdown or maintenance of primary containment:

a. Separation

b. Distance

c. Redundancy

d. Backup

e. Self-elimination

f. Size

g. Low pressure

h. Barrier

i. Testing condition

j. Scarcity of usage

k. Safe area

l. Minimum size

3.6.2.2 High-Energy Pipe Break Analyses

The high-energy pipe-break analyses for those systems identified in Subsection 3.6.2.1.3.1 are reported in this section.

3.6.2.2.1 Main Steam Line Break in Steam Tunnel

The evaluation of the consequences of a break in the main steam line was carried out as described below.

Breaks of main steam lines in the turbine building were not subject to detailed evaluation because the equipment located in the Turbine Building steam tunnel is either not required for safe shutdown or has been analyzed and found to be able to perform its shutdown function in the conditions present after a break. The only consequences of such a break would be the backflow of steam into the auxiliary building comparable to that resulting from a break in the steam tunnel.

3.6.2.2.1.1 Review of Potential Damage

A review of the potential damage resulting from the break of a main steam line was carried out in accordance with Subsection 3.6.2.1.6 to identify the need for a dynamic pipe whip or jet impingement analysis. The review took into consideration the equipment required to ensure a safe shutdown as discussed in Subsection 3.4 and summarized in Table 3.6-1. The results of this review are documented in Reference 24.

3.6.2.2.1.2 Pipe Break Analysis

The main steam lines outside the containment are routed in an enclosed concrete tunnel through the auxiliary building and into the turbine building, as shown in Figure 3.6-20. The steam tunnel, which serves to isolate the main steam lines from most of the plant safety-related equipment, is provided with relief doors to alleviate pressures that would result in the event of a pipe rupture. In accordance with the criteria given in Subsection 3.6.2.1.2.2, longitudinal and circumferential design-basis ruptures were postulated at each end of each elbow in the main steam lines between the outboard isolation valves and the steam tunnel exit to the turbine building. Critical crack breaks were also postulated at all adverse locations in this piping. As an alternative to postulating ruptures between the containment and outboard isolation valves, the piping was designed to meet the normal operating stress criteria presented in Subsection 3.6.2.1.2.2, and was provided with rigid restraints to limit transmission of bending and torsional loads through the valves in the event of a downstream rupture. The postulated rupture locations are shown in Figure 3.6-20.

Analyses were performed to assess quantitatively the pipe whip, jet impingement, compartment pressure and steam environment effects of these postulated ruptures. Since the routing of all four main steam lines in the tunnel is nearly identical, in most cases the analyses were performed for only one line and the results extrapolated to the other three. The predicted rupture effects were evaluated to identify cases in which unacceptable damage to structures, systems or components required for shutdown could result. Finally, designs were developed for modifications required to prevent the occurrence of any unacceptable damage.

A description of the rupture effects analyses, the damage evaluations, and the required modifications for the main steam lines in the steam tunnel is given in the following sections.

3.6.2.2.1.2.1 Short-Term Blowdown Analysis

Blowdown analyses were performed for postulated main steam line breaks using the methods presented in Reference 25. The resulting thrust time-histories were then used as input for subsequent pipe whip and jet impingement analyses, using the methods outlined in Subsection 3.6.3.

To determine conservatively the thrusts resulting from the postulated ruptures, the following assumptions were made:

a. The reservoir pressures in the main steam system were assumed to be 1060 psia

b. On the reactor side of the breaks, the flow limiters were assumed to be the only resistance to flow

c. On the turbine side of the breaks, the lines between the break and the header were considered the only resistance to flow.

For the analyses, the RPV and the main steam header were assumed to be reservoirs. Since the routing of the four main steam lines in the area of interest is essentially the same, only one line was analyzed. The analyses were carried out for a time sufficient to allow for the use time of all restraint reactions. Typical thrust time histories resulting from the analyses are shown in Figure 3.6-21.

3.6.2.2.1.2.2 Pressurization and Environmental Analyses

These analyses were performed to predict maximum compartment pressures and steam environment conditions resulting from postulated design-basis ruptures in the steam tunnel. The tunnel is provided with two sets of pressure relief doors for venting. The upper set of pressure relief doors opens into the turbine building; the lower set opens into the first floor of the auxiliary building.

The environmental analyses were evaluated for uprated power conditions. Changes in the governing parameters were used to scale the affected environmental conditions. This resulted in a small increase in peak temperatures and pressure.

Immediately after a postulated main steam line break in the steam tunnel, saturated steam will flow from the break. However, due to the rise in reactor water level during blowdown, a flashing two-phase mixture of steam and water will soon begin to flow out of the break. The mass flow rate of the two-phase mixture is considerably higher than that of the saturated steam and continues until terminated by the closure of the MSIV.

The steam exiting from the break will pressurize the tunnel, and the pressure relief doors will be forced open by the pressure differential between the tunnel and the adjacent compartments. Within a few milliseconds, a mixture of steam and air will be flowing through the upper and lower sets of pressure relief doors into the turbine building and the first floor auxiliary building.

The operating condition analyzed is based on MSIV closure time of 10.5 sec. The break mass used in the calculation is shown in Figure 3.6-22 (Reference 26). Flow from the upstream side of the break was calculated assuming critical flow at the flow limiter of the broken line. Flow from the downstream side of the break is supplied by the 52 in. manifold which is in turn supplied by the three unbroken steam lines. Flow through the downstream side of the break is limited by critical flow at the end of the broken pipe.

A break in one of the main steam lines in the steam tunnel would affect only the steam tunnel, first floor Auxiliary Building, and the Turbine Building (Figure 3.6-23). Break mass and energy would be vented directly to the first floor Auxiliary Building through the lower steam tunnel pressure relief doors and then to the Turbine Building through large openings in the east wall of the first floor Auxiliary Building. Mass and energy would also be released directly from the steam tunnel to the Turbine Building second floor through the upper pressure relief doors.

Isolation of the main steam line break would be initiated almost immediately as the pressure drop across the flow restrictor for the broken line exceeds the setpoint of redundant pressure differential trip units, which send a signal to the nuclear steam supply systems (NSSS). The NSSS deenergize the main steam isolation valve solenoids for the broken and unbroken lines initiating valve closure. In addition, redundant steam tunnel high temperature leak detection trip units would also initiate MSIV closure.

The environmental responses for the steam tunnel and first floor Auxiliary Building due to a main steam line break are discussed in this section. The Turbine Building second floor confined area is included in the main steam line break model. However, the environmental response of this area is not discussed since failure of the safety related instrumentation and third MSIVs located in the Turbine Building will not prevent safe shutdown of the plant.

The plot of break flow versus time used in the computer model is shown in Figure 3.6-22. The "steps" in this plot correspond to the times at which a two-phase mixture reaches the upstream and downstream ends of the break. As can be seen in the plot, closure of the isolation valve does not begin to affect the break flow until approximately 8.6 seconds. The MSIV is closed at 10.5 seconds. However, the break flow is continuous to 13.0 seconds due to expansion of the inventory of steam and water in the piping downstream of the MSIV.

Pressure response versus time plots are shown in Figures 3.6-24 and 3.6-27 for the steam tunnel and Auxiliary Building first floor. The peak pressure value of 5.1 psig occurs in the steam tunnel at 8.6 seconds after the break. The time corresponds to the end of the highest plateau of the break flow curve. The pressure profile decreases from this peak to a negative pressure at 13.0 seconds due to condensation of the steam on cooler wall and floor surfaces.

The peak first floor Auxiliary Building pressure of 0.9 psig occurs at approximately 0.8 second after the break.

The plots of temperature versus time for the steam tunnel and first floor Auxiliary Building are shown in Figures 3.6-25 and 3.6-28, respectively. The temperatures in both rooms climb to the saturation temperature for the given room pressure as the air is exhausted through the vent openings. The peak temperatures are 228°F and 215°F for the steam tunnel and first floor Auxiliary Building, respectively. The relative humidity in both rooms reaches 100 percent within 0.5 second and remains at this level for the duration of the evaluation. Plots of humidity versus time are shown in Figures 3.6-26 and 3.6-29. The effect of the resultant temperature and humidity on safe shutdown equipment has been evaluated.

3.6.2.2.1.2.3 Pipe Whip Evaluation

Analyses were performed to assess pipe whip consequences on safety-related structures, systems, and components in the steam tunnel. Since most considerations and potential problems in this area are common to the main steam and feedwater lines, these systems will be discussed together in this section.

The calculated blowdown thrust forces for main steam and feedwater ruptures are given earlier in this subsection. The methods used in the pipe whip analyses are given in Subsection 3.6.3. Details of the structural evaluation for pipe whip impact are also given in Subsection 3.6.3; a brief summary of results of this evaluation follows:

a. Loads from pipe whip impact from main steam and feedwater line breaks could cause failure of the lower tunnel floor at elevation 583 ft 6 in.

b. Loads from pipe whip impact from a main steam line break could cause failure of the 4-ft 4-in. west wall between the steam tunnel and reactor building.

c. Pipe whip from a break of either line could induce unacceptable stresses in the isolation valves on the pipe between the valves and the primary containment

d. The remaining steam tunnel walls, the upper tunnel floor at elevation 626 ft 6 in., and the tunnel ceiling are all adequately designed to withstand pipe whip impact.

Designs were developed for main steam and feedwater line pipe whip restraints as described later in this subsection. The restraints are intended to prevent occurrence of the unacceptable consequences identified in Items a and b above. In addition, the combined action of the pipe whip restraints and the anchor framework just outside of main steam and feedwater flued heads (anchor framework is designed for pipe-break loads) prevents damage to the isolation valves, to the containment penetrations, and to the containment shell, caused by the transmission of bending and torsional loads through the piping in the event of a postulated rupture.

3.6.2.2.1.2.4 Jet Impingement Evaluation

Jet impingement effects were postulated for the main steam and feedwater line breaks in the vicinity of the isolation valves. The dynamic force on each valve was calculated in accordance with the methods presented in Subsection 3.6.3 and a stress evaluation of each valve and its interconnected piping was undertaken. It was determined that, although the MSIVs and feedwater check valves could safely withstand the dynamic impingement force, the HPCI, steam drain, and RCIC isolation valves' motor operator linkage and structure would be subjected to stress levels above the allowable value of 2 S_m (42,000 psia). In addition, the tunnel floor at elevation 583 ft 6 in. would fail as a result of jet impingement. A direct jet impingement would cause failure of the lower pressure relief doors; however, the jet would not cause failure to the 4-ft 4-in. shield wall outside the doors in the auxiliary building. The pressure relief doors are not required for safe shutdown.

To protect against the loss of function of the HPCI, RCIC, and steam drain isolation valves as a result of the postulated pipe break, jet impingement barriers were incorporated as a part of the pipe restraint system as shown in Figure 3.6-30, Sheets 1 through 3. A jet impingement barrier is also provided to protect the tunnel floor.

3.6.2.2.1.2.5 Structural Evaluation

Steam Tunnel Description

The lower portion of the steam tunnel (Figures 3.6-31 and 3.6-32) is 32 ft 9 in. long by 30 ft 0 in. wide with a 2-ft-thick slab floor at elevation 583 feet 6 in. All floor and wall slabs are doubly reinforced. The lower floor slab contains No. 7 steel reinforcing bars at 12 in. and is supported on 27WF160 and 27WF145 I-beams spaced 7°30' running in a radial pattern from the containment center. The north and south walls are reinforced concrete 4 ft 8 in. thick containing No. 9 steel reinforcing bars at 12 in. The west wall (next to the containment) is 4 ft 4 in. thick with No. 7 steel reinforcing bars at 12 in. The east wall contains 20 pressure relief panels 3 ft 6 in. by 5 ft 6 in. Directly outside the pressure relief panels, in the auxiliary building, is a 4-ft 4-in.-thick reinforced-concrete wall containing No. 8 steel reinforcing bars at 12 in. Joining this wall are 3-ft-thick side walls containing No. 7 steel reinforcing bars at 12 in. This outside structure provides radiation shielding.

Entrance to the lower portion of the steam tunnel may be made through a personnel door or through an equipment passage. The 3 x 7 ft personnel door is accessible to the outside through a side alcove. It is a seal-tight steel door, designed to withstand a 2.5 psig inward pressure and an outward pressure greater than 7 psi. The 6 x 8 ft equipment passage may be opened from the inside only by removing solid concrete shield blocks, unbolting a 3/8-in. steel plate, and then removing concrete shielding plank from the side of the aisle. This closure was designed for a 2.5 psi inward pressure and outward pressure greater than 7 psi (Reference 26).

Effects of Pipe Whip and Jet Impingement

The evaluation of tunnel structural elements for pipe whip impact and jet impingement loads was based on the criteria given in Subsection 3.6.2.1.5.2 and the methods given in Subsection 3.6.3.

The calculated blowdown thrust forces for main steam and feedwater ruptures, the methods for determination of jet impingement loads from these thrust forces, and the methods used in the pipe whip analyses are also given in Subsection 3.6.3. Yield line theory was used for analysis of concrete slabs (References 28 through 31).

Pipe Whip Restraints and Jet Impingement Shields

The main steam, feedwater, and HPCI steam piping in the steam tunnel are equipped with two restraint assemblies. These assemblies, acting in conjunction with the anchor framework just outside the main steam, feedwater, and HPCI steam flued heads, prevent the occurrence of the unacceptable consequences identified earlier in this section. Final designs are shown in Figure 3.6-30, Sheets 1 through 3, and are discussed below.

The restraint assemblies each consist of an assembly of six elastically designed plane frames situated around the main steam, feedwater, and HPCI steam piping immediately adjacent to the drywell penetrations, and an assembly of six energy-absorbing U-bolt devices situated on the main steam and feedwater lines directly above and east of the lower frames. The lower framework assembly is equipped with jet impingement shielding. Functions of these assemblies are listed below.

a. The lower assembly frames act as normal operating pipe supports and as pipe whip restraints. Lateral and vertical pipe motion is prevented; axial motion is permitted. The assembly frames are shimmed to fit around the pipes, allowing for out-of-roundness and diametric expansion

b. The lower assembly frames and shields prevent the following:

    1. Pipe whip onto, and subsequent failure of, the steam tunnel floor at elevation 583 ft 6 in.

    2. Jet impingement onto, and subsequent failure of, the steam tunnel floor at elevation 583 ft 6 in., and the HPCI, RCIC, and steam drain isolation valves

    3. Overloading of the isolation valves, piping between the isolation valves and the drywell penetrations, the drywell penetrations, and the containment shell, caused by transmission of pipe-rupture loads by the main steam, feedwater, and HPCI steam piping, acting individually or with the upper assembly.

c. The upper assembly restraints clear the pipes during all normal operating conditions, and act only in the event of a rupture. They prevent the following:

    1. Pipe whip onto, and subsequent failure of, the north, south, and west walls of the steam tunnel, adjacent to their locations

    2. Overloading of the isolation valves, piping between the isolation valves and drywell penetrations, the drywell penetrations, and the containment shell as described in Item b above. These restraints act with the lower assembly.

Final design of the lower and upper assemblies was based on the criteria given in Subsection 3.6.2.1.5.2, and the methods given in Subsection 3.6.3. The material used for the lower assembly members and the U-bolt attachment structures in the upper assembly is ASTM A 588 steel. The energy-absorbing U bolts used in the upper assembly are made of A479, type 304, stainless steel. Blowdown thrusts used for the design are shown in Figures 3.6-33 and 3.6-34. A value of 1.26 PA (where P is the operating pressure of the line and A is the flow area of the pipe) was used for the design blowdown thrust for all HPCI steam piping restraints. The thrusts were assumed to act instantaneously and to remain constant for the duration of the blowdown event.

Effect of Tunnel Pressurization on Steam Tunnel Structures

As indicated in Subsection 3.6.2.2.1.2.2, the calculated pressure resulting from a main steam line break in the steam tunnel was 5.1 psig for an MSIV closure time of 10.5 sec. The pressure resulting from the conservative 10.5 sec closure time of the MSIVs does not have an adverse effect on the structures.

The lower floor at elevation 583 ft 6 in. is the critical structure in the tunnel. It was designed for the load combinations and acceptance criteria shown in Table 3.6-5. The floor slab has been evaluated for a maximum pressure of 5.1 psig and found to be adequate.

All other structural elements of the steam tunnel will withstand higher pressures.

3.6.2.2.2 Feedwater Line Breaks in Steam Tunnel

The evaluation of the consequence of a break in the feedwater lines was carried out as described below.

3.6.2.2.2.1 Review of Potential Damage

As in the case of the main steam line break, a review of the potential damage caused by a feedwater line break was made to identify the need for a dynamic pipe whip or jet impingement analysis. The results of this review are documented in Reference 24.

In view of these results, a dynamic pipe whip and jet impingement analysis was carried out.

3.6.2.2.2.2 Feedwater Break Analysis

The feedwater lines outside the containment are routed in the same concrete tunnel through the Auxiliary Building and into the Turbine Building. In accordance with the criteria given in Subsection 3.6.2.1.2.2, longitudinal and circumferential design-basis ruptures were postulated at each end of each elbow in the feedwater lines between the outboard isolation check valves and the steam tunnel exit to the turbine building. Critical crack breaks were postulated at all adverse locations in this piping. As an alternative to postulating ruptures between the containment and outboard isolation check valves, the piping was designed to meet the normal operating stress criteria presented in Subsection 3.6.2.1.2.2, and was provided with restraints to limit transmission of bending and torsional loads through the valves in the event of an upstream rupture. The postulated rupture locations are shown in Figure 3.6-35.

Analyses were performed to assess quantitatively the pipe whip, jet impingement, and flooding effects of these postulated ruptures. Since the routing of the two feedwater lines in the tunnel is nearly identical, in most cases the analyses were performed for only one line and results extrapolated to the other line. The predicted rupture effects were evaluated to identify cases in which unacceptable damage to structures, systems, or components required for shutdown could result. Finally, designs were developed for modifications required to prevent occurrence of unacceptable damage.

A description of the rupture effects analyses, the damage evaluations, and the required modifications for the feedwater lines in the steam tunnel is given in the following sections.

3.6.2.2.2.2.1 Short-Term Blowdown Analysis

Blowdown analyses were performed for the postulated feedwater line breaks using the methods presented in Reference 25. The resulting thrust time-histories were then used as input for subsequent pipe whip and jet impingement analyses using the methods outlined in Subsection 3.6.3.

To determine conservatively the thrusts resulting from the postulated ruptures, the following assumptions were made:

a. The reservoir pressures for the feedwater line were assumed to be 1135 psia

b. Only the so-called "slower wave" (Reference 25) was considered in evaluating "wave force" (Reference 25) components on the open segments of piping.

As the routing of the two feedwater lines in the area of interest is essentially the same, only one line was analyzed.

As a result of check-valve closure, the thrust time-histories have the following characteristics.

a. Reactor side circumferential break thrusts do not reach a steady value that is larger than the initial value because the check valves between the breaks and the RPV close during the blowdown event

b. Longitudinal break thrusts have the same steady-state values as pump side circumferential break thrusts at the same location because check-valve closure prevents feeding of the break from the reactor side.

The analyses were carried out for a period of time sufficient to allow for the rise time of all restraint reactions. Typical thrust time-histories resulting from the analyses are shown in Figures 3.6-33 and 3.6-34.

3.6.2.2.2.2.2 Flooding Analysis

This subsection presents the analysis of flooding effects due to a postulated feedwater pipe rupture outside containment. The various locations that could be flooded are discussed as well as the assumptions and flow events following postulated pipe breaks. The results of the analysis are presented last.

Figure 3.6-36 shows the bottom floor of the steam tunnel and the first floor of the Auxiliary Building. The lower west portion of the steam tunnel is shown in Figure 3.6-35. The tunnel is provided with a lower pressure relief door composed of steel panels that individually swing up on hinges. Along the east wall of the Auxiliary Building first floor are large openings to the Turbine Building. The bottoms of these openings are located 6 ft above the floor.

Figure 3.6-37 shows a block diagram of the main features of the feedwater system. During normal plant operation, the condensate is pumped from the hotwell through various subsystems such as polishing demineralizers and the feedwater heaters to the reactor.

The entire feedwater system is classified as high energy. The two feedwater lines are routed from the reactor feed pumps on the first floor of the Turbine Building up to the No. 6 feedwater heaters located on the Turbine Building third floor. The feedwater lines are headered together downstream of the No. 6 heaters and then branch again into two parallel lines to drop down through the third floor slab into the second floor where they enter the steam tunnel.

The lines penetrate the steam tunnel upper pressure relief doors and continue parallel to the main steam lines until they enter primary containment.

As with the main steam lines, only that portion of the feedwater lines on the reactor side of the upper pressure relief doors is considered for pipe break. Feedwater breaks upstream of the doors are not evaluated since no safety-related equipment is located in the Turbine Building except for the third set of MSIVs which are not required for mitigation of break effects. Adequate seals exist to prevent Turbine Building breaks from affecting the Reactor and Auxiliary Buildings.

The analysis of the feedwater line break scenario assumes failure of the feedwater startup control valve in the open position. This single failure was selected to maximize flooding in affected areas of the plant. The feedwater line break would not be isolated until the condensate and heater feed pumps trip on low hotwell level. Operator action was not considered for the 8-1/2 minutes during which water flows through the postulated pipe break.

The basic flood model for the feedwater line break in the steam tunnel is shown in Figure 3.6-38. The fluid released from the feedwater break would be dispersed over the steam tunnel floor. A portion of the break fluid would flash to steam and pressurize the steam tunnel for a short period following the break.

The water dispersed over the steam tunnel floor would begin to drain to the northeast corner room sump. However, the break flow will greatly exceed the capacity of the floor drains and the steam tunnel flood depth would continue to increase. At a flood elevation of 584 ft 9 in. water would begin to flow through the equipment drain in the steam tunnel to the southeast corner room sump. As the steam tunnel flood depth continues to rise, the head of water would open the pressure relief doors allowing break fluid to enter first floor Auxiliary Building.

The first floor Auxiliary Building floor drains have been capped. Therefore, flow through the drain lines in this room does not occur.

As the first floor Auxiliary Building flood depth continues to increase, flow through numerous equipment drains whose funnels are at various elevations would occur. The equipment drains from first floor Auxiliary Building header join those from the steam tunnel, which lead to the southeast corner room.

The steam tunnel and first floor Auxiliary Building are watertight to flood depths of 10 and 4 feet, respectively. Therefore, these are the only rooms directly affected by flooding. Flooding in other rooms results only from floor and equipment drain flow. Floor and equipment drains flow to the northeast and southeast corner rooms, respectively, will exceed the capacity of the sumps. Therefore, the flood depth in these rooms will continue to increase. If the flood depths increase to a certain level, flooding of the torus and HPCI rooms will occur.

Table 3.6-6 indicates break flow rate and the sequence of events following a feedwater line break in the steam tunnel. The table is based on the assumption that the feedwater startup control valve failed in the open position. The sequence of events after the break is discussed below.

Immediately following the break, feedwater flow increases to 43,000 gpm, tripping the reactor feed pumps on low suction pressure (300 psig). Tripping the reactor feed pumps does not decrease the break flow because the heater feed, heater drain, and condensate pumps are capable of maintaining this flow rate. This flow rate remains constant until 57 seconds after the break when the fast closure reactor feedwater pump discharge valves are closed. Flow is then diverted through the startup level control valve and the break flow is reduced to 21,800 gpm. At 64.0 seconds, the heater number 5 level control valves are fully closed and heater drain pump flow is isolated from the break. As a result, the break flow decreased to 20,000 gpm and remains at this level until 447 seconds, at which time the condensate and heater feed pumps trip on low hotwell level totally isolating the break. The inventory of water in the piping downstream of the reactor feed pumps is then assumed to be discharged from the break over a one-minute period. The steam tunnel drains to the southeast corner room through equipment drains and the northeast corner room through floor drains.

The steam tunnel and Auxiliary Building first floor, where the Reactor Building closed cooling water (RBCCW) heat exchanger room is located, reach peak flood depths of 4 ft 6 in. and 3 ft 10 in., respectively. The steam tunnel and RBCCW heat exchanger room flood elevations then begin to decrease due to flow through the steam tunnel floor and equipment drains (DRN). The RBCCW heat exchanger room drains to the southeast corner room of the Auxiliary Building.

The northeast corner room peak flood depth of 7 ft is reached due to floor drain flow into the sump. Equipment drain flow causes flooding in the southeast corner room to a depth of 14 ft approximately 10 hours after the line break, at which point water would begin to spill into the torus and HPCI rooms. The torus room flood depth reached 14.8 in. and the HPCI room would reach a peak flood depth of 78.8 in. based on worst case door failure combinations for the given room.

The safe shutdown path considers systems necessary to scram the reactor, depressurize the reactor, and to establish and maintain core cooling utilizing the residual heat removal (RHR) system. For feedwater line breaks, the availability of offsite power maximizes the consequential effects of flooding. Therefore, offsite power was assumed to be available. If offsite power is not available, the condensate and heater feed pumps will trip, thus ending the break scenario sooner. No water will be lost from the reactor, whether offsite power is available or not, since the feedwater check valves are designed to close immediately. HPCI will restore water level to compensate for the loss of feedwater flow. The vessel can be manually depressurized by using the main steam safety relief valves. After pressure reduction, the operator places the RHR system (Division 1 or Division 2) in the low pressure coolant injection mode. The residual heat removal service water (RHRSW) system is used as the heat sink in the RHR cooling mode.

The evaluation shows the first floor Auxiliary Building (RBCCW heat exchanger room), the northeast corner room (Division 1 core spray and reactor core isolation cooling (RCIC)), the southeast corner room (Division 2 core spray), high pressure coolant injection (HPCI) and the torus room flooded. Under these conditions the plant can achieve safe shutdown by using the main steam safety relief valves and both divisions of RHR.

3.6.2.2.2.2.3 Effects of Feedwater Jet Impingement and Pipe Whip

The analyses of the effects of pipe whip and jet impingement for feedwater line rupture in the steam tunnel have been discussed previously, in conjunction with the analyses for main steam line ruptures in the steam tunnel, in Subsections 3.6.2.2.1.2.3 and 3.6.2.2.1.2.4.

3.6.2.2.2.2.4 Structural Evaluation

The evaluation of structural components for ability to withstand loads resulting from postulated feedwater line ruptures in the steam tunnel closely parallels that previously described in Subsection 3.6.2.2.1.2.5 for postulated main steam line ruptures, and details will not be repeated here.

The pipe whip impact and jet impingement loads resulting from a postulated feedwater line rupture could cause failure of the lower tunnel floor at elevation 583 ft 6 in.; however, the restraints described in Subsection 3.6.2.2.1.2.5 are intended to prevent this failure. The tunnel walls, upper floor at elevation 626 ft 8 in., and ceiling are adequately designed to withstand the pipe whip impact and jet impingement loads that would result from the postulated feedwater line ruptures.

The 1.9 psig maximum lower tunnel floor hydrostatic pressure that could result from flooding after a postulated feedwater line break is less than the 5.1 psig steam pressure that would follow a postulated steam line break with a 10.5 sec MSIV closure. It has been previously determined that the lower tunnel floor is adequately designed to withstand this pressure. Pressurization effects from flashing feedwater are also bounded by the main steam line break.

The structural evaluation also indicated that the auxiliary building is adequately designed to withstand the 1.7 psig hydrostatic pressure resulting from maximum possible flooding on the first floor.

3.6.2.2.3 High Pressure Coolant Injection System

The evaluation of the consequences of a pipe break in the HPCI system was carried out as described below and in references 24, 33 and 64.

The only portion of the HPCI system piping that is classified as high energy is the HPCI turbine steam supply line. This line exits from the primary containment into the steam tunnel near the main steam and feedwater lines. The line then drops through the steam tunnel floor to the torus area, where it is routed adjacent to the torus before entering the HPCI pump room (room SB7), where the line connects to the HPCI turbine.

Figure 3.6-39 shows the HPCI turbine steam supply line and its relation to building structures and components.

3.6.2.2.3.1 Review of Potential Damage

As in the case of the main steam and feedwater line breaks, a review of the potential damage resulting from a break in the HPCI steam line was carried out in accordance with Subsection 3.6.2.1.6 to identify the need for a dynamic pipe whip or jet impingement analysis. The review took into consideration the equipment required to ensure a safe shutdown, as discussed in Subsection 3.6.2.1.4 and listed in Table 3.6-1.

This review, combined with other considerations such as existing pipe restraints, established the following areas of concern with respect to pipe whip and/or jet impingement:

a. Effect on the HPCI isolation valve and containment penetration

b. Effect on the torus in the vicinity of the HPCI line in the torus room.

In addition, the environmental effects investigation was limited to a break in the HPCI room.

3.6.2.2.3.2 Pipe-Break Analysis

The criteria given in Subsection 3.6.2.1.2.2 have been applied in determining the locations of the postulated pipe breaks. The terminal end break locations are defined at the connection to the outboard containment isolation valve and the connection to the HPCI turbine stop valve. The Class 1 piping between the containment and the outboard isolation valve is designed to meet the normal operating stress criteria presented in Subsection 3.6.2.1.2.2. The detailed stress analysis of the piping determined that the predicted stresses at all locations between the terminal ends are substantially less than the stress limits established in Subsection 3.6.2.1.2.2. Therefore, no arbitrary intermediate pipe breaks were postulated per Generic Letter 87-11.

The postulated break locations in the HPCI steam supply line are included in Figure 3.6-39. With the criteria of Subsection 3.6.2.1.2.3.c, only circumferential-type breaks are postulated at each break. As required, analyses were performed to assess quantitatively the pipe whip, jet impingement, and environmental effects of these postulated ruptures.

3.6.2.2.3.2.1 Blowdown Analyses

Blowdown analyses were performed for postulated HPCI turbine steam supply line breaks using the PRTHRUST program as discussed in Subsection 3.6.3. To conservatively maximize blowdown thrust forces for input to the pipe whip and jet impingement calculations, the full-size 10-in. line was assumed to be initially at rated reactor pressure conditions (1060 psia saturated steam, zero flow). The primary containment isolation valves were assumed to be in their normal operating position, and the HPCI turbine stop valve was assumed to be closed.

The analyses were carried out for a period of approximately 0.5 sec, a time sufficient to develop maximum pipe whip and jet impingement response. It should be noted that this length of time is insufficient to allow activation of operable components, and the balance of plant systems (other than the broken system) continue to operate in the normal way.

The blowdown thrust was calculated as a function of time for circumferential breaks at each of the locations indicated in Figure 3.6-39. These results were then used in jet impingement and pipe whip analyses according to the methods described in Subsection 3.6.3 and discussed below.

Characteristically, the thrust is equal to line pressure times area at the instant of rupture, and increases thereafter as fluid is accelerated from the break. Within a short time, however, choking takes place at the break, and thrust drops sharply as the line pressure decays. The HPCI turbine side-break forces decay to zero as the steam in the line is depleted. Reactor side-break forces decay to a quasi-steady-state thrust, controlled by choking in the shutoff valve in the 1-in. bypass line around the outboard isolation valve.

A "longer-term" blowdown analysis was conducted as the initiating calculation of the environmental conditions resulting from a postulated break in a high energy piping system as described in Subsection 3.6.3.

A preliminary hand calculation showed that a full blowdown of the 10-in. steam line, under the assumption of loss of offsite power and failure of the dc isolation valve, together with a startup time of 10 sec for the diesel generator to initiate closure of the ac isolation valve, would result in unacceptably high pressures and temperatures in the reactor building. To avoid this situation, a 1-in. bypass line and shutoff valve around the outboard isolation valve were incorporated into the system with the normal mode of the outboard isolation valve in the closed position, as shown in Figure 3.6-39. The analysis incorporated the 1-in. bypass line.

The following assumptions were made for the "longer-term" blowdown analysis:

a. The line is initially at rated reactor pressure conditions of 1060 psia and saturated steam

b. The inboard steam line isolation valve and bypass line isolation valve are in their normally open positions

c. The outboard isolation valve is in its normally closed position

d. The HPCI turbine stop valve is closed

e. The dc valve in the 1-in. bypass line fails to close

f. Choke flow through the bypass line takes place until the inboard isolation valve is closed by operator action or high area temperature.

Immediately after the postulated break, steam from the downstream side of the outboard isolation valve is rapidly released from the break, and choke flow through the 1-in. bypass line continues to release steam until the inboard isolation valve is closed.

3.6.2.2.3.2.2 <u>Environmental Analysis</u>

Environmental analyses were performed using the "CVPT-REPORT" and “KITTY6” programs (described in Subsection 3.6.3) to predict the maximum compartment pressures, temperatures, and humidities resulting from postulated design-basis ruptures of the HPCI steam supply line. The steam line traverses three compartments: the steam tunnel, the torus room, and the HPCI pump room. The environmental analyses were evaluated for uprated power conditions.

<u>The Steam Tunnel (Room 109)</u>

No specific analysis was performed for HPCI steam line breaks in the steam tunnel, as the resulting conditions are bounded by the main steam line breaks discussed in Subsection 3.6.2.2.1 on the basis of mass flow and steam conditions.

Torus Room (Room SB2)

There are no postulated HPCI steam line breaks in the torus area.

HPCI Pump Room (Room SB7)

An environmental analysis was performed for a postulated break in the HPCI pump room, and the environmental response of affected compartments was determined. The analytical model showing the control volumes and the flow paths is provided in References 33 and 64.

Results of Analysis for HPCI Line Break in HPCI Room

After the postulated break, the HPCI pump room (SB7) would rapidly fill with steam and vent to room B7 and southeast corner room (SB6 and B6). Room B6 is connected via a stairwell to the first floor which, in turn, is connected to the second, third, and fourth floors via staircases and the equipment hatchway.

The compartmental (control volume) pressure and humidity responses are provided in Reference 33, and the temperature responses are provided in Reference 64. An evaluation has been performed to include the effect of power uprate. Maximum calculated pressure and temperature are 1.23 psig and 183°F in the HPCI pump room.

3.6.2.2.3.2.3 Pipe Whip Evaluation

Consideration was given to the effects of HPCI steam line pipe whip on safety-related structures, systems, and components in the steam tunnel, torus area, and the HPCI room. Where required, analyses were performed to assess pipe whip consequences. See Reference 24.

Effects in Steam Tunnel (Room l09)

Without a pipe whip restraint, the HPCI line pipe whip in the steam tunnel could potentially damage the outboard isolation valve and the primary containment penetration. However, this was anticipated in consideration of the restraints for the steam and feedwater lines in the tunnel, and restraints were provided for the HPCI line as well. The restraint design is discussed in Subsection 3.6.2.2.1.2.5.

Effects in Torus Room (Room SB2)

There are no postulated HPCI steam line breaks in the torus area.

Effects in HPCI Room (Room SB7)

The shutdown capability for the HPCI steam line break in the HPCI pump room is demonstrated in Reference 24.

The concrete structural components within the room were found to be adequately designed to withstand pipe whip impact loads.

Damage Evaluation

Damage evaluation and modifications instituted as a result of the pipe whip evaluation are discussed in Subsection 3.6.2.2.3.2.5.

3.6.2.2.3.2.4 Jet Impingement Evaluation

As in the case of pipe whip, the evaluation was carried out for each area through which the HPCI steam line passes. The evaluation and results are described in Reference 24.

Effects in Steam Tunnel (Room 109)

Jet impingement could adversely affect the HPCI outboard isolation valve. However, the jet impingement shields provided for protection against main steam line breaks would also protect against the HPCI line breaks.

With the inclusion of the 1.75-in. jet deflector plate provided for the main steam line break, jet impingement would not adversely affect any of the concrete structures.

Effects in Torus Room (Room SB2)

There are no postulated HPCI steam line breaks in the torus area.

Effects in HPCI Pump Room (Room SB7)

The shutdown capability for the HPCI steam line break in the HPCI pump room is demonstrated in Reference 24.

The concrete structural components were found to be adequately designed to withstand jet impingement loads.

Damage Evaluation

Damage evaluation and modifications instituted as a result of the jet impingement evaluation are discussed in Subsection (3.6.2.2.3.2.5).

3.6.2.2.3.2.5 System, Component, and Structural Damage Evaluation

An evaluation (Reference 24) was made of the direct damage resulting from HPCI steam line pipe whip or jet impingement. The evaluation included effects on various systems, components, and structures, as well as impact on the ability to achieve a safe shutdown. Plant features that mitigate the consequences of HPCI steam line breaks are described below.

Steam Tunnel (Room 109)

As indicated in Subsection 3.6.2.2.3.2.3, provision of HPCI steam line pipe restraints was incorporated into the steam line/ feedwater line restraint system. The restraints, together with the piping sleeve provided where the steam line passes through the steam tunnel floor, preclude unacceptable damage to the HPCI isolation valve or the primary containment penetration. Incorporated with the restraints is a shield to protect the isolation valve from jet impingement.

Torus Room (Room SB2)

There are no postulated HPCI steam line breaks in the torus area.

Evaluation of Environmental Effects on Systems, Components, and Structures

The equipment, including active and passive components, listed in Table 3.6-1 as required for attaining and maintaining a safe-shutdown condition after a postulated break in the HPCI steam line, was evaluated for its functional capability under the pressure, temperature, and humidity conditions resulting from the pipe break. It was determined that the environmental conditions would not adversely affect the operation of the required systems.

HPCI steam line breaks in the steam tunnel or HPCI pump room would not result in a reactor protection system (RPS) trip because the bypass line around the outboard isolation valves would severely minimize the RPV inventory loss. Normal shutdown procedures would be used in the case of HPCI steam line breaks.

It is concluded, therefore, that with the bypass line around the outboard isolation valve (Subsection 3.6.2.2.3.2.1) and the pipe whip restraints (Subsection 3.6.2.2.3.2.5) incorporated into the design, a break in the HPCI steam supply line will not jeopardize the ability to attain and maintain a safe shutdown.

3.6.2.2.4 Reactor Core Isolation Cooling System

The evaluation of the consequences of a break in the RCIC system was carried out as described below and in References 24, 33 and 64.

The only portion of the RCIC piping system that is classified as high energy is the RCIC steam supply line. This line exits from the primary containment into the steam tunnel near the main steam and feedwater lines. The line then drops through the steam tunnel floor to the torus area, where it is routed adjacent to the torus before entering the north core spray and RCIC pump room. Figure 3.6-41 shows the RCIC steam supply line configuration and its relation to building structures and components.

3.6.2.2.4.1 Review of Potential Damage

A review of the potential damage resulting from a break in the RCIC steam supply line was carried out in accordance with Subsection 3.6.2.1.6 to identify the need for a dynamic pipe whip or jet impingement analysis. The review took into consideration the equipment required to ensure a safe shutdown as discussed in Section 3.6.2.1.4 and Table 3.6-1.

In addition, the environmental effects investigation was conducted for a break in the northeast corner room, as discussed in Subsection 3.6.2.2.4.2.

3.6.2.2.4.2 Pipe-Break Analysis

The criteria given in Subsection 3.6.2.1.2.2 have been applied in determining the locations of the postulated pipe breaks. The terminal end break locations are defined at the connection to the outboard containment isolation valve and the connection to the RCIC turbine stop valve. The Class 1 piping between the containment and the outboard isolation valve is designed to meet the normal operating stress criteria presented in Subsection 3.6.2.1.2.2. The detailed stress analysis of the piping determined that the predicted stresses at all locations between the terminal ends are less than the stress limits established in Subsection 3.6.2.1.2.2. Therefore, no intermediate break locations have been postulated between the terminal ends.

The postulated break locations in the RCIC steam supply line are included in Figures 3.6-41 and 3.6-42. With the criteria of Subsection 3.6.2.1.2.3, Item c., only circumferential-type breaks are postulated at each break location. As required, analyses were performed to assess quantitatively the pipe whip, jet impingement, and environmental effects of these postulated ruptures.

3.6.2.2.4.2.1 Blowdown Analysis

For the pipe whip analysis, blowdown thrust loads at all critical break locations, as identified above, were calculated using an ideal gas model to determine the forcing functions. For jet impingement analyses, the exit plane thrust was calculated as 1.26 PA, where P is the fluid saturation pressure and A is the pipe flow area.

A "longer-term" blowdown analysis was conducted to determine the environmental conditions resulting from a postulated break in a high-energy piping system, as described in references 33, 64, and 65.

The following assumptions were made for the "longer-term" blowdown analysis:

a. The line is initially at rated reactor pressure conditions of 1060 psia and saturated steam

b. The line isolation valves are in their normally open position

c. The RCIC turbine stop valve and the motor-operated bypass valve are closed

d. The dc-powered isolation valve fails to close

e. The ac-powered isolation valve closes 29 sec after the line break. Closure time includes diesel generator startup and valve closure time, Reference 65.

f. Choked flow occurs at the most limiting restriction.

Immediately after the postulated break, the flow from the upstream side of the break increases rapidly to the critical flow for the break, and decreases with closing of the isolation valve. Steam flow rate as a function of time after the postulated break is provided in References 33 and 65.

3.6.2.2.4.2.2 Environmental Analysis

Environmental analyses were performed using the "CVPT-REPORT" and “KITTY6” programs (described in Subsection 3.6.3) to predict maximum compartment pressures, temperatures, and humidities resulting from postulated design-basis ruptures of the RCIC steam supply line. The steam line traverses three compartments: the steam tunnel, the torus room, and the RCIC pump room. The environmental analyses were evaluated for uprated power conditions.

The Steam Tunnel (Room 109)

No specific analysis was performed for RCIC steam line breaks in the steam tunnel, as the resulting conditions are bounded by those produced by the main steam line breaks discussed in Subsection 3.6.2.2.1.2.2 on the basis of mass flow and steam conditions.

Torus Room (Room SB2)

There are no postulated RCIC steam line breaks in the torus room.

RCIC Pump Room (Room SB5)

An environmental analysis was performed for postulated RCIC line breaks in the northeast corner room, and the environmental response of affected compartments was determined. The environmental analysis model showing the control volumes and the flow paths are provided in References 33 and 64.

Results of Environmental Analysis

For RCIC Steam Line Break in Torus Room

There are no postulated RCIC steam line breaks in the torus area.

For RCIC Steam Line Break in the RCIC Pump Room

After the postulated break, the RCIC pump room SB5 would rapidly fill with steam and, via a stairwell, vent to the basement of the reactor building, which in turn is connected to the first, second, third, and fourth floors via staircases and the equipment hatchway. Steam is also vented from the first floor to remaining corner rooms (B1/SB1, B3/SB3, B6/SB6) via staircases and equipment hatches. Each corner room vents to the torus room through pressure relief seals that open at 0.1 psid. Steam is also vented from the torus room to the pipe tunnel through open penetrations and from the RCIC pump room via pressure relief seals which open at 0.1 psid. A vent opening between the pipe tunnel and the first floor auxiliary building is provided and is designed to open at 0.33 psid to vent steam to the auxiliary building and subsequently to the turbine building.

The compartmental (control volume) pressure and humidity responses are provided in Reference 33, and the temperature responses are provided in Reference 64. Calculated pressure and temperature are 0.48 psig and <237°F in the RCIC pump room.

3.6.2.2.4.2.3 Pipe Whip Evaluation

Analyses were performed (Reference 24) to assess pipe whip consequences on safety-related structures, systems, and components in the main steam tunnel, in the event of a postulated design basis break of the RCIC steam line.

Blowdown thrust loads were calculated using the methods discussed in Subsection 3.6.2.2.4.2.1. The methods used in conducting the pipe whip analysis, and the details of the structural evaluation for pipe whip impact, are given in Subsection 3.6.3.

A summary of the results of the pipe whip evaluation for each room traversed by the RCIC steam line is presented below.

Effects in Steam Tunnel (Room 109)

Loads due to pipe whip could damage the outboard RCIC steam isolation valve. The containment penetration assembly would not be damaged, due to the existence of a separate containment penetration flued head support structure.

Effects in Torus Room (Rooms SB2 and B2)

There are no postulated RCIC steam line breaks in the torus area.

Effects in RCIC Pump Room

An evaluation of the effects of pipe whip is described in Reference 24.

Effects on Structural Components

The reinforced-concrete structures were found to be adequately designed to withstand pipe whip impact loads when analyzed in accordance with Subsection 3.6.3.

3.6.2.2.4.2.4 Jet Impingement Evaluation

As in the case of pipe whip, an evaluation (Reference 24) was carried out for each room traversed by the RCIC steam line. The jet impingement loads from the calculated thrust forces were determined by the methods described in Subsection 3.6.3.

Effects in Steam Tunnel

Jet impingement loadings due to a break in the RCIC steam line in the steam tunnel could damage the top works of the RCIC outboard isolation valve and render it inoperative.

Effects on Structural Components

The reinforced-concrete structures were found to be adequately designed to withstand jet impingement loads when analyzed by the methods described in Subsection 3.6.3.

3.6.2.2.4.2.5 System, Component, and Structural Damage Evaluation

An evaluation (Reference 24) was made of the direct damage resulting from the RCIC steam line break on various systems, components, and structures as well as its impact on the ability to attain a safe shutdown. The plant features that mitigate the consequences of RCIC steam line breaks are described below.

Steam Tunnel

The RCIC outboard isolation valve could be damaged by either pipe whip or jet impingement. Referring to Table 3.6-1, it is seen that a general requirement for a safe shutdown is to maintain the integrity of the primary containment. If the inboard isolation valve is assumed to fail in accordance with assumption d.2. of Subsection 3.6.2.1.1.1 and the outboard valve is damaged, the primary containment would be lost, representing an unacceptable condition.

To mitigate this condition, the RCIC steam line in the steam tunnel is equipped with a steel-plate pipe restraint that will prevent pipe whip from imposing unacceptably high loadings on the outboard isolation valve. The restraint design concurrently provides protection against jet impingement on the isolation valve top works, thereby ensuring its operability.

Torus Room

There are no postulated RCIC steam line breaks in the torus area.

Pipe Whip Restraints and Jet Impingement Shields

Designs for pipe whip restraints and jet impingement shields to protect from unacceptable damage resulting from RCIC steam line breaks in the steam tunnel are described below.

Steam Tunnel

The design for the pipe whip restraint and the jet impingement shield is as follows. The restraint is of the close-clearance plate type and, together with the piping sleeve previously provided where the pipe passes through the steam tunnel floor, serves to prevent the line from deflecting excessively in either the torsional or bending modes and hence prevents the development of excessive stresses in the outboard RCIC steam line isolation valve or in the piping between the isolation valve and the primary containment penetration flued head. Due to its configuration, this restraint also serves to protect the RCIC outer isolation valve from jet impingement.

Torus Room

There are no postulated RCIC steam line breaks in the torus area.

3.6.2.2.4.3 Evaluation of Ability To Attain and Maintain a Safe Shutdown

A break in the RCIC steam supply line in the steam tunnel may result in activation of the high-temperature signal in the steam tunnel, with concurrent closure of the MSIVs and subsequent RPS trip (scram). Therefore, this evaluation assumes loss of offsite power, as indicated in Subsection 3.6.2.1.1.1.

All of the systems and components listed in Table 3.6-1 as required for a safe shutdown will be operable. All required redundant components will be available, with the exception of Division I core spray, which is assumed to have failed, together with the RCIC system for the pipe break in the RCIC pump room.

As a result of reactor scram and primary containment isolation, reactor shutdown cannot be attained by normal shutdown procedures for a break in the steam tunnel, although they could and probably would be used for the break in the RCIC pump room.

The RCIC system will be automatically isolated by activation of one of the three signals listed in Table 3.6-1. Reactor depressurization can be achieved through use of the HPCI system to maintain water level and remotely operated relief valves to depressurize the RPV.

On depressurization, Division II core spray and LPCI would be available to maintain water level. Suppression pool cooling and maintaining a long-term safe shutdown can be accomplished by operation of the RHR system.

Applying the single failure criterion, operable redundant or backup systems are available to ensure that each required function is carried out. If HPCI is unavailable, depressurization can be accomplished by the ADS alone, while coolant water inventory is maintained at an acceptable level. Division II core spray or LPCI independently can maintain acceptable water levels after depressurization, and redundancy in RHR will ensure suppression pool cooling and the ability to maintain a long-term safe shutdown.

It is concluded, therefore, that a break in the RCIC steam supply line will not jeopardize the ability to attain and maintain a safe shutdown.

3.6.2.2.5 Reactor Water Cleanup System (RWCU)

The reactor water cleanup (RWCU) system removes water from the reactor recirculation system for decontamination by a demineralizer system and then returns the water to the reactor through the feedwater system. The RWCU line leaves the containment, entering the second floor of the reactor building, as shown in Figure 3.6-43. From this area, the line divides into two smaller lines that feed the RWCU recirculation pumps. The water is pumped through heat exchangers to a demineralizer system. The cleaned-up water is reheated by the heat exchangers and enters the feedwater system in the steam tunnel after being routed through a pipe chase to the torus area.

3.6.2.2.5.1 Review of Potential Damage

As in the preceding evaluations, a review of the potential damage resulting from a break in the RWCU piping system was carried out in accordance with Subsection 3.6.2.1.6 to identify the need for a dynamic pipe whip or jet impingement analysis.

Based on this review, the following concerns were identified regarding pipe whip and jet impingement:

a. Effect on outboard isolation valves and the primary containment penetration in room 224

b. Effects on the RWCU line isolation valves that connect to the feedwater line in the steam tunnel

c. Effects on the torus and other Category I systems and components in the torus room.

In addition, the environmental conditions and effects resulting from an RWCU line break required evaluation.

3.6.2.2.5.2 Pipe-Break Analysis

The criteria given in Subsection 3.6.2.1.2.2 have been applied in determining the locations of the postulated pipe breaks. The Class 1 piping between the containment and the outboard isolation valve is designed to meet the normal operating stress criteria presented in Subsection 3.6.2.1.2.2.

The postulated break locations in the seismically analyzed portion of the RWCU water line are included in Figure 3.6-44. With the criteria of Subsection 3.6.2.1.2.3, Item c., only circumferential-type breaks are postulated at each break. As required, analyses were performed to assess quantitatively the pipe whip, jet impingement, and environmental effects of these postulated ruptures.

3.6.2.2.5.2.1 Blowdown Analysis

A short-term blowdown analysis was performed for a rupture of an RWCU line downstream of the outboard isolation valve in room 224 and in room 219, using hand calculation methods to determine thrust loads for use in evaluating the potential damage due to pipe whip and jet impingement. Although pipe breaks are no longer postulated in room 224, this break analysis bounds other postulated breaks in room 219. To conservatively maximize blowdown thrust, the lines were assumed to be at the reactor recirculation line normal operating conditions.

Initially, the thrust would be equal to the line pressure at the time of the break times the break area. The thrust would then rapidly rise to a steady-state force of 1.26PA, where P is the fluid saturation pressure (910 psia) and A is the break area. The steady-state force would equal 27,300 lb and was taken as a constant value in the subsequent damage evaluation.

Similar evaluations were conducted for the postulated break in the torus room. Blowdown force for a postulated rupture of the RWCU line at anchor G33-3245-G34 was performed using Fauske's two-phase flow model (Reference 35). To conservatively maximize the blowdown forces, the line was assumed to be at maximum operating temperature and pressure (532°F and 1244 psia).

A "longer-term" blowdown analysis was conducted to determine the environmental conditions resulting from a postulated break in the RWCU system. Although a pipe break immediately downstream of the RWCU suction line outboard isolation valve is no longer postulated, this break analysis was used as a bounding case for other postulated breaks.

The following assumptions were made for the "longer-term" blowdown analysis:

a. The line is initially at 1060 psia and 534°F.

b. Deleted

c. The ac-powered isolation valve closes 23 sec after detection of the line break. This time includes instrument and loop response time and valve closure time. References 64 and 65.

Immediately after the postulated break, the flow rate from the upstream side of the break will consist of an initially high flow rate during the inventory blowdown followed by a smaller rate during steady-state blowdown. This flow rate is provided in References 33 and 65.

3.6.2.2.5.2.2 Environmental Analysis

Environmental analyses were performed using the "CVPT-REPORT" or “KITTY6” programs (described in Subsection 3.6.3) to predict the maximum compartment pressures, temperatures, and humidities resulting from postulated design-basis ruptures of the RWCU high-energy line. The environmental analyses were evaluated for uprated power conditions.

RWCU Pump Rooms (Rooms 217 and 218)

An environmental analysis was performed for an RWCU line break in pump room B, and the environmental response of affected compartments was determined. The environmental response for a break in pump room A would be similar. The environmental analysis model showing the control volumes and flow paths is provided in Reference 33.

RWCU Holdup and Heat Exchanger Rooms (Rooms 224 and 219)

An environmental analysis was performed for an RWCU line break in the hold-up pipe room 224. Although pipe breaks in room 224 are no longer postulated, this break analysis bounds all other postulated breaks in room 219; therefore, was maintained as a bounding analysis. No analysis was performed for the heat exchanger room 219. The environmental analysis models showing the control volumes and flow paths are provided in References 33 and 64.

Torus Room (SB2)

An environmental analysis was performed for an RWCU line break in the torus room. RWCU system isolation is automatically initiated following the break. The redundant

RWCU system flow comparator instrumentation is assumed to fail. However, the torus room RWCU leak detection thermocouple setpoints are reached.

Steam Tunnel (Room 109)

No pipe breaks are postulated for the RWCU line in the steam tunnel. Conditions in the steam tunnel are bounded by the main steam line breaks discussed in subsection 3.6.2.2.1.

Results of Analysis for RWCU Break in Room 218

Both pump rooms feature a stacked brick wall on the east side for shielding purposes only. On collapse of the stacked brick wall, break mass and energy is vented to the second floor, which is connected via staircases and the equipment hatchway to the first, third, and fourth floors of the reactor building. No essential equipment is located in the missile path of the collapsing wall. The maximum calculated temperature in pump room B was 214°F at a pressure of 0.97 psig. All other areas are bounded by the RWCU break in room 224.

Results of Analysis for RWCU Break in Room 224

Although pipe breaks in room 224 are no longer postulated, this break analysis bounds breaks in room 219. The heat exchanger (219) and holdup (224) rooms feature a common stacked brick wall on the west side, 24 in. thick, for shielding purposes only. Break mass and energy is vented to the reactor building on collapse of the stacked brick wall in the same manner as room 218. The maximum temperatures and pressures calculated for the hold-up pipe room and heat exchanger room were <216°F and 9.7 psig and <215°F and 1.18 psig. The reactor building second floor temperature and pressure calculated were 156°F and 1.14 psig. All other areas showed equal or lower temperatures and pressures.

The maximum pressure of 9.7 psig predicted for room 224 exceeded the capacity of the wall between the hold-up room and the heat exchanger room. The short term environmental analysis was updated to bound the actual RWCU breaks in the RWCU heat exchanger room (instead of a bounding break of the largest pipe in the worst location) and to recompute the differential pressure across the wall between the hold-up room and the heat exchanger room. The computer code COMPARE was utilized in the analysis. The revised analysis calculated a maximum differential pressure of 1.5 psid across the wall which is acceptable.

Results of Analysis for RWCU Break in Torus Room

A steady-state temperature of 191°F with 100 percent humidity would result in the torus room. The pressures in the reactor building are bounded by the RCIC steam line break environmental conditions, as discussed in Subsection 3.6.2.2.4.

3.6.2.2.5.2.3 Pipe Whip Evaluation

Consideration was given to the effects of RWCU-water-line pipe whip on safety-related structures, systems, and components in the rooms where the high-energy portion of the water lines coexists with such structures and equipment. The results of these evaluations are reported in Reference 24.

3.6.2.2.5.2.4 Jet Impingement Evaluation

As in the case of pipe whip, an evaluation was carried out to assess how jet impingement resulting from a break in the RWCU high-energy water line would affect safety-related structures, systems, and components. The results of these evaluations are reported in Reference 24.

3.6.2.2.5.2.5 Evaluation of System, Component, and Structural Damage

An evaluation was made of the damage resulting directly from the RWCU water line break. The evaluation included effects on various systems, components, and structures as well as impact on the ability to achieve a safe shutdown. The RWCU line in the torus room was provided with pipe whip restraints to protect the torus from the effects of design-basis breaks. A review of the consequences of pipe whip and jet impingement loadings resulting from a postulated break in the RWCU piping determined that no important structural element would be damaged by pipe whip or jet impingement.

Pipe Whip Restraints and Jet Impingement Shields

Design of restraints was based on the criteria given in Subsection 3.6.2.3.2 and the methods given in Subsection 3.6.2.1.5. The restraints provided for the RWCU pump discharge line in the torus room direct the thrust loads resulting from postulated breaks into the reactor building walls and prevent the broken line from impacting the torus. An analysis to determine member sizes was made assuming a conservative steady-state thrust loading, applied instantaneously and assumed to be constant for the entire blowdown event. A dynamic impact factor of 2 was assumed to account for the sudden nature of the loading.

Evaluation of Environmental Effects on Systems, Components, and Structures

Equipment, including active and passive components, listed in Table 3.6-1 as required for attaining and maintaining a safe-shutdown condition after a postulated break in the RWCU system was evaluated with respect to its functional capability under the pressure, temperature, and humidity conditions resulting from the pipe break. It was determined that the environmental conditions would not adversely affect the operation of the required systems.

The structural framing of the reactor/auxiliary building was analyzed in accordance with the methods described in Subsection 3.6.3 for the effects of RWCU water line breaks. The interior walls and slabs of the compartment were analyzed for the effects of pressurization in accordance with the methods described in Subsection 3.6.3, and were found to be acceptable. The maximum differential pressure between the external walls and atmospheric pressure is well within the 3-psig tornado design pressure differential.

In view of the above, no adverse consequences due to environmental effects of the RWCU line break have been identified.

3.6.2.2.5.3 Evaluation of Ability To Attain and Maintain a Safe Shutdown

A break in the RWCU water line would not result in a turbine generator trip or an RPS trip. Therefore, this evaluation assumes that offsite power is available.

Referring to Table 3.6-1, all systems and components listed as required for a safe shutdown will be operable, as will all redundant components. No single component failure can be assigned that would preclude attaining and maintaining a safe shutdown.

It is concluded, therefore, that a break in the RWCU water line will not jeopardize the ability to attain and maintain a safe shutdown.

3.6.2.2.6 Control Rod Drive System

An evaluation of the consequences of a pipe break in the control rod drive (CRD) system was carried out as described below.

Those portions of the CRD system piping outside the containment that are classified as high energy are the 1-in. insert and 3/4-in. withdraw lines running from the hydraulic control units to the drives, and the 2-in. water charging line.

The piping from the hydraulic control units to the scram discharge system is pressurized to high-energy conditions less than 0.1 percent of the system operating time. The balance of the time this portion of piping is vented to the atmosphere. Accordingly, in conformance to NRC guidance, it is permissible to treat this portion of the piping as moderate-energy piping. The consequences of a break in moderate-energy lines are addressed in Subsection 3.6.2.3. In response to additional NRC comments on a break in this portion of piping, GE published two generic evaluations (References 36 and 37), and the BWR Owners Group submitted a report on scram discharge pipe integrity (Reference 38). The applicability of these evaluations to Fermi 2 was addressed in Reference 39, along with additional plant-unique information as needed to address the NRC comments. The conclusion of the studies indicates that the mechanical quality, maintenance procedures, operator actions, and existing system performance are sufficient to satisfactorily guarantee scram discharge piping system integrity. In addition, even if a break were to occur, it was shown that the break would contribute negligibly to the risk of core uncovering.

3.6.2.2.6.1 Review of Potential Damage

As in the case of the previously discussed systems, a review of the potential damage resulting from a break in any of the CRD piping was conducted in accordance with the methods described in Subsection 3.6.2.1.6. The review took into account the equipment required for a safe shutdown, as discussed in Subsection 3.6.2.1.4 and Table 3.6-1.

This review (Reference 24) established that there would be no adverse consequences resulting from any break of the CRD piping.

Environmental effects were not analyzed for a break in the CRD line because of the highly subcooled nature of the water. There would be no flashing of the liquid that escapes the postulated break and consequently no adverse environmental response.

3.6.2.2.6.2 Pipe-Break Analyses

3.6.2.2.6.2.1 Blowdown Analyses

Blowdown thrust loads used to assess the effects of pipe whips were generated by conservative hand calculational methods, as described in Subsection 3.6.3.

3.6.2.2.6.2.2 Environmental Analyses

Effects related to the spraying or flooding of components in the area of a CRD line break have been considered and are included in the analysis of moderate-energy systems provided in Subsection 3.6.2.3.

3.6.2.2.6.2.3 Pipe Whip Evaluation

Analyses were performed to assess the pipe whip consequences on the safety-related structures, systems, and equipment that might be damaged as a result of a design-basis break in the CRD piping.

Blowdown thrust loads were hand-calculated, as discussed in Subsection 3.6.2.2.6.2.1. The methods used in the structural evaluation for pipe whip impact are discussed in Subsection 3.6.3.

As a result of the review of the consequences of pipe whip caused from breaks in the CRD system piping, it was determined that no safety-related structural elements would be damaged by a whipping pipe.

3.6.2.2.6.2.4 Jet Impingement Evaluation

Because of the highly subcooled condition of the water in the CRD piping outside the primary containment, the line would depressurize almost instantaneously after a design-basis break. The flow rate out of the break would be limited by the capacity of the CRD pumps (260 gpm at 0-ft head), and flow velocity from the break will not exceed 11 fps. As a result, no unacceptable consequences due to jet impingement loadings could be identified for a design-basis break in the CRD system outside the primary containment.

The effects of crack breaks in this piping are the effects expected from breaks in moderate-energy systems, that is, spraying and flooding.

3.6.2.2.6.2.5 Evaluation of System, Component, and Structural Damage

An evaluation was made of the direct damage resulting from a break in the CRD piping system on various systems, components, and structures, and its impact on the ability to achieve a safe shutdown. This evaluation was made by reference to Subsections 3.6.2.1.4 and 3.6.2.1.5, and Table 3.6-1. No adverse consequences were identified.

3.6.2.2.6.3 Evaluation of Ability To Attain and Maintain a Safe Shutdown

There would be no adverse consequences resulting from a break in the CRD piping that could preclude attaining and maintaining a safe shutdown.

3.6.2.2.7 Reactor Building Heating System

An evaluation of the consequences of a pipe break in the reactor building heating system was carried out as described below. The piping of this system is routed through most of the floors and rooms in the reactor and auxiliary buildings. Only the steam lines of the building heating system are classified as high energy.

3.6.2.2.7.1 Review of Potential Damage

As in the case of the previously discussed systems, a review (Reference 24) of the potential damage resulting from a break in any of the building heating steam lines was conducted in accordance with the methods described in Subsection 3.6.1. The review took into account the equipment required for a safe shutdown, as discussed in Subsection 3.6.2.1.4 and Table 3.6-1.

The pressure in the reactor building heating steam lines (15 psig) is less than that which can cause a pipe to whip from postulated breaks. Subsequently, blowdown analyses, pipe whip evaluations and jet impingement evaluations were not performed. The effects of cracks in this piping are the same as those expected from breaks in moderate-energy systems; that is, spraying. An evaluation of the spraying effects on various systems, components, and structures did not identify any adverse consequences. The environmental effects of temperature and humidity were also analyzed for a break in the building heating steam lines. Secondary containment isolation valves are provided in the steam supply lines to isolate the steam source on indication of a break.

3.6.2.2.7.2 Evaluation of Ability To Attain and Maintain a Safe Shutdown

The loss of the building heating system does not affect the safe shutdown capability of the reactor.

3.6.2.2.8 Non-Safety-Grade Systems

A review of plant safety with regard to high-energy pipe breaks was performed using the format established by the BWR Owners Group in response to H. R. Denton's letter, Potential Unreviewed Safety Question on Interaction Between Non-Safety-Grade Systems and Safety-Grade Systems.

From this review, Edison has concluded that no identified safety action would be negated by the failure of non-safety equipment resulting from the environmental effects of a high-energy pipe break. The only minor area of concern is the temperature effects of the pipe break on the level instrumentation sensing lines, and this has been addressed and resolved in the generic BWR report, NEDO-24708.

This review indicates that no previously established safety limits would be violated by the environmental effects.

It is desirable that operator action be taken to quickly mitigate the effects of the failures in most cases.

The specific systems and areas considered are included in Table 3.6-8.

3.6.2.3 Moderate-Energy Pipe-Break Evaluation

Moderate-energy piping systems, as defined in Subsection 3.6.2.1.2.1 and listed in Subsection 3.6.2.1.3.2 have been evaluated (Reference 24) for postulated through-wall leakage cracks (refer to Subsection 3.6.2.1.1.2 for the method and Subsection 3.6.2.1.2.4 for the design bases for the crack size and location). The components and/or equipment required for the safe shutdown of the reactor were evaluated and, if necessary, provided with measures to protect and ensure their operability. The evaluation for the moderate-energy piping systems encompasses an analysis of both flooding and spraying effects.

The consequences of flooding would depend on the crack size, crack flow rate, drainage rate of the compartment, and the location within the compartment of the components required for safe shutdown. An accumulation rate (defined as the crack flow rate minus the drainage rate) was determined for each compartment, and the potential for water accumulation in each compartment was examined. If accumulation posed a flooding threat to components or equipment, an examination was undertaken to determine the possibility of damaging each component within that compartment and the acceptability of such damage. Where drain paths exist such that water accumulation may occur in adjacent compartments, an evaluation of the components and equipment damage in those compartments was carried out.

The consequences of spraying would depend on the spray distance and the spray angle. The spraying distance was determined for the highest-pressure line within each compartment in which components and/or equipment are located.

In many cases, numerous break locations were selected within each compartment so as to maximize the effect on any one component and/or equipment within that compartment. An examination of each component and/or equipment required for safe reactor shutdown was completed to determine the acceptability of damage. In either analysis, whether for the case of flooding or for the case of spraying, the basic problem was to establish whether the effect of a postulated leakage crack has the potential of preventing the safe shutdown of the reactor when combined with a random failure of a single component.

3.6.2.3.1 Analytical Procedure

A step-by-step procedure was used to determine which of the components within the reactor/auxiliary building and the residual heat removal (RHR) complex could have the potential of being damaged by either flooding or spraying. The steps include listing components and/or equipment required for safe shutdown, located in areas affected by spray or flooding. On the basis of the crack flow rates (Subsection 3.6.2.3.3.) and spray distances (Subsection 3.6.2.3.3.) for each postulated crack, a determination was made as to which of these components could fail as a result of spray or submergence. Finally, the ability to achieve safe shutdown was evaluated, assuming a single active failure in addition to the failures caused by spray and flooding (Subsection 3.6.2.1.1.2).

3.6.2.3.2 Evaluation Guidelines

The basic guidelines used in evaluating the effects of flooding or spraying were as follows:

a. All water pipes were assumed to have the required pressure to produce a spray that would reach the most distant walls in direct line of the spray; if unacceptable damage would result, this assumption was reviewed and validated

b. All valve operator motors have NEMA 4 enclosures and were not assumed to fail from water spray

c. All motors other than valve operators were evaluated on an individual basis. If they were in open drip-proof enclosures, they were assumed to fail when exposed to a spray

d. All motors were assumed to fail if submersed due to flooding, except the subbasement floor drain sump motors

e. Cables are waterproof and would be unaffected by flooding or water spray

f. Motor control centers and switchgear were assumed to fail if sprayed or if submersed due to flooding

g. Essential instruments that are NEMA 4 rated were not assumed to fail from water spray

h. All instruments were assumed to fail if submersed due to flooding

i. Essential local terminal boxes that are NEMA 4 rated were not assumed to fail from spraying effects

j. All terminal boxes were assumed to fail if submersed due to flooding.

3.6.2.3.3 Analytical Methods

As indicated in the analytical procedures (Subsection 3.6.2.3.1) for flooding and spraying, calculations must be performed to find the crack flow rate and the spraying distance for postulated through-wall leakage cracks. The methods used to determine these parameters are described in Reference 24.

3.6.2.3.4 Results of Evaluation

From the evaluations of flooding and spraying effects, it was possible to identify certain components and/or equipment that required protective measures to prevent their loss of function as a result of the pipe crack.

3.6.2.3.4.1 Protective Measures To Mitigate Flooding Effects

3.6.2.3.4.1.1 Residual Heat Removal Complex

The QA Level I components located in the RHR complex are

a. Standby diesel generators (four)

b. RHR service water pumps (four)

c. Emergency equipment service water system (EESWS) pumps (two)

d. Diesel generator service water (DGSW) pumps (four)

e. RHR cooling towers (four)

f. 4160-V switchgear, 480-V switchgear, and motor control centers

g. Standby diesel generator heating, ventilation and air conditioning (HVAC) systems.

Because of the structural arrangement and openings for drainage, a moderate-energy pipe rupture in those areas cannot cause a flooding problem.

3.6.2.3.4.1.2 Reactor/Auxiliary Building

Subbasement Flooding

The floor drains, open stairwells, and other openings (equipment hatches, pipe chases, and various penetrations) provide ample access to the subbasement for any water that would leak from a postulated pipe crack occurring on any floor of the reactor/ auxiliary building. Therefore, the analysis of damage from flooding in the subbasement covers floods that would result from pipe breaks on other floors as well as from breaks in the subbasement.

Flooding in the subbasement can occur in the torus room (room SB2), any of the corner rooms (rooms SB1, SB3, SB5, or SB6) or the HPCI room (room SB7). The maximum flooding rate in the torus room would result from an RHR pump discharge line. Other moderate-energy lines in the torus room would have a lower leakage rate, and thus an evaluation of the RHR leak represents the envelope for leaks from other systems. Secondary containment isolation valves are provided in the torus water management system (TWMS) return line to the torus to isolate the condensate systems on indication of leakage (see Figure 9.2-13).

The maximum flooding rate in a corner room would result from an RHR return line. Other moderate-energy lines in the corner rooms would have a lower leakage rate. Flooding in a corner room due to leakage from an operating system in that room would be self-contained and self-limiting since it would stop when the pump stops due to submergence of the pump motors. For the case of a leak in one RHR division corner room while the other division is in operation, the presence of a leak would be readily identified by the sump level indication and the system would be turned off.

Leaks of a magnitude great enough to cause flooding would be detected by water line pressure and flow instrumentation, leak-detection instrumentation, or the activation of the sump pumps and sump overflow alarms.

Shutdown Capability Evaluation

Indications that there is water leakage into the subbasement in excess of specified limits call for an immediate controlled shutdown. Since there would be no turbine generator or RPS trip resulting from a moderate-energy pipe break, offsite power would be available and shutdown would be carried out by normal shutdown procedures.

In cases of moderate-energy piping leakage, operator action would be required to identify the leaking system and the location of the leak. Instrumentation of various types is available to allow the operator to identify the leaking system. In most cases, location of the leak would require a search of the areas traversed by the identified leaking system. Some of the safety-related equipment in the subbasement prone to damage by flooding is the equipment in the specific corner rooms indicated in Figure 3.6-13 and the various pump suction valves in the torus room. If a leak in an RHR pump discharge line resulted in flooding in the torus room and the flooding proceeded at the fastest rate, more than 2 hr would be required to flood the motor operators of the RHR suction valves; thus the operator would have adequate time to locate and terminate the leak without damage to any safety-related system or component.

In applying the single-component failure criterion, the only system required for normal shutdown that could be made inoperable is the RHR system. Should a leak develop in one division of the RHR, and the single-failure criterion were applied to the RHR divisional cross-tie valve, both divisions of the RHR would be disabled since continued operation of the nonleaking division would force water out of the leaking division and there would be no method of determining the leaking division. In this event, the primary system would be held at low-pressure hot standby until the malfunctioning valve were repaired and closed so that the leaking division could be identified. Once this were accomplished, the system could be taken down to a cold shutdown using the redundant RHR division, where the single-failure criterion is not applied in accordance with Subsection 3.6.2.1.1.2.

Loss of equipment in any single corner room together with an assumed single failure would not preclude attaining and maintaining a safe shutdown.

<u>Basement</u>

The basement consists of the four corner rooms enclosing various instrument racks. The flow rate from pipe cracks in any room in the basement would be relatively small; in all cases the water would drain to the subbasement rooms, and the evaluation for the subbasement flooding applies.

<u>First Floor</u>

Water accumulation could occur in a few areas on this floor; however, no safe-shutdown equipment is located in these areas. In other areas, sufficient floor drains and/or other openings (doors and stairwells) are provided to limit the accumulation to a few inches in depth. The water that leaked on this floor would eventually flow down to the subbasement, and the evaluation for the subbasement flooding applies.

Where RBCCW supplemental cooling supply and return piping to the Division I EECW loop passes along the floor between the control rod drive hydraulic control units, the HCUs have been evaluated for the impact of spray and jet impingement and found not to be impacted as a result of postulated cracks in this piping.

<u>Second Floor</u>

In general, the water that leaked on this floor from a postulated pipe crack in moderate-energy piping would drain to the subbasement area through the floor openings and/or stairwells, where its effect would be smaller than that of the postulated pipe crack in the subbasement.

Moderate-energy lines located in room 209 have the potential of damaging nearby electrical equipment as a result of postulated pipe cracks. These lines were shrouded to preclude this possibility.

Third Floor

The third floor contains the control room complex (room 308), motor control center (room 320), RPS motor-generator sets (rooms 32l and 322), switchgear (room 323), and batteries (rooms 325 and 326). The doors from room 3l8 provide access to the control room complex and to the switchgear room through the motor control center room. These rooms contain equipment that is sensitive to moisture and is required for the safe reactor shutdown.

Room 320 contains emergency equipment cooling water (EECW) supply and return lines that are routed near the motor control center cabinets, divisional cable trays, and fire protection system elements. The lines are shrouded to prevent these items from being sprayed and to prevent any significant water accumulation in room 320 in the event of a leakage crack in the lines. Room 318 contains no safety-related equipment. Water from a crack in either the EECW lines or fire protection header located in this room would spread directly to the turbine building. No significant accumulation of water in room 318 will be experienced. The closure strips that have been installed at the bottom of doors leading from room 318 into room 308 or room 320 will prevent water from spreading into the control room. In addition, small leakage allowed through the door in room 320 will not affect the equipment.

Fourth Floor

The secondary containment ventilation room (room 416) contains four pipes of about equal potential for causing flood damage. Two pathways are available for the spread of water; the open stairway (room 319) leading down to room 318, and the equipment hatch located over room 320. Water retained on the floor of room 416 will spread over the large floor area. No significant accumulation will be experienced.

Water descending the stairway will spread over the floor surface of room 318 into the turbine building. No significant accumulation will be experienced.

Water could leak into room 320 via the small holes in the equipment hatch. The water would spread over the room 320 floor surface and into adjacent rooms (rooms 321, 322, 323, 325, and 326). These rooms contain safety-related equipment that could be affected by water spray or accumulation. Therefore, a perimeter curb has been installed around the hatch. Also, a plastic cover has been installed over the hatch. The fire-protection line that is routed over the hatch area has also been shrouded. These measures will prevent water from leaking into room 320.

Fifth Floor

There is little effect on reactor safe shutdown from a postulated pipe crack in a moderate-energy piping system on this floor. Sufficient floor drains and openings to floors below (staircases and equipment hatches) have been provided to prevent flooding from postulated leaks. The water from this floor going to floors below has less effect than that of a postulated pipe crack on those floors. However, the following are some of the rooms that have been modified to mitigate the effect of a postulated pipe crack on this floor.

A maximum flow rate of 78 gpm from a postulated pipe crack of a moderate-energy reactor building closed cooling water (RBCCW) line is postulated in room 509. A floor drain is provided to prevent any water accumulation; however, there are duct penetrations leading to the control room below. To mitigate any possibility of water flowing through these penetrations, each penetration was sealed after its installation. A pipe crack can also lead to water flow down the stairwell leading to room 3l9 and thus to room 318.

3.6.2.3.4.2 Protective Measures To Mitigate Spraying Effects

3.6.2.3.4.2.1 Residual Heat Removal Complex

The safety-related components and equipment located in the RHR complex were discussed in Subsection 3.6.2.3.4.1.1. Water spray can adversely affect some safety-related equipment. Because of the separation arrangement, however, this damage would be limited to one division in any one system. Since the single-failure criterion is not applied to a redundant train of a dual-purpose system damaged by a pipe crack, the ability to achieve and maintain a safe shutdown is not jeopardized.

3.6.2.3.4.2.2 Reactor/Auxiliary Building

Following the procedure presented in Subsection 3.6.2.3.1, components and equipment required to ensure a safe shutdown and having the potential of being damaged as a result of a water spray from a moderate-energy pipe crack were identified. See Subsection 3.6.2.3.4.1.2.

3.6.2.4 Conclusions

3.6.2.4.1 High-Energy Piping Systems

Following the criteria described in Subsection 3.6.2.1, the main steam, feedwater, HPCI steam, RCIC steam, RWCU, building heating steam line, and CRD systems were evaluated for the effects of pipe rupture outside primary containment. This evaluation, described in Subsection 3.6.2.2, encompassed the effects of pipe whip, jet impingement, flooding, and environmental effects. The conclusions that were reached are described below.

3.6.2.4.1.1 Pipe Whip Effects

The effects of the unrestrained motion of segments of the afore-mentioned piping systems, due to thrust loads developed at postulated piping breaks, have been investigated. Pipe restraint designs have been installed that will mitigate the adverse effects of the pipe whip.

3.6.2.4.1.2 Jet Impingement Effects

The effects of jet impingement from the postulated piping breaks have been investigated. To mitigate the consequences of jet impingement, several postulated break locations are equipped with jet impingement shields to protect the affected systems, structures, and equipment.

3.6.2.4.1.3 Environmental and Flooding Effects

The environmental effects, including flooding and the effluent of a steam/air mixture, of postulated high-energy piping breaks have been investigated. It has been concluded that the components and equipment required for effecting and maintaining a safe shutdown are protected and the main control room will remain habitable.

3.6.2.4.1.4 Pressurization Effects

The pressurization effects of a postulated high-energy line break on the main steam tunnel, reactor/auxiliary building structures, and certain components and equipment have been investigated. The components and equipment required to effect and maintain a safe shutdown are protected and the main control room will remain habitable.

3.6.2.4.2 Moderate-Energy Piping Systems

Following the criteria described in Subsection 3.6.2.1, moderate-energy piping systems were evaluated for the effects of throughwall leakage cracks. This evaluation, described in Subsection 3.6.2.3, encompassed the effects of flooding and spraying. The conclusions reached are described below.

In the event of substantial flooding in the torus room, resulting primarily from an RHR pump discharge line leak, but including other lines as well, operator action is required to terminate the leakage. Maintenance work would be required to achieve and maintain a cold shutdown if the single-component failure criterion were applied to the cross-tie valve between the two RHR divisions.

Flooding into the third floor control room or switchgear room containing equipment required for a safe shutdown has been evaluated. Provisions were made to ensure that such flooding cannot occur either directly from other areas of the third floor or by leakage through penetrations from above. In addition, moderate-energy lines presently located in the third floor switchgear room are shrouded to preclude the possibility of spraying and flooding.

3.6.3 Analysis Methods and Procedures

The methods and procedures used for the evaluation of pipe breaks of high-energy systems outside containment are presented in Subsections 3.6.3.1 through 3.6.3.4.

3.6.3.1 Blowdown and Environmental Effects Analyses

An analysis was performed to predict system blowdown response for each of the postulated high-energy line ruptures. The blowdown information is used to determine pipe whip forces and jet impingement characteristics, and may also be used as input in a number of other thermal-hydraulic analyses, depending on the requirements and problems anticipated for a particular break. In cases where structural damage could result from overpressure caused by rupture, the blowdown flow results are input in a compartment pressurization analysis. In situations where building temperature and humidity in the postbreak environment could damage required electrical, instrumentation, or control equipment, the blowdown flow results are input in a building environment analysis. Postulated ruptures resulting in release of significant amounts of subcooled water were evaluated for effects of flooding in the building housing the broken lines.

The criteria and methods to be used for the thermal-hydraulic analyses are described below.

3.6.3.1.1 Blowdown Analysis

The blowdown analyses may be characterized as either short term or long term in nature, depending on the purpose for which the results are used. Thrust data required for pipe whip analysis can be obtained in short-term analyses (about 500 msec real time) since maximum piping response is reached within this time. Mass and energy flow data required for compartment pressurization, environmental, and flooding analyses must be based on longer term blowdown information since the severity of these effects may increase with continuing flow from the break.

Typically, the duration of short-term analyses is insufficient to allow activation of operable components (other than check valves), and the balance of plant systems (other than the inoperable system) continue to operate in the normal way. For the long-term analyses, consideration must be given to action of operable components, interaction of other systems with the broken system, and the effects of shutdown of the reactor.

The following general criteria govern the blowdown analysis:

a. Analyses shall consider flow from both sides of the break

b. Discharge coefficients shall equal 1.0 for all breaks

c. Credit shall be taken for flow limiters, line restrictions, and pipe friction as applicable

d. Breaks shall be assumed to occur instantaneously

e. The initial conditions for a break shall be the worst-case operational condition.

3.6.3.1.2 PRTHRUST Program

The blowdown analyses were performed using the computer program PRTHRUST (Reference 40). The PRTHRUST program is a modification of RELAP3 (Reference 41), the AEC's presently accepted LOCA analysis code (Reference 42) for the specific requirements of pipe rupture analysis. In PRTHRUST, the fluid system is mathematically modeled as an assemblage of control volumes interconnected by flow paths. Characteristics of control volume include the state of the contained fluid and possible energy addition. Control volumes are used to model such components as pressure vessels, steam generators, heat exchangers, and the piping volumes. Flow paths are used to interconnect control volumes and may include operable valves, check valves, fills, and pumps. The program allows actuation of operable devices, such as valves, to be triggered at a specific time or based on a physical signal such as pressure or flow at a point in the system. The variation in pump performance under transient conditions is considered. A core model is available for cases in which transient reactor performance effects blowdown.

Initial values for the problem are taken as steady-state operating conditions for the system. The transient is initiated by instantaneously opening a leak in the system. The solution proceeds by step-by-step integrations of the governing fluid equations with time. The requirement for conservation of mass and energy in a volume is satisfied at each time step. State properties in the volumes are calculated using thermodynamic state equations and the ASME steam tables (Reference 43) for subcooled, saturated or superheated fluids. The flow rate for flow paths between volumes is calculated using both the one-dimensional momentum equation and Moody's two-phase choked flow model (Reference 44). The lesser of the two flows is assumed to govern.

The PRTHRUST program output includes time-history values of mass flow, pressure, temperature, enthalpy, and other thermodynamic quantities at specified points in the system and at the break. This information is suitable for input in subsequent jet impingement, compartment pressurization, environmental, and flooding analyses as required.

The program also calculates break thrust as a function of time. This calculation is facilitated by placement of control volume(s) in the model between the break and the piping elbow(s) nearest the break, and is carried out using the following equation.

$$T = T_{pt} + T_{mt} + T_a = (P_t - P_e)A + \frac{\rho A V^2}{g} + \frac{\Delta(MV)}{\Delta t g} \quad (3.6\text{-}5)$$

where

T	=	total thrust at break
T_{pt}	=	pressure thrust
T_{mt}	=	momentum thrust
T_a	=	thrust due to fluid acceleration
P_t	=	throat pressure at break
P_e	=	ambient pressure
A	=	break area
ρ	=	density
V	=	velocity
g	=	Newton's constant
M	=	mass
t	=	time

Throat pressure is given by the Moody correlation (Reference 44) for choked flow, and is taken as equal to ambient pressure for nonchoked flow. The momentum change term is equal to zero for steady flow.

3.6.3.1.3 Building Pressurization and Environmental Analyses

In cases where compartment pressure or steam environment resulting from a postulated rupture could result in damage to structures, systems, or equipment required for safe shutdown, an analysis was performed to assess the magnitude of these effects. Only the worstcase break in each compartment was analyzed.

The mass and energy input from the break was determined by a long-term blowdown analysis, assuming the most adverse reactor operating conditions. The analyses of longterm compartment pressures and environment are generally performed using the CONTEMPT-LT or KITTY6 computer program. Where expedient, however, conservative hand calculated mass and energy balance is used. The failure point of all components (doors, vents, walls, etc.) that may alter fluid flow paths are determined and the effects of such failures considered in the analyses. The initial positions of all doors or other movable vents are assumed to be in the most adverse normal condition. A discharge coefficient of 0.6 is conservatively assumed for all vent areas unless another value can be justified analytically.

3.6.3.1.4 CONTEMPT-LT Computer Program

The CONTEMPT-LT computer program (Reference 45) predicts the time-history of pressure, temperature, and humidity response in a group of interconnected compartments, resulting from a high energy pipe break in one compartment. Input break characteristics are the mass and energy flows computed in the blowdown analyses. In the program, compartments are represented by control volumes, and are interconnected by flow paths representing the venting areas. Venting to the outside may also be considered. Vents may be opened at a specified pressure to represent active components or failure of components such as doors.

Each control volume is separated into variable liquid and vapor regions. While each region is assumed to be uniform temperature, the liquid and vapor temperature may be different. Mass and energy transfer between the two regions is permitted, based on condensation and/or evaporation correlations, as applicable. The program solution is by step-by-step integration of the mass conservation and energy equations with time.

The program also includes the capability to perform one-dimensional heat conduction calculations. This capability can be used to account for the heat sinking effect of various building components by specifying appropriate initial and boundary conditions. The effect of building venting and leakage can be accounted for through use of available correlations for flow through small and large openings.

The CONTEMPT-LT output includes values of building pressure, temperature, and relative humidity in each compartment as a function of time.

3.6.3.1.5 Flooding Analyses

The analyses for flooding effects were carried out using hand calculational methods. Input was taken from mass and energy flow results of blowdown analyses for pressurized water lines. Consideration was also given to flooding caused by steam condensation as determined in the building environment analyses described in the preceding section. The flooding analysis for a particular compartment was performed only for the worst-case break for that compartment, although possible secondary effects, such as rupture of a second line by a whipping pipe, were considered.

The most adverse system operating conditions were assumed at the time of the break. In determining the mass of water released, failure of the active component leading to the maximum release was assumed.

The flooding analysis method was based on determination of compartment free volumes as functions of elevation, along with available drainage capability and flow-path characteristics between connecting compartments.

Using this information along with calculated water flow rates, the maximum water level and rate of water-level rise was determined for each compartment.

3.6.3.1.6 CVPT-REPORT Computer Program

The CVPT-REPORT computer program is used for the determination of compartment pressure and environmental response due to a postulated instantaneous pipe rupture in a high-energy piping system.

CVPT-REPORT is a design and analysis tool that can treat one compartment or 16 interconnected compartments of a completely arbitrary arrangement. Interconnecting paths modeled as orifices or pipe sections are accepted by the program. The break flow rate from the postulated break is determined by the program throughout the incident by considering input conditions of reactor vessel steady-state fluid conditions, pipe losses to the break point, flow restrictions, and isolation valve actuation cycles, including closing times and flow choking.

The CVPT-REPORT program uses a step-by-step integration method of the mass conservation and energy equations with time to obtain the environmental response of each compartment during the incident. State properties in the compartments are calculated using thermodynamic uniform-flow, uniform-state equations and the ASME steam tables for wet, saturated, or superheated steam. The flow rate for flow paths between compartments is calculated using the Darcy formula for compressible flow through orifices or a formula for compressible isothermal flow in pipelines. For postulated breaks where the process fluid is saturated or superheated steam, the mass flow rate out of the break is calculated by using a formula for the choked flow of a compressible gas through an isentropic nozzle. For cases where the process fluid is saturated liquid, the mass flow rate out of the break is determined by using the Darcy formula for the discharge of fluid through valves. The heat-transfer effect to the compartment walls is considered through incorporation of the Uchida heat transfer coefficients for a range of air-to-steam ratios (Reference 46).

The CVPT-REPORT program output includes time-history response of mass flow rate, and the pressure, temperature, and humidity of each compartment. This information is suitable for input into any subsequent long-term analysis or structural and component damage analysis.

3.6.3.1.7 KITTY6 Computer Program

The KITTY6 computer program is used to determine transient temperature and pressure responses in various areas of the reactor building for the HELB and LOCA accident cases. This problem is a transient heat transfer problem which depends upon the initial conditions, the boundary conditions and the characteristics of the system. The problem is solved numerically using the computer program KITTY6.

KITTY6 calculates node properties and path heat flow and mass flow rates for transients in user specified solid and/or fluid channel configurations. Paths between nodes may be used to model conductive, convective and radiative heat transfer and mass and enthalpy transport. In the compressible fluid system (CFS) of the model, elevation effects may be accounted for, compressible fluid flow paths may be represented as either of inertial or non-inertial (pseudo-steady) types, and limitation of flow path rates to the choking flows may be elected. Water and as many as five noncondensible gas species may be treated in the CFS. Provided the configuration specification, material properties, boundary conditions, internal heat generation rates, selected fluid flow rates and required printout times, KITTY6 computes and prints the node properties (temperature, pressure, density, composition and mass flow rates), and path mass and energy flow rates at user specified time intervals.

KITTY6 was utilized to re-evaluate environmental conditions resulting from HPCI, RCIC and RWCU breaks as referenced in the previous sections. The original short-term analyses for these breaks were not impacted by the revised environmental analyses.

3.6.3.2 Jet Impingement Analysis

This section defines analytical methods used for performing the jet impingement evaluation. The effects of jet impingement were considered for all longitudinal design-basis, longitudinal crack, and circumferential crack breaks. The jet impingement effects resulting from fluid discharge from both ends of the severed pipe were considered for all circumferential design-basis breaks, unless it could be shown that the two ends are sufficiently restrained to prevent offset after rupture. The sweep of the jet was considered for all design-basis breaks subject to pipe whip.

The break opening configuration for postulated design-basis breaks is defined in Subsection 3.6.1.1.2. The fluid jet is assumed to fan out to form a circular or prismatic cone issuing from the break opening. Jet impingement pressure on a target struck by the jet is calculated by determining total thrust at the break, and assuming this total integrated thrust remains constant at any plane of interest in the cone.

The axis of the jet is parallel to the pipe axis for design-basis circumferential breaks and perpendicular to the pipe axis for all other break types. The characteristics of the jet shapes for the various postulated break types are shown in Figure 3.6-45.

3.6.3.2.1 Total Thrust Load

The calculation of thrust on the pipe after rupture was described in the blowdown analysis. Using the principle of conservation of momentum, the steady-state jet thrust can be equated to the thrust on the pipe. This conclusion has been confirmed by Moody (Reference 47).

The total jet thrust for breaks can be equated to the maximum in the quasi-steady-state region of the thrust time curve. The rise time for the jet thrust can be taken as the time to reach this quasi-steady-state peak. The initial peak in the thrust-time curve, which is caused by acceleration of fluid from the pipe, does not influence the jet impingement load on a target. A graph showing determination of total thrust from the PRTHRUST results is shown in Figure 3.6-46.

3.6.3.2.2 Coning Angle

The fluid jets from steam or flashing water breaks were assumed to fan out with a constant half angle of 12.5°. The experimental basis for this assumption is found in Reference 48, which includes photographs of jets of both steam (about 100 psia saturated) and water (at 2250 psia and 550°F). For subcooled, nonflashing breaks, a jet divergence angle of 10° was assumed.

3.6.3.2.3 Jet Temperature

The jet temperature will vary with distance from the jet source. However, jet temperature will not exceed the stagnation temperature based upon an isenthalpic expansion of jet fluid. The jet temperature limits are

For steam line breaks:

$$T_j = 330°F \tag{3.6-6}$$

For liquid line breaks:

$$T_j = 240°F \text{ maximum} \tag{3.6-7}$$

$$= \text{Fluid temperature if less than } 240°F \tag{3.6-8}$$

where

T_j = jet stagnation temperature (°F)

3.6.3.2.4 Target Loading

The normal load applied to a target by the jet issuing from a postulated break may be expressed as

$$F = T\frac{A_i}{A_j} S_F D_{LF} \tag{3.6-9}$$

where

T = total thrust of jet (lb_f)

A_i = cross-sectional area of jet intercepted by target structure

A_j = total cross-sectional area of jet at target structure

S_F = shape factor

D_{LF} = dynamic load factor

The total thrust T has been defined previously in the blowdown analysis. The ratio A_i/A_j represents the proportion of the total mass flow interrupted by the target structure. The dynamic load factor D_{LF} accounts for the rapid application of the load. A dynamic load factor of two should be used in the absence of an analysis justifying a lower value.

The shape factor S_F depends on the projected section and orientation of the target struck by the jet and is a measure of the target's potential for changing the momentum of the jet.

Typical shape factors for perpendicular impingement at turbulent flow conditions are

a. 0.5 to 0.6 for piping spans up to ten diameters

b. 0.4 for spherical shapes

c. 0.2 for ellipsoidal shapes (stream lined)

d. 1.25 for flat plates.

The force determined using the above formula and factors represents the integral of a uniform pressure applied normally to the target impinged upon by the jet.

3.6.3.3 Pipe Whip Analysis

A pipe whip analysis was carried out for each of the design-basis ruptures postulated in high-energy lines identified as requiring this type of analysis. This analysis serves initially to identify situations in which whipping pipes could cause unacceptable damage to systems, equipment, or structures, and later to develop locations and design loads for restraints required to prevent unacceptable pipe whips. The steps in the pipe whip analysis are as follows:

a. An analysis is performed to predict piping response to the rupture thrust load. This analysis determines whether maximum moments and torque in the piping exceed values necessary to cause plastic hinging and whether a sufficient number of plastic hinges form to effect a plastic failure mechanism or pipe whip

b. If pipe whip takes place, the trajectory of the whipping pipe is traced to identify impact on systems, equipment, or structures. Mechanical, HVAC, electrical, instrumentation, and control components are considered to fail upon impact by whipping pipes, unless stress analyses justify otherwise. Structural components are evaluated to determine whether pipe whip impact causes failure in accordance with the methods described in this section and in Subsection 3.6.3.4

c. Locations selected for pipe restraints prevent the occurrence of unacceptable pipe whips. Sizing analyses were performed to determine stiffness and strength characteristics of these restraints.

d. Finally, a dynamic response analysis of the complete piping system and identified restraints was performed. This analysis verified the fact that unacceptable pipe whips do not occur in the restrained system, and provided maximum reaction loads for use in final design of the restraints.

The criteria for the pipe whip evaluation, and the analytical formulation of the analyses described above, is given in the following sections.

3.6.3.3.1 Criteria for Analysis

The following general criteria were applied in the pipe whip evaluation:

a. The dynamic nature of the piping thrust load shall be considered. In the absence of analytical justification to the contrary, a dynamic load factor of 2.0 shall be used

b. Nonlinear (elastic-plastic strain hardening) pipe and restraint material properties shall be considered. Pipe whip shall be considered to take place on attainment of a hinge mechanism in which maximum fiber strain reached 50 percent of that strain corresponding to maximum stress in a one-dimensional tensile test

c. Pipe whip was considered to result in unrestrained motion of the pipe along a path governed by the hinge mechanism and the direction of the vector thrust of the break force. A maximum of 180° rotation was assumed to take place about any hinge

d. The effect of rapid strain rate on material properties was considered. In the absence of justification to the contrary, a l0 percent increase in yield and ultimate stress under dynamic load was assumed

e. Variations between lower- and upper-bound material properties were considered in the most conservative fashion. For example, use of lower-bound properties provided a conservative prediction of pipe whip, while use of upper-bound properties was conservative for determination of maximum restraint loads. In the absence of data justifying the contrary, lower bounds were taken as minimum guaranteed properties, with a 40-percent statistical increase for upper-bound properties

f. Where possible, required pipe whip protection was provided by designing normal operating pipe restraints to withstand pipe rupture loads. Pipe whip restraints required at locations where resultant piping thermal stress would preclude use of rigid supports were designed with an initial clearance sufficient to allow. free thermal expansion of the pipe. The clearance restraints used a deformable energy-absorbing component retained by a support substructure. Energy-absorbing components were designed to withstand pipe impact without exceeding 50 percent of ultimate capacity. Rigid supports and support substructures were designed in accordance with the criteria given in Subsection 3.6.2.1.5.

3.6.3.3.2 <u>Preliminary Pipe Whip Evaluation</u>

The methods in this section were used to determine whether pipe whip takes place for a given postulated rupture, and to determine the kinetic energy of whipping pipes on impact with a target.

A pipe whip occurs when a hinge mechanism forms in the system that has a structural resistance less than the applied thrust force. The mechanisms consist of straight runs of pipe connected by fittings (elbows, etc.) that yield under a combination of internal moment and torsion. The condition for formation of a plastic hinge at a given location in a piping system is

$$\left(\frac{iM^2}{M_{ult}} + \frac{T}{T_{ult}}\right)\ 2 \geq 1 \qquad (3.6\text{-}10)$$

where

M = applied moment

= $\sqrt{M_l^2 + M_2^2}$

M_1, M_2 = moment components in plane perpendicular to pipe centerline

M_{ult} = ultimate moment

i = stress intensification factors for elbows, tees, etc.

T = applied torque

T_{ult} = ultimate torque

The ultimate moment and torque are limited by the allowable 50 percent uniform ultimate strain. Expressions for these quantities, based on assumed elastic-linear strain-hardening material properties, are given by the following:

$$M_{ult} = \sigma_y Z_p + (\sigma_{ult} - \sigma_y) Z_e \qquad (3.6\text{-}11)$$

$$T_{ult} = t_y z_{tp} + (t_{ult} - t_y) Z_{te} \qquad (3.6\text{-}12)$$

where

σ_{ult} = tensile stress corresponding to 50 percent of strain at maximum tensile stress

σ_y = tensile yield stress

Z_p = plastic bending section modulus

= $(4/3)(r_o^3 - r_i^3)$

Z_e = elastic bending section modulus

= $\frac{\pi(r_o^4 - r_i^4)}{4r_o}$

t_y = shear yield stress

= $\sigma_y/2$ (maximum shear theory)

t_{ult} = ultimate shear stress

= $\sigma_{ult}/2$

Z_{te} = elastic torsion section modulus

= $\frac{\pi(r_o^4 - r_i^4)}{2r_o}$

Z_{tp} = plastic torsion section modulus

= $\frac{2\pi}{3}(r_o^3 - r_i^3)$

r_o = pipe outside radius

r_i = pipe inside radius

Formation of a sufficient number of hinges produces a mechanism that moves under the action of the blowdown force and is resisted by the constant limit load and inertia of the mechanism. The resulting motion may be determined using simple kinematic formulas.

As an example, the motion resulting from a postulated longitudinal break at joint B is to be determined for the piping system shown in Figure 3.6-47.

The structural resistance (R_e) can be determined by subjecting the mechanism to a virtual displacement (w). Equating the work done by the limit load to the strain energy dissipated in the yield hinges (all hinges assumed to have the same plastic moment) results in

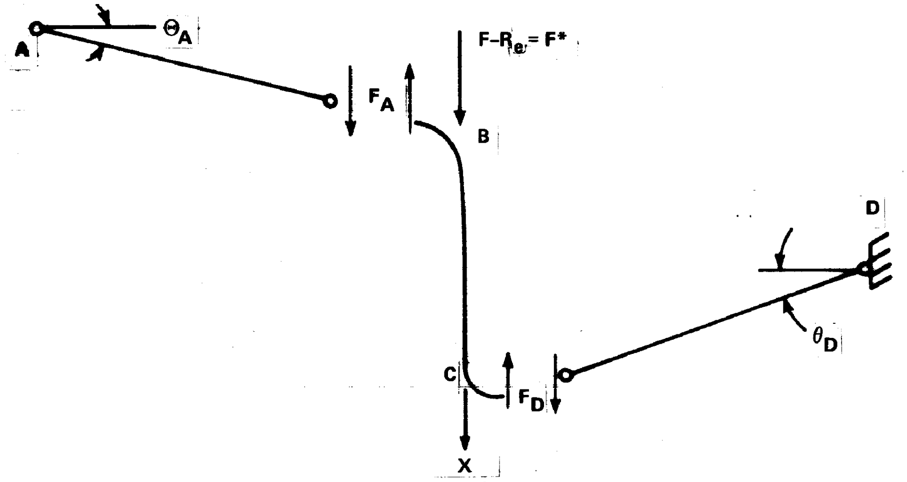
$$R_e(w) = 2M_p\theta_A + 2M_p\theta_D \qquad (3.6\text{-}13)$$

Substituting $\theta_A = {}^{w}/_{L_A}$, $\theta_D = {}^{w}/_{L_C}$, and eliminating w yields

$$R_e = 2M_p\left(\frac{1}{L_A} + \frac{1}{L_C}\right) \qquad (3.6\text{-}14)$$

If the blowdown force exceeds R_e, the system is unstable and pipe whip takes place. Section A-B will rotate clockwise about point A, and Section C-D will rotate counterclockwise about point D; in other words, the trajectory is determined assuming the runs are inextensional.

Impact velocity is determined by writing the dynamic equilibrium equations for the hinge mechanism subject to the action of the net force (blowdown force less structural resistance).



For Section A-B

$$F_A L_A = I_{AB}\ddot{\theta}_A \qquad (3.6\text{-}15)$$

For Section B-C

$$F - R_e - F_A - F_D = M_{BC}\ddot{\chi} \qquad (3.6\text{-}16)$$

For Section C-D

$$F_D L_C = I_{CD}\ddot{\theta}_D \qquad (3.6\text{-}17)$$

where

I = rotational mass moment of inertia of pipe about one end

$= \frac{ML^2}{3}$

M = mass of pipe

L = length of pipe

Substituting the expression for mass moment of inertia and $\theta = \chi L$, and combining the three equations, results in

$$F^* = F - R_e = \left[\frac{M_{AB}}{3} + M_{BC} + \frac{M_{CD}}{3}\right]\ddot{\chi} = M^*\ddot{\chi} \qquad (3.6\text{-}18)$$

where

F^* = apparent force

M^* = apparent mass of mechanism

Having the force F^* and the apparent mass M^*, the velocity and kinetic energy at any displacement d is given by

$$V_d = \sqrt{2\frac{F^*}{M^*}d} \qquad (3.6\text{-}19)$$

and the kinetic energy is given by

$$(KE)_d = \frac{M^*V_d^2}{2} = F^*d \qquad (3.6\text{-}20)$$

= Work done during displacement

The formulation described above can be altered very simply to evaluate the case of a circumferential (guillotine) break at point C. In this case, the limit load for Section C-D is given by

$$R_e = \frac{M_p}{L_{CD}} \qquad (3.6\text{-}21)$$

where

F^* = $F - R_e$ as before

M^* = $\frac{1}{3}M_{CD}$

and the equations for velocity and kinetic energy at any displacement can be applied as before.

The formulation above can thus be used to evaluate whether pipe whip takes place, the trajectory of the whip if formed, and the kinetic energy of the whipping pipe on impact with a target.

3.6.3.3.3 Preliminary Design of Pipe Whip Restraints

The preliminary design of pipe whip restraints designed to maintain contact with the pipe during all operating conditions were carried out using the SAP IV computer program (Reference 49). The preliminary design of pipe whip restraints designed with an initial clearance between pipe and restraints were carried out using the RAP computer program (Reference 50). The descriptions of these programs and methods for their use are given below.

3.6.3.3.4 RAP Computer Program

The RAP program performs a time-step integration solution of the dynamic equilibrium equation for a mass (the pipe) subjected to a force-time-history (the blowdown force) impacting a bilinear strain-hardening viscous-damped spring (the restraint). The pipe mass assumed in RAP is the apparent mass of the whip mechanism as described in the preceding section. The solution makes use of kinematic relationships among accelerations, velocities, and displacements at the beginning and end of each time step to reduce the second-order differential equation of motion to a form that may be solved algebraically over the time increment (References 50 and 51).

The incremental equilibrium equation for the system shown in Figure 3.6-48 is the following:

$$M\Delta\ddot{X} + C\Delta\dot{X} + K\Delta X = \Delta F \qquad (3.6\text{-}22)$$

where

M	=	effective mass of pipe
C	=	viscous damping coefficient
K	=	restraint stiffness
F	=	applied blowdown force
X	=	displacement
Δ	=	an increment of succeeding quantity
·	=	superscript indicating derivative w.r.t. time

In addition, if the acceleration of the mass is assumed to change linearly over a time step, the following relationships can be written:

$$\Delta\ddot{X}_{N+1} = \frac{6}{DT^2}\Delta X_{N+1} - \frac{6}{DT}\dot{X}_N - 3\ddot{X}_N \qquad (3.6\text{-}23)$$

$$\Delta\dot{X}_{N+1} = \frac{3}{DT}\Delta X_{N+1} - 3\dot{X}_N - \frac{DT}{2}\ddot{X}_N \qquad (3.6\text{-}24)$$

where the subscript "N" represents a quantity taken at the "Nth" time step and DT is the length of the time step. If Equations 3.6-23 and 3.6-24 are substituted into Equation 3.3-22, the result is

$$\left\{\frac{6M}{DT^2} + \frac{3C}{DT} + K\right\} \Delta X_{N+1} = \Delta F + \left\{\frac{6M}{DT} + 3C\right\}\dot{X}_N + \left\{3M + \frac{DT(C)}{2}\right\}\ddot{X}_N \qquad (3.6\text{-}25)$$

which can be solved for ΔX_{N+l}, since $\dot{X}_N$ and $\ddot{X}_N$ are known initial conditions at the beginning of the time step. Having ΔX_{N+l}, Equations 3.6-23 and 3.6-24 can be used to determine the change in acceleration and velocity during the time increment. At each step during the incremental process, the status of the restraint is checked to determine whether it is detached, elastically loading, plastically loading, or elastically unloading, and appropriate changes are made to the initial gap, yield deflection, and restraint stiffness.

3.6.3.3.5 SAP IV Computer Program

SAP IV (Reference 49) is a finite element computer program for linear elastic analysis of arbitrary three dimensional structures. The program includes a variety of beam, plate, shell, and solid elements. The program can consider applied static loads, thermal expansion, seismic response spectra, and time-history dynamic loads. The program numerical techniques and core storage allocation methods have been designed to permit analysis of largescale problems at reasonable cost, although smaller problems can be solved with no loss in efficiency.

3.6.3.3.6 Final Analysis of Piping System and Restraints

The final system analysis was performed to verify that the restraints selected fulfill their intended function in preventing unacceptable pipe whips, and to provide final design loads on the restraints. The final system analysis was carried out using the PIPERUP computer program (Reference 52).

The PIPERUP computer program performs nonlinear dynamic analysis of piping systems subjected to rupture thrust forces. PIPERUP can be used both to predict formation of pipe whips and to determine loads on piping anchors and pipe whip restraints. The program is based on the finite element method of analysis, with the piping represented as an assemblage of straight and curved beam elements, and the restraints as axial and rotational springs. The solution is a time-step integration of the system equations of motion.

Piping element stiffness is arranged to permit representation of elastic and linear strain-hardening material properties. Each element is initially represented as a combination of these sub-elements, whose sum stiffness equals the elastic stiffness of the pipe. If at a given time-step the element internal forces are detected to exceed the yield capacity of the pipe, one of the subelements is hinged, such that the stiffness of the remaining two subelements corresponds to the strain-hardening modulus of the material. The analysis is then continued; if the internal forces are later detected to exceed the ultimate capacity of the pipe, the second subelement is hinged, leaving a single subelement with a very small stiffness. Prediction of the yield and ultimate hinge transitions is based on a formulation derived in accordance with the von Mises theory, which considers biaxial bending and torsional stresses. In the event that unloading occurs from the plastic region, such unloading is along the elastic line (isotropic strain-hardening model). Prediction of a plastic collapse mechanism, or pipe whip, is based on detection of excessive deflections.

The modeling of restraints in the analysis can include initial gaps, and elastic and linear strain-hardening stiffnesses. The effects of impact on restraint loading are accounted for automatically in the solution technique.

Program output includes time-history values of deformation, internal loads, material strains, restraint reactions, and identification of pipe whip mechanisms.

3.6.3.4 Structural Analysis

Structural analyses were performed to assess the ability of essential plant structures and structural components to withstand loads resulting from postulated ruptures. These analyses included

a. Analysis of structural components and structures for pipe whip impacts and jet impingements

b. Analysis of structural components for compartment pressure, temperature, and hydrostatic flooding loads

c. Analysis of piping anchor structures for pipe-break loads

d. Analysis of structures and structural support systems for pipe whip restraint and jet impingement barrier reaction loads.

The criteria governing acceptability of postulated rupture loads on essential structures and structural components have been given in Subsection 3.6.2.1.5. As indicated in that section, the structural analyses are generally performed using limit analysis techniques, such as collapse load analysis for beams and frames and yield line theory for concrete slabs, which account for resistance of structural elements into their plastic range. The description of these techniques follows.

3.6.3.4.1 Characteristics of Pipe Rupture Loads

The structural loads resulting from pipe rupture can, in general, be categorized as either impulsive or impactive in nature. The time variation of impactive dynamic loads is dependent on the initial kinetic energy of the impacting body, and on the stiffness and inertial resistance of the impacting body and the structure to which the loads are applied. The time variation of impulsive dynamic loads is determined independently by factors other than structural mass or stiffness. The jet impingement, compartment pressure, and pipe restraint reaction loads resulting from pipe rupture are impulsive, while loads applied by whipping pipes are impactive. In situations where the applied force-time function is known, structural response can be computed accurately using time-history analysis techniques. This is, in fact, the case for all impulsive loads, and for certain impactive load cases. It is also possible to obtain simplified conservative solutions for many cases of practical interest. The analysis for impactive loads can be made using energy and momentum balance methods. The solution for impulsive loads can also be obtained using energy methods, or by equivalent static analysis using dynamic load factors.

3.6.3.4.2 Energy Balance Methods

Solution for structural response by energy methods is predicated on the equality:

Work Done on System = Energy Absorbed by System

The energy is absorbed as strain energy by the structure, and is equal to the area under the resistance-displacement curve (Figure 3.6-49) for the structure under load, or

$$E_S = \int_0^{X_m} R(X)dx \qquad (3.6\text{-}26)$$

where

E_s = strain energy absorbed

$R(X)$ = resistance-displacement function at point of loading

dx = deflection

X_m = maximum deflection under load

If the assumption of elastic-perfectly plastic-material properties are made, the energy absorbed is given by (see Figure 3.6-49)

$$E_s = R_e\left(X_m - \frac{1}{2}X_e\right)$$

where

R_e = resistance at yield

X_e = deflection at yield

For an elastic-plastic structure subject to initial loads, the energy absorbed is given by (see Figure 3.6-49)

$$E_s = [R_e - R_o]\left[x_m - \frac{x_e + x_o}{2}\right] \quad (3.6\text{-}27)$$

where

R_o = equivalent resistance required for initial loads

x_o = displacement associated with R_o

3.6.3.4.3 Evaluation of Resistance-Displacement Functions

The evaluation of structure-resistance-displacement functions can be carried out using standard limit analyses techniques for most cases of practical interest. Acceptable methods for determination of resistance-displacement functions are demonstrated in two commonly encountered examples, as follows:

a. Point load on fixed-fixed beam - The resistance load at full yield for a fixed-fixed beam loaded at the center (Figure 3.6-50) can be determined using the principle of virtual work (References 53 and 54).

$$R_e = \frac{8M_P}{L} \quad (3.6\text{-}28)$$

where

M_P = maximum section strength

L = length of beam

The deflection at yield X_e is given by

$$X_e = \frac{R_e L^3}{192EI} \quad (3.6\text{-}29)$$

where

E = Young's modulus

I = section moment of inertia

For a steel beam, M_p is given by (see Reference 22)

$$M_p = F_y z \tag{3.6-30}$$

where

F_y = yield stress

z = plastic section modulus

For a concrete beam, M_p is given by (see Reference 55)

$$M_p = 0.9\left[(A_S - A'_s)F_y\left(d - {}^{a}/_{2}\right) + A'_s F_y (d - d')\right] \tag{3.6-31}$$

where

A_s = tensile steel reinforcing area

A'_s = compressive steel reinforcing area

F_y = steel yield stress (may be increased by dynamic increase factors)

d - d' = distance between tensile and compressive reinforcing

$a = \frac{(p-p')F_y d}{0.85 f'_c}$

The above formula is predicated on the assumption that the beam is under-reinforced, or

$$(p - p') \leq 0.75 p_b \tag{3.6-32}$$

where

p = ratio of tensile steel

p' = ratio of compressive steel

p_b = balanced steel ratio

Computation of deflection for a concrete beam shall be based on the average moment of inertia for the cracked and uncracked sections, which may be approximated by (Reference 31)

$$I_a = \frac{bd^3}{2}(5.5p + 0.083) \tag{3.6-33}$$

where

b = width of beam

d = effective depth

All information necessary to quantify resistance- displacement curves for the fixed-fixed beam shown in Figure 3.6-50 is now present. Although the resistance of the beam in this example was governed by bending capability of the section, it should be noted that bending/shear interaction may substantially influence resistance of other systems, particularly where loads are applied near supports. Methods that may be used for other beam configurations and for frames are described in References 22, 53, and 54.

b. Point load on concrete slab - The resistance- displacement functions for concrete slabs may be found using yield line theory. As an example, typical yield line patterns for rectangular slabs under point loads are shown in Figure 3.6-51 (Reference 55). The yield load for a complete circular fan-type failure in an isotropically reinforced rectangular slab with equal tensile and compressive reinforcing steel is given by Reference 56.

$$R_e = 4\pi M_p \tag{3.6-34}$$

and the deflection at yield is given by

$$X_e = \frac{\alpha R_e a^2}{EI_a}(1 - \gamma^2) \tag{3.6-35}$$

The term α is dependent on the slab length-to-width ratio and may be obtained from Reference 57. Concrete slabs should also be checked for punching shear failure (see Reference 56) particularly where loads are applied close to edge supports.

3.6.3.4.4 Time-History Analysis Methods

The preceding section has described simplified analysis methods in which the energy absorption capability of the affected structure or structural component is compared to the initial kinetic energy of an impacting body or to the work done by an impulsive force. Application of the simplified methods generally requires use of conservative assumptions concerning the nature of the motive force and the strength of the structure. If necessary, the degree of conservatism can be reduced by use of a more accurate time-history analysis solution. Available time-history solutions range from simple single degree of freedom (first mode) approximations to highly detailed elasto-plastic finite element models. Nearly all time-history methods compute response to a specified force-time function, although a few solutions are available for specified initial structural velocities. Determination of forcing functions for impulsive loads is discussed in Subsection 3.6.3.4.7 and for impactive loads in Subsection 3.6.3.4.8.

3.6.3.4.5 Single Degree of Freedom Solutions

Methods are presented in Reference 31 for time-history analysis of single degree of freedom systems. Figures 2.7 through 2.9 of Reference 31 may be used to determine peak response to applied rectangular pulse, triangular pulse and ramp forcing functions for elastic systems. Where Figures 2.7 through 2.9 of Reference 31 are used to compute peak response, such response remained within elastic limits for the materials. Figures 2.23 through 2.26 of Reference 31 may be used to determine peak response to applied rectangular pulse, triangular pulse, and ramp forcing functions for elasto-plastic systems. Where Figures 2.23 through 2.26 of Reference 31 are used to compute peak response, the assumed resistance-displacement function was computed in accordance with methods described for the evaluation of resistance-displacement functions. It should be noted that the validity of the one degree of freedom response curves in Reference 31 is predicated on the assumption that dominant response occurs in the structure fundamental mode. The validity of this assumption was verified by application of the curves using the second structure mode. If peak response in the second mode exceeds 10 percent of that in the fundamental mode, a more detailed representation of the structure was used, as described below.

3.6.3.4.6 Numerical Methods of Structural Analysis

For structural problems in which the assumptions required to perform simplified analysis are excessively inaccurate or conservative, more general techniques are available in the form of automated discretization techniques. The two most common discretization techniques are finite element, wherein the structural continuum is modeled as an assemblage of discrete regions, and finite difference wherein the differential equations governing structural behavior are satisfied at discrete points. In either case, the result of the discretization process is a system of equations, generally of a size well beyond the scope of hand computations.

SAP (Structural Analysis Program) (Reference 49), a finite element computer program, was used to perform linear elastic dynamic analysis of complex structures and structural components.

3.6.3.4.7 Analysis for Impulsive Loads

As indicated previously, the analysis for impulsive loads can be carried out using energy balance techniques. As an example, the work done by an instantaneously applied constant magnitude impulsive force F in displacing a structure from rest to a maximum displacement X_m can be equated to the energy absorbed by the structure.

$$FX_m = R_e\left(\frac{X_e}{2} + (X_m - X_e)\right) \quad (3.6\text{-}36)$$

The structure does not fail if the maximum displacement X_m is less than the ultimate displacement, or

$$X_m \leq \mu X_e \quad (3.6\text{-}37)$$

By substituting and rearranging the two equations, we obtain the minimum required structural resistance as

$$R_e \geq \frac{F\mu}{\mu - {}^1\!/_2} \quad (3.6\text{-}38)$$

where

R_e = resistance at yield

F = applied force

μ = allowable ductility ratio

It should be noted that this solution is always conservative in that it neglects both decrease in response due to finite rise time of the impulsive force, and the strain-hardening resistance of the structure. A more definitive solution may be obtained using the time-history analysis methods. Since the rise time of most impulsive loads (jet impingement and compartment pressure) resulting from pipe rupture substantially exceeds the fundamental period of the target structures, a one degree of freedom analysis is normally sufficient. Where time-history methods are used to compute response to impulsive loads, acceptability of maximum response will be governed by the ductility ratios.

3.6.3.4.8 Analysis for Impactive Loads

Overall structural response to impactive loads, such as whipping pipes, is dependent on the initial kinetic energy of the impacting body and the inertial (mass) and stiffness characteristics of the impacting body and target structure. It is appropriate to categorize impact problems in terms of the relative "hardness" (stiffness and inertial resistance) of the impacting body and target structure. Where the target structure is harder than the impacting body, the loading applied to the structure will be determined by the collapse of the impacting body. Where the impacting body is harder than the target structure, the loading on the target structure will be determined by the course of embedment of the body into the structure. Both of these cases approach what is termed plastic impact in mechanics. However, if the hardness of the impacting body and target structure is nearly equal, an elastic impact occurs. The solution for pipe impact problems may be obtained by energy/momentum balance methods, by time-history analysis methods, or by a combination of these two methods.

3.6.3.4.8.1 Analysis Using Energy and Momentum Balance

The analysis using energy and momentum balance is based on equating energy imparted to the target structure after impact to the maximum resultant strain energy. Using conservation of momentum to determine target velocity after impact we obtain

$$V_t = \frac{V_s m(1+e)}{M+m} \tag{3.6-39}$$

where

V_t = target velocity after impact

m = effective mass of striking body (pipe)

M = effective mass of target structure

e = coefficient of restitution

V_s = velocity of striking body

Knowing target velocity after impact, the kinetic energy of the target after impact E_t is given by

$$E_t = \frac{MV_t^2}{2} \tag{3.6-40}$$

Solution for maximum response is then found by equating the initial kinetic energy plus the work done by external forces to the strain energy at maximum displacement. For an elastic-perfectly-plastic system subject to impact, and an instantaneously applied constant magnitude force F, this equation is

$$\frac{MV_t^2}{2} + FX_m = R_e\left(\frac{X_e}{2} + (X_m - X_e)\right) \tag{3.6-41}$$

The structure does not fail if the maximum deflection X_m obtained from the equation above is less than the ultimate deflection, or

$$X_m \leq \mu X_e \tag{3.6-42}$$

Alternatively, the structure survives if energy absorption required is less than the structure's energy absorption capability, that is,

$$\frac{MV_t^2}{2} + F\mu X_e \leq \left(R_e X_e \left(\mu - \frac{1}{2}\right)\right) \tag{3.6-43}$$

The acceptability of pipe whip impact can be conservatively evaluated for all cases based on the above equations and the following conservative assumptions:

a. Impact is elastic (e = 1)

b. The effective mass of the whipping pipe equals one-third of the mass between adjacent hinge(s) making up the pipe whip mechanism (Reference 59) for sections of pipe impacting side-on, and full mass for sections of pipe impacting end-on

c. The velocity of the pipe at impact is taken either from the piping dynamic analysis or determined using kinematic relationships

d. The effective mass of the target corresponds to that of a circular plug through the target thickness with diameter equal to pipe diameter plus target thickness (Reference 60). If the target is a beam, plug width may not exceed beam width

e. The resistance-displacement function for the target structure is computed using the energy balance methods

f. Acceptability of impact is governed by the allowable ductility factor in Table 3.6-2.

The simplified method is conservative both in assuming elastic impact (ignoring energy absorbed in local plastic deformation of the pipe and target structure on impact), and in assuming a lower limit target effective mass. A more definitive analysis is obtained by using more complex time-history analysis methods, as described in the following section.

3.6.3.4.8.2 Combined Time-History and Energy Balance Methods

In these methods, a time-history forcing function characterizing impact is established based on local deformation of the impacting body or target structure during impact. By taking into account local deformation during impact, the conservatism noted in the preceding section in assuming fully elastic impact is removed. By applying the computed forcing function in a structure dynamic response analysis, a realistic value of structure effective mass can be determined, based on the failure mechanism determined for the structure.

The case of a "hard" body impacting a relatively "soft" structure has been treated in Reference 61. This case would correspond, for example, to a heavy-walled pipe striking a thin shell.

The case of a relatively "soft" body striking a "hard" structure would correspond to a whipping pipe striking a massive concrete structure. The formulation for this analysis was derived from methods presented for evaluation of response to aircraft impact in Reference 62 and 63, and is presented below.

The force applied during impact includes a component due to blowdown thrust and an impulse component. The blowdown thrust component is simply that calculated in the thermal-hydraulic analysis previously described. The impulse component is that portion of the wall reaction that removes the kinetic energy from the pipe. It can be calculated by considering the change in momentum as the pipe crushes from length L + ΔX to length L, as shown in Figure 3.6-52.

Impulse = Change in momentum

$$F\Delta t = M_p(V - \Delta V) - \left(M_p V + \mu_S \Delta X V\right) \quad (3.6\text{-}44)$$

$$F\Delta t = -M_p \Delta V - \mu_S \Delta X V \quad (3.6\text{-}45)$$

$$F = -M_p \frac{\Delta V}{\Delta t} - \mu_S \frac{\Delta X}{\Delta t} V \quad (3.6\text{-}46)$$

$$F = -M_p a - \mu_S V^2 \quad (3.6\text{-}47)$$

Taking a force balance on the uncrushed portion of pipe

$$M_p a = K_p = \text{Crushing strength of pipe} \quad (3.6\text{-}48)$$

Then the force on the wall, F_W is

$$F_W = -F = K_p + \mu_S V^2 \quad (3.6\text{-}49)$$

where

F = $F(t)$ = impulse reaction applied to structure after impact

K_P = $K_P(x)$ = crushing strength of pipe

μ_S = $\mu_S(x)$ = mass of pipe stopped per unit of deflection

X = $X(t)$ = total distance crushed

t = time

V = velocity of uncrushed portion of pipe

M_P = mass of uncrushed portion of pipe

a = acceleration of uncrushed portion of pipe

This equation can be conservatively evaluated to find the impact force time-history based on the following assumptions:

a. The impulse reaction is applied to a target structure area with a maximum dimension not exceeding pipe diameter

b. Pipe crushing strength is based on local collapse of the pipe walls up to the point where the pipe is fully "flattened." Pipe crushing strength after "flattening" is limited to the lesser of piping or target ultimate compressive stress

c. The mass of pipe stopped can be calculated using geometric considerations up to the point where the pipe is fully "flattened." The mass of the pipe stopped after "flattening" is equal to one-third of the mass between adjacent plastic hinges for sections of pipe impacting side-on and full mass for sections of pipe impacting end-on

d. The impulse force drops to zero when the integral impulse force by time applied is equal to the initial momentum of the impacting pipe.

A typical force-time history after impact, generated in the above fashion, is shown in Figure 3.6-53. The force is equal to the blowdown force at first contact between the pipe and structure, and begins to increase thereafter as the pipe crushes. Once the pipe is fully crushed, the force rises to the limit of pipe or wall compressive strength. Once the pipe momentum is exhausted, the force drops again to the level of the blowdown thrust.

The force-time history thus determined is then applied in a time-history response analysis previously described. The time-history analysis is carried out up to formation of a plastic collapse mechanism in the target structure (up to limit load). Acceptability of the target structure response is determined using an energy balance as follows.

Work done on structure + kinetic energy of structure

= Maximum strain energy of structure

$$\int_{X_e}^{X_m} F dz + {}^{1}/_{2}\, MV^2 = R_e(X_m - X_e) \qquad (3.6\text{-}50)$$

where

F = applied force (see Figure 3.6-53)

Z = deflection

X_e = deflection at onset of collapse (from time-history analysis)

X_m = maximum deflection

M = effective mass of target structure (see Table 5-l from Reference 59, and Table 3.6-8)

V = velocity of target structure at onset of collapse (from time-history analysis)

R_e = limit resistance of structure

As long as F falls below R_e, the pipe kinetic energy will be reduced during impact with the structure by conversion to strain energy. Acceptability is again governed by

$$X_m \leq \mu X_e \qquad (3.6\text{-}51)$$

where μ is taken from Table 3.6-2.

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

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TABLE 3.6-1 SYSTEMS AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN

Component or System

General Requirements

- Primary containment
- Pressure relief device
  - Safety/relief valves
  - Pressure suppression pool (passive)
- Main control room complex and control room air conditioning including intake radiation monitoring equipment
- Electrical power
  - Offsite power
  - Standby ac power
  - Emergency dc power
- Scram protection (reactor protection system)
- Control rod drive system (portion required for scam)
  - a. Turbine control valve fast signal, or
  - b. Reactor low water level signal, etc.[a]
- Core cooling
- Incident detection circuitry (start ECCS)
- RHR torus cooling mode (one loop)
- RHR service water to available RHR heat exchanger
- Core water to:
  - Diesel generator jacket cooling
  - RBCCW or EECW available to RHR pump motors
  - RBCCW or EECW available to RHR room coolers
- Instrumentation
  - Reactor water level indication
  - Temperature indication
- Control air system (noninterruptible portion – 1 division)

TABLE 3.6-1 SYSTEMS AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN

Specific Requirements

For Main Steam Line Break

Flow restrictors (passive)

Isolation system control; incident detection circuitry

a. High temperature in main steam line tunnel

b. High steam line flow, or

c. Reactor low water level

Main steam line isolation valves

Feedwater check valves

Core cooling

a. HPCI

b. RHR plus remote-operated SRVs

Equipment cooling water (RBCCW or EECW) to room coolers

For Feedwater Line Break

Feedwater check valves

Isolation system control: incident detection circuitry

a. High temperature in main steam line tunnel, or

b. Reactor low water level

Main steam line isolation valves

Core cooling

a. HPCI

b. RHR

Equipment cooling water (RBCCW or EECW) to RHR room coolers

For High Pressure Coolant Injection System (HPCI) Steam Line Break

Isolation system control; incident detection circuitry

a. High temperature in HPCI steam line chase, or

b. High HPCI steam line flow, or

c. HPCI turbine steam line low pressure

HPCI isolation valves

Core cooling

a. RHR plus remote-operated SRVs

TABLE 3.6-1 SYSTEMS AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN

RBCCW or EECW to room coolers

For Reactor Core Isolation Cooling System (RCIC)

Steam Line Break

Isolation systems control; incident detection circuitry

a. High temperature in RCIC steam line chase, or

b. High RCIC steam line flow, or

c. RCIC turbine steam line low pressure

RCIC isolation valves

Core cooling

a. HPCI plus remote-operated relief valves

b. RHR

RBCCW or EECW to HPCI room coolers

For Reactor Water Cleanup System (RWCU) Line Break

Isolation system control; incident detection circuitry

a. Flow imbalance, or

b. Low reactor water level, or

c. High temperature in RWCU pipe chase

RWCU isolation valves

Core cooling

a. HPCI and remote-operated relief valves, or

b. RHR plus remote-operated relief valves

RBCCW or EECW to HPCI room cooler

RBCCW or EECW to RCIC room cooler

---

[a] RPS trip signals resulting from loss of coolant:
1. Reactor vessel low water level
2. Main steam isolation valve closure
3. Primary containment (drywell) high pressure.

TABLE 3.6-2 MAXIMUM DUCTILITY FACTORS

1. Tension reinforced concrete beams and slabs (flexure controls design)
   $\mu = \frac{0.05}{p}$; $\mu \leq 12.5$

2. Doubly reinforced concrete beams and slabs (flexure controls design)
   $\mu = \frac{0.05}{p - p'}$; $\mu \leq 15.0$

3. Concrete beams and slabs in region requiring shear reinforcement
   a. Shear carried by concrete and stirrups $\mu = 1.3$
   b. Shear carried completely by stirrups $\mu = 3.0$

4. Concrete columns $\mu = 1.3$

5. Structural steel tension members $\mu = 0.5 \frac{\varepsilon_u}{\varepsilon_y}$

6. Structural steel flexural members
   a. Open sections (I, WF, T, etc.) $\mu \leq 12.5$
   b. Closed sections (pipe, box, etc.) $\mu \leq 25.0$
   c. Members where shear governs design $\mu \leq 6.0$

7. Structural steel columns
   $\mu \leq 1.0$ for $\ell/r < 30$
   $\mu \leq 3.0$ for $30 \leq \ell/r \leq 60$
   $\mu \leq 6.0$ for $\ell/r > 60$

Notes

A_s = Area of tension reinforcement
A'_s = Area of compressive reinforcement
b = Width of section
d = Depth of section to reinforcement
p = Percentage tensile reinforcement
p' = Percentage compression reinforcement
$\mathcal{E}u$ = Uniform ultimate strain of material
$\mathcal{E}y$ = Strain at yield of material
ℓ = Effective length of column
r = Radius of gyration
(See AISC-69 Specifications)

TABLE 3.6-3 DYNAMIC INCREASE FACTORS (DIF)

I.	Reinforced or Prestressed Concrete	DIF
	Concrete	
	Compression	1.25
	Diagonal tension and direct shear (punch out)	1.0
	Bond	1.0
	Reinforcing Steel	
	Tension	1.2
	Compression	1.2
	Diagonal tension and direct shear (stirrups)	1.0
II.	Structural Steel	
	Flexure and tension	1.2
	Compression	1.2
	Shear	1.0

TABLE 3.6-4 REASONS FOR EXCLUSIONS

Index No.	Reasons	Explanation
(R1)	Separation	System in separate compartment.
(R2)	Distance	System separated by distance but in the same compartment.
(R3)	Redundancy	System function can be performed by two or more identical units.
(R4)	Back-up	System function can be replaced by the function of a different system.
(R5)	Self-eliminating	Pipe rupture caused damage only to the system itself.
(R6)	Size criteria	Pipe of an equal or larger diameter and equal or heavier wall thickness than the broken pipe is considered not damaged.
(R7)	Low pressure	Pressure inside the pipe is too low to cause a pipe whip.
(R8)	Barrier	System protected by barrier.
(R9)	Testing condition	Pipe line used only at testing condition or emergency condition, etc.
(R10)	Scarcity of usage	Duration of operation of the pipe is less than 2 percent of the duration of reactor operation.
(R11)	Safe area	Pipe routing in area where no system related with safe shutdown is located.
(R12)	Minimum size	Pipe smaller than 4 in. is not required for the analysis of longitudinal pipe break or pipe equal to or less than 1 in. is not required for the analysis of circumferential break.

[a] Index number used for permanent identification of components excluded from further consideration.

TABLE 3.6-5 LOADING COMBINATIONS FOR ELASTIC DESIGN OF STEEL STRUCTURES AND ULTIMATE STRENGTH OF CONCRETE STRUCTURES (STEAM TUNNEL)[a]

Load Combination Number	Overall Loading Equation[b]
	Elastic Design of Steel Structures
1	$1.5\ S = D + L + T_a + R_a + P_a$
2	$1.5\ S = D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + Feqo$
3	$1.5\ S = D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + Feqs$
	Ultimate Strength of Concrete Structures
1	$U_1 = D + L + T_a + R_a + 1.0\ P_a$
2	$U_1 = D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ Feqo$
3	$U_1 = D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ Feqs$

Symbols

D	Dead load of the structure, including any permanent equipment loads
Feqo	Loads generated by operating-basis earthquake
Feqs	Loads generated by the safe-shutdown earthquake
L	Live loads
P_a	Compartmental pressure due to pipe break
R_a	Pipe reaction under thermal conditions due to pipe rupture and including pipe reactions during normal operating conditions
T_a	Thermal loads due to pipe rupture and including thermal loads during normal operating conditions
Y_j	Jet impingement due to pipe rupture
Y_m	Missile effects due to pipe rupture
Y_r	High-energy pipe break reactions
S	Section strength based on elastic design methods and the allowable stresses as described in the AISC
U_1	Section strength based on ultimate strength design methods as described in ACI 318-63

[a] Loads not applicable to a particular system under consideration may be deleted.

[b] Effects for time-dependent loads will be superimposed accordingly.

TABLE 3.6-6 FLOW AND EVENTS POSTULATED FOR FEEDWATER BREAK

Approximate Time (sec)	Flow Rate (gpm)	Event
0		Break
0+	43,000	o Feedwater flow jumps to 43,000 GPM o RFP tripped on low suction pressure
7	43,000	o Reactor water level (L3) trip initiates a SCRAM trip o SCRAM trip initiates closure of feedwater pump discharge valves o Post SCRAM feedwater control automatically put into service
14	43,000	Steam tunnel leak detection system temperature exceeded.
22	43,000	After an 8.0 second instrument channel response time MSIV closure is initiated.
26	43,000	Vessel water level (L2) trip initates HPCI and RCIC operation. RCIC is not taken credit for in the scenario, since it is not environmentally qualified and because the NE corner room is affected by flooding.
32	43,000	After a 10 second MSIV closure time, the MSIVs are fully closed.
56	43,000	After the HPCI initiation signal, HPCI reaches rated flow within 30 to 60 sec. Fifty-six sec corresponds to the assumption of a 30 sec response time.
57	21,800	After an 8-second delay (in addition to the 30-second post scram delay) and a 12-second closure time, fast closure valves V12-2531 and V12-2532 are closed and all flow is forced through the start up level control valve.
64	20,000	The number 5 feedwater heater level control valves are closed and flow through the heater drain pumps is isolated from the break.
117	20,000	The RFP slow closure discharge valves are fully closed (V12-2503, V12-2504). This has no effect on the break flow but is noted to provide assurance that failure of the fast closure valves would not be as severe as failure of the start up control valve. Closure of these valves requires 80 seconds plus a 30 second delay.

TABLE 3.6-6 FLOW AND EVENTS POSTULATED FOR FEEDWATER BREAK

Approximate Time (sec)	Flow Rate (gpm)	Event
279	20,000	Reactor water level is restored by HPCI, which would close the startup control valve had it not been assumed to fail. Two hundred seventy-nine sec corresponds to the assumption of a 30 sec response HPCI time. This time may be up to 30 sec longer, assuming a 60 sec HPCI response time.
447	20,000	Condensate and heater feed pumps trip on low hotwell level and pumped flow is assumed to decrease to zero.
		The water inventory (13725 gal.) downstream of the RFPs is assumed to be discharged from the break over 1 minute period by gravity flow.
507	0	The water inventory in the piping is totally discharged.
	185,734 gallons	

TABLE 3.6-7 FEEDWATER LINE BREAK IN STEAM TUNNEL MAXIMUM FLOOD HEIGHT

Affected Area	Flood Elevation, ft	Flood Depth, in.
Steam Tunnel	587.98	53.7
RBCCW Room	587.31	45.8
NE Corner Room	546.74	80.9
SE Corner Room	554.08	169.0
Torus Room	541.23	14.8
HPCI Room	546.57	78.8

Note: This table lists maximum flood heights for each area, maximum heights do not occur simultaneously for all rooms.

TABLE 3.6-8 NON-SAFETY SYSTEMS NOT INVOLVED IN THE HIGH-ENERGY PIPE ANALYSIS

Rod worth minimizer

Plant process computer

Area radiation monitors

Transient recording and analysis (TRA)

Offgas

Radwaste solidification

Heat-tracing

Fuel-handling equipment

Fuel pool cooling

Maintenance monorails and hoists

Seismic measurement equipment

Turning gear

Generator

Generator hydrogen seal oil

Generator cooling

Generator buses

Generator excitation

Demineralized water

Sampling

Plant heating

Heating and process steam

Security

Communications

Integrated leak-rate test

Cooling tower

Screen wash

Circulating water screens and trash rakes

Hot machine shop

Switchyard

Tornado roof vents

Plant lighting

TABLE 3.6-9 EQUIVALENT MASS FOR COLLAPSED SECTIONS

Member	m_e
Beam or one way slab uniformly distributed load	
Restrained at supports	0.667 m
Simple at supports	0.667 m
Beam or one way slab concentrated load at center	
Restrained at supports	0.333 m
Simple at supports	0.333 m
Rectangular slab (b x a) [a, b] uniformly distributed load	
Restrained at four sides	$\frac{1}{2}\ \Sigma m_{\Delta} + \frac{(4b-3a)}{6b-4a}\ \Sigma m_{\text{▱}}$
Simple at four sides	$\frac{1}{2}\ \Sigma m_{\Delta} + \frac{(4b-3a)}{6b-4a}\ \Sigma m_{\text{▱}}$
Rectangular slab with concentrated load	$\frac{1}{6}\ \Sigma m_{\Delta}$

(a) $\frac{b}{2} \leq a \leq b$

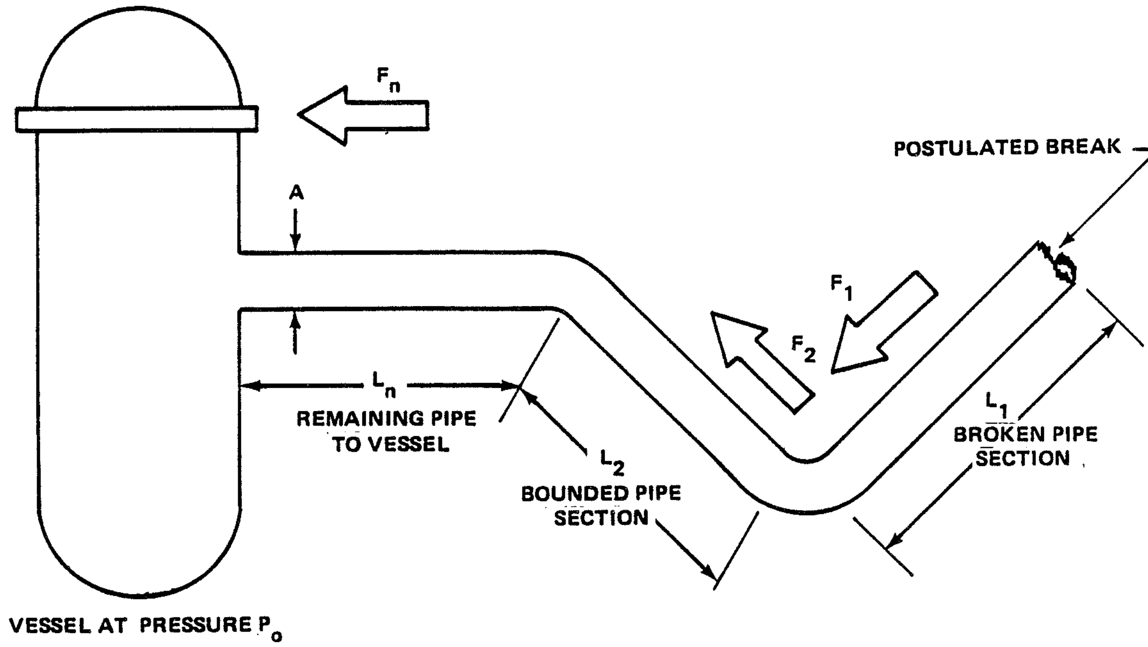
(b) $\frac{b}{2} \geq 1.0$

Notes

M = total mass of beam or slab

m_{Δ} = mass of triangular sections in yield line pattern

$m_{\text{▱}}$ = mass of trapezoidal sections in yield line pattern



A	UNIFORM PIPE FLOW AREA
F_1	BROKEN PIPE REACTION
F_2 — F_n	BOUNDED PIPE REACTIONS

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FIGURE 3.6-1

PRESSURE VESSEL AND BROKEN PIPE CIRCUMFERENTIAL BREAK

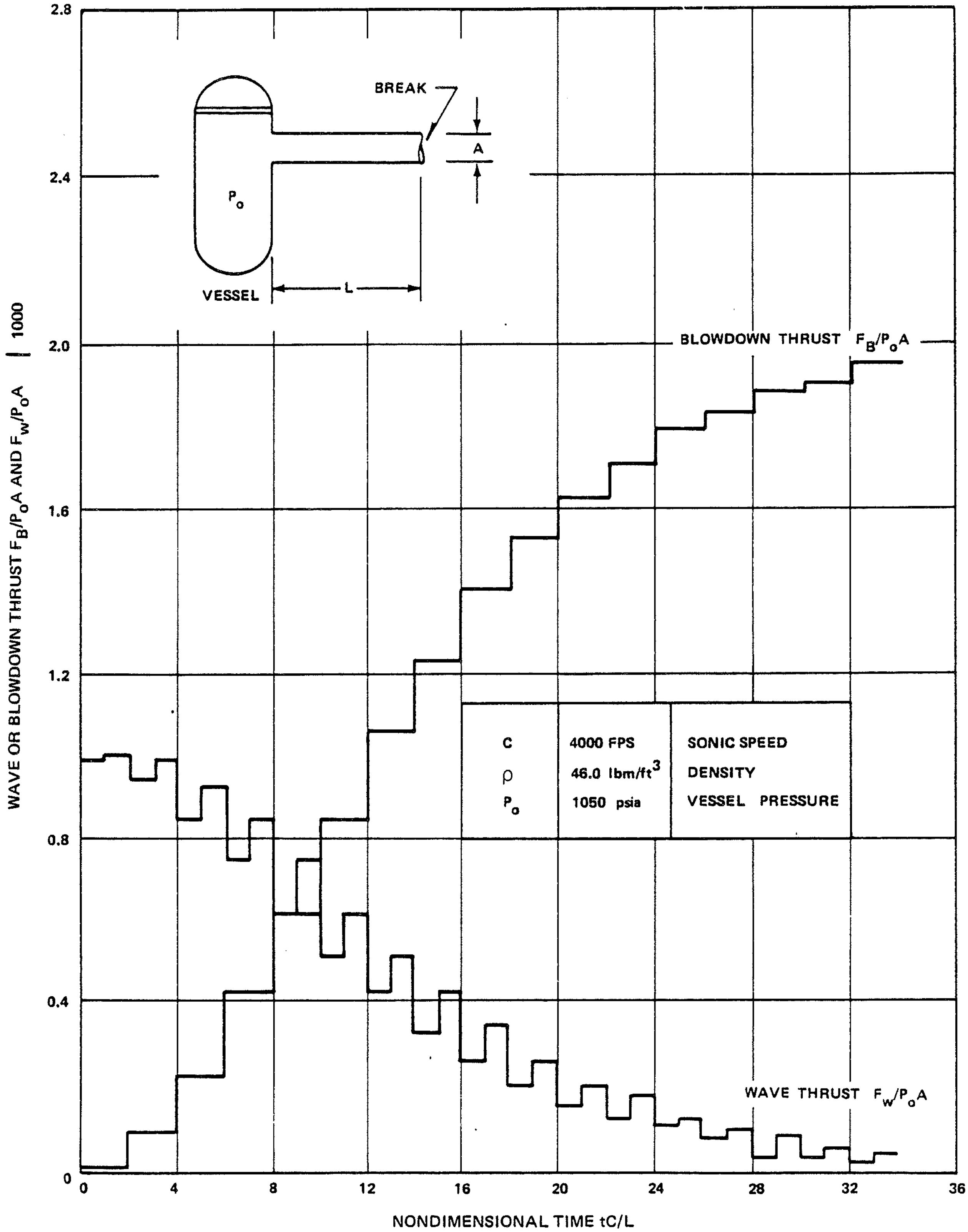

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FIGURE 3.6-2

THRUST COMPONENTS, SINGLE STRAIGHT PIPE, NONFLASHING LIQUID

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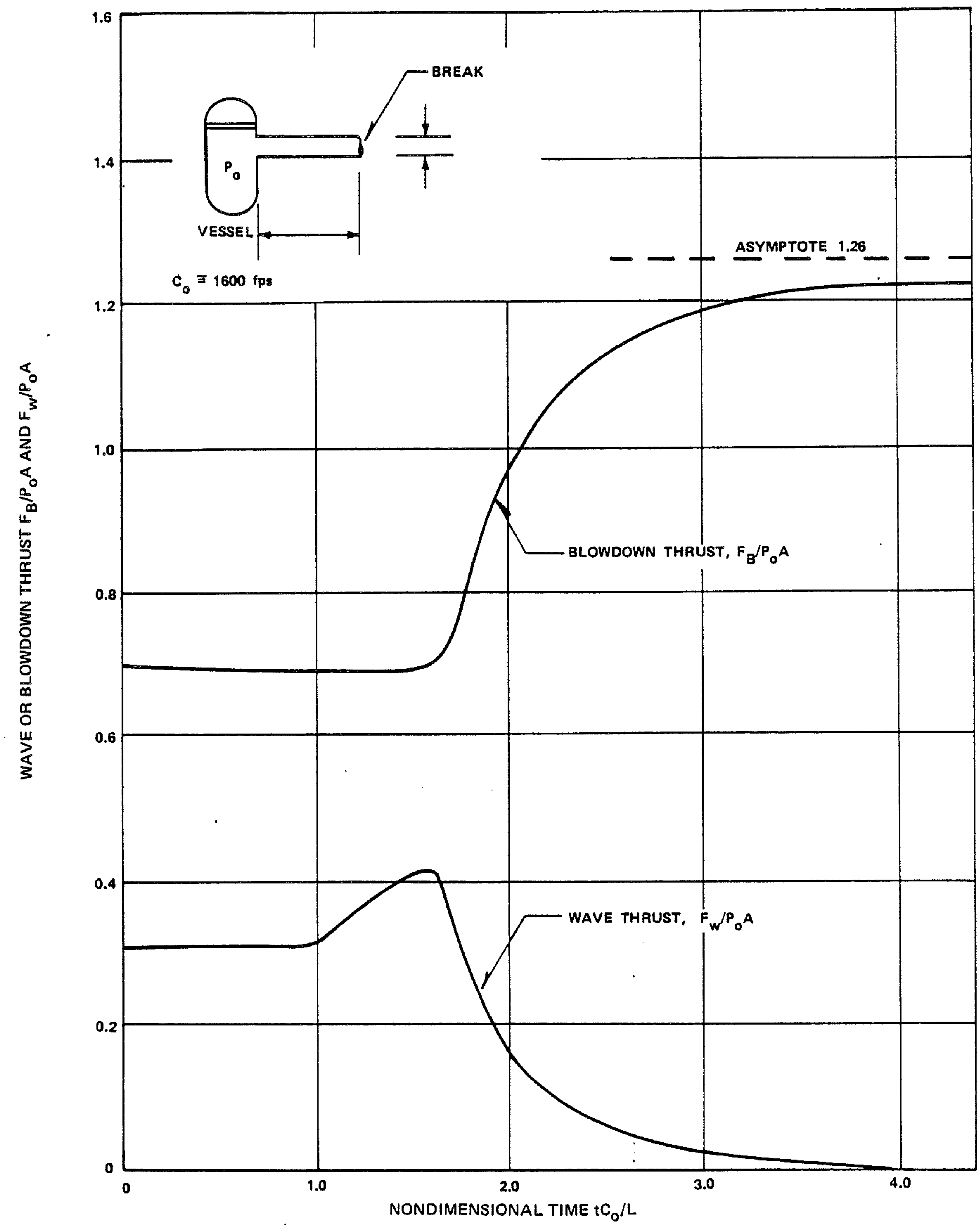

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FIGURE 3.6-3

THRUST COMPONENTS, SINGLE STRAIGHT PIPE, STEAM

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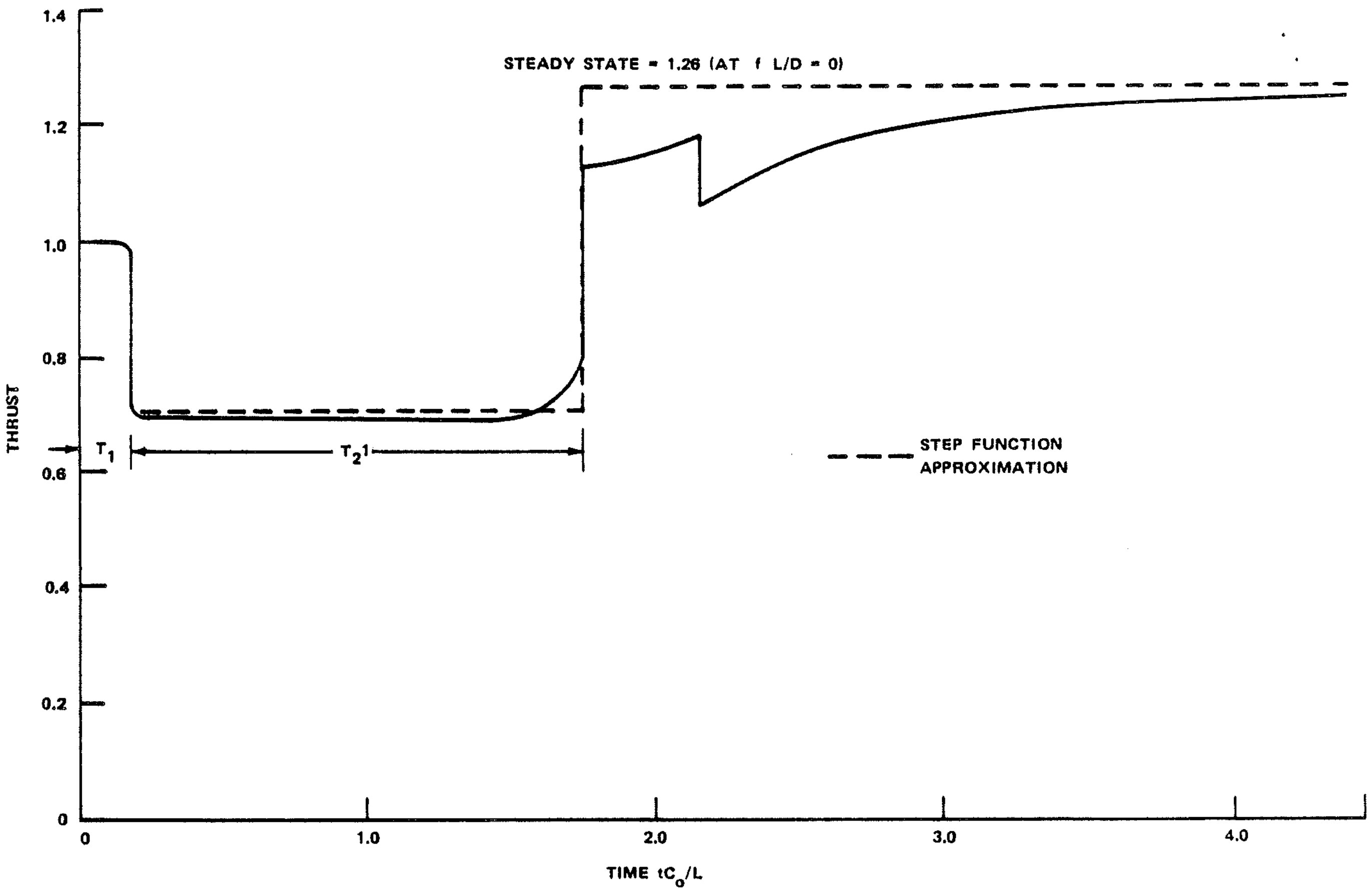

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FIGURE 3.6-4

TIME-THRUST DIAGRAM FOR A LINE CONTAINING STEAM – CIRCUMFERENTIAL BREAK

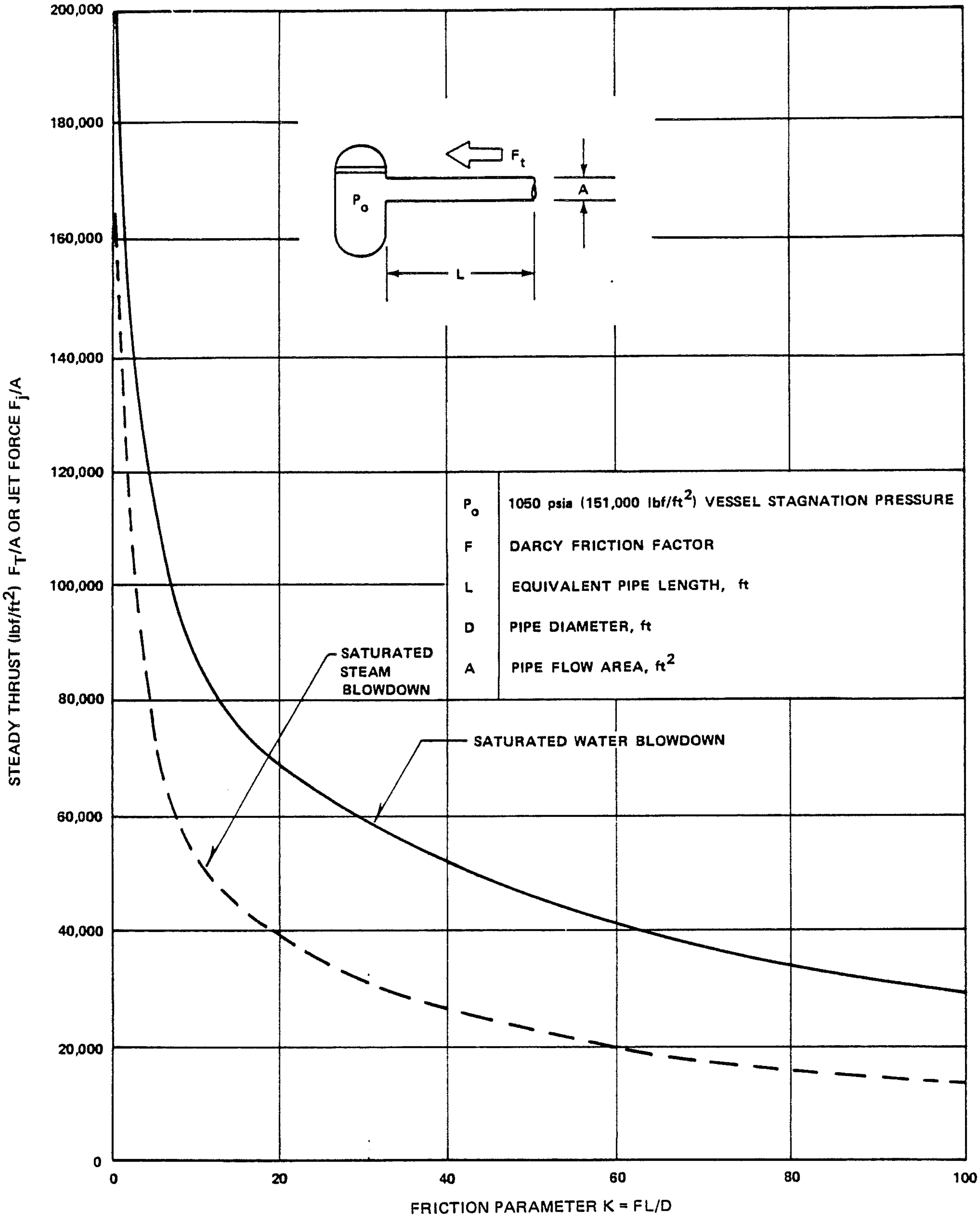

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FIGURE 3.6-5

STEAM-WATER TOTAL STEADY THRUST PER UNIT AREA

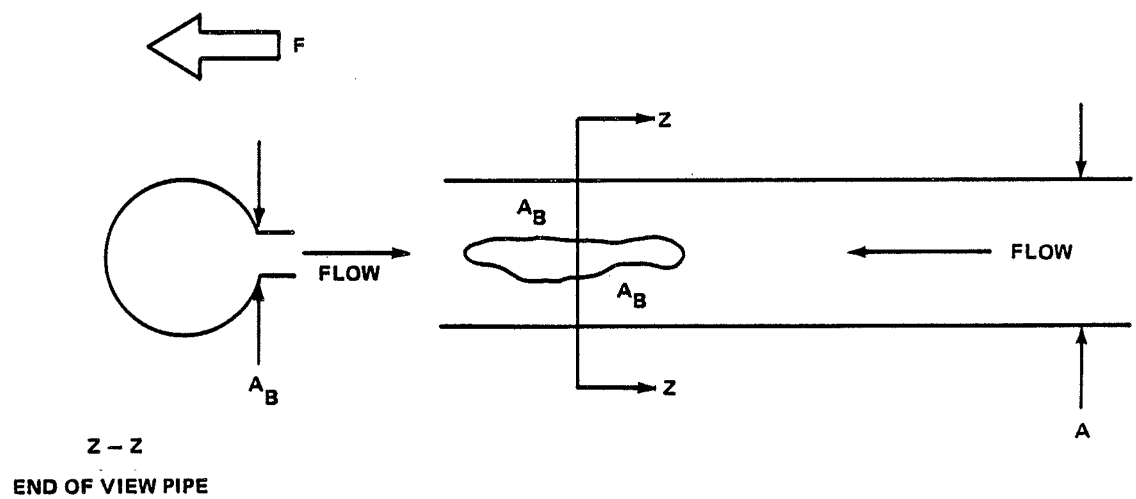

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FIGURE 3.6-6

LONGITUDINAL BREAK

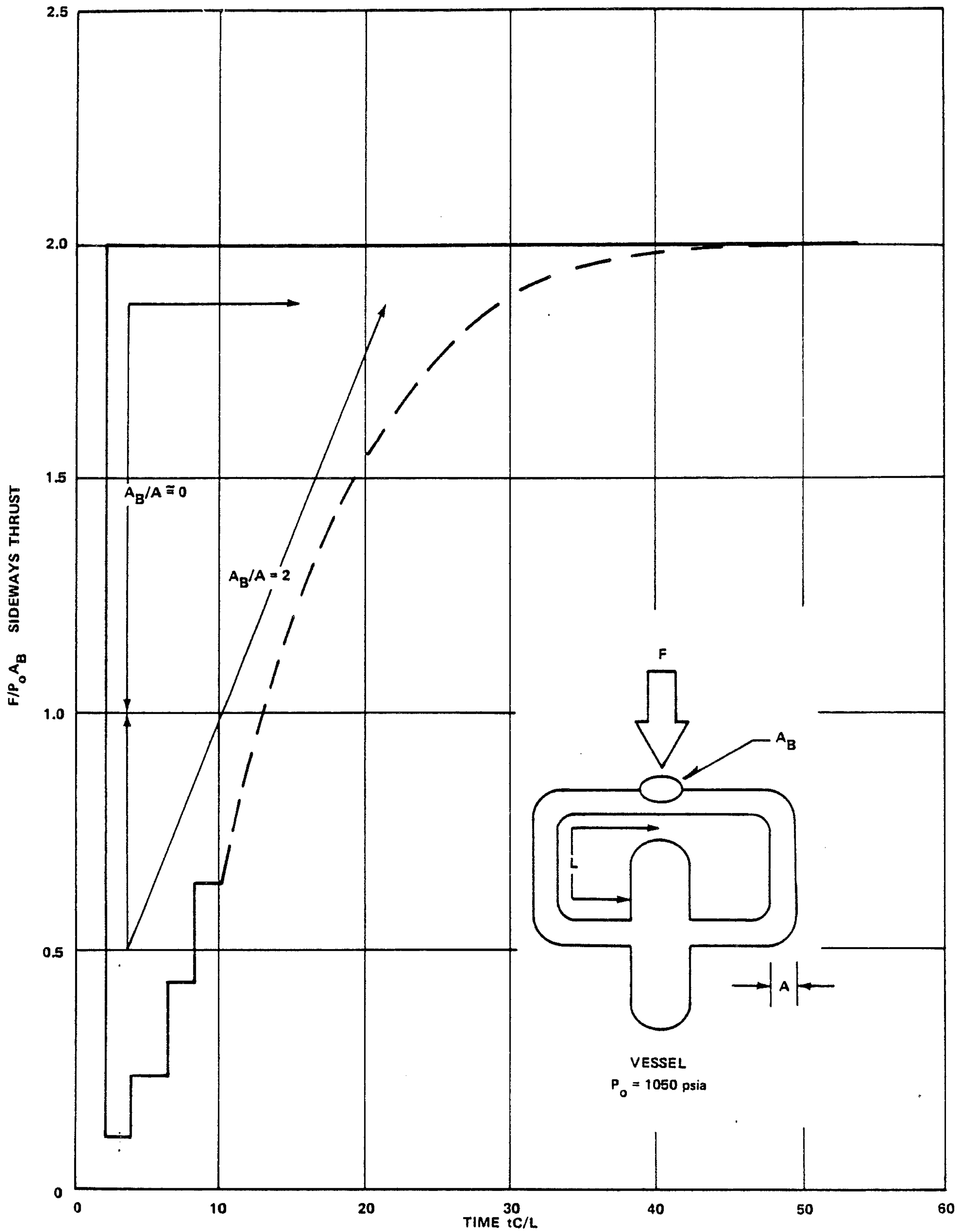

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FIGURE 3.6-7

THRUST FOR LONGITUDINAL BREAK IN WATER PIPE

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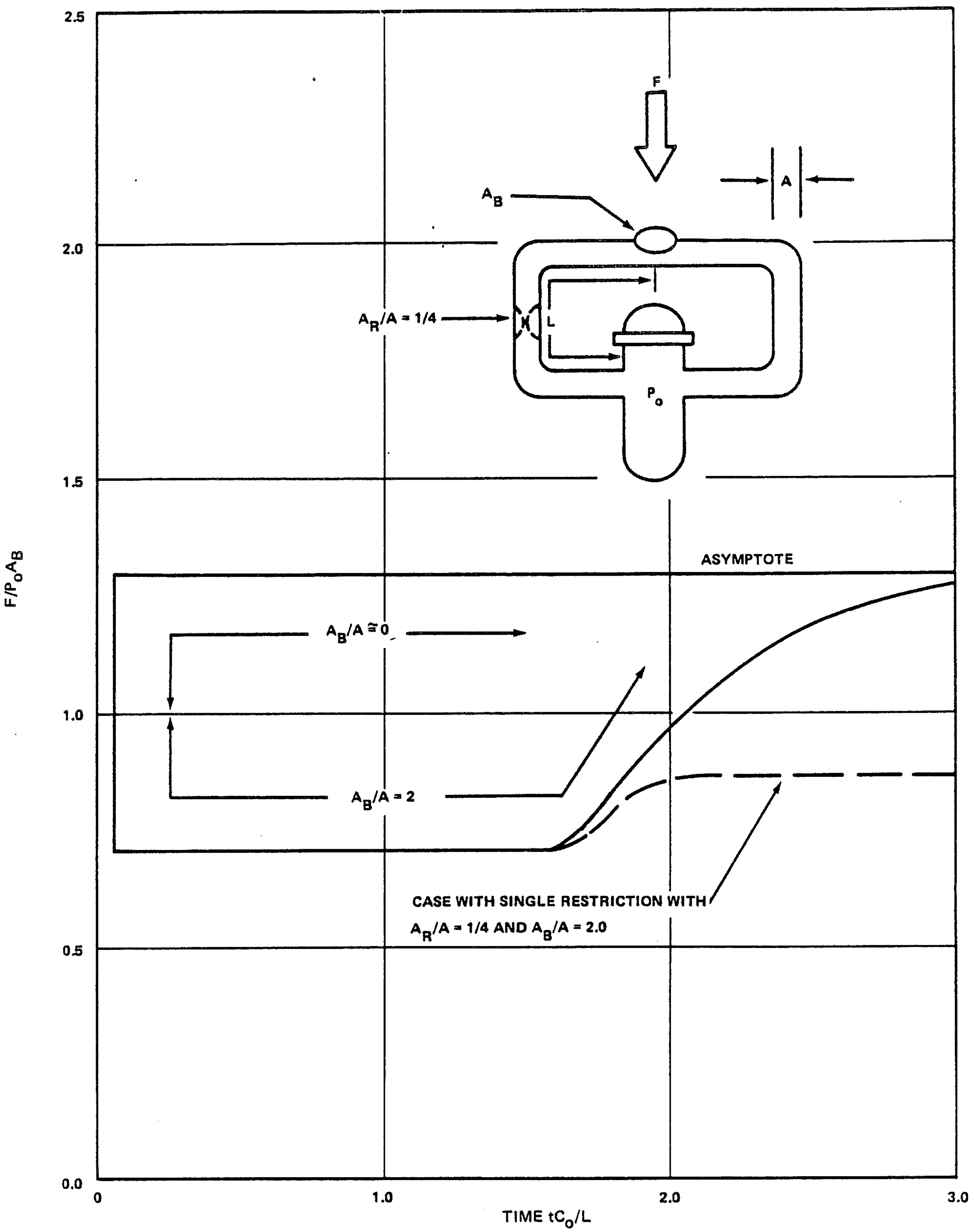

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FIGURE 3.6-8

THRUST FOR LONGITUDINAL BREAK IN STEAM PIPE EXCLUDING EFFECTS OF FRICTION

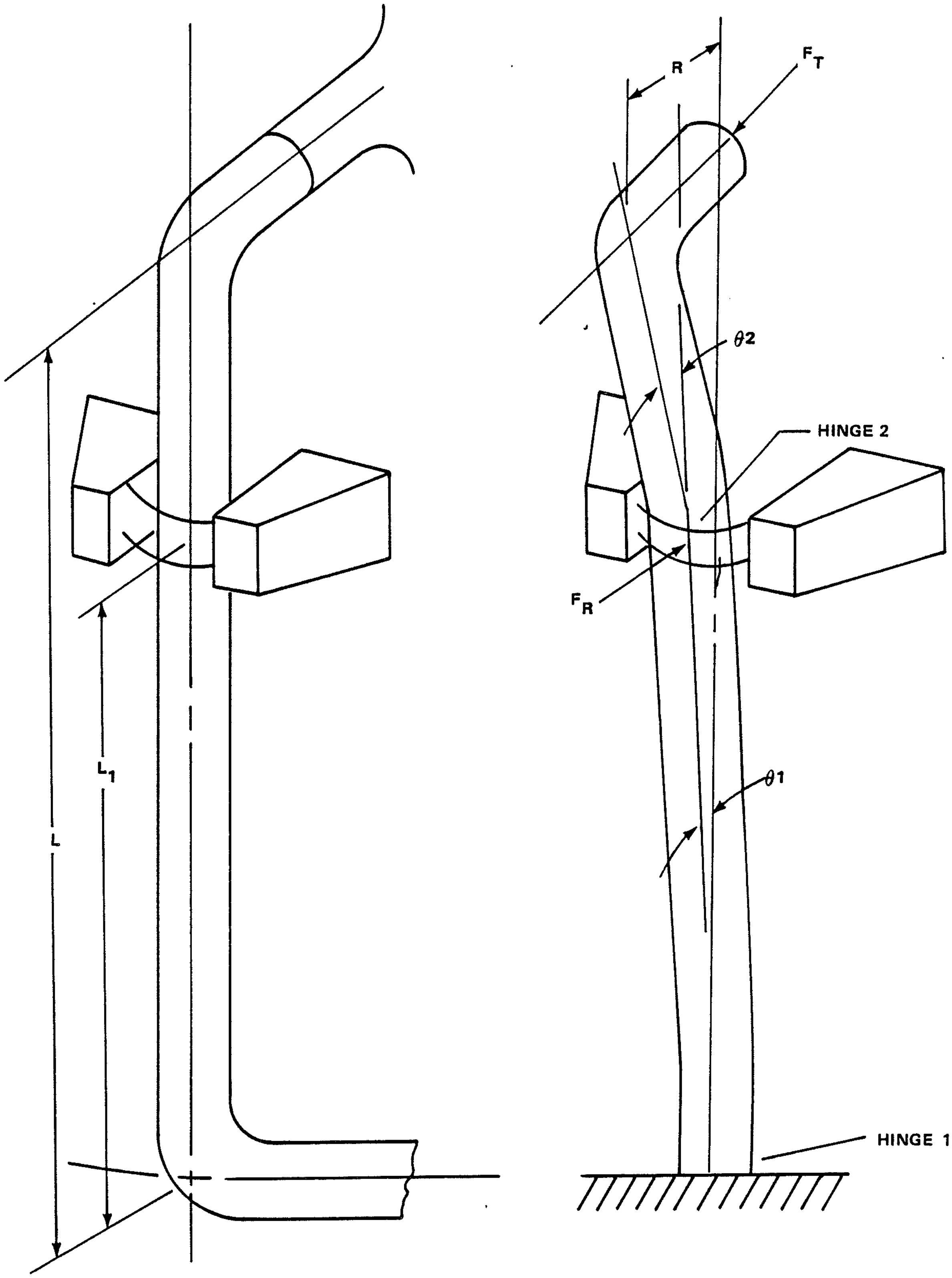

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FIGURE 3.6-9

PIPE/RESTRAINT CIRCUMFERENTIAL BREAK MODEL

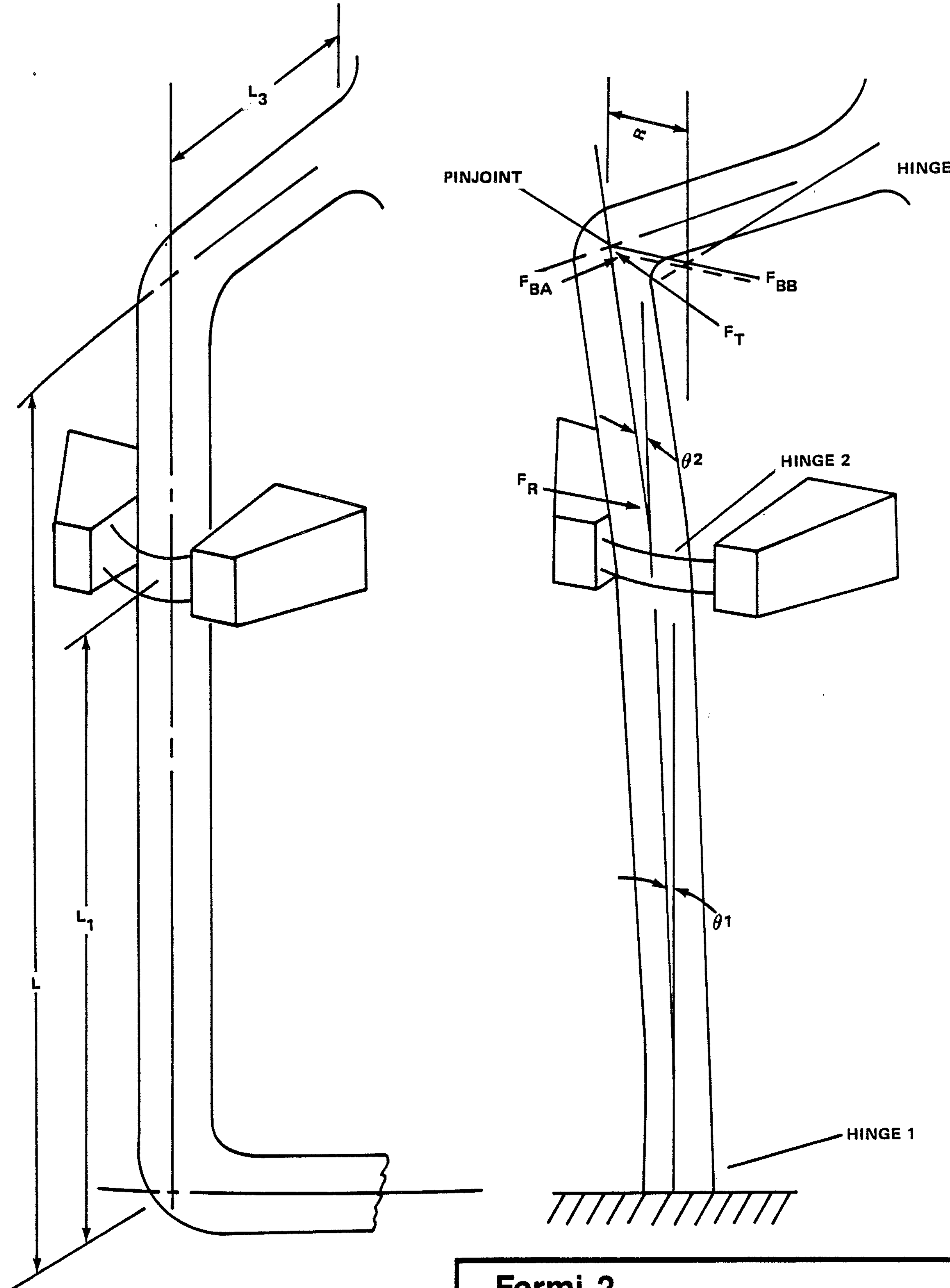

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FIGURE 3.6-10

PIPE/RESTRAINT LONGITUDINAL BREAK MODEL

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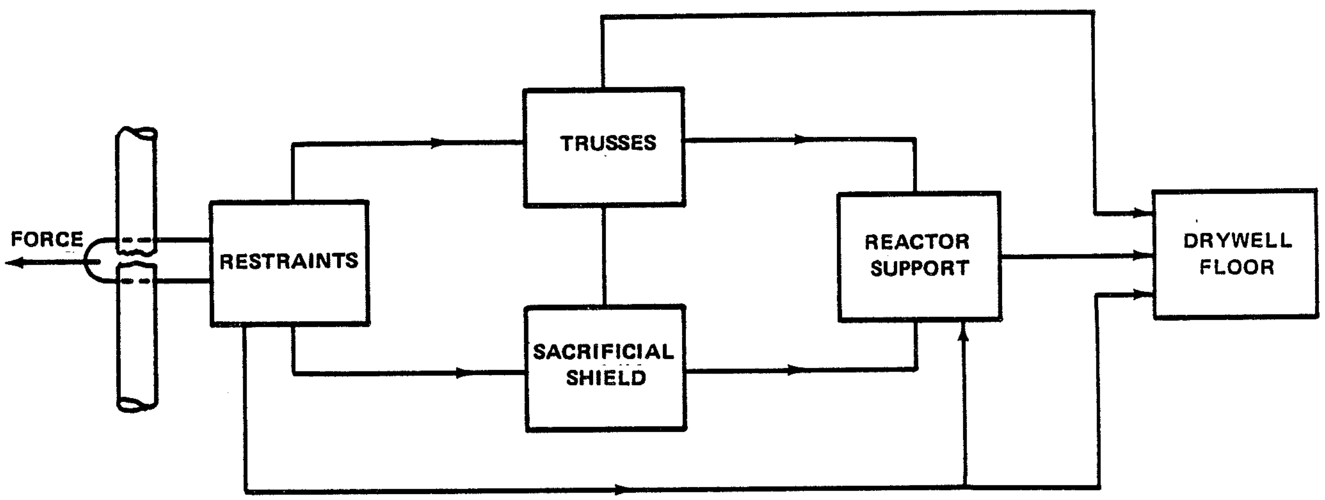

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FIGURE 3.6-11

PIPE WHIP RESTRAINT SUPPORT SYSTEM SCHEMATIC

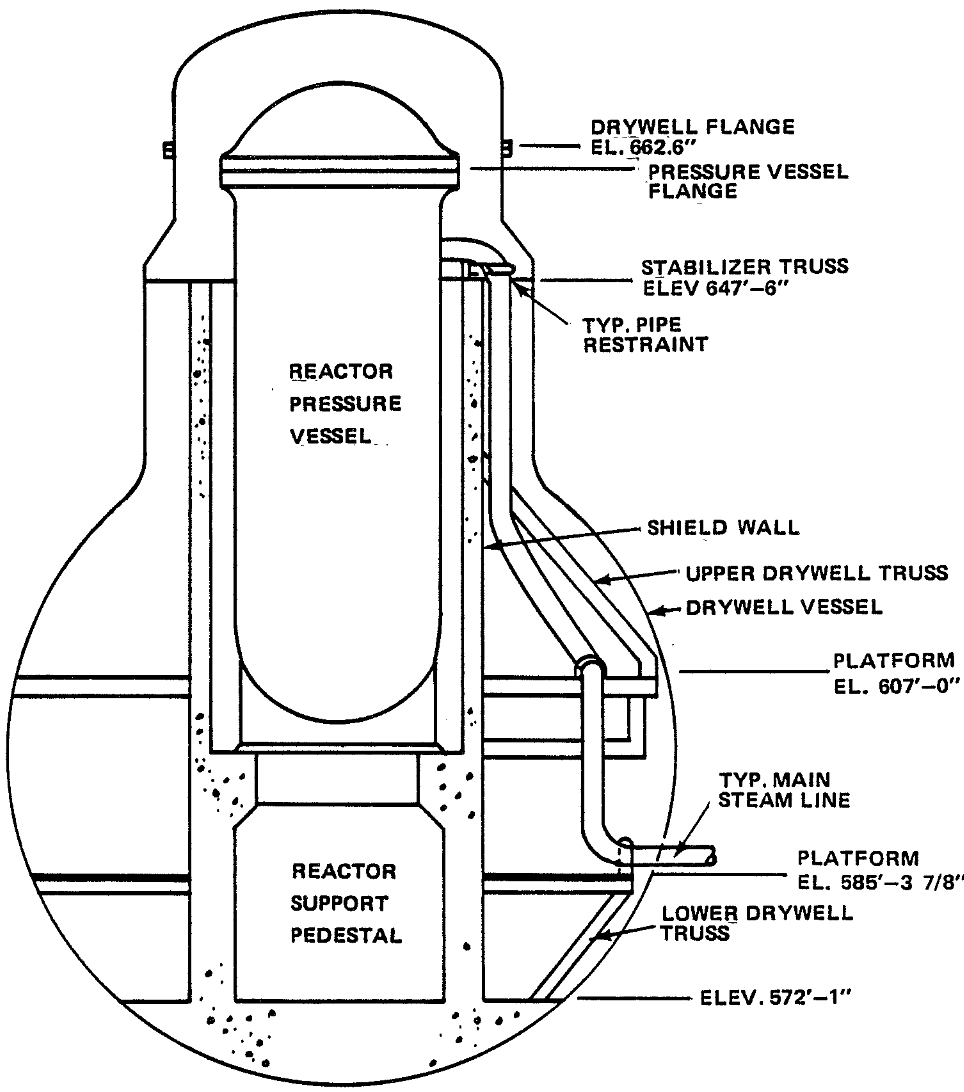

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FIGURE 3.6-12

PIPE WHIP RESTRAINT SUPPORT SYSTEM
DRYWELL SECTION

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Figure Intentionally Removed
Refer to Plant Drawing RX BLDG ROOM LOCATIONS

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FIGURE 3.6-13

SELECTIVE ROOM LOCATIONS
REACTOR BUILDING SUBBASEMENT
ELEVATION 540.0 FT

Figure Intentionally Removed
Refer to Plant Drawing RX BLDG ROOM LOCATIONS

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FIGURE 3.6-14

SELECTIVE ROOM LOCATIONS
REACTOR BUILDING BASEMENT
ELEVATION 565.0 FT

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FIGURE 3.6-15

SELECTIVE ROOM LOCATIONS
REACTOR BUILDING FIRST FLOOR
ELEVATION 583.5 FT

Figure Intentionally Removed
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FIGURE 3.6-16

SELECTIVE ROOM LOCATIONS
REACTOR BUILDING SECOND FLOOR
ELEVATION 613.5 FT

Figure Intentionally Removed
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FIGURE 3.6-17

SELECTIVE ROOM LOCATIONS
REACTOR BUILDING THIRD FLOOR
ELEVATIONS 643.5 FT AND 641.5 FT

Figure Intentionally Removed
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FIGURE 3.6-18

SELECTIVE ROOM LOCATIONS
REACTOR BUILDING FOURTH FLOOR
ELEVATION 659.5 FT

Figure Intentionally Removed
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FIGURE 3.6-19

SELECTIVE ROOM LOCATIONS
REACTOR BUILDING FIFTH FLOOR
ELEVATIONS 677.5 FT AND 684.5 FT

Figure Intentionally Removed
Refer to Plant Drawing STEAM TUNNEL SKETCH

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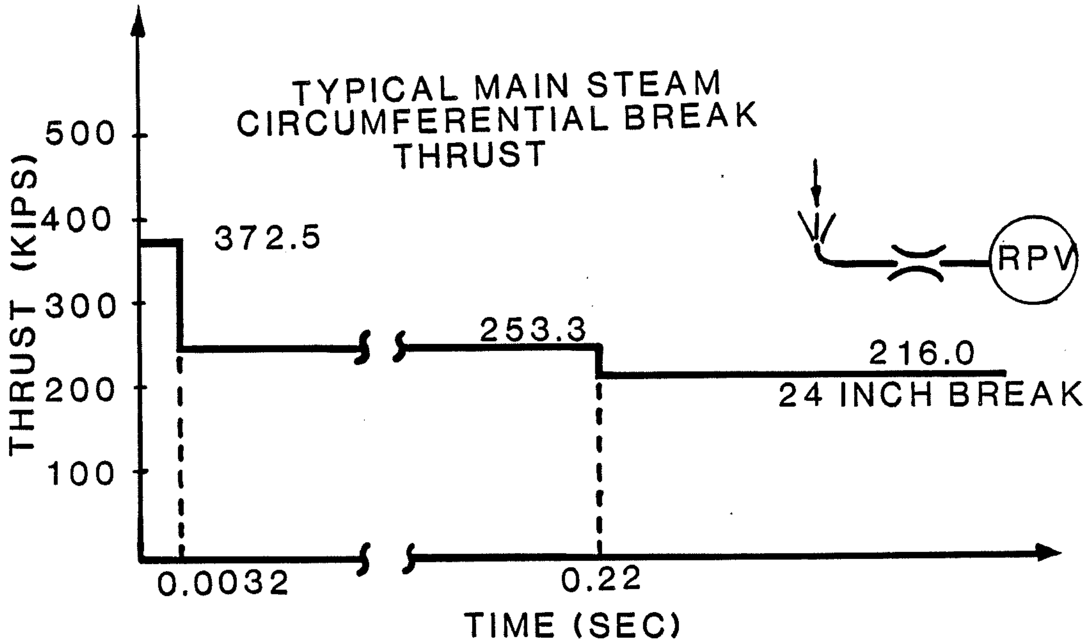
FIGURE 3.6-20

MAIN STEAM PIPING IN STEAM TUNNEL



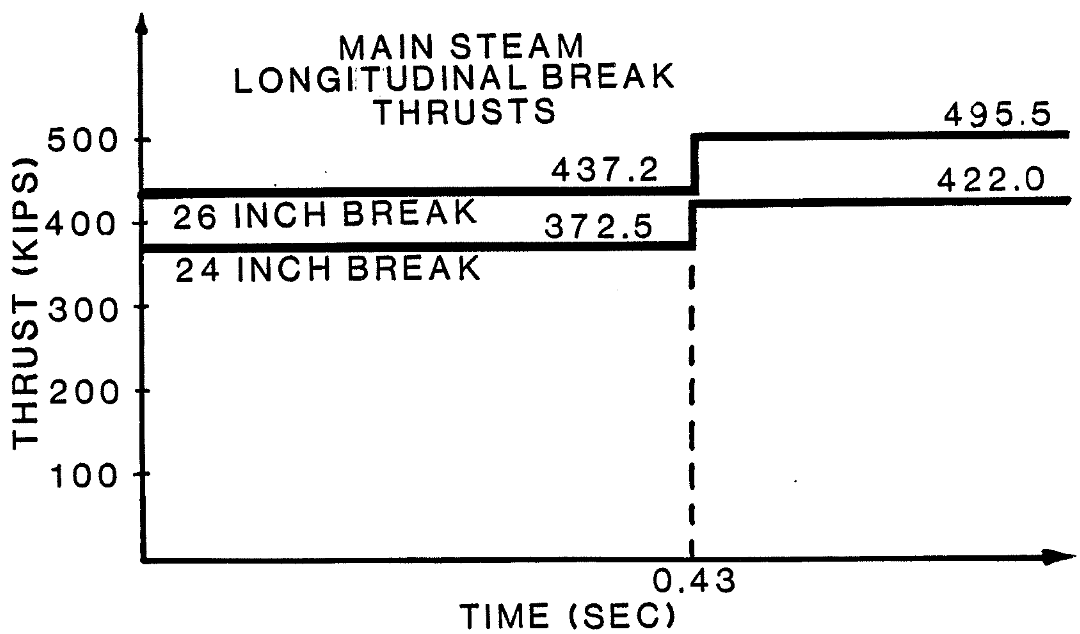


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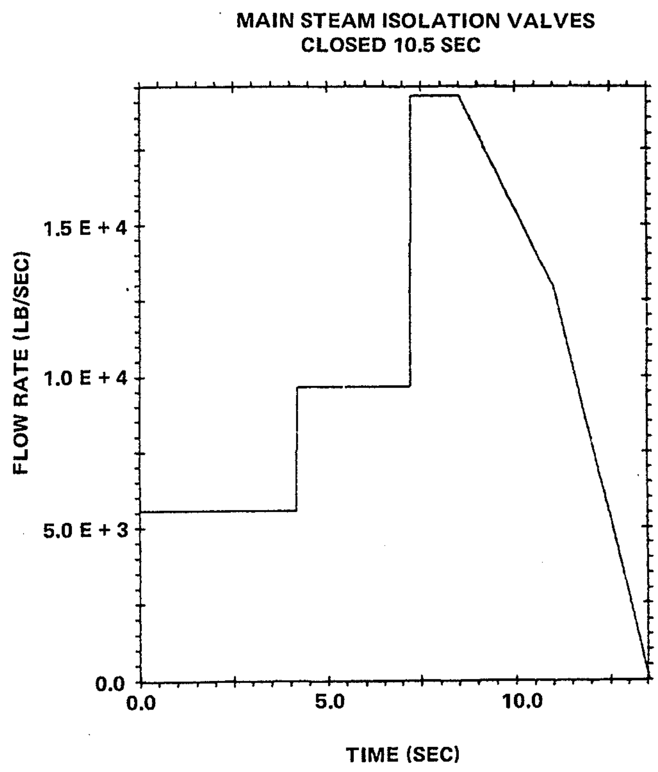
FIGURE 3.6-21

TYPICAL THRUST TIME HISTORIES
MAIN STEAM LINE BREAK



Note: The above blowdown history corresponds to 1050 psia dome pressure. For 1060 psia dome pressure (power uprate conditions), the blowdown rate will be 1% higher.

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FIGURE 3.6-22

BREAK FLOW RATES AFTER MAIN STEAM LINE BREAK

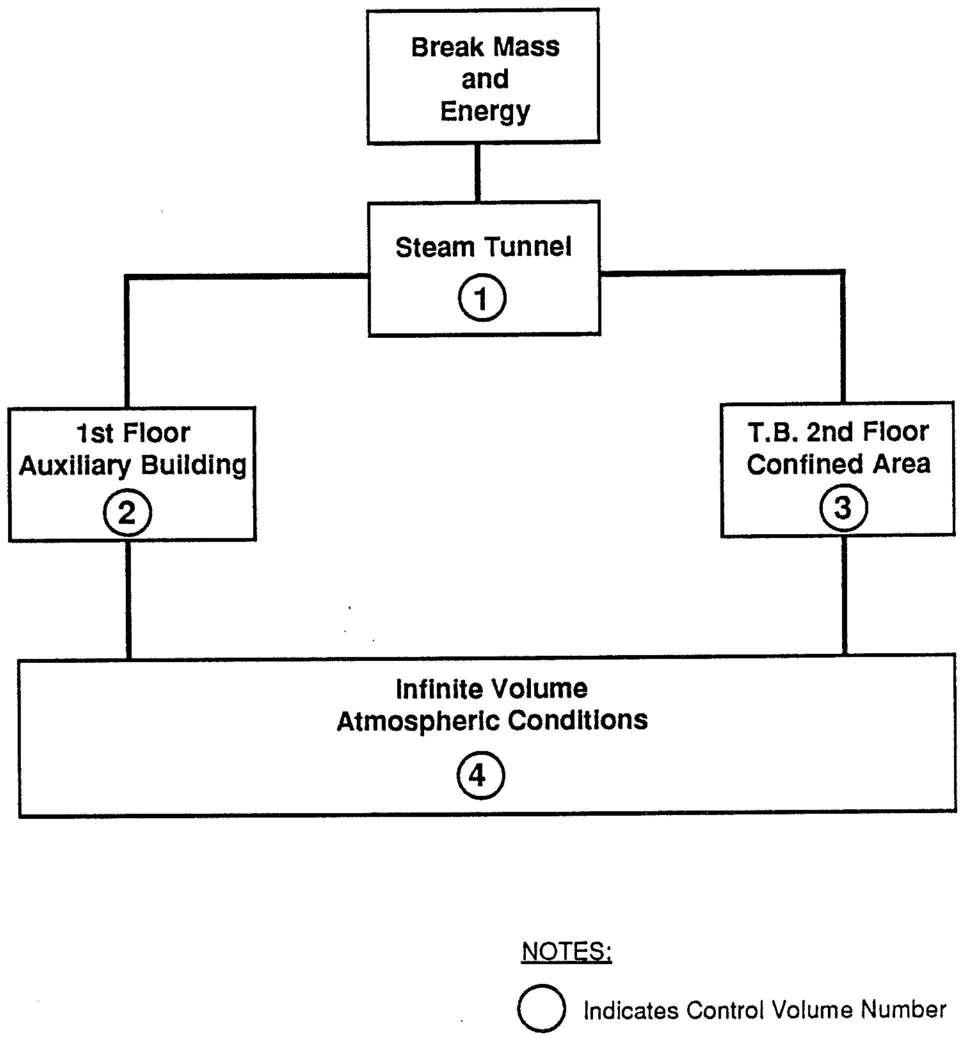

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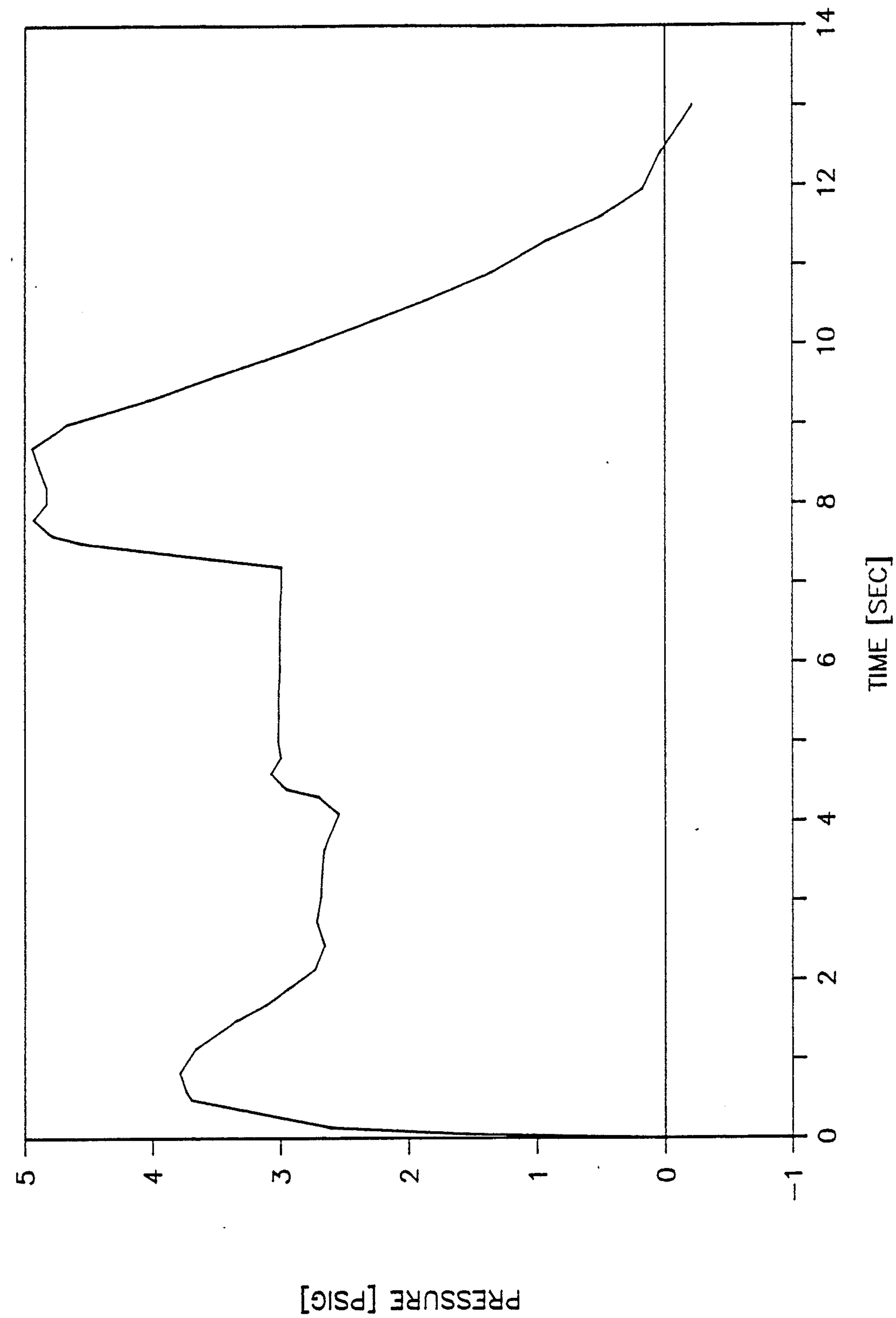
FIGURE 3.6-23

MAIN STEAM LINE BREAK IN THE
STEAM TUNNEL - MATHEMATICAL MODEL



Note: The pressure transient corresponds to 1050 psia steam dome pressure. For 1060 psia dome pressure (power uprate conditions), the peak pressure will be 5.1 psig.

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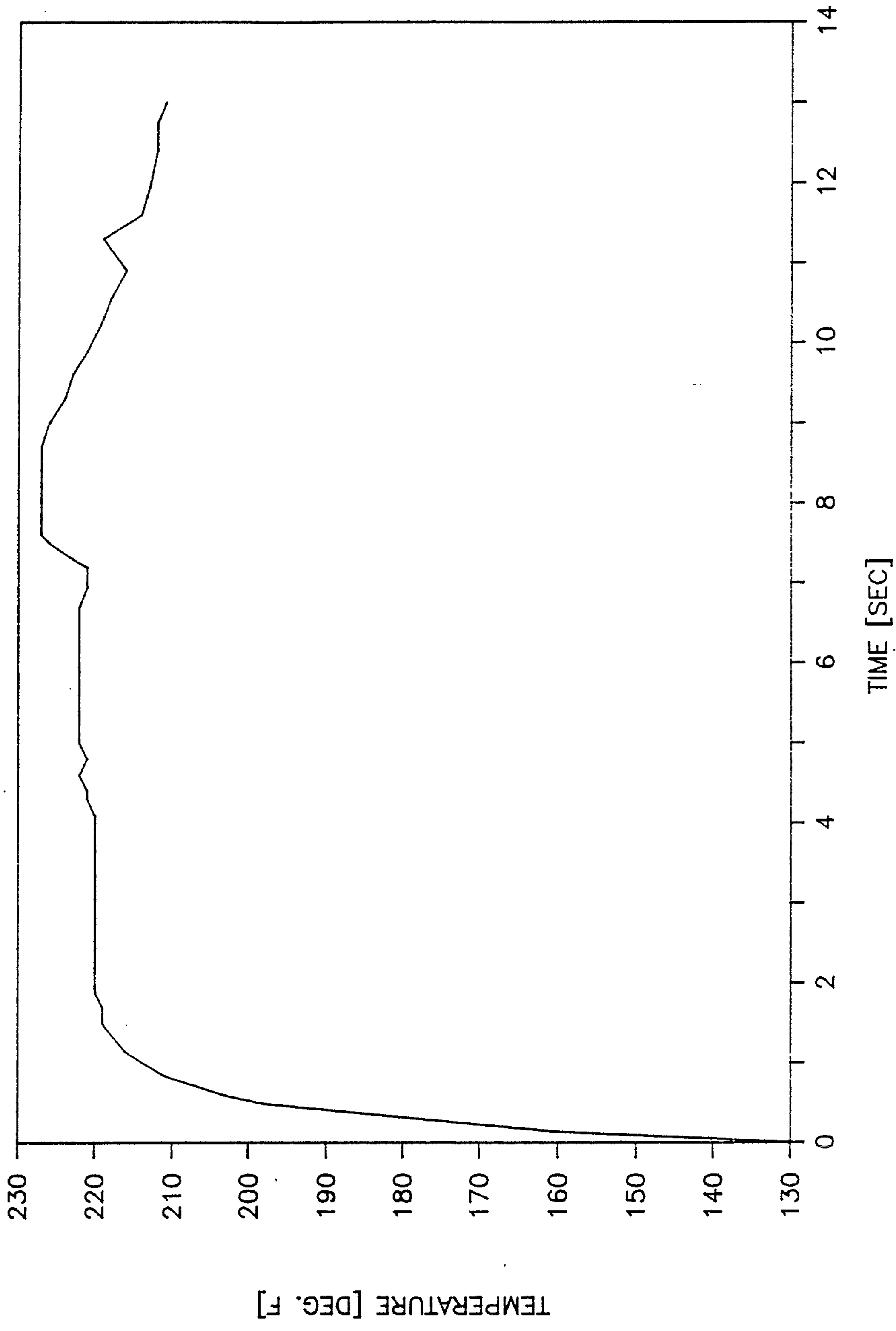
FIGURE 3.6-24

STEAM TUNNEL PRESSURE AFTER STEAM LINE BREAK



Note: The temperature transient corresponds to 1050 psia steam dome pressure. For 1060 psia dome pressure (power uprate conditions), the peak temperature will be 228°F.

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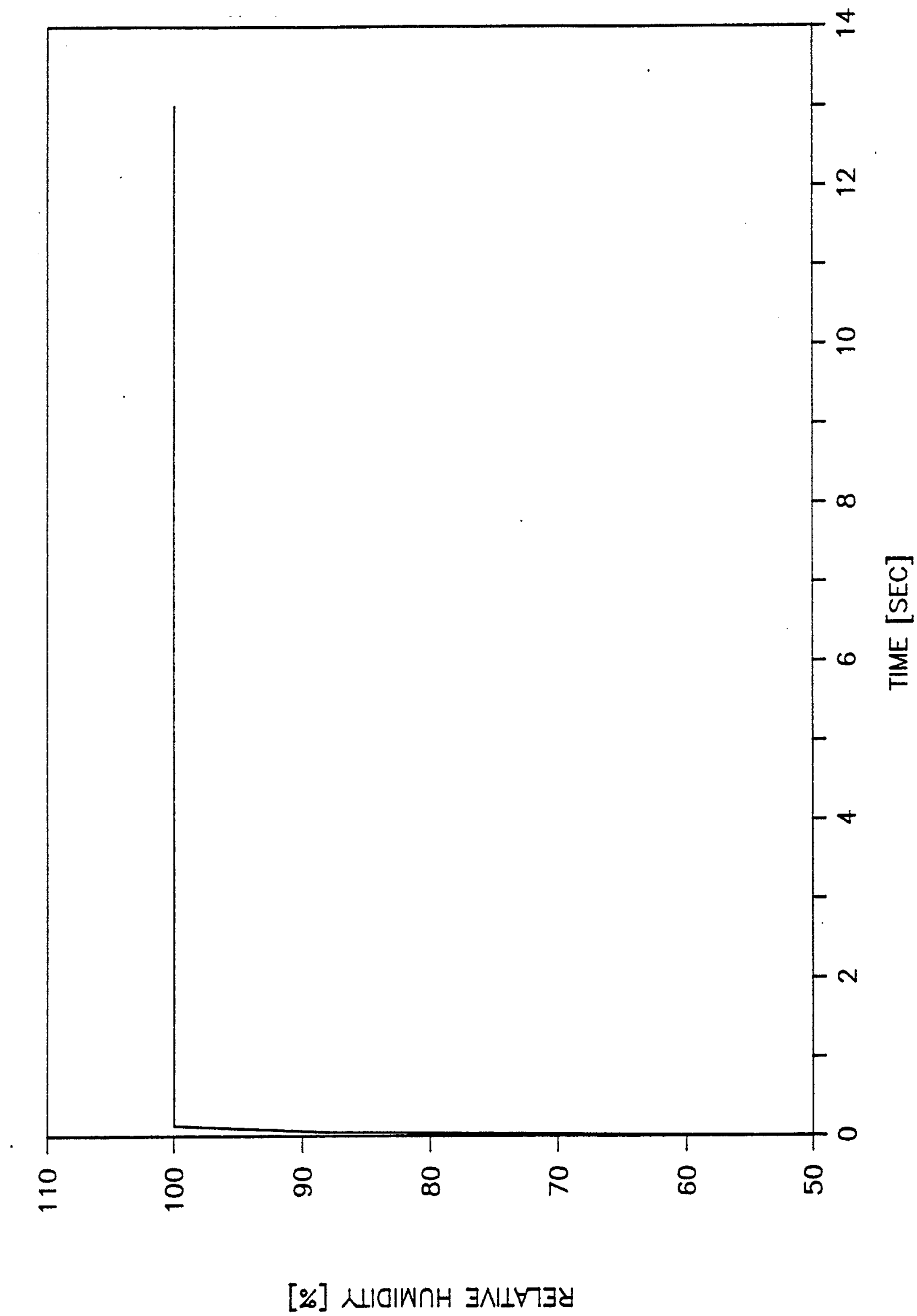
FIGURE 3.6-25

STEAM TUNNEL TEMPERATURE AFTER MAIN STEAM LINE BREAK



Note: The relative humidity profile corresponds to 1050 psia steam dome pressure. For 1060 psia dome pressure (power uprate conditions), the profile remains the same.

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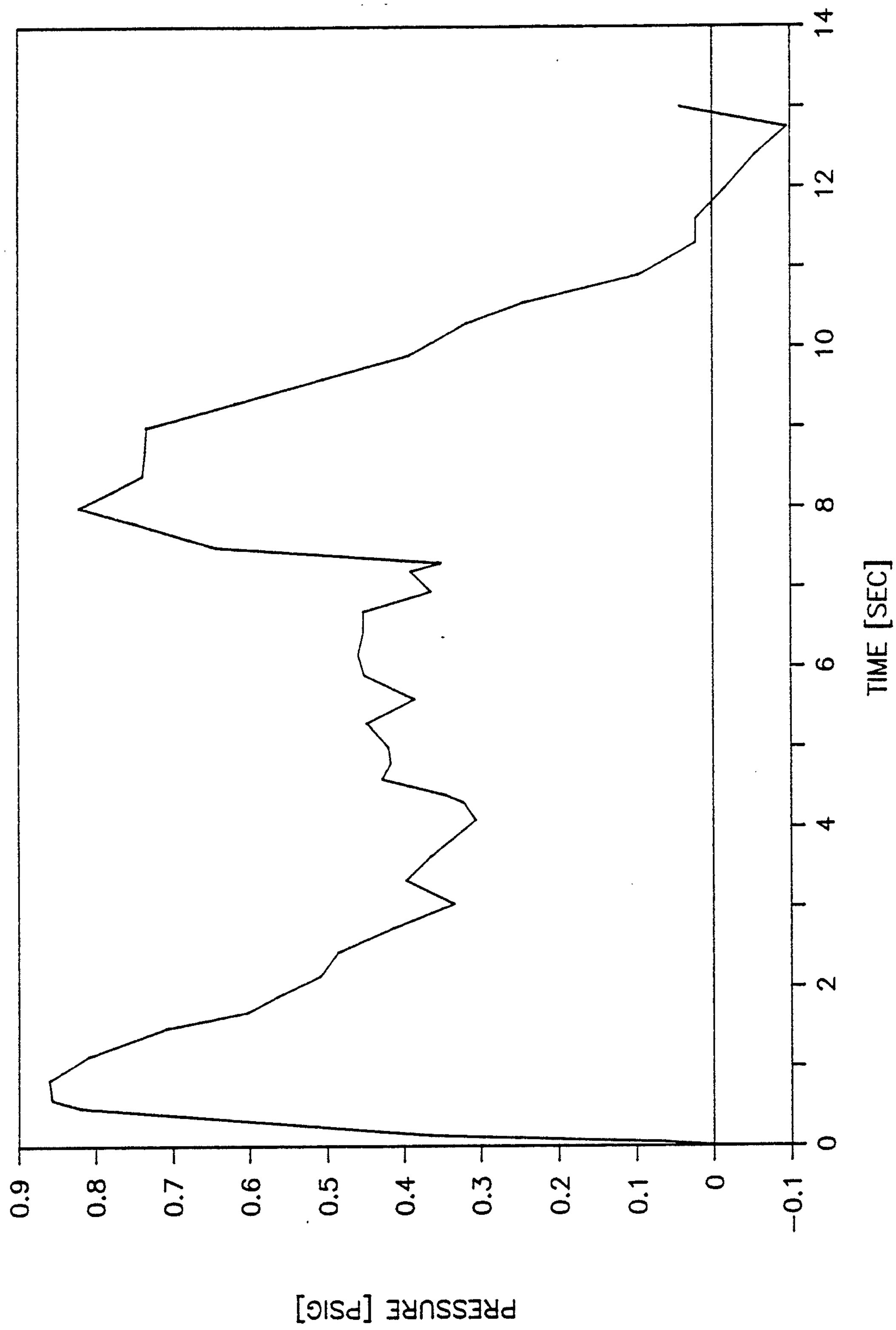
FIGURE 3.6-26

STEAM TUNNEL RELATIVE HUMIDITY AFTER MAIN STEAM LINE BREAK



Note: The pressure profile corresponds to 1050 psia steam dome pressure. For 1060 psia dome pressure (power uprate conditions), the peak pressure will be 0.9 psig.

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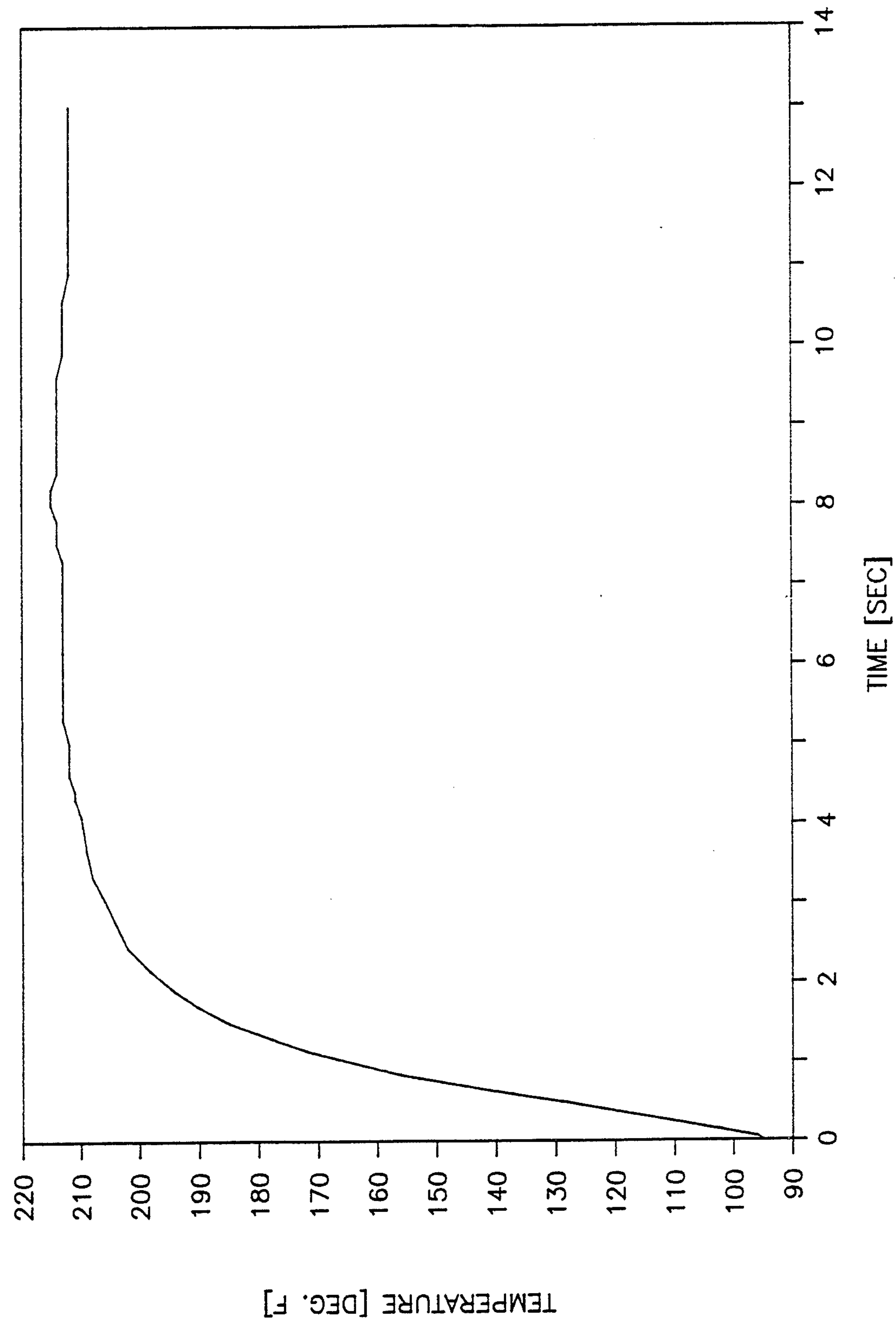
FIGURE 3.6-27

FIRST FLOOR AUXILIARY BUILDING PRESSURE AFTER MAIN STEAM LINE BREAK



Note: The temperature profile corresponds to 1050 psia steam dome pressure. For 1060 psia dome pressure (power uprate conditions), the temperature profile is practically unaffected.

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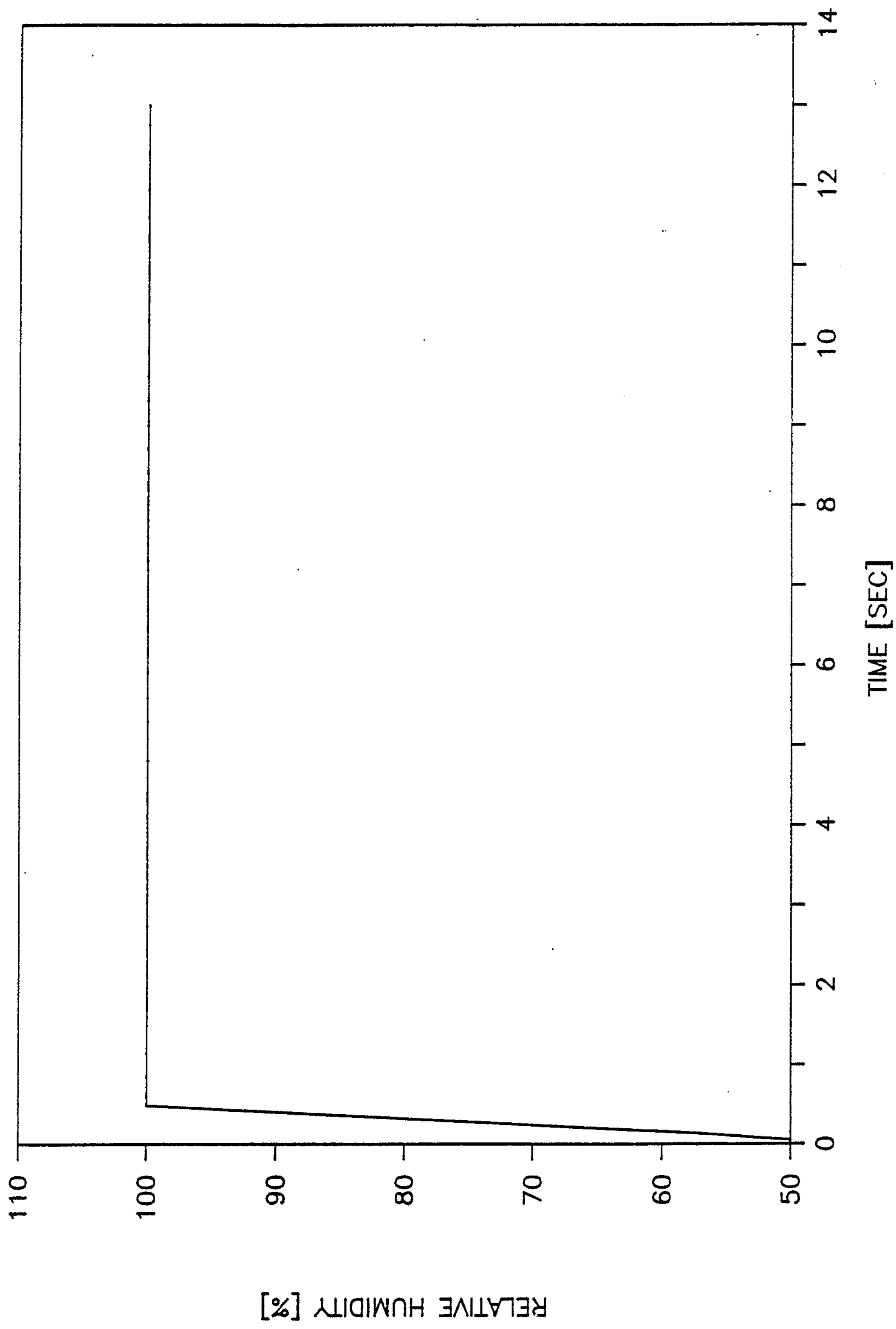
FIGURE 3.6-28

FIRST FLOOR AUXILIARY BUILDING TEMPERATURE AFTER MAIN STEAM LINE BREAK



Note: The relative humidity profile corresponds to 1050 psia steam dome pressure. For 1060 psia dome pressure (power uprate conditions), the relative humidity profile is practically unaffected.

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FIGURE 3.6-29

FIRST FLOOR AUXILIARY BUILDING RELATIVE HUMIDITY AFTER MAIN STEAM LINE BREAK

Figure Intentionally Removed
Refer to Plant Drawing C-2546

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FIGURE 3.6-30, SHEET 1

RESTRAINT STRUCTURE IN STEAM TUNNEL

Figure Intentionally Removed
Refer to Plant Drawing C-2539

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FIGURE 3.6-30, SHEET 2

RESTRAINT STRUCTURE IN STEAM TUNNEL

Figure Intentionally Removed
Refer to Plant Drawing C-2538

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FIGURE 3.6-30, SHEET 3

RESTRAINT STRUCTURE IN STEAM TUNNEL

Figure Intentionally Removed
Refer to Plant Drawing STEAM TUNNEL SKETCH

FIGURE 3.6-31

STEAM TUNNEL – PLAN VIEW

Figure Intentionally Removed
Refer to Plant Drawing STEAM TUNNEL SKETCH

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FIGURE 3.6-32
STEAM TUNNEL

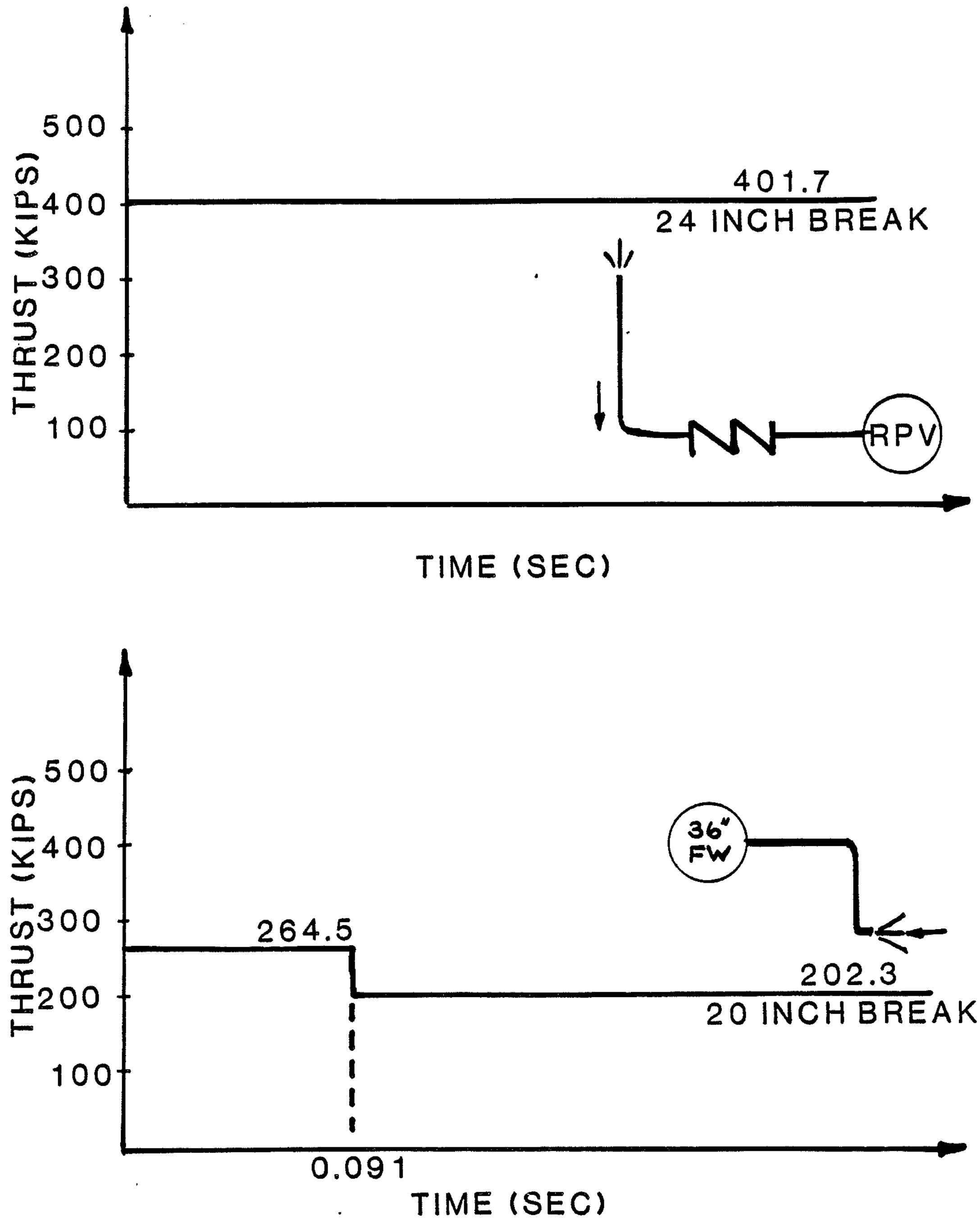

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FIGURE 3.6-33

TYPICAL FEEDWATER CIRCUMFERENTIAL BREAK THRUST

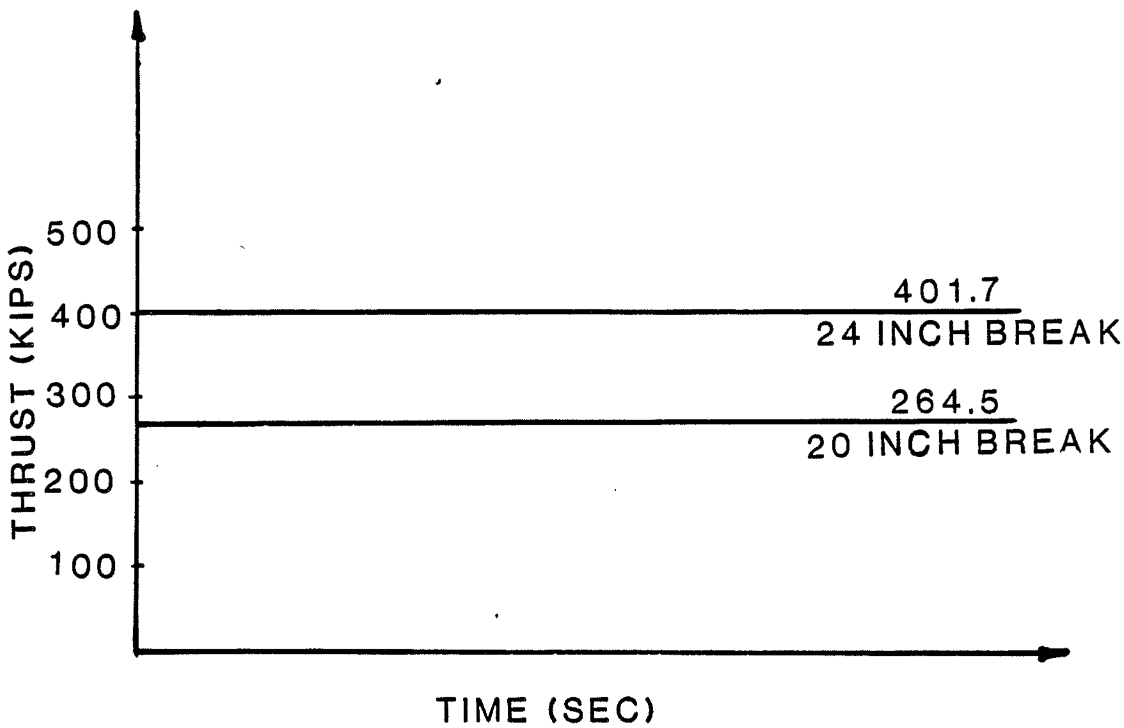

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FIGURE 3.6-34

FEEDWATER LONGITUDINAL BREAK THRUSTS

Figure Intentionally Removed
Refer to Plant Drawing STEAM TUNNEL FW PIPE

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FIGURE 3.6-35

FEEDWATER PIPING IN STEAM TUNNEL

Figure Intentionally Removed
Refer to Plant Drawing AUX BLDG FIRST FLOOR

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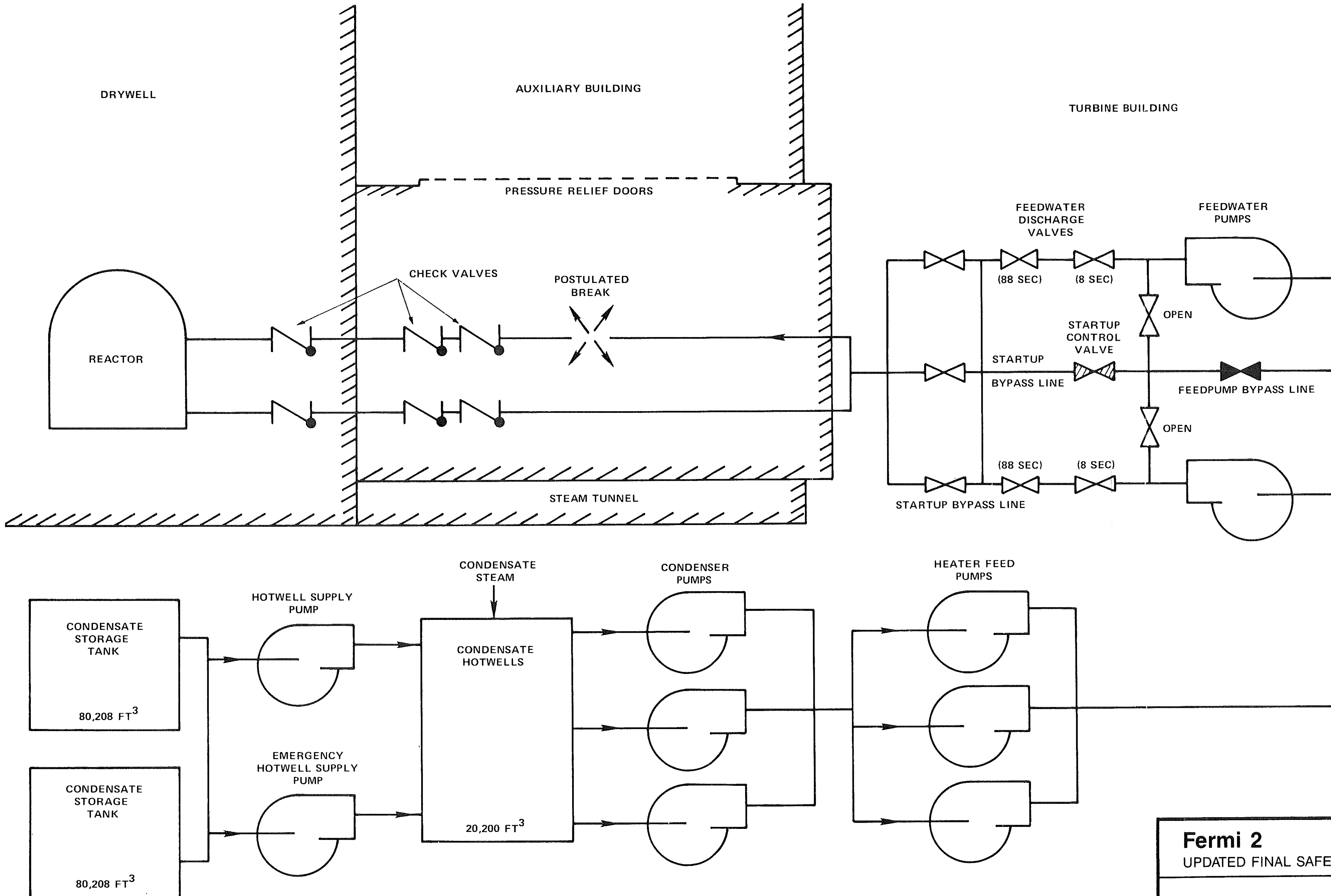
FIGURE 3.6-36

REACTOR/AUXILIARY BUILDING FIRST FLOOR



NUCLEAR SERVICES CORPORATION DRAWING

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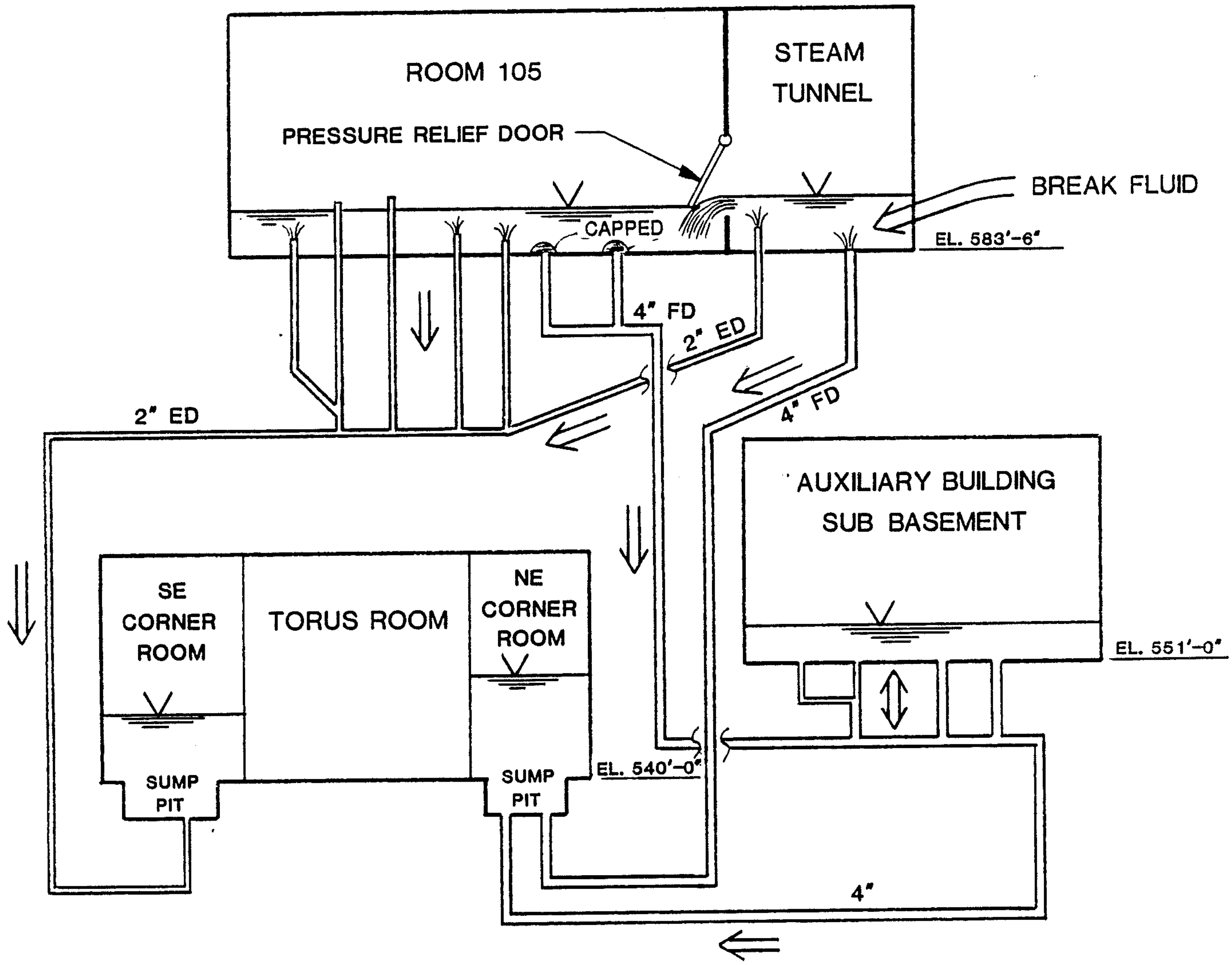
FIGURE 3.6-37

TYPICAL FEEDWATER LINE BLOWDOWN FORCE MODEL – CIRCUMFERENTIAL BREAK



NOTES:

1. ⇒ INDICATES FLOW DIRECTION
2. THIS FIGURE IS NOT TO SCALE
3. ⊻ INDICATES WATER SURFACE
4. FD - FLOOR DRAIN
   ED - EQUIPMENT DRAIN

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FIGURE 3.6-38

STEAM TUNNEL AND AUXILIARY BUILDING FIRST FLOOR FLOODING LEVEL VERSUS TIME FEEDWATER LINE BREAK

Figure Intentionally Removed
Refer to Plant Drawing HPCI SKETCH

Figure Intentionally Removed
Refer to Plant Drawing HPCI SKETCH

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FIGURE 3.6-40

PIPE-WHIP RESTRAINT LOCATIONS ON
HIGH-PRESSURE COOLANT INJECTION
STEAM LINE

Figure Intentionally Removed
Refer to Plant Drawing RCIC SKETCH

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FIGURE 3.6-41

REVISED ROUTING OF REACTOR CORE ISOLATION COOLING STEAM LINE

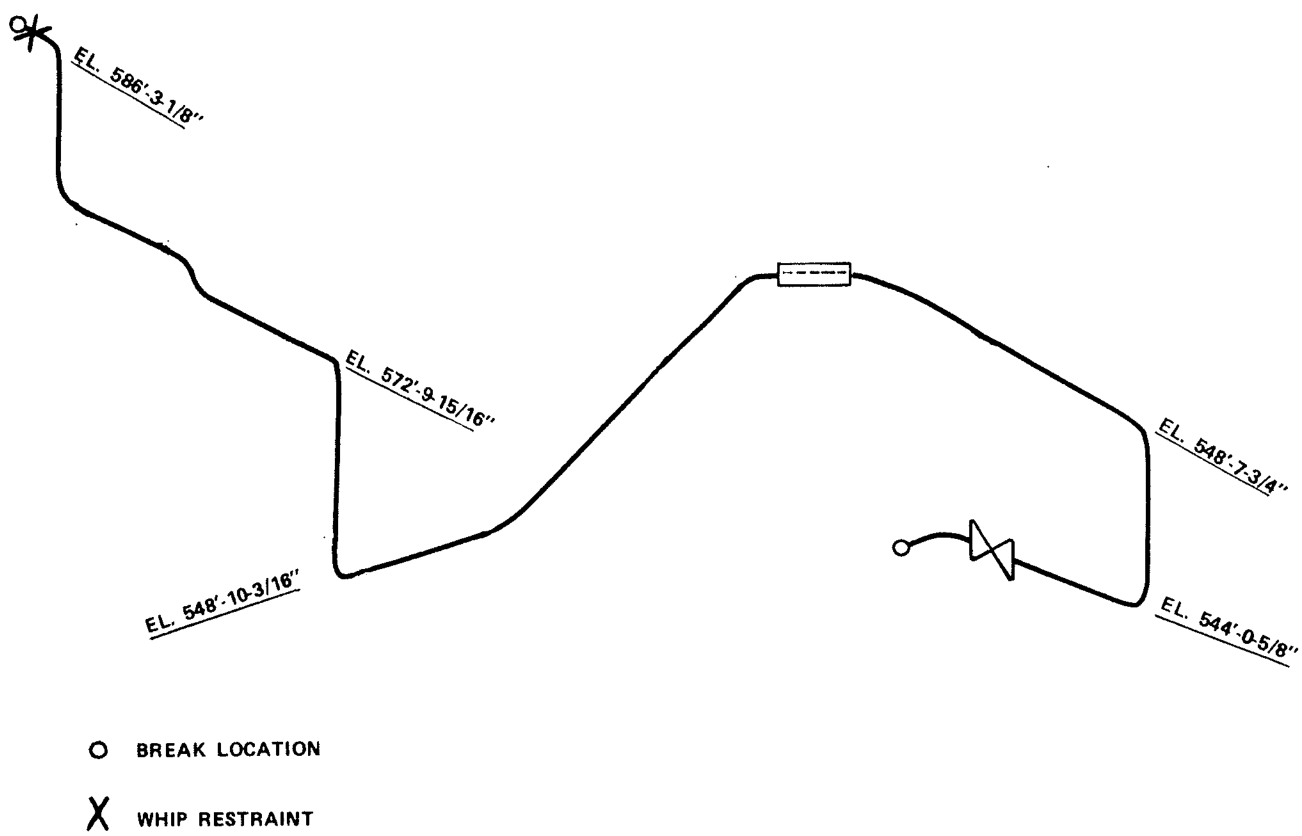

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FIGURE 3.6-42

REACTOR CORE ISOLATION COOLING STEAM LINE PIPE BREAK AND RESTRAINT LOCATIONS

Figure Intentionally Removed
Refer to Plant Drawing RWCU SKETCH

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FIGURE 3.6-43

REACTOR WATER CLEANUP LINE ROUTING AND MODIFICATIONS ON REACTOR BUILDING SECOND FLOOR

Figure Intentionally Removed
Refer to Plant Drawing RWCU SKETCH

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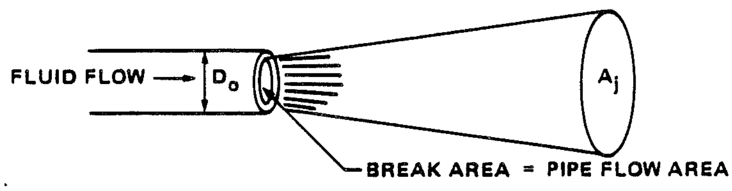
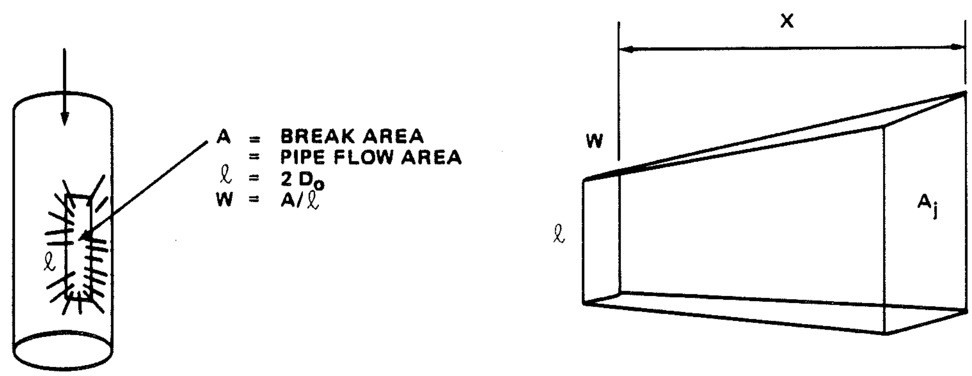
FIGURE 3.6-44

REACTOR WATER CLEANUP PUMP DISCHARGE
RESTRAINT LOCATIONS AND REVISED ROUTING

(a) DESIGN BASIS CIRCUMFERENTIAL BREAK



(b) DESIGN BASIS LONGITUDINAL BREAK



(c) CRITICAL CRACK BREAK

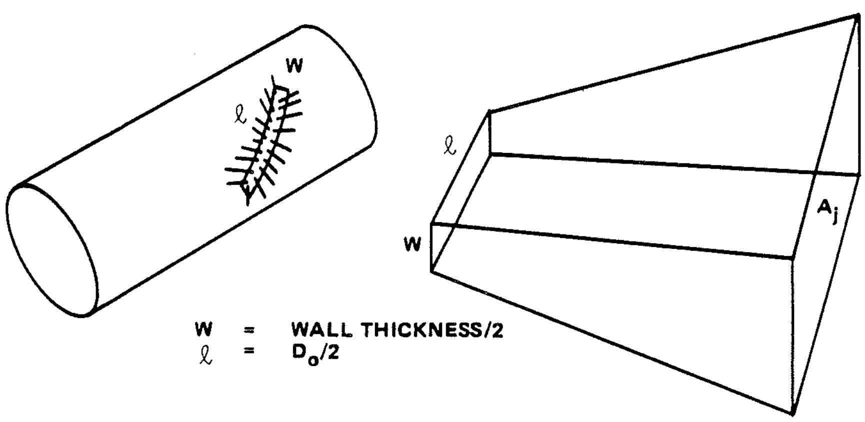


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FIGURE 3.6-45

JET CHARACTERISTICS

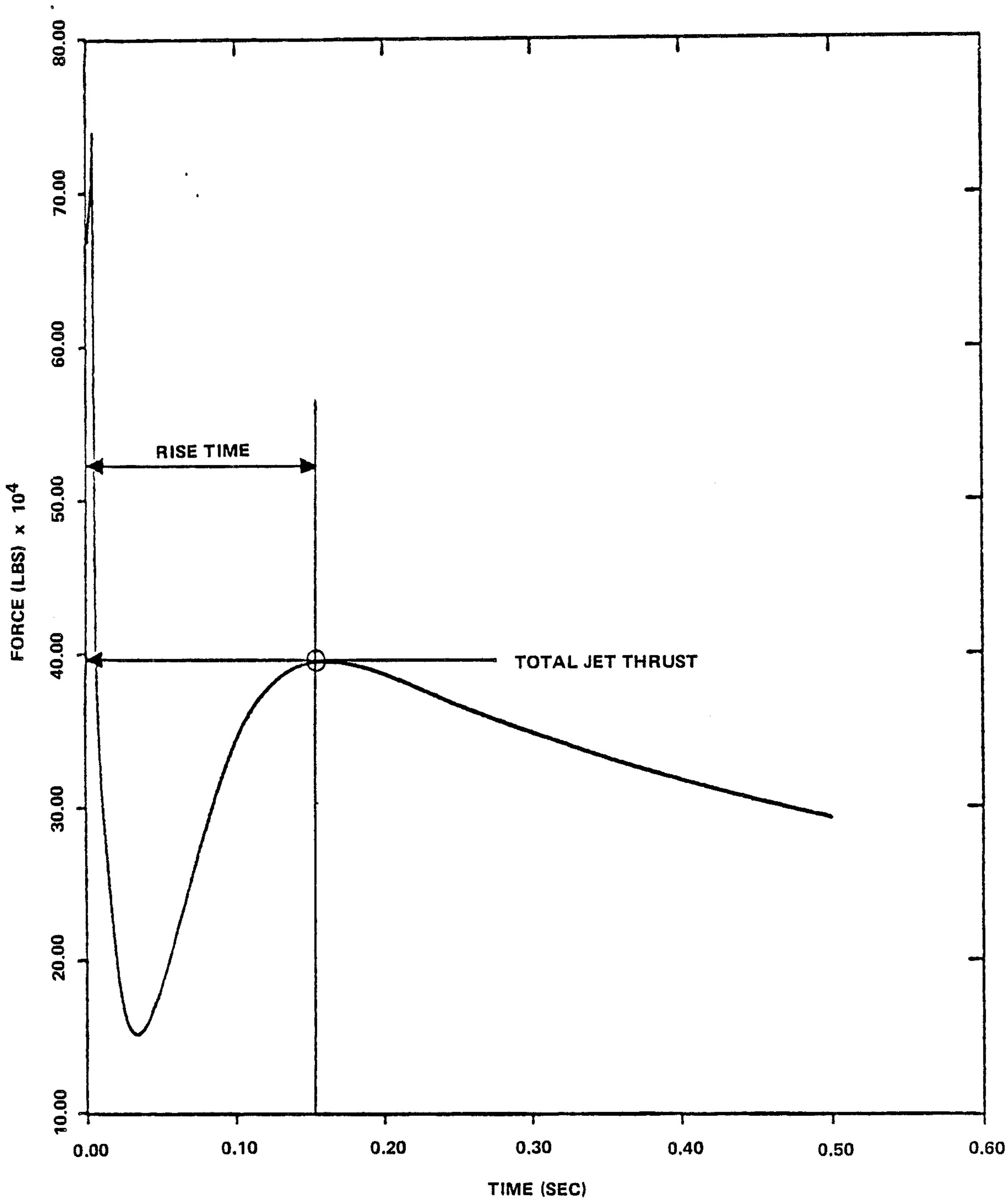

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FIGURE 3.6-46

DETERMINATION OF TOTAL JET THRUST

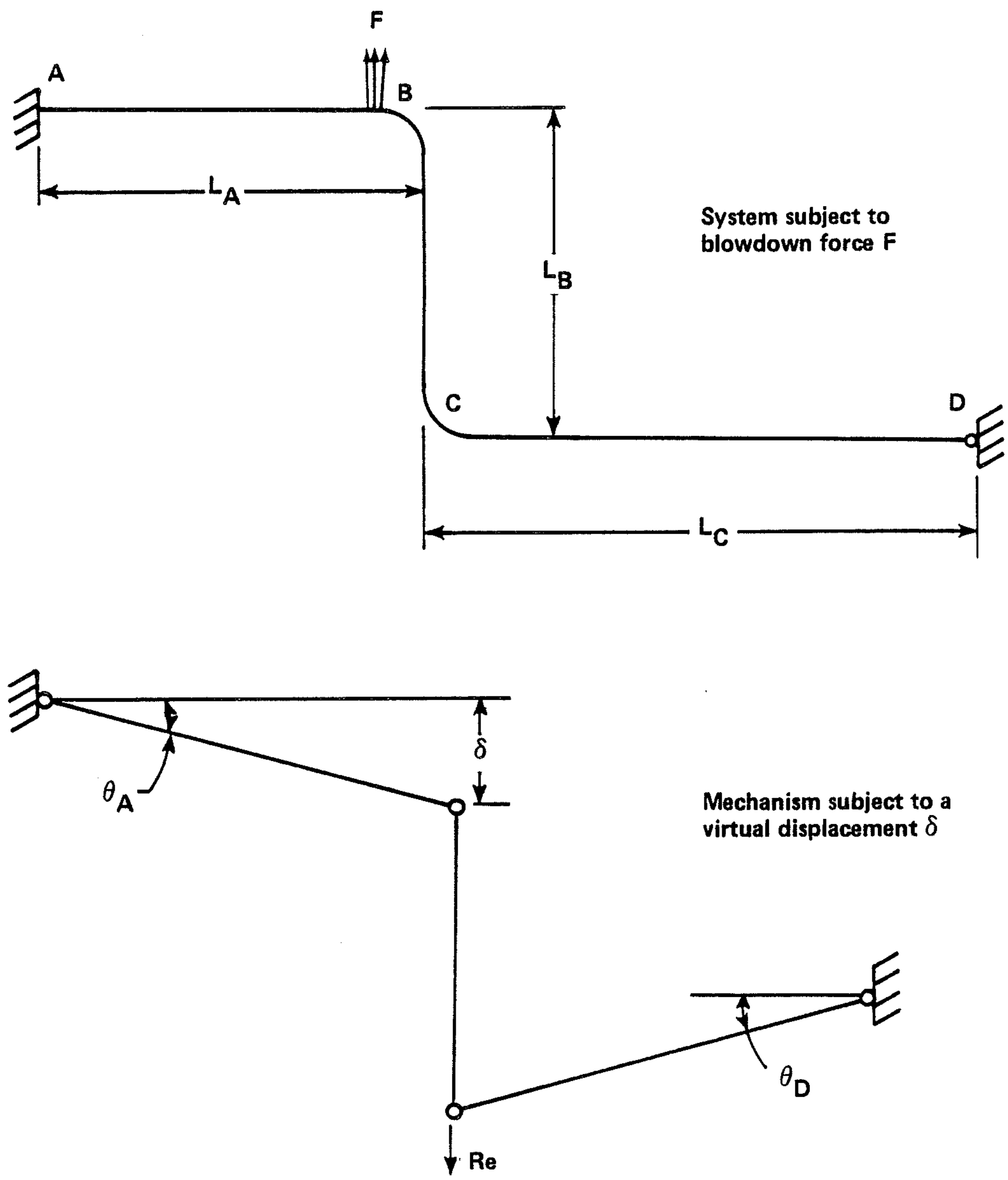

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FIGURE 3.6-47

MOTION AT POSTULATED LONGITUDINAL BREAK

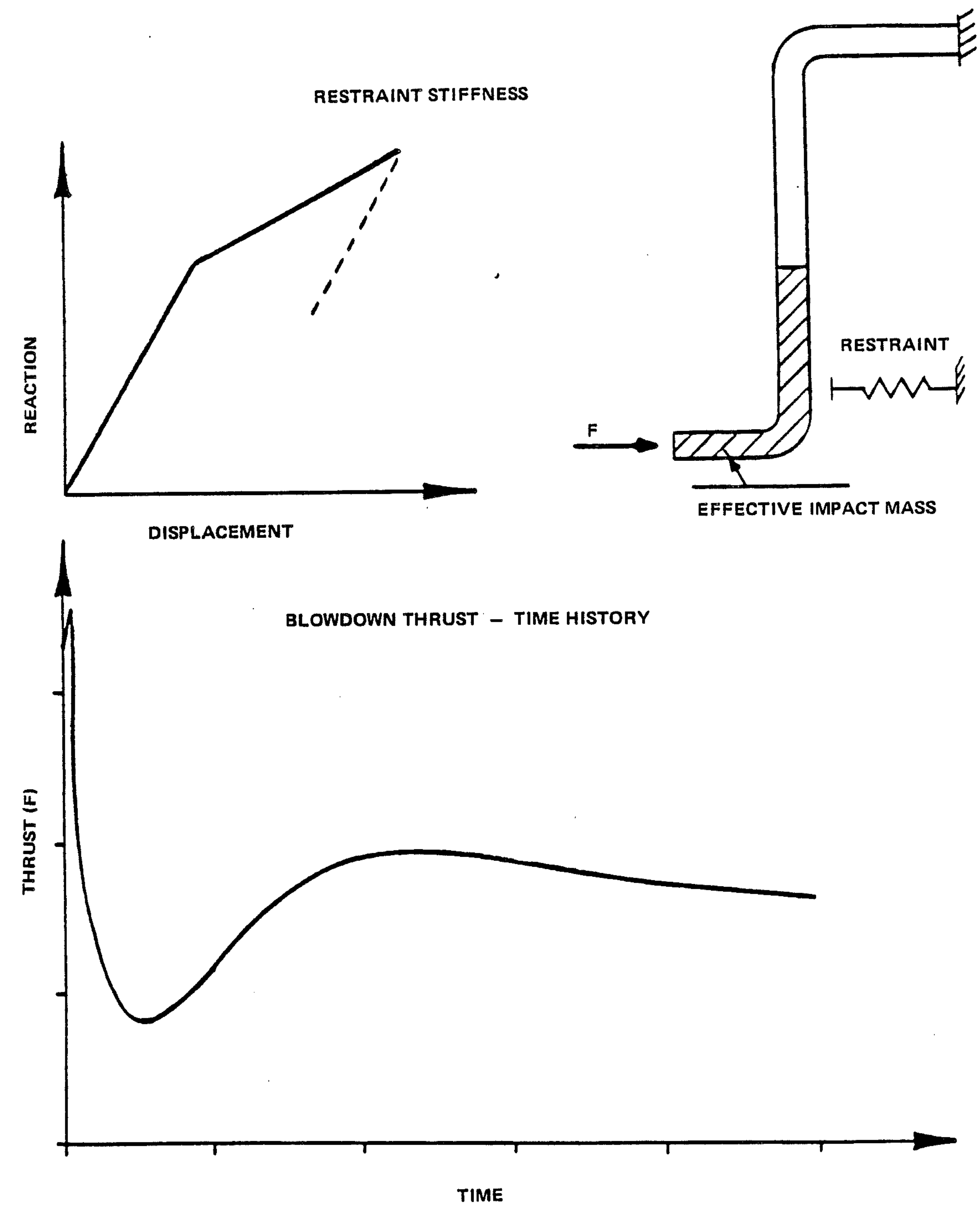

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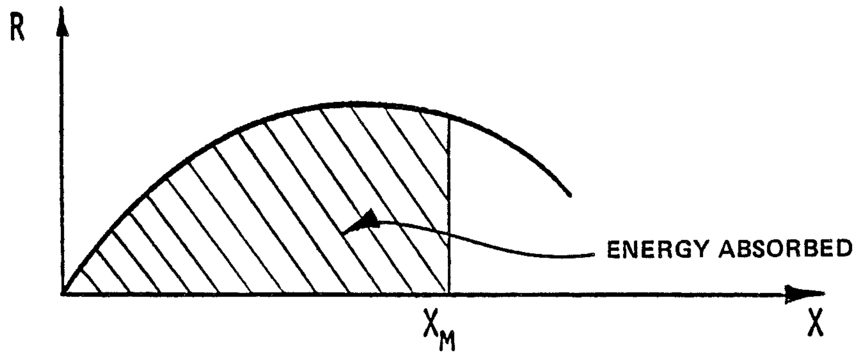
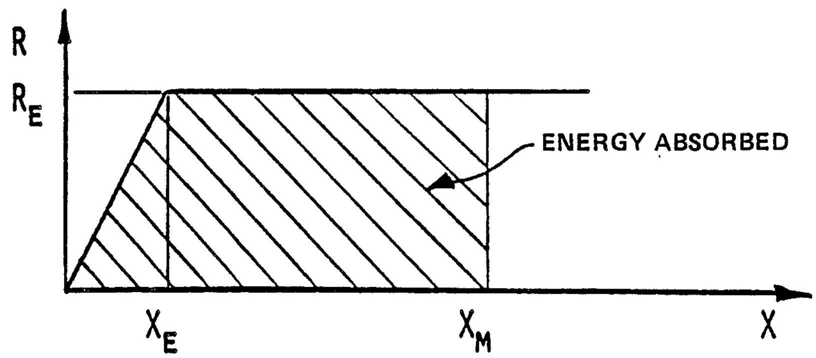
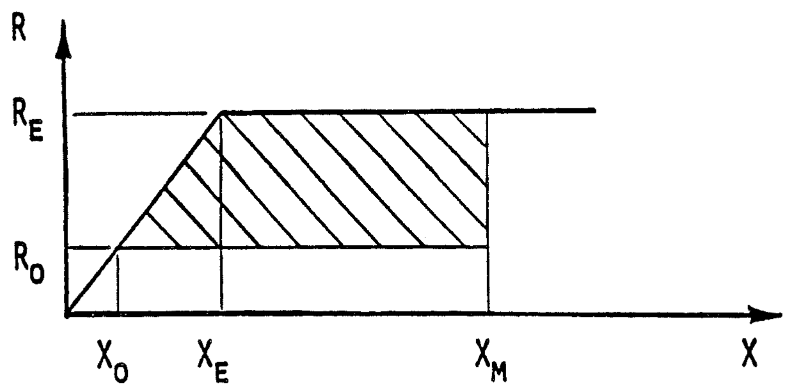
FIGURE 3.6-48
RESTRAINT IMPACT



A) NONLINEAR STRUCTURAL SYSTEM



B) ELASTO-PLASTIC STRUCTURAL SYSTEM



C) ELASTO-PLASTIC SYSTEM WITH INITIAL LOADS

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FIGURE 3.6-49

RESISTANCE-DISPLACEMENT CURVES

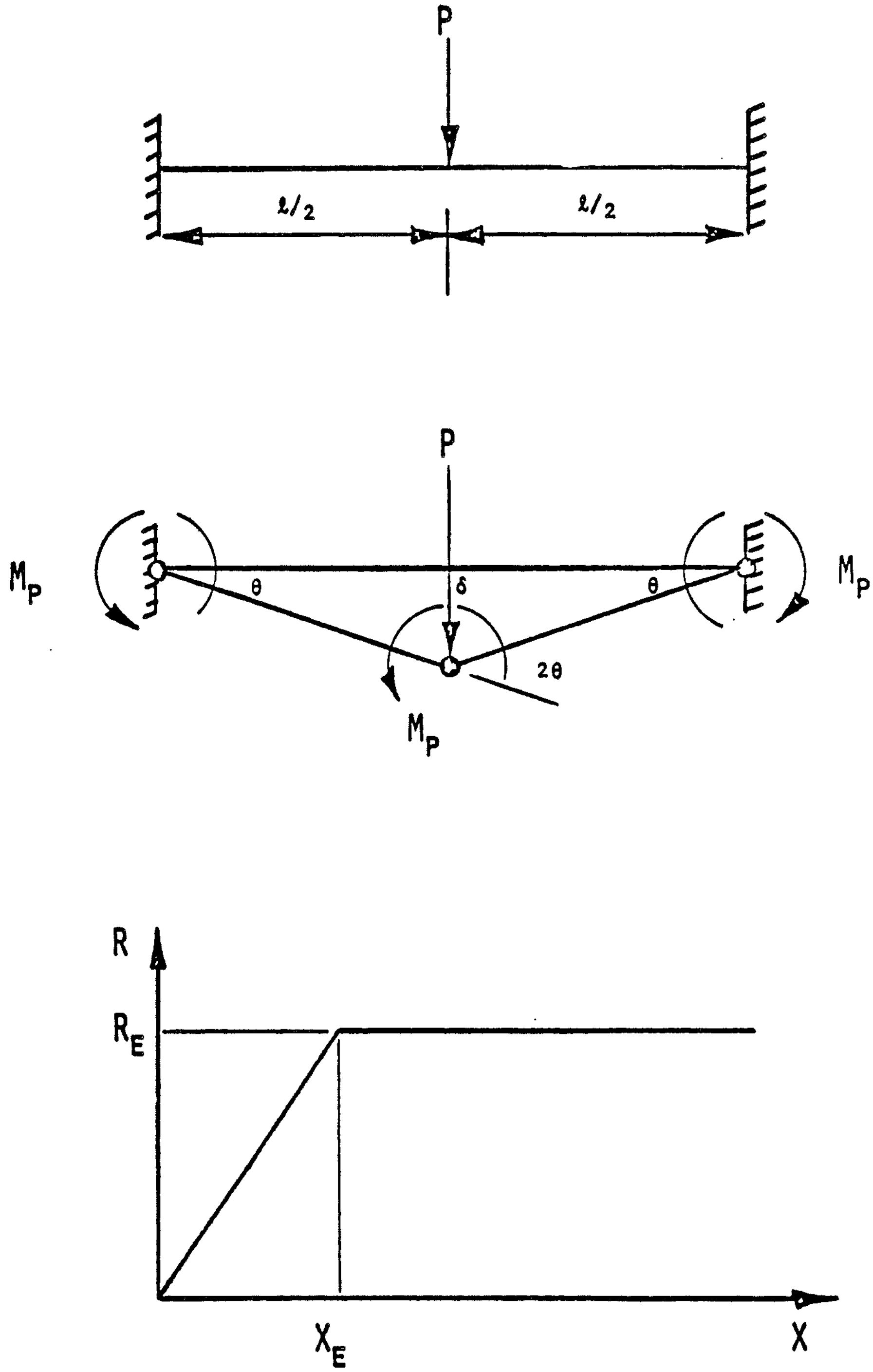

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FIGURE 3.6-50

RESISTANCE-DISPLACEMENT FOR FIXED-FIXED BEAM

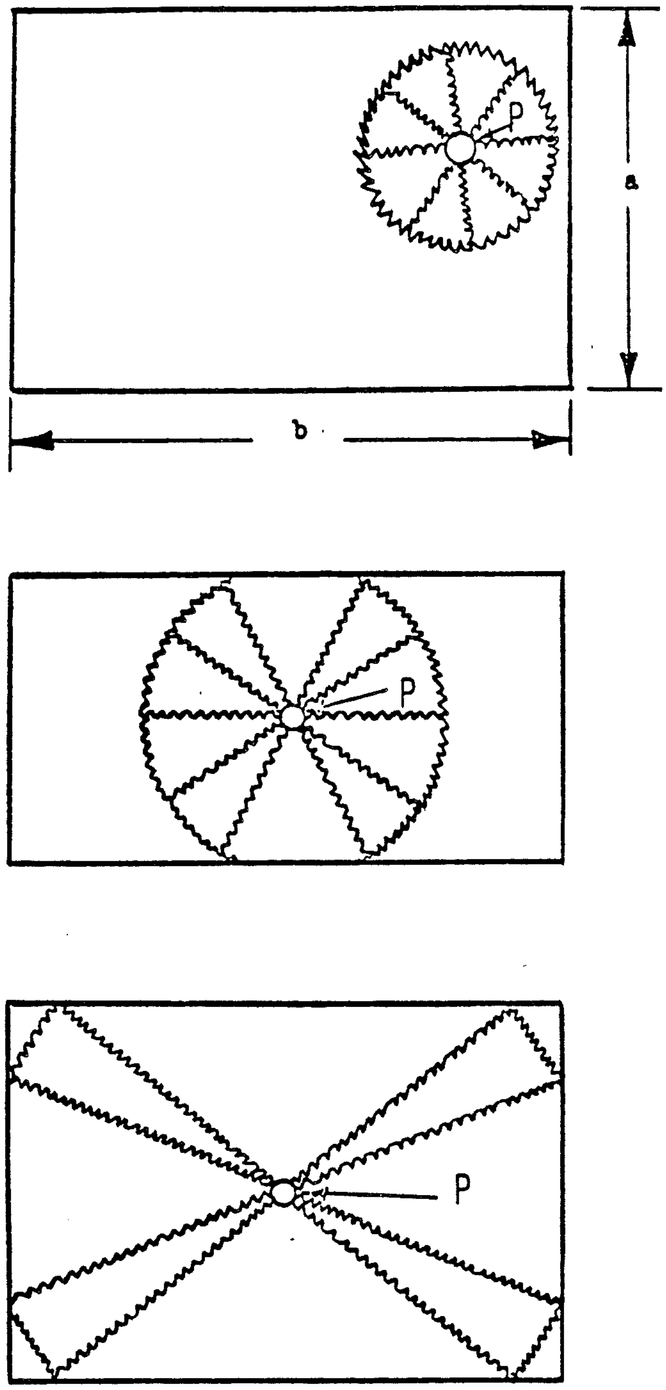

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FIGURE 3.6-51

YIELD LINE PATTERNS FOR SLABS SUBJECT TO POINT LOADS

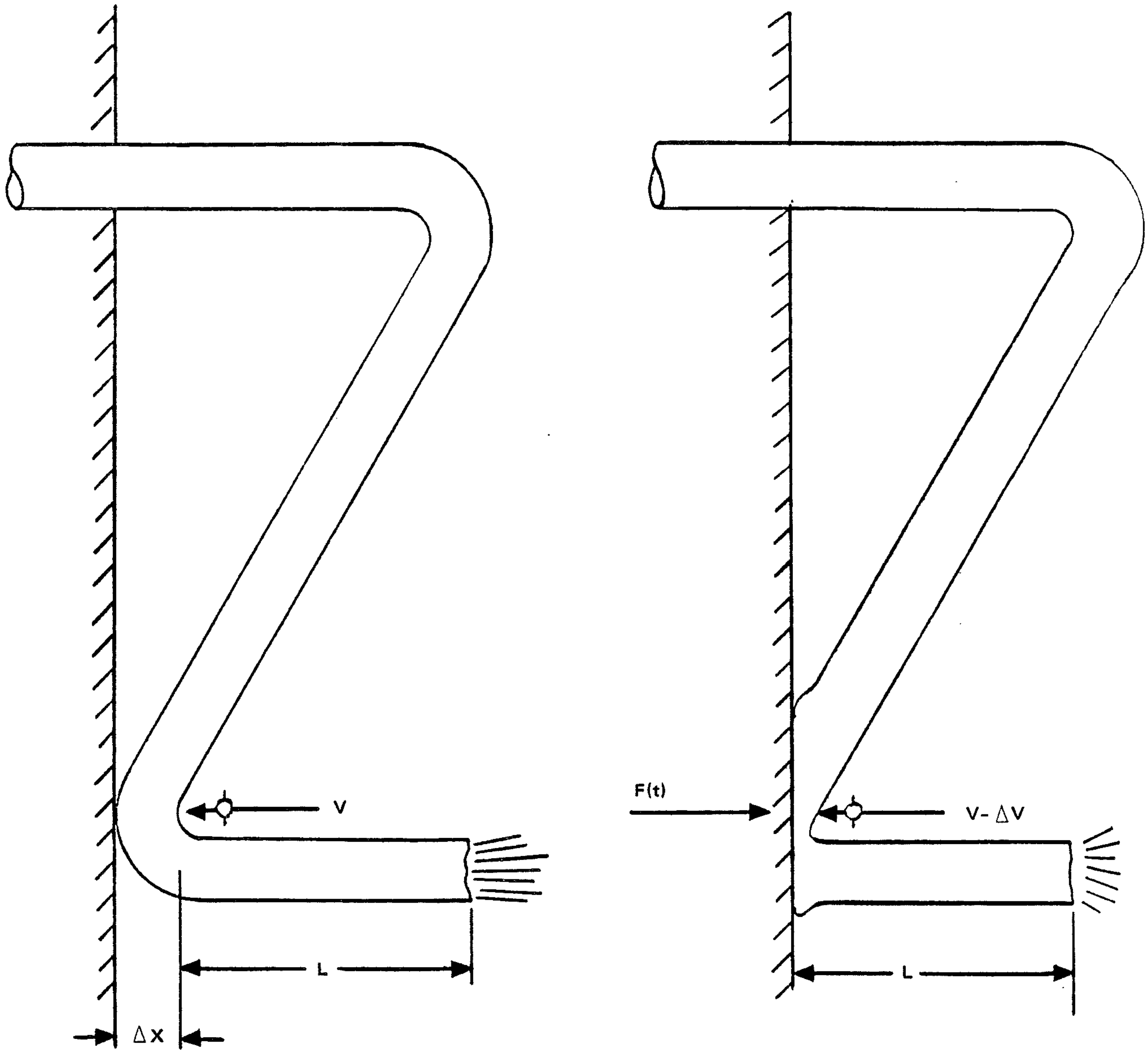

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FIGURE 3.6-52

CALCULATION OF IMPACT TIME-HISTORY

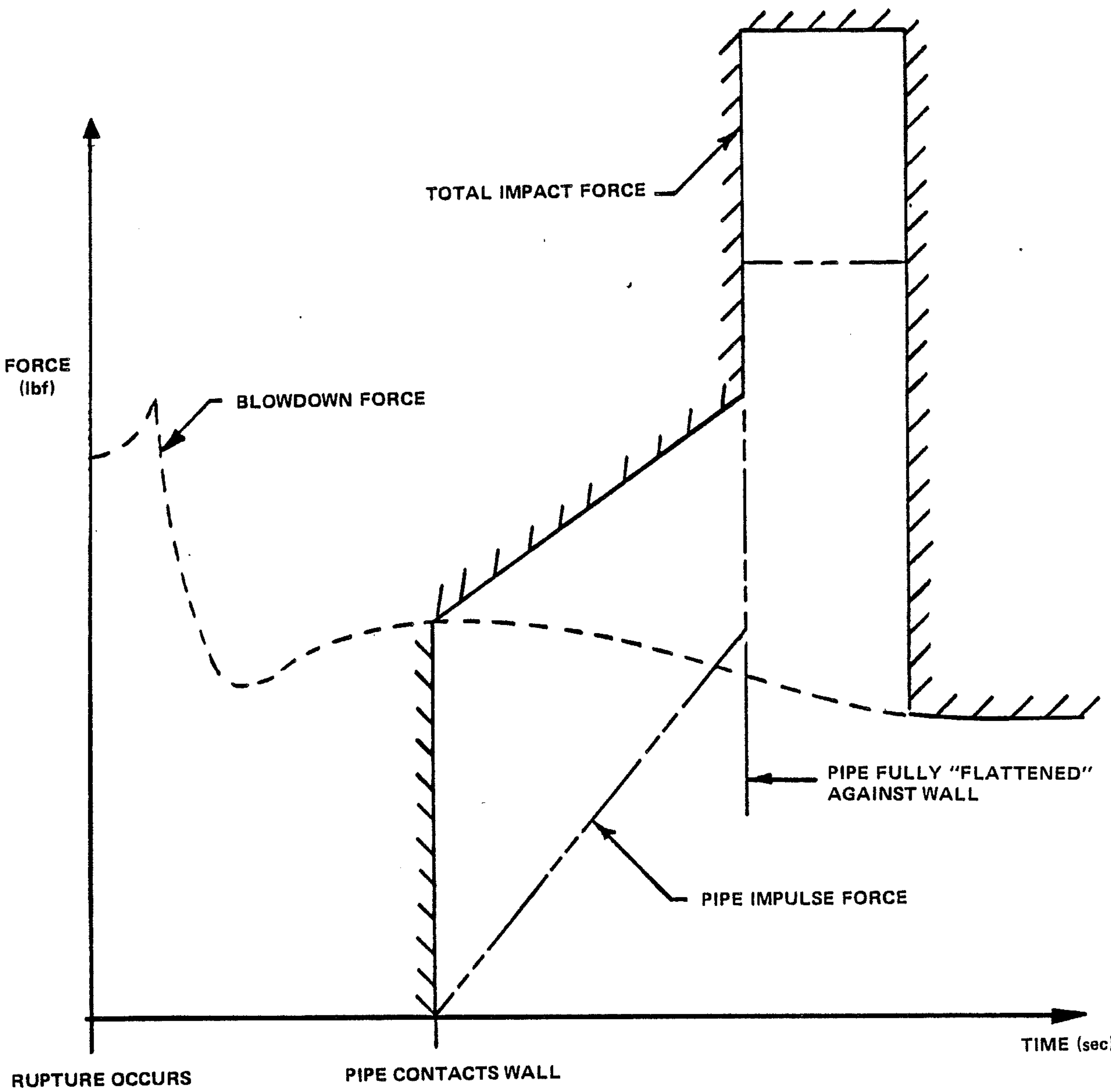

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FIGURE 3.6-53

IMPACT TIME-HISTORY

3.7 SEISMIC DESIGN

For purposes of seismic design, structures, systems, and components are categorized as follows:

Category I: Safety-Related Structures, Systems, and Components

Plant structures, systems, and components, including their foundations and supports, that are required to be designed to remain functional in the event of a safe-shutdown earthquake (SSE) as described in Regulatory Guide 1.29, are designated Category I. A detailed discussion on the design Category I structures and components is provided in the following sections.

Category II/I: Non-Safety-Related Items in a Safety-Related Envelope

Non-safety-related components--control, electrical, mechanical, or structural--in a safety-related envelope are designated Category II/I. The continued functioning of these items is not required, but their failure could reduce the functioning of plant Category I items. Hangers and supports for Category II/I components carrying safety-related items are Category I.

Category II/I items are designed to maintain their structural and mounting integrity. For normal (operating) loads, the maximum stresses in components are required to remain within code-specified allowable limits. Components may be stressed beyond the yield limit stress during SSE loading. A reasonable limit, depending on material capability, is placed on the allowable ductility ratio. Test and/or analysis may be performed to establish Category II/I component ductility levels to be satisfactory under postulated loads.

Nonseismic: Non-Safety-Related Structures and Associated Non-Safety-Related Components

Structures and components designated as Nonseismic are designed by the appropriate state-of-the-art methods.

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra

The design-basis earthquake (DBE) as referred to in the Preliminary Safety Analysis Report (PSAR) and other documents is referred to herein as the SSE.

The results of the seismological studies performed by Dames & Moore (D&M) for Fermi 2 are summarized below.

Confirmatory site-specific earthquake evaluations were recently completed by Weston Geophysical to reaffirm the acceptability of the established Fermi 2 facility aseismic design bases.

Site-specific spectra were developed from real time-history data representing quakes with a magnitude never to be exceeded at the site and subsurface conditions similar to those at the site.

The site is located in one of the seismically stable regions in the United States. As shown by Figure 3.7-1, no earthquake epicenter has been located closer than about 15 miles and only nine earthquakes have been reported within 50 miles of the site since the beginning of the 19th century. None of these shocks were greater than Intensity V on the Modified Mercalli Scale (Table 3.7-1). Twelve earthquake epicenters of Intensity V or greater have been reported within 50 to 100 miles of the site and another 27 of Intensity V or greater were located at distances between 100 and 200 miles. These more distant shocks ranged up to Intensity VIII.

The closest reported earthquake of Intensity V or greater was in the 1877 shock west of Detroit, Michigan. This earthquake caused no damage near its epicenter and was not felt in the vicinity of the site. The remaining eight earthquakes, within 50 miles of the site, were of Intensity IV or smaller and none were larger than Intensity III at the site. The largest regional shocks occurred in 1937 near Lima, Ohio, in 1977 near Celina, Ohio, in 1980 near Sharpsburg, Kentucky, and in 1986 near Perry, Ohio. Although these shocks may have had epicentral intensities as great as VIII, none were greater than Intensity IV shocks at the site. The effect of these shocks in Michigan was not great and no damage resulted.

Although several shocks have been felt at the site within about the past two centuries, the maximum intensity at the site has not exceeded IV. None of the recorded earthquakes caused any damage at or near the site (Subsection 2.5.2).

With few exceptions, the significant earthquakes reported in the region can be associated with well-defined geologic structural zones (Subsection 2.5.2). To the north and east of the site, earthquakes are scarce and appear to be related to anticlinal structure in northern Michigan. To the west of the site, earthquake activity has consisted of infrequent minor shocks that can be related to faulting in southern Wisconsin and northern and central Illinois. To the south, earthquakes are believed to be related to the confluence of the Findlay, Cincinnati, and Kankakee Arches. There are no known faults within 25 miles of the site.

The site response spectra for the operating-basis earthquake (OBE) and the SSE, for the horizontal direction, are shown in Figures 3.7-2 and 3.7-3. Vertical ground motion for the SSE and OBE are taken as 2/3 (0.667) of the maximum horizontal ground acceleration. The maximum ground acceleration for horizontal motion for the SSE is 0.15g and for the OBE is 0.08g. These earthquakes for the stable Fermi site are very conservative and were selected jointly between Edison and the AEC staff and have received their acceptance (see Safety Evaluation by DRL, May 17, 1971). Earthquake history and other pertinent information on site geology and seismology are included in Section 2.5.

3.7.1.2 Design Response Spectra Derivation

3.7.1.2.1 General

The shapes of the OBE and SSE spectra essentially conform to the 1940 El Centro, California spectra with minor embellishments to accommodate the 1949 Olympia, Washington, and the 1935 Helena, Montana, experiences. The spectra are anchored at horizontal zero period accelerations of .08 and .15g respectively with corresponding vertical accelerations of .05 and .10g.

Internal equipment response spectra were derived based on detailed time-history analysis of the buildings subjected to numerous time-history base excitations. Time histories were employed in addition to those used to describe the shape of the basic ground spectra to ensure a broadband frequency content for equipment aseismic qualification purposes. In this regard, scaled earthquake records were used for generating the internal equipment response spectra. These internal building location spectra are arrived at by averaging the results obtained from four scaled earthquake records. The four earthquakes and their horizontal time-history records are the following:

a. N-S - El Centro, Calif., May 18, 1940

b. N-S - El Centro, Calif., December 30, 1934

c. S-80-W - Olympia, Wash., April 13, 1949

d. N-21-E - Taft, Calif., July 21, 1952.

Ground response spectra for a system with 2 percent of critical damping have been generated for each of the previous earthquake records.

In the generation of the ground spectra for each record, 60 periods from 0.1 sec to 1.0 sec were considered.

To determine what time duration of each record is required to ensure that maximum responses on the floor slab were obtained, response spectra were generated for each record using different time lengths of the records, all starting from zero time. The durations of the record required to give maximum responses in the period range of interest have been determined to be as follows:

a. 1940 El Centro - 7 sec

b. 1934 El Centro - 13 sec

c. 1949 Olympia - 20 sec

d. 1952 Taft - 10 sec.

Each earthquake record was scaled so that the area under the acceleration response spectra, obtained from the record duration previously indicated, between the periods 0.1 sec and 1.0 sec, equaled the area under the recommended OBE spectra between the corresponding periods for a 2 percent-damped system. The ground accelerations obtained by the previous scaling procedure, used to simulate the horizontal OBE, are as follows:

a. 1940 El Centro - 0.053g

b. 1934 El Centro - 0.078g

c. 1949 Olympia - 0.077g

d. 1952 Taft - 0.062g.

Response spectra from the earthquake records scaled to simulate the horizontal OBE were plotted over the recommended OBE spectra and are presented in Figures 3.7-4 through 3.7-7. The maximum ground accelerations for SSE were obtained by multiplying the previous values by two. The resulting ground response spectra for OBE and SSE are shown in Figures 3.7-2 and 3.7-3, respectively. The vertical components of the four previously described earthquakes were used to generate vertical spectra. Again the duration of each record required to give maximum responses in the period range of interest was determined. The durations of each record used in generating response spectra are as follows:

a. 1940 El Centro - 10.0 sec

b. 1934 El Centro - 10.8 sec

c. 1949 Olympia - 12.5 sec

d. 1952 Taft - 12.5 sec.

To determine the scaled vertical ground acceleration, a_{VS}, the following relationship has been used:

$$a_{VS} = a_{HS} * \frac{a_V * A_V}{a_H * A_H} \qquad (3.7\text{-}1)$$

where

a_{HS} = scaled horizontal ground acceleration

a_V = actual vertical ground acceleration

a_H = actual horizontal ground acceleration

A_V = actual area of vertical ground response spectrum (0.lg maximum, acceleration)

A_H = actual area of horizontal ground response spectrum (0.1g maximum, acceleration)

Maximum vertical ground accelerations used to generate vertical internal building spectra for OBE are as follows:

a. 1940 El Centro - 0.0204g

b. 1934 El Centro - 0.0240g

c. 1949 Olympia - 0.0256g

d. 1952 Taft - 0.0395g.

Maximum vertical ground accelerations for SSE were obtained by multiplying the previous values by a factor of two. The OBE vertical spectra are shown in Figure 3.7-8. The SSE vertical spectra are shown in Figure 3.7-9.

3.7.1.2.2 Supplementary Seismic Evaluation

In response to requests for information from the NRC Geosciences Branch, a site-specific ground response spectrum was developed, exhibiting a significantly higher ground response than the SSE ground response. Structures, systems, and components required for cold shutdown have been reevaluated for this higher site-specific earthquake, and the plant's capability to safely shut down has been confirmed. A detailed description of the evaluation program, analytical results, and conclusions can be found in the Supplementary Seismic Evaluation Report, Detroit Edison Report No. EF2-53,332 (Reference 1). Additional information on certain details of the analysis (provided in response to NRC questions) and the results of additional analyses performed subsequent to the Reference 1 report are listed in Reference 2.

Site-specific ground motion spectra were developed from real time-history data as previously described. Category I structures were then proven to adequately resist this excitation by means of a response spectrum evaluation equivalent in technique to that initially used for facility design purposes.

Internal equipment spectra were, however, generated from a synthesized ground motion time history, rather than the averaged real time histories used for original internal spectral generation purposes.

These supplementary evaluations reaffirmed the original facility aseismic design basis acceptability.

3.7.1.2.3 Response Spectra and Seismic Analysis Methods Used for Piping Snubber Reduction

OBE and SSE seismic loads for drywell piping snubber reduction purposes were analyzed using the following method:

> ASME Code Case N-411-1 damping values were applied with the uniform support motion response spectra analysis method in accordance with Regulatory Guide 1.84, Revision 27. Closely spaced modes were combined in accordance with Regulatory Guide 1.92.

This method included high-frequency modes per NUREG-1061, Volume 4, recommendations. The total combined response of high-frequency modes is combined by the Square Root of the Sum of the Squares (SRSS) method with the total combined response from lower-frequency modes to determine the overall structural peak response. When Code Case N-411-1 was used for earthquake loads, it was not mixed with Regulatory Guide 1.61 damping criteria for the same load case. Seismic inertia and anchor movement loads were combined by the SRSS method.

New building response spectra using Regulatory Guide 1.60 ground motion input were developed using the containment model used for the Cycle 3 fuel load. Direct generation of response spectra and Code Case N-397 spectra broadening techniques were not used in developing the new response spectra. The horizontal OBE spectra was anchored at 0.08g and the SSE spectra was anchored at 0.15g. Response spectra peaks were broadened by +15 percent.

These methods and spectra were applied in the seismic analysis of selected drywell piping systems in order to reduce the number of snubber supports.

3.7.1.3 Critical Damping Values

The damping values (expressed as a percentage of critical damping) of common structures and equipment in the Fermi 2 plant are listed in Table 3.7-2. The damping values used for the Fermi 2 project are in some cases higher than those specified in Regulatory Guide 1.61. These higher values were taken from Reference 3, prior to the issuance of Regulatory Guide 1.61. The damping values for HVAC systems, as delineated in Reference 19, were used for the CCHVAC System and SGTS duct and duct support revalidation effort. The specified systems and subsystems which were not listed were classified within one of the items in the table. Other damping values have been used when justified by specific data such as data obtained by testing.

3.7.1.4 Bases for Site-Dependent Analysis

No site-dependent analysis was necessary for Fermi 2 since the Fermi site is founded on bedrock (Subsection 3.7.1.6).

3.7.1.5 Soil-Supported Category I Structures

As described in Section 2.5 and Subsection 3.7.1.6, all Category I structures are supported directly on bedrock.

3.7.1.6 Soil-Structure Interaction

Structures Founded on Rock

Category I structures at Fermi 2 are founded on bedrock. A study was completed for Fermi 2 structures founded on rock (Reference 4) in which it was shown that for the Fermi site, soil-structure interaction was insignificant. The findings of this study are in agreement with the conclusions drawn by other researchers who report that soil-structure interaction is significant only when the shear wave velocity of the soil is less than 1000 fps (References 5 and 6). Since the shear wave velocity of the rock at the Fermi site is 7600 fps, it can be safely assumed, in accordance with the literature (References 5 and 6) and finite element analysis undertaken (Reference 4), that the Fermi 2 medium behaves as a rigid foundation. Therefore, the spectra developed for the bedrock represent the response to the base excitation.

3.7.2 Seismic System Analysis

3.7.2.1 Seismic Analysis Methods

3.7.2.1.1 General Description

The calculation of the dynamic response of a nuclear power plant complex subjected to an earthquake loading can generally be divided into two broad areas of analysis. The first is the analysis of major buildings and structures which house and/or support Category I systems and components. The second is the analysis of Category I systems and components. This subsection deals with the first area of analysis: seismic system analysis.

The necessity for division into two categories is that it is not practical to accomplish the analysis of major structures, systems, and components contained therein in a single dynamic analysis. The analysis is completed in steps. Major seismic systems, such as Category I structures, are modeled and analyzed. The motion of major structures, obtained from their analysis, is then used as the forcing function in the dynamic analysis of smaller Category I systems and components.

The classification of major buildings and structures, and Category I systems and components, is complicated by the fact that all systems and components that possess sufficient mass and stiffness to influence the dynamic behavior of major buildings and structures must be incorporated in the analysis of the major buildings and structures.

Seismic systems are defined as those systems in contact with the ground and thus are excited directly by the site response spectra, or the equivalent time-history motion. For each seismic system, there is a corresponding dynamic model. Seismic systems are discussed in this subsection and they include the reactor/ auxiliary building, residual heat removal (RHR) complex, buried piping, and buried electrical ducts. Subsystems are those in contact with or coupled to the seismic system and thus are excited by the response spectra derived from the system analysis. Subsystem analysis is discussed in Subsection 3.7.3, where the specific analyses for piping, reactor pressure vessel (RPV) and its internals, components, cable tray supports, cranes, racks, ventilation ducts, and tanks are contained in Subsections 3.7.3.6, 3.7.3.15, 3.7.3.16, 3.7.3.17, and 3.7.3.19.

The following criteria are used for system and subsystem decoupling.

a. If the mass of a component or equipment is less than 1 percent of the mass of its supporting structure, the component or equipment is treated as a subsystem and its mass may not be included in the system model

b. If the mass of a component or equipment is between 1 and 10 percent of the mass of its supporting structure, an approximate model of the component or equipment is included in the system model. Later, detailed subsystem analyses are made for this component or equipment

c. If the mass of a component or equipment is more than 10 percent of the mass of its supporting structures, a detailed model of the component or equipment is included in the system model.

3.7.2.1.2 Analysis of Building Structure Systems

To determine the exact dynamic forces acting on a structure, the accelerations (and, therefore, the displacements) of every mass particle must be evaluated. As any real structure's mass is distributed over the spatial extent of the structure, an infinite number of coordinates is required to describe the motion of every mass particle when the structure is subjected to a dynamic load. Calculation of time-dependent displacements at every point in a complex structure is impossible, but the analysis can be simplified by the judicious selection of a limited number of displacement components or coordinates. In dynamic structural analysis, two different assumptions are used to specify the deflected shape of a structure. These are referred to as the lumped-mass approach and the distributed-coordinate approach. The lumped-mass approach is the most convenient and versatile method to use in analyzing complex structural configurations found in a nuclear power plant. This approach was used in the seismic analysis of the Fermi 2 plant structures.

In the lumped-mass idealization, it is assumed that the entire mass of the structure is concentrated at a number of discrete points. A six-degree-of-freedom lumped mass would be general, in the sense that the discrete mass would possess all possible degrees of freedom. But in many structures, certain degrees of freedom may be neglected because the mass-stiffness configuration of the structure is such that these neglected degrees of freedom would not give rise to significant inertia forces if they were considered.

Recognition of the degrees of freedom in a structure that do not contribute to its dynamic response simplifies the modeling of the degrees of freedom that do contribute to the dynamic response of the structure. A series of computer programs, developed and validated for the analysis of nuclear power plant structures, are used to analyze Category I building structures. The criteria used in developing these programs are (1) consideration of the degrees of freedom encountered in the dynamic analysis of a nuclear power plant; (2) ease of inputting mass-stiffness properties from the structural drawings; and (3) ease of using output in structural design. The three programs used in the dynamic analysis of major structures are

a. Dynamic Seismic Analysis of Shear Structures (DSASS)

b. Matrix Analysis of Seismic Stress (MASS-IV)

c. Dynamic Analysis of Structures (DYNAS).

Each of these programs was used in the analysis of a specific type of structure. The program DYNAS can be used for the analysis of any type of structure, system, or equipment. All three programs use the modal method of analysis of a lumped-mass model, but the stiffness properties that interconnect the masses are read in the programs differently, because each program considers different degrees of freedom of the masses. The forcing function can either be acceleration spectra or a time-dependent base acceleration record. The descriptions of these programs are presented in Section 3.13.

The seismic motion of all Category I structures has been determined by applying the earthquake ground motions to appropriate dynamic models. In general, interaction between Category I and nonseismic structures has been eliminated by providing separate foundations for the structures. Also, rattlespace between abutting buildings has been provided so that seismic motion between buildings will be unimpeded.

Throughout the analysis of building structures, the coordinate directions are defined as the x, y, and z axes. The x and y axes denote the two principal horizontal directions and the z axis denotes the vertical direction.

3.7.2.1.2.1 Criteria Used in Modeling Techniques

Horizontal Analysis

The site response spectra presented in Figures 3.7-2 and 3.7-3 have been interpreted as one horizontal component of the OBE and the SSE, respectively.

These spectra are based on the free field vibratory accelerations, before plant structures are in place, at the elevation of the foundation of the structure being analyzed.

Action of the two horizontal components of the ground motion has been considered by analyzing the dynamic models for excitations parallel to the principal horizontal axes of the model. The model used is a discrete-lumped-mass, dynamic model having coupled modes; that is, a static force in one principal direction results in modal displacements in the other principal direction. For models in which the displacements of the two horizontal principal directions were statically coupled, analysis for excitations parallel to a model's two horizontal principal axes, has been accomplished by

a. The response spectra method of analysis (used for the design of structures modeled in system analyses), which involves

1. Analyzing the model for x-excitation

2. Analyzing the model for y-excitation

3. Combining the results of Steps 1 and 2 by the following equation:

$$\sigma_d = \sqrt{\sigma_{cx}^2 + \sigma_{cy}^2} \quad \text{square root of the sum of squares} \qquad (3.7\text{-}2)$$

where

σ_d = design seismic stress

σ_{cx} = stress component from x-direction seismic excitation

σ_{cy} = stress component from y-direction seismic excitation

b. The time-history method of analysis is used to generate response spectra for subsystem analyses by

1. Analyzing the model for x-excitation

2. Computing the average of the four x-excitations

3. Analyzing the model for y-excitation

4. Computing the average of the four y-excitation

5. Plotting the maximum of the average x-spectra and average y-spectra.

Site-specific internal building spectra were developed using a single synthesized time history for analysis. In this work, the spectra were thus generated directly and no averaging was necessary.

The horizontal dynamic analysis was performed using a shear structure system, a frame structure system, and a combined shear-frame structure system. A description of these analysis systems are as follows:

a. <u>Shear structure system</u> - The plant building structures are complex systems, asymmetric in plan, with heavy concrete slabs at the various floor elevations. These slabs are interconnected with numerous concrete shear walls and/or heavy cross-braced steel members. The overall height dimensions are smaller than the plan dimensions. This low height-to-plan ratio indicates that under lateral loads the predominant deformations of the long shear walls are shear deformations. Consequently, the relative rotations of the slabs about horizontal axes do not cause significant deformations; but, due to asymmetrical mass-stiffness distribution, rotation of the slabs about a vertical axis does occur when this type of structure is subjected to lateral loads. Since the predominant deformation of this type of structure under horizontal seismic loading is a horizontal shear deformation of the walls, it has been referred to as a shear structure system

Figure 3.7-10 shows a simplified shear structure system and the x-y-z axis system where the z-axis is vertical and the x- and y-axes are parallel to the principal axes of the structures. The significant deformations of the structure under horizontal seismic excitation are described with three coordinates, X, Y, and θ_z. These three degrees of freedom describe the motion of the concrete slab. Neglect of the θ_x, θ_y, and Z degrees of freedom implies that the slab mass moves in a horizontal plane

In describing the shear structure system model, the words "model slab" are substituted for the words "lumped-mass," because the mass of the actual structure was simulated in the model with virtually infinite rigid slabs located at the elevations of the major floor slabs and roof of the structure

The mass of the walls between two floors was lumped to the floors that they connect. The mass of equipment supported on slabs in the actual structure is included in the calculated mass of the virtually infinite rigid slabs. The actual slabs are considered to be infinitely rigid in their own planes. The rigid body motions of the model slabs consist of three degrees of freedom: horizontal translation in two perpendicular directions and rotation about a vertical axis. The model slabs are interconnected by weightless elastic springs that possess stiffness in the x- or y-direction and simulate the shear walls and vertical bracing in the structure. These springs are distributed horizontally on the model slabs so that the torsional stiffness interconnecting two slabs is approximated

Since the ends of the springs are considered to be horizontally distributed on the spatial extent of the model slabs, the model slabs are not point masses. Rather, they may be thought of as rigid bodies with horizontal dimensions only, because the mass of the actual structure has been considered to be lumped in the planes of the model slabs. This is the advantage of the slabspring model over the lumped-mass frame model

Three coordinates are required to describe the motion of each model slab. Therefore, three mass parameters are determined for each model slab. These mass parameters for the ith slab of the model are

1. M_{xi}, associated with x-translation
2. M_{yi}, associated with y-translation
3. $I_{\theta i}$, associated with the rotation about a vertical axis.

The mass parameters associated with x-translation and y-translation are the same and are equal to the mass of the slab. The mass polar moment of inertia, θ_z, is about a vertical axis through the centroid of the slab

To evaluate the stiffness of the structural components that interconnect slabs, the following assumptions are made.

1. All floor and roof slabs are considered rigid in their own planes; no point can displace another point relative to it on the same slab

2. Walls interconnecting slabs offer resistance only to relative displacement of slabs parallel to their line of action
3. The stiffness of small reinforced-concrete columns or walls and steel framing other than the braced bents is neglected, because their stiffness is small compared to the stiffness of shear walls.

When resisting lateral loads applied parallel to the long dimension, most walls act as short, deep beams; therefore, the contribution of shear to the deflection is considered in calculating the stiffness of a wall

The stiffness of steel framing which acts as springs was evaluated with conventional elastic frame or truss analysis computer programs such as EASE (Section 3.13)

Dynamic analysis of the shear structure systems was accomplished with the computer program DSASS. The input to DSASS is compiled by using the code, Spring Slab Analysis (SSANA). The centroid, total weight, and the weight moment of inertia about the vertical centroidal axis of the slabs; the spring constants; and the location of springs with respect to the slab centroid were calculated by SSANA. The description and analytical details of programs SSANA and DSASS are in Section 3.13.

b. Frame structure system - In the shear structure system, the motion of the structure's mass is restricted to a horizontal plane. For many structural systems under dynamic loading, motions are not restricted to a horizontal plane, and all six possible degrees of freedom of the discrete masses are required to describe the dynamic behavior of the structure. Dynamic analysis of this type of structure was accomplished by the program MASS-IV. This is a general frame program that can be used to analyze a plane frame, truss, grid, space truss, and space frame

c. Combined shear-frame structure system - The shear-type structures with three degrees of freedom for each slab mass and the frame-type structures with six degrees of freedom for each mass could both be present in a building system. The analysis of a coupled shear-frame structure was performed by DYNAS, which combines the features of DSASS and MASS-IV. Rigid or flexible frame members are used to connect the joints of the frame members to the slab centroids where interconnections exist.

Vertical Analysis

The dynamic behavior of a building in the vertical direction is a function of the wall axial stiffness, the floor system flexural stiffness, and the mass distribution. An examination of the vertical mass distribution of a building structure shows that there are mass concentrations at the floor elevations. A plane-frame model was developed to simulate the behavior of the building in the vertical direction.

Figure 3.7-11 shows an example of a plane frame, typical of that used to simulate a building's dynamic response in the vertical direction. The horizontal members in the model simulate the flexural stiffness of the floor systems. The lumped masses shown on the schematic simulate the mass of the building structure and the mass of equipment supported by the structure. Although only two wall systems are shown in Figure 3.7-11, any number of wall systems can be incorporated in an analysis. The number of wall systems depends upon the layout of the structure to be analyzed. The mass distribution in the model consisting of the actual structure's mass concentrated at floor slab elevation is distributed between the walls and the horizontal members. This mass distribution is used because part of the actual structure's mass moves with the walls, whereas part of the mass motion is amplified because of slab flexibility. The flexural stiffness of the horizontal members is adjusted to represent the stiffness of the actual floor systems. Since a floor system consists of slabs of various thicknesses, beams, and openings, and since it is supported by interior and exterior walls, many periods of vibration occur in the floor system at a single elevation when a building structure is subjected to vertical seismic loading. Therefore, the periods of vibration of a floor system cannot be simulated by a single horizontal member in a frame model. For this reason, a multimember-mass system has been used to simulate a complex floor system (see upper level on Figure 3.7-11). Each member-mass system is adjusted to have frequency characteristics matching one of the calculated frequencies of the actual slab system. The analysis of the vertical model is performed by the MASS-IV program.

Response spectra are generated at each mass of the system used to represent a slab. These spectra are plotted on a single plot and enveloped with a smooth curve. The floor slabs were designed by using seismic coefficients obtained from the rigid end (frequency response greater than 33 Hz) of the resulting spectra. Vertical seismic stresses in the building walls were obtained from the vertical members of the vertical models. Equipment supported on a slab was designed using the resulting spectra as the vertical seismic load. Equipment located near walls was designed using response spectra generated on masses located on the vertical members.

3.7.2.1.2.2 Description of Mathematical Models

Horizontal Seismic Analysis

The massive stiff floor slab-shear wall configurations of the reactor/auxiliary building (Figures 3.7-12 through 3.7-14) and the RHR complex are modeled as a slab-spring system. The slabs, treated as infinitely rigid in their own planes, are interconnected by weightless linear elastic springs used to simulate the stiffness of shear walls within the structural system, as described in Subsection 3.7.2.1.2.1.

Rotations about the horizontal axes could be significant in the reactor containment portion of the reactor/auxiliary building. Since these degrees of freedom, θ_x and θ_y, cannot be modeled with the slab model, a conventional three-dimensional frame analytical model is used to model the containment shield, the containment vessel, the RPV and internals, the reactor support pedestal, and the biological shield. The lumped masses in this portion of the model are allowed X, Y, θ_x, θ_y, and θ_z degrees of freedom, and are interconnected with frame members. The slab model and the frame model are connected by axial springs at various elevations to represent the behavior of the actual structure more accurately. The configuration of this model is shown in Figure 3.7-15 except for the model of the RPV and its internals, which is shown in Figure 3.7-16. The RPV is supported by the reactor pedestal at Mass 29 and laterally supported at Masses 26 and 32 by the refueling bellows and stabilizer, respectively. The seismic methods and analysis procedures for the RPV and its internals are described in Subsection 3.7.3.15.

The reactor building crane bridge and associated steel structures (between the fifth floor and the roof) have been simulated in the horizontal dynamic model as shown in Figure 3.7-17. The model is based on the assumption that the crane would be parked at the end bay during a seismic event. The vertical lines represent steel columns connected by rigid members at the bottom and top ends to the mass centroids of Slabs 5 and 6, respectively. (NOTE: Subsequent to the original analysis, an analysis (Reference 22) was performed which qualifies the crane girder steel superstructure interior support columns for the crane deadload plus rated load combined with either wind or seismic loads. The additional analysis assumes that the overhead crane is located anywhere along the crane's travel path to maximize the member stresses.)

The mass parameters of the reactor/auxiliary building slabs in the dynamic model are presented in Tables 3.7-3 and 3.7-4. These mass properties are calculated by considering the mass of the actual structure concentrated at the slab elevations, and distributed laterally on virtual infinitely rigid slabs in accordance with the lateral mass distribution in the structure. Therefore, each model slab represents the concrete of, and equipment on, actual slabs and the tributary mass of equipment structure between slabs. Both the translational and rotational inertia of the actual structure are taken into consideration.

The lumped masses in the frame part of the reactor/auxiliary building model are calculated from the physical properties of the containment and reactor support system, and are also presented in Table 3.7-4. However, the properties of the masses for the RPV are not included in this table.

The stiffness elements that interconnect the lumped masses of the dynamic models are shown schematically in Figure 3.7-15 for the reactor/auxiliary building and Figure 3.7-18 for the RHR complex. The solid vertical lines interconnecting slabs represent groups of linear elastic springs that simulate the stiffness of the walls in the structural complex. The walls in the building complex that are considered to act as springs are shown in Figures 3.7-19 through 3.7-24 for the reactor/auxiliary building, and Figures 3.7-25 through 3.7-27 for the RHR complex with walls parallel to the X-axis treated as X-springs and walls parallel to the Y-axis treated as Y-springs.

Each wall or group of walls considered to act as a spring in this analysis is assigned a six-digit identification number which is shown on the figures. For any identification number that does not have six digits, leading zeros are implied. The digits of the identification number convey the following information:

a. <u>First two digits</u> - slab number that the lower end of the spring is connected to

b. <u>Second two digits</u> - slab number that the upper end of the spring is connected to

c. <u>Third two digits</u> - ith spring with its lower end connected to the slab given by the lst two digits (if the 3rd two digits form an even number, the wall is a Y-spring and if these two digits form an odd number, the wall is an X-spring).

Frame members in the reactor containment portion of the model are represented on Figure 3.7-15 with dashed lines. The properties of these members are calculated from the physical properties of the primary containment, the reactor support pedestal, and the biological shield. Table 3.7-5 presents the properties and the topography of the frame members of the reactor/auxiliary building model, except for the members of the RPV part of the frame model. To eliminate assigning artificial horizontal distances between the centroids of masses in the containment and pedestal-shield cantilevers, the stiffness of the connections represented by horizontal dotted lines in Figure 3.7-15 is given in Table 3.7-6 as stiffness coefficients.

To evaluate the stiffness of the structural components that interconnect the masses of the shear models shown in Figures 3.7-15 and 3.7-18, the following assumptions have been made:

a. All points on the same slab translate in the horizontal plane passing through the mass-center of the slab and the slab rotates only about the vertical axis

b. The walls offer resistance to relative displacements between slabs only in their longitudinal direction.

When resisting lateral loads applied parallel to the long dimension, most walls act as short, deep beams; therefore, the contributions of both flexure and shear to the deflection must be considered in calculating the stiffness of a wall. The stiffness of an individual wall was calculated by the following formula:

$$K = \frac{1}{\Delta} \qquad (3.7\text{-}3)$$

$$\Delta = \frac{1.2h}{GA} + \frac{h^3}{12EI} \qquad (3.7\text{-}4)$$

where

h	=	height of wall
I	=	moment of inertia of wall for bending about centroidal axis perpendicular to length of wall
A	=	cross-sectional area of wall
E	=	elastic modulus of concrete
G	=	shear modulus of concrete
K	=	stiffness
Δ	=	deflection of wall due to a unit force

Vertical Seismic Analyses

No attempt was made to set up a three-dimensional model on account of the excessive number of degrees of freedom. The vertical dynamic model of the building was developed on the basis that the amplification in the vertical direction is a function of the axial stiffness of the walls and bending stiffness of the beam-slab system.

The vertical stiffness is due mainly to two structural systems in each model. They are

a. Reactor/auxiliary building model

   1. The reactor containment shield (right side of Figure 3.7-28)

   2. Reactor/auxiliary building walls (left side of Figure 3.7-28).

b. RHR complex model

1. The cooling tower walls (right side of Figure 3.7-29)
2. The RHR building wall (left side of Figure 3.7-29).

The two wall systems are connected by the reactor building floor slab at all the floor elevations for the reactor building model and at Elevation 617 ft for the RHR complex model. The auxiliary building floor slab is represented by a single-degree-of-freedom system connected to the joints of the reactor/auxiliary building wall system at each elevation.

In the dynamic model formulated for these analyses, the masses can displace, relative to one another, with one degree of freedom in the vertical direction. The mass parameters are calculated in the following manner:

a. The masses are concentrated at joints (as shown in Figures 3.7-28 and 3.7-29) and interconnected by weightless linear springs that simulate the stiffness of the slabs or walls
b. In general, the wall masses are lumped equally to the nearest joints
c. For the slabs, it has been assumed that one-third of the total slab mass is effective; the remaining mass of the slab was lumped with the wall mass at that elevation
d. The mass of the reactor containment shield includes only the mass of concrete and contributory slab mass.

The determination of the spring stiffness and the modification of the original model, to simulate higher mode contribution of slabs, is described as follows:

a. <u>Wall springs</u> - For the wall system, the effective area is the sum of the areas of all the individual walls at a particular level. The walls that are connected monolithically with the top and bottom slab only, provide vertical stiffness. For the reactor containment shield, the effective area is that of a circular ring. In cases where the radius changes between two mass points, the average area is used
b. <u>Slab springs</u>
    1. <u>Slab natural frequency</u> - The stiffness of a member representing a slab in the vertical model simulates the lowest natural frequency of the slab. As a grid model is analyzed to determine the lowest natural frequency of a typical slab, the frequencies of other floors are determined based on the grid analysis and standard formulae
    2. <u>Modification of the model</u> - To determine the response spectrum of the slab at a particular level, the model is modified at that level to include the multi-degree behavior of the slab system. The stiffness and mass properties of slabs at other levels are not changed and correspond to the lowest fundamental frequency of the slab at that level. The modified vertical model for determining slab response spectrum at Elevation 684 ft 6 in. of the reactor /auxiliary building is shown in Figure 3.7-30. The slab system at this level consists of six masses and the springs on each side are connected to the same wall joint. The total effective slab mass is

divided by the number of masses and is assigned to each individual mass. In this case, Mass l simulates the stiffness of the lowest natural frequency of the slab as calculated before, and the rest of the masses were assigned frequencies higher than the calculated natural frequency at regular intervals. The highest frequency assigned, 30 Hz (Reference 25), has negligible amplification. Similarly, the stiffness parameters of the auxiliary building slab system are determined. The model for Elevation 6l3 ft 6 in. of the reactor/ auxiliary building is shown in Figure 3.7-31. The model for the RHR complex is shown in Figure 3.7-29.

3.7.2.1.2.3 Analysis of Mathematical Models for Structures

To determine the free vibrational characteristics of the dynamic models, the model equation for a multi-degree lumped-mass system may be written as

$$[\mathrm{M}]\{\ddot{\mathrm{x}}\} + [\mathrm{K}]\{\mathrm{x}\} = 0 \qquad (3.7\text{-}5)$$

where

[M]	=	mass matrix
[K]	=	stiffness matrix
$\{x\}$, $\{\ddot{x}\}$	=	displacement, acceleration vectors

where the mode shapes and frequencies are solved in accordance with

$$[\mathrm{K} - \omega_n^2 \mathrm{M}]\{\phi_n\} = 0 \qquad (3.7\text{-}6)$$

This set of equations has as eigenvalues the squares of the circular natural frequencies, ω_n. Associated with each frequency is a mode shape ϕ_n which may be arranged as one of the columns of the matrix $[\phi]$.

The modal participation factors are given by

$$[\Gamma]\, \mathrm{n_i} = \frac{[\phi]^{\mathrm{T}}[\mathrm{M}]\,[\mathrm{D}]_{\mathrm{i}}}{[\phi]^{\mathrm{T}}\,[\mathrm{M}]\,[\phi]} \qquad (3.7\text{-}7)$$

where

$[\Gamma]n_i$	=	participation factor
$[\phi]^T$	=	transpose of mode shape vector for nth mode
$[D]_i$	=	earthquake direction vector referring to direction i

The response of the system in one mode, A_i, is given by

$$A_{\mathrm{i}} = \begin{Bmatrix} \mathrm{a_i^1} \\ \mathrm{a_i^2} \\ \vdots \\ \mathrm{a_i^n} \end{Bmatrix} = \mathrm{T_i}\,\{\phi_{\mathrm{i}}\}\mathrm{R_i} \qquad (3.7\text{-}8)$$

where

$\{\phi_i\}$ = one column of the matrix $[\phi]_n$ corresponding to the mode

T_i = corresponding element in column matrix of $[T_i]$

R_i = response of a single-degree-of-freedom system of period T_i and damping ratio B_i from specified ground response spectrum for the site

At any mass coordinate in the system, the total response, $\frac{k}{a}$, is given by

$$\frac{k}{a} = \sqrt{\sum_{i=1}^{n}\left(a_i^k\right)^2} \qquad (3.7\text{-}9)$$

where

a_i^k = response at coordinate k in ith mode

n = number of modes

Floor slab-shear wall type structures are modeled as a slab spring system in which the mass of the structure is lumped at floor slab elevations. The weight of each shear wall is lumped equally between the floor slab above and the floor slab below. The containment vessel, the concrete shield wall, the sacrificial shield, and the reactor support pedestal were modeled with a sufficient number of lumped masses so that all the modes up to 33 Hz can be extracted. The number of lumped masses for these elements was at least three more than the number of modes below 33 Hz as determined from closed form solutions.

Dynamic analysis has been carried out to include all the significant response modes.

The applicable stress/deformation criteria are described in Subsection 3.8.4.

3.7.2.1.2.4 Basis for Computing Combined Responses

In the original design performed in 1971, horizontal and vertical seismic effects were not combined in the structural design. In subsequent analyses, the effects of two statistically independent time histories were added algebraically and then combined with the vertical component effect by the SRSS rules.

3.7.2.1.3 Buried Electrical Ducts

3.7.2.1.3.1 General

There are two sets of Category I ductbanks between the RHR complex and the Reactor/Auxiliary building, with a Division I and Division II ductbank in each set. The first set of ductbanks was installed during plant construction. The analysis of these Category I buried electrical ducts is described in the following paragraphs.

The forces in the duct due to wave propagation in soil and rock are determined (Reference 7). The duct design is required to take into account the relative seismic displacements at its anchor points with the building, in addition to the strains induced due to wave propagation in the surrounding soil. The anchorage of the duct with the building and manhole is designed to be flexible such that 1 in. maximum displacement in any direction is allowed for analytical purposes. Thus, if the relative displacement of the buildings and duct at the anchorage is less than the maximum allowable displacement of 1 in., the flexural strains are only due to wave propagation.

The analytical procedure used to evaluate the seismic influence on buried electrical ducts considers the soil condition at the plant site. The method of analysis follows the method of Reference 8.

The second set of Category I 4160-V electrical ductbanks run between the RHR cable vaults and the Reactor/Auxiliary building cable vaults. These ductbanks (including the cable vaults and manholes) are designed as Seismic Category I components. The buried portion of the ductbank is designed for seismic response effects utilizing the approach identified in ASCE 4-98, as endorsed by NUREG-0800 Standard Review Plan, section 3.7.3 Revision 3 (March 2007). This approach is an extension of the Fermi 2 methodology used for the first set of Category I ductbanks. This extended methodology provides an accurate analytical means for the prediction of the seismic responses of buried structural components, specifically at bends and other geometric discontinuities, precluding the need for physical measures (such as loosely compacted sand) at these discontinuities. The ductbanks connecting to rigid structural components, such as the manholes and the transition vaults at the RHR and Auxiliary buildings, are provided with one inch physical gaps, similar to that for the first set of Category I ductbanks, to preclude locked in stresses due to potential differential displacements during and after postulated seismic events.

3.7.2.1.3.2 Analysis

The design of the ducts ascertains that the stresses caused by the strains do not exceed the acceptable safe limits in the event of an SSE.

The maximum axial strain in the straight portion of a duct has an upper bound equal to the maximum strain in the surrounding soil in the direction of the duct. If the wave length is much larger than the straight portion of the duct, the maximum strain in the duct is assumed to be uniform along the duct run. However, in cases where the duct is very long, the duct displaces relative to the surrounding soil because of strain incompatibility between the soil and the duct. The relative displacement between the soil and the end of the duct is determined by deducting frictional restraint to the movement of the duct from the upper-bound soil displacement in the direction of the duct.

The effect of axial displacement of a straight portion of duct relative to the soil, at bends and at juncture points, is evaluated by the "beam on elastic foundation" concept. To obtain forces in the bend, each bend is subjected to the relative displacement as obtained previously. A fill of well-graded, loosely compacted sand is provided on either side of the bend to avoid concentration of forces around the bend due to stiff subgrade, and to distribute the subgrade stresses uniformly.

The design of the new Category I 4160-V electrical ductbanks that run between the RHR cable vaults and the Reactor/Auxiliary building cable vaults uses a more conservative approach at bends and elbows. In the analysis, the elbow is treated as an inflexible structure, whereas longitudinal and traverse legs are treated as flexible structures, as outlined in the ASCE Report “Seismic Response of Buried Pipes and Structural Components” (1983) (Referenced in NUREG-0800, Standard Review Plan 3.5.3). Therefore, loosely compacted sand fill is not required on either side of the ductbanks bends.

The concrete structures of the new Category I 4160-V electrical ductbanks were analyzed using the guidance in accordance with ACI 349-01 “Code Requirements for Nuclear Safety

Related Concrete Structures" and Regulatory Guide 1.142 "Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)".

For both the new and existing ductbanks, effects of flexural strains were evaluated at points of maximum possible curvature between the duct attachment points. As commonly observed, the strains associated with such effects were found to be negligibly small from the practical design standpoint.

The corresponding effects of flexural strains were similarly evaluated by means of calculation from the maximum possible curvature between the duct attachment points. As commonly observed, the strains associated with such effects were found to be negligibly small from the practical design standpoint.

3.7.2.1.4 Seismic Design of Category I Buried Piping

The seismic analysis of buried piping located between the RHR complex and the reactor/auxiliary building is performed in exactly the same way as the seismic analysis of buried electrical ducts described in Subsection 3.7.2.1.3. The stresses/strains do not exceed the acceptable safe limits in the event of an SSE.

During the course of safety evaluation review, at the request of the NRC, additional information on this was submitted. Included was Reference 9, which discussed lateral pressure and the analysis of buried piping, and which forwarded a February 3, 1970, D&M report, Reference 10. Also included were References 11 and 12, which added to the information provided by References 9 and 10.

The Category I structures and buried pipes and conduits have been structurally reassessed for the effect of the SSE.

In reference to LOCA stresses, the pertinent information on original load combinations and respective stress components, including those resulting from a LOCA, have been presented in Chapter 4 and in appendixes of the reassessment report.

The three components of the earthquake have been considered in the reassessment report. Two horizontal components have been considered to be acting simultaneously, and the vertical component has been added as an absolute sum or square root of the sum of the squares as appropriate.

A damping value of 7 percent has been used in accordance with Regulatory Guide 1.61, since the Category I structures are of reinforced-concrete construction, and the structural elements are highly stressed for the site-specific earthquake loading.

3.7.2.2 Natural Frequencies and Response Loads

The analysis of the models developed in Subsection 3.7.2.1 yields the natural frequencies, mode shapes, and modal responses of the overall system. These results are presented for both the horizontal and vertical analysis.

3.7.2.2.1 Reactor/Auxiliary Building

3.7.2.2.1.1 Horizontal Analysis

a. Frequencies and mode shapes - The periods, mode shapes, and dynamic response of the lumped-mass system are computed with the use of DYNAS. Table 3.7-7 presents the summary of the first 20 modal periods, the modal participation factors for X-direction base excitation, and the modal participation factors for Y-direction base excitation

b. Response spectrum - The program DYNAS was used to perform the time-history analysis of the dynamic model, damped with 2 percent and 5 percent of critical damping for the OBE and SSE, respectively, and to generate response spectra at selected mass centroids for E-W and N-S base excitations. Newmark's β-method of numerical integration for a linear system with time-dependent input base motion, combined with modal superposition, was used to obtain the motions of the lumped masses. The time-histories of the mass motions were not printed out of computer storage because of the large quantity of data, but rather response spectra generating subroutine used the stored slab motions to generate response spectra for specified masses

   Separate spectra curves are not plotted for the N-S excitation and the E-W excitation; rather, at each spectra period for a given spectra damping, the average response from the four N-S excitations from Subsection 3.7.1, and the average response from the four E-W excitations from Section 3.7.1 were calculated, and the maximum of the averages was plotted. The plotted spectra curves, with their valleys and peaks, were smoothed by enveloping the peaks with the envelope at a peak extending ten percent, on the period scale, to either side of the peak. The resulting smooth curves are presented in Figures 3.7-32 through 3.7-55 for OBE and Figure 3.7-56 through 3.7-79 for SSE

c. Displacement response - Table 3.7-8 summarizes the probable displacements obtained from this analysis.

3.7.2.2.1.2 Vertical Analysis

a. Frequencies and mode shapes - The vertical model shown in Figures 3.7-28, 3.7-30, and 3.7-31 has been analyzed by the MASS-IV program. Table 3.7-9 lists the periods and participation factors for 24 modes for the model shown in Figure 3.7-30. The variation of main structural period in models shown in Figures 3.7-28, 3.7-30, and 3.7-31 is negligible

   The vertical analysis was used to generate response spectra for the design of Category I equipment located at different floor levels. The forces in the structure are also determined by the response spectra method.

   The slabs and shear walls of the reactor building and the reactor containment are designed to withstand these forces due to vertical excitation

b. Response spectrum - A computer program, MASS-IV, was used to analyze the vertical models and generate vertical response spectra.

Response spectra were generated at two elevations (Elevation 613 ft 6 in. and 684 ft 6 in.) at the reactor containment shield, reactor/auxiliary building wall, reactor building slab, and auxiliary building slab for OBE (2 percent structural damping) and SSE (5 percent structural damping). The spectra at other elevations were not generated, but were classified in one of the two levels. At each period considered in the spectra generation process, the average response from the four earthquakes was calculated. These averages were plotted. The rough curves were smoothed by enveloping the peaks and extending 20 percent to either side of a peak. The vertical response spectra are presented in Figures 3.7-80 through 3.7-88 for OBE and in Figures 3.7-89 through 3.7-97 for SSE.

3.7.2.2.2 Residual Heat Removal Complex

3.7.2.2.2.1 Horizontal Analysis

a. Frequencies and mode shapes - The periods and mode shapes and dynamic responses of the lumped-mass system are computed by the use of DYNAS. A summary of the model periods and modal participation factors for the x and y excitations is presented in Reference 13

b. Response spectrum - The program DYNAS was used to perform the time-history analysis of the dynamic model, damped with 2 percent of critical damping, and generate response spectra at selected mass centroids for E-W and N-S base excitations. Newmark's β-method of numerical integration for a linear system with time-dependent input base motion, combined with modal superposition, was used to obtain the motions of the lumped masses. The time-histories of the mass motions were not printed out of computer storage because of the large quantity of data, but rather response spectra generating subroutine used the stored slab motions to generate response spectra for specified masses

   Separate spectra curves were plotted for the N-S excitation and the E-W excitation; at each spectra period for a given spectra damping, the average response from the four N-S excitations and the average response from the four E-W excitations were calculated

   The plotted spectra curves, with their valleys and peaks, were smoothed by enveloping the peaks with the envelope at a peak extending 10 percent, on the period scale, to either side of the peak's period

   The representative resulting smooth curves are presented in Figures 3.7-98 through 3.7-101 for OBE and in Figures 3.7-102 through 3.7-105 for SSE.

3.7.2.2.2.2 Vertical Analysis

a. Frequencies and mode shape - The periods, mode shapes, and dynamic response of the lumped-mass system are computed by the use of MASS-IV. A summary of the modal periods and modal participation factors for the x and y excitations is presented in Reference 13

b. Response spectrum - MASS-IV was used to analyze the vertical models and to generate vertical response

Response spectra were generated at two elevations (Elevation 590 ft, 0 in. and Elevation 617 ft, 0 in.) at the building walls and the building slabs for OBE (2 percent structural damping) and SSE (5 percent structural damping)

At each period considered in the spectra generation process, the average response from the four earthquakes was calculated. These averages were plotted. The rough curves were smoothed by enveloping the peaks with a smooth curve, which, as the period scale, extends 20 percent to either side of a peak. The representative vertical response spectra are presented in Figures 3.7-106 through 3.7-110 for OBE and in Figures 3.7-111 through 3.7-115 for SSE.

3.7.2.3 Procedures Used To Lump Masses

For dynamic analysis, Category I equipment was represented by lumped-mass systems that consist of discrete masses connected by weightless springs. The criteria used to lump masses were

a. Because the number of modes of a dynamic system is controlled by the number of masses used, the number of masses was chosen so that all significant modes are included

b. Mass was lumped at any point where a significant concentrated weight is located. Examples are the motor in the analysis of pump motor stand, and the propeller in the analysis of pump shaft

c. If the equipment has a free end overhang span whose flexibility is significant compared to the center span, a mass was lumped at the overhang span

d. When a mass was lumped between two supports, it was located at a point where the maximum displacement was expected to occur. This tends to conservatively lower the natural frequencies of the equipment. Similarly, in the case of live loads (mobile) and a variable support stiffness, the location of the load and the magnitude of support stiffness were chosen so as to yield the lowest frequency content for the system. This is to ensure conservative dynamic loads since equipment frequencies are such that the floor spectra peak is in the lower frequency range.

Slab masses were lumped in accordance with the procedures described in Subsection 3.7.2.1.2.1.

3.7.2.4 Rocking and Translational Response Summary

The site response spectra developed for Fermi 2 are the bedrock spectra. Since the Fermi 2 Category I structures are founded directly on bedrock, the rocking and translational effect due to soil structure interaction is not applicable to this location. See Subsection 3.7.1.6 for a description of the studies that document that the Fermi 2 site behaves as a rigid foundation.

3.7.2.5 Method Used To Couple Soil With Seismic System Structures

Fermi 2 Category I structures are all founded on bedrock and do not require an evaluation of soil structure interaction (Subsection 3.7.1.6).

3.7.2.6 Development of Floor Response Spectra

3.7.2.6.1 Introduction

If a structure is subjected to an earthquake, the base of a subsystem (or equipment) mounted on a floor slab or wall experiences the motion of the slab or wall. This motion may be significantly different from the input motion at the base of the structure. Therefore, the response spectra used in the analysis of the structure are not directly applicable to the analysis of subsystems mounted in the structure unless the subsystem element is modeled in the dynamic model of the structure. Also, unless the subsystem element is a rigid mass, rigidly connected to the slab or wall, the motion of the subsystem is different from the motion of the slab or wall, because the subsystem element is a flexible elastic system that responds dynamically to the motion of the slab. For these reasons, the motion experienced by a subsystem is the structure's base excitation modified as a function of the structure's characteristics, the subsystem's characteristics, and the mode of attachment to the structure.

To establish explicit slab or wall motions, applicable to development of subsystem design criteria, time-history forcing functions were used to excite the building models used in the system analysis. Resulting time-history slab or wall motions were used to generate response spectra for the analysis of subsystems supported in the building.

3.7.2.6.2 Horizontal Response Spectra

The seismic models used to generate the response spectra at various building elevations are described in Subsection 3.7.2.1.2. The base input forcing functions are described in Subsection 3.7.1.2 and shown in Figure 3.7-2 for OBE, and 3.7-3 for SSE. Site-specific analyses were completed using statistically independent synthesized time histories in orthogonal directions. The response spectrum evaluations for the reactor/auxiliary building and the RHR complex are described in Sutbsection 3.7.2.2.2. Representative reactor/auxiliary building horizontal response spectra for selected building elevations are shown in Figures 3.7-32 through 3.7-55 for OBE and Figures 3.7-56 through 3.7-79 for SSE. Representative RHR complex response spectra are shown in Figures 3.7-98 through 3.7-101, and 3.7-102 through 3.7-105 for OBE and SSE, respectively. The spectra ensemble defining the facility aseismic design bases is described in Subsection 3.7.2.2.1.1.

3.7.2.6.3 Vertical Response Spectra

The scaled time-history forcing functions for the vertical direction (Subsection 3.7.1.2) were used to perform time-history analyses of the vertical seismic models described in Subsection 3.7.2.l.2.2. A single synthesized time history was used for the site-specific evaluation.

The procedure for determining subsystem response spectra in the vertical direction is the same as for the horizontal direction, as described in Subsections 3.7.2.2.1.2 and 3.7.2.2.2.2.

In this case, response spectra were generated for uncoupled time-history motion in the vertical or z direction.

The resulting reactor/auxiliary building vertical response spectra for selected building elevations are shown in Figures 3.7-80 through 3.7-88 for OBE and in Figures 3.7-89 through 3.7-97 for SSE. The RHR complex response spectra are shown in Figures 3.7-106 through 3.7-110 and 3.7-111 through 3.7-115 for OBE and SSE, respectively.

3.7.2.7 Differential Seismic Movement of Interconnected Components

The effects of differential movements of interconnected components due to seismic disturbance were considered in the seismic analysis of the piping systems and components where they contribute significantly to the overall response (Subsection 3.7.3.6).

All means and mechanisms of interconnection are designed to limit the applicable stress and deformation to within the ASME Section III Code Allowable Limits.

3.7.2.8 Effects of Variations on Floor Response Spectra

The increase in peak width, to account for variations in structural properties and damping, is described in Subsection 3.7.2.2. Variations in material properties are described in Subsections 3.7.2.15.2 and 3.7.2.15.3.

3.7.2.9 Use of Constant Load Factors

Vertical seismic system multi-mass dynamic models were used to obtain vertical response loads for the seismic design of Category I structures, systems, and components (Subsection 3.7.2.1). A constant load factor was used only when it was established that the structure, system, and/or component under consideration was rigid.

3.7.2.10 Method Used To Account for Torsional Effects

Category I structures may have natural torsional modes of vibration due to eccentricities between the centers of rigidity and centers of mass of the structural elements. As described in Subsection 3.7.2.1.2.2, the torsional response was accounted for by interconnecting the slab with weightless resisting elements, parallel to the x and y axes, distributed on the slabs as the shear walls are distributed in the structure.

3.7.2.11 Comparison of Responses

The forces obtained from the response spectrum method of analysis were used in the design of structural components of the building. The floor response spectra were generated by time-history analyses (Figures 3.7-32 through 3.7-115). Comparisons of accelerations were made at various elevations in the building to ensure that the floor response spectrum was obtained from a seismic load equivalent to or greater than the seismic load specified by the site response spectra.

3.7.2.12 Methods for Seismic Analysis of Earth Structures

The design of Fermi 2 does not include Category I earth structures except the shore barrier which was designed to meet Category I requirements. The shore barrier was analyzed using the computer code ICES-SLOPE. Details on slope stability analysis of the shore barrier were provided to the NRC staff during their safety evaluation review (see Subsection 3.4.4.5 and References 14 through 17).

3.7.2.13 Methods To Determine Category I Structures/Overturning Moments

The overturning moments induced by seismic excitation were computed by applying the inertia forces determined in Subsection 3.7.2.1, with vertical inertia forces taken upward, reducing the structure's effective weight. The inertia force on each mass was determined by computing the square-root-of-sum-of-squares of the modal acceleration contributions for that mass. Tensile base reactions were not allowed.

3.7.2.14 Analysis Procedure for Damping

Structural damping is energy loss due to internal friction within the structural material and at connections. The damping force is a function of the intensity of motion and the stress levels induced in the system. Damping is also highly dependent on the makeup of the structural system and the energy absorption mechanisms within the system. Considerable energy is also absorbed at cracked surfaces when the elements on each side of the crack can move relative to one another. In the linear dynamic analysis, the procedure used to account properly for the previous damping in different elements of a coupled system model was as follows:

a. The structural damping of the various elements of the model was first specified. These values are referred to as the damping ratios (Bi) of the various components making up the complete systems

b. A modal analysis of the linear system model was performed. This results in a modal column matrix (ψ) normalized such that $\psi T\ M\psi = I$; where M is the mass matrix, I is the identity matrix, and ψT is the transpose of ψ

c. Using the kinetic energy of the individual components as a weighting function, the following equation was used to obtain a suitable damping ratio (Bi) for the ith mode.

$$B_i = \psi^T \left[B_j\right] M\psi \qquad (3.7\text{-}10)$$

The diagonal terms of this matrix product are the modal damping ratios (B_i) of the coupled system. The damping ratios (B_j) of the individual substructures making up the complete system under investigation were used as input to [B_j] in order to calculate B_i.

3.7.2.15 Miscellaneous Considerations

3.7.2.15.1 Parametric Study

To evaluate the effects of the variation in mass-stiffness parameters on the seismic response of the building systems analyzed, several cases were studied by varying the original stiffness properties, or original mass properties, or both.

3.7.2.15.2 Structural Material Parameter

The modulus of elasticity, e_c, for concrete is taken as

$$e_c = (W^{1.5})33\sqrt{f_c'} \qquad (3.7\text{-}11)$$

where

W = density of concrete, lb/ft^3

f_c' = specified compressive strength of concrete, $lb/in.^2$

The modulus of elasticity of nonprestressed steel reinforcement and steel structures is taken as 29×10^6 lb/in^2.

3.7.2.15.3 Interconnecting Category I and Other Structures

No Category I and nonseismic structures are integrally connected. The nonseismic structure is provided with sufficient seismic rattlespace or a flexible boundary layer to ensure that there is no effect on the adjacent Category I structure.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Determination of Number of Earthquake Cycles

Seismic loading cycles were considered for those Category I systems requiring fatigue analysis by applicable codes. The number of seismic cycles at maximum load per seismic event used on the various Category I systems and components varies from 5 to 250 cycles, depending on the component's natural frequency. In addition, the magnitude of the cyclic load varies with each component. The stated number of loading cycles was determined by actually performing a time-history analysis of reactor systems subjected to the full durations of the El Centro, Taft, and Olympia earthquakes. The number of cycles selected was always conservative with respect to the usage factor; an example of these cycles is presented as follows.

a. For components - ASME Section III NB-3650 requires that a number of earthquake cycles used in the analysis of ASME III Code components be specified as part of the design mechanical loads. The following criteria were used for all equipment within the jurisdiction of this code:

    1. A total of two OBEs and one SSE was assumed during the lifetime of the plant

    2. For conservative component design, structures were assumed to cycle (full sign reversal) 20 times per earthquake

3. Systems and components classified as relatively rigid with respect to local structural response will "ride" with the structure and are thus assigned 20 stress cycles per earthquake

4. If the system and/or component is relatively flexible (fundamental frequency equal to or less than 50 percent of structural fundamental frequency), a 20-cycle criterion governs.

b. For piping systems - The dynamic analysis using the floor response spectra as input motion performs an actual cycle count of the first mode vibration. The duration of the earthquake is taken as 10 sec and the result is adjusted to consider reduced stress range cycles. The number of effective cycles determined varies around 200. For valves and many other mechanical items, a conservative number of 250 is used.

For the GE-supplied equipment, the reactor/auxiliary building dynamic model was excited by the same time-histories as previously specified. The modal response was truncated such that the response of three different frequency bandwidths could be studied: 0-10 Hz, 10-20 Hz, and 20-50 Hz. This was done to give a good approximation to the cyclic behavior expected from structures with different frequency content. Using the results from the three earthquakes and averaging the results from several different points on the dynamic model, the cyclic behavior was formed (Table 3.7-10).

Independent of earthquake or component frequency, 99.5 percent of the stress reversals occur below 75 percent of the maximum stress level, and 95 percent of the reversals lie below 50 percent of the maximum stress level. This relationship is shown in Figure 3.7-116.

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake was found in the following manner:

a. The fundamental frequency and peak seismic loads were found by a standard seismic analysis

b. The number of cycles which the component experiences were found from Table 3.7-10 according to the frequency range within which the fundamental frequency lies

c. For fatigue evaluation, 1/2 percent (0.005) of these cycles are conservatively assumed to be at the peak load, and 4.5 percent (0.045) at three-quarter peak. The remainder of the cycles will have negligible contribution to fatigue usage, as their resultant stresses are well below the fatigue limits set forth in the ASME B&PV Code Section III.

The SSE has the highest level of response. However, the encounter probability of the SSE is so small that it was not necessary to postulate the possibility of more than one SSE during the life of a plant. Fatigue evaluation due to the SSE was not necessary since it is an emergency condition and thus not required by ASME B&PV Code Section III.

The OBE is an upset condition and therefore, must be included in fatigue evaluations according to ASME B&PV Code Section III. Investigation of seismic histories in the PSARs of many plants show that during a 40-year life (See Appendix B, Section B, for discussion of operation beyond the original design plant life) it is probable that five earthquakes with intensities one-tenth of their individual prescribed SSE intensity, and one earthquake approximately 20 percent of their individual prescribed SSE intensity, will occur. Therefore, the probability of even an OBE is extremely low. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, one OBE intensity earthquake was postulated for fatigue evaluation. Table 3.7-11 shows the calculated number of fatigue cycles and the number of fatigue cycles used in design.

3.7.3.2 Basis for Selection of Forcing Frequencies

Amplified response spectra (floor) developed for horizontal (two directions) and vertical direction earthquakes was the basic source of seismic design accelerations. As noted in Subsections 3.7.1.2 and 3.7.3.6, seismic accelerations are selected from the amplified response spectra based on natural frequency calculations for the component or system. All frequencies in the range of 0.25 to 33 Hz were considered in the analysis and testing of structures, systems, and components.

3.7.3.3 Root-Mean-Square Basis

The term "root-mean-square basis" is not to be used in the procedure for combining modal responses. The SRSS is used to describe the method of combining modal responses when used herein and is described as follows:

$$R = \sqrt{\sum_{i=1}^{n} (R_i)^2} \qquad (3.7\text{-}12)$$

where

R = combined response

R_i = response in the ith mode

n = number of modes considered in the analysis

3.7.3.4 Procedure for Combining Modal Responses

When a response spectrum method of analysis is used to analyze a system or subsystem, the maximum response (displacements, accelerations, shears, and moments) in each mode is calculated independently of time; whereas, actual modal responses are nearly independent functions of time, and maximum responses in different modes do not necessarily occur simultaneously. The maximum possible response is given by the sum of the maximum modal responses without regard to sign. It has been shown that the probable maximum response is 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, except in combining closely spaced in-phase modes of vibration.

These closely spaced coupled modes of vibration are detected by computing the model's modal responses and then using both the square-root criterion and the absolute-sum criterion in combining modes. In many locations in a complex model, both criteria give nearly equal results, indicating that a single mode is contributing to the response. If the two criteria give results that differ by a large amount, then more than one mode is contributing to the response. The modes that contribute are examined; if they are closely spaced coupled modes, they are combined using the absolute-sum criterion and are treated as a single mode when combined with the rest of the modes using the square-root criterion.

When the time-history method of seismic analysis is used, the physical displacements, accelerations, shears, and moments due to each mode are added algebraically at each instant of time. Hence, no criterion concerning the method of combining loads from the individual modes needs to be set.

3.7.3.5 Significant Dynamic Response Modes

All significant modes were included in modal dynamic analysis. Generally, the number of significant modes varied between 10 and 30 modes.

A static analysis was used in seismic design if the component, structure, or equipment was essentially rigid or could be properly represented by a single-degree-of-freedom system. If it was rigid, the static load was based on the zero-period acceleration.

If it could be properly represented by a single-degree-of-freedom system, the static load was based on the acceleration corresponding to the natural period of the system. Using the peak of the floor spectrum curve was a conservative approach.

When a static analysis based on the peak floor spectrum curve was used in the seismic design of a component, structure, or piece of equipment that could not be represented by a single-degree-of-freedom system, an amplification factor was used to bound anticipated multi-mode phenomena, or it was ensured that the fundamental natural period of the system was far enough from the period corresponding to the peak value. Therefore, the participation of the expected following modes would not cause the resultant acceleration to exceed the peak value used in the static analysis.

3.7.3.6 Design Criteria and Analytical Procedures for Piping

3.7.3.6.1 Introduction

All Category I piping was seismically analyzed by either a simplified analysis or a multi-degree dynamic analysis, depending on its quality group and nominal size, as shown in Table 3.7-12. The loading combinations correspond to various stress criteria; this is also shown in Table 3.7-12.

3.7.3.6.2 Design Spectra and Anchor Movement

Two orthogonal horizontal earthquake motions and one vertical earthquake motion were considered. The two horizontal earthquake spectra were distinctly applied in north-south and east-west directions along with a vertical response spectra. These spatial results were combined for each point in the piping model by the method of the SRSS.

Modal responses in seismic response analysis were combined using the methods described in NRC Regulatory Guide 1.92, Revision 1. All modes with frequencies of 33 Hz or less were considered.

In cases where more than one response spectrum was applied to a subsystem (i.e., if the system is supported from locations in the structure having different response spectra), an envelope of all applicable response spectra was applied to the subsystem. When GE piping systems are anchored and supported at points with different excitations, the multiple response spectra method was used.

Secondary stresses of piping systems due to seismic anchor movements were computed by using the maximum relative displacements in two horizontal and vertical directions as the boundary conditions at support (anchor) points. The computed secondary stresses were added absolutely to other stresses in accordance with the procedures specified in ANSI B-31.7, Nuclear Piping and ASME B&PV Code Section III-1971.

For the GE-supplied piping, the maximum value of the modal displacement was used in the static calculation of the stresses due to relative displacements in the response-spectrum method. Therefore, the mathematical model of the equipment was subjected to a maximum displacement at its supporting points obtained from the modal displacements. This procedure was repeated for the significant modes of the structure (modes contributing most to the total displacement response at the supporting point). The total stresses due to relative displacement were obtained by combining the modal results using the SRSS method. Since the maximum displacement for different modes does not occur at the same time, the SRSS method is a realistic and practical method.

3.7.3.6.3 Simplified Analysis

When simplified seismic analysis was used for piping, the system is restrained such that the combined seismic stress of the system (SRSS of all three excitations) is less than 7000 psi for the OBE. The methods used and their limitations are presented in Subsection 3.9.2.7. For equipment and piping supplied or analyzed by GE, a simplified dynamic analysis was not used.

3.7.3.6.4 Dynamic Analysis

The general procedure for the modal analysis response-spectrum method for piping systems is described in Subsection 3.7.3.16. 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. The flexibility matrix of each beam element includes axial, bending, shear, and torsional flexibilities. The size of the stiffness matrix for each piping structural element is 12 by 12, since six forces and moments and six deflections and rotations are considered by the piping flexibility program in each of the two nodes of an element.

The unrestrained general stiffness matrix [K] of a dynamic structural model is condensed to a square reduced-stiffness matrix [k]. The purpose of this procedure is to exclude rigid constraints and to condense rotational stiffness coordinates into dependent coordinates of the translational displacement stiffness matrix.

After development of stiffness and mass matrices, natural frequencies and their associated modal shapes are determined by solution of the following equation:

$$\{[k] - \omega_i^2\,[m]\}\,[Q_i] = 0 \qquad (3.7\text{-}13)$$

where

$[k]$ = square reduced-stiffness matrix

$[\omega_i]$ = natural frequencies of system (i = 1,2,...n)

$[m]$ = mass matrix

$[Q_i]$ = mode shape vector associated with ith mode

The ω values and $[Q_i]$ matrix for each of the n modes are computed (i = 1, 2,...n, where n equals degrees of freedom of the piping system dynamic structural model). For the acceleration response spectrum method of analysis, the maximum displacements in global coordinates are shown as:

$$\{y_{max_n}\} = [Q]\,\{q_{max}\} \tag{3.7-14}$$

where

$[Q]$ = square matrix containing eigenvectors for each mode

$\{qmax\}$ = $[\omega_n^2]^{-1}\,[S_a]\,[M_n]^{-1}\,[Q]^T\,[m]D$

$[M_n]$ = generalized mass = $[Q]^T[m][Q]$

$\{D\}$ = direction vector

$[S_a]$ = matrix of special acceleration values

The maximum displacement equation can be rewritten as:

$$\{y_{max}\} = [Q][\omega_n^2]^{-1}[S_a]\{\Gamma\}_n \tag{3.7-15}$$

where

$\{\Gamma\}_n$ = participation factor of system

$\{\Gamma\}_n$ = $[M_n]^{-1}[Q]^T[m]\{D\}$

Inertia forces for each mass point are then calculated from

$$\{F_{max}\}_n = [m]_n[Q][\omega_n^2]_d\{q_{max}\}_n \tag{3.7-16}$$

(nxn)(nxd)(dxd) (dx1)

where

d = number of modes considered

The computation of internal moments at each mass node represents maximum seismic inertial responses due to excitations of vertical amplified response spectrum and horizontal amplified response spectrum applicable to the piping system. The stresses due to the inertia forces were determined using the SRSS of the horizontal responses and the vertical response.

The relative displacement between anchors was determined from the dynamic analysis of the structures. The results of the relative anchor point displacements were used for a static analysis to determine the additional stresses due to relative anchor point displacements as described in Subsection 3.7.3.6.2.

All of the calculations outlined in this subsection, except for those of the GE scope of supply, were performed by using the computer program AutoPIPE, PIPSYS, or NUPIPE, for the analysis of a three-dimensional piping system (Section 3.13).

3.7.3.6.5 Allowable Stress

Allowable stresses in the piping caused by an earthquake are in accordance with the ASME Code Section III. Internal moments and forces, computed in Subsection 3.7.3.6.4 as the seismic responses of the piping system, were then combined with deadweight, pressure, thermal, and other mechanical loads to complete the stress analysis of all Category I and some nonseismic piping.

For ASME Code Class 1 piping larger than 1-in. nominal pipe size, stress intensities and cumulative usage factors of the piping system were computed based on formulations specified in ASME Code Section III-1971, NB-3653. For ASME Code Class 1 piping, 1-in. nominal pipe size and smaller, the stress intensities were computed based on formulations specified in the ASME Code Section III-1971, NC-3650.

General seismic design and analysis criteria for ASME Code Classes 1, 2, and 3 are defined in Table 3.7-13. For additional information, see Section 3.9.

Allowable stresses in the earthquake restraint components such as shock suppressors are in accordance with stress limits established by American Institute of Steel Construction (AISC)-1969 for original plant design, subsequent design codes are as described in subsection 3.9.2.2.5.2.

3.7.3.7 Basis for Computing Combined Responses

The two horizontal components and one vertical component of ground motion are accounted for in the following manner:

a. Components - The procedure described in Subsection 3.7.3.16 for Category I component analysis in combining the dynamic responses from horizontal and vertical amplified response loading was based on

   1. Static analysis - The sum of the horizontal plus the vertical responses

   2. Dynamic analysis - The SRSS of the two horizontal modal responses and vertical modal responses.

b. Piping systems - The procedure described in Subsection 3.7.3.6 for Category I piping analysis in combining the dynamic responses from horizontal and vertical amplified response loading was based on the SRSS of the two horizontal spatial responses and the vertical spatial response.

Alternatively, for subsystems or components under the GE scope of supply, the two horizontal components and one vertical component of ground motion can be accounted for in the following manner: Two sets of seismic results are obtained.

First, the maximum value of the horizontal component of the earthquake is assumed to act in one horizontal direction simultaneous with the vertical component, and the loads are computed for this combination. Next, the maximum value of the horizontal component of the earthquake is assumed to act perpendicular to the direction previously assumed and simultaneous with the vertical component, and loads are computed for this combination. The larger of these two loads at each point in the system is used for design.

This method of analysis is based on the fact that the seismologist specified the maximum resultant value of the horizontal component of the earthquake when specifying the horizontal component of the SSE. Using this method, it is conservatively assumed that the horizontal and vertical components of the earthquake response occur simultaneously.

3.7.3.8 Amplified Seismic Responses

Constant load factors were not used for vertical floor response in the seismic design of Category I components. As described in Subsection 3.7.1.2, amplified response spectra (floor) were developed for horizontal (two directions) and vertical seismic excitation. Components and systems were designed for the combination of operating loads acting simultaneously with horizontal and vertical seismic loads based on these response spectra. As noted in Subsection 3.7.2.1, three directions of earthquake motion were considered.

In the simplified dynamic analysis described in Subsection 3.7.3.9 for Category I piping, constant load factors based on applicable amplified response spectra were used as the vertical and horizontal amplified floor response loading.

3.7.3.9 Use of Simplified Dynamic Analysis

Simplified dynamic analysis methods for piping are discussed in Subsection 3.9.2.7.

3.7.3.10 Modal Period Variation

The modal period variation was considered in the derivation of floor response spectra curves by widening the peaks of those curves (Subsection 3.7.2.6).

3.7.3.11 Torsional Effects of Eccentric Masses

If the torsional effect of the valve operator was likely to have a significant effect on the results of an analysis, the operator's mass and moment arm were included in the mathematical model. However, if the pipe stress due to the torsional effect was expected to be less than 500 lb/in^2, the offset moment due to the operator was neglected.

3.7.3.12 Piping Outside Containment Structure

Category I piping located outside the containment, but not buried, was analyzed so that allowable piping and structural stresses were not exceeded due to differential movement at support points, at containment penetrations, and at entry points into other structures, as specified in Subsection 3.7.3.6.

3.7.3.13 Interaction of Other Piping With Category I Piping

For systems that are partially Category I, the seismically qualified portion of the system extends to the first seismic constraint (anchor) beyond the isolation valves that separate the safety-related from the nonseismic portions of the system. The isolation valve(s) that defines the operational boundary location between seismic and nonseismic portions of the system is identified on the respective piping and instrumentation diagram(s). The specific constraint beyond the isolation valve that is included in the seismic analysis of the piping system would be identified on the "dash-2 version" of the system-piping isometric drawing(s). These isometric drawings are commonly referred to as the system hanger drawings. The hanger drawings and required seismic analyses are retained as permanent plant records.

3.7.3.14 Field Location of Supports and Restraints

The field location of seismic supports and restraints for piping and piping systems was so selected as to keep the seismic stresses and deflections below the allowable limits. The following procedure was used to ensure that the seismic constraints were actually applied consistent with the assumptions used in the seismic analysis of the piping:

a. The seismic analyst recommended approximate locations for seismic restraints

b. The piping designer and/or the field engineer found an exact location for each restraint (including the methods of attachment) and notified the seismic analyst of these locations by generating as-built field sketches

c. The final seismic analysis was performed using the agreed-upon locations to arrive at piping loads

d. The piping stress analysis was performed to ensure that applicable code limits are not exceeded.

The field location of seismic supports and restraints for GE Category I piping and piping system components was selected to satisfy the following two conditions:

a. The location selected must furnish the required response to control stresses and deflections to allowable limits

b. Adequate building strength for attachment of components must be available.

The final location of seismic supports and restraints for Category I piping, piping system components, and equipment, including the placement of snubbers, was checked against the drawings and instructions issued by the engineer. An additional examination of these supports and restraining devices by an engineer competent in the design of Category I systems and components was made to ensure that the location and characteristics of these supports and restraining devices were consistent with the dynamic and static analyses of the systems.

3.7.3.15 Seismic Analysis for the Reactor Pressure Vessel, Fuel Elements, Control Rod Assemblies, and Control Rod Drives

The seismic loads on the RPV and internals were based on a dynamic analysis of the reactor/auxiliary building, with the appropriate forcing function supplied at ground level. The seismic model of the RPV and internals is given in Figure 3.7-16.

This mathematical model consists of lumped masses connected by elastic (linear) members. Using the elastic properties of the structural components, the stiffness properties of the model were determined. This included the effects of both bending and shear. To facilitate hydrodynamic mass calculations, several mass points (fuel, shroud, vessel) were selected at the same elevation. The various lengths of control rod drive (CRD) housings were grouped into the two representative lengths shown. These lengths represent the longest and shortest housings in order to adequately represent the full range of frequency response of the housings.

The high fundamental natural frequencies of the CRD housings result in very small seismic loads. Furthermore, the small frequency differences between the housings due to the various lengths result in negligible differences in dynamic response. Hence, the modeling of intermediate length members becomes unnecessary. Not included in the mathematical model are light components such as jet pumps, in-core guide tubes and housings, sparger, and their supply headers. This reduces the complexity of the dynamic model.

The presence of fluid and other structural components (e.g., fuel within the RPV) introduces a dynamic coupling effect. Dynamic effects of water enclosed by the RPV were accounted for by introduction of a hydrodynamic mass matrix. This matrix served to link the equations of motion acceleration terms of points at the same elevation in concentric cylinders with a fluid entrapped in the annulus. The seismic model of the RPV and internals had two horizontal coordinates for each mass point considered in the analysis. The remaining translational coordinate (vertical) was excluded because the vertical frequencies of RPV and internals were well above the significant horizontal frequencies. Furthermore, all support structures, buildings, and containment walls have a common centerline, making the coupling effects negligible.

The vertical seismic loads acting on the structures within the RPV are based on a separate vertical dynamic analysis.

The multi-node mathematical model used represents the RPV, RPV internals, pedestal, and the shield wall by lumped masses and a set of springs idealizing both the inertial and stiffness properties of the system. Between mass points, the structural properties are reduced to uniform beam segments of crosssectional area, effective shear area, and moment of inertia. The two rotational coordinates about each node point were excluded because of the momentary contribution of rotary inertia from surrounding nodes. Since all deflections were assumed to be within the elastic range, the rigidity of some components was accounted for by equivalent linear springs.

The shroud support plate was loaded in its own plane during a seismic event, and hence was extremely stiff. Therefore, it was modeled as a rigid link in the translational direction. The shroud support gussets and the local flexibilities of the vessel and shroud contribute to the rotational flexibilities, and were thus modeled as an equivalent torsional spring. The foundation mat was considered to be fixed. The effect of the water inside the RPV was included in the vertical model by adding concentrated mass to the node points in the mathematical model.

The seismic analysis was performed by a modal super-position time-history analysis. Design calculations were made using one of the following: peak loads or accelerations from the response time histories; amplified response spectra appropriately broadened; or peak displacements created by each natural mode of the structure. Table 3.7-14 lists several of the seismic loads on the RPV and RPV internals.

3.7.3.16 Seismic Analysis of Components

3.7.3.16.1 General

All Category I equipment has been documented for seismic adequacy. Depending on equipment location, the basic source of seismic design data is either the ground response spectrum or the amplified response spectrum, derived through a dynamic analysis of the relevant structure.

The uncertainties in the calculated values of fundamental structural frequencies due to reasonable variations in the structural properties are taken into account in the use of amplified response spectra. The peak resonant period value(s) in the amplified response spectra was developed as described in Subsection 3.7.2.6.

Three principal methods of documenting adequacy for Category I components are

a. Analysis

b. Analysis and testing

c. Testing.

Static Analysis

Static analysis was used for equipment that could be characterized as a relatively simple structure. This type of analysis involves the multiplication of the equipment or component weight times the applicable acceleration value (direction-dependent loading) to produce forces that have been applied at the center of gravity in the horizontal and vertical directions. A stress analysis of equipment components, such as feet, hold-down bolts, and other structural members, has been performed to determine their adequacy.

In the specification of equipment for static analysis, two or more sets of acceleration data were provided, the choice of which set to use being dependent on the equipment's fundamental natural frequency. The relevant response curves were reviewed to determine a "cutoff frequency" which bounds the rigid range from the resonance range of the response curves. Components having fundamental natural frequencies above the cutoff frequency were analyzed to rigid range response accelerations.

For components having a fundamental natural frequency below the cutoff frequency, analysis was based on response accelerations that were not less than those indicated by the amplified response curves over the full frequency range of the component. If the fundamental mode of the component fell within any of the resonant response peaks, and if the component cannot be characterized as a single degree-of-freedom system, the resonant peak response acceleration was used.

Each of the three defined directions of earthquake input (two horizontal and one vertical taken orthogonally) was evaluated separately. The calculated results of the analyses were superimposed on an SRSS of the maximum horizontal with the vertical basis. The particular response values to be combined are optional (i.e., acceleration, force, stress) but must remain consistent throughout.

Dynamic Analysis

A detailed dynamic analysis was performed when component complexity or dynamic interaction precluded static analysis, or when static analysis had been too conservative.

To fully describe the behavior of a component subjected to dynamic loads, infinite numbers of coordinates are required. Since calculation at every point of a complex model is impractical, the analysis is simplified by a selection of a limited number of mass points. The lumped-mass approach is used in the dynamic analysis. In the lumped-mass idealizations, the main structure is divided into substructures, and the masses of these substructures are concentrated at a number of discrete points. The nature of these substructures and the stiffness properties of the corresponding modeling elements determine the minimum spacing of the mass points and the degrees of freedom to associate with each point. In accordance with the minimum spacing requirements, the analyst could then choose, for the model, particular mass points reflecting predominant masses of the components that give significant contribution to the total response.

In cases for which some dynamic degrees of freedom do not contribute to the total response, static or kinematic condensation was used in the analysis.

The normal mode approach was used for dynamic seismic analysis of components. Natural frequencies, eigenvectors, participation factors, and modal member-end forces and moments of the undamped structure were calculated. The system of equations that describe the free vibrations of an n-degree-of-freedom undamped structure is:

$$[\mathrm{M}]\{\ddot{\mathrm{X}}\} + [\mathrm{K}]\{\mathrm{X}\} = 0 \qquad (3.7\text{-}17)$$

where

$[\mathrm{M}]$	=	mass matrix
$[\mathrm{K}]$	=	stiffness matrix
$\{\mathrm{X}\}, \{\ddot{\mathrm{X}}\}$	=	displacement, acceleration vectors

The mode shapes and frequencies were solved in accordance with:

$$[\mathrm{K} - \omega_n^2 \mathrm{M}]\{\phi\}_n = 0 \qquad (3.7\text{-}18)$$

where

ω_n^2	=	frequency of nth mode
$\{\phi\}_n$	=	mode shape vector for nth mode

Eigenvector-eigenvalue extraction routines, such as Householder-QR, Jacobi reduction, and inverse iteration, are used, depending upon the total number of dynamic degrees of freedom and the number of modes desired.

For each mode, the participation factor for the specific direction "i" is defined by:

$$\Gamma_{n_i} = \frac{[\phi]^T[\mathrm{M}][\mathrm{D}]_i}{[\phi]^T[\mathrm{M}][\phi]} \qquad (3.7\text{-}19)$$

where

Γ_{n_i}	=	participation factor shape vector for nth mode in ith direction
$[\phi]^T$	=	transpose of mode
$[\mathrm{D}]_i$	=	earthquake direction i

The modal member-end forces and moments were determined by:

$$[F_m]_n = [K_m]\,[\phi]_n \qquad (3.7\text{-}20)$$

where

K_m = member stiffness matrix

For each modal frequency, the corresponding response acceleration was determined for a given level of equipment damping from the applicable response curve. Modes within the broadened response peak were assigned the peak resonant response value.

The maximum response for each mode was found by computing

$$[\ddot{X}] = \Gamma_{n_i} R_{n_i} [\phi]_n$$

$$[\dot{X}] = \frac{1}{\omega_n}[\ddot{X}]_n$$

$$[X] = \frac{1}{\omega_{n^2}}\,[\ddot{X}]_n \qquad (3.7\text{-}21)$$

$$[F]_n = \frac{\Gamma_n R_{n_i}}{\omega_n^2}\,[F_m]_n$$

where

$[\ddot{X}]_n$ = modal acceleration for nth mode

$[\dot{X}]_n$ = modal velocity for nth mode

$[X]_n$ = modal displacement for nth mode

$[F]_n$ = moment vectors for nth mode

R_{n_i} = spectral acceleration for nth mode in ith direction

The basis for combination of modal responses is described in Subsection 3.7.3.4.

Each of the three defined directions of earthquake input (two horizontal and one vertical taken orthogonally) were evaluated separately. The calculated results of the maximum horizontal and vertical directions were combined on an SRSS basis. The particular response values to be combined are optional (i.e., acceleration, force, stress) but must remain consistent throughout.

Testing

For tested equipment that has an operability function, the Fermi 2 requirements supplement other applicable industry standards (such as IEEE-344-1971, Section 3.10) or provide guidance for testing where no such codes are available. Equipment packages or components were shown to be adequate either by being tested individually, as part of a simulated structural section, or as part of an assembled module or unit. In any case, the minimum acceptance criteria were

a. No loss of function, or ability to function, during and/or after the proposed test, as required

b. No structural/electrical failure (i.e., connections and anchorages) that would compromise component integrity

c. No adverse or maloperation during and after the proposed test that could result in an improper safety action.

Equipment vendors and suppliers were required to formulate programs for qualifying the equipment in accordance with the conditions specified in the earthquake design requirements.

Sinusoidal, sine beat, and random input tests were accepted as methods of seismic qualification based on the particular component location, structure, and floor response characteristics. Structures, particularly at lower elevations, exhibit a broad frequency range response similar to the ground motion during an earthquake. This broad range frequency motion is filtered at higher structural elevations, and response becomes more sinusoidal in nature. Knowledge of the floor response characteristics of the structure generally dictates the requirements for testing. Periodic testing is applicable where periodic floor motion is indicated and, conversely, random input testing is most applicable for broad frequency range input to components. Periodic testing can be used to develop multiple peak floor responses, as well as single peak, providing sufficiently high force is used.

Conservative periodic (sine wave) inputs to the tested component have been specified regardless of floor input characteristics since the test requires a sine sweep throughout the full frequency range at "zero period" response levels associated with relevant floor and building locations, as well as the generally required resonance dwells at discovered equipment resonance requencies. Other less conservative but generally acceptable testing techniques (periodic) have been reviewed to ensure conservatism of test results.

Either single or multiaxis test results are considered acceptable. While multiaxis tests, with some definition of "most conservative phasing" are ideal, the availability of testing machines and techniques capable of attaining this ideal is severely limited.

General testing guidance criteria specified for components include the following:

a. Sinusoidal testing

1. A frequency scan (2 octaves per minute maximum) at a constant acceleration level is performed for as much of the range between 1 and 35 Hz as practicable or justified. The objective of this test is to determine the natural frequencies and amplification factors of the tested equipment and its critical components or appurtenances and to ensure general seismic adequacy over the full frequency range of interest. The acceleration inputs used are the maximum rigid range accelerations indicated by the relevant response spectrum curves

2. A dwell test of the equipment at its fundamental natural frequency is included at the acceleration values specified previously in Item 1. Additionally, other frequencies are selected if amplification factors of 2.0 or more are indicated. A minimum 15-sec duration is considered acceptable for each dwell.

b. Sine beat testing

A sine beat test is performed in conjunction with a sine scan and is an alternative to the dwell portion of the program outlined previously in Item 2. The sine beat test is performed at natural frequencies and bands of large amplification identified during the sine scan. The duration and peak amplitude of the beat for each particular test frequency are chosen to most nearly produce a magnitude of equipment response equivalent to that produced by the particular floor response spectrum at justifiable damping levels

Current practice indicates that a minimum of 10 cycles per beat should be used unless it can be shown that a lower number of cycles is sufficient to duplicate or exceed the response spectra for the equipment at the appropriate location. Five sine beats with a time delay between beats are commonly used

c. Random motion testing

Random excitation may be used for components. The excitation is controlled to provide a test response spectrum that is required to envelope the required response spectrum

Additionally, as stated in Subsection 3.10.1.1, components purchased after the issuance of IEEE-344-1975 are specified to be qualified to the requirements of that standard.

3.7.3.16.2 Category I Equipment

In the analysis of the building systems, the Category I equipment was lumped with the building floor on which the equipment is supported. The equipment was analyzed as a secondary system, and the model simulating the equipment was excited by the floor response spectra obtained from the time-history analysis of the building. However, the equipment model was included in the building model if the mass of the equipment was large enough to cause significant change in the building response.

Equipment was idealized as a mathematical model consisting of lumped masses connected by elastic members or springs. Results for some selected large generic nuclear steam supply system (NSSS) Category I equipment are given in Table 3.7-15. Seismic loadings due to two orthogonal horizontal directions and the vertical direction were combined as detailed in Subsection 3.7.3.7.

When the equipment was supported at more than two points located at different elevations in the building, the response spectrum at the elevation near the center of gravity of the equipment was chosen as the design spectrum for the GE equipment. An envelope of each applicable spectrum was developed for the equipment.

The relative displacement between supports was determined from the dynamic analysis of the structure. The relative support point displacements were used for a static analysis to determine the additional stresses due to support displacements. Further details are given in Subsection 3.7.3.6.2.

The seismic design criteria for Category I equipment and components are described in Section 3.9.

3.7.3.17 Cable Tray Support Systems

3.7.3.17.1 Introduction

A cable tray and its attachment to a building comprise a structural system used to support electrical cables in a power plant. This subsection describes some of the aspects that are considered in designing cable tray supports to meet seismic criteria. A cable tray system's response characteristics, its modal periods of vibration, its relation to the seismic load, and its floor response spectra determine how the system is analyzed to ensure that it meets the seismic criteria.

3.7.3.17.2 Analysis and Design

The cable trays and cable tray support system were evaluated to withstand forces caused due to dead load, live load, and seismic conditions.

The following combinations of dead load, live load, and earthquake were investigated and checked to determine the most severe condition:

a. Dead load of various components with allowable stresses according to AISC specifications. The dead load on cable trays consists of cables, trays, and attachments. In the case of hangers, it includes the dead weight of hangers also. The original cable tray design loading was 40lb/ft^2 generally, except in the relay room area, where it was 50 lb/ft^2. An on-going program was later established to monitor the actual weight of cables in the trays and to account for fire wrap, conduit and air drop loads. Cable tray design load is adjusted to reflect these actual loads. See Subsection 8.3.1.4.3 for additional information

b. Dead load plus a concentrated live load of 200 lb at the mid-span was specified for all trays with the exception of those in the drywell. For drywell trays, a dead load plus a concentrated live load of 250 lb was specified

c. Dead load plus earthquake.

The cable trays and the support system were modeled as a multidegree-of-freedom system with the mass of the cables plus tray lumped at the levels at which they are supported.

For vertical excitation, the fundamental period of vibration was computed by using a simplified model of a continuous beam with hinged ends. This approximation was found to be consistent with the numerous models studied for this purpose.

The response spectrum obtained from the analysis of the building was used in determining the response of the cable tray support.

The horizontal and vertical seismic excitations were assumed to be acting simultaneously along the principal axis on the cable tray system. The seismic response was computed by taking the SRSS of the individual responses.

It was observed that contribution due to nonfundamental modes was negligible, and hence the effect of closely spaced modes was negligible also.

The design was based on the 1968 edition of the "Specifications for the Design of Cold-Formed Steel Structural Members."

In the design specification for cable trays, deadweight loading did not include the weight of fire wrapping material or any other attachments, such as the top hat cover, which were subsequently added. Accordingly, hanger modifications were made where necessary, and the structural adequacy of the cable trays was verified.

3.7.3.18 Safeguard Against Derailing the Reactor/Auxiliary Building Crane

The crane is safeguarded against derailing in the three principal directions of seismic movement.

The crane was subjected to a detailed analysis (results reported in Reference 18). Seismic responses of the crane to an SSE based on the crane's fundamental frequency in the vertical and two horizontal directions (perpendicular and parallel to girder) in the loaded and unloaded conditions were determined on the basis of the reactor building seismic response spectra. Vertical accelerations did not exceed 0.431g and horizontal accelerations did not exceed 0.65g for the loaded and unloaded crane in all positions. Thus, no uplift is encountered.

In the horizontal direction parallel to the runway, the crane is regarded as a suspended mass in space. Maximum seismic acceleration is limited by the friction forces of the crane's wheels. In the parked position, the crane is locked to the runway by means of electrically operated locking bars on both sides of the crane. These bars are designed to secure the crane in a stationary position in the event of a tornado strike or horizontal seismic forces (Figure 3.8-32).

In the horizontal direction perpendicular to the runway, the crane bridge wheels have sufficient play on their axles to accommodate thermal movement and seismic deflection of the crane supporting structure. In the event that seismic deflections exceed axle play, the insides of the girders are provided with seismic end stops impacting on the runway structure.

The trolley is equipped with seismic end stops to prevent excessive movement perpendicular to its runway. The trolley is not restrained of movement parallel to its runway. The movement is expected to be minimal as only wheel friction forces are transmitted and also due to the stabilizing effect of the cable and hook assembly, which acts as a pendulum.

3.7.3.19 Other Subsystems

This subsection refers to the structural subsystems such as cranes, racks, ventilation ducts, and tanks. If the subsystem is idealized as a single-degree-of-freedom system, the forces in each direction are determined by applying, through the center of gravity, a static force equal to the weight of the subsystem multiplied by a frequency-dependent multiplier obtained from the floor response spectrum curve. In all other cases, the subsystem is modeled as a multidegree system with an adequate number of lumped masses that predict the true dynamic response of the subsystem. For tanks, the dynamic effect of fluid oscillations is considered in both the horizontal and vertical directions.

The Control Center (CC) HVAC System and Standby Gas Treatment System (SGTS) duct and duct supports were revalidated to demonstrate their structural adequacy under the combined effects of dead load, internal duct pressure (normal operating and maximum credible), and three-directional seismic (OBE and SSE) loads in accordance with the requirements and acceptance criteria contained in Reference 20.

The horizontal and vertical seismic excitations used for the revalidation of the CCHVAC and SGTS duct and duct supports were based on Figures 3.7-36 through 3.7-41, 3.7-83, 3.7-85, 3.7-86 and 3.7-87 for OBE, and on Figures 3.7-60 through 3.7-65, 3.7-92, 3.7-94, 3.7-95 and 3.7-96 for SSE. For revalidation of the duct systems under OBE effects, damping values of 4% and 2% were used for rectangular and round duct, respectively. For SSE effects, damping values of 7% and 4% were used for rectangular and round duct respectively.

Structural acceptance criteria for CCHVAC and SGTS duct and duct supports were based on the minimum published yield and ultimate strengths of the duct and duct support materials. Straight duct segment maximum stresses were limited to 0.9 Fy of the duct material for SSE effects (0.6 Fy for OBE effects) in accordance with ANSI/ASME-N509-1980 (Reference 21). Duct support allowable member stresses were governed by Table 3.8-19 for structural steel. Duct support anchorages (base plates and anchors) were also evaluated for adequacy. References 22 and 23 were used for the expansion anchor acceptance criteria. To conform to these acceptance criteria, duct system structural modifications were made where necessary, and the structural adequacy of the systems was verified.

3.7.4 Seismic Instrumentation Program

3.7.4.1 Comparison With Regulatory Guide 1.12

A seismic instrumentation program has been implemented to monitor and record the input motion and behavior of Fermi 2 in the event of an earthquake. The instrumentation program described below meets the intent of Regulatory Guide 1.12, Revision 1. (See Subsection A.1.12 for regulatory guide compliance statement.)

The seismic event recording system conceived and designed for Fermi 2 was documented in January 1972, prior to the issuance of Regulatory Guide 1.12. The project reviewed the Fermi 2 earthquake recording system for compliance with the requirements of Regulatory Guide 1.12, Revision 1, and concluded that the intent of Regulatory Guide 1.12 was satisfied. The seismic monitoring system is classified as Seismic Category II/I; however, it is designed, tested, mounted and maintained in a manner that gives a high degree of confidence that it will function during and after a seismic event of the Fermi 2 SSE. Seismic Category II/I is consistent with Regulatory Guide 1.29.

3.7.4.2 Location and Description of Instrumentation

Strong motion triaxial accelerographs are installed in two different reactor/auxiliary building locations. One of the accelerographs measures the response of the free field at the building foundation. The other device establishes the anticipated excitation to the reactor building containment structure and major internal equipment.

Strong motion triaxial response spectrum recorders are additionally installed at six seismically interesting plant locations. Three of these passive devices are contained in the reactor/auxiliary building, one at the free field/foundation location, adjacent to the active accelerograph, one in the relay room on the second floor, and one at the top of the reactor/

auxiliary building on the fifth floor. The other three response spectra recording devices are installed in the RHR complex. One of the devices is installed to measure the excitation to the diesel generators and the RHR pumps, a second device is installed to measure the excitation to the items at higher RHR complex elevations, and the third device is installed at a location in the RHR complex to measure the excitation experienced at the mechanical draft cooling towers.

3.7.4.2.1 Active Sensors

Active earthquake-recording instrumentation has been provided to measure and record the basic ground motion time-history acceleration, as well as the seismic excitation of the reactor/auxiliary building complex foundation and the primary containment structural elements including major internal equipment items. The complete system consists of an active seismic recording system and an active seismic playback system. The active seismic recording system consists of two triaxial accelerometers and a digital recorder. Both triaxial accelerometers are installed with the same geometrical orientation. The active seismic playback system consists of a computer and printer.

A seismic trigger activates the seismic recording system and indicates to control room personnel that a seismic event has occurred. The trigger is initiated from the High Pressure Coolant Injection (HPCI) room accelerometer, where free field and building foundation excitations are established. The seismic trigger senses any acceleration above a preset limit, 0.01g, and activates the recording system.

3.7.4.2.1.1 Active Instrumentation Locations

Triaxial accelerographs responding to acceleration excitation in three mutually perpendicular axes have been installed at two locations, as shown in Figure 3.7-117. The recording axes directions coincide with each other. A vertical axis is used, as well as two horizontal axes corresponding to the mutually orthogonal primary directions of the reactor/auxiliary building structure. The specific instrument locations are identified as follows:

a. Reactor/auxiliary building subbasement in the HPCI room (Figure 3.7-117, Location 1). This record is used for direct comparison with the ground motion and reactor building earthquake design excitation. This single triaxial earthquake accelerogram is used to establish not only the ground motion, but also the building foundation excitation, since it has been established that soil-structure interaction effects are negligible at Fermi 2

b. At the bottom of the RPV pedestal, adjacent to the floor at the base of the drywell (Figure 3.7-117, Location 2). This record is used to establish the primary containment element excitation, anticipated RPV motions, and the environment for major containment structure equipment items.

3.7.4.2.1.2 Active Instrumentation Specifications

Over the frequency range of interest (0.1 to 40 cps), the output of the seismic transducer is a voltage proportional to acceleration. This voltage is filtered and conditioned such that the overall sensitivity of the channel is approximately 2.5 V/g.

The seismic trigger is activated at the .01g level. This trigger not only initiates recording by the accelerometers, but also activates an operator indication and on-line monitor of the free field, subbasement time-history excitation. All data is stored in a unique file in memory, which is subsequently analyzed and available for evaluation.

3.7.4.2.2 Passive Sensors

Passive earthquake recording instrumentation has been provided throughout the complex to measure various ground motion and in-structure response spectra. These directly measured triaxial spectra may be used for comparison with basic facility design spectra without the need for intermediate data reduction. The passive instrumentation serves as a backup for the active sensors, and provides basic definitions of reactor/auxiliary building and RHR complex input motion and response phenomena. In addition, this instrumentation provides a direct definition of internal equipment environments in the Category I structures, as well as basic information defining the Category I structure and internal equipment response.

The complete system comprises 18 response-spectrum recorders (six triaxial spectrum recorders) that are identical in configuration and orientation and differ only in their installation location.

3.7.4.2.2.1 Passive Instrumentation Locations

Triaxial response spectrum recorders, which respond to accelerations in three mutually perpendicular axes, have been installed at six locations, as illustrated in Figures 3.7-117 and 3.7-119. These devices have been installed so that the directions of the recording axes coincide. One recording axis is vertical, and two are horizontal, corresponding to the mutually orthogonal directions of both the reactor/auxiliary building and RHR complex. The specific passive instrumentation locations are described as follows:

a. At the reactor/auxiliary building subbasement adjacent to the active accelerograph in the HPCI room (Figure 3.7-117, Location 1). Spectra generated at this location will be used to evaluate the recorded seismic spectra relative to the corresponding facility operating bases response spectra. These data will assist in the determination of the need to shut down the facility after an earthquake and will also be used for possible subsequent comparison with ground motion spectra generated from the active accelerometer records

b. At the reactor/auxiliary building second floor relay room (Figure 3.7-117, Location 4). These spectra will define the in-structure equipment environment at an intermediate height for investigation of critical Category I equipment

c. At the reactor/auxiliary building fifth floor (Figure 3.7-117, Location 5). This device will define the in-structure equipment environment spectra at an upper level for investigation of Category I elements at this structural elevation

d. At a critical location in the RHR complex (Figure 3.7-119, Location 6). This instrument will define the environment for investigation of the structural and equipment response for this Category I structure at the emergency diesel generator and RHR system pump location

e. At a critical location in the RHR complex (Figure 3.7-119, Location 7). This instrument will define the environment for the investigation of the structural and equipment response of this Category I structure at an elevated equipment location

f. At a third critical location in the RHR complex (Figure 3.7-119, Location 8). This instrument will define the environment for the investigation of the structural and equipment response of this Category I structure at an upper elevation corresponding to the sensitive region for mechanical draft cooling tower equipment apparatus.

3.7.4.2.2.2 Passive Instrumentation Specifications

There are 12 sensing elements included in each single response spectrum recorder. These elements provide a resolution of 1 percent of full scale at damping between 1 and 3 percent of critical, and are accurate in a temperature range of -50°C to +85°C. A tabular summary of representative reed frequencies and nominal full-scale acceleration limits follows:

Reed Number	Nominal Frequency (cps)	Nominal Full Scale Acceleration Limit (g)
1	2.0	±1.6
2	2.5	±2.5
3	3.2	±4
4	4.0	±6
5	5.0	±10
6	6.4	±16
7	8.0	±24
8	10.1	±34
9	12.7	±42
10	16.0	±64
11	20.2	±81
12	25.4	±90

3.7.4.3 Operator Seismic Event Notification and Recording System

The active seismic recording system is equipped with an earthquake event indicator which has been placed in the facility control room (Figure 3.7-117, Location 3). This event indicator notifies the control room operator that an excitation has occurred at the facility foundation level in excess of the 0.01g trigger setpoint.

Concurrent with this operator notification, the seismic trigger automatically activates the recording system and on-line monitor in the facility relay room (Figure 3.7-117, Location 4).

An earthquake is considered to have occurred if the trigger mechanism and companion event indicator are actuated. Post-earthquake data evaluation and reduction activities ensue in accordance with established project procedures.

The minimum system recording time is limited only by memory and will provide a minimum of 25 minutes of continuous recording. The lengths of pre-event and post-event memory are user selectable and recorded for ease of analysis. Continuous system actuation capability is ensured by an internal battery, which remains “trickle” charged from an ac power line. Minimum system accuracy is ± 8%.

The seismic recording system has playback capability enabling the facility operators to immediately obtain the representative acceleration time-history.

3.7.4.4 Post-Earthquake Evaluation Activities

An earthquake is considered to have occurred if the trigger mechanism is activated (.01g or larger) with attendant control room indication. Essential post-earthquake evaluation activities are summarized by the flow chart included in Figure 3.7-120.

In accordance with the requirements of Appendix A to 10 CFR 100, if the earthquake excitation exceeds that described by the facility OBE spectra, the reactor must be shut down to cease operation in as timely a manner as possible. The sequence of events by which this shutdown decision is made is described in Subsection 3.7.4.4.1. Subsequent earthquake data reduction and analysis activities are described in Subsections 3.7.4.4.2 and 3.7.4.4.3.

3.7.4.4.1 Immediate Operational Decision

Immediately upon signal indication of earthquake occurrences, the control room operator verifies that there are no abnormal changes in critical plant parameters as indicated by operational instrumentation. If any malfunctions are indicated by the instruments, shutdown is initiated as dictated by the severity of the malfunction.

In the absence of instrumentation-indicated malfunctions, plant personnel go to the relay room to examine the active earthquake system records, and to the HPCI subbasement location to extract the ground motion spectra from the passive measurement device.

In examining the active information, the earthquake ground motion response time-history from the subbasement location which is played on-line may immediately be evaluated as to whether or not the observed peak acceleration exceeds the OBE (.08g horizontal, .05g vertical). If the observed peak acceleration is greater than the OBE value, controlled shutdown activities are initiated.

Three directional response spectra are extracted from the passive earthquake recording device in the HPCI room subbasement location by recording the observed acceleration record associated with each of the tuned reeds at their various response frequencies. The spectra obtained from the passive recording device are compared with the facility OBE spectra, and if the response observed at any measured frequency exceeds that corresponding OBE level, facility shutdown is initiated. If not, the remaining passive data are extracted from all the passive instrumentation, the passive gages are reset, and the facility continues to operate with no further earthquake data investigation required.

If it is determined during the data operational evaluation process that the facility is to be shut down, data reduction and analysis activities ensue as described in the following subsections.

3.7.4.4.2 Earthquake Data Reduction

If the active or passive earthquake recording instrumentation indicates that the OBE design level was exceeded, data-reduction activities commence. In this regard, concurrent passive and active data reduction is accompanied by physical facility structural and component inspection.

All active data are reduced by generation of detailed acceleration time-histories from each active instrument. These time-histories are subsequently used to generate response spectra for all active instrument locations.

Passive spectral instrumentation exists at six varied facility locations. The 18 resulting passive measured response spectra are plotted after extraction of the necessary raw information from the recording devices.

Detailed inspection activities are documented for all Category I items, and any malfunctions or permanent distortions in the apparatus are recorded.

A document is prepared summarizing and presenting the reduced data for further evaluation purposes. Detailed earthquake data analysis activities are described in the following subsection.

3.7.4.4.3 Earthquake Data Analysis

Data reduction activities result in assembly of facility response spectra for representative locations and elevations at the Fermi 2 site in the Category I structures. These spectra are compared with the established facility SSE spectra for initial evaluation purposes. If the recorded event spectra do not exceed the facility established SSE spectra, no further investigation is necessary, and facility operations may resume. Certain essential structures and components were reassessed to a site-specific earthquake spectrum (larger than the SSE spectrum). Such items can be screened out of the investigation in a similar manner.

If there are spectra that exceed the SSE facility spectra at some facility locations, all Category I items in this proximity are noted and specifically evaluated with respect to the observed excitations. For each of these items identified, actual fragility level capability will be documented and compared with the excitation environment recorded. Items assessed to be satisfactory in this evaluation may be considered acceptable for continuing plant use.

If an item fragility level is equal to or less than the earthquake event excitation level recorded, a detailed dynamic analysis and/or system test combined with comprehensive item inspection will be required to establish whether or not the particular item is satisfactory for continuing facility service. If in this investigation it is established that tolerable permanent deformation or damage was sustained, the item will be considered satisfactory for continuing use. If not, the item shall be fully refurbished or a new item must be procured and installed for continuing plant operation.

3.7.5 Seismic Design Control

3.7.5.1 Introduction

Category I systems and components are designed to perform their intended function during and after the specified earthquakes. Category I items, at various locations in the reactor/auxiliary building and the RHR complex, are capable of withstanding the seismic excitation specified in the specifications (Subsection 3.7.5.2). Rational analyses or test results as described in this specification validating the seismic performance of all Category I items were submitted to Edison. Furthermore, independent documented reviews are performed to validate the adequacy of the seismic designs performed and to ensure the compatibility of such designs. The seismic design control procedure is outlined in Table 3.7-16.

All Category I structures, systems, and components were reviewed on an item-by-item basis. Nonseismic structures, systems, and components were examined to ensure that they do not adversely interact with close-proximity Category I items. General Electric-supplied items are subjected to a rigorous independent design review. The independent reviewer assists in an audit of GE seismic design documentation only, since independent design review is performed by GE.

Items that have significant mass and size relative to the building in which they are located were analyzed coupled to the structure itself to appropriately consider interaction effects. The RPV, primary containment, and crane are so considered in Subsection 3.7.2.1.2.2.

Items that are small enough relative to the building in which they are located so as not to influence the dynamic response of the building itself are considered uncoupled from the building. These items are validated to be capable of withstanding the earthquake excitation defined by the response of the building at the location where they are attached.

3.7.5.2 Seismic Performance Specification

To ensure that the various Fermi 2 vendors provide seismically adequate systems and components, a seismic performance specification was prepared.

It was specified in the seismic performance portion of the component specification that items mounted directly to a building are validated as capable of withstanding the excitation from the building at the location where they are attached. Design requirements are delineated in the specification. There are also other components that are attached to systems attached to the building rather than to the building itself. These components cannot be validated to the building excitation since it is necessary to consider the influence of the response of the system to which the component is attached.

In many cases, the component was procured as a part of the total system. A total system validation was required in this situation and the component validation was undertaken to the levels indicated as appropriate from the system analysis.

When a rigid component must be procured apart from the system to which it belongs, it is validated by the vendor to the mounting amplified acceleration or to the maximum acceleration on the response spectrum applicable for the parent system. This is conservative since essentially the assumption is made that the system is in resonance with the building. Nonrigid components are examined by the seismic design reviewer on an item-by-item basis.

A procurement specification is prepared for each item required for Fermi 2. Seismic provisions are accounted for by reference to the seismic performance portion of the specification. Generally, the location of the item being purchased is delineated so that the appropriate validation spectra are selected from the performance specification, unless plant-wide use qualification is required.

In addition, the potential vendors are informed that they must submit a description of their proposed validation with their basic bid package. A vendor is not selected until his seismic design approach is reviewed and found acceptable.

The seismic environment for items mounted directly to the structures is defined as a function of the item location in terms of vertical and horizontal response spectra. The vertical and horizontal excitations are assumed to act simultaneously. Figures 3.7-32 through 3.7-115 define the response spectra for both the reactor/auxiliary building and the RHR complex. Enveloping spectra have also been generated for plant-wide use qualification.

The seismic environment for rigid components not mounted directly to the structure, but rather mounted to a system that is connected to the structure, are validated to the peak acceleration indicated on the appropriate response spectrum. If the component is considered part of the system connected to the floor, then it is validated to the system acceleration obtained directly.

3.7.5.3 Seismic Acceptance Criteria

3.7.5.3.1 Validation Procedures

The seismic capability of vendor-supplied items is validated by either a rational dynamic response analysis or a suitable dynamic system test, or some combination of both as hereinafter specified.

3.7.5.3.2 Dynamic Response Analysis

The rational dynamic response analysis conforms to standard techniques of engineering mechanics. Stress and deformation of all elements of the vendor-supplied items are examined in accordance with the design criteria as shown in Sections 3.8 and 3.9. The vendor seismic dynamic response analysis is submitted to Edison for approval before acceptance of the items. The analysis submitted to Edison includes the following:

a. Description of the mathematical model used in the analysis

b. Description of the determination of properties such as the model mass distribution, damping, and stiffness characteristics

c. Development of the dynamic response analysis equations of motion

d. Discussion of experimental investigation supporting the given model and equations of motion

e. Description of the way the seismic input is applied in the analysis

f. Description of the solution techniques for the equations of motion

g. Evaluation of the seismic capability of the equipment including calculations of stress and deformation levels.

3.7.5.3.3 Dynamic System Test

Where dynamic system tests are made to verify the acceptability of the vendor-supplied items in accordance with the design criteria as shown in Sections 3.8 and 3.9, the tests impose upon the equipment a dynamic test environment equal to or greater than that specified in the earthquake criteria at all frequencies. Acceptable test environments are achieved by use of a controlled shaker table or a shock machine. Other test techniques are acceptable if the input is suitably defined. Before the testing is undertaken, a test procedure document is submitted to Edison for approval. This document contains a complete description of the testing to be done including descriptions of the following:

a. Method of measurement of the test environment including descriptions of active and passive instrumentation and techniques used in generating response spectra

b. Method of measurement of the response of the equipment including descriptions of active and passive instrumentation and operational testing

c. Method of deciding the adequacy of the equipment.

After the dynamic system tests are performed and before the acceptance of the equipment, a report summarizing the results of the testing is submitted to Edison for approval. The report includes pertinent test data as well as an analysis of the test data.

3.7.5.4 Independent Review

An independent review of the seismic design approach proposed by the various vendors is performed. In review of a proposed analytical validation, the approach is accepted or modifications are recommended. It is possible that for some items no analytical validation could be acceptable. In this case, it is recommended that the vendor be required to provide a test validation.

During the independent review of a proposed test validation, the approach is accepted or modifications are recommended.

The vendor then updates and modifies his seismic design package until it is accepted without any recommended modifications. The complete results of the seismic analysis either by testing or by calculations are documented in a clear and concise format and submitted.

The documentation submitted generally includes the following:

a. The abstract describes the purpose of the test or calculations and gives a brief description of the problem

b. The conclusions summarize the results obtained from the test or calculations. A concise statement is made regarding the conclusion reached, which is related to the purpose of the test or calculations

c. Both data and assumptions are listed. In the case of testing, the documentation includes the type of test machine used, the loads considered, and the efforts made to idealize the actual case in preparing the test. In the case of calculations, the documentation includes the loads considered, the weights used, the damping values chosen, and the assumptions used to convert the design criteria to actual loads, stresses, or displacements

d. A description of the test or the method of analysis is included. In the case of testing, the type of test, the input motion, and the generated response spectrum from this motion are presented. In the case of calculations, the analytical method, all analytical equations and their derivation from basic principles, any assumptions made to idealize boundary to initial conditions, the limitations of the applicability of the analysis (if any), and documentation to establish the validity of any computer program used are stated

e. The documentation outlines the results of the test or analytical calculations. In the case of testing, the measurements obtained, their interpretations, and numerical or graphical form of the test results are shown. In the case of calculations, design calculations as well as figures and sketches for the mathematical model showing loads, resultant forces, and displacements (if possible) are presented

f. Design drawings of the component and its support, including all necessary dimensions, are provided.

3.7.6 Testing of General Electric-Supplied Equipment

For GE-supplied essential mechanical equipment, two types of tests were used in the dynamic testing of equipment: free vibration and forced vibration tests. Dynamic analysis was also used for qualification of components. A description of the qualification methods is given below.

3.7.6.1 Free Vibration Test

This test was performed on equipment whose response is dominated by the fundamental mode. The critical damping ratio and fundamental frequency were determined from this test and were used to verify or supplement calculated values used in dynamic analysis of this equipment. This test was not used alone to demonstrate dynamic capability.

In this test an initial displacement or initial velocity was imparted to the equipment. The initial displacement was introduced by forcibly displacing the equipment and then suddenly releasing the force. The initial velocity was obtained by applying an impulse. Accelerometers or strain gages were mounted on the equipment. After first ensuring that the equipment was vibrating in its primary mode, the critical damping ratio was calculated from the logarithmic decrement.

3.7.6.2 Forced Vibration Test

The equipment was mounted on a shake table or driven by an eccentric shaker. The critical damping ratios, resonant frequencies, and the equipment's functional capability were determined.

The critical damping ratio of the equipment was determined by applying a sinusoidal acceleration and measuring the forced response curve (amplitude versus forcing frequency). The critical damping ratio was then calculated by using the half-power method, fitting a theoretical forced response curve through the data points, or direct reading of the resonant amplification. The vibratory motion used was such that the vibratory loads equaled or exceeded seismic loads represented by the applicable floor spectra. When testing was the only method used to demonstrate functional capability of equipment, the mounting conditions were simulated and the equipment was operating during and after the tests.

When the seismic testing is supplemented by analysis, the seismic stresses are added to those from normal and accident conditions in the appropriate loading combinations in order to ensure that the equipment will perform its required safety functions. Each type of equipment is examined individually to provide this assurance.

As an example of the approach required for extremely complicated geometrical configurations, the tests performed on the HPCI turbine are summarized below.

The major structures of the HPCI and reactor core isolation cooling (RCIC) turbines were qualified by dynamic analysis. The turbine-control-unit components were qualified by dynamic testing on a shake table with electrical and hydraulic systems functional. The actual mounting brackets were simulated in the test mounting. Vibration in all three perpendicular axes (two horizontal and one vertical) was accomplished by orienting the equipment in three directions on a horizontal shake table. A resonant search was made from 1 to 200 Hz, and the components with substantial resonances below 33 Hz were modified before the functional qualification test was performed. These modifications were applied to the standard design. This equipment was then tested with a sinusoidal input of 1.6g and then 3.0g for at least 30 sec at each of the arbitrary frequencies of 10, 15, and 23 Hz in each of the three perpendicular directions, with all systems operational. Since there were no functional failures, the equipment was deemed qualified for up to 3.0g horizontal or vertical maximum floor acceleration for all frequencies 33 Hz and below.

When required, all tests conducted will use methods and procedures comparable to those in the foregoing example. Furthermore, the amplitudes supplied at the support brackets will be equal to or greater than the levels predicted by system dynamic analysis.

3.7 SEISMIC DESIGN

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3.7 SEISMIC DESIGN

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TABLE 3.7-1 MODIFIED MERCALLI INTENSITY (DAMAGE) SCALE OF 1931 (Abridged)

I. Not felt except by a very few under especially favorable circumstances (I Rossi-Forel Scale).

II. Felt only by a few persons at rest, especially on upper floors of buildings. Delicately suspended objects may swing (I to II Rossi-Forel Scale).

III. Felt quite noticeably indoors, especially on upper floors of buildings, but many people do not recognize it as an earthquake. Standing motorcars may rock slightly. Vibration like passing of truck. Duration estimated (III Rossi-Forel Scale).

IV. During the day felt indoors by many, outdoors by few. At night some awakened. Dishes, windows, doors disturbed; walls make creaking sound. Sensation like heavy truck striking building. Standing motorcars rocked noticeably (IV to V Rossi-Forel Scale).

V. Felt by nearly everyone, many awakened. Some dishes, windows, etc., broken: a few instances of cracked plaster; unstable objects overturned. Disturbance of trees, poles, and other tall objects sometimes noticed. Pendulum clocks may stop (V to VI Rossi-Forel Scale).

VI. Felt by all, many frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight (VI to VII Rossi-Forel Scale).

VII. Everybody runs outdoors. Damage negligible in buildings of good design and construction; slight to moderate in well-built ordinary structures; considerable in poorly built or badly designed structures; some chimneys broken. Noticed by persons driving motorcars (VIII Rossi-Forel Scale).

VIII. Damage slight in specially designed structures; considerable in ordinary substantial buildings with partial collapse; great in poorly built structures. Panel walls thrown out of frame structures. Fall of chimneys, factory stacks, columns, monuments, walls. Heavy furniture overturned. Sand and mud ejected in small amounts. Changes in well-water. Persons driving motorcars disturbed (VIII + to IX Rossi-Forel Scale).

IX. Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb; great in substantial buildings, with partial collapse. Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken (IX+ Rossi-Forel Scale).

X. Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations; ground badly cracked. Rails bent. Landslides considerable from riverbanks and steep slopes. Shifted sand and mud. Water splashed (slopped) over banks (X Rossi-Forel Scale).

XI. Few, if any (masonry) structures remain standing. Bridges destroyed. Broad fissures in ground. Underground pipelines completely out of service. Earth slumps and land slips in soft ground. Rails bent greatly.

XII. Damage total. Waves seen on ground surfaces. Lines of sight and level distorted. Objects thrown upward into the air.

TABLE 3.7-2 DAMPING VALUES

Item	Percent of Critical Operating-Basis Earthquake	Safe-Shutdown Earthquake
General		
Equipment and large-diameter piping	0.5	1.0
Small-diameter piping	0.5	1.0
Welded and H.S. bolted steel framed structures	2.0	5.0
Bolted and riveted steel framed structures	5.0	10.0
Welded structural Assemblies (equipment and supports)	2.0	4.0
Reinforced-concrete structures	2.0	5.0
Specific		
Reactor pressure vessel	2.0	2.0
CRD housing	3.5	3.5
Fuel	7.0	7.0
Drywell-building (coupled)	2.0	5.0
CCHVAC and SGTS Rectangular Ducts and Duct Supports	4.0	7.0
CCHVAC and SGTS Round Ducts and Duct Supports	2.0	4.0

TABLE 3.7-3 THE REACTOR/AUXILIARY BUILDING COORDINATES OF MASS CENTROIDS

Mass		Coordinates of Centroid X	Y	Z
1	Slab model	13.93	1.42	43.50
2		-0.15	11.79	73.50
3		-0.08	10.45	107.50
4		+3.85	-1.18	119.50
5		+14.08	4.83	144.50
6		-42.56	+0.23	195.50
7		-91.44	-105.27	43.50
8		-91.44	-105.27	68.00
9		62.25	32.79	157.50
10	Frame model	-99.69	0.23	195.50
11		-99.69	0.23	144.50
12		-23.69	0.00	32.08
13		-23.69	0.00	43.16
14		-23.69	0.00	73.50
15		-23.69	0.00	90.16
16		-23.69	0.00	107.50
17		-23.69	0.00	119.50
18		-23.69	0.00	144.50
19		-23.69	0.00	39.92
20		-23.69	0.00	57.00
21		-23.69	0.00	74.33
22		-23.69	0.00	85.00
23		-23.69	0.00	96.00
24		-23.69	0.00	107.50
25		-23.69	0.00	118.66
26		-23.69	0.00	122.50
27		-23.69	0.00	135.90
28		-23.69	0.00	44.33
29		-23.69	0.00	57.90
30		-23.69	0.00	66.00
31		-23.69	0.00	87.90
32		-23.69	0.00	107.50

TABLE 3.7-4 THE REACTOR/AUXILIARY BUILDING MASS PROPERTIES

Mass Number	Description	Elevation	X	Kips Y	Z	Kip – ft^2 θ_x	θ_y	θ_z
1	Reactor building 1st floor	583 ft 6 in.	30,162	30,162	-			271,238,976
2	Reactor building 2nd floor	613 ft 6 in.	24,024	24,024	-			151,530,000
3	Reactor building 3rd floor	641 ft 6 in.	20,224	20,224	-			124,675,856
4	Reactor building 4th floor	659 ft 6 in.	17,993	17,993	-			113,462,096
5	Reactor building 5th floor	684 ft 6 in.	20,628	20,628	-			122,659,328
6	Reactor building roof	735 ft 6 in.	684	684	-			2,310,000
7	Equip. access building floor	583 ft 6 in.	809	809	-			217,400
8	Equip. access building roof	608 ft 0 in.	458	458	-			122,000
9	Auxiliary bay roof	669 ft 6 in.	9,076	9,076	-			20,475,808
10	Upper crane support	735 ft 6 in.	0	0	-			0
11	Lower crane support	684 ft 6 in.	0	0	-			0
12	Reactor support pedestal	572 ft 1 in.	6,776	6,776	-	3,196,640	3,197,640	5,408,625
13	Containment shield	583 ft 6 in.	2,772	2,772	-	1,923,671	1,923,671	3,552,663
14	Containment shield	613 ft 6 in.	2,951	2,951	-	1,903,922	1,903,922	3,463,821
15	Containment shield	630 ft 3 in.	1,222	1,222	-	373,070	373,070	539,254
16	Containment shield	647 ft 6 in.	892	892	-	168,779	168,779	303,694
17	Containment shield	659 ft 6 in.	3,953	3,953	-	238,502	239,502	389,470
18	Containment shield	684 ft 6 in.	3,598	3,598	-	169,475	169,475	259,431
19	Containment vessel	579 ft 10 in.	137	137	-	42,568	42,568	81,232
20	Containment vessel	597 ft 0 in.	200	200	-	42,785	42,785	81,324

TABLE 3.7-4 THE REACTOR/AUXILIARY BUILDING MASS PROPERTIES

Mass Number	Description	Elevation	X	Kips Y	Z	Kip – ft^2 θ_x	θ_y	θ_z
21	Containment vessel	614 ft 4 in.	61	61	-	26,240	26,240	49,924
22	Containment vessel	625 ft 0 in.	91	91	-	12,231	12,231	24,177
23	Containment vessel	636 ft 0 in.	35	35	-	7,412	7,412	13,304
24	Containment vessel	647 ft 6 in.	30	30	-	6,008	6,008	11,351
25	Containment vessel	658 ft 8 in.	43	43	-	5,209	5,209	9,946
26	Containment vessel	662 ft 6 in.	32	32	-	3,008	3,008	6,001
27	Containment vessel	675 ft 11 in.	66	66	-	3,319	3,319	6,638
28	Reactor support pedestal	584 ft 4 in.	465	465	-	49,966	49,966	68,198
29	Reactor support pedestal	597 ft 11 in.	297	297	-	28,165	28,165	45,541
30	Biological shield	606 ft 0 in.	191	191	-	23,595	23,595	35,894
31	Biological shield	662 ft 11 in.	262	262	-	33,613	33,613	49,074
32	Biological shield	664 ft 6 in.	123	123	-	15,242	15,242	23,047

TABLE 3.7-5 THE REACTOR/AUXILIARY BUILDING MEMBER PROPERTIES

Member	From	To	Area (ft^2)	Moment of Inertia (ft^4) I_x and I_y	I_z	Elastic Modulus K. S. F.	Poisson's Ratio	Shear Factor
1	94	12	4,656.0	1,725,569	3,451,138	552,000	0.17	2.0
2	12	13	2,475.0	1,397,911	2,795,822	552,000	0.17	2.0
3	13	14	1,699.0	1,112,287	2,224,574	552,000	0.17	2.0
4	14	15	1,329.0	493,748	987,496	552,000	0.17	2.0
5	15	16	814.0	138,363	276,727	552,000	0.17	2.0
6	16	17	814.0	138,363	276,727	552,000	0.17	2.0
7	17	18	814.0	138,363	276,727	552,000	0.17	2.0
8	12	19	24.2	8,381	16,762	4,175,000	0.27	2.0
9	19	20	21.2	10,901	21,803	4,175,000	0.27	2.0
10	20	21	15.6	8,193	16,386	4,175,000	0.27	2.0
11	21	22	33.2	8,039	17,078	4,175,000	0.27	2.0
12	22	23	19.9	3,833	7,663	4,175,000	0.27	2.0
13	23	24	10.8	2,050	4,100	4,175,000	0.27	2.0
14	24	25	14.5	2,537	5,075	4,175,000	0.27	2.0
15	25	26	13.0	1,771	3,542	4,175,000	0.27	2.0
16	26	27	13.0	1,771	3,542	4,175,000	0.27	2.0
17	12	28	315.0	25,421	50,841	552,000	0.17	2.0
18	28	29	364.0	27,741	55,482	552,000	0.17	2.0
19	29	30	8.2	766	1,532	4,175,000	0.27	2.0
20	30	31	8.2	766	1,532	4,175,000	0.27	2.0
21	31	32	8.2	766	1,532	4,175,000	0.27	2.0

TABLE 3.7-6 THE REACTOR/AUXILIARY BUILDING STIFFNESS COEFFICIENTS

Stiffness Element	Stiffness Coefficients K/ft			K-ft/rad		
	x	y	z	θ_x	θ_y	θ_z
$K_{1,13}$	∞	∞	-	0.	0.	∞
$K_{2,14}$	∞	∞	-	0.	0.	∞
$K_{3,16}$	∞	∞	-	0.	0.	∞
$K_{4,17}$	∞	∞	-	0.	0.	∞
$K_{5,18}$	∞	∞	-	0.	0.	∞
$K_{16,24}$	∞	∞	-	0.	0.	∞
$K_{24,32}$	2.33×10^5	2.33×10^5	-	0.	0.	2.36×10^8
$K_{26,\ REACTOR}$	3.20×10^4	3.20×10^4	-	0.	0.	0.30×10^8
$K_{32,\ REACTOR}$	4.80×10^4	4.80×10^4	-	0.	0.	0.10×10^8

TABLE 3.7-7 REACTOR/AUXILIARY BUILDING SUMMARY OF THE FIRST 20 MODAL PERIODS AND PARTICIPATION FACTORS (HORIZONTAL MODEL)

Mode	Period, sec	Participation Factors, X-Excitation	Y-Excitation
1	0.6583	-0.00054	0.00153
2	0.3686	-0.00029	0.00082
3	0.3479	0.00113	-0.00315
4	0.3116	-0.00132	0.00363
5	0.2832	7.35972	-3.01693
6	0.2829	3.07018	6.70456
7	0.2221	3.58447	-14.67056
8	0.2219	-12.66471	-3.11176
9	0.2011	17.41747	-46.04075
10	0.1994	-26.94208	-20.12656
11	0.1877	-14.71599	27.16832
12	0.1845	-49.52908	-13.47238
13	0.1674	0.19769	-0.78225
14	0.1673	2.78885	0.47093
15	0.1597	0.12897	-1.17272
16	0.1597	-0.01402	0.00499
17	0.1548	-0.41594	15.01317
18	0.1527	0.00384	0.03227
19	0.1223	-0.54652	1.35398
20	0.1177	3.22427	0.85541

TABLE 3.7-8 REACTOR/AUXILIARY BUILDING PROBABLE MAXIMUM DISPLACEMENTS

Horizontal Displacement (ft)[a]

	OBE		SSE	
Mass No.	X-Excit. X-Displ. (ft)	Y-Excit. Y-Displ. (ft)	X-Excit. X-Displ. (ft)	Y-Excit. Y-Displ. (ft)
1	0.00110	0.00095	0.00157	0.00136
2	0.00257	0.00245	0.00366	0.00349
3	0.00376	0.00260	0.00535	0.00469
4	0.00430	0.00376	0.00611	0.00534
5	0.00483	0.00340	0.00686	0.00594
6	0.03502	0.03240	0.05082	0.04623
7	0.00149	0.00117	0.00209	0.00164
8	0.00330	0.00143	0.00465	0.00201
9	0.00491	0.00500	0.00692	0.00705
10	0.03500	0.02780	0.05080	0.04040
11	0.00490	0.00310	0.00689	0.00465
12	0.00070	0.00040	0.00094	0.00060
13	0.00110	0.00070	0.00157	0.00102
14	0.00250	0.00160	0.00357	0.00252
15	0.00310	0.00210	0.00436	0.00322
16	0.00380	0.00270	0.00536	0.00419
17	0.00430	0.00320	0.00610	0.00485
18	0.00480	0.00390	0.00689	0.00600
19	0.00090	0.00060	0.00126	0.00086
20	0.00150	0.00100	0.00208	0.00152
21	0.00220	0.00160	0.00314	0.00238
22	0.00260	0.00190	0.00371	0.00284
23	0.00310	0.00220	0.00441	0.00341
24	0.00380	0.00270	0.00536	0.00419
25	0.00420	0.00300	0.00602	0.00466
26	0.00440	0.00320	0.00625	0.00483
27	0.00490	0.00350	0.00698	0.00539
28	0.00100	0.00060	0.00140	0.00097

TABLE 3.7-8 REACTOR/AUXILIARY BUILDING PROBABLE MAXIMUM DISPLACEMENTS

Horizontal Displacement (ft)[a]

	OBE		SSE	
Mass No.	X-Excit. X-Displ. (ft)	Y-Excit. Y-Displ. (ft)	X-Excit. X-Displ. (ft)	Y-Excit. Y-Displ. (ft)
29	0.00140	0.00090	0.00212	0.00164
30	0.00180	0.00120	0.00263	0.00200
31	0.00290	0.00190	0.00409	0.00306
32	0.00380	0.00270	0.00540	0.00415

Vertical Displacement[a]

Floor Elevation	OBE	SSE
583 ft 6 in. (1st floor)	0.00013	0.00020
613 ft 6 in. (2nd floor)	0.00026	0.00039
641 ft 6 in. (3rd floor)	0.00035	0.00052
659 ft 6 in. (4th floor)	0.00039	0.00057
684 ft 6 in. (5th floor)	0.00044	0.00064

[a] Displacements are relative to the base of the structure.

TABLE 3.7-9 THE REACTOR/AUXILIARY BUILDING SUMMARY OF PERIODS AND PARTICIPATION FACTORS (VERTICAL MODEL)

Mode	Period	Participation Factors
1	0.08136	53.79
2	0.06564	13.71
3	0.06304	-9.36
4	0.06219	-3.69
5	0.05874	27.35
6	0.05554	0.16
7	0.05520	-8.99
8	0.05368	6.22
9	0.05019	3.98
10	0.04999	-2.60
11	0.04985	-4.52
12	0.04925	-1.12
13	0.04575	7.67
14	0.04552	0.22
15	0.04533	-0.48
16	0.04426	7.71
17	0.04162	0.20
18	0.04156	-4.41
19	0.03947	-2.99
20	0.03846	-1.39
21	0.03826	-5.00
22	0.03572	0.79
23	0.03564	-4.34
24	0.03331	2.47

TABLE 3.7-10 NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED DURING A SEISMIC EVENT

Frequency band (Hz)	0 to 10	10 to 20	20 to 50
Number of seismic cycles	168	359	643

TABLE 3.7-11 FATIGUE EVALUATION DUE TO SEISMIC LOAD

Component	Calculated No. of Cycles at Peak Stress	Design No. of OBE Cycles at Peak Stress
1. Reactor pressure vessel		
Vessel	< 3	10
Shroud support	< 3	10
Skirt	< 3	10
2. Category I piping		
Recirculation lines	< 3	60
Steam lines	< 3	60

TABLE 3.7-12 PIPING SYSTEM SEISMIC CRITERIA FOR PIPING LOCATED INSIDE BUILDING STRUCTURES

Group Classification	Type of Earthquake	Type of Seismic Analysis	Combined Stress Calculations	Stress Criteria
A (Size 1-1/4 in. NPS and larger)	OBE[a]	Dynamic response spectra	ASME Section III NB-3650	Normal and upset condition
	SSE[b]	Dynamic response spectra	ASME Section III NB-3650	Emergency and faulted condition
A (Size 1 in. NPS and smaller)	OBE	Response spectra	ASME Section III NC-3650	Normal and upset condition
	SSE	Response spectra	ASME Section III NC-3650	Emergency and faulted condition
B and C (Size 4 in. NPS and smaller)	OBE	Simplified dynamic analysis or dynamic response spectra	ASME B&PV Code - 1971 Section III, Sub-section NC-3650	Upset condition
	SSE	Simplified dynamic analysis or dynamic spectra response	ASME B&PV Code - 1971 Section III, Sub-section NC-3650	Emergency or faulted condition
D+	Unclassified but seismic Group B and C seismic criteria are used.			
D	None	None	ANSI-B-31.1.0	
B and C (Size 5 in. NPS and larger)	OBE	Dynamic Response Spectra	ASME B&PV Code - 1971 Section III, Sub-section NC-3650	Upset Condition
	SSE	Dynamic Response Spectra	ASME B&PV Code - 1971 Section III, Sub-section NC-3650	Emergency or faulted condition

[a] OBE = operating-basis earthquake.

[b] SSE = safe-shutdown earthquake.

TABLE 3.7-13 SEISMIC DESIGN LIMITS FOR CATEGORY I EQUIPMENT

Type of Earthquake	ASME Code Equipment	Loading Combination[a]	ASME Code Operating Category and Design Limits	Deflection Criteria[b]	Seismic Test Criteria[c]
OBE	Active	0.5 SSEL + OCL	Upset	For active equipment	For active equipment
OBE	Passive	0.5 SSEL + OCL	Upset	For passive equipment	For passive equipment
SSE	Active	SSEL + OCL + DSL	Upset or Emergency or Faulted	For active equipment	For active equipment
SSE	Passive	SSEL + OCL + DSL	Emergency or Faulted	For passive equipment	For passive equipment

[a] OCL stands for operating conditions loads, the loads acting on the equipment in each condition to which the equipment is subjected in accordance with "Normal Conditions" as defined by the ASME B&PV Code Section III, 1971.

SSEL stands for safe-shutdown earthquake loads, the seismic loads to which the equipment is subjected during the SSE.

DSL stands for other dynamic loads, such as relief valve blowdown loads. Earthquake loads are combined with otl dynamic loads as described in section 3.9.1.6.3.

[b] Deflection Criteria for Active Equipment: The deflection of any point on the equipment due to all applicable loads shall not impair the function of the equipment or any other Category I active equipment.

Deflection Criteria for Passive Equipment: The deflection of any point on the equipment due to all applicable loads shall not impair the function of any Category I active equipment.

[c] Test Criteria for Active Equipment: The equipment shall perform its intended function during and after the seismic test. Monitoring devices shall be installed during the test to verify that the equipment satisfies the above criteria. In cases where this is not possible, the equipment shall be tested for operation after the seismic test, and realistic engineering evidences which show that the equipment will function during the seismic test shall be presented.

Test Criteria for Passive Equipment: The equipment shall be inspected and checked after the seismic test to ensure that the pressure boundary integrity has been maintained.

TABLE 3.7-14 COMPARISON OF THE DESIGN AND COMPUTED HORIZONTAL SEISMIC LOADS OF REACTOR PRESSURE VESSEL AND INTERNALS

(Some Representative Values)

Location	Seismic Loads X-Excitation	Seismic Loads Y-Excitation	Allowable Loads
Top guide shear	74	116	687
Core plate shear	66	113	687
Stabilizer force (Total)	205	186	2,400
Max. fuel moment: Total	2,410	6,040	32,200
Per bundle	3.15	7.91	
Max. shroud moment	179,000	120,000	207,000
Max. shroud shear	732	537	1,184
Max. vessel skirt moment	101,000	106,000	1,152,000
Vessel skirt shear	286	280	2,600

Units: Moment - in-kip; Shear - kip; Force - kip

COMPARISON OF THE MAXIMUM SSE LOAD ON REACTOR VESSEL AND INTERNALS DUE TO VERTICAL EARTHQUAKE

Component	Seismic Load (kips)	Allowable Load (kips)
Shroud Support-Axial	183	>183[a]
Vessel Skirt-Axial	594	>594[a]

[a] That is, calculated loads result in stresses that are lower than allowable stress.

TABLE 3.7-15 COMPARISON OF CALCULATED SEISMIC LOADS TO DESIGN SEISMIC LOADS OF CATEGORY I EQUIPMENT, SSE CONDITION

(Some Generic NSSS Large Category I Items)

		Calculated Results		
	Equipment	Natural Frequency (Hz)	Seismic Loads	Design Seismic Load
1.	HPCI pump and turbine	>33	0.43g	1.5g
2.	RCIC pump and turbine	>33	0.43g	1.5g
3.	SLC tank	>33	0.8g	1.5g
4.	Spent-fuel racks	≈ 9[a]	0.46g	1.5g
5.				
6.	New-fuel racks	18.75[a]	0.22g	1.5g
7.	Refueling platform[b]	1.3	20,600 psi	36,000 psi
8.	Control room panels	Seismic adequacy determined by test		
9.	Fuel prep machine	>.79	0.1g	1.5g
			Fermi 2 only	
10.	RHR heat exchanger	>15	0.6g	1.5g
11.	Hydraulic control unit	>10.4	0.6g	4.9g

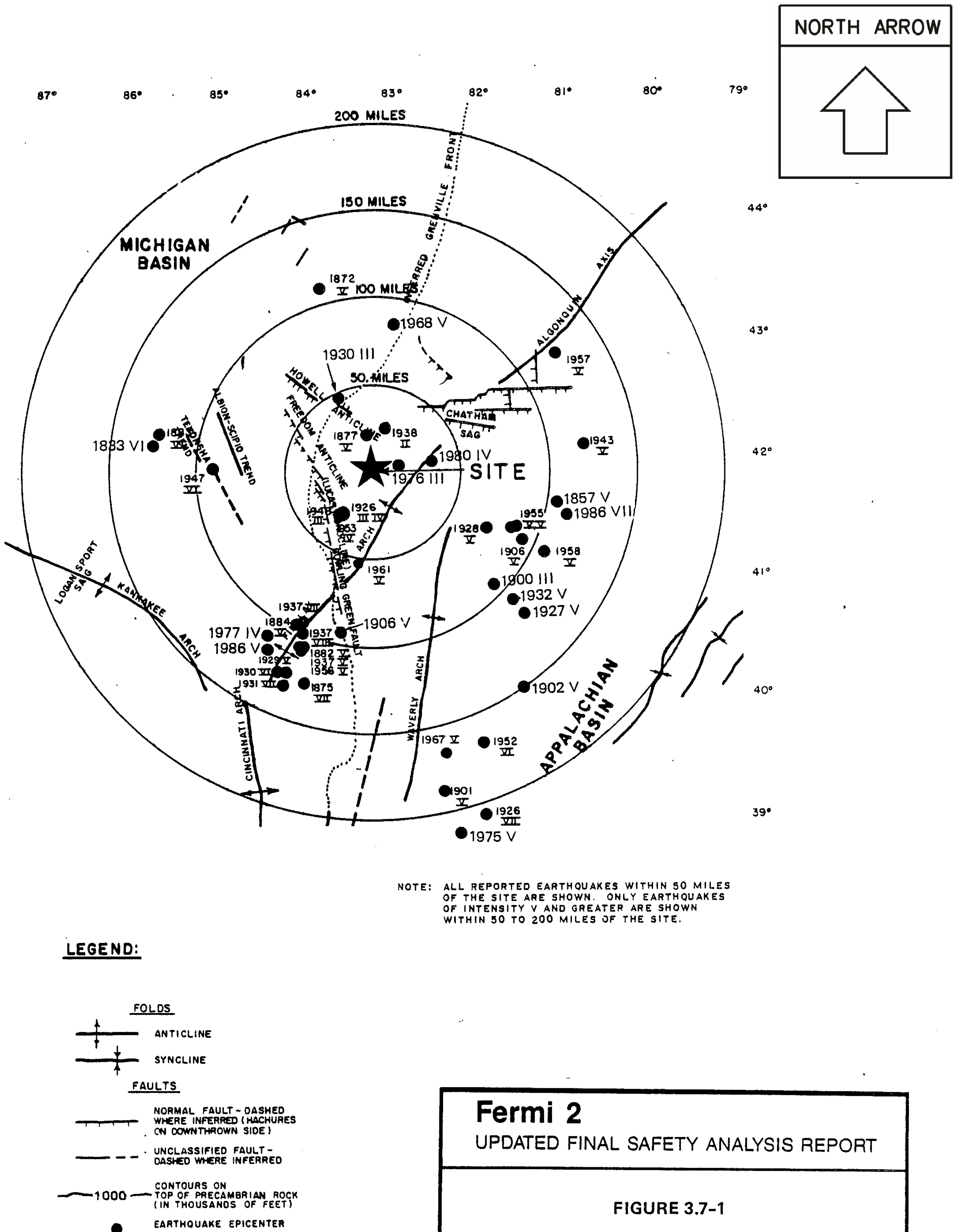
[a] Two percent Damping Calculated Lowest Natural Frequency.

[b] The refueling platform has been reclassified as Seismic Category II/I.

TABLE 3.7-16 SEISMIC DESIGN CONTROL ACTION ITEMS

(General Block Diagram)

Item	Responsibility	Description of Action			
1	Edison engineer (EE)	Prepare specification of vendor equipment			
2	EE	Transmit specification to the independent reviewer (IR) for aseismic performance review			
3	Independent reviewer (IR)	Review aseismic performance specification			
4	EE	Submit equipment specification to vendor (including the IR's review comments)			
5	Vendor	Select method of validation			
		Analytical	Testing		
		(or combination of both)			
6	Vendor	Perform analysis	Develop test procedures		
7	EE	Transmit report to the IR	Transmit test procedures to IR		
8	IR	Review report	Review test procedures		
9	EE		Action on test procedures:		
			Approved	Disapproved	
			OK for testing	Modify and resubmit	
10	Vendor		Perform test and submit report to Edison		
11	EE		Submit vendor test report to IR		
12	IR		Review report		
13	IR	Transmit documented review of report to Edison			
14	EE	Action on Analysis Report:		Action on Test Report:	
		Approved	Disapproved	Approved	Disapproved
15	Vendor		Revise analysis and resubmit		Perform revised test and resubmit
16	EE	File approved vendor validation package and IR report			
17	Vendor	File Edison's aseismic design approval			



NOTE: ALL REPORTED EARTHQUAKES WITHIN 50 MILES OF THE SITE ARE SHOWN. ONLY EARTHQUAKES OF INTENSITY V AND GREATER ARE SHOWN WITHIN 50 TO 200 MILES OF THE SITE.

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FIGURE 3.7-1

EPICENTER MAP

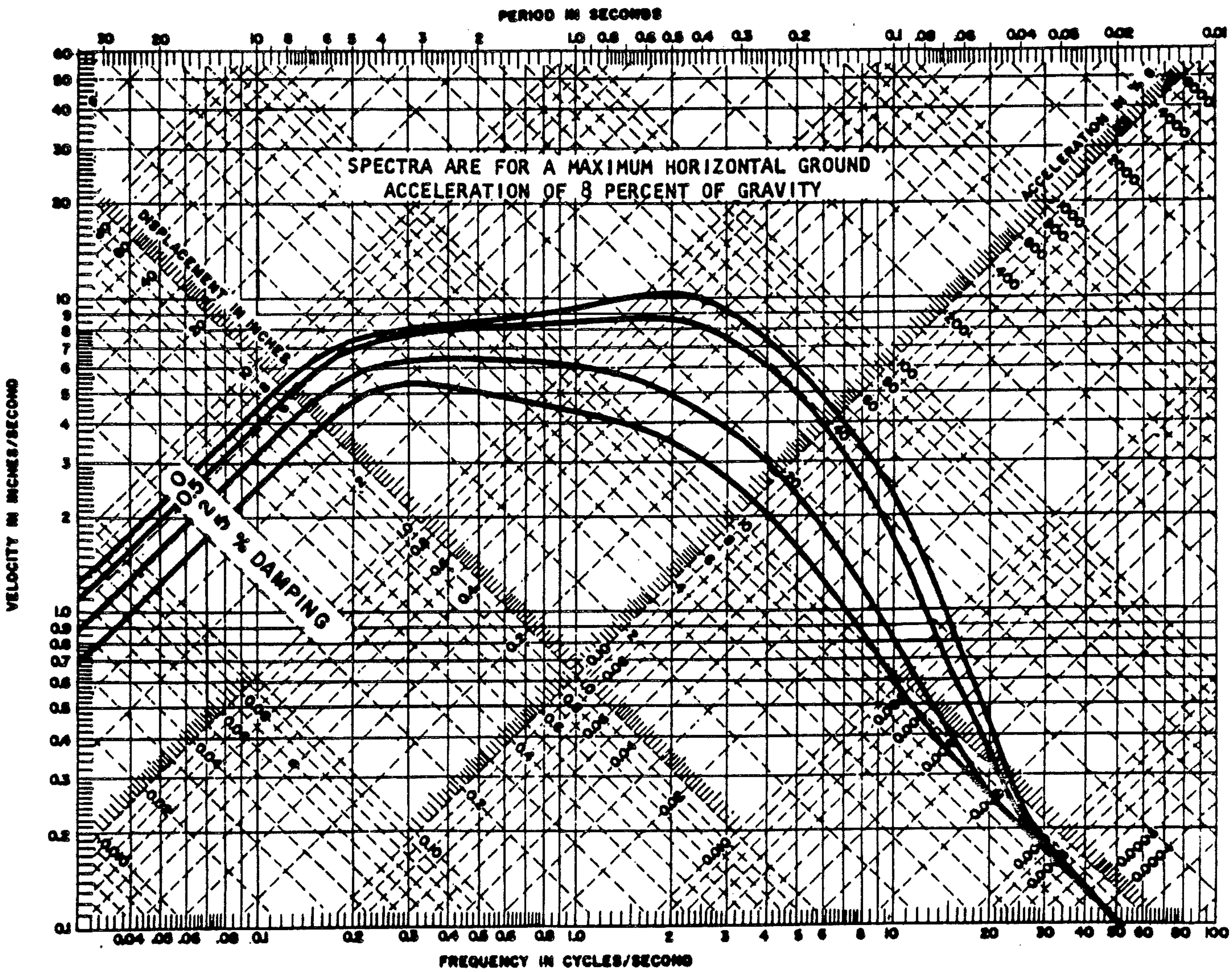

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FIGURE 3.7-2

HORIZONTAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
GROUND LEVEL

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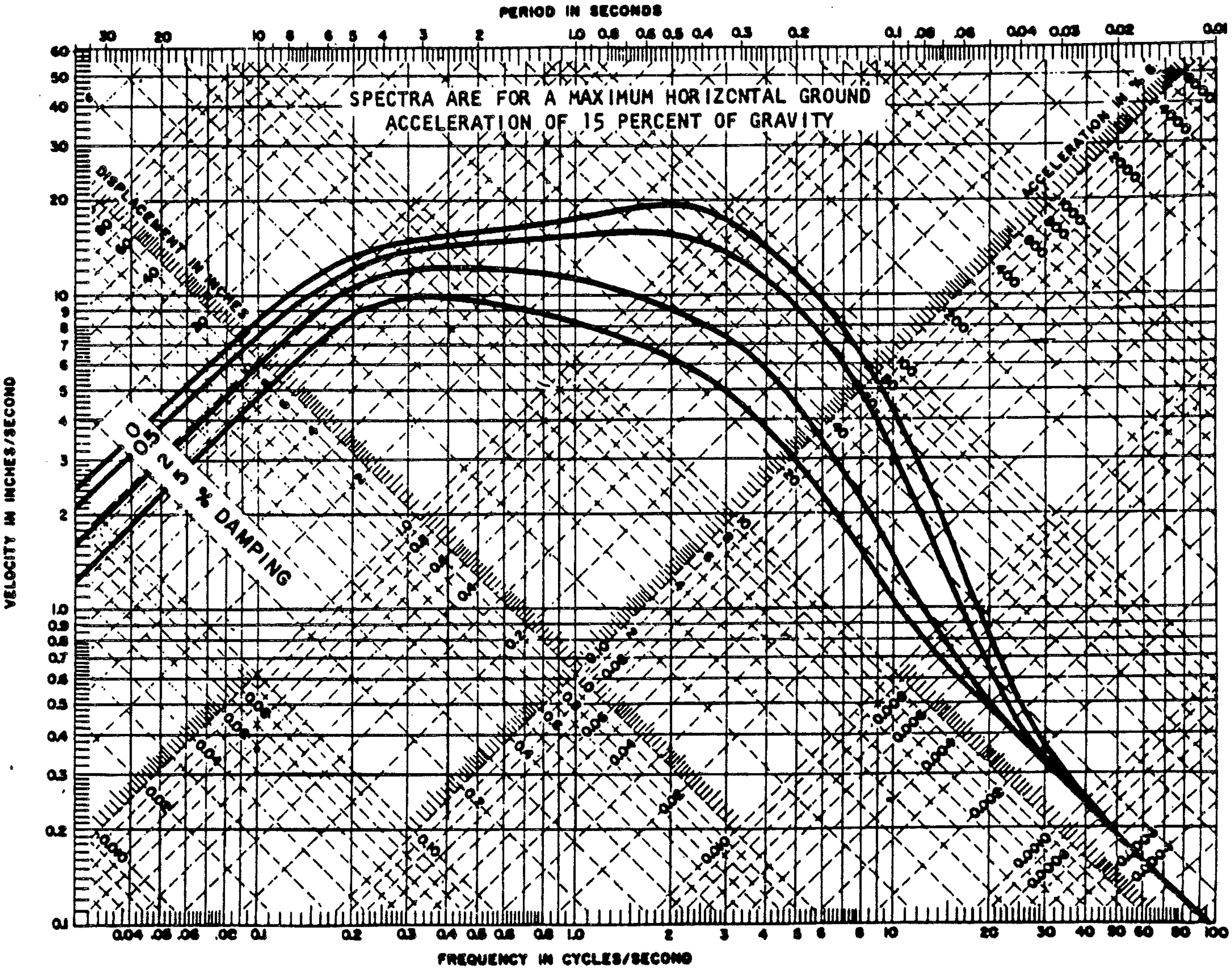

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FIGURE 3.7-3

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
GROUND LEVEL

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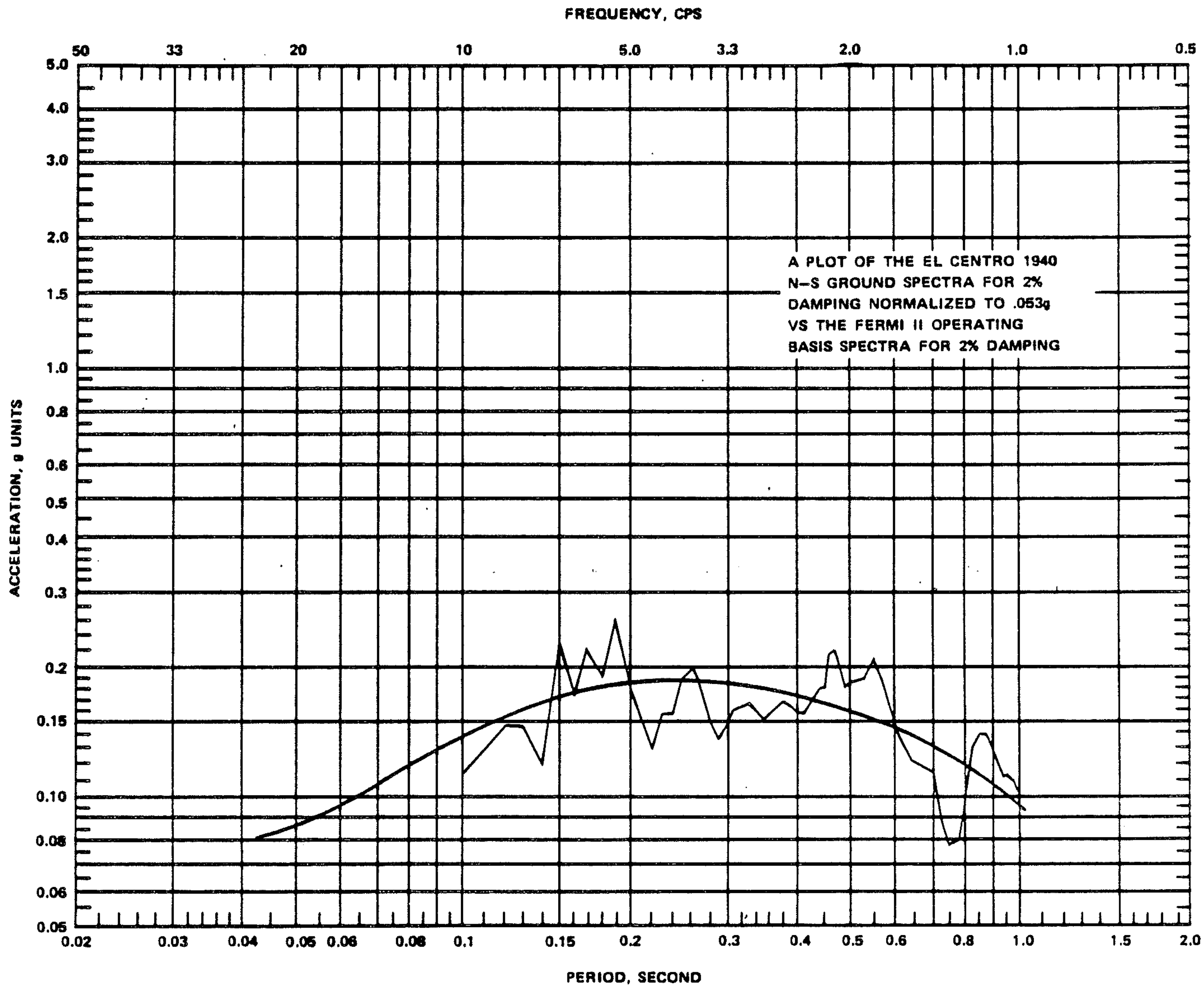

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FIGURE 3.7-4

EL CENTRO 1940 – N-S GROUND SPECTRA

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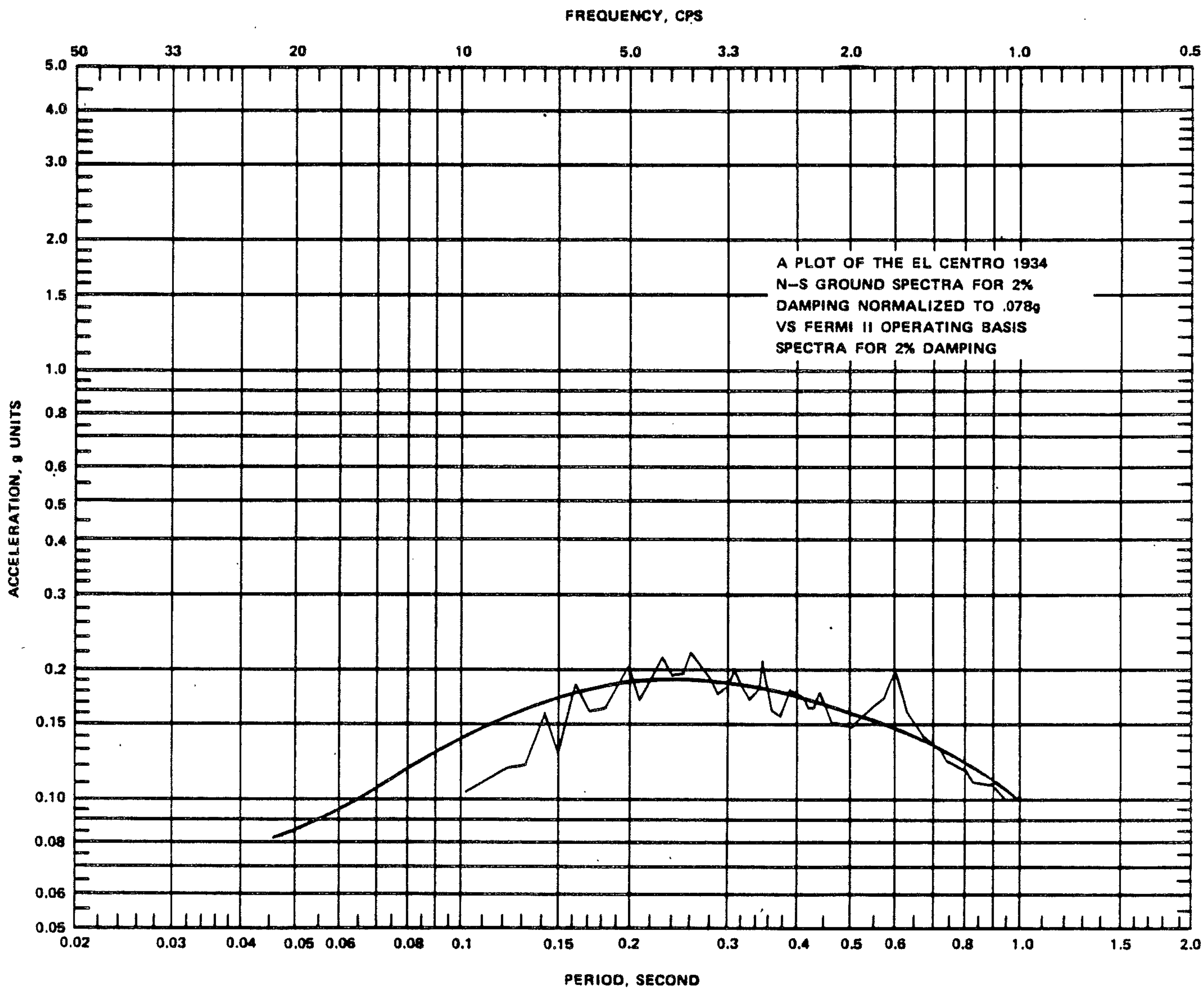

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FIGURE 3.7-5

EL CENTRO 1934 – N-S GROUND SPECTRA

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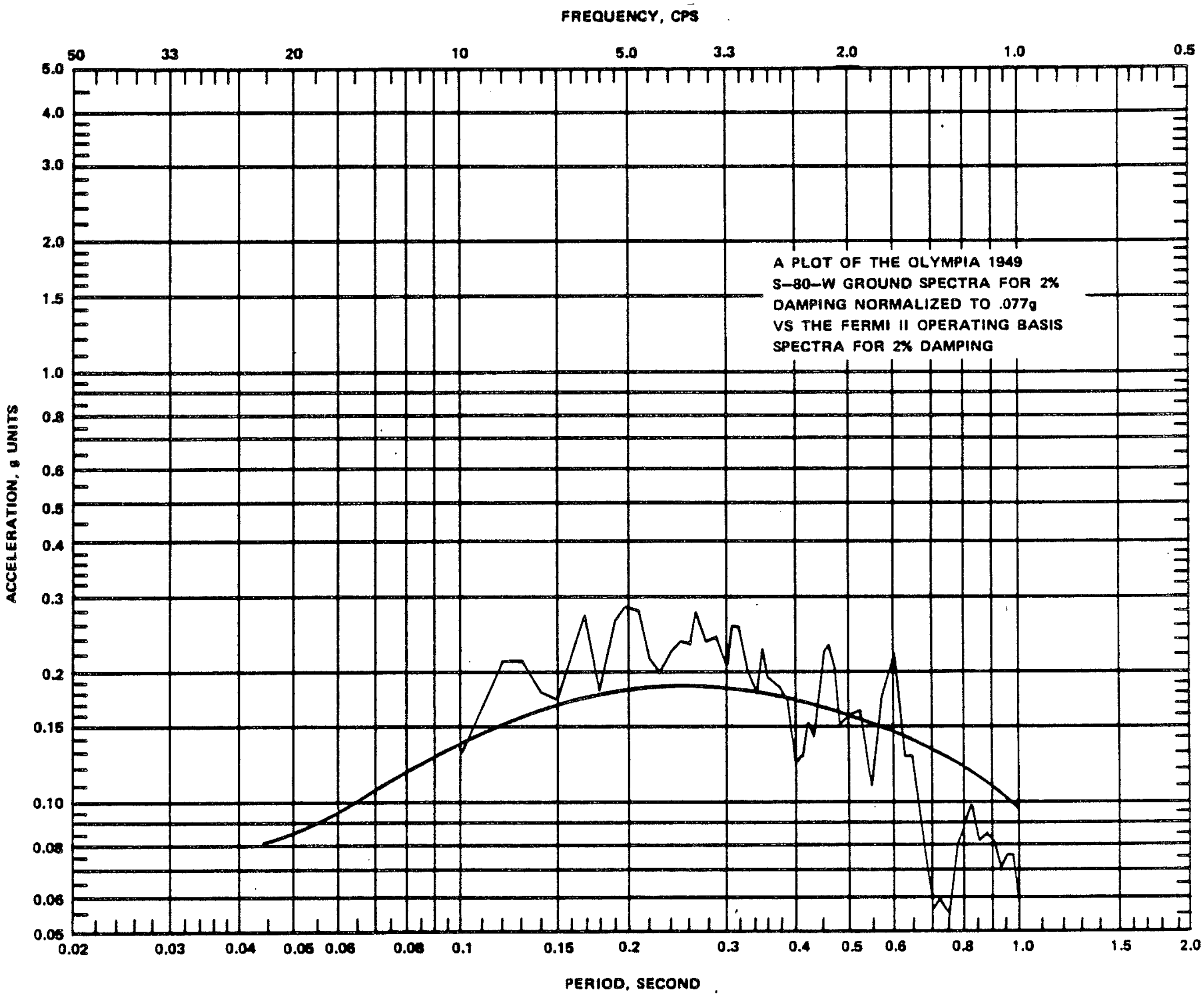

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FIGURE 3.7-6

OLYMPIA 1949 – S-80-W GROUND SPECTRA

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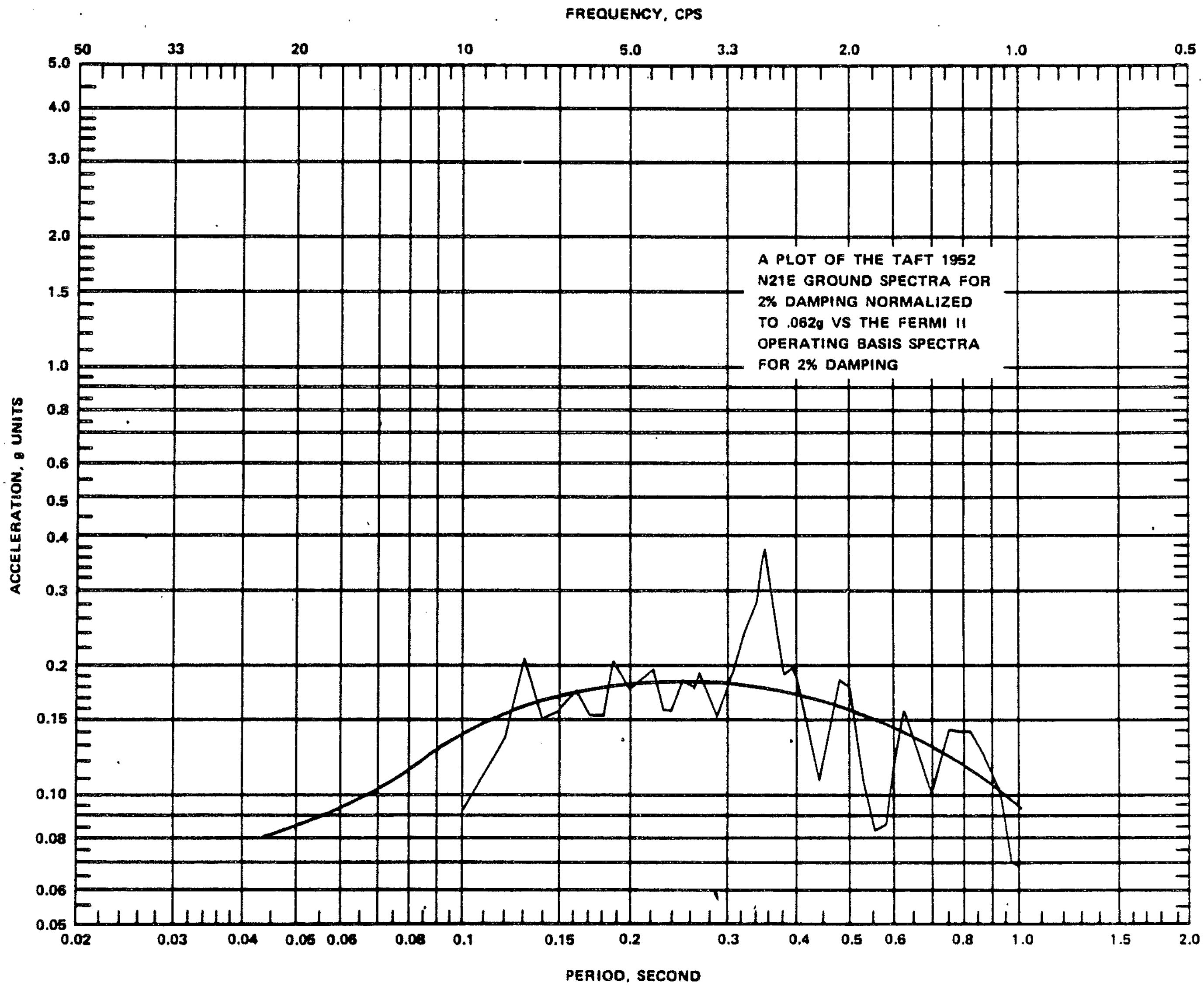

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FIGURE 3.7-7

TAFT 1952 – N-21-E GROUND SPECTRA

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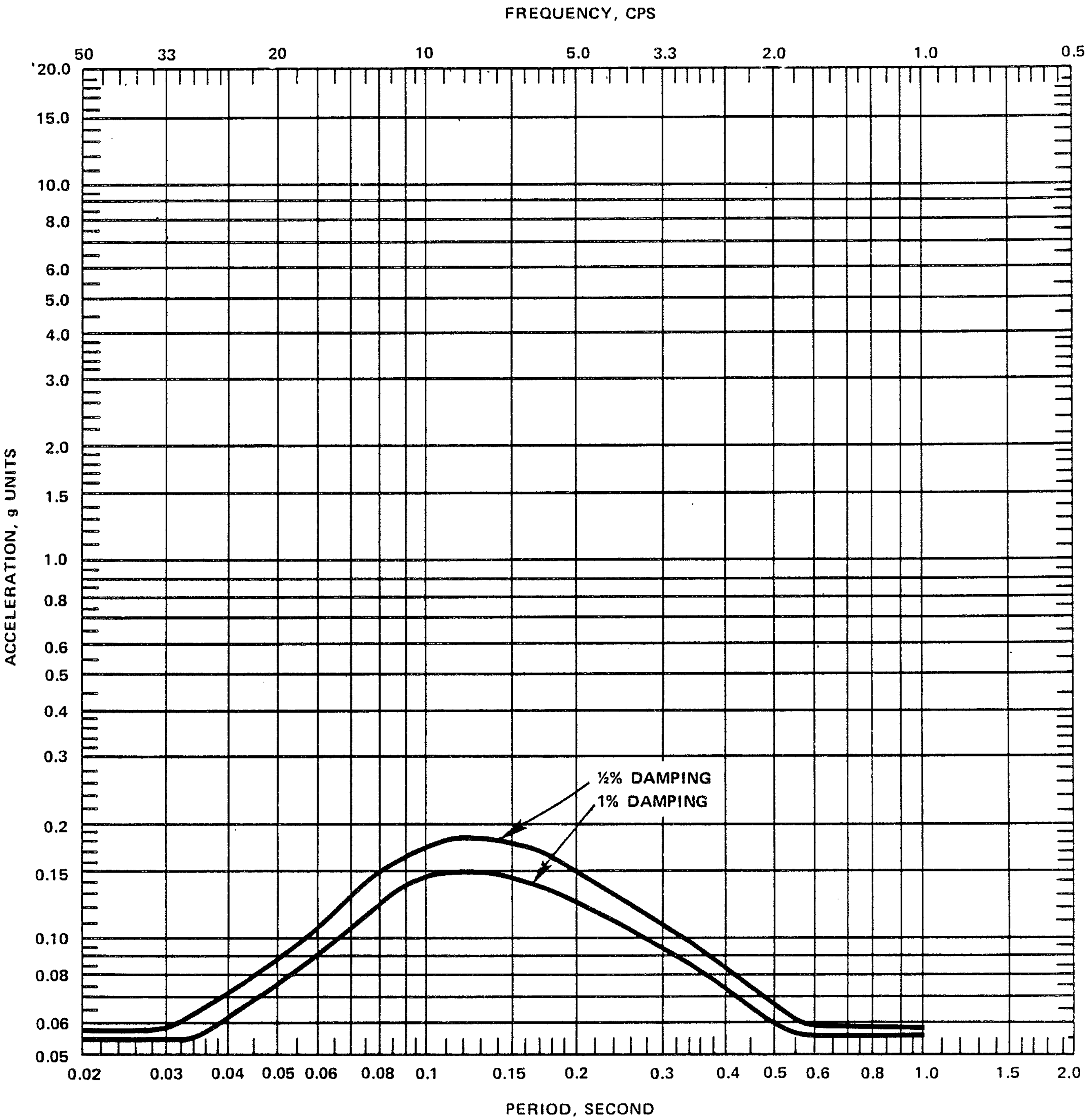

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FIGURE 3.7-8

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
GROUND LEVEL

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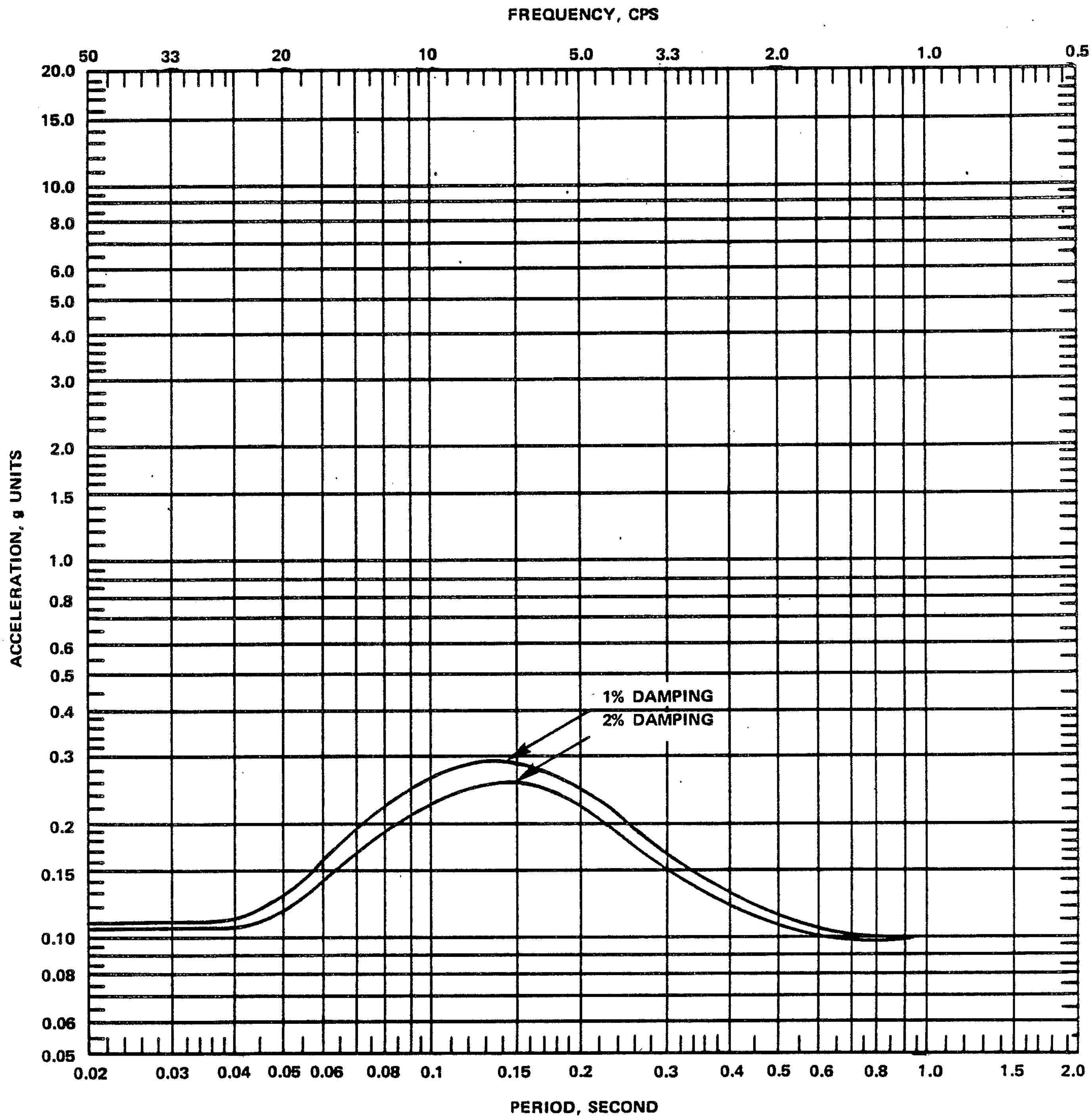

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FIGURE 3.7-9

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
GROUND LEVEL

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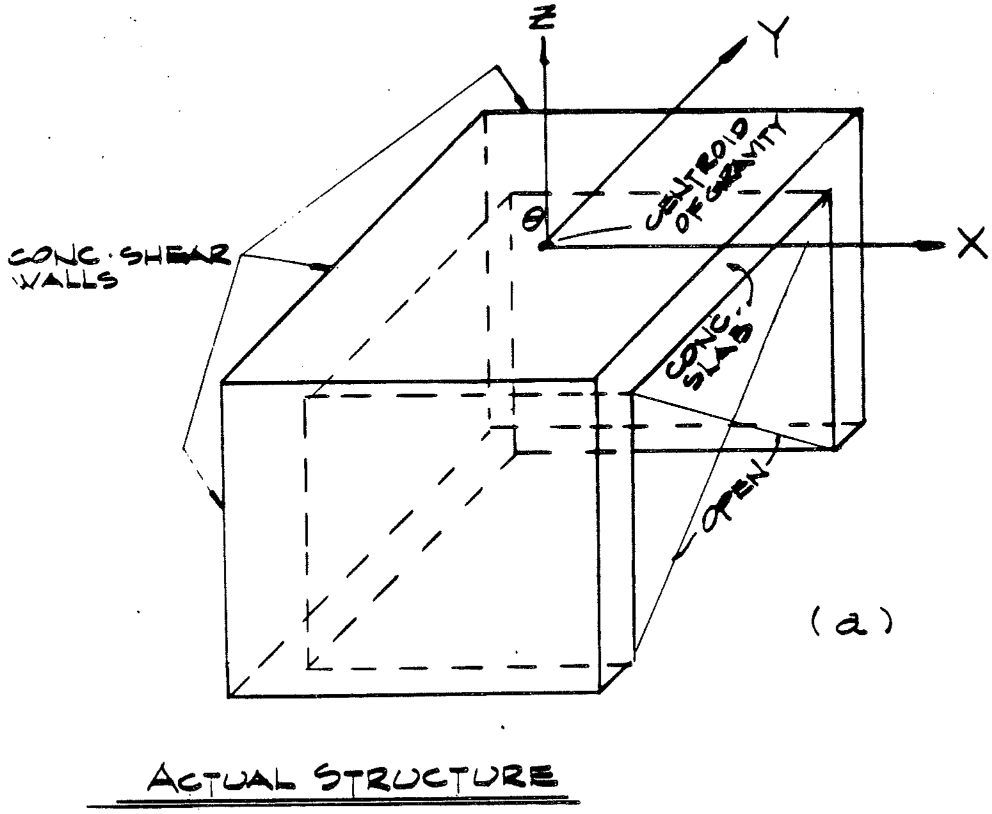

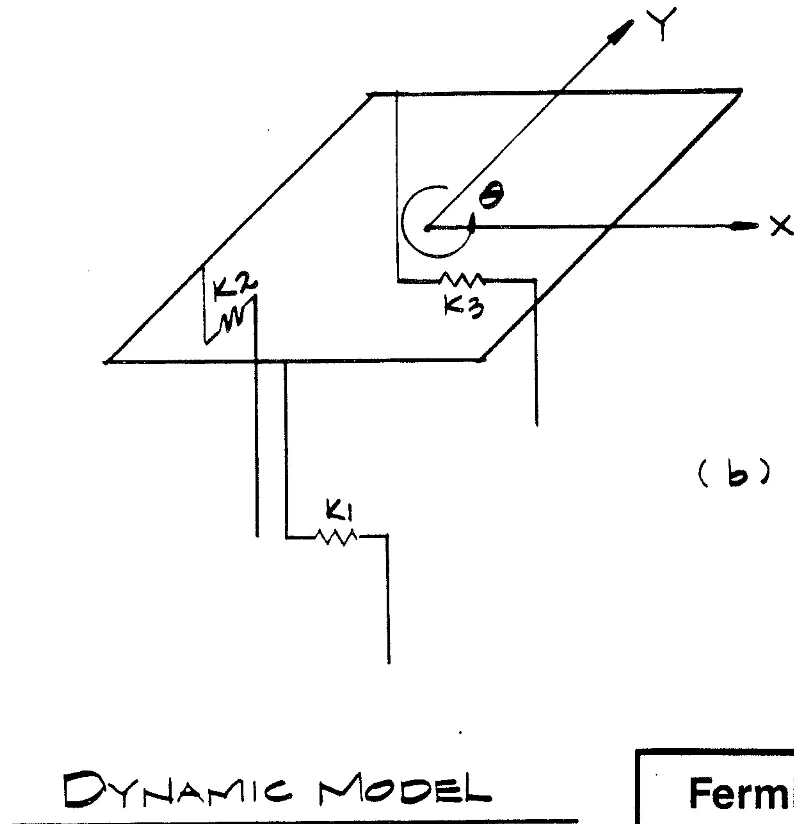

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FIGURE 3.7-10

TYPICAL SLAB SPRING DYNAMIC MODEL

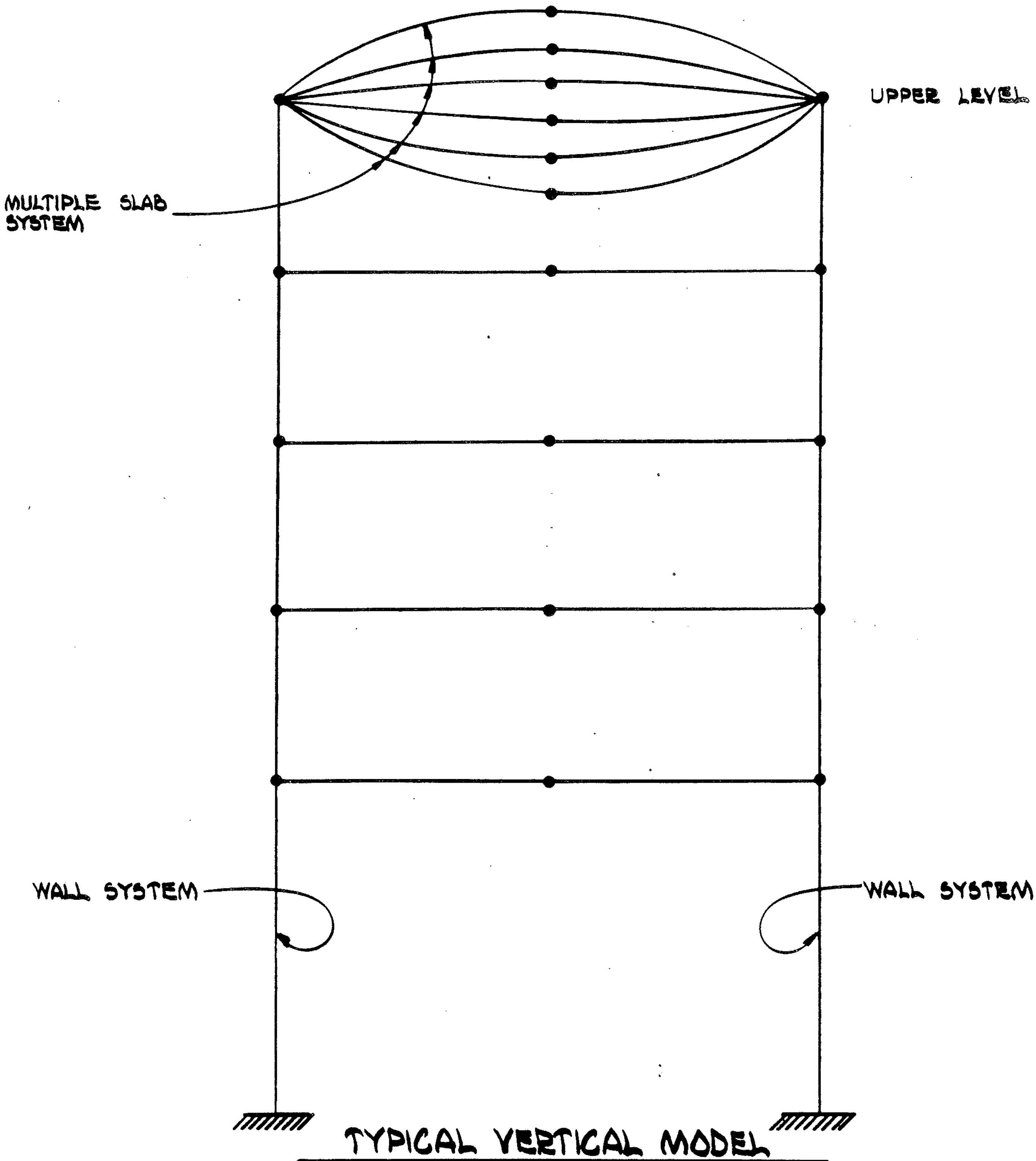

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FIGURE 3.7-11

TYPICAL VERTICAL MODEL

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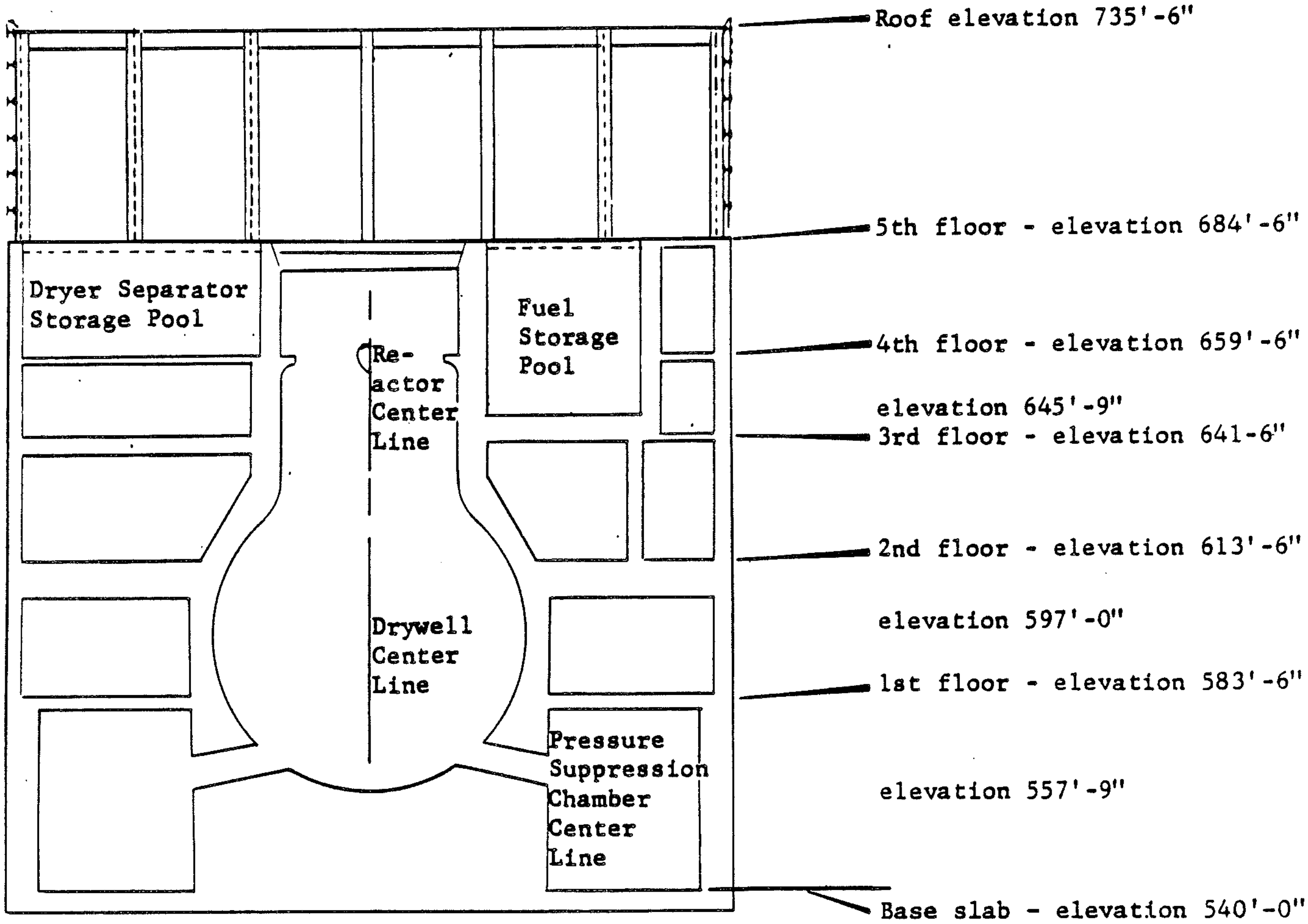

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FIGURE 3.7-12

REACTOR/AUXILIARY BUILDING – NORTH-SOUTH SECTION THROUGH REACTOR CENTERLINE LOOKING WEST

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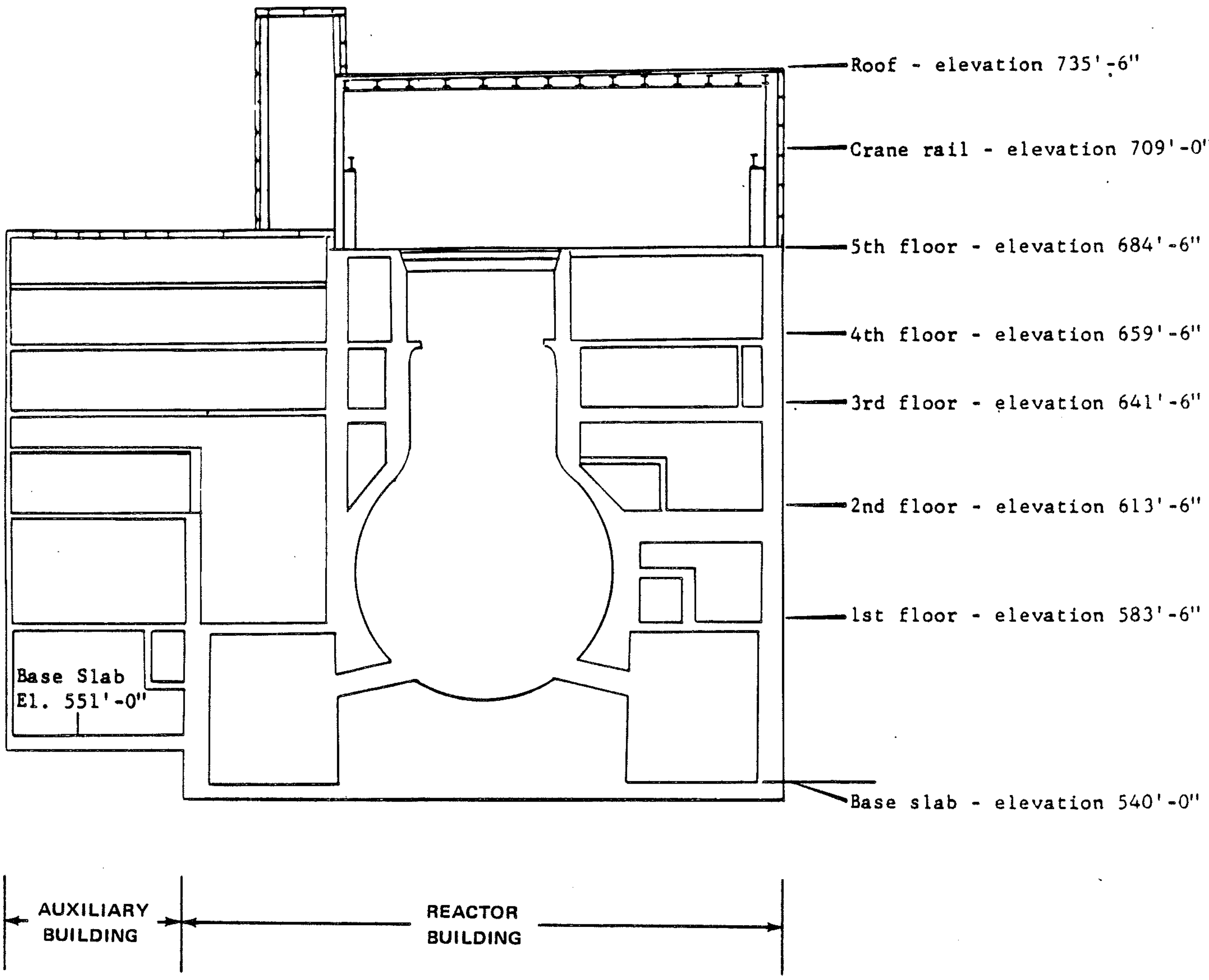

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FIGURE 3.7-13

REACTOR/AUXILIARY BUILDING – EAST-WEST SECTION THROUGH REACTOR CENTERLINE LOOKING SOUTH

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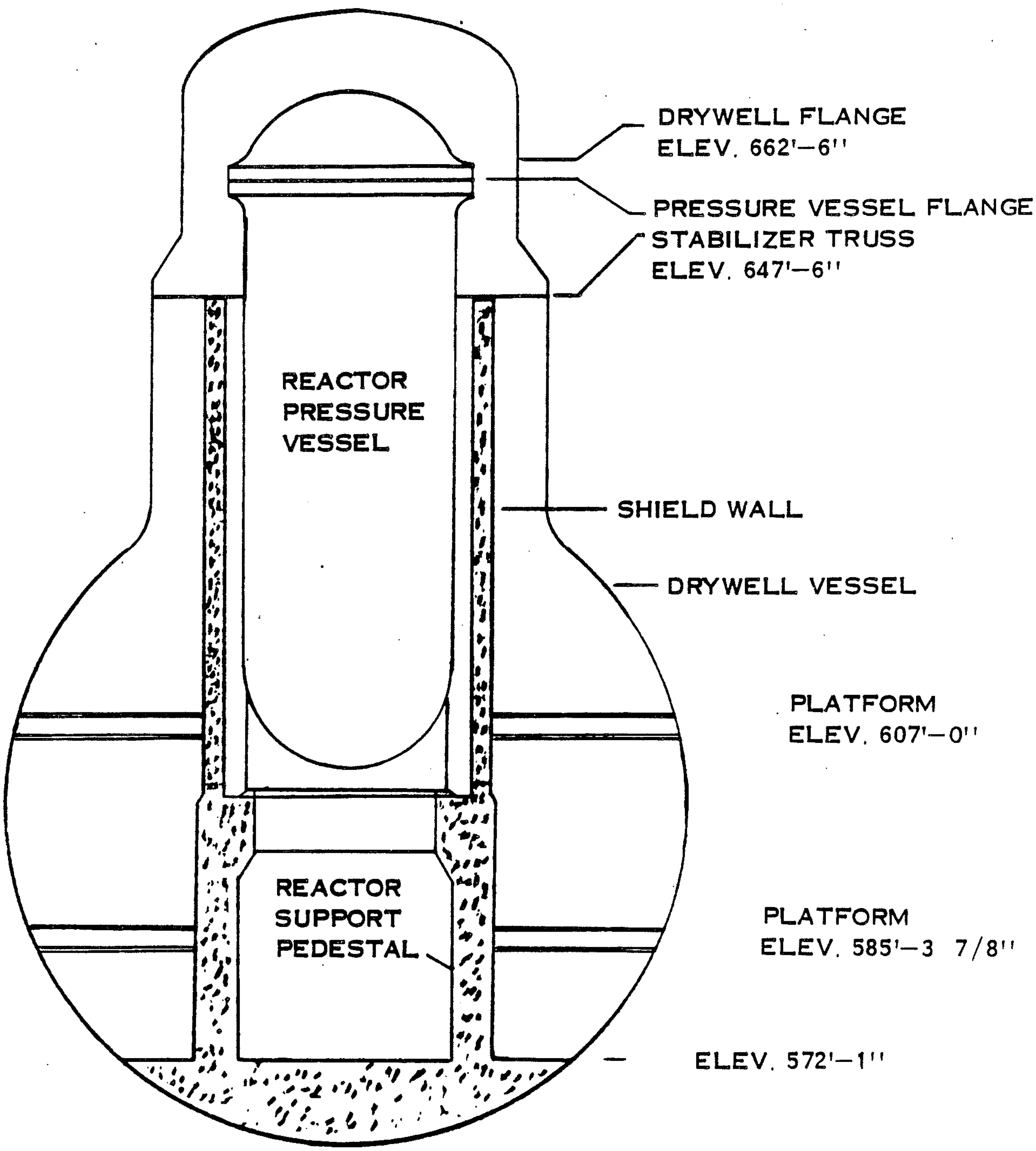

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FIGURE 3.7-14

SECTION THROUGH CENTERLINE OF DRYWELL

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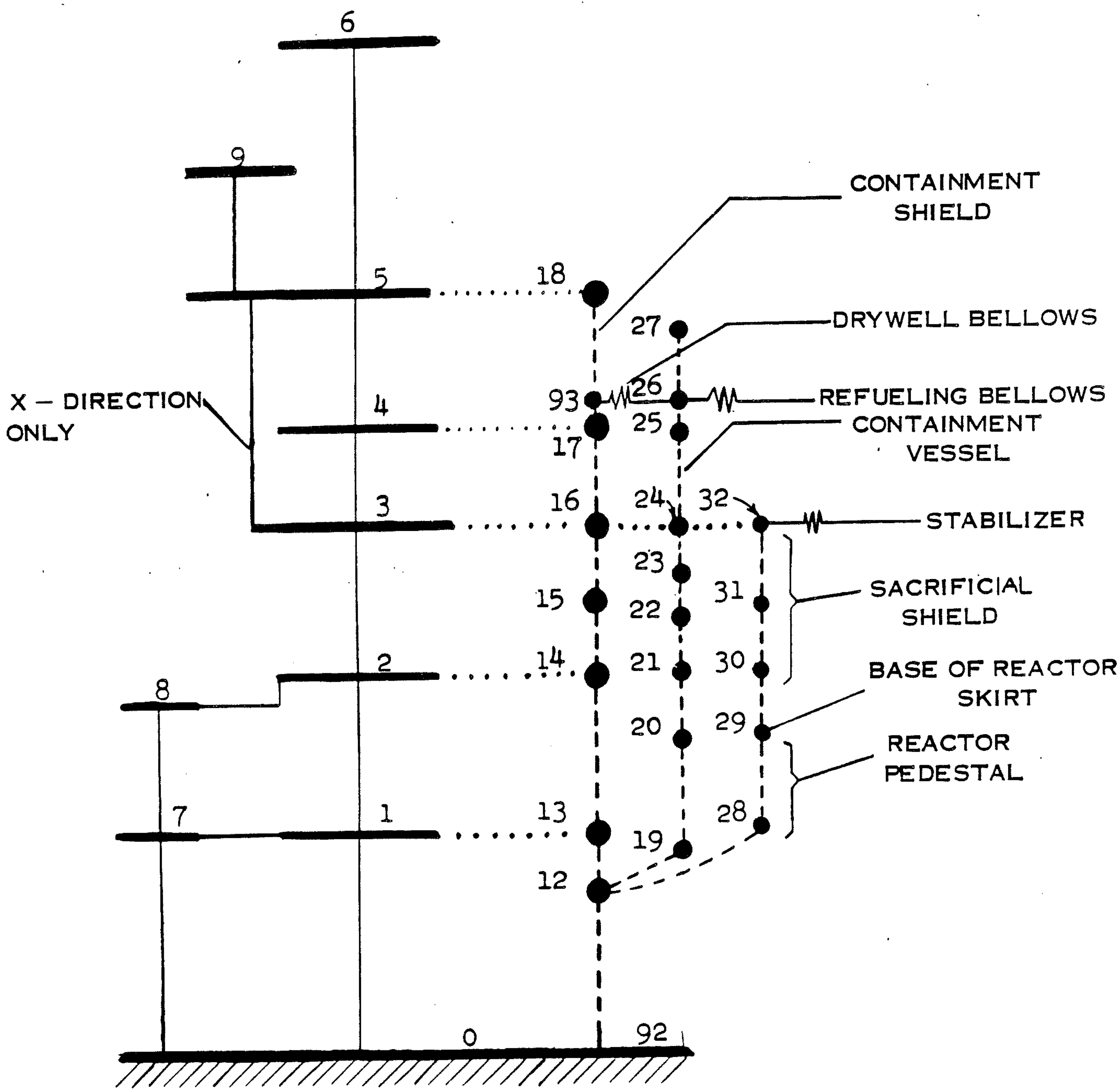

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FIGURE 3.7-15

REACTOR/AUXILIARY BUILDING – HORIZONTAL DYNAMIC MODEL

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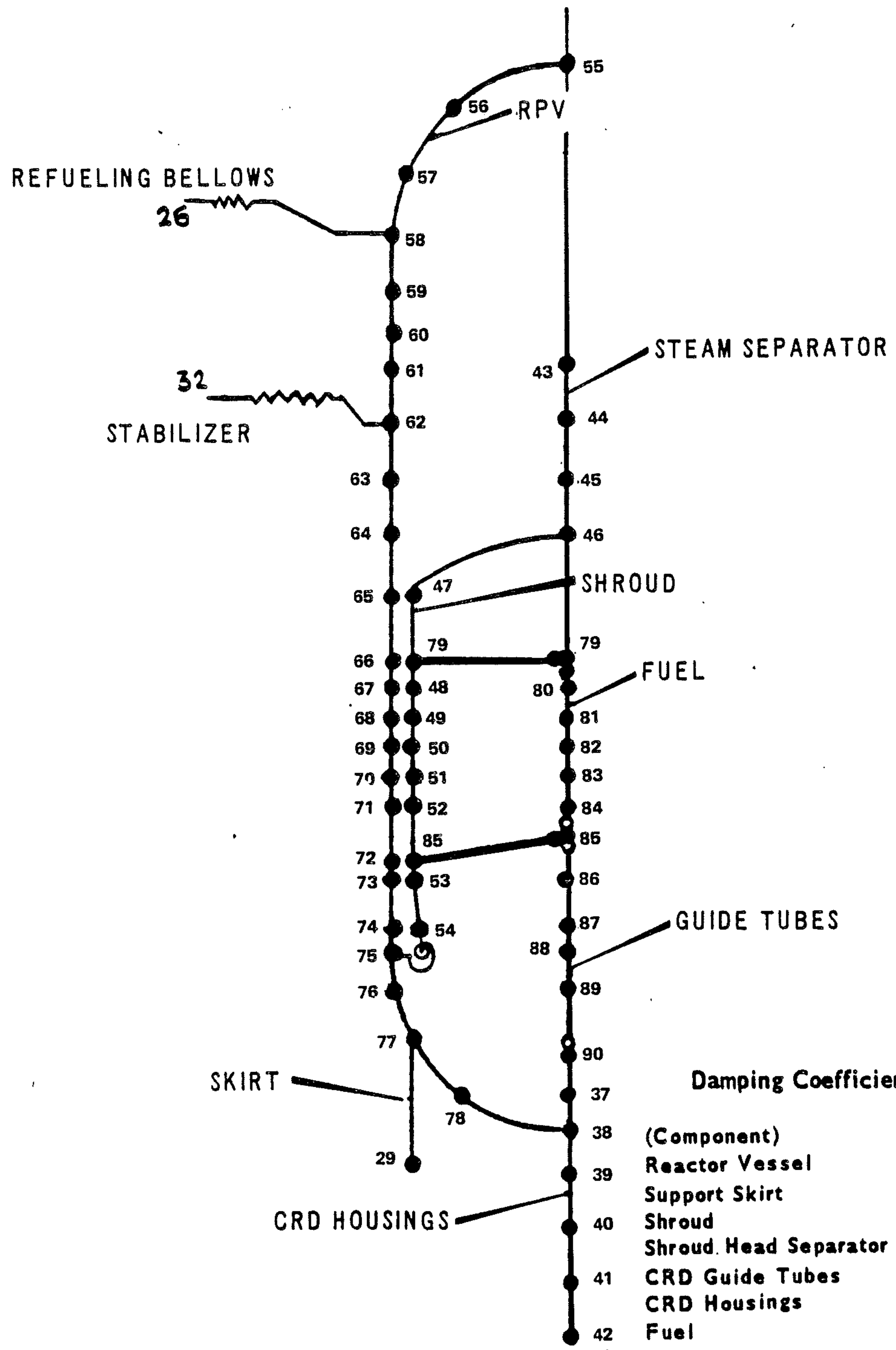

Damping Coefficients for OBE & DBE

(Component)	(% Critical Damping)
Reactor Vessel	2.0
Support Skirt	2.0
Shroud	2.0
Shroud Head Separator	2.0
CRD Guide Tubes	2.0
CRD Housings	3.5
Fuel	7.0

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FIGURE 3.7-16

MATHEMATICAL MODEL – REACTOR PRESSURE VESSEL AND INTERNALS

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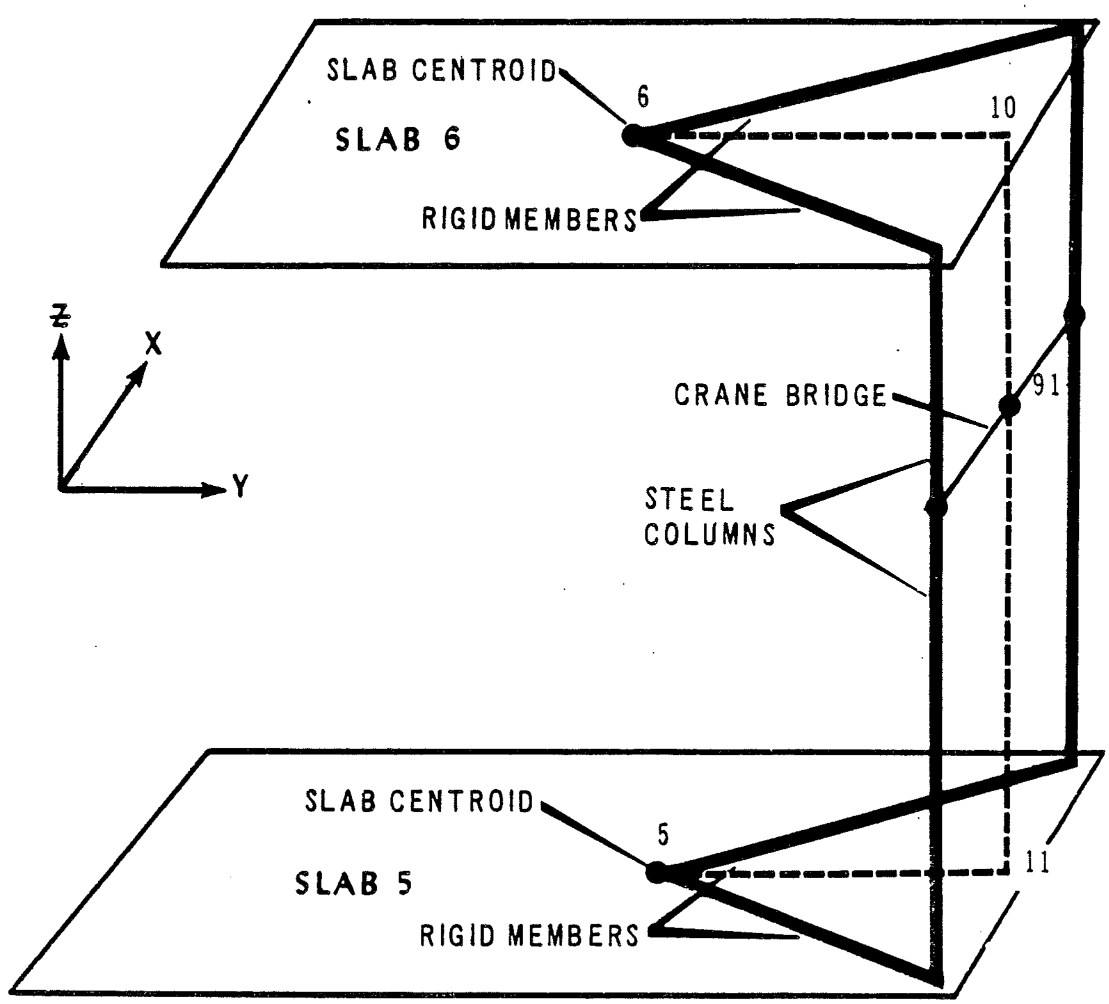

NOTE:

▬▬▬▬ SIMPLIFIED SEISMIC MODEL OF CRANE

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FIGURE 3.7-17

REACTOR/AUXILIARY BUILDING CRANE MODEL

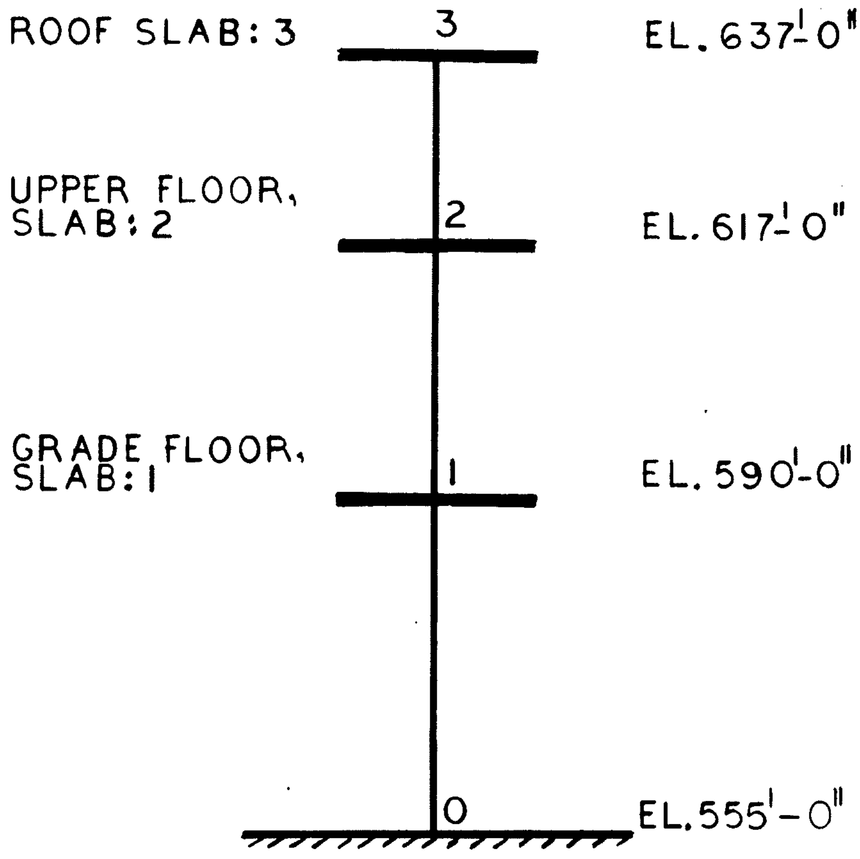

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FIGURE 3.7-18

RESIDUAL HEAT REMOVAL COMPLEX DYNAMIC MODEL FOR HORIZONTAL EXCITATION

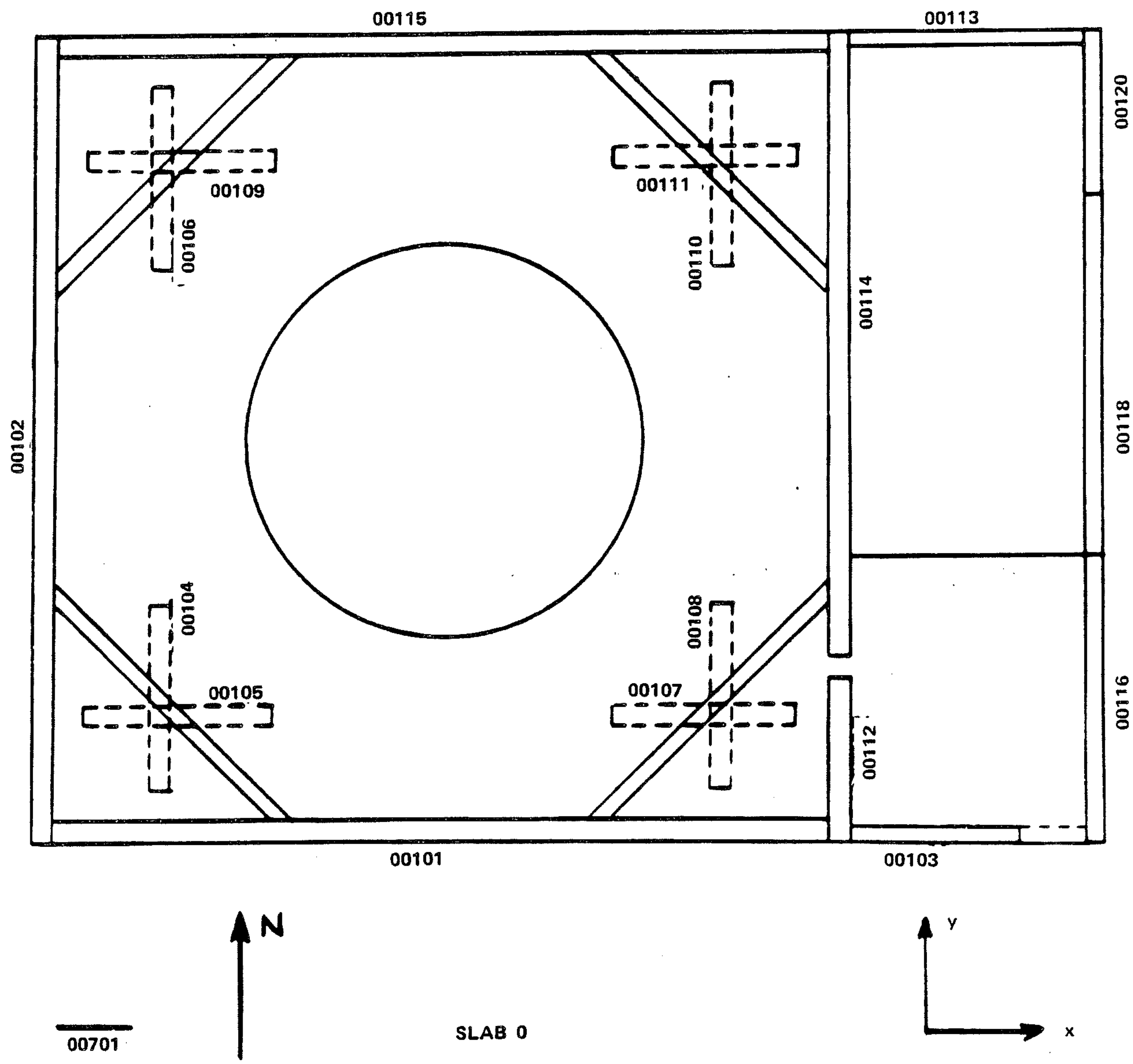

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FIGURE 3.7-19

SEISMIC ANALYSIS – REACTOR/AUXILIARY BUILDING BASE SLAB – ELEVATION 540.0 FT

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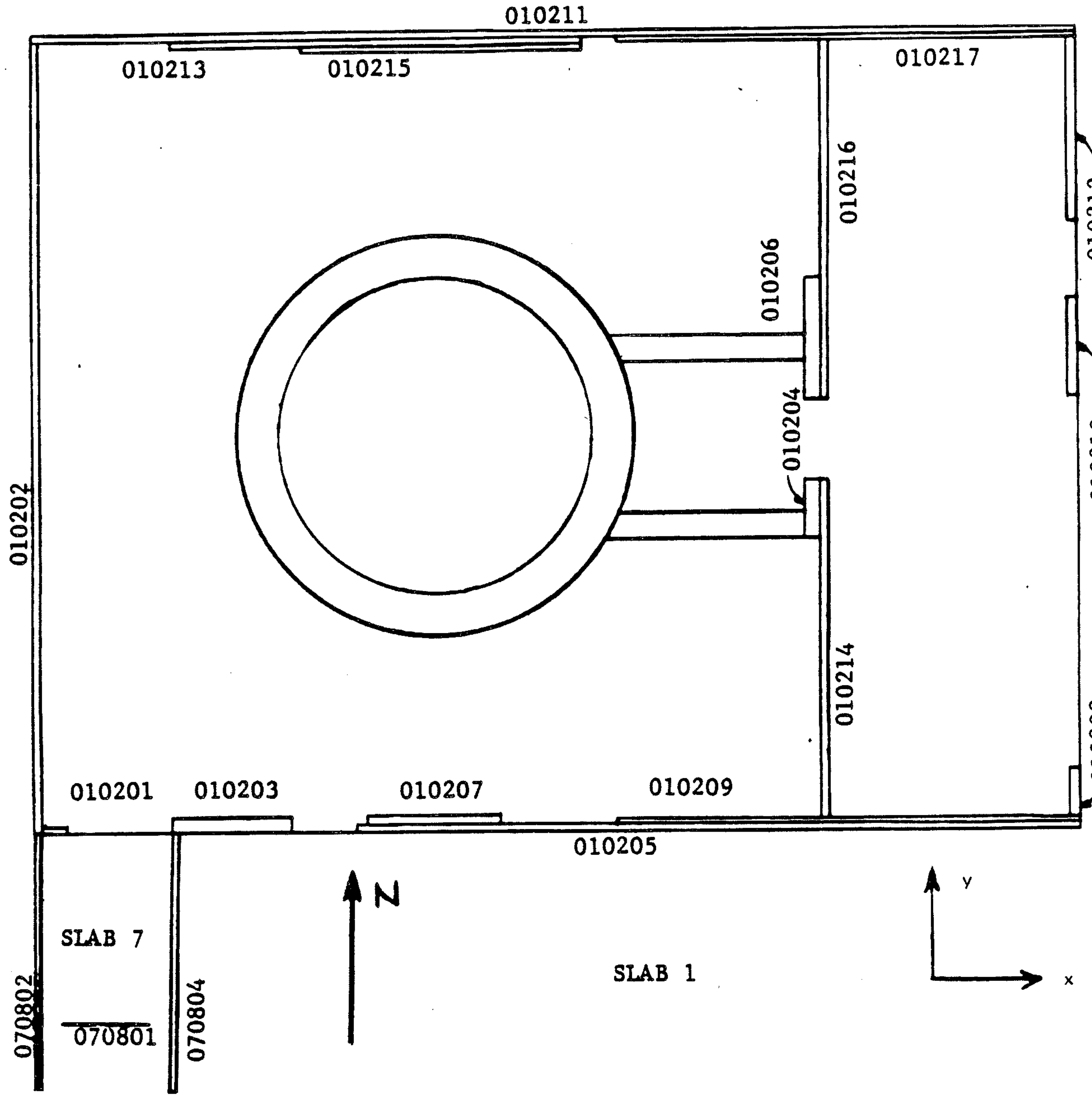

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FIGURE 3.7-20

SEISMIC ANALYSIS – REACTOR/AUXILIARY BUILDING FIRST FLOOR – ELEVATION 583.5 FT

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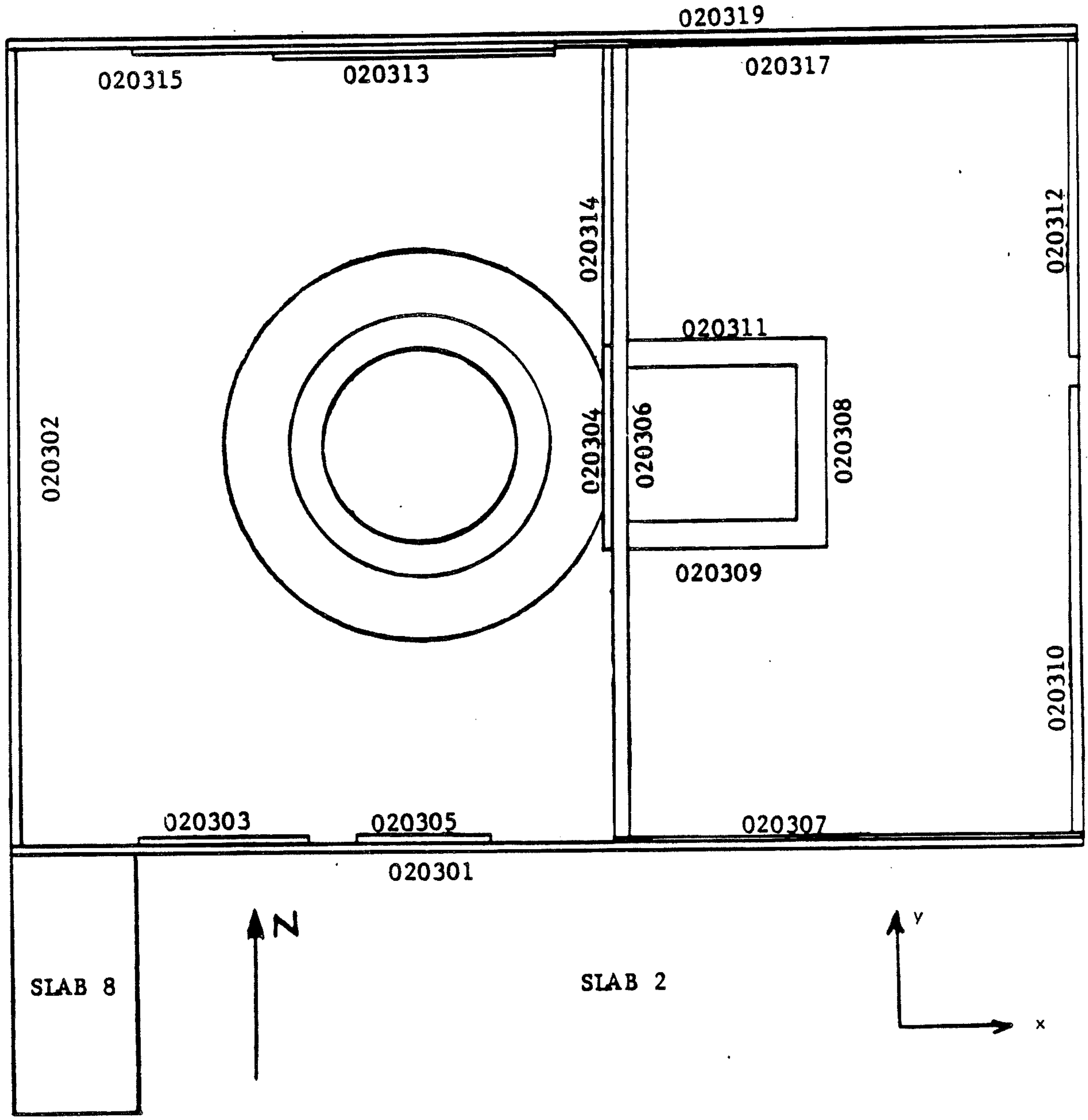

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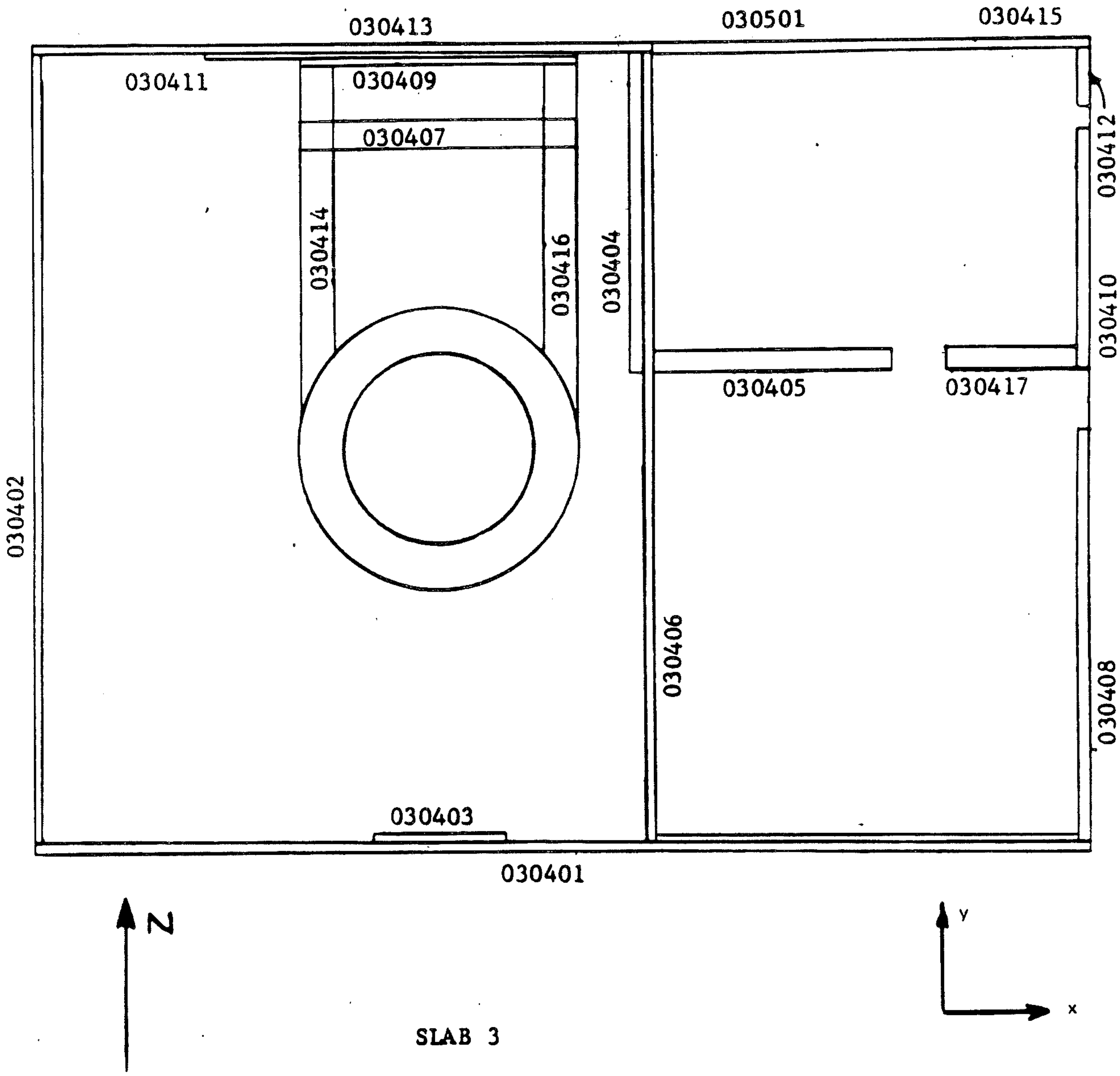
FIGURE 3.7-21

SEISMIC ANALYSIS – REACTOR/AUXILIARY BUILDING SECOND FLOOR – ELEVATION 613.5 FT

SARGENT & LUNDY REPORT NO. SL-2682



SLAB 3

Fermi 2
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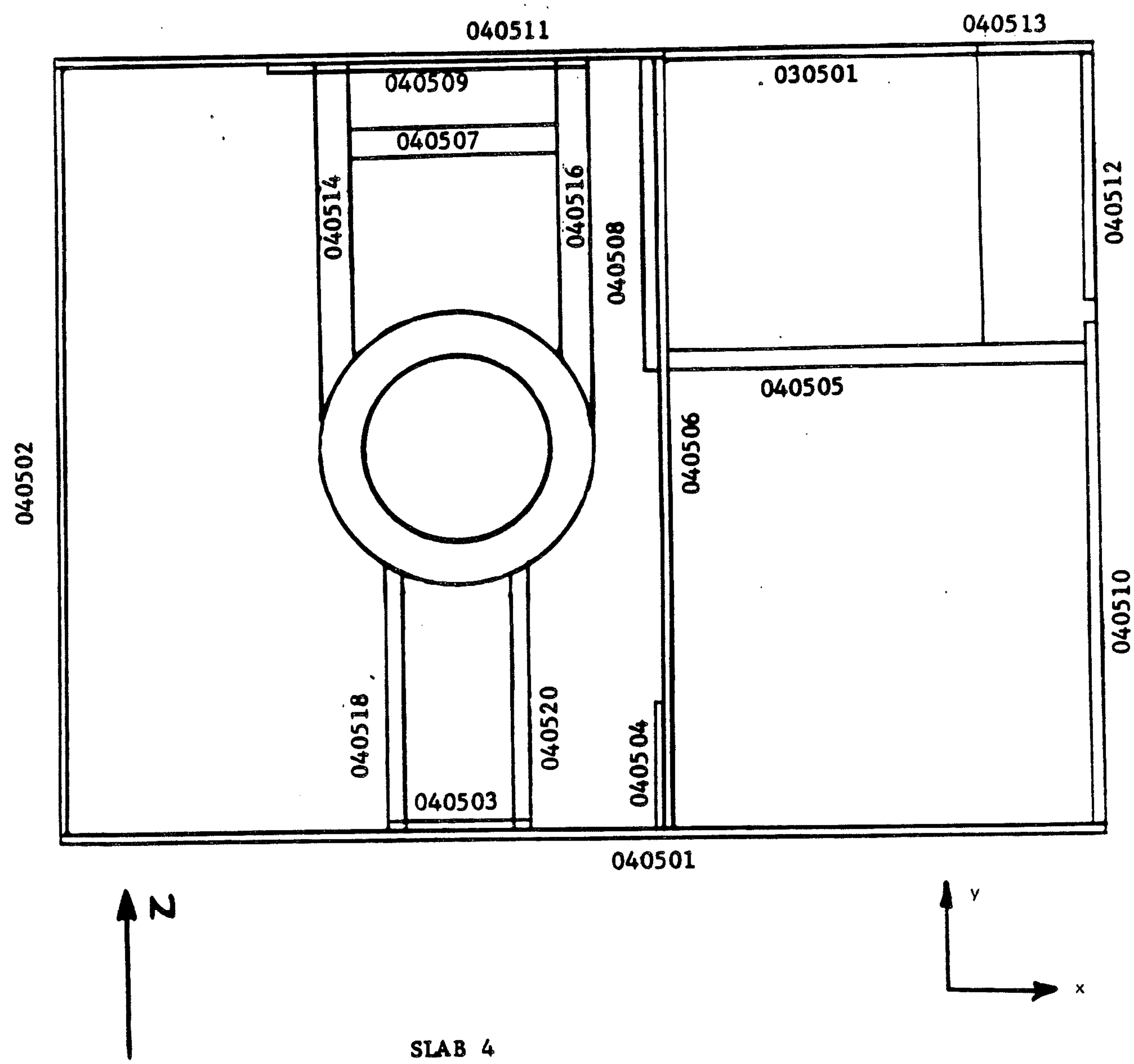
FIGURE 3.7-22

SEISMIC ANALYSIS – REACTOR/AUXILIARY BUILDING THIRD FLOOR – ELEVATION 641.5 FT



SARGENT & LUNDY REPORT NO. SL-2682

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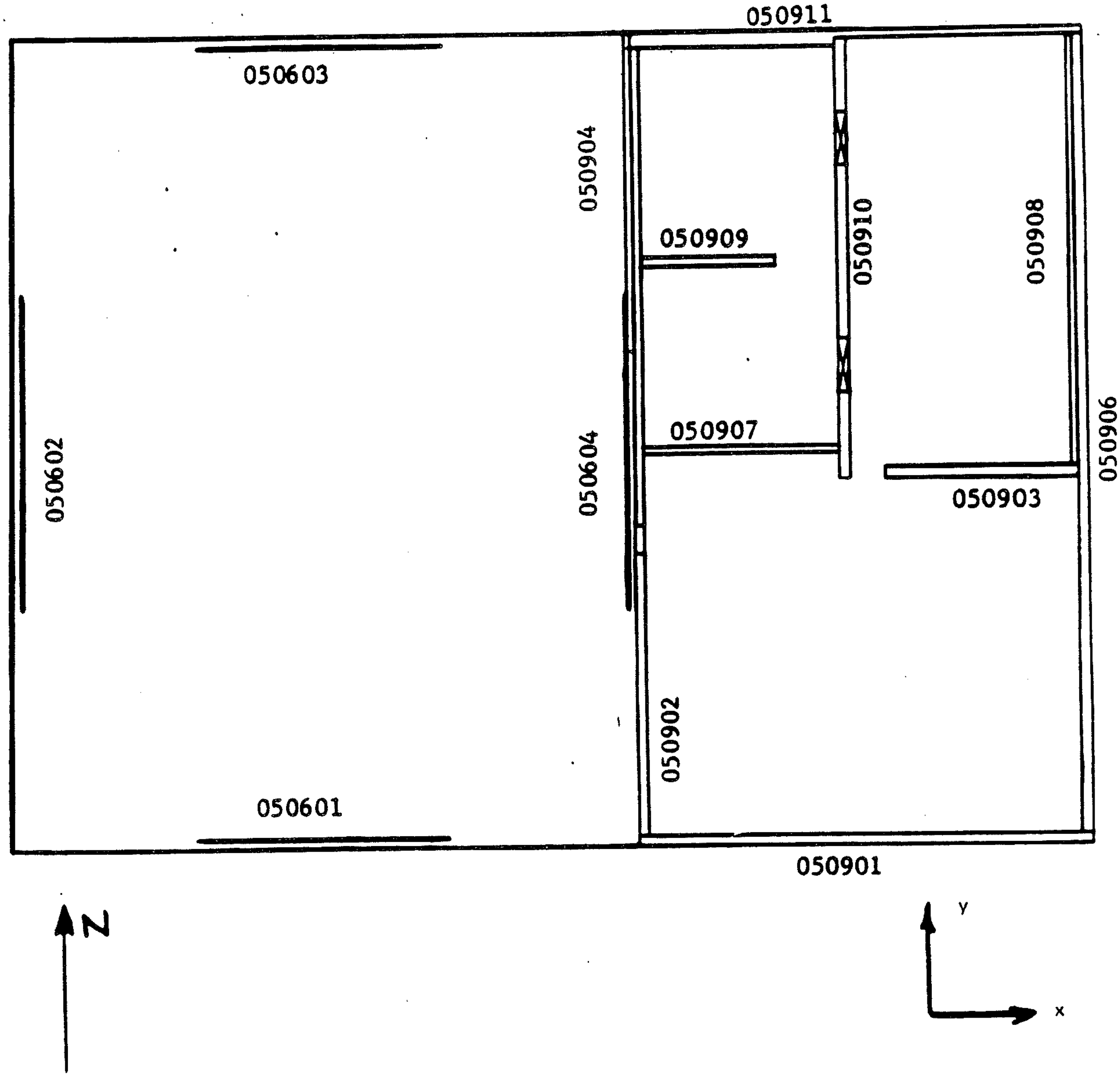
FIGURE 3.7-23

SEISMIC ANALYSIS – REACTOR/AUXILIARY BUILDING FOURTH FLOOR – ELEVATION 659.5 FT



SLAB 5

Fermi 2
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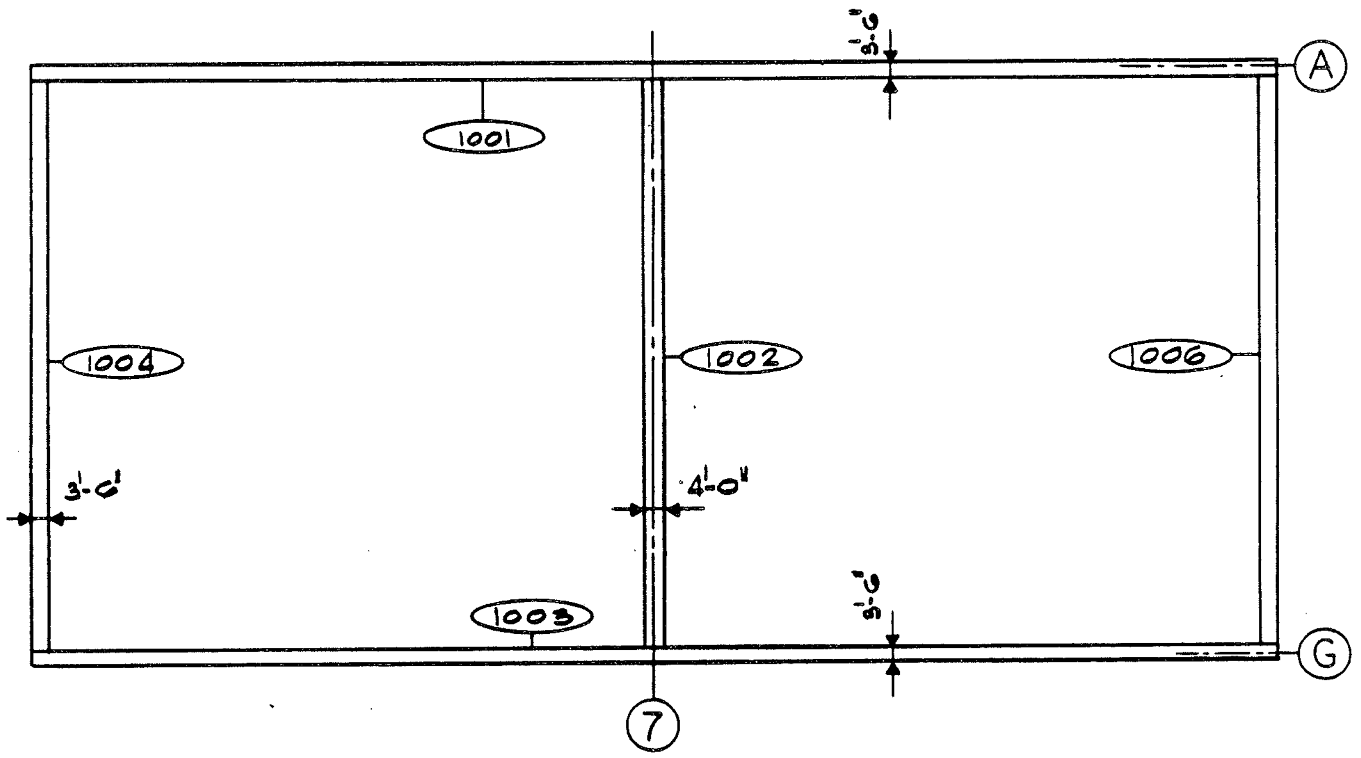
FIGURE 3.7-24

SEISMIC ANALYSIS – REACTOR/AUXILIARY BUILDING FIFTH FLOOR – ELEVATION 684.5 FT



BASEMENT FLOOR

SLAB-0 ELEV. 555'-0"

N

Fermi 2

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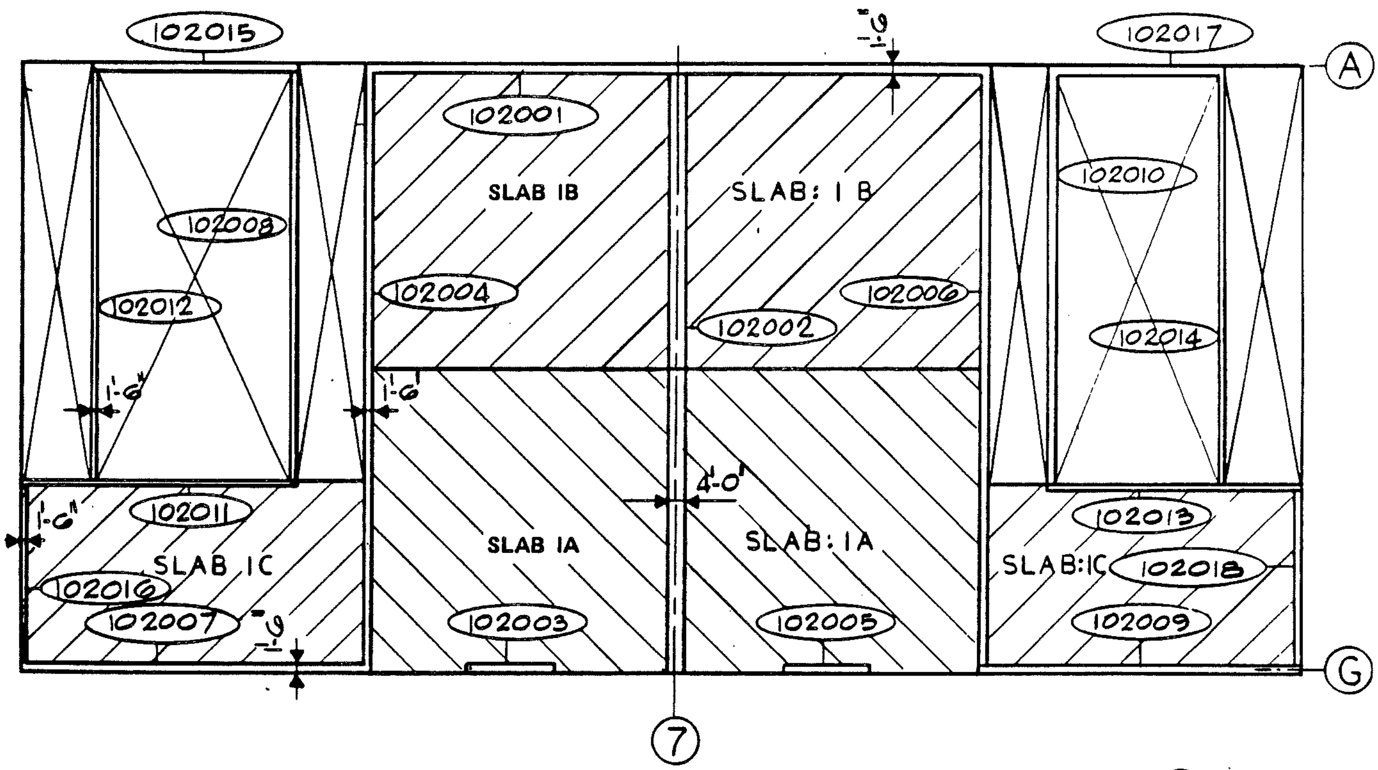
FIGURE 3.7-25

RESIDUAL HEAT REMOVAL COMPLEX BASEMENT FLOOR – SLAB 0 – ELEVATION 555.0 FT

SARGENT & LUNDY REPORT NO. SL-3147



GRADE FLOOR

SLAB-1 ELEV. 590'-0"

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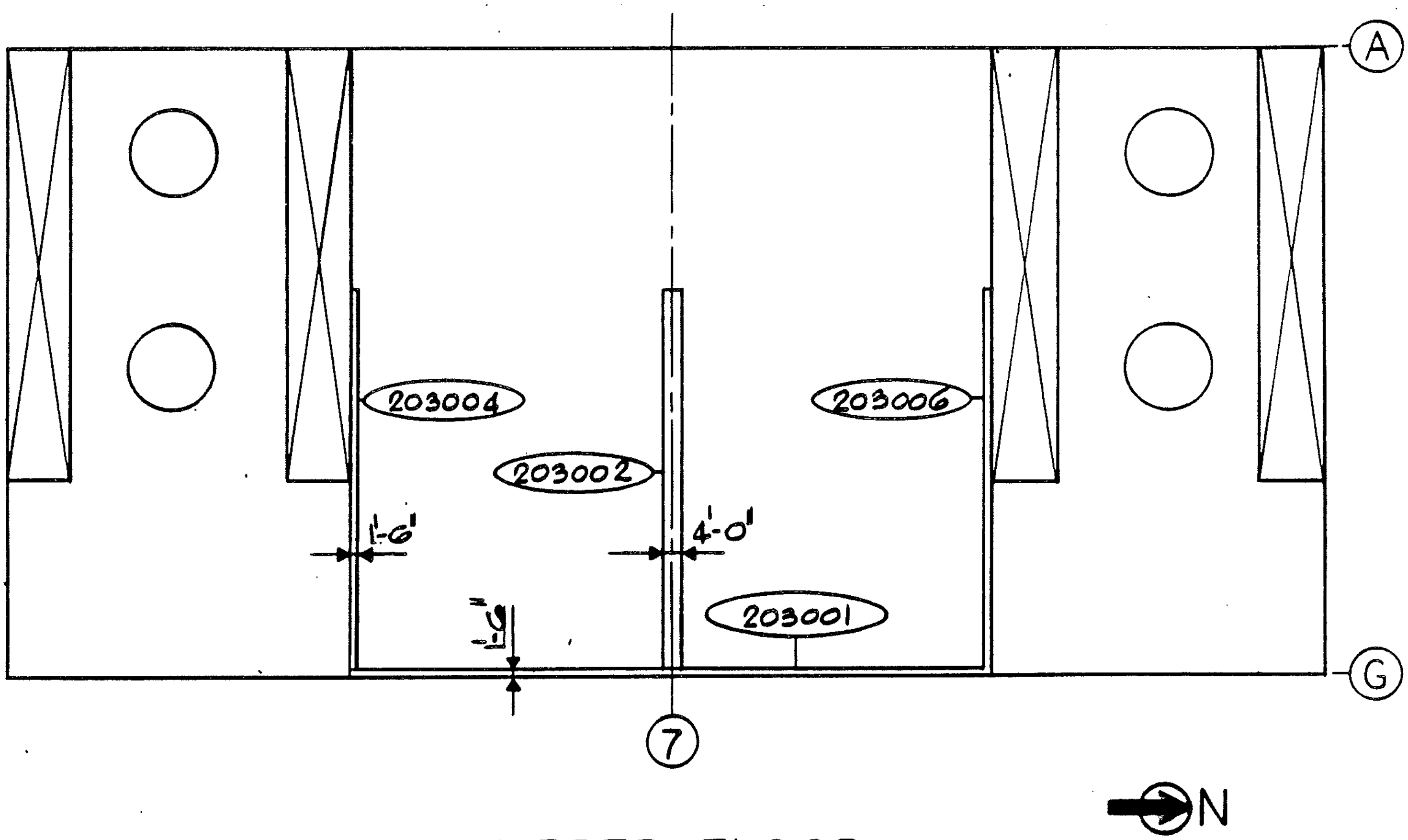
FIGURE 3.7-26

RESIDUAL HEAT REMOVAL COMPLEX GRADE FLOOR – SLAB 1 – ELEVATION 590.0 FT

SARGENT & LUNDY REPORT NO. SL-3147



UPPER FLOOR

SLAB 2 ELEV. 617'-0"

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FIGURE 3.7-27

RESIDUAL HEAT REMOVAL COMPLEX UPPER FLOOR – SLAB 2 – ELEVATION 617.0 FT

SARGENT & LUNDY REPORT NO. SL-3147

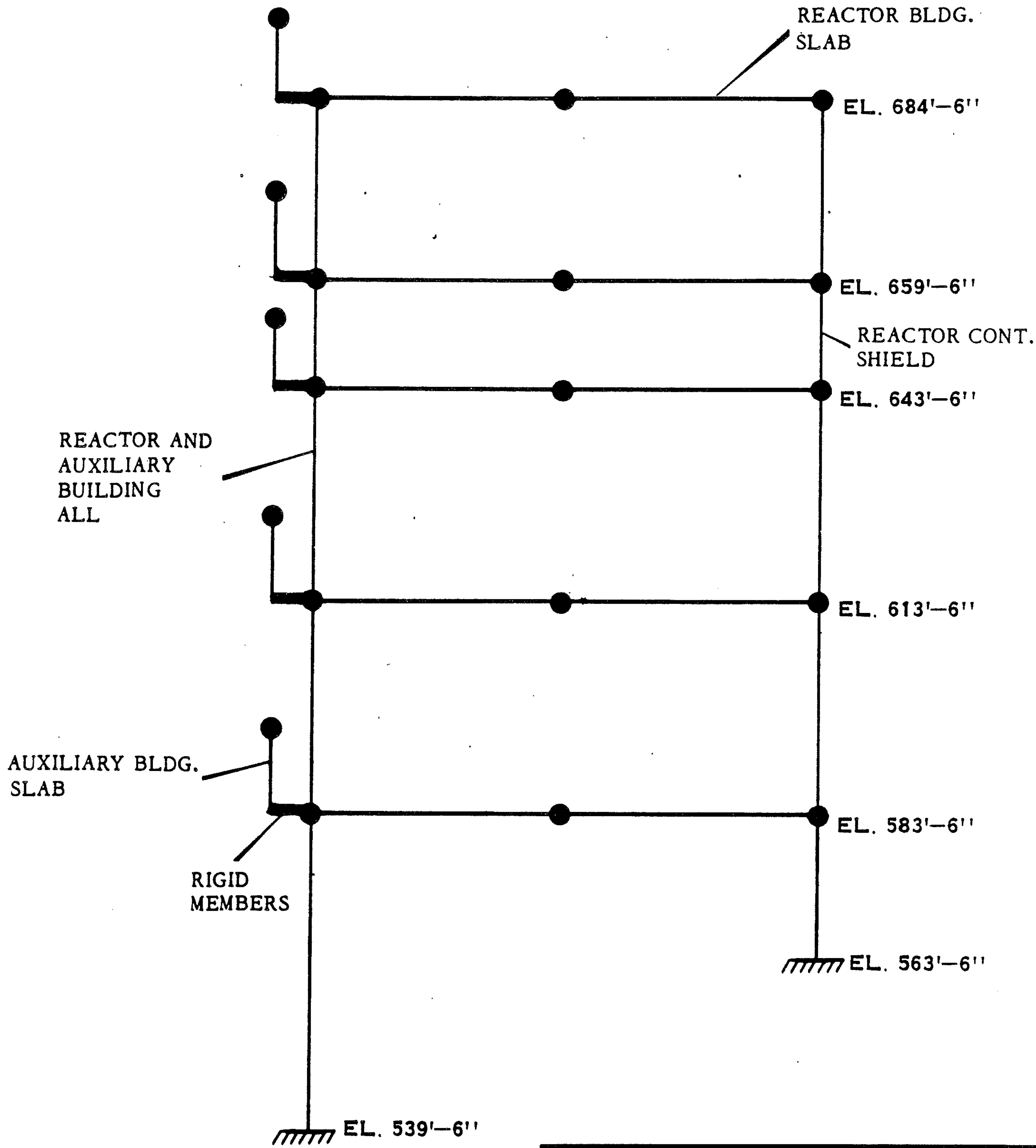

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FIGURE 3.7-28

REACTOR/AUXILIARY BUILDING VERTICAL MODEL GENERAL

SARGENT & LUNDY REPORT NO. SL-2682

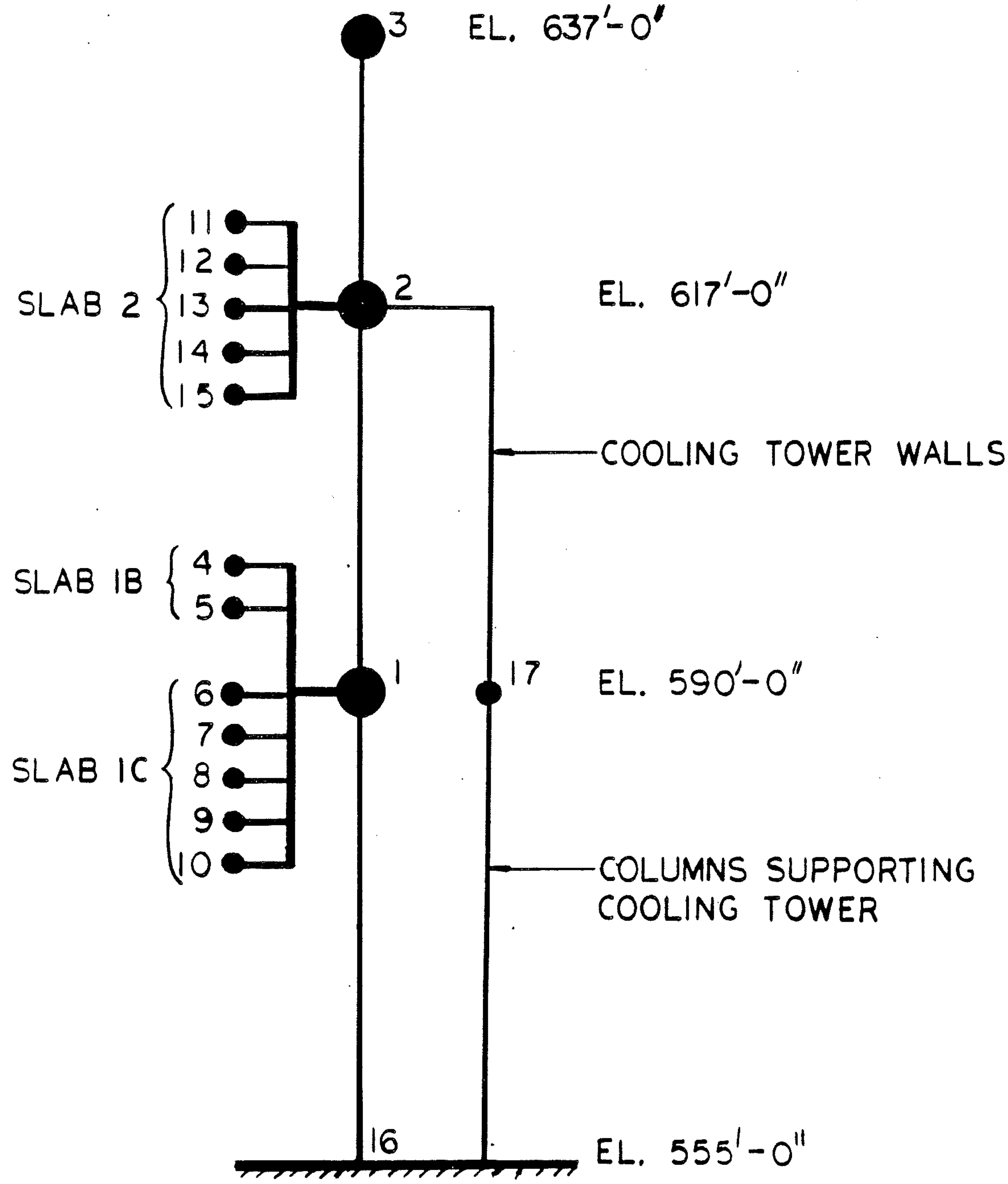

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FIGURE 3.7-29

RESIDUAL HEAT REMOVAL COMPLEX DYNAMIC MODEL FOR VERTICAL EXCITATION

SARGENT & LUNDY REPORT NO. SL-3147

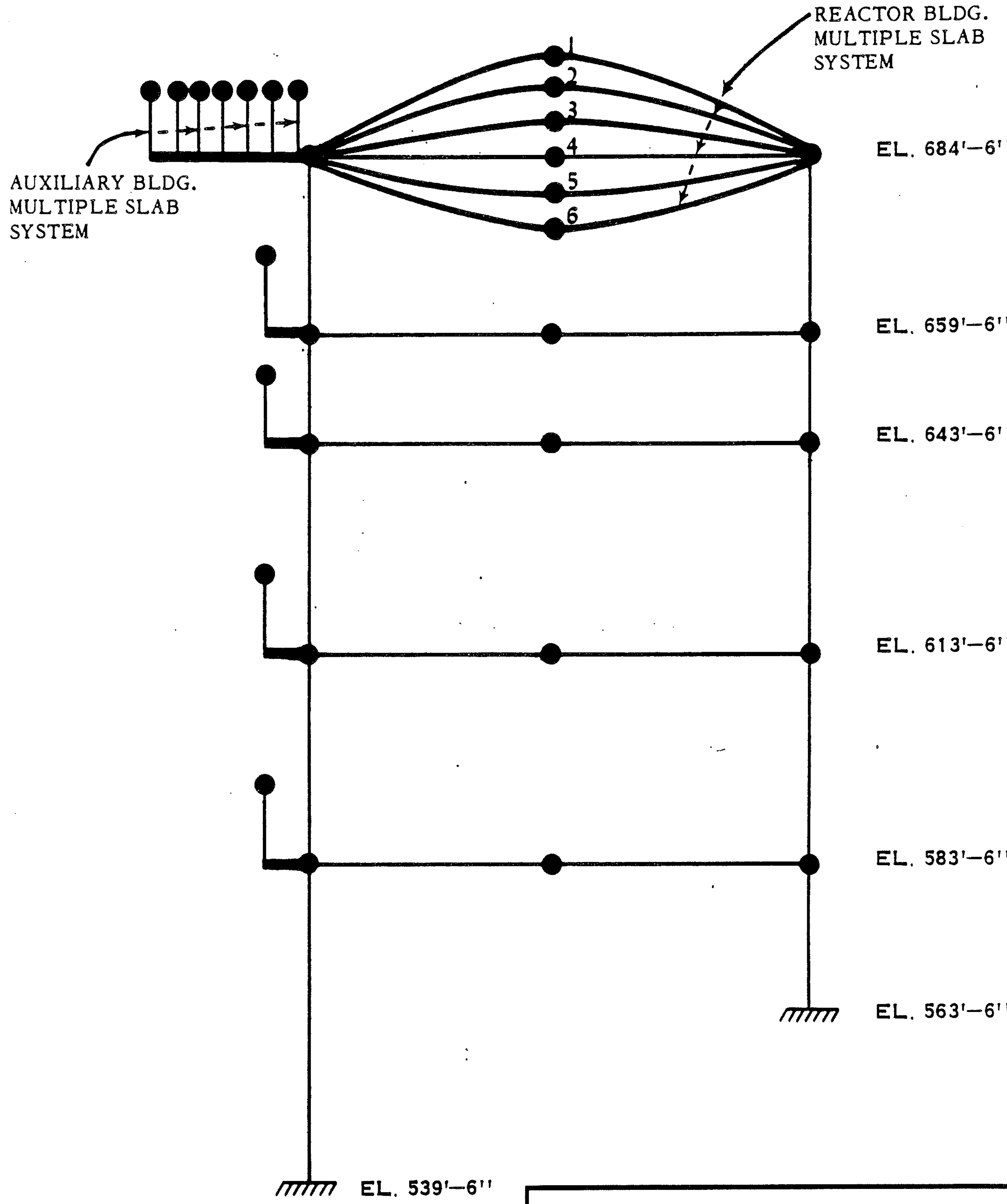

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FIGURE 3.7-30

REACTOR/AUXILIARY BUILDING VERTICAL MODEL FOR GENERATING SPECTRUM AT ELEVATION 684.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

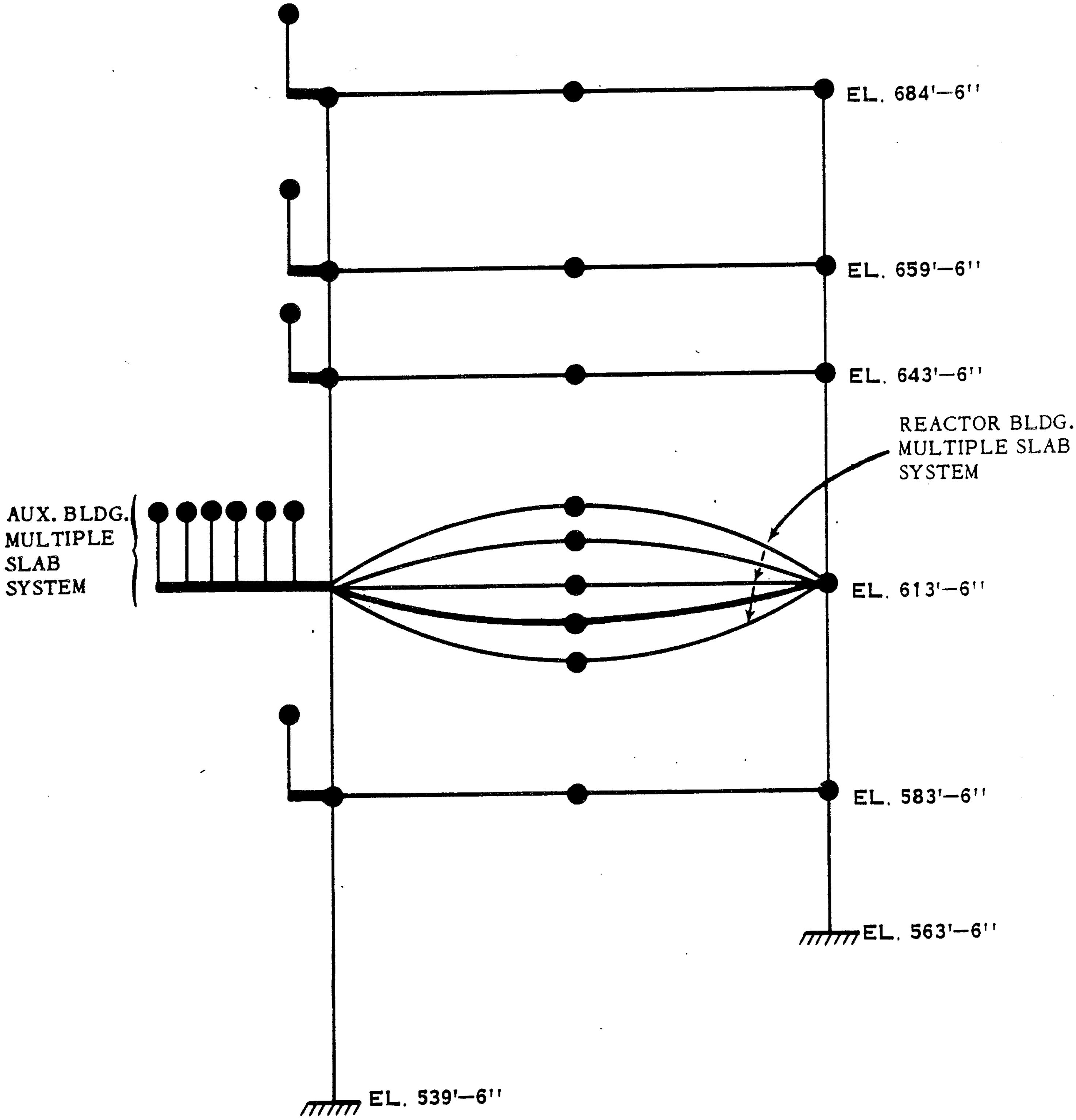

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FIGURE 3.7-31

REACTOR/AUXILIARY BUILDING VERTICAL MODEL FOR GENERATING SPECTRUM AT ELEVATION 613.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

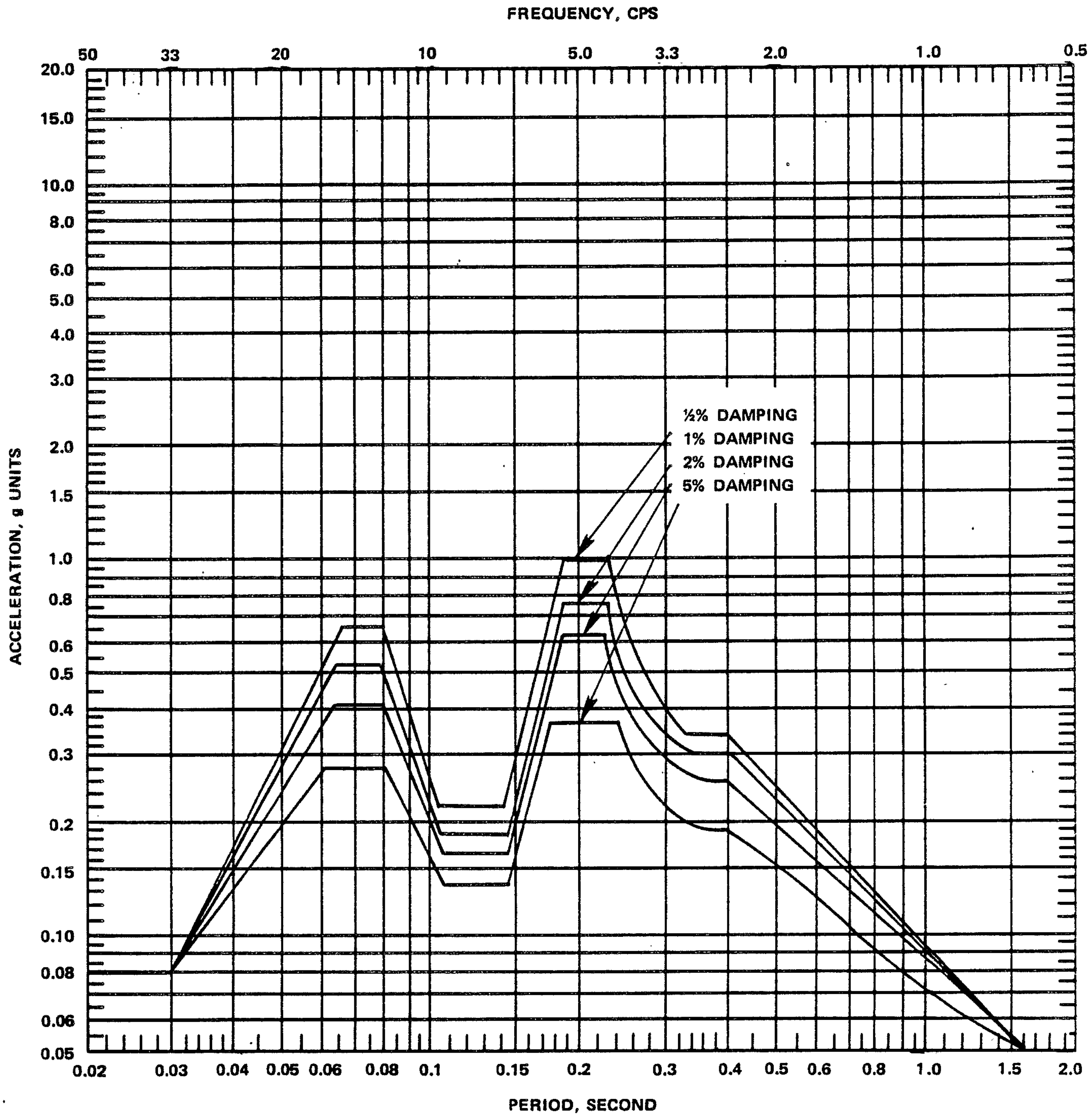

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FIGURE 3.7-32

HORIZONTAL FLOOR RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
ELEVATION 583.5 FT – SLAB 1
REACTOR/AUXILIARY BUILDING NORTH-SOUTH
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

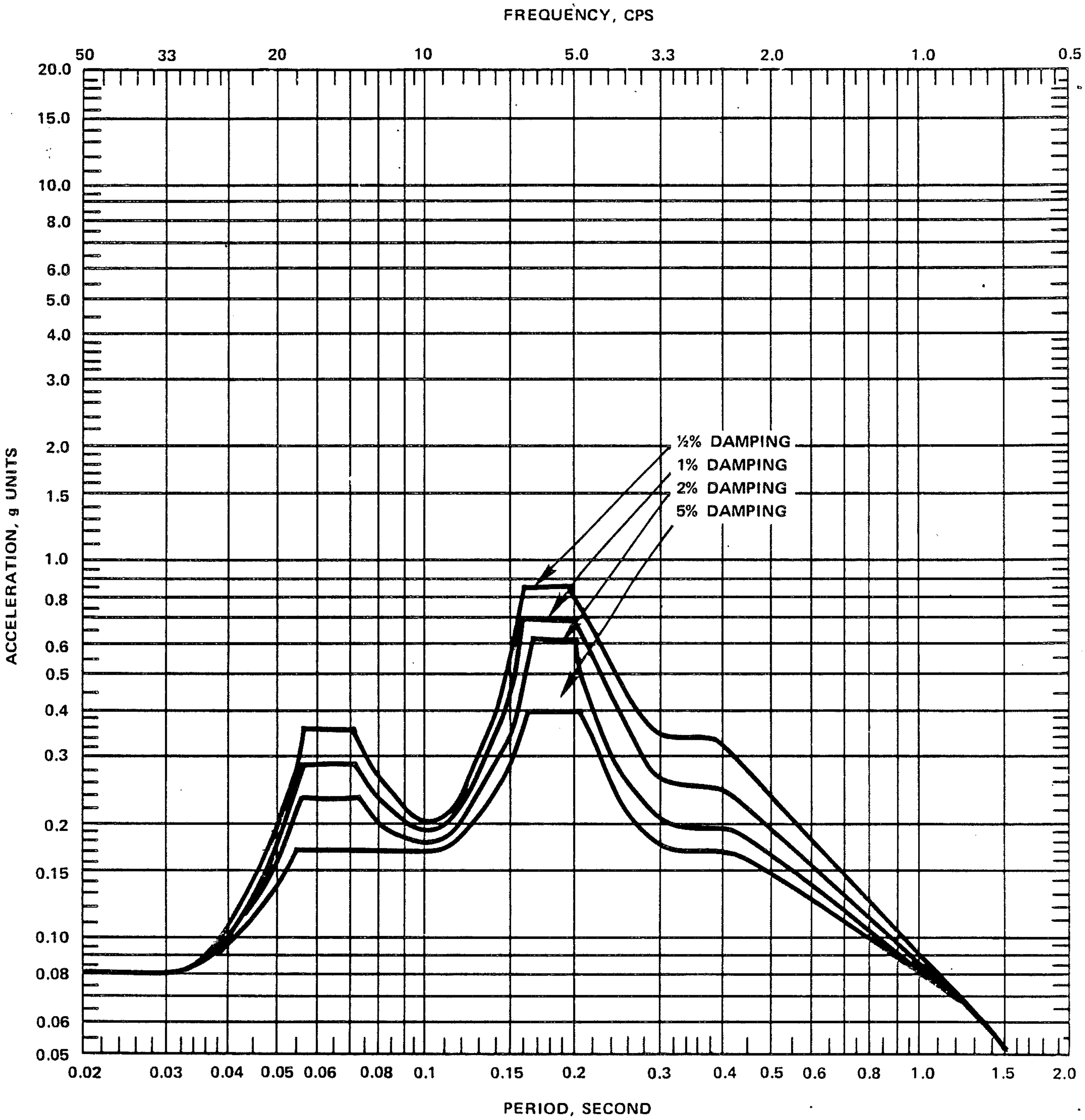

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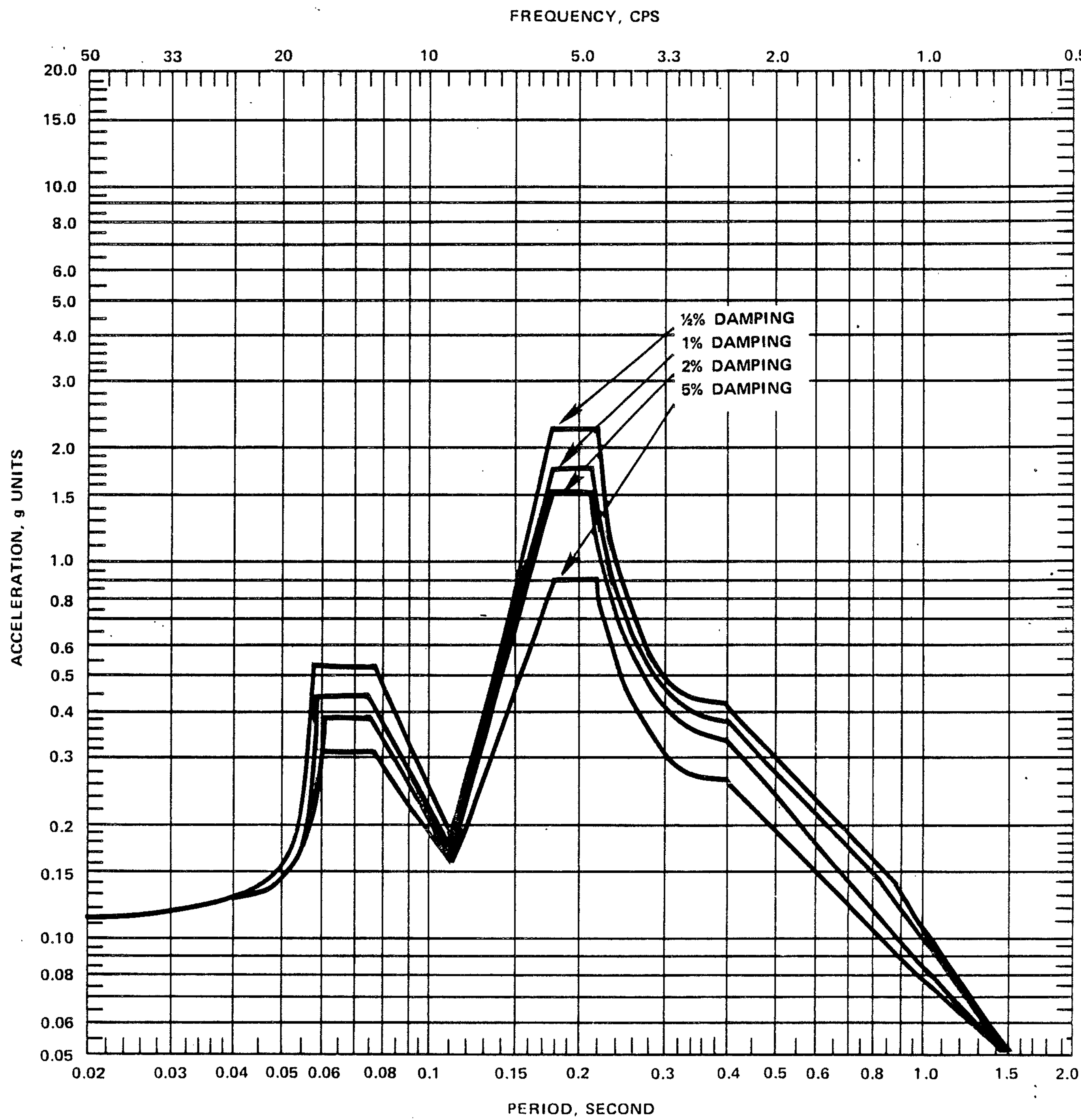
FIGURE 3.7-33

HORIZONTAL FLOOR RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
ELEVATION 583.5 FT – SLAB 1
REACTOR/AUXILIARY BUILDING EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682



Fermi 2

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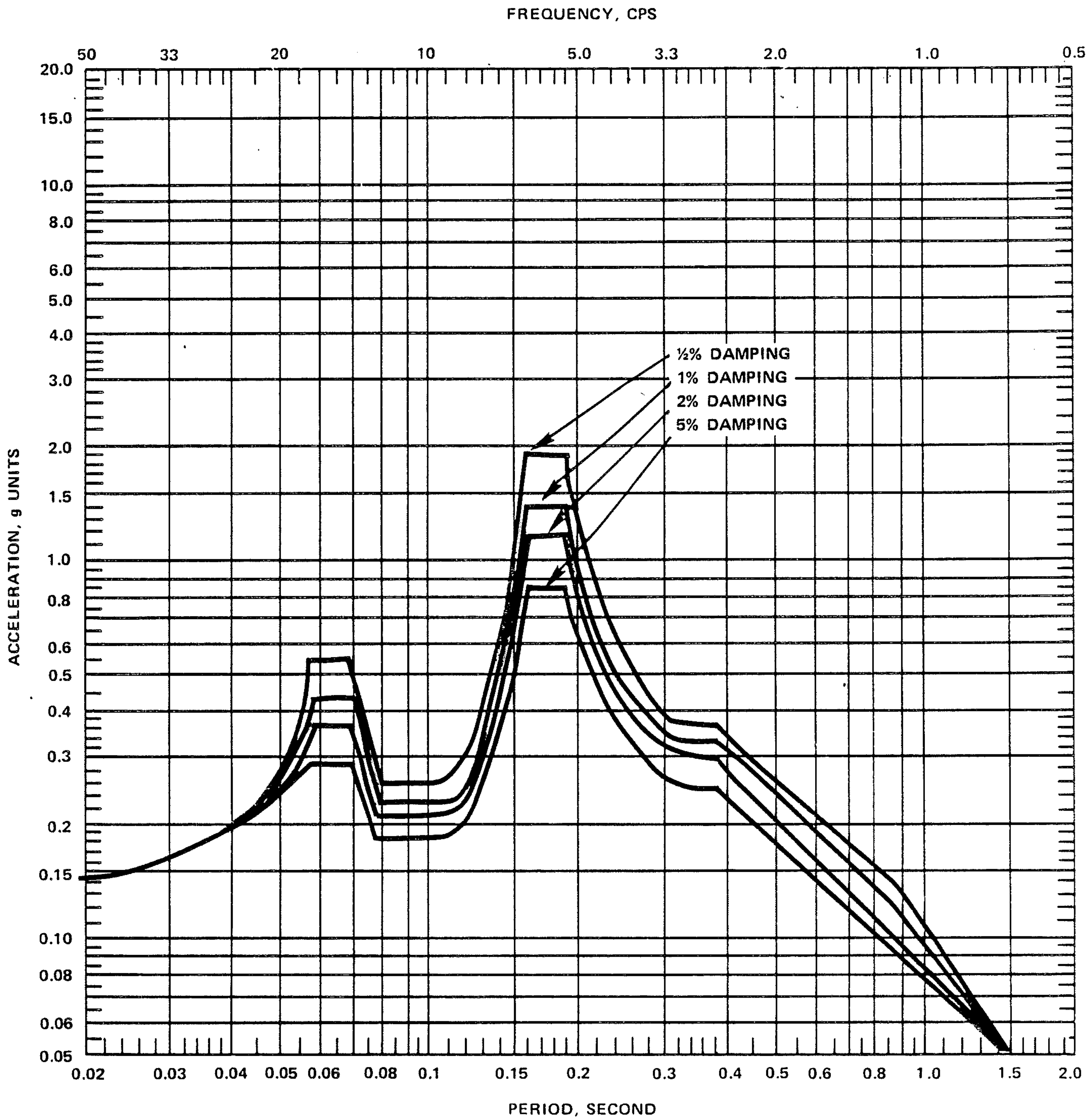
FIGURE 3.7-34

HORIZONTAL FLOOR RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
ELEVATION 613.5 FT – SLAB 2
REACTOR/AUXILIARY BUILDING NORTH-SOUTH
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682



SARGENT & LUNDY REPORT NO. SL-2682

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FIGURE 3.7-35

HORIZONTAL FLOOR RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
ELEVATION 613.5 FT – SLAB 2
REACTOR/AUXILIARY BUILDING EAST-WEST
COMPONENT

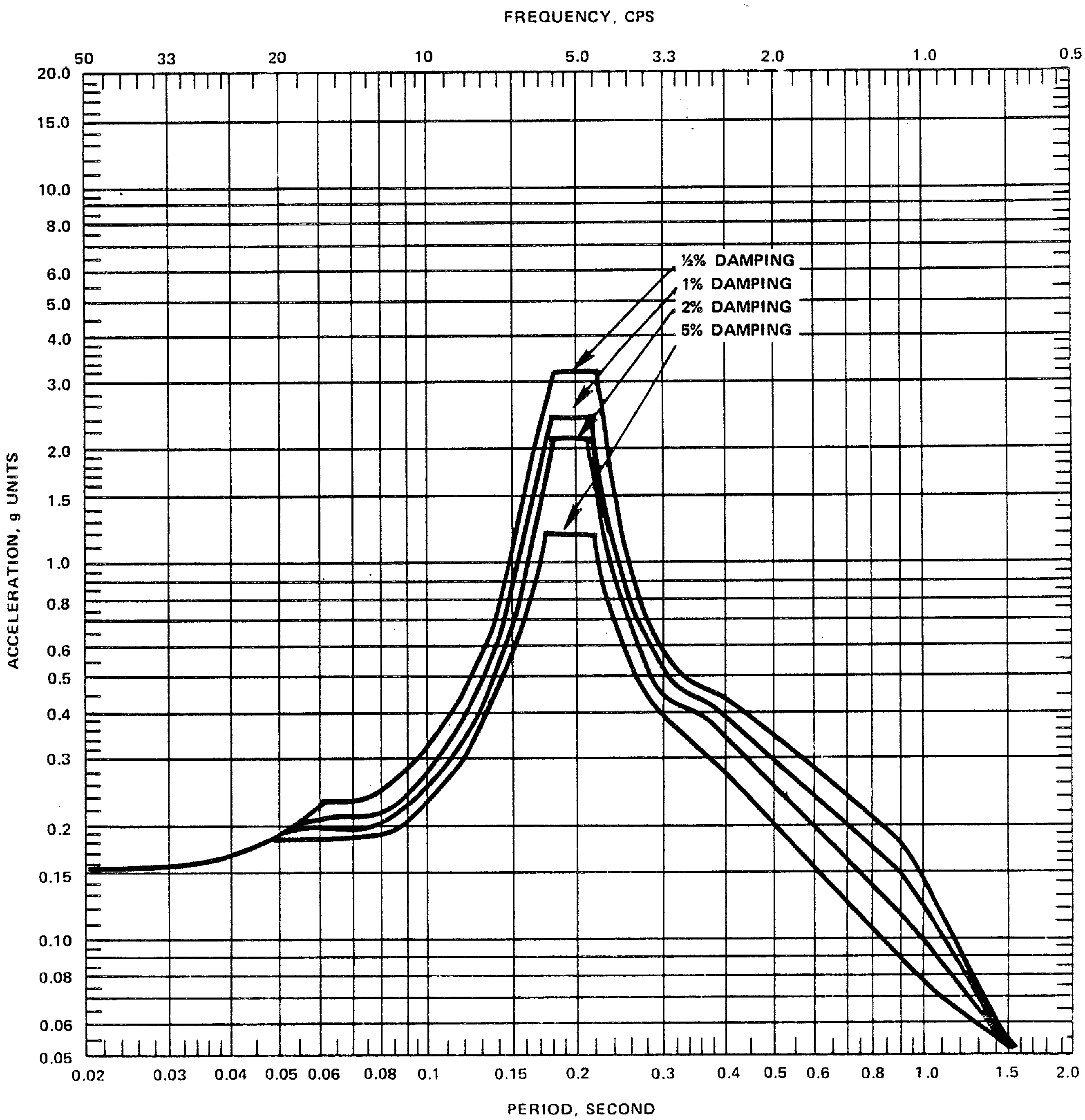

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FIGURE 3.7-36

HORIZONTAL FLOOR RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
ELEVATION 641.5 FT – SLAB 3
REACTOR/AUXILIARY BUILDING NORTH-SOUTH
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

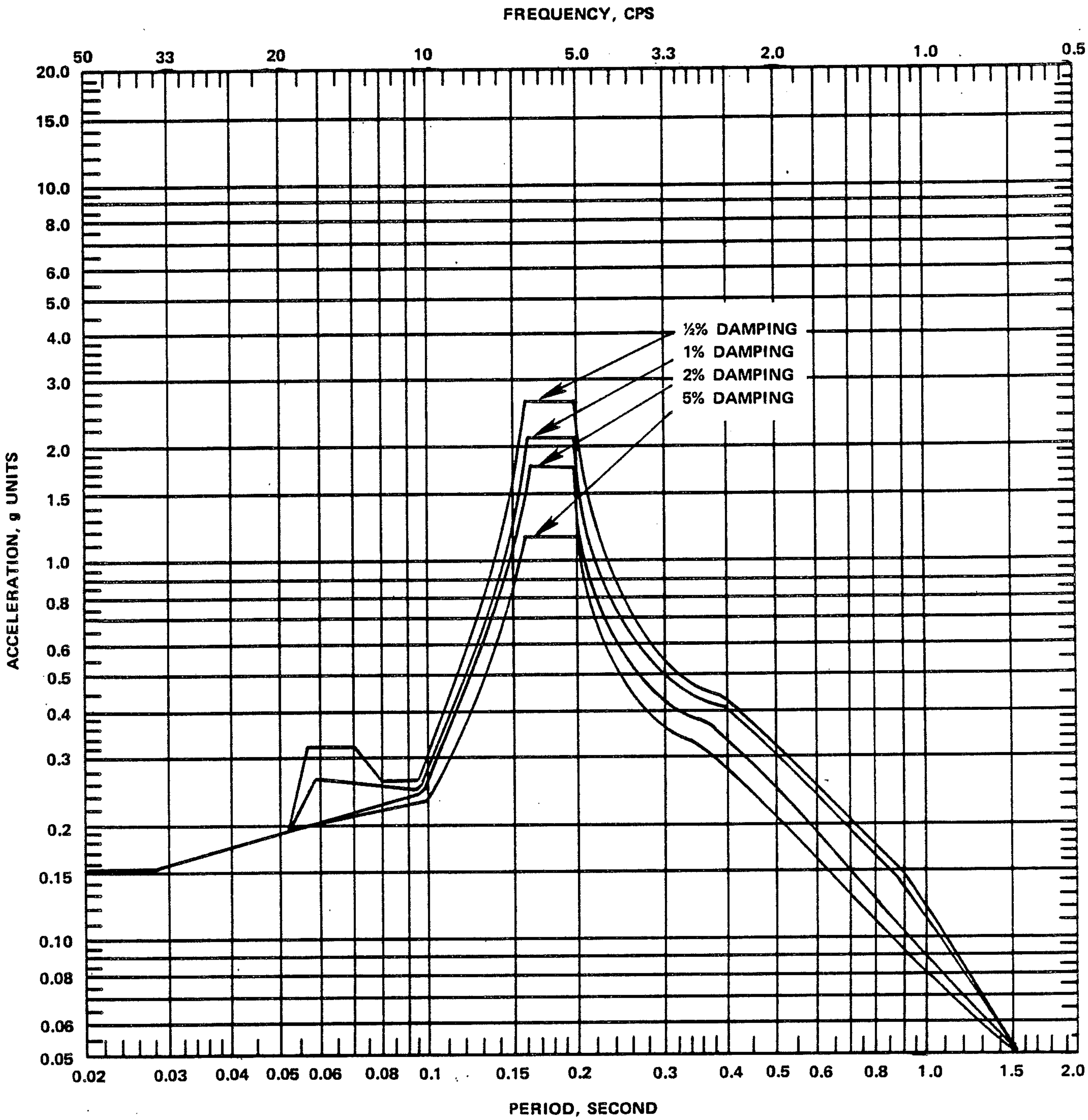

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FIGURE 3.7-37

HORIZONTAL FLOOR RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
ELEVATION 641.5 FT – SLAB 3
REACTOR/AUXILIARY BUILDING EAST-WEST
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

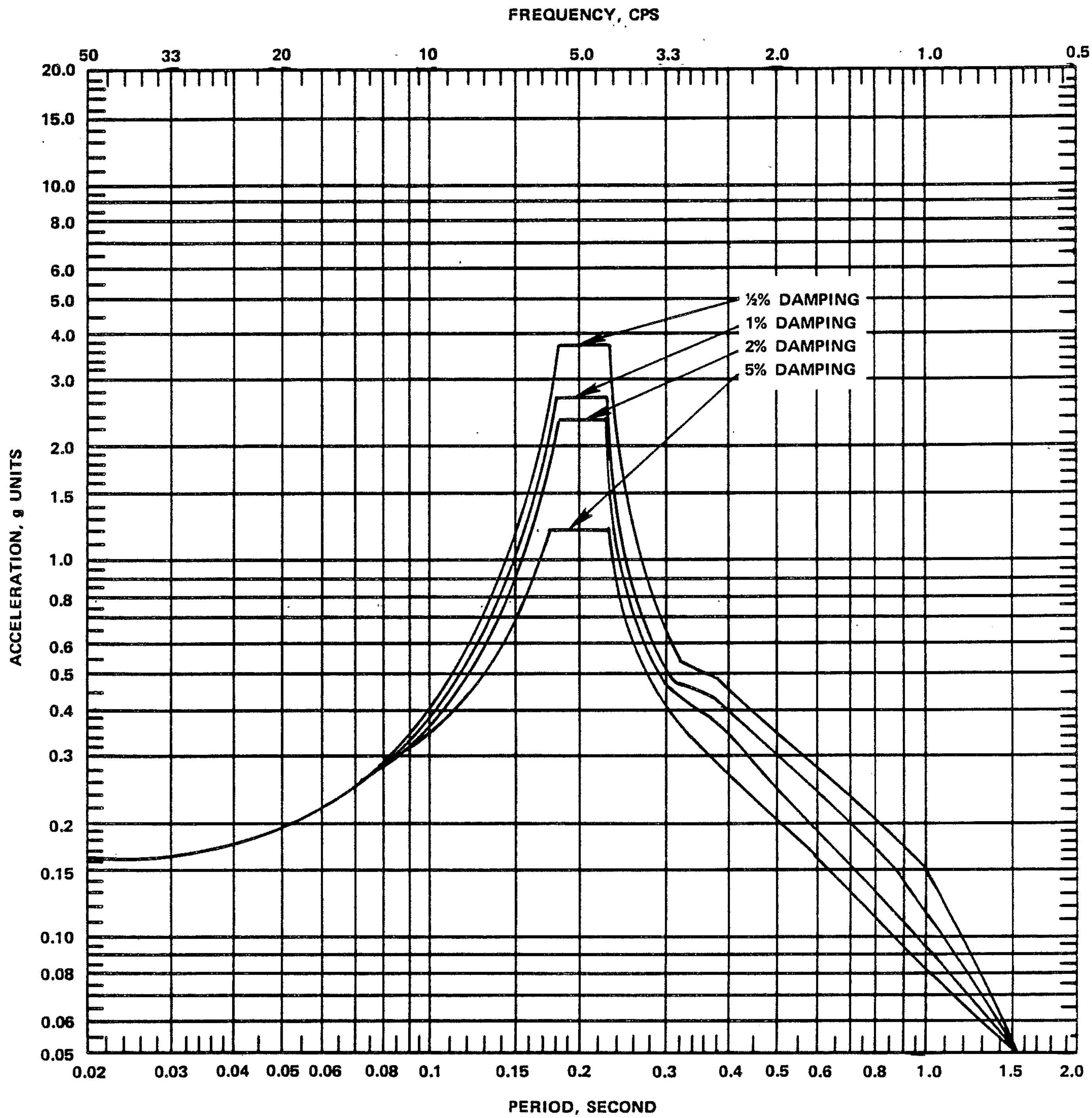

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FIGURE 3.7-38

HORIZONTAL FLOOR RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
ELEVATION 659.0 FT – SLAB 4
REACTOR/AUXILIARY BUILDING NORTH-SOUTH
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

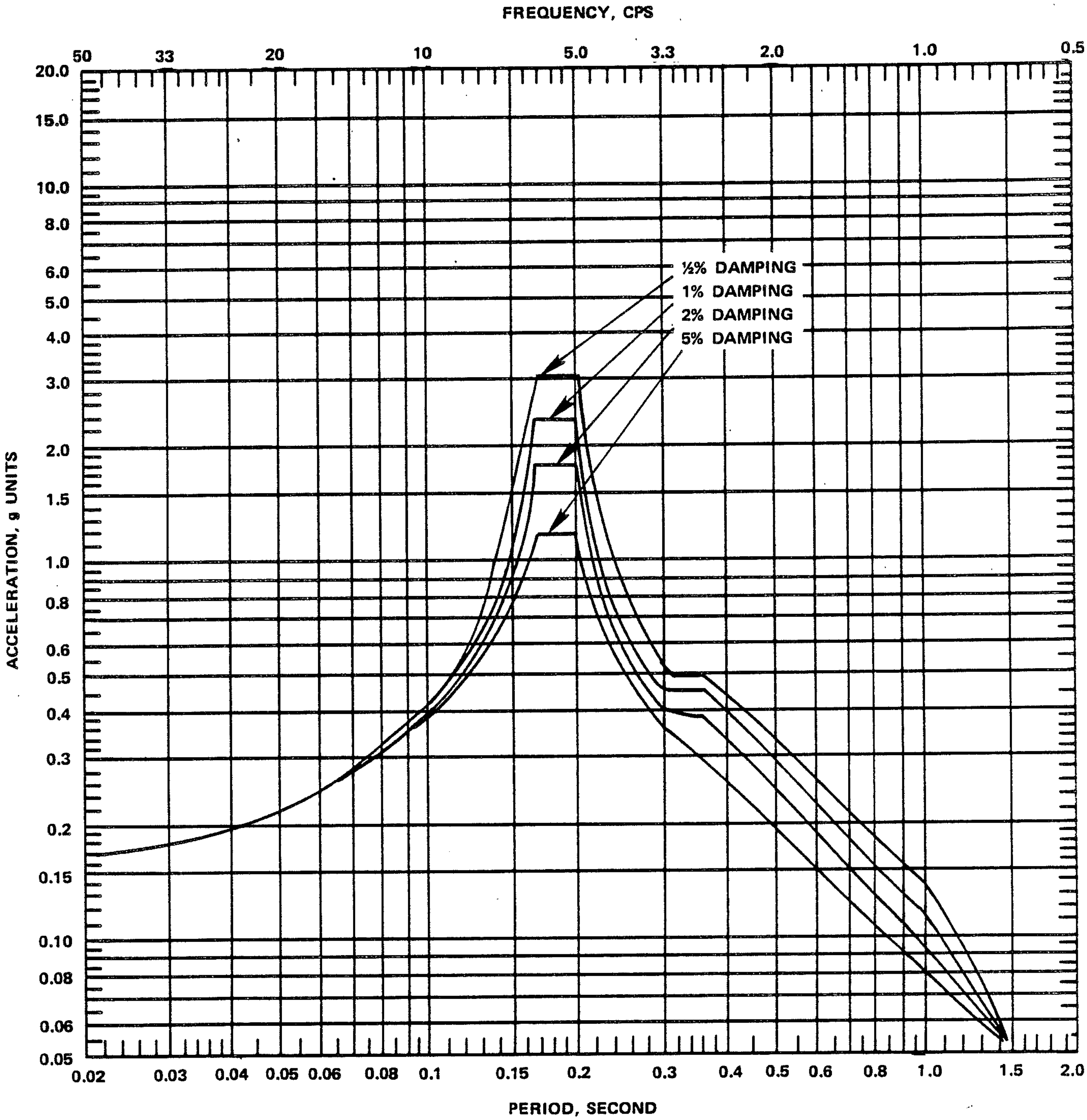

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FIGURE 3.7-39

HORIZONTAL FLOOR RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
ELEVATION 659.0 FT – SLAB 4
REACTOR/AUXILIARY BUILDING EAST-WEST
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

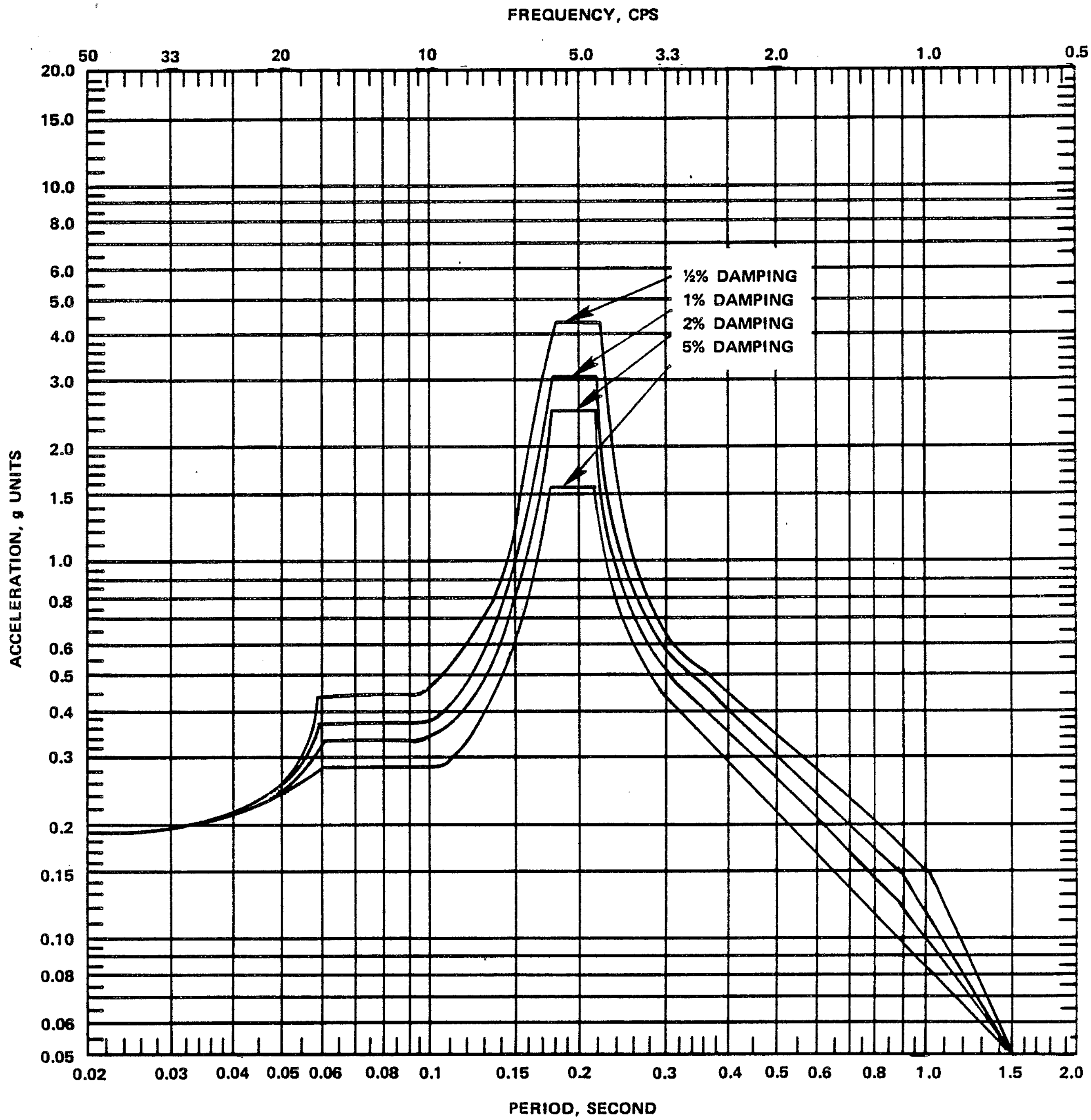

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FIGURE 3.7-40

HORIZONTAL FLOOR RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
ELEVATION 684.5 FT – SLAB 5
REACTOR/AUXILIARY BUILDING NORTH-SOUTH
COMPONENT

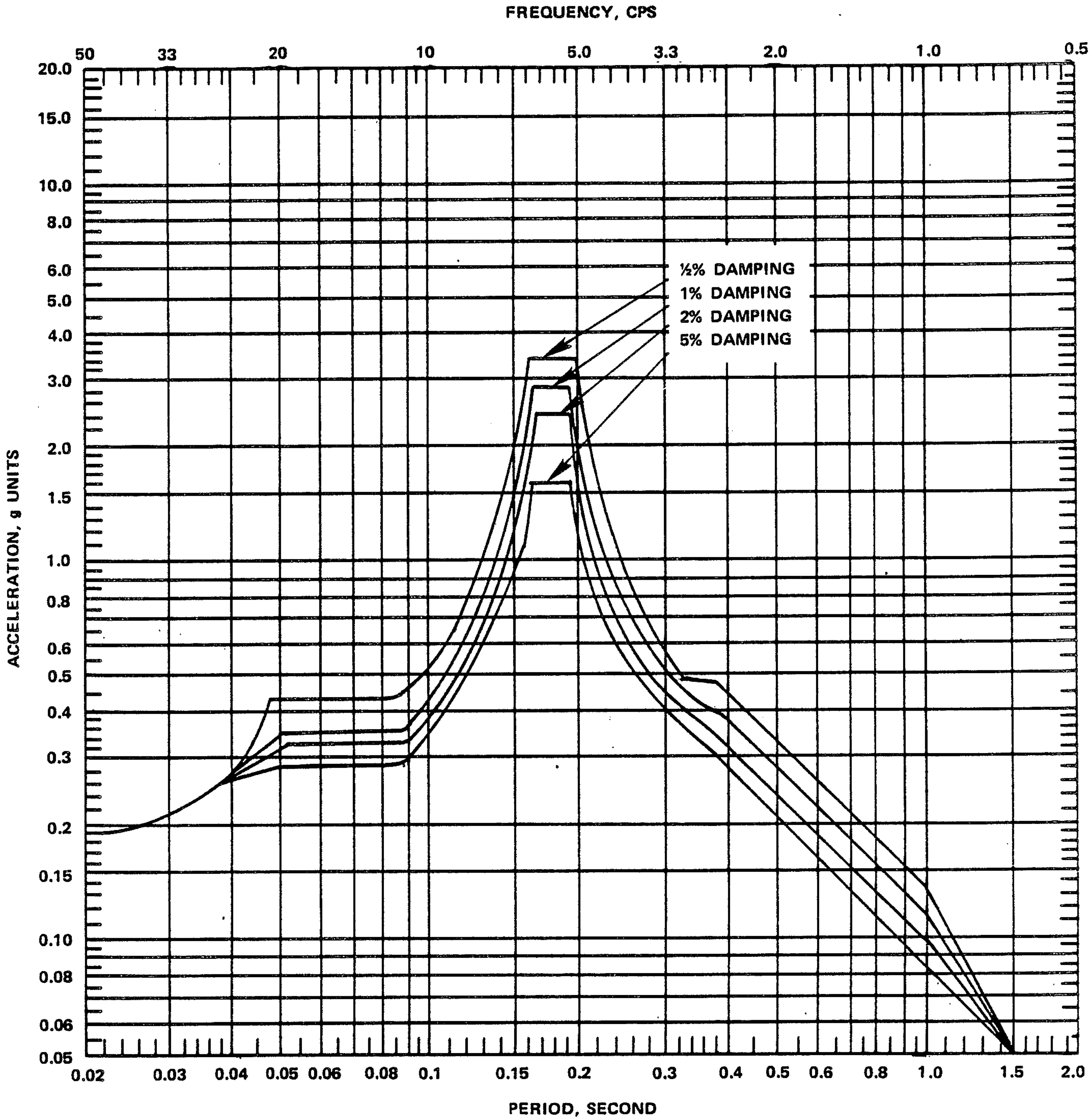

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FIGURE 3.7-41

HORIZONTAL FLOOR RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
ELEVATION 684.5 FT – SLAB 5
REACTOR/AUXILIARY BUILDING EAST-WEST
COMPONENT

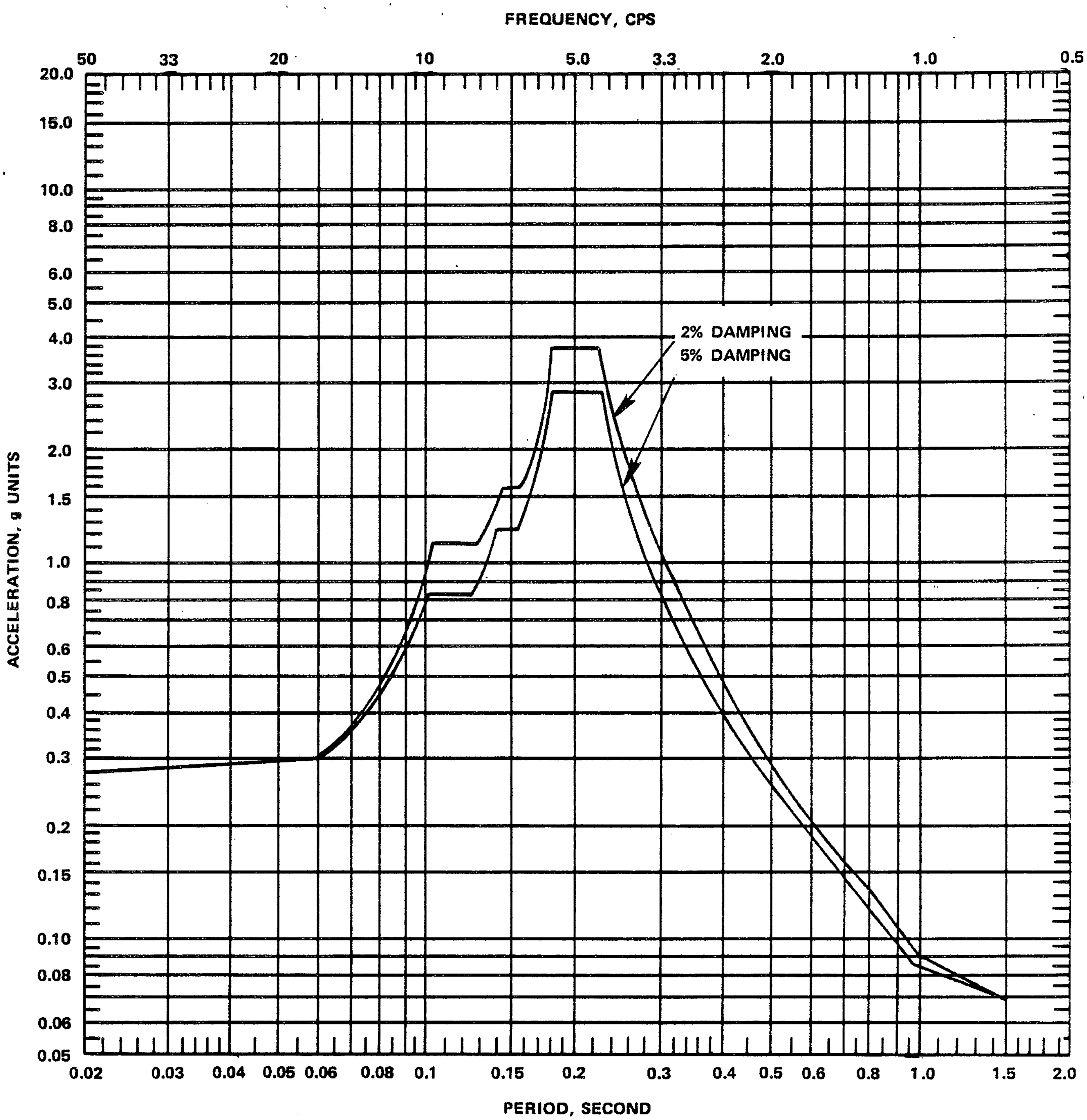

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FIGURE 3.7-42

REACTOR/AUXILIARY BUILDING HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE AT CRANE RAIL – CRANE ADJACENT TO COLUMN ROW 17 – NORTH-SOUTH COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

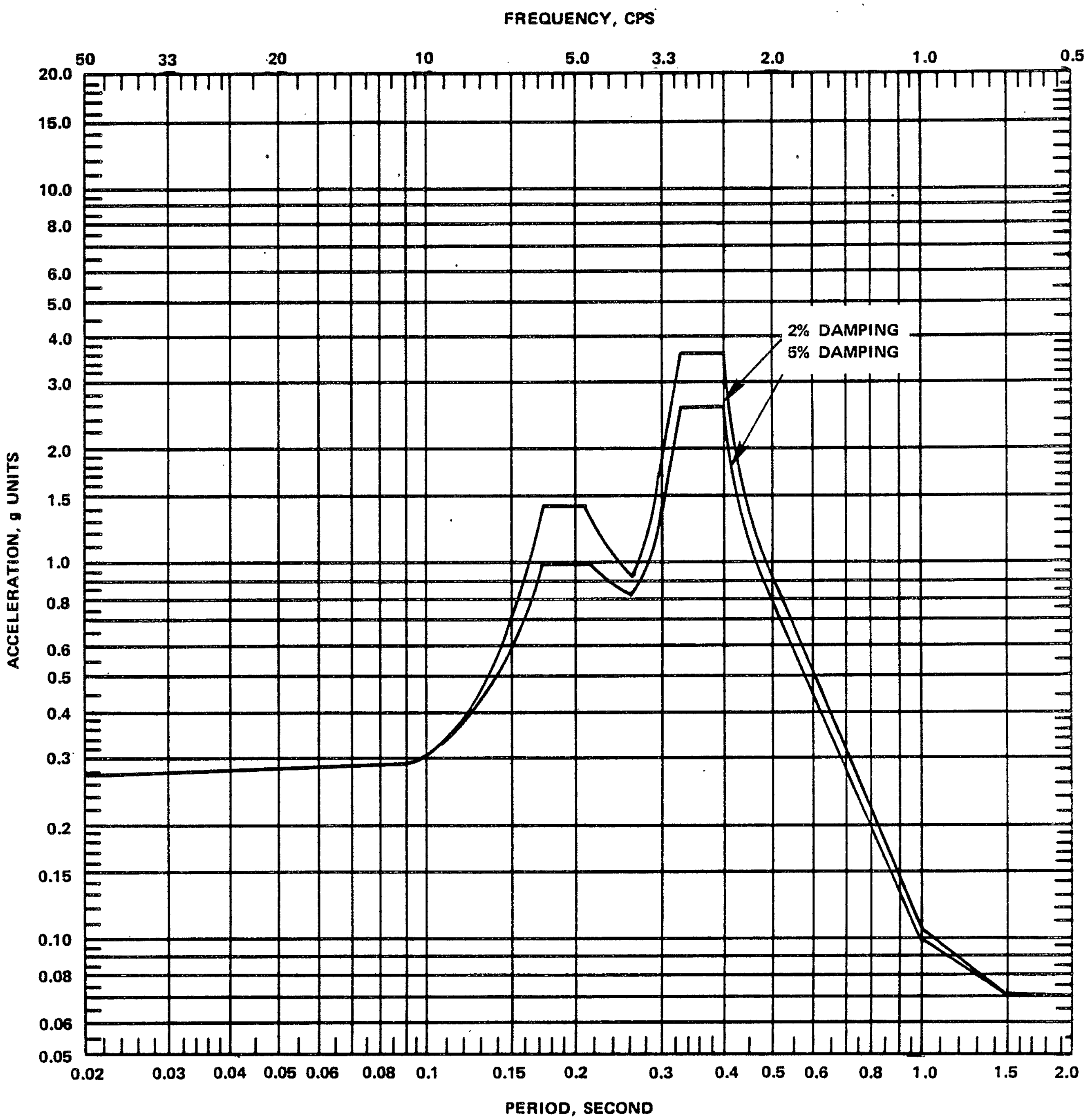

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FIGURE 3.7-43

REACTOR/AUXILIARY BUILDING HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE AT CRANE RAIL – CRANE ADJACENT TO COLUMN ROW 17 – EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

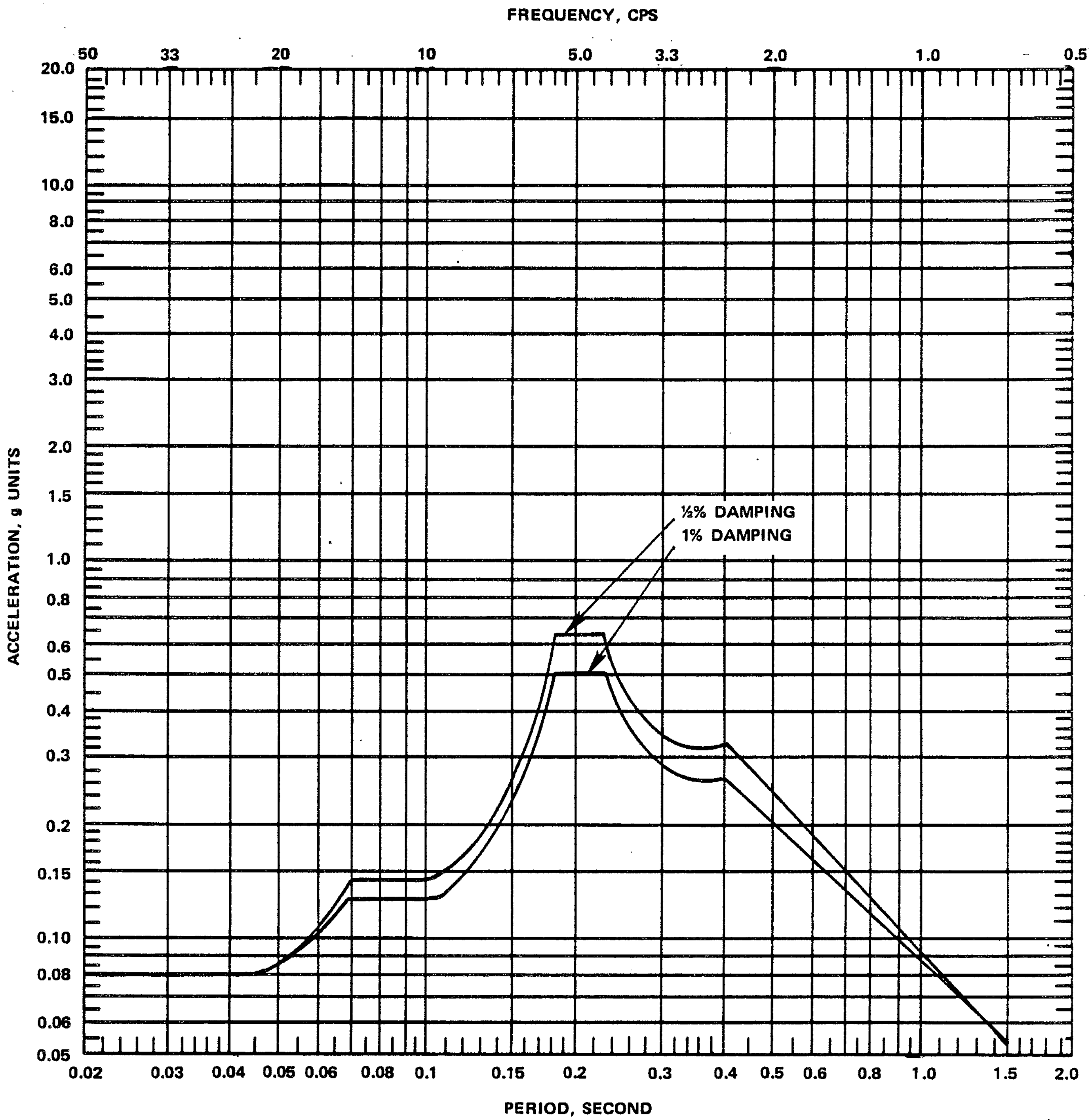

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FIGURE 3.7-44

HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE – DRYWELL CONTAINMENT 18.0 FT BELOW REACTOR PRESSURE VESSEL INVERT NORTH-SOUTH COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

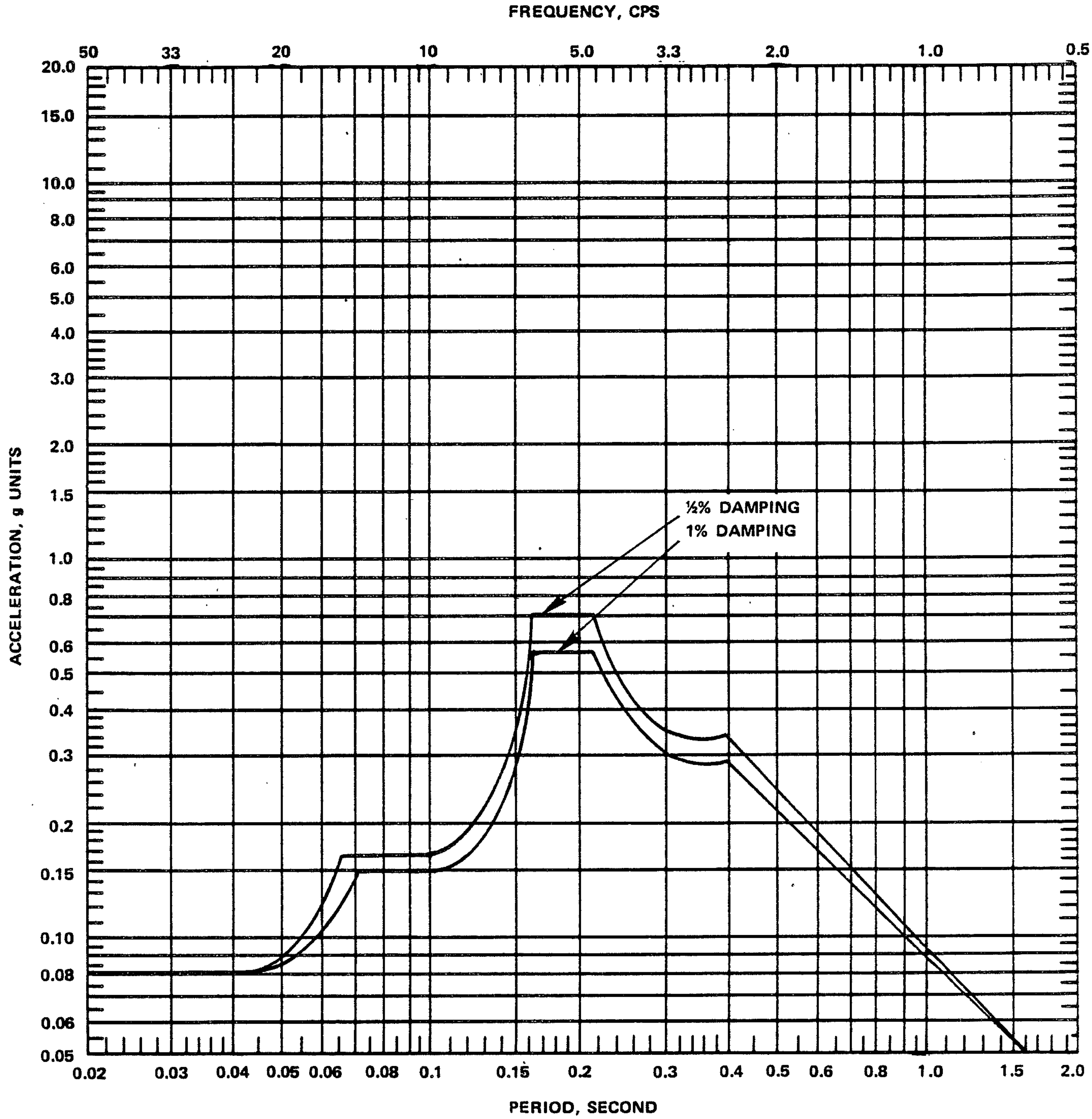

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FIGURE 3.7-45

HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE – DRYWELL CONTAINMENT 18.0 FT BELOW REACTOR PRESSURE VESSEL INVERT EAST-WEST COMPONENT

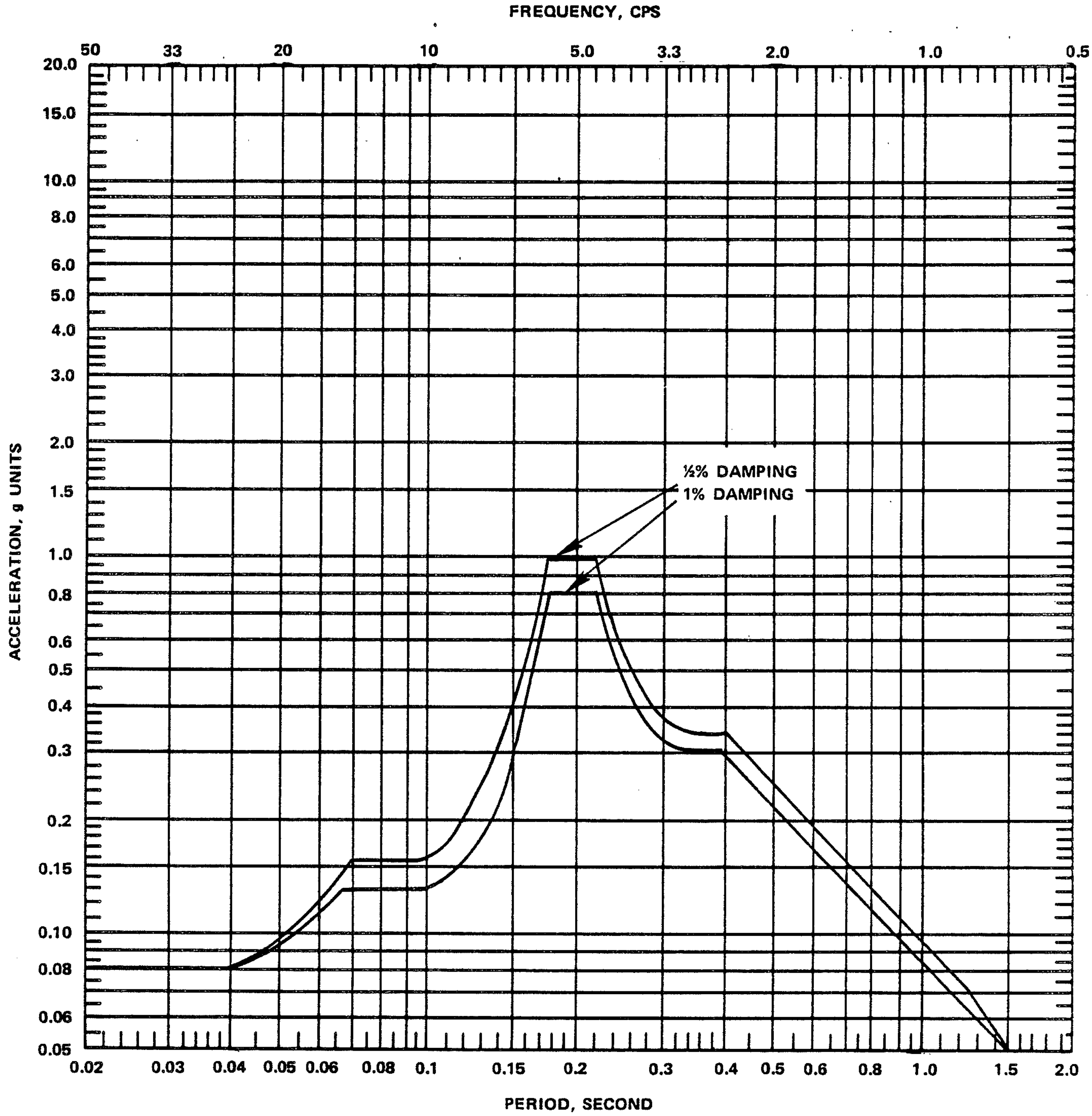

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FIGURE 3.7-46

HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE – DRYWELL CONTAINMENT 6.0 FT BELOW REACTOR PRESSURE VESSEL INVERT NORTH-SOUTH COMPONENT

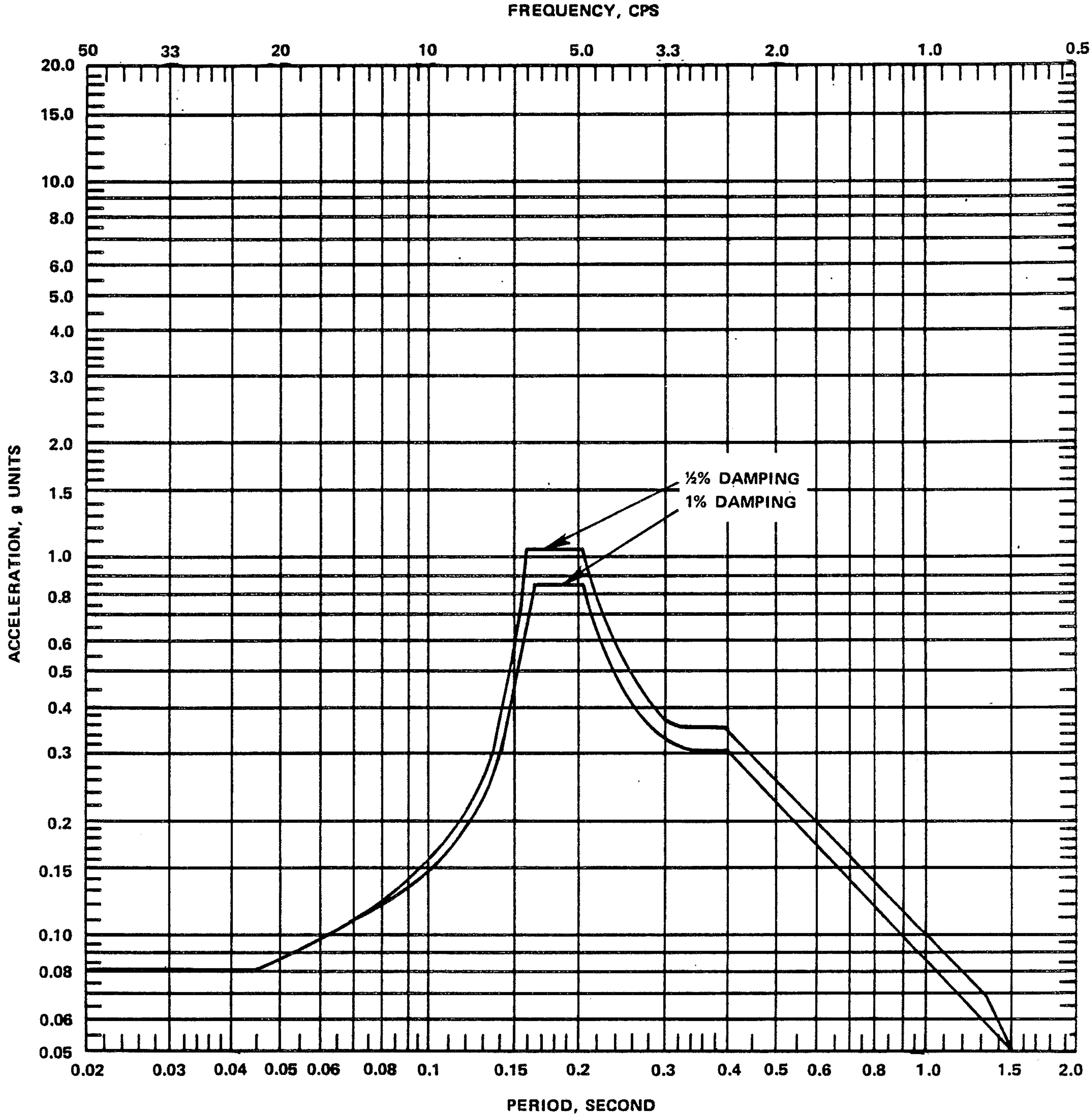

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FIGURE 3.7-47

HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE – DRYWELL CONTAINMENT 6.0 FT BELOW REACTOR PRESSURE VESSEL INVERT EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

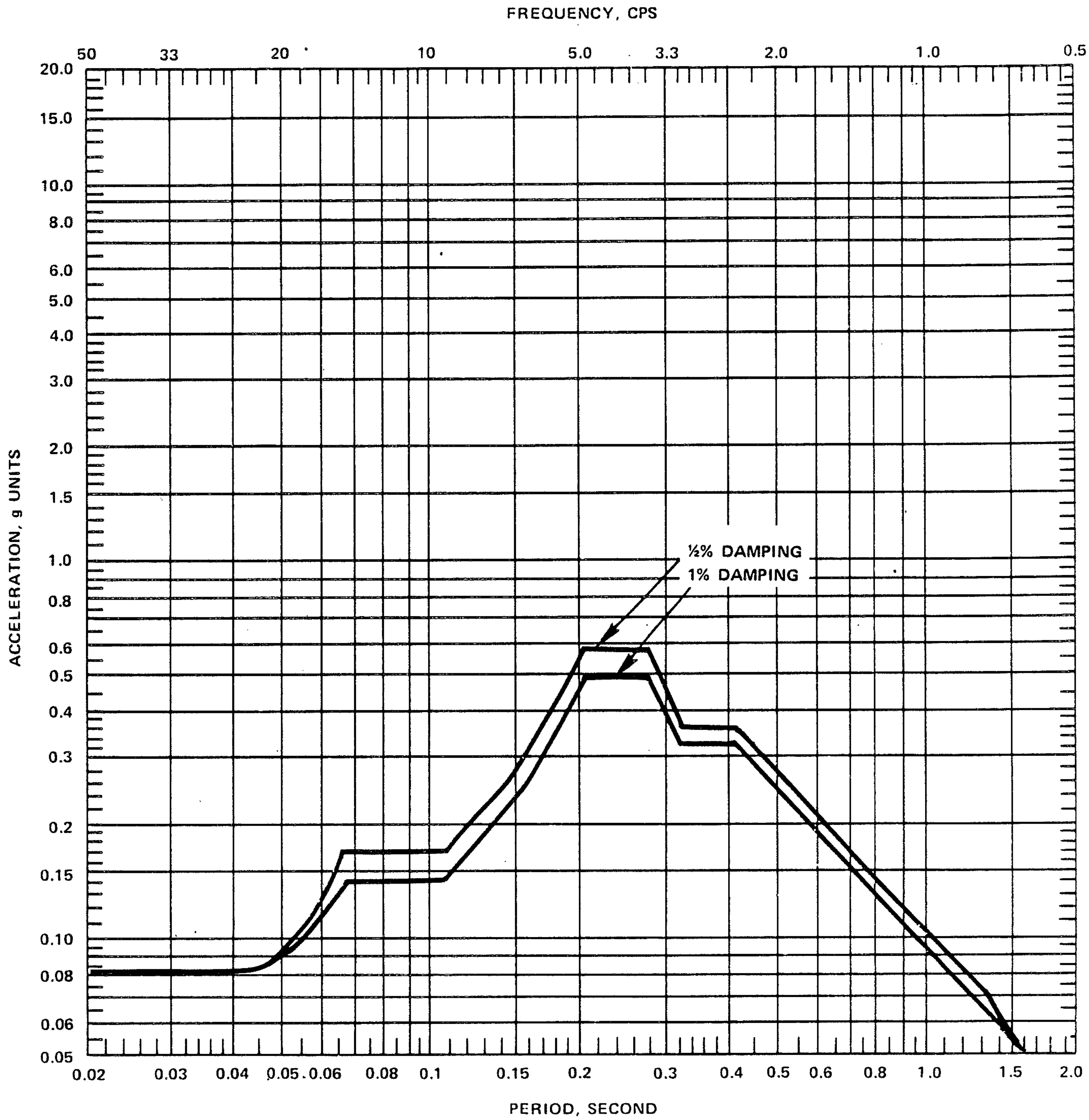

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FIGURE 3.7-48

HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE – REACTOR PEDESTAL 18.0 FT BELOW REACTOR PRESSURE VESSEL INVERT NORTH-SOUTH COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

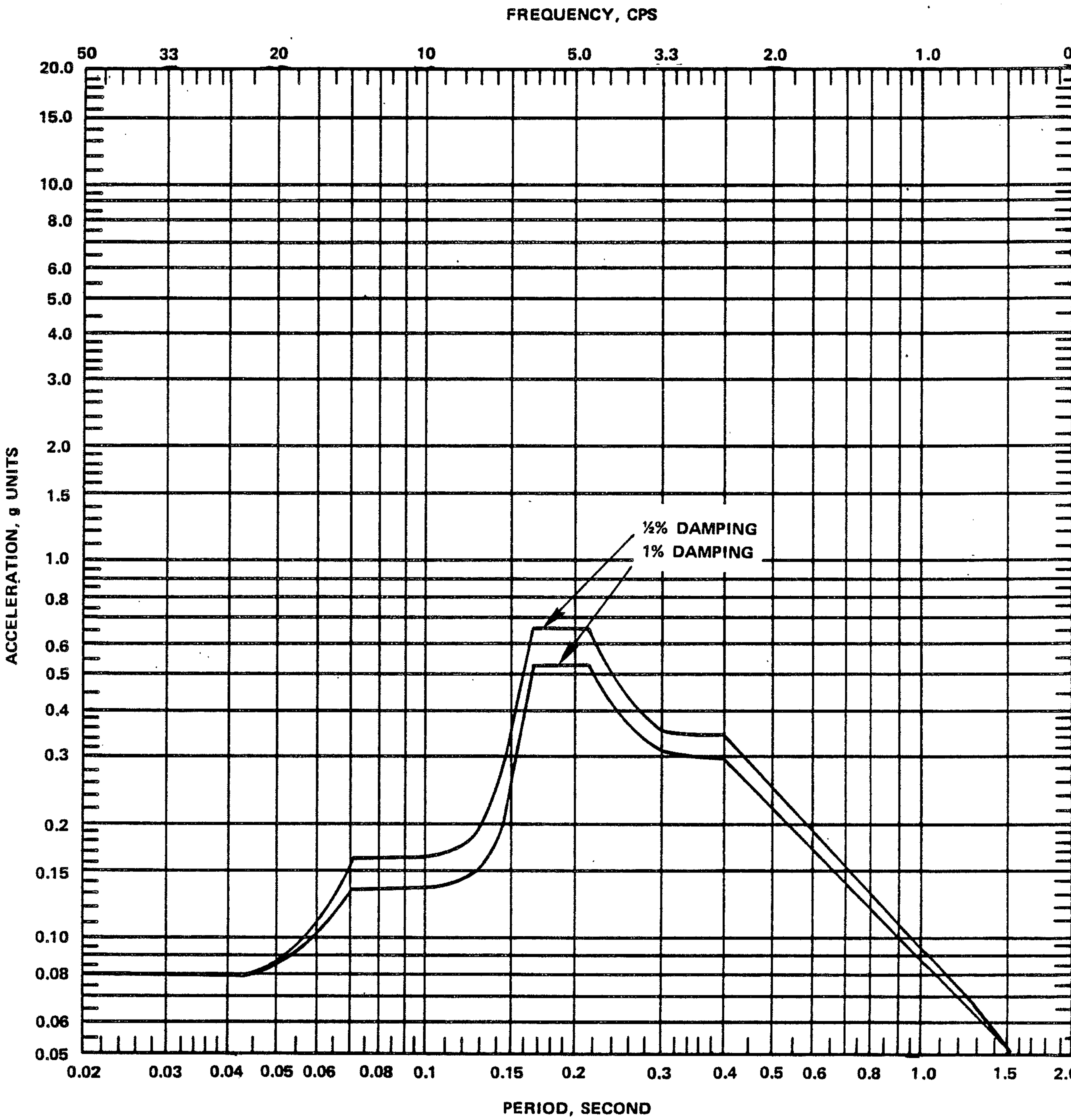

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FIGURE 3.7-49

HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE – REACTOR PEDESTAL 18.0 FT BELOW REACTOR PRESSURE VESSEL INVERT EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

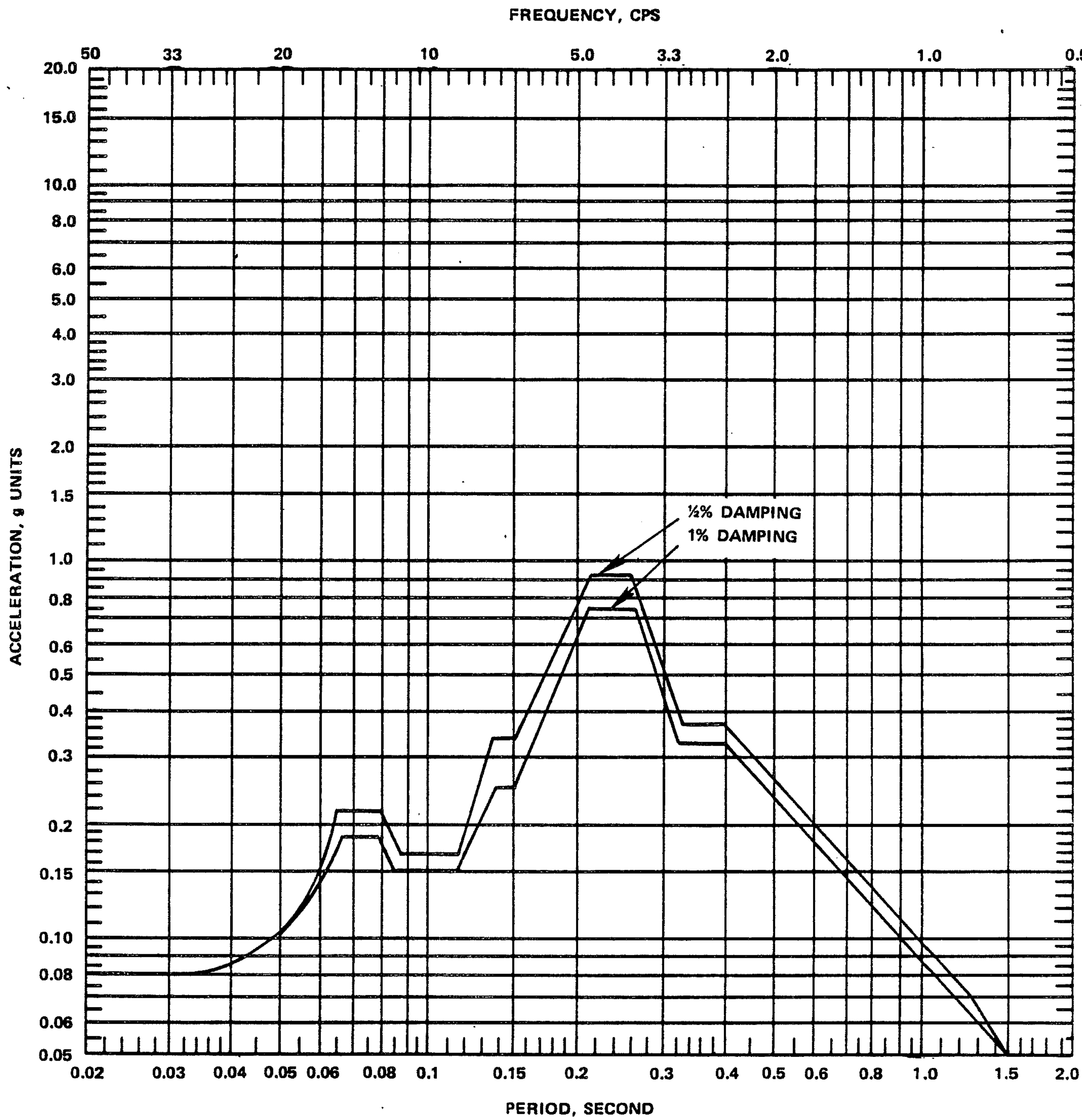

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FIGURE 3.7-50

HORIZONTAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE – TOP OF
REACTOR PEDESTAL – NORTH-SOUTH COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

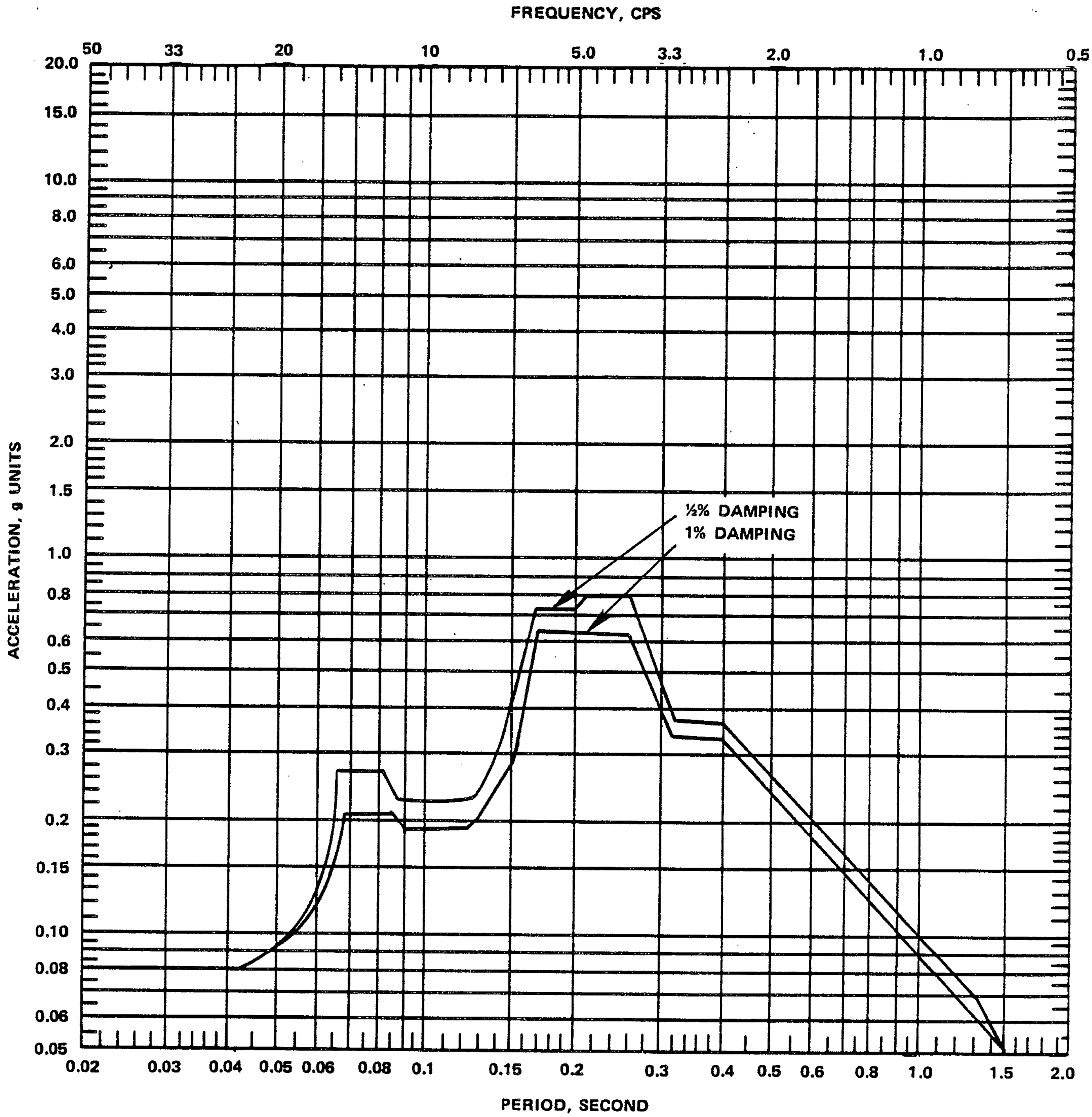

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FIGURE 3.7-51

HORIZONTAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE – TOP OF
REACTOR PEDESTAL – EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

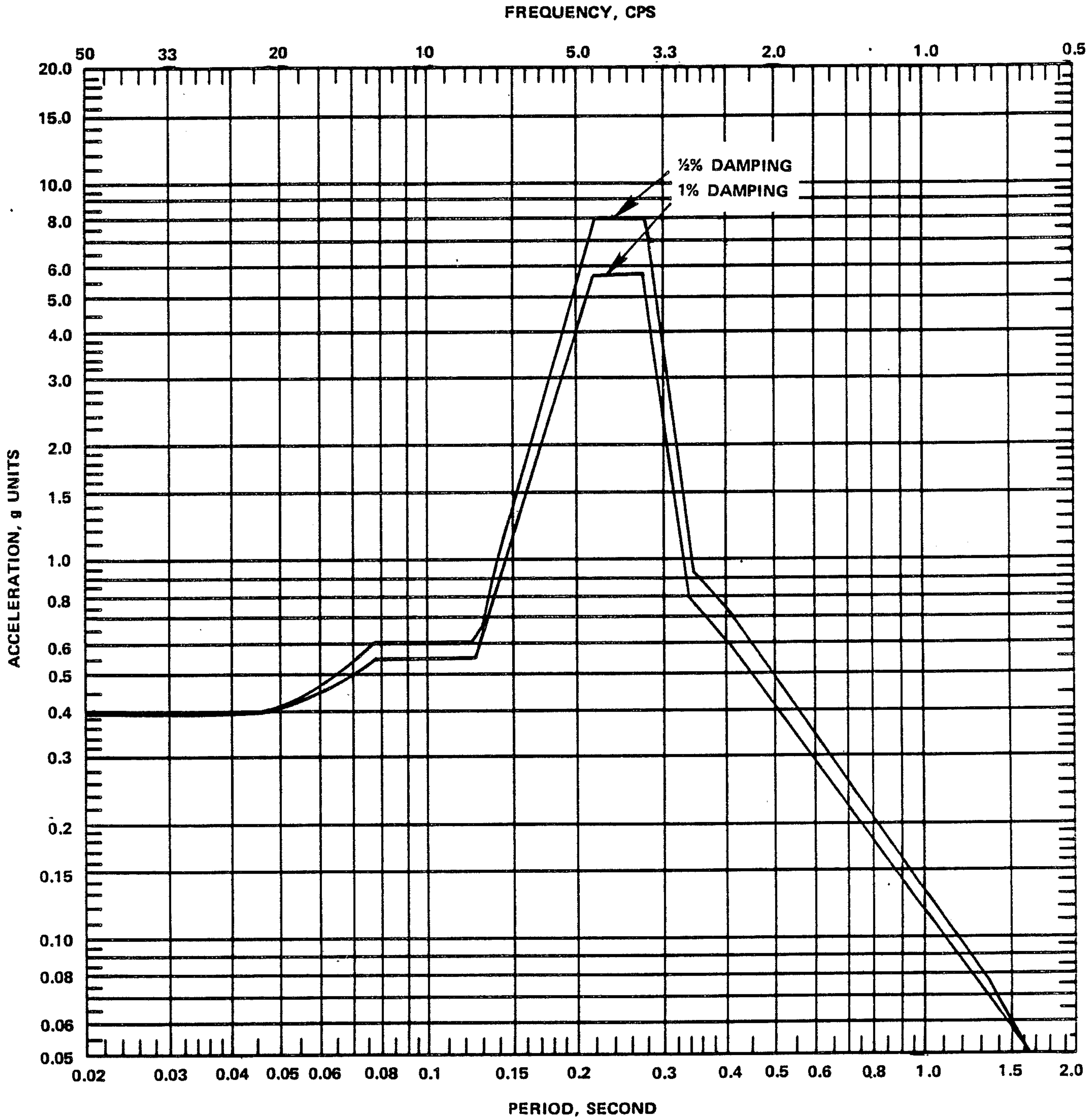

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FIGURE 3.7-52

HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE – REACTOR PRESSURE VESSEL 14.0 FT ABOVE REACTOR PRESSURE VESSEL INVERT NORTH-SOUTH COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

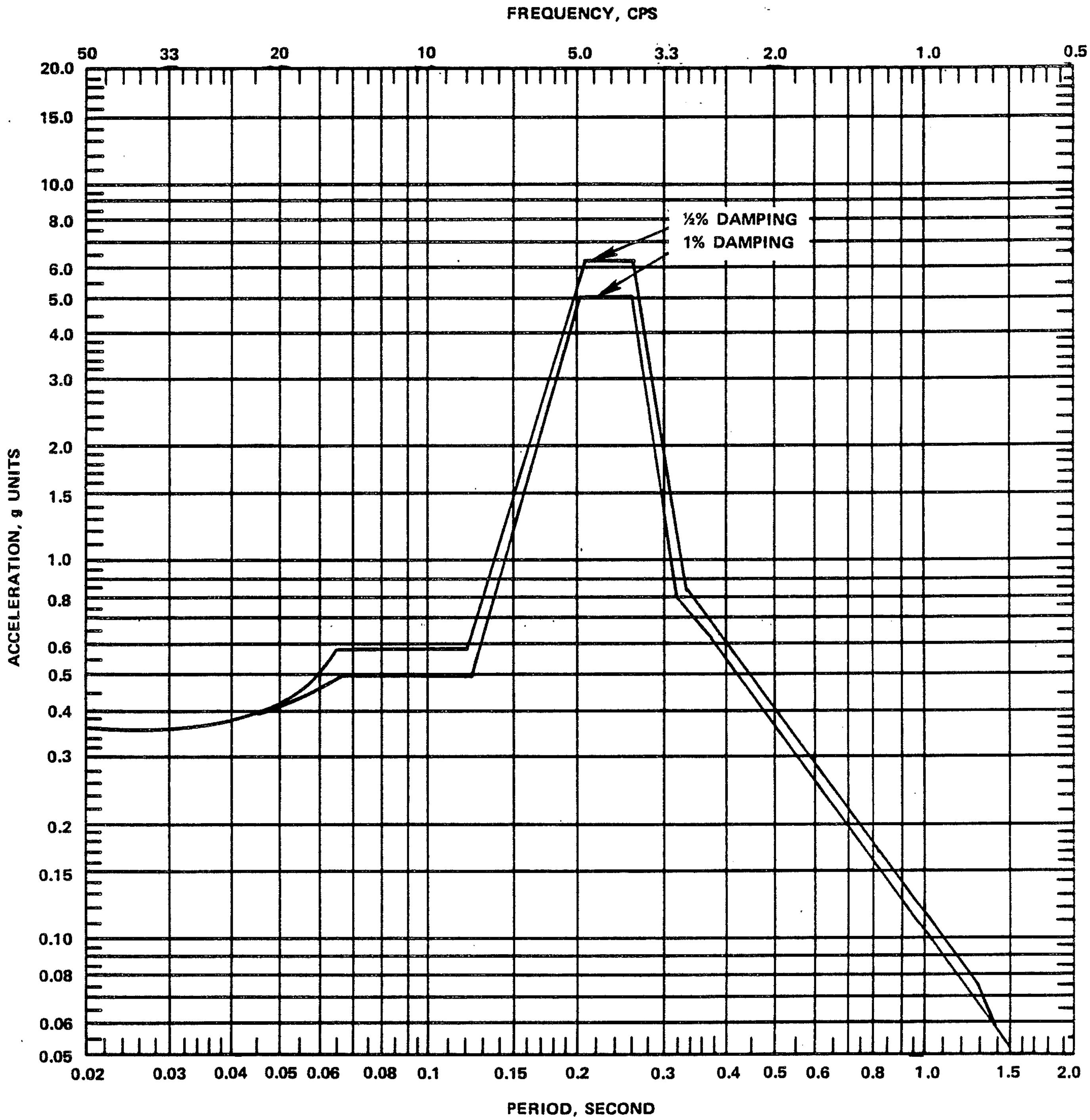

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FIGURE 3.7-53

HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE – REACTOR PRESSURE VESSEL 14.0 FT ABOVE REACTOR PRESSURE VESSEL INVERT EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

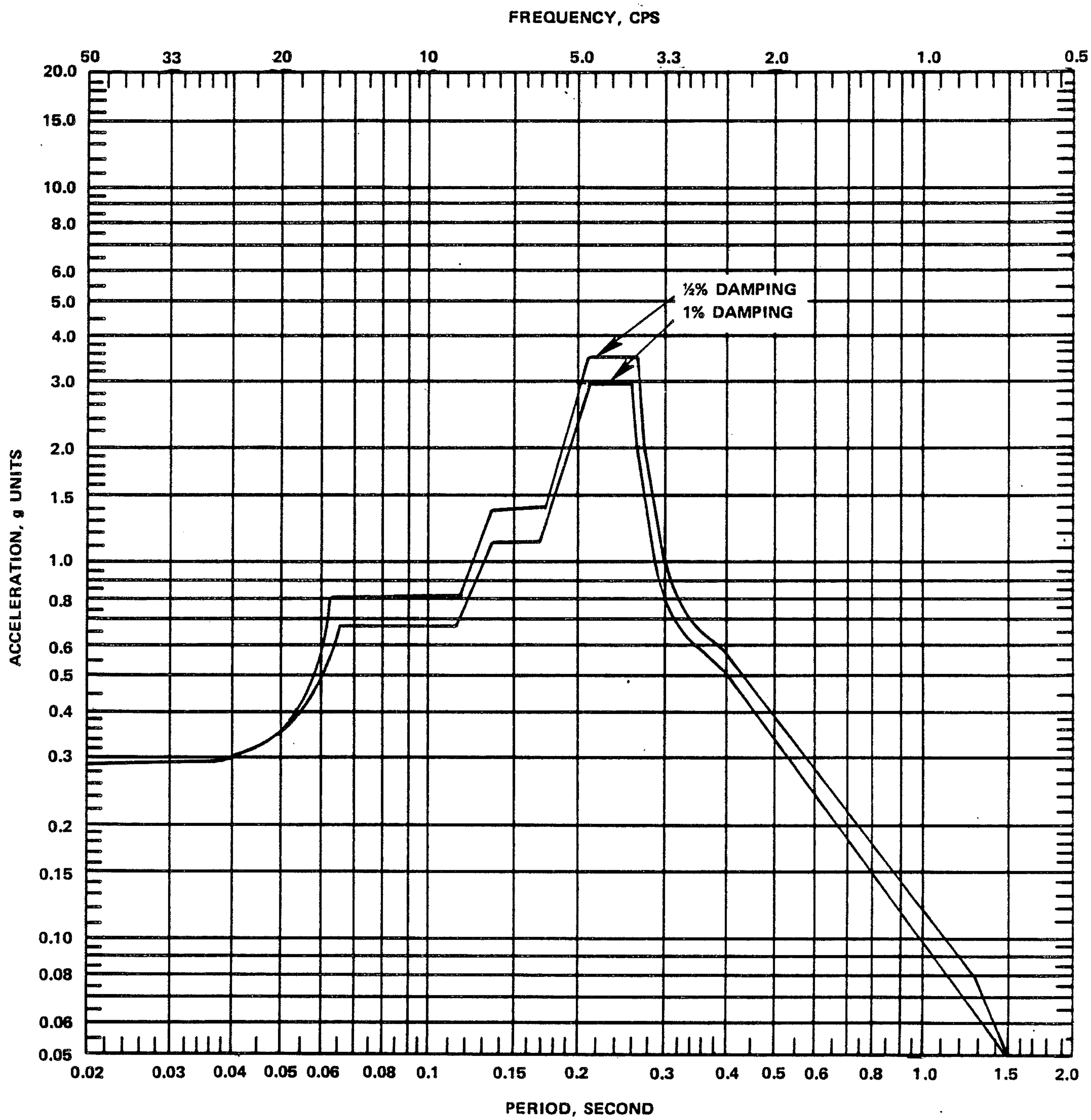

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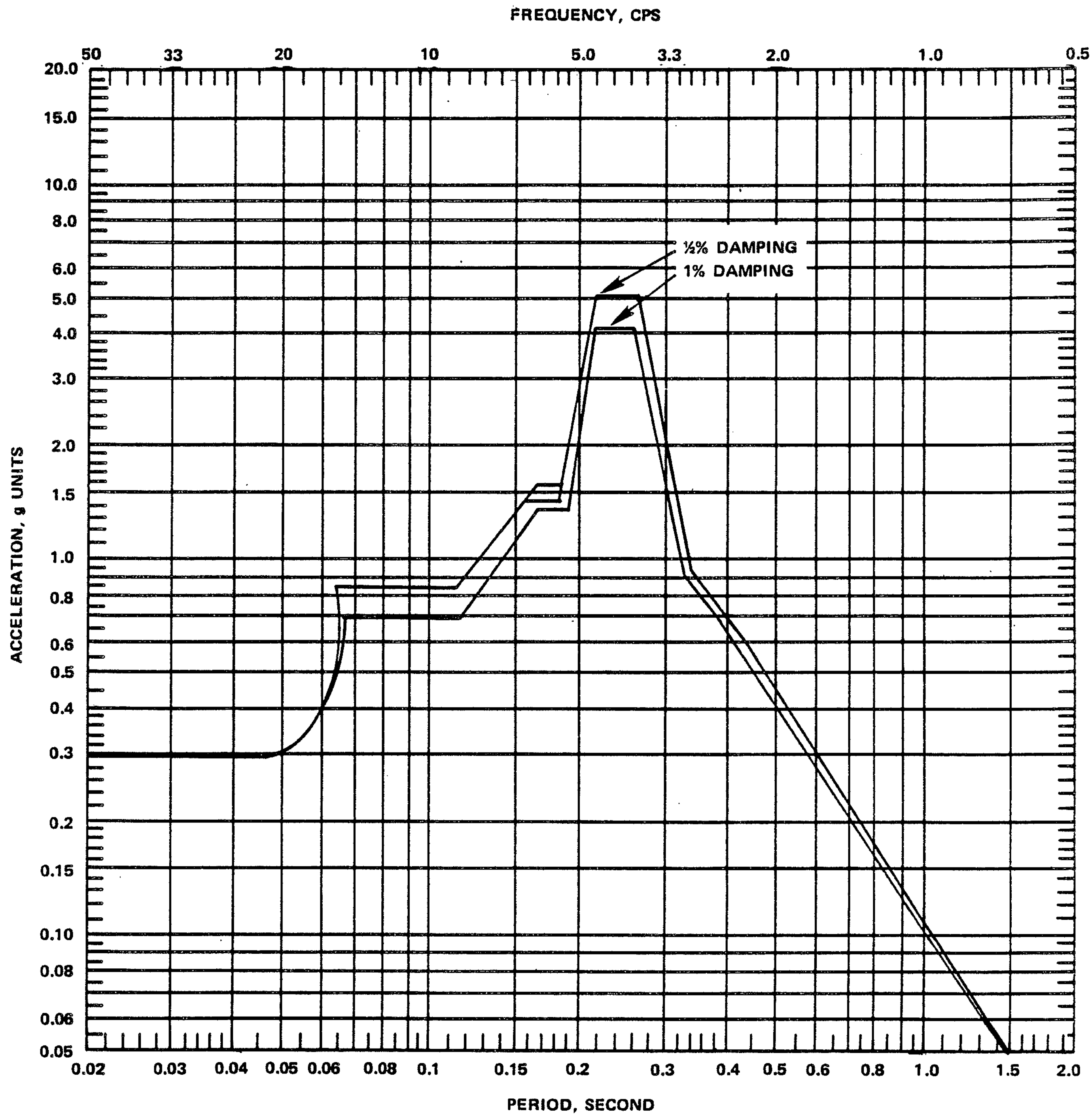
FIGURE 3.7-54

HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE – REACTOR PRESSURE VESSEL 54.0 FT ABOVE REACTOR PRESSURE VESSEL INVERT NORTH-SOUTH COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682



Fermi 2

UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7-55

HORIZONTAL RESPONSE SPECTRA – OPERATING-BASIS EARTHQUAKE – REACTOR PRESSURE VESSEL 54.0 FT ABOVE REACTOR PRESSURE VESSEL INVERT EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

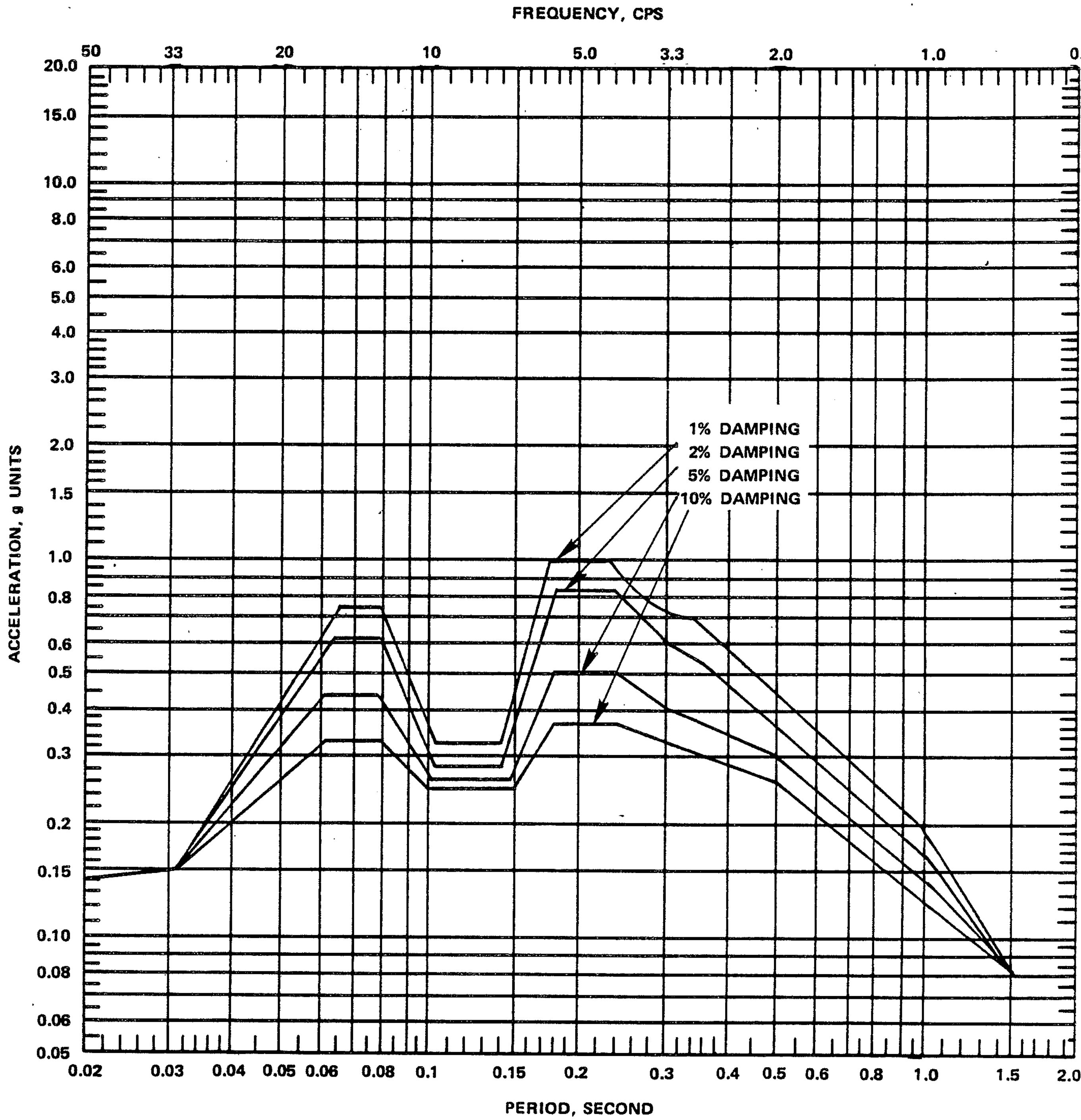


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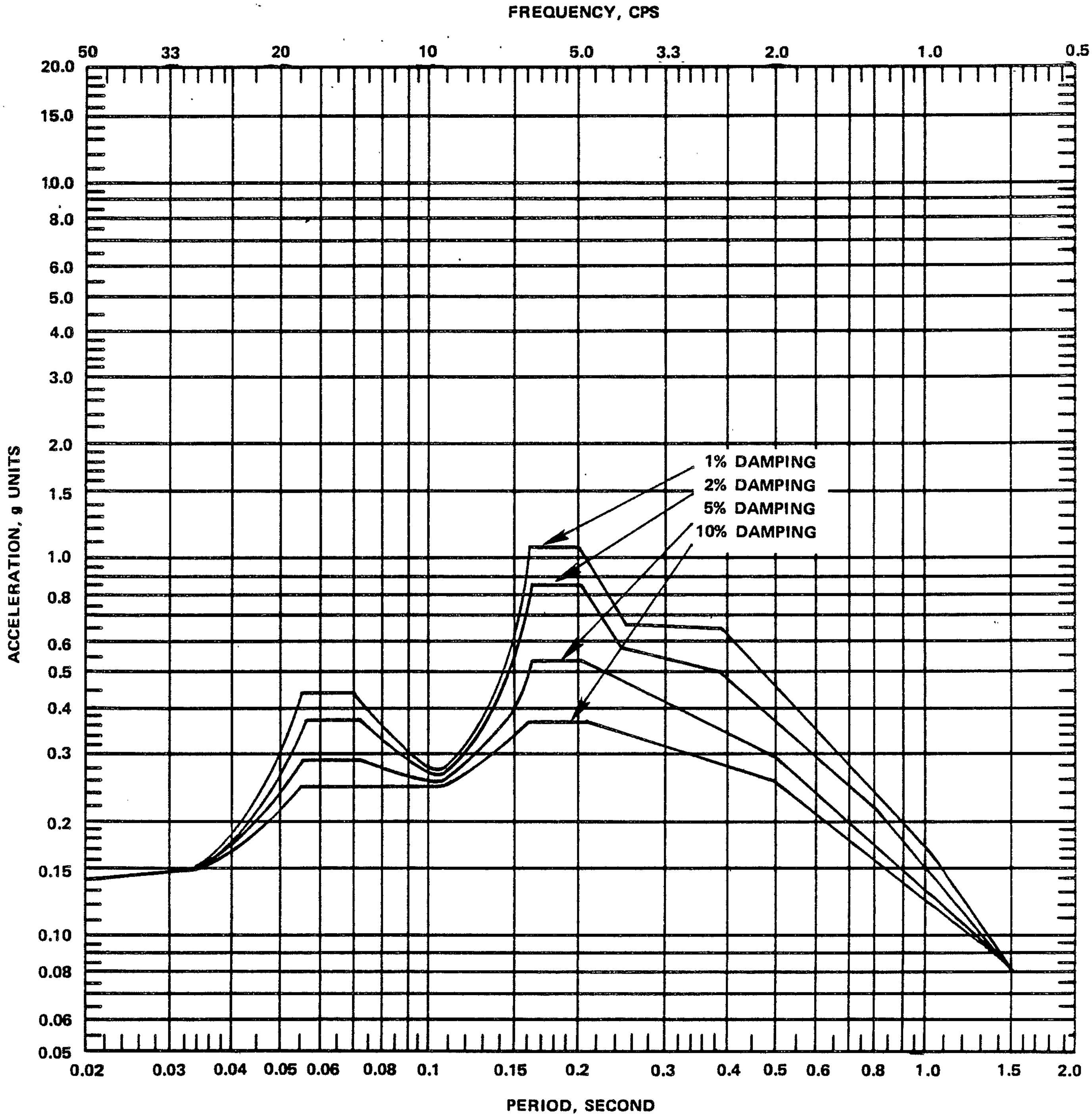
FIGURE 3.7-56

HORIZONTAL FLOOR RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
ELEVATION 583.5 FT – SLAB 1
REACTOR/AUXILIARY BUILDING NORTH-SOUTH COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682



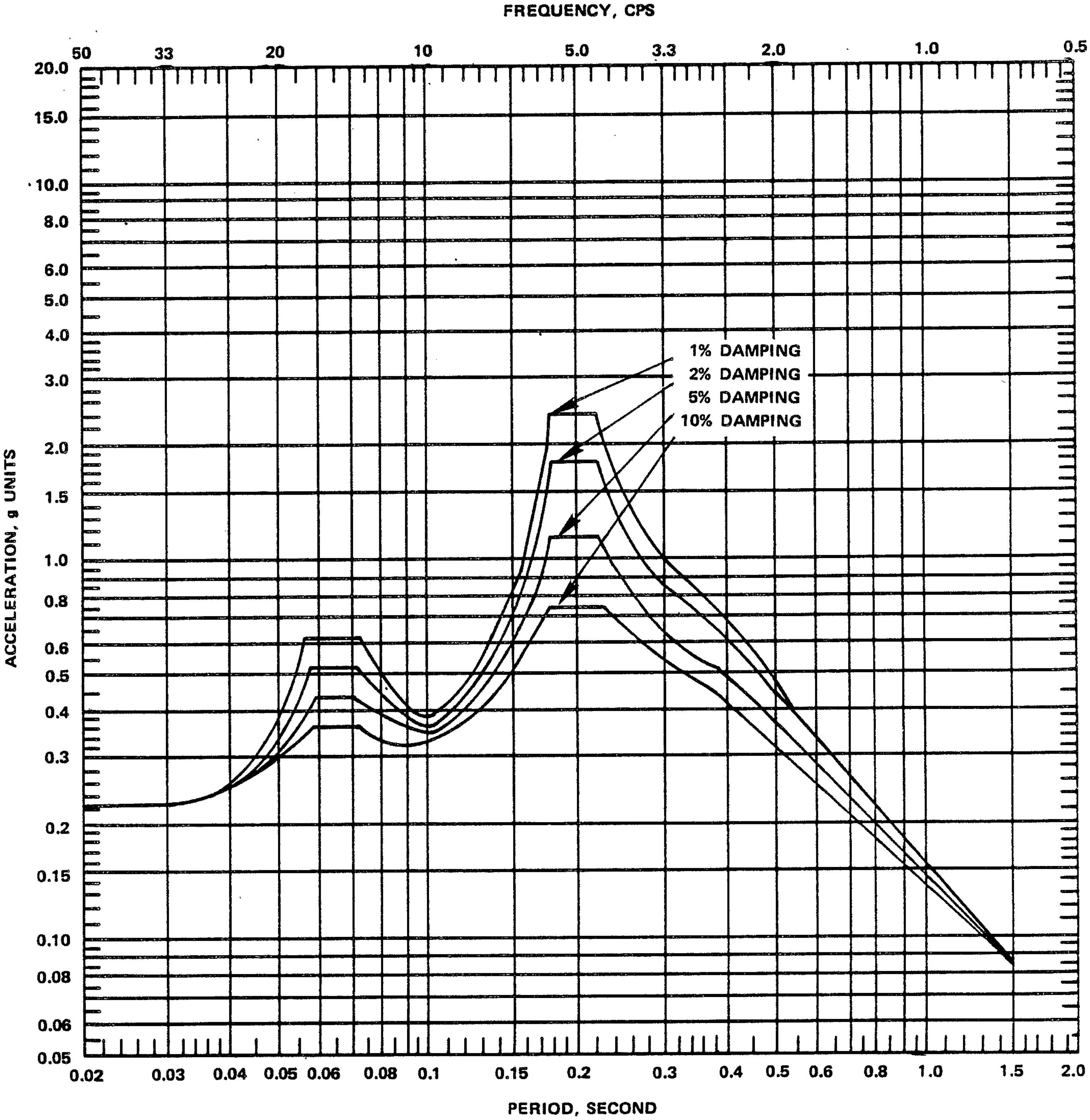
Fermi 2

UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7-57

HORIZONTAL FLOOR RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
ELEVATION 583.5 FT – SLAB 1
REACTOR/AUXILIARY BUILDING EAST-WEST
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682



Fermi 2

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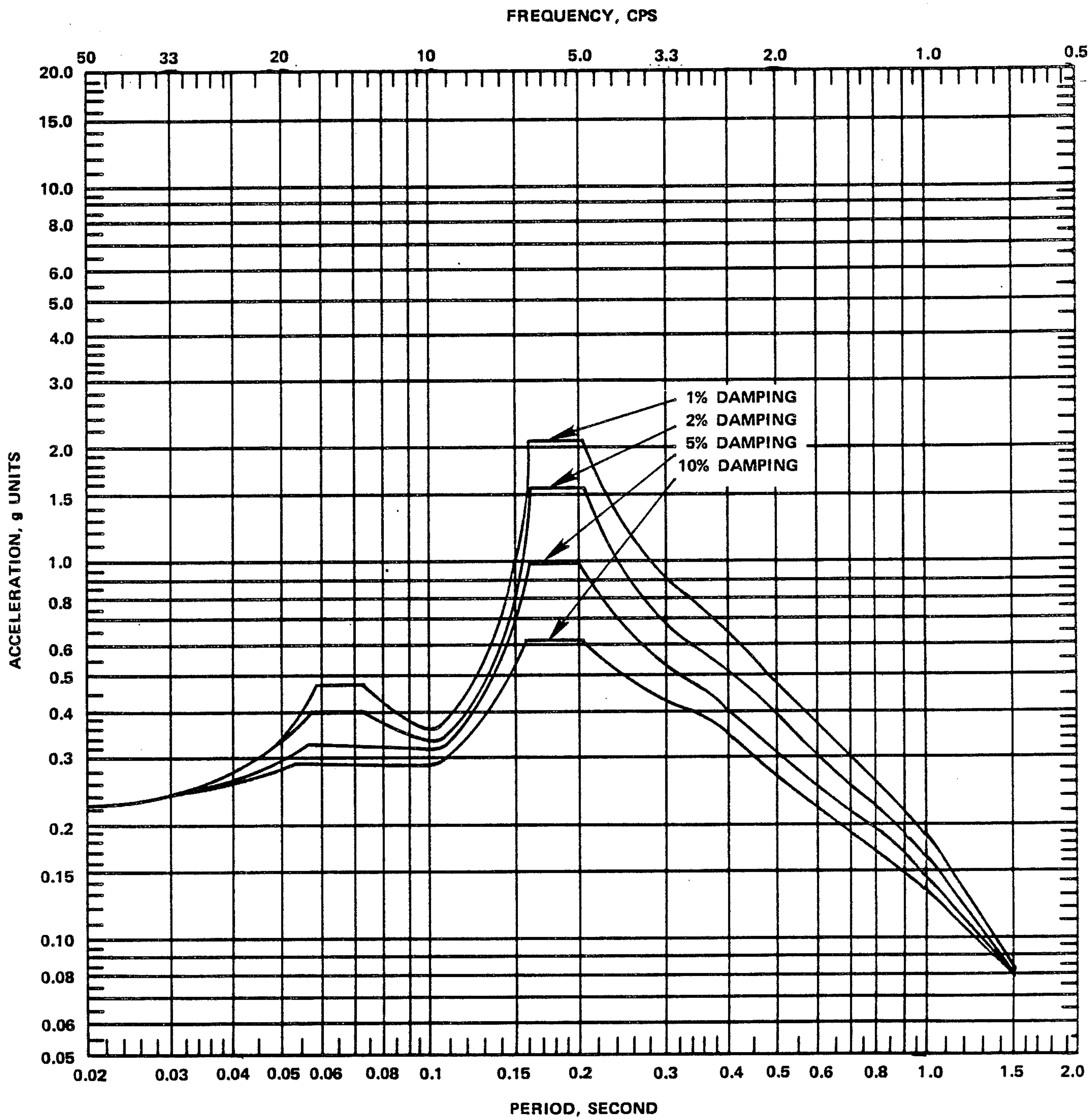
FIGURE 3.7-58

HORIZONTAL FLOOR RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
ELEVATION 613.5 FT – SLAB 2
REACTOR/AUXILIARY BUILDING NORTH-SOUTH
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682



Fermi 2

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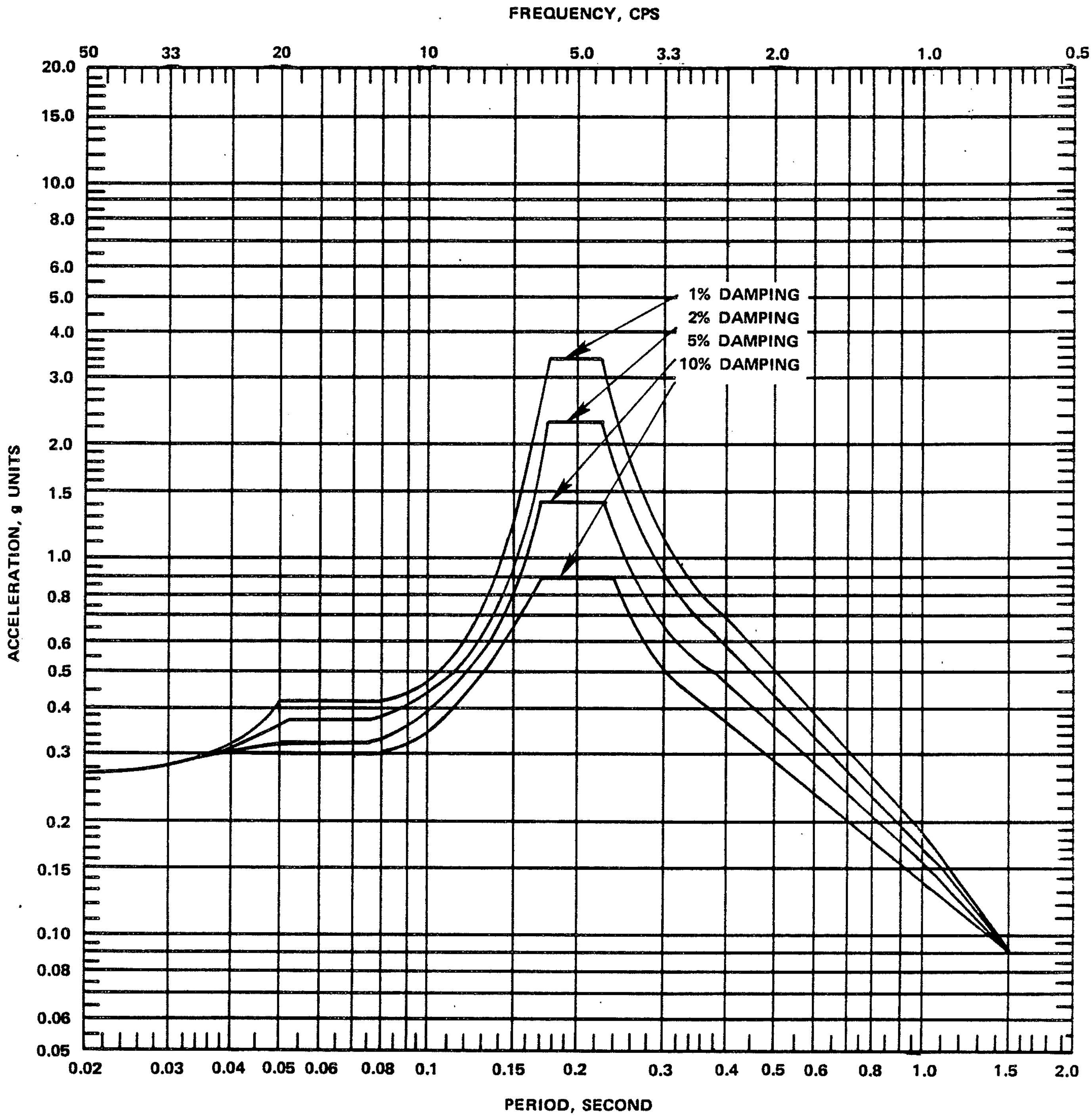
FIGURE 3.7-59

HORIZONTAL FLOOR RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
ELEVATION 613.5 FT – SLAB 2
REACTOR/AUXILIARY BUILDING EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682



Fermi 2

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FIGURE 3.7-60

HORIZONTAL FLOOR RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
ELEVATION 641.5 FT – SLAB 3
REACTOR/AUXILIARY BUILDING NORTH-SOUTH
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

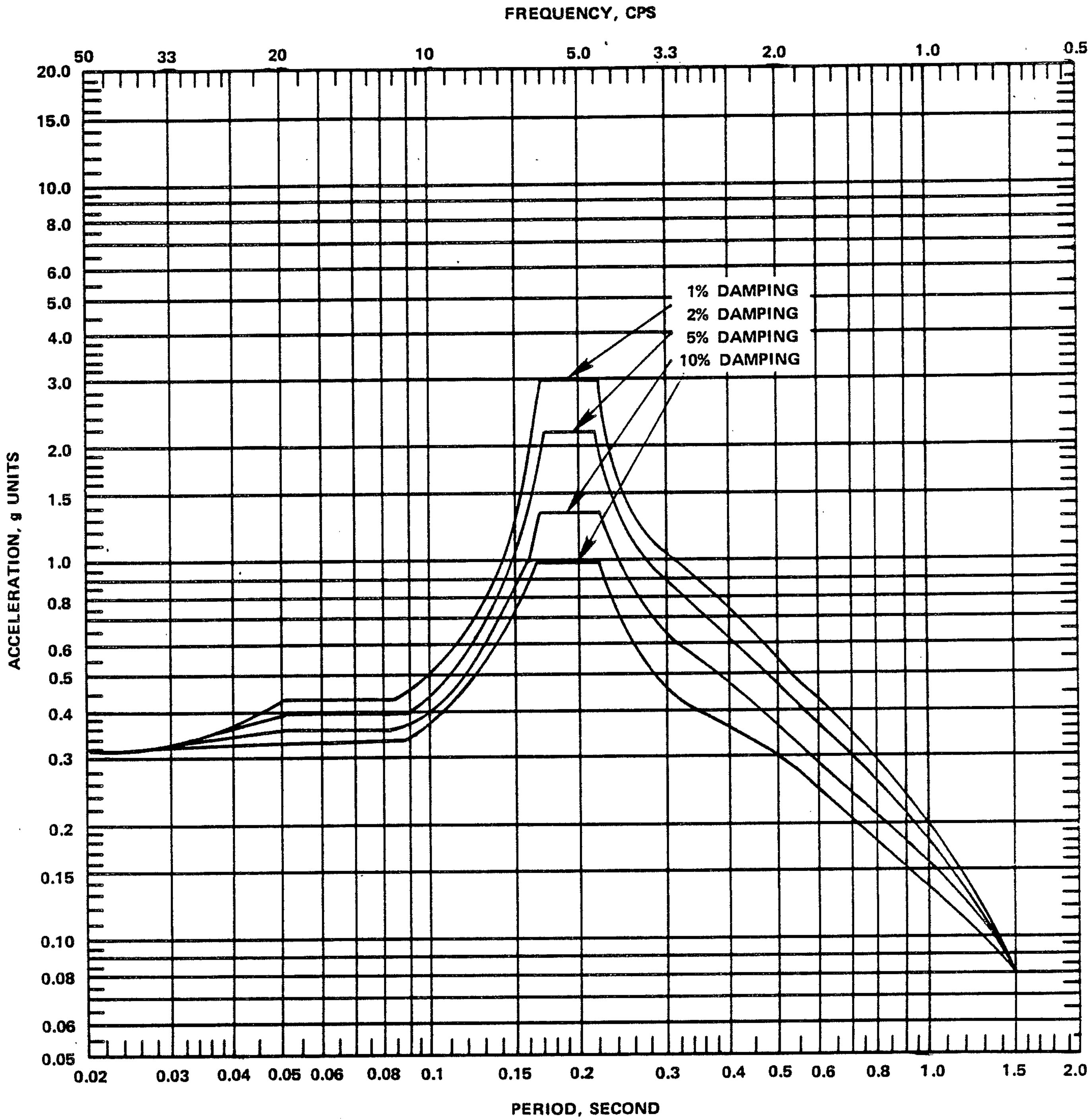

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FIGURE 3.7-61

HORIZONTAL FLOOR RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
ELEVATION 641.5 FT – SLAB 3
REACTOR/AUXILIARY BUILDING EAST-WEST
COMPONENT

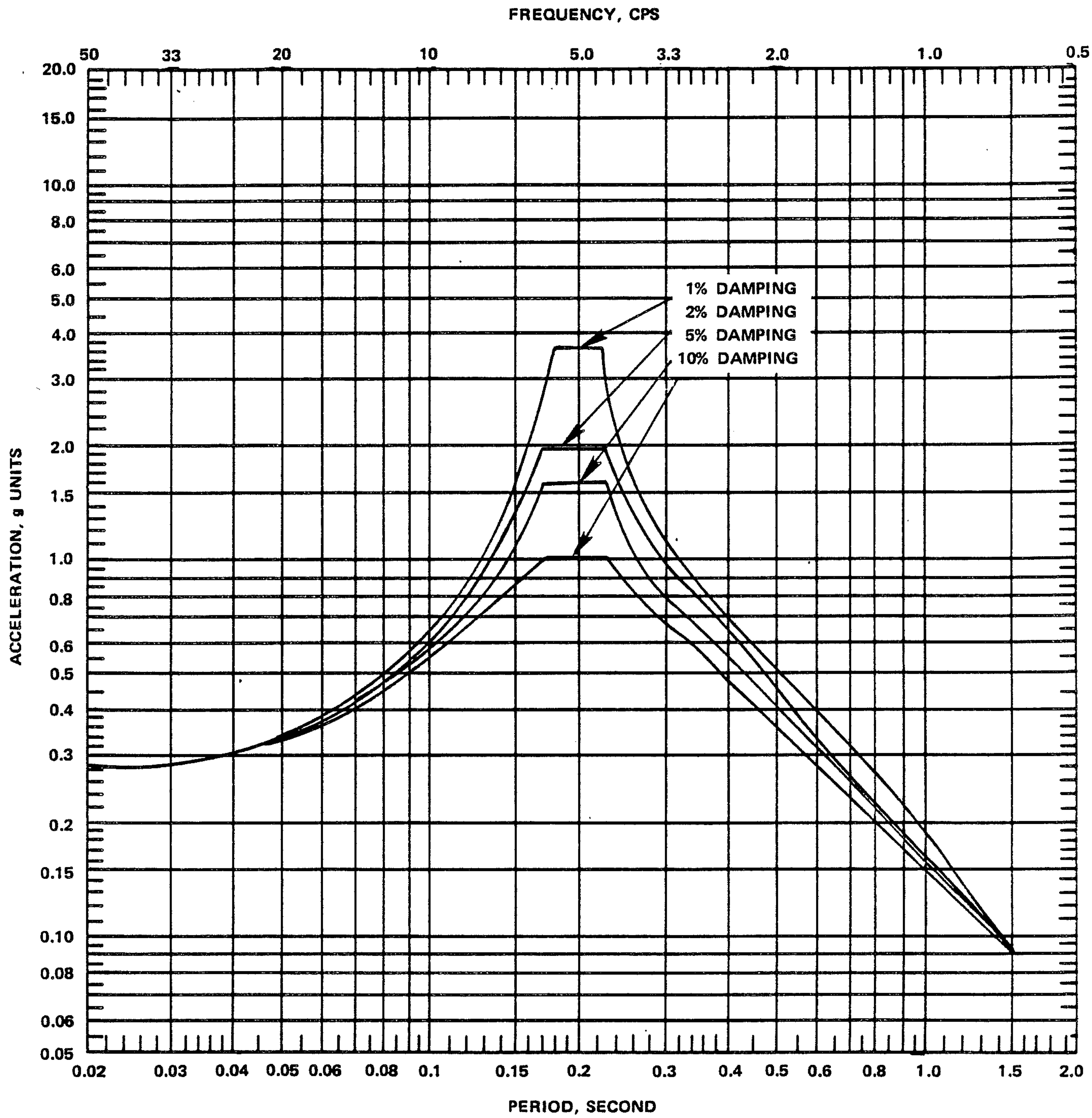

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FIGURE 3.7-62

HORIZONTAL FLOOR RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
ELEVATION 659.0 FT – SLAB 4
REACTOR/AUXILIARY BUILDING NORTH-SOUTH
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

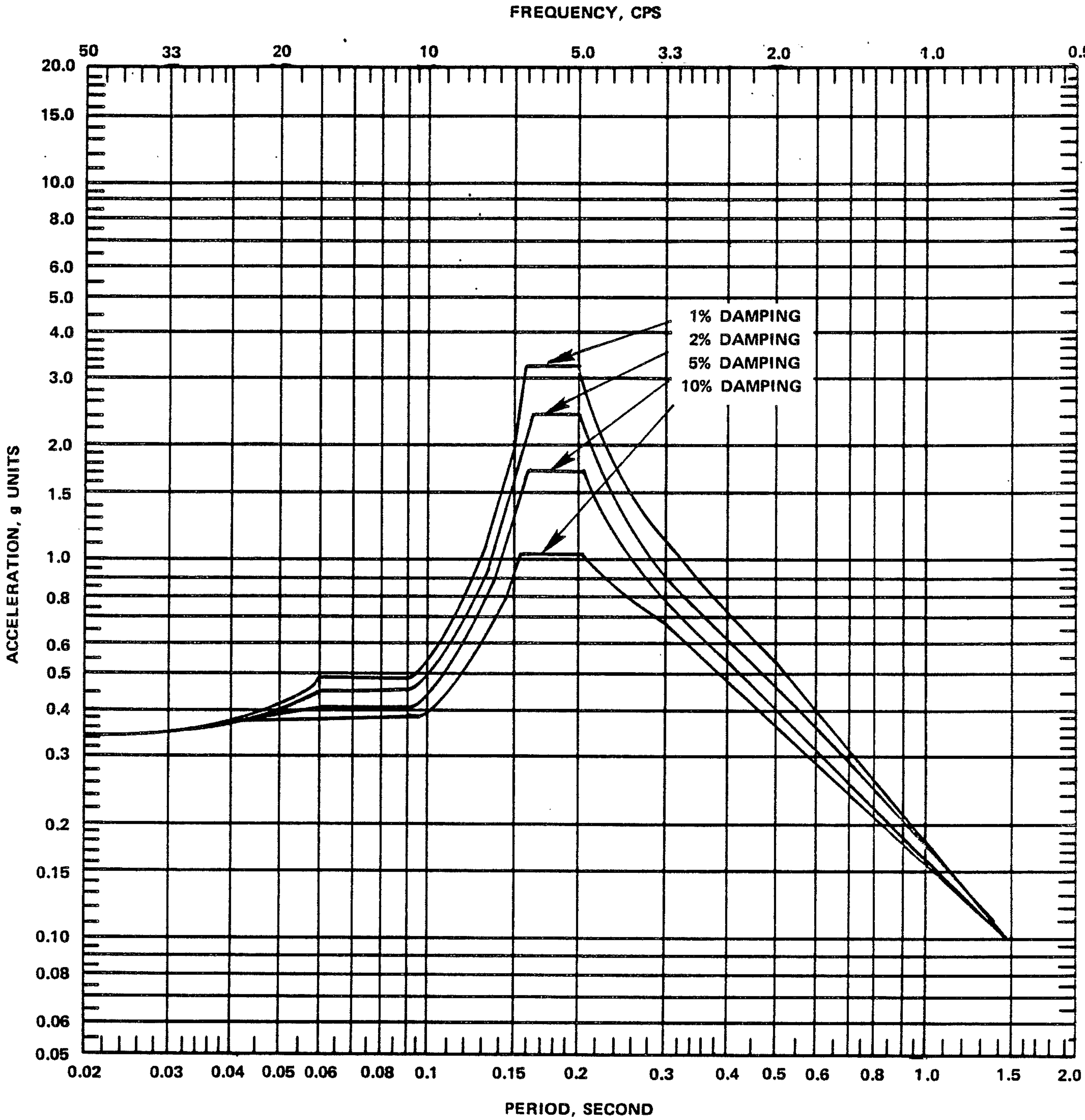

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FIGURE 3.7-63

HORIZONTAL FLOOR RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
ELEVATION 659.0 FT – SLAB 4
REACTOR/AUXILIARY BUILDING EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

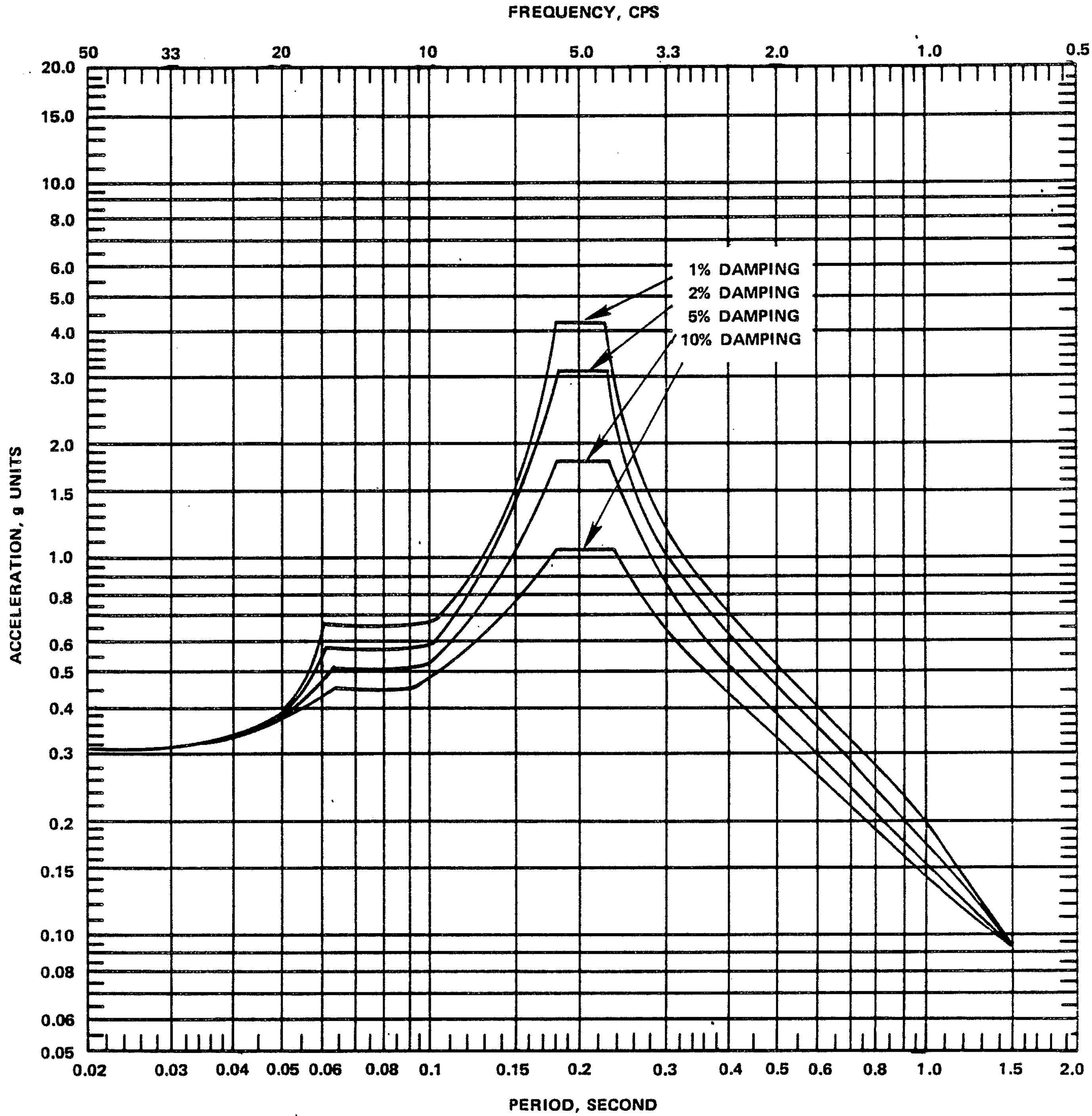

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FIGURE 3.7-64

HORIZONTAL FLOOR RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
ELEVATION 684.5 FT – SLAB 5
REACTOR/AUXILIARY BUILDING NORTH-SOUTH
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

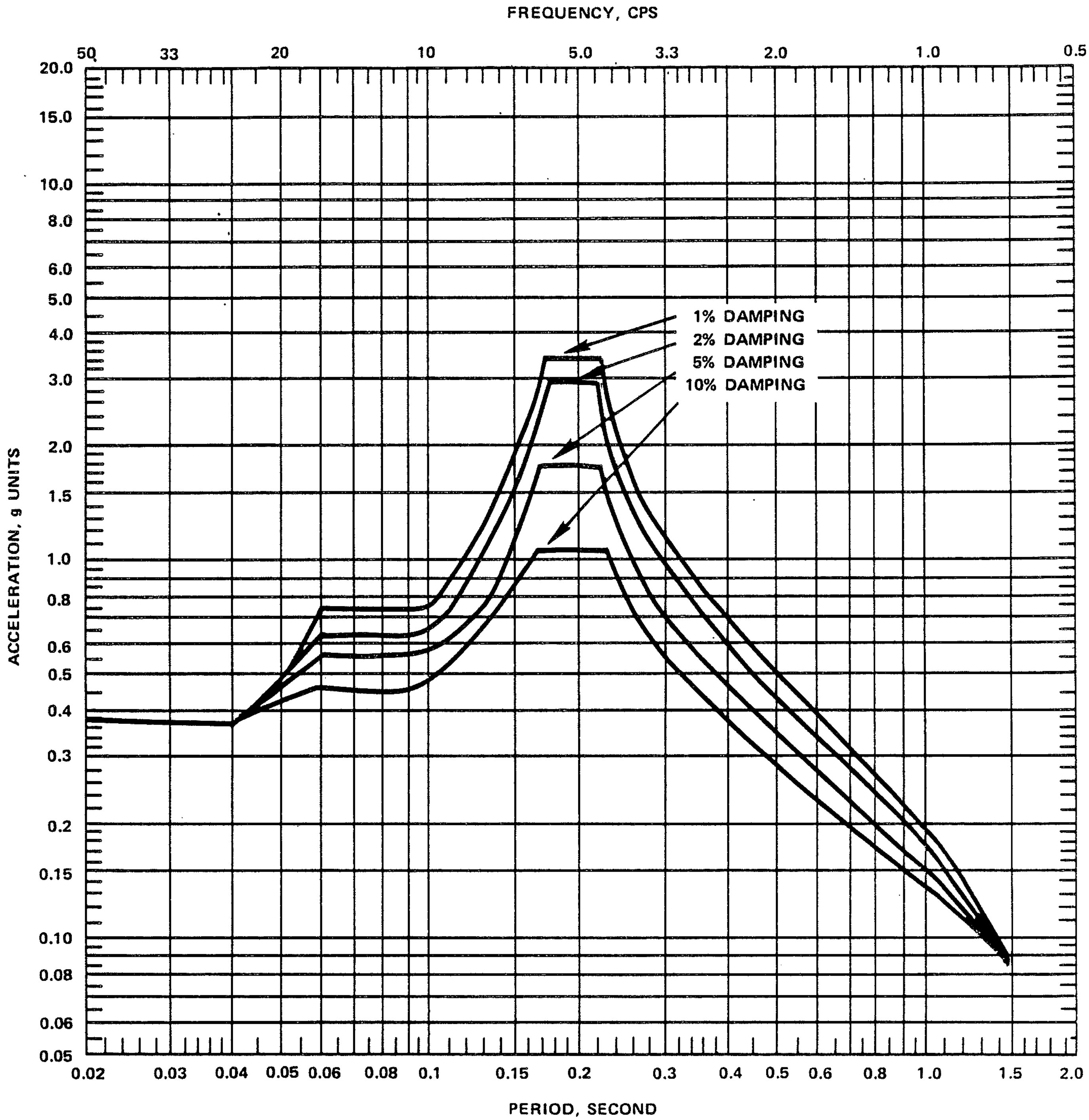

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FIGURE 3.7-65

HORIZONTAL FLOOR RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
ELEVATION 684.5 FT – SLAB 5
REACTOR/AUXILIARY BUILDING EAST-WEST
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

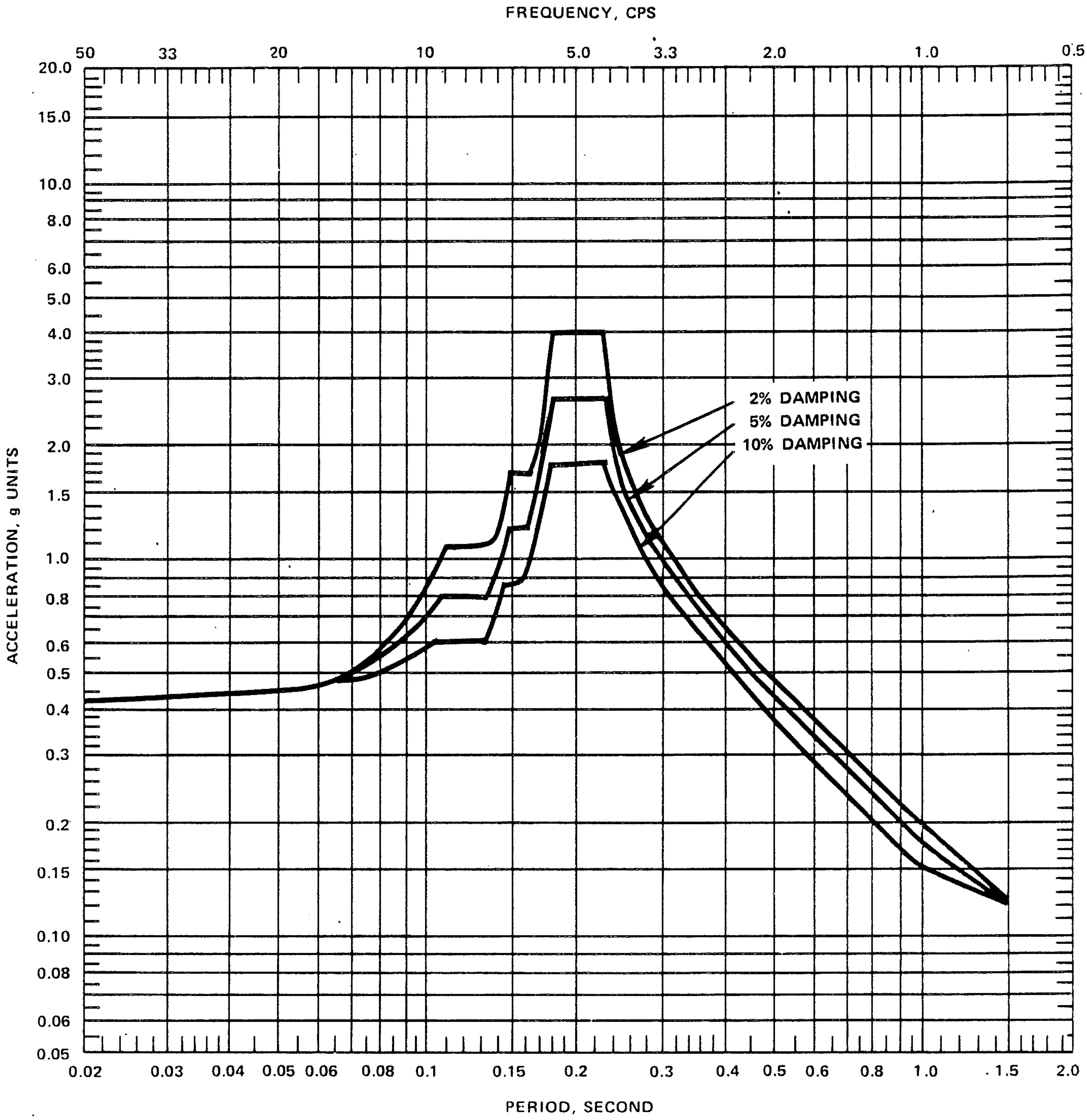

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FIGURE 3.7-66

HORIZONTAL RESPONSE SPECTRA REACTOR BUILDING CRANE – SAFE-SHUTDOWN EARTHQUAKE – CRANE ADJACENT TO COLUMN ROW 17 – NORTH-SOUTH COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

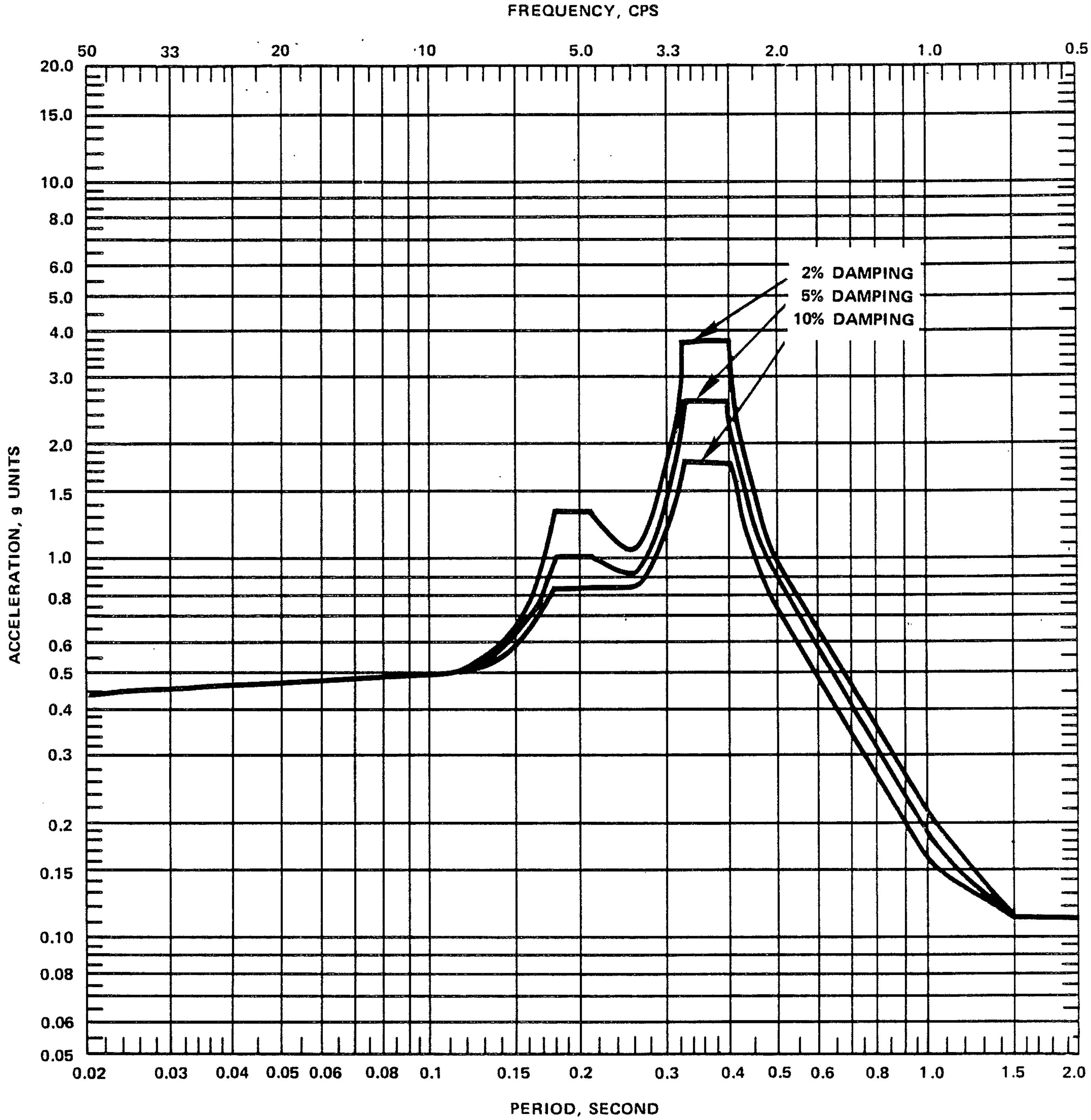

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FIGURE 3.7-67

HORIZONTAL RESPONSE SPECTRA REACTOR BUILDING CRANE – SAFE-SHUTDOWN EARTHQUAKE – CRANE ADJACENT TO COLUMN ROW 17 – EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

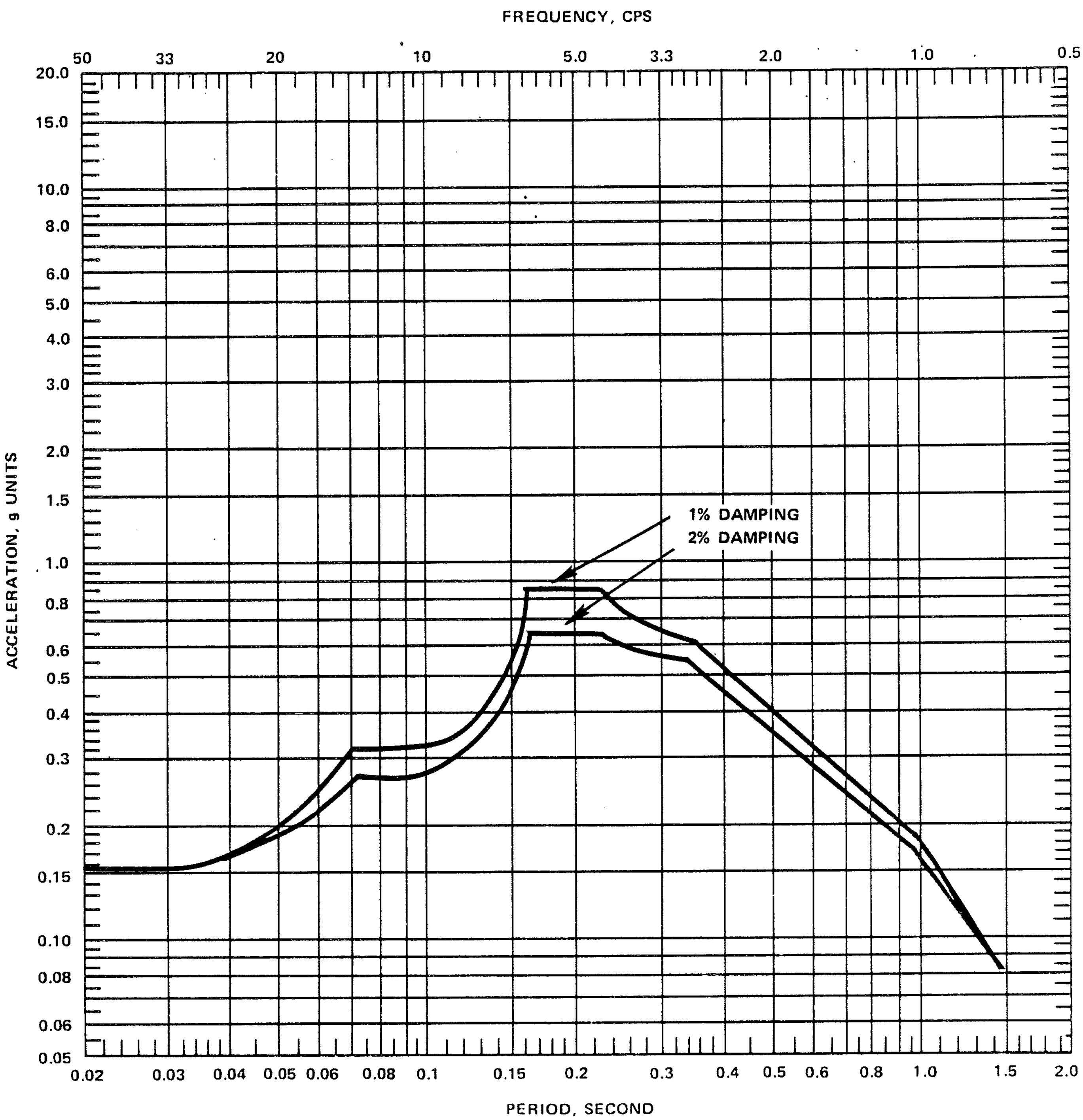

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FIGURE 3.7-68

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – DRYWELL
CONTAINMENT 18.0 FT BELOW REACTOR PRESSURE
VESSEL INVERT – NORTH-SOUTH COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

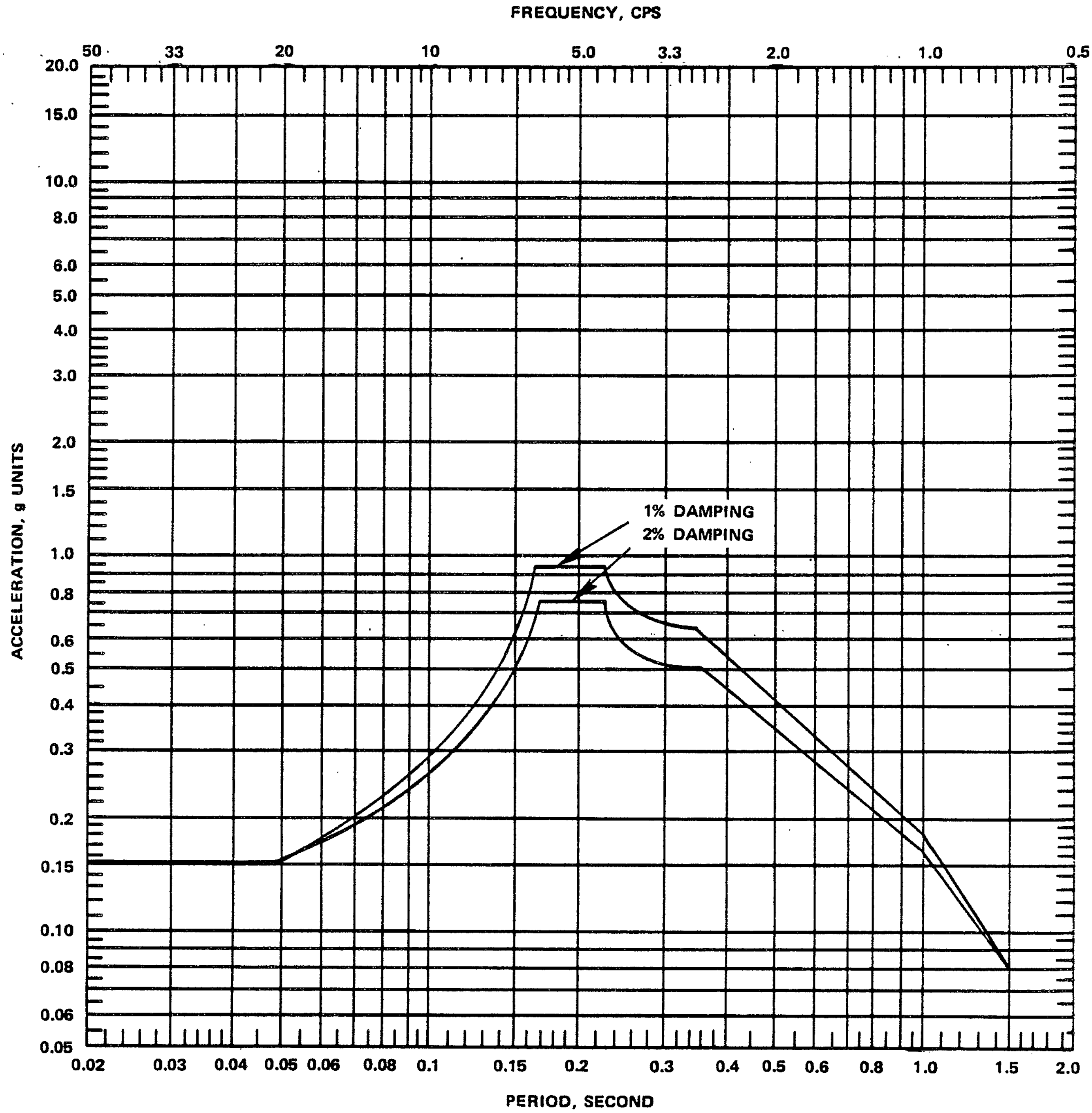

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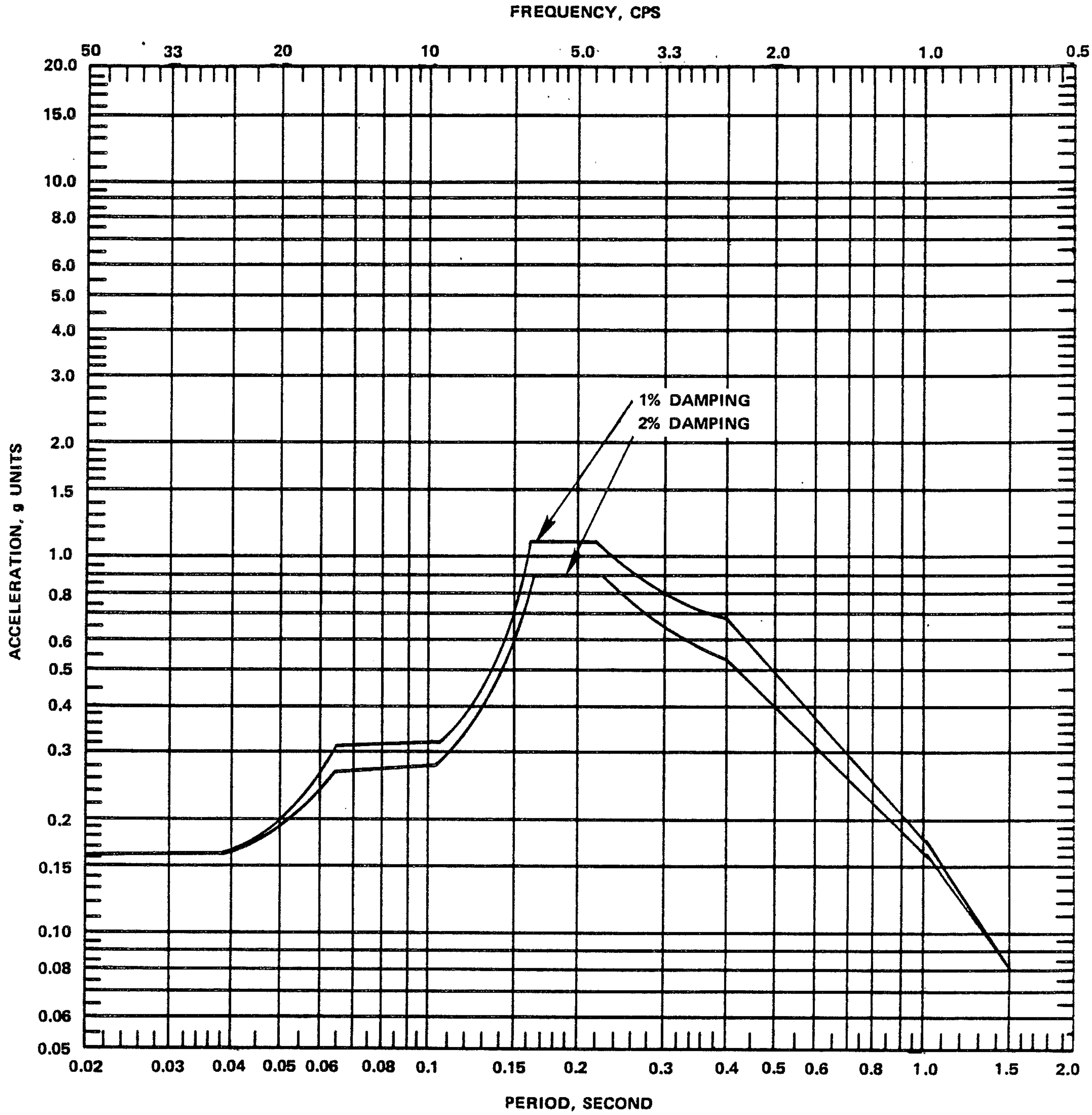
FIGURE 3.7-69

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – DRYWELL
CONTAINMENT 18.0 FT BELOW REACTOR PRESSURE
INVERT – EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682



SARGENT & LUNDY REPORT NO. SL-2682

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FIGURE 3.7-70

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – DRYWELL
CONTAINMENT 6.0 FT BELOW REACTOR PRESSURE
VESSEL INVERT – NORTH-SOUTH COMPONENT

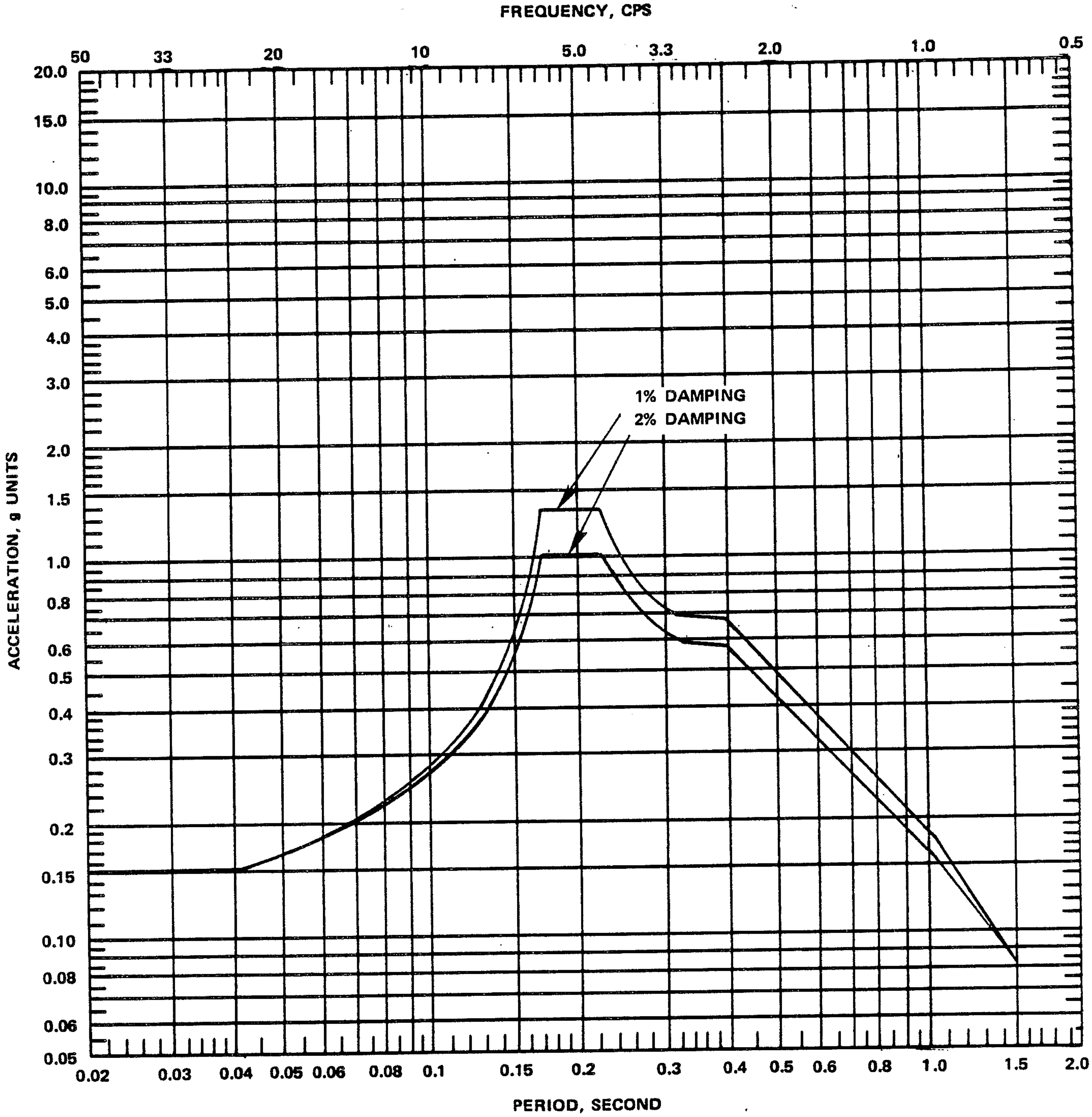

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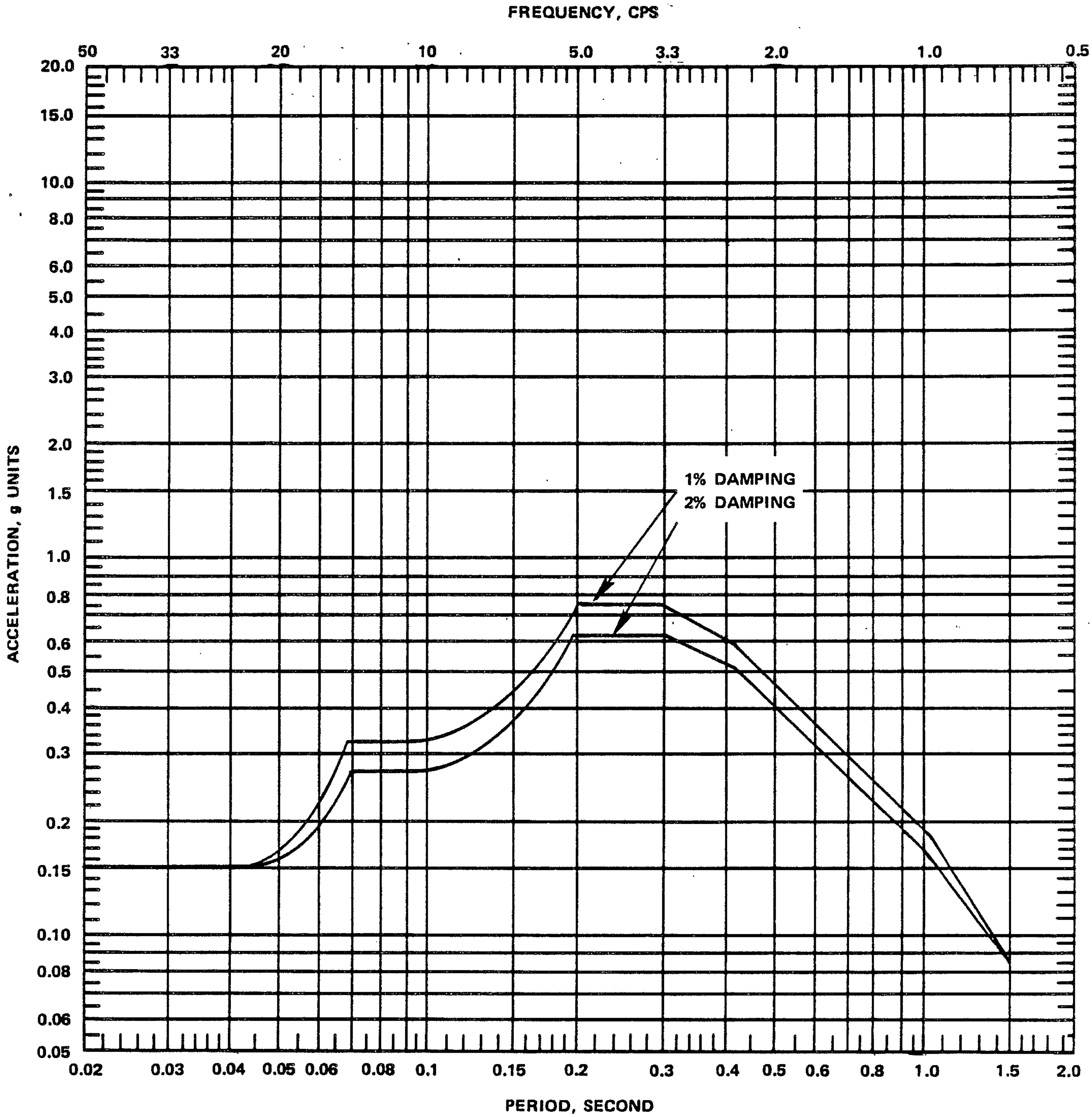
FIGURE 3.7-71

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – DRYWELL
CONTAINMENT 6.0 FT BELOW REACTOR PRESSURE
VESSEL INVERT – EAST-WEST COMPONENT



SARGENT & LUNDY REPORT NO. SL-2682

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FIGURE 3.7-72

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – REACTOR
PEDESTAL 18.0 FT BELOW REACTOR PRESSURE
VESSEL INVERT – NORTH-SOUTH COMPONENT

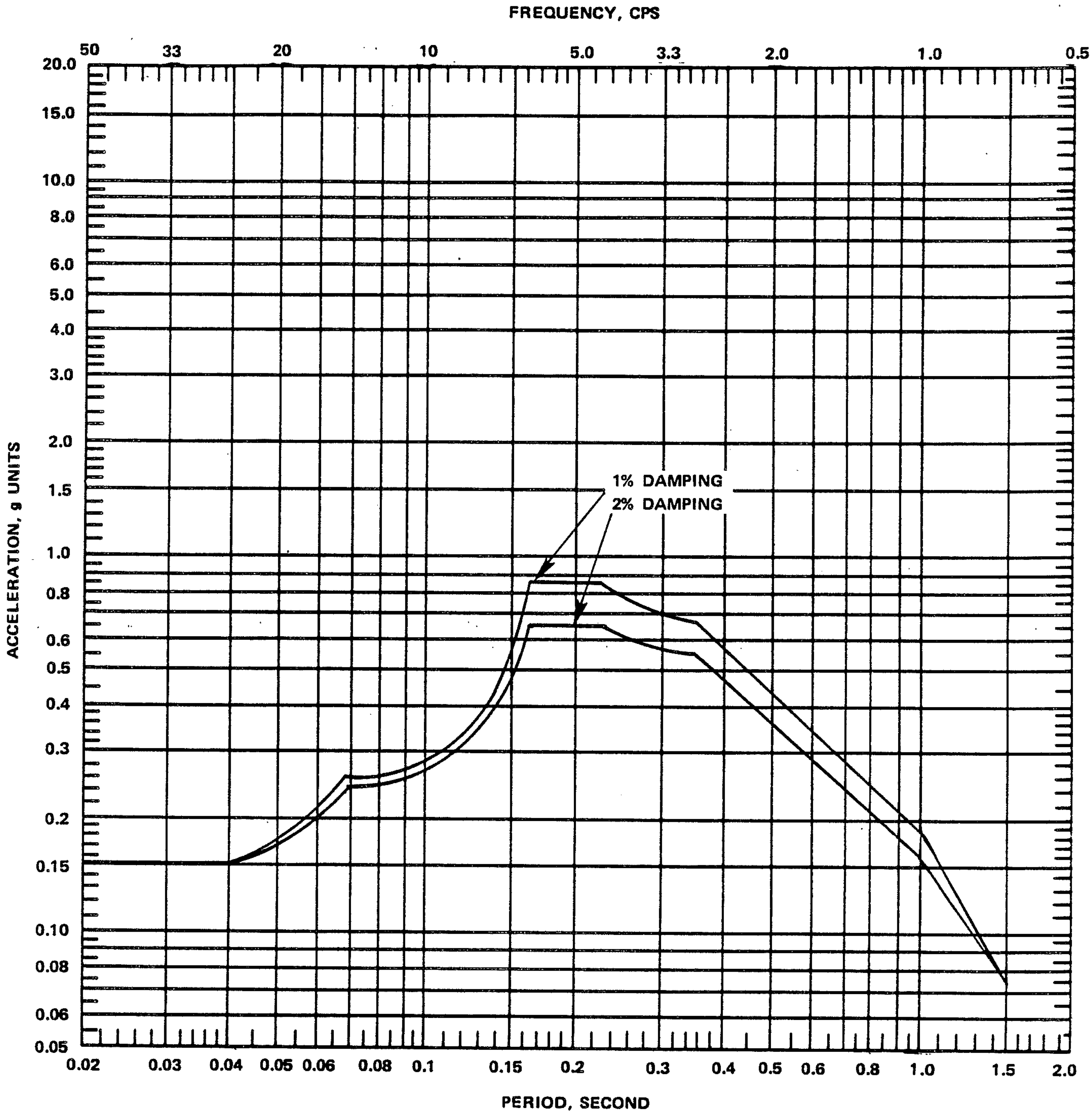

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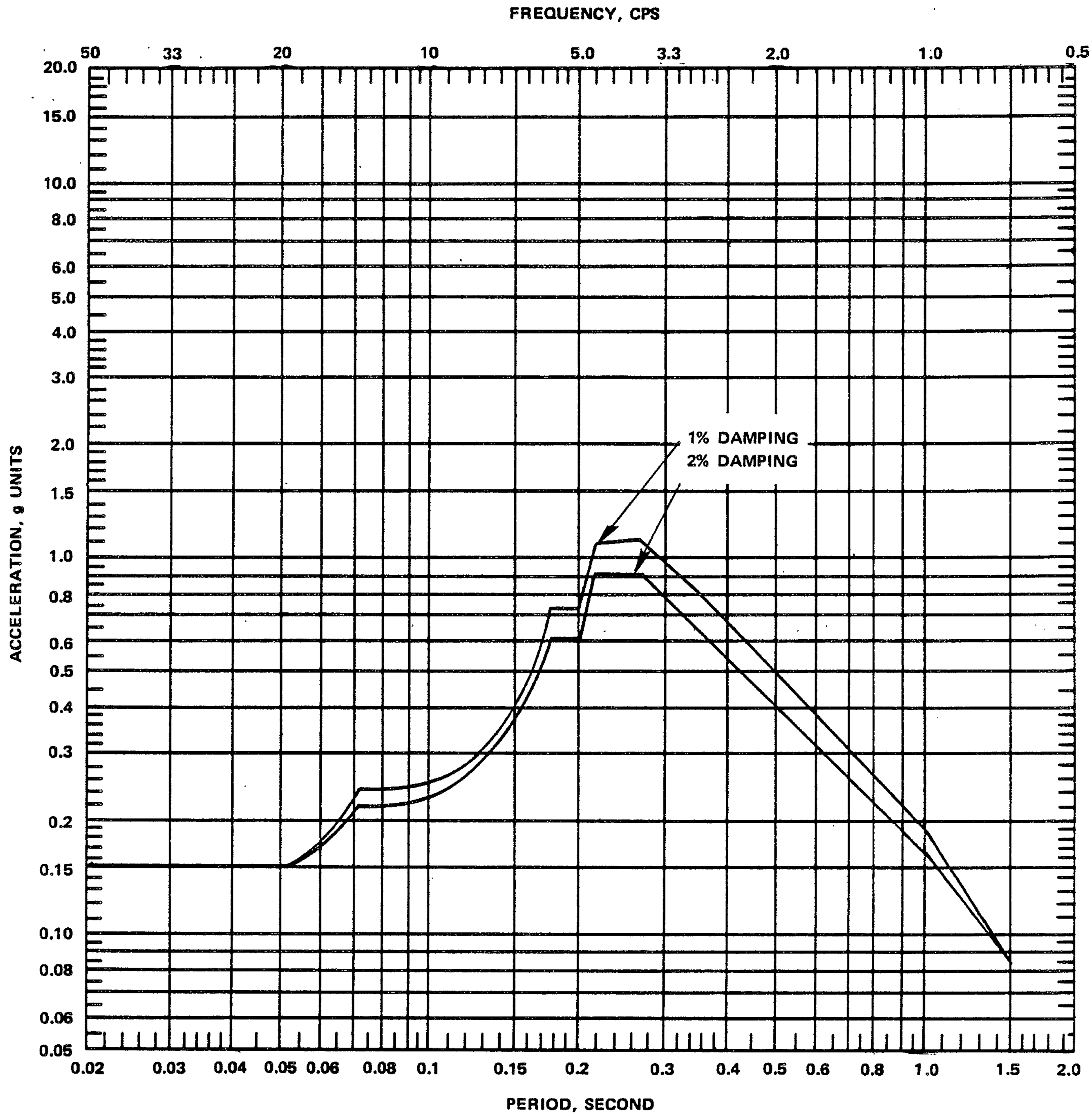
FIGURE 3.7-73

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – REACTOR
PEDESTAL 18.0 FT BELOW REACTOR PRESSURE
VESSEL INVERT – EAST-WEST COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682



SARGENT & LUNDY REPORT NO. SL-2682

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FIGURE 3.7-74

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
TOP OF REACTOR PEDESTAL
NORTH-SOUTH COMPONENT

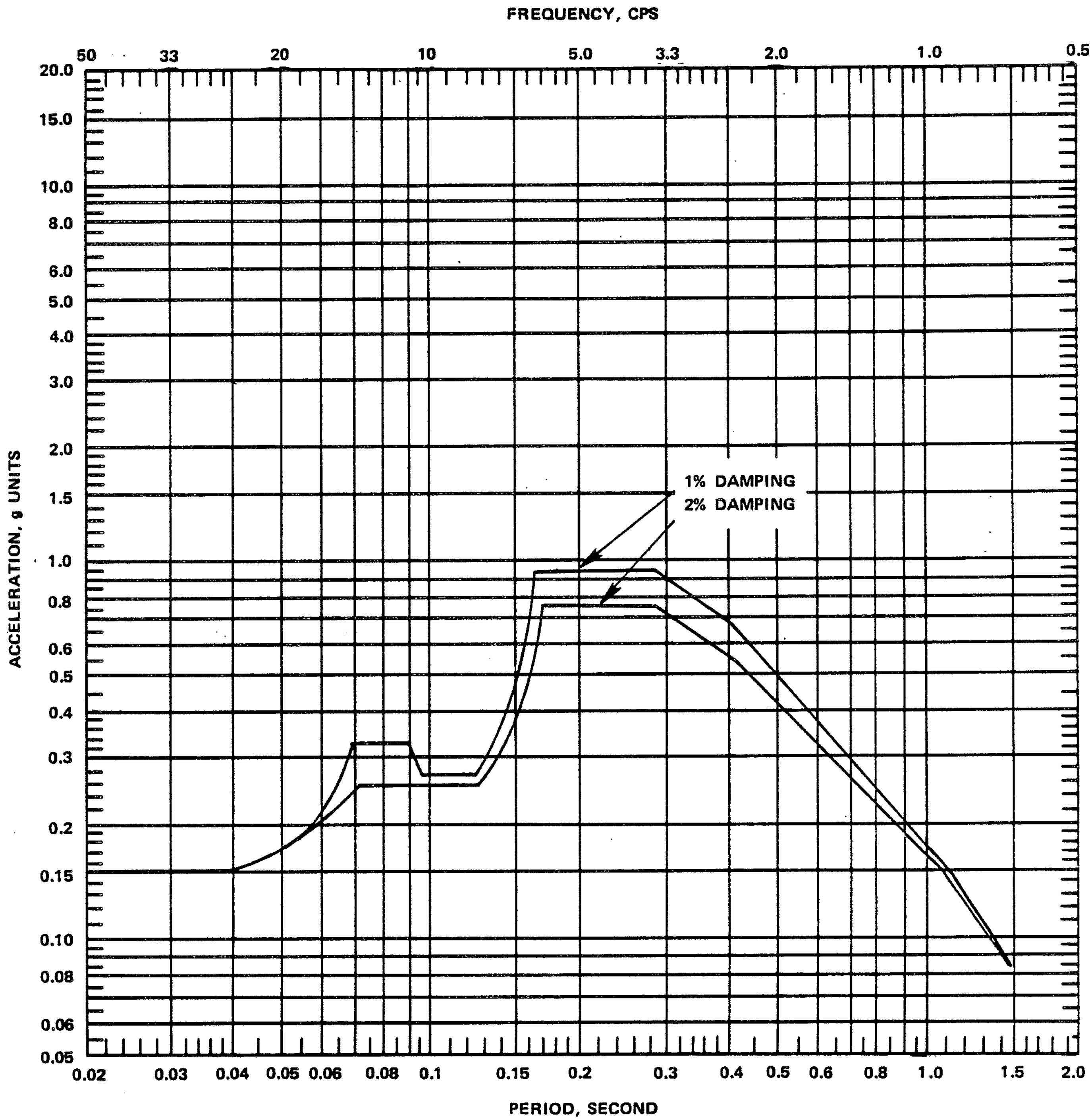

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FIGURE 3.7-75

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
TOP OF REACTOR PEDESTAL
EAST-WEST COMPONENT

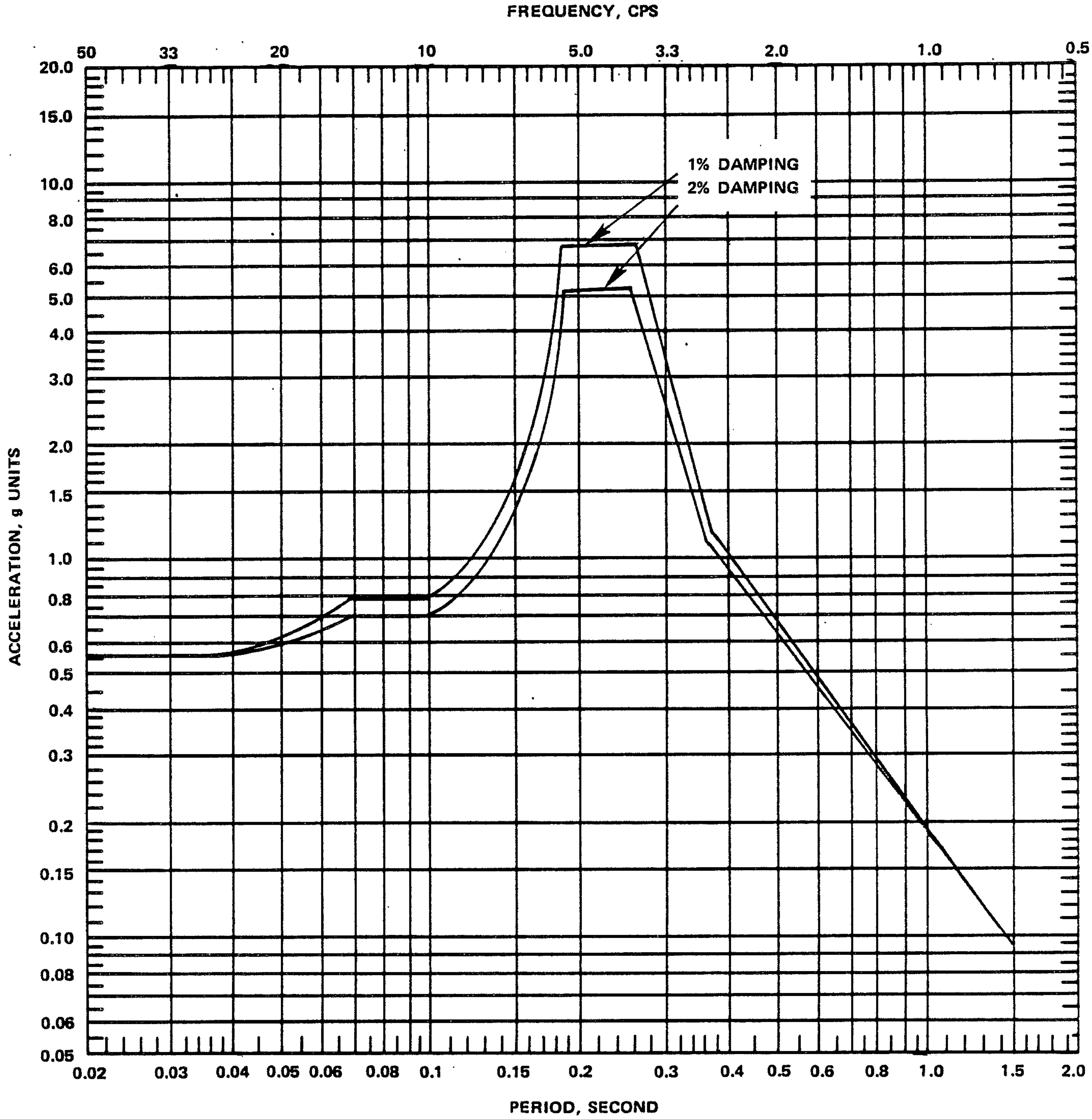

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FIGURE 3.7-76

HORIZONTAL RESPONSE SPECTRA SAFE-SHUTDOWN EARTHQUAKE – REACTOR PRESSURE VESSEL 14.0 FT ABOVE REACTOR PRESSURE VESSEL INVERT – NORTH-SOUTH COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

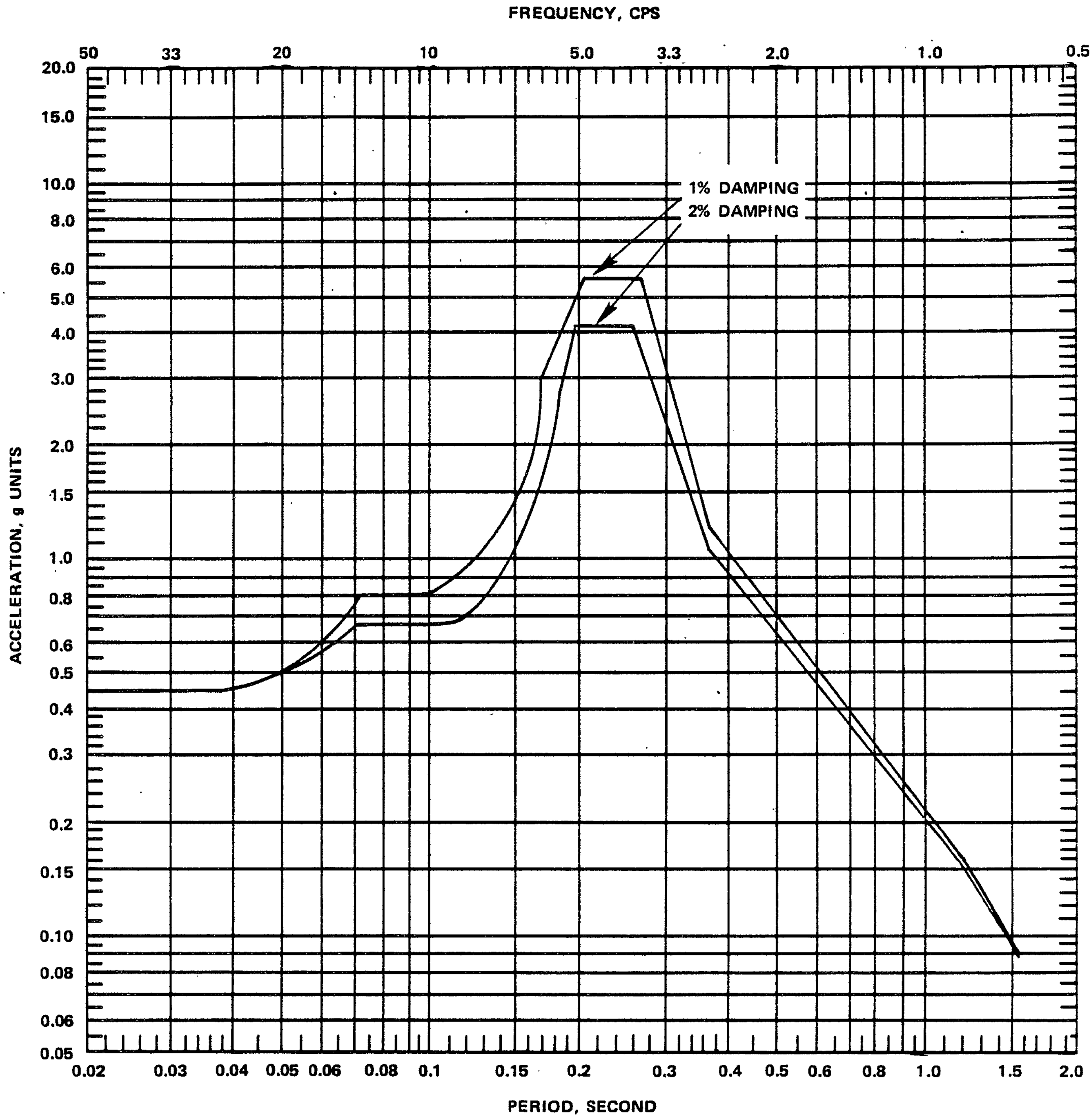

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FIGURE 3.7-77

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – REACTOR
PRESSURE VESSEL 14.0 FT ABOVE REACTOR
PRESSURE VESSEL INVERT – EAST-WEST
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

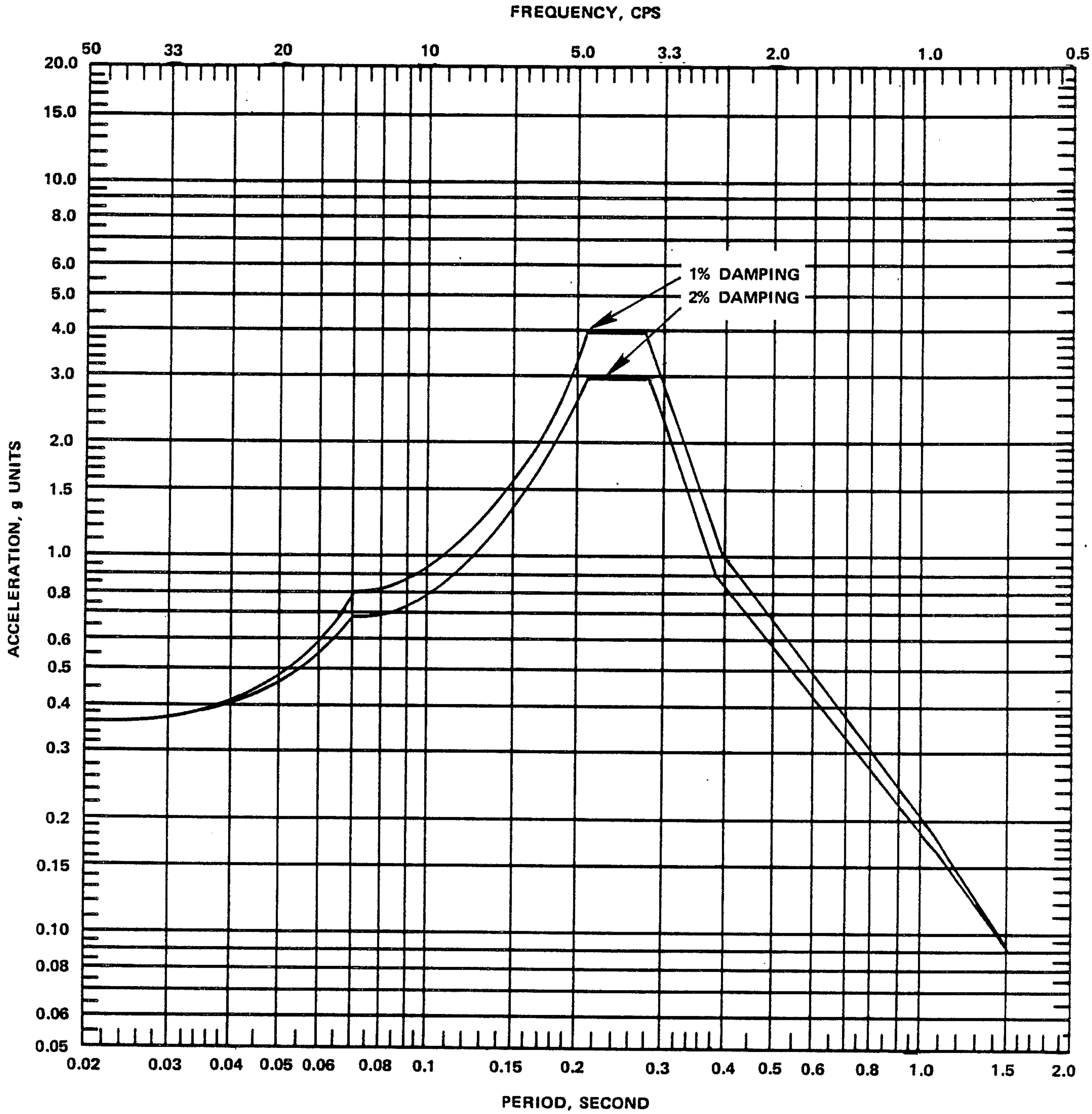

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FIGURE 3.7-78

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – REACTOR
PRESSURE VESSEL 54.0 FT ABOVE REACTOR
PRESSURE VESSEL INVERT – NORTH-SOUTH
COMPONENT

SARGENT & LUNDY REPORT NO. SL-2682

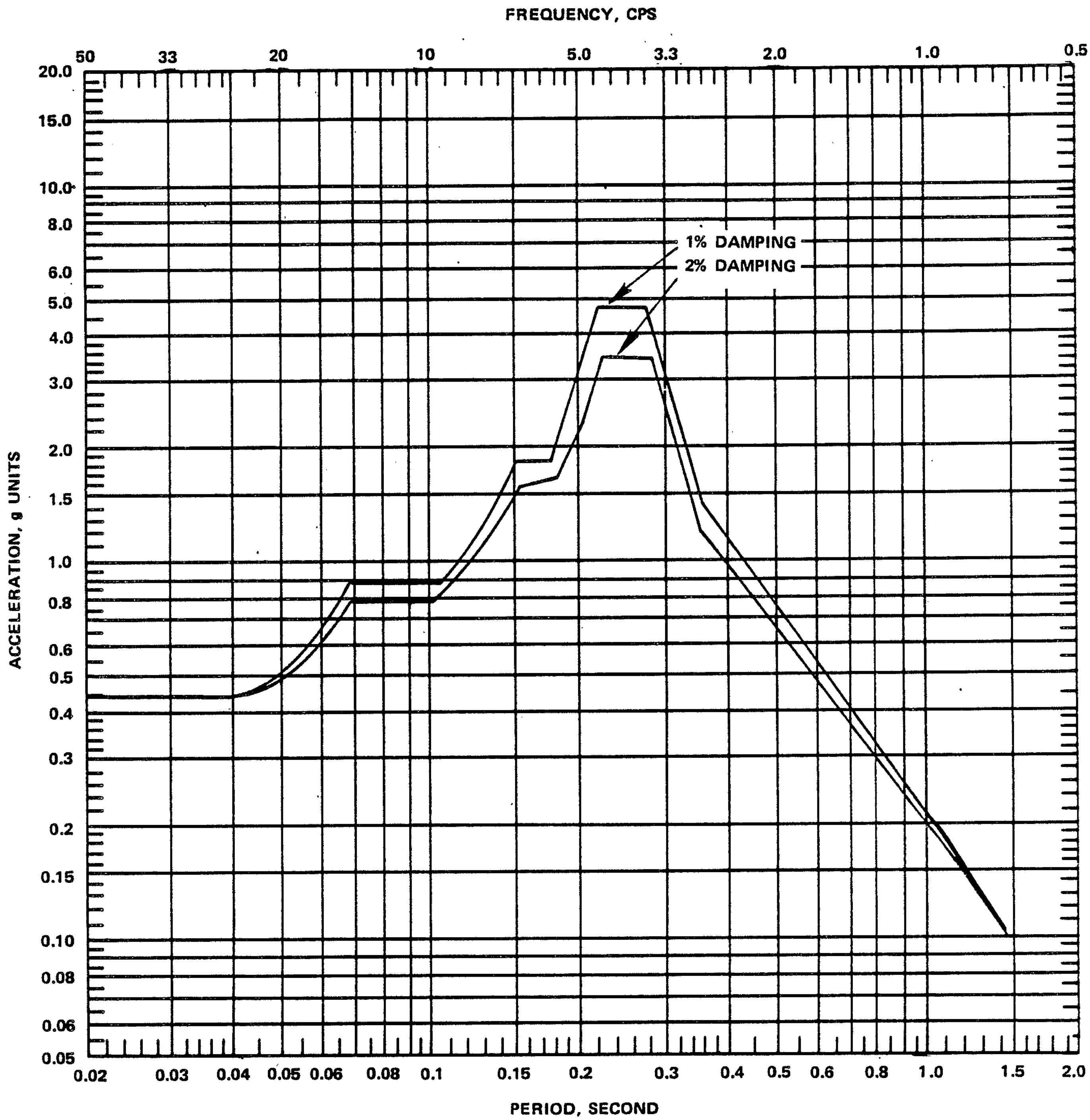

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FIGURE 3.7-79

HORIZONTAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – REACTOR
PRESSURE VESSEL 54.0 FT ABOVE REACTOR
PRESSURE VESSEL INVERT – EAST-WEST
COMPONENT

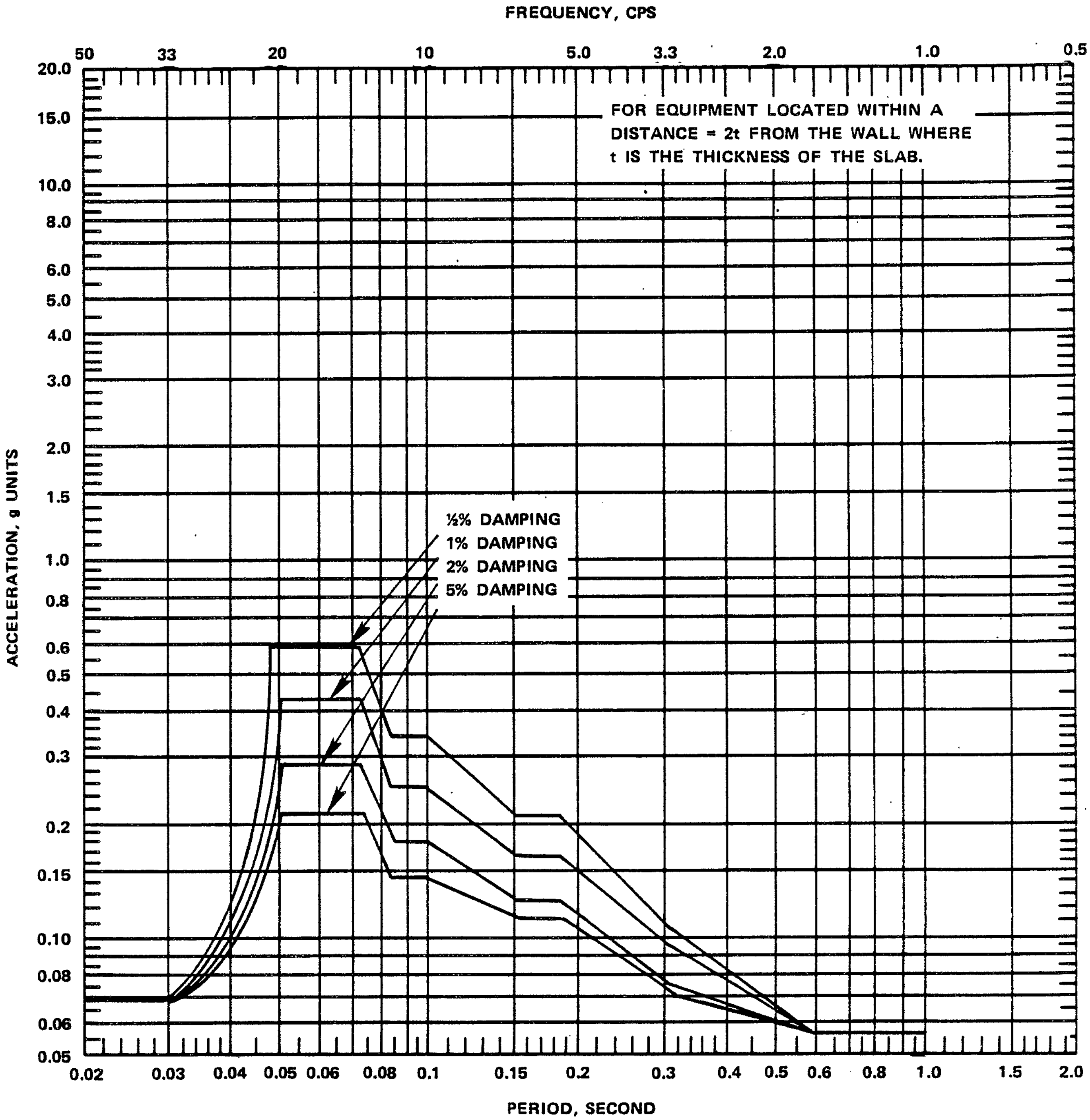

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FIGURE 3.7-80

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
REACTOR CONTAINMENT SHIELD
ELEVATIONS 583.5 FT AND 613.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

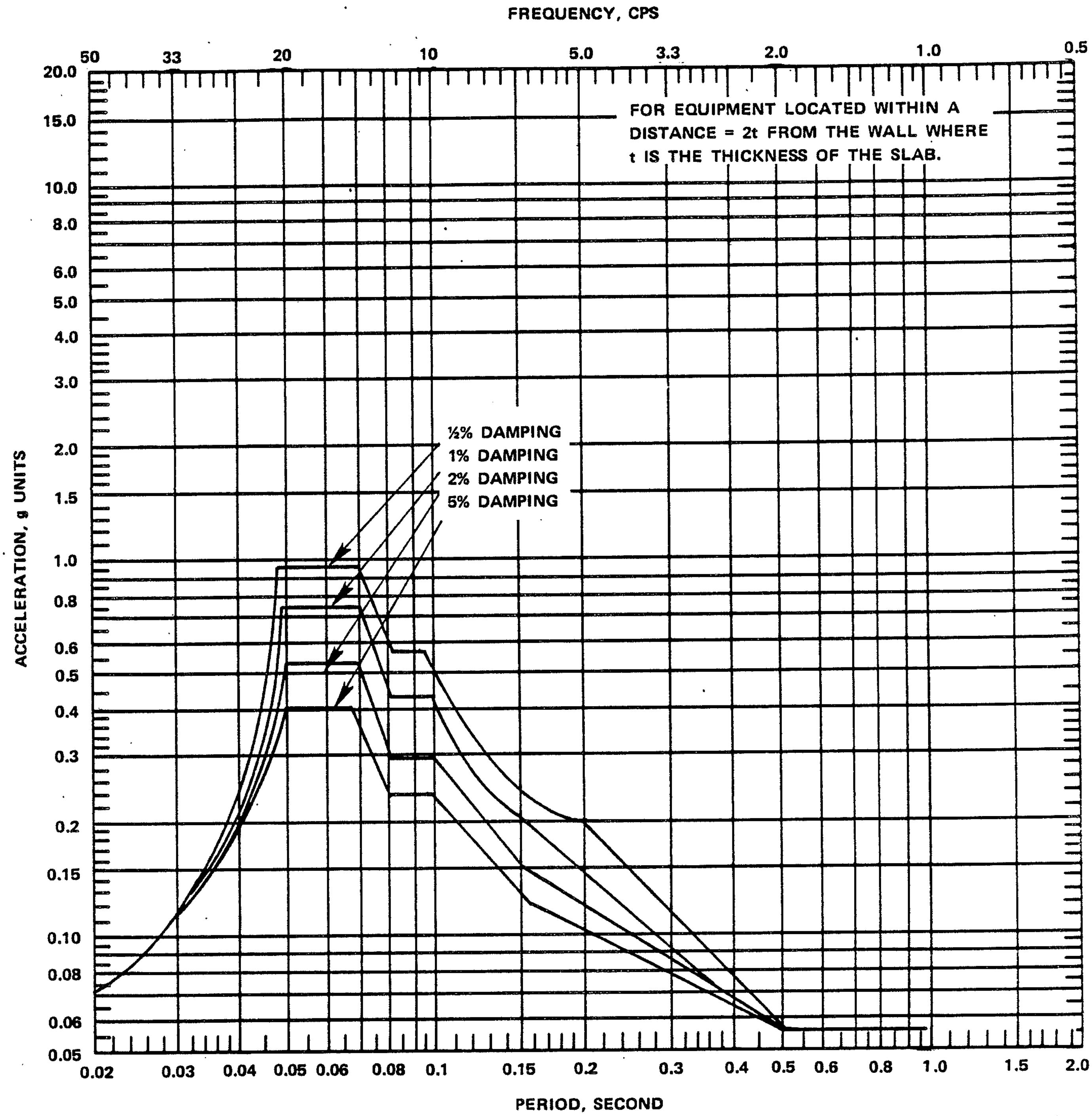

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FIGURE 3.7-81

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
REACTOR CONTAINMENT SHIELD
ELEVATIONS 643.5 FT, 659.5 FT, AND 684.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

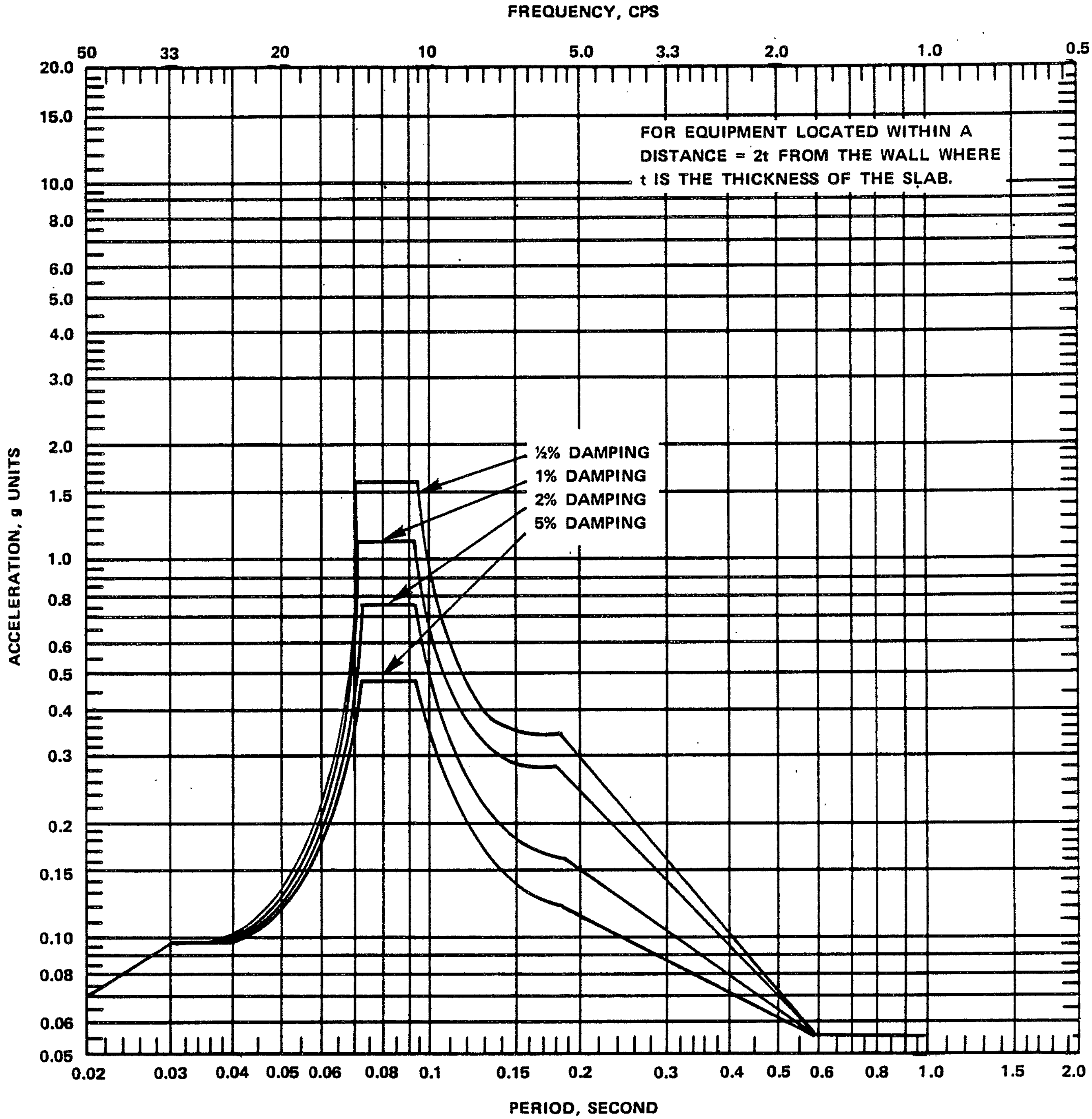

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FIGURE 3.7-82

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
REACTOR/AUXILIARY BUILDING WALL
ELEVATIONS 583.5 FT AND 613.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

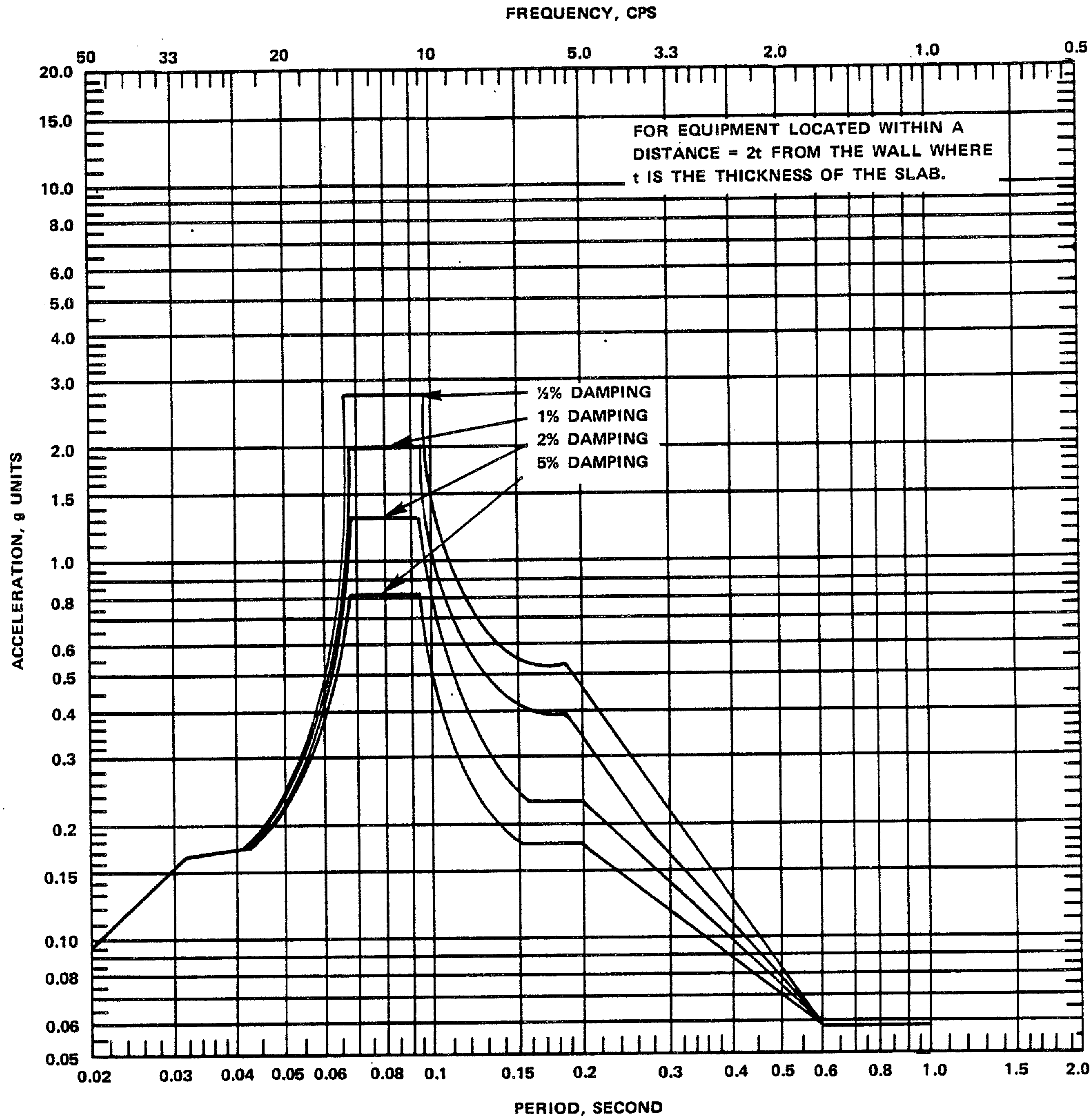

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FIGURE 3.7-83

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
REACTOR/AUXILIARY BUILDING WALL
ELEVATIONS 641.5 FT, 659.5 FT, AND 684.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

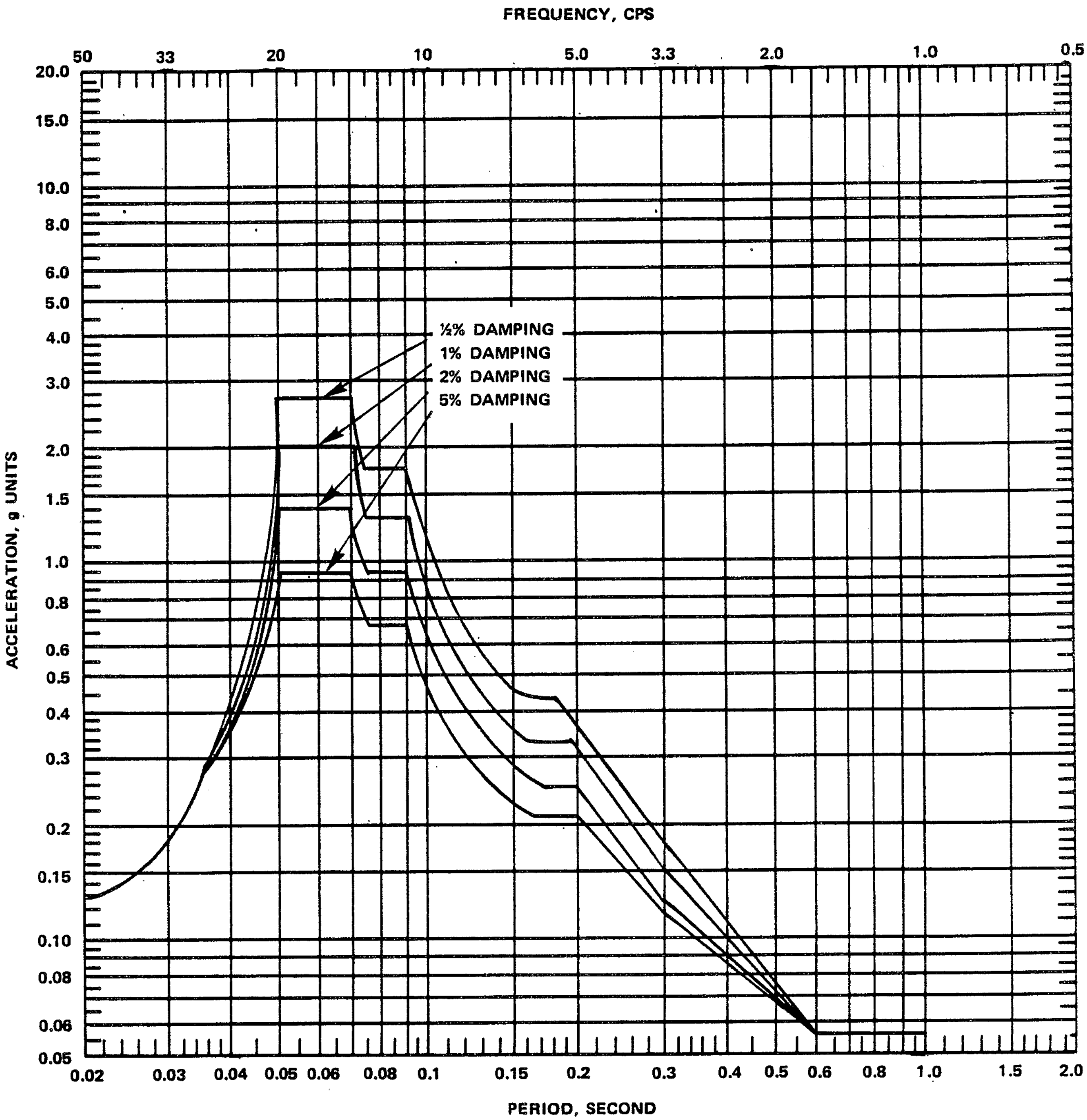

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FIGURE 3.7-84

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
REACTOR BUILDING SLAB
ELEVATIONS 583.5 FT AND 613.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

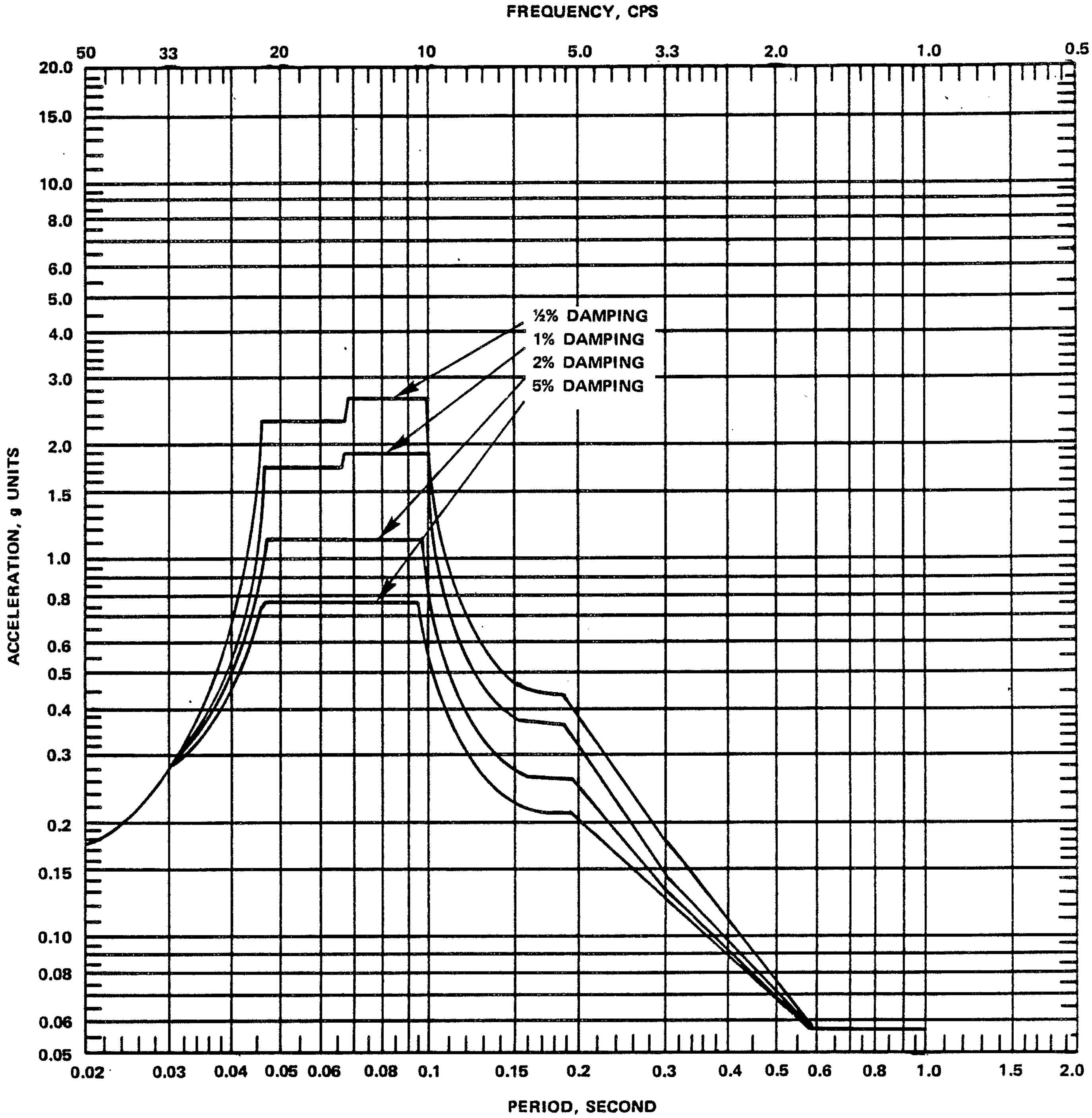

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FIGURE 3.7-85

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
REACTOR BUILDING SLAB
ELEVATIONS 641.5 FT, 659.5 FT, AND 684.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

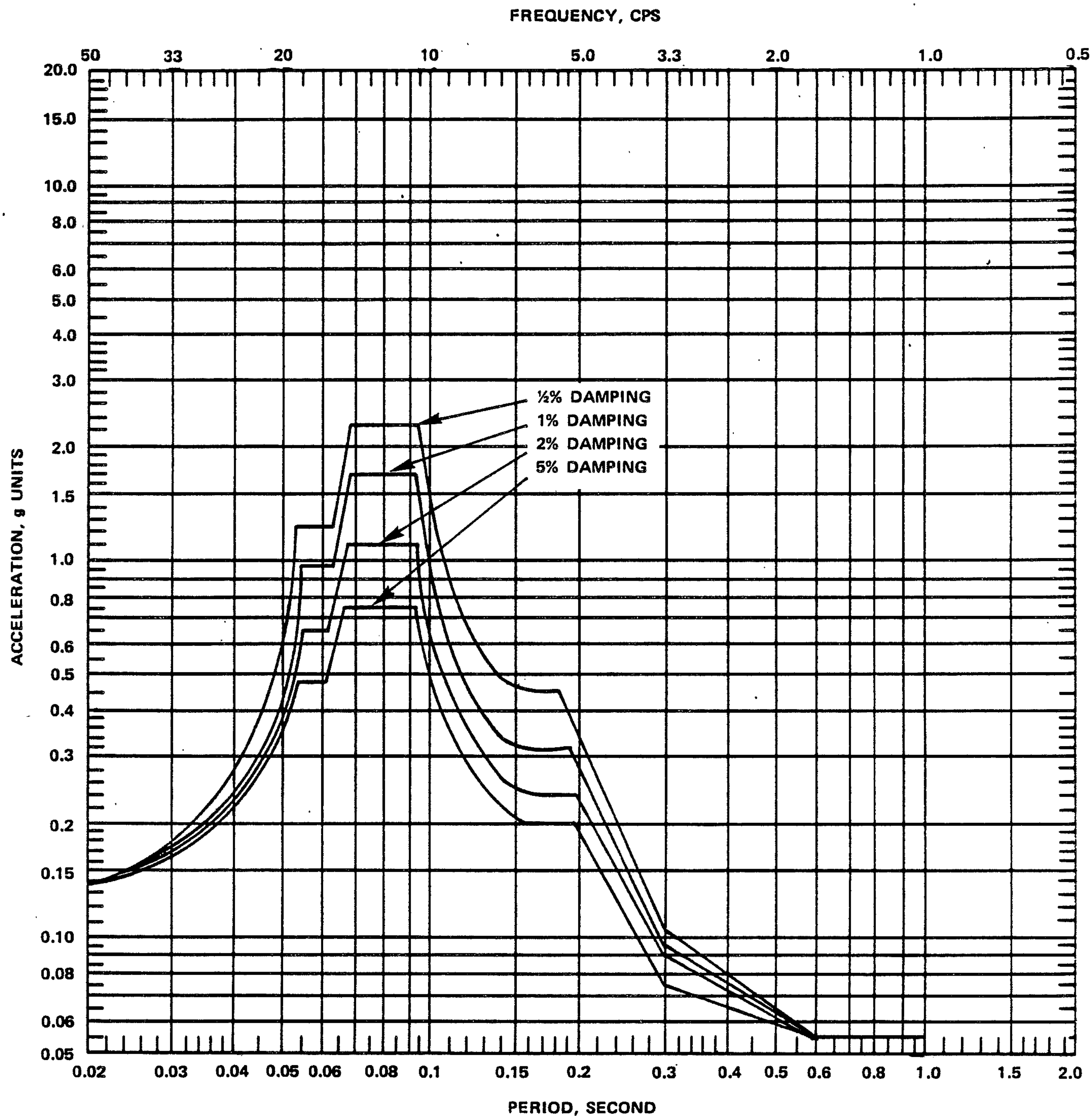

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FIGURE 3.7-86

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
AUXILIARY BUILDING SLAB
ELEVATIONS 583.5 FT, 613.5 FT, AND 659.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

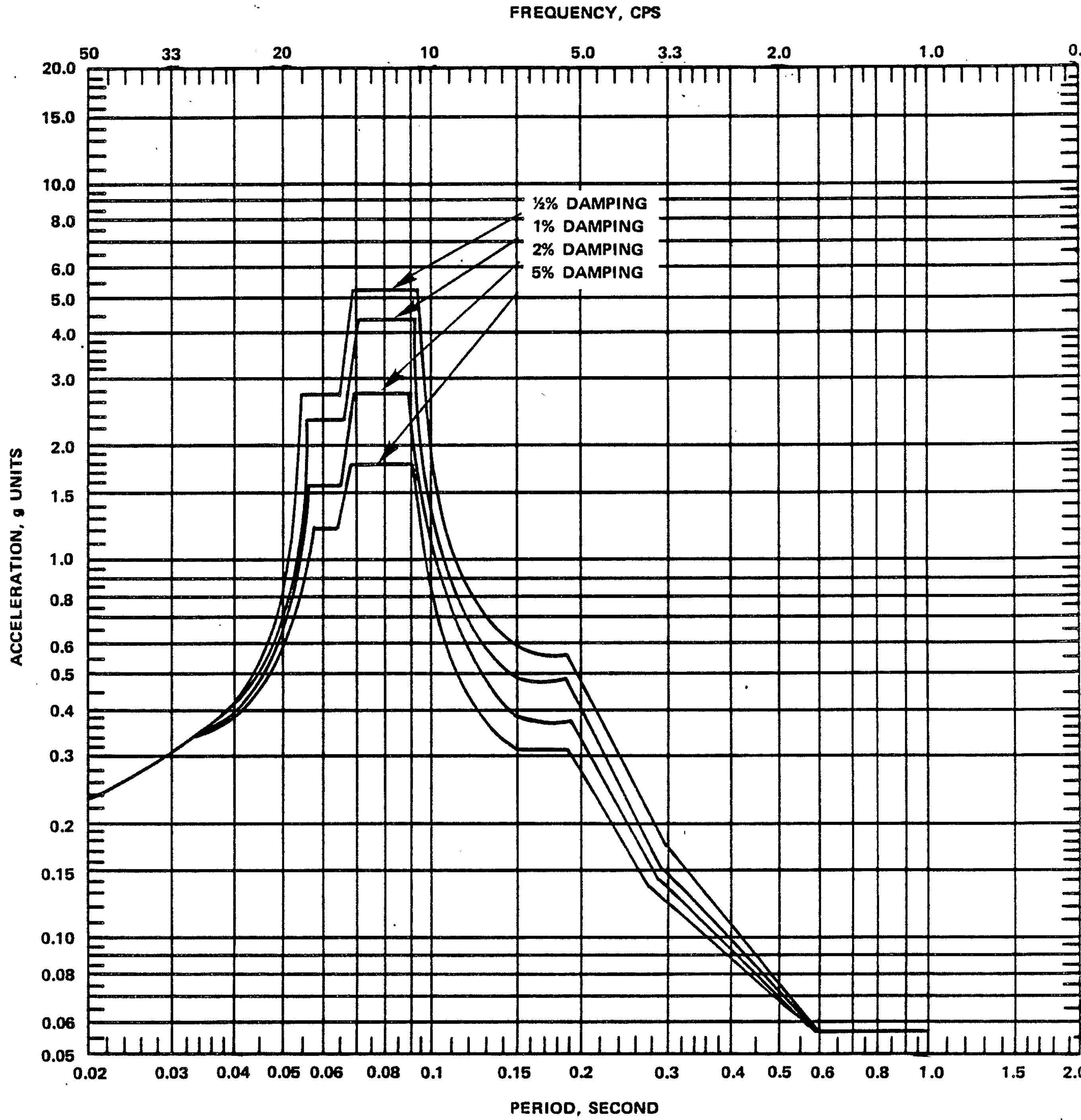

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FIGURE 3.7–87

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
AUXILIARY BUILDING SLAB
ELEVATIONS 643.5 FT AND 677.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

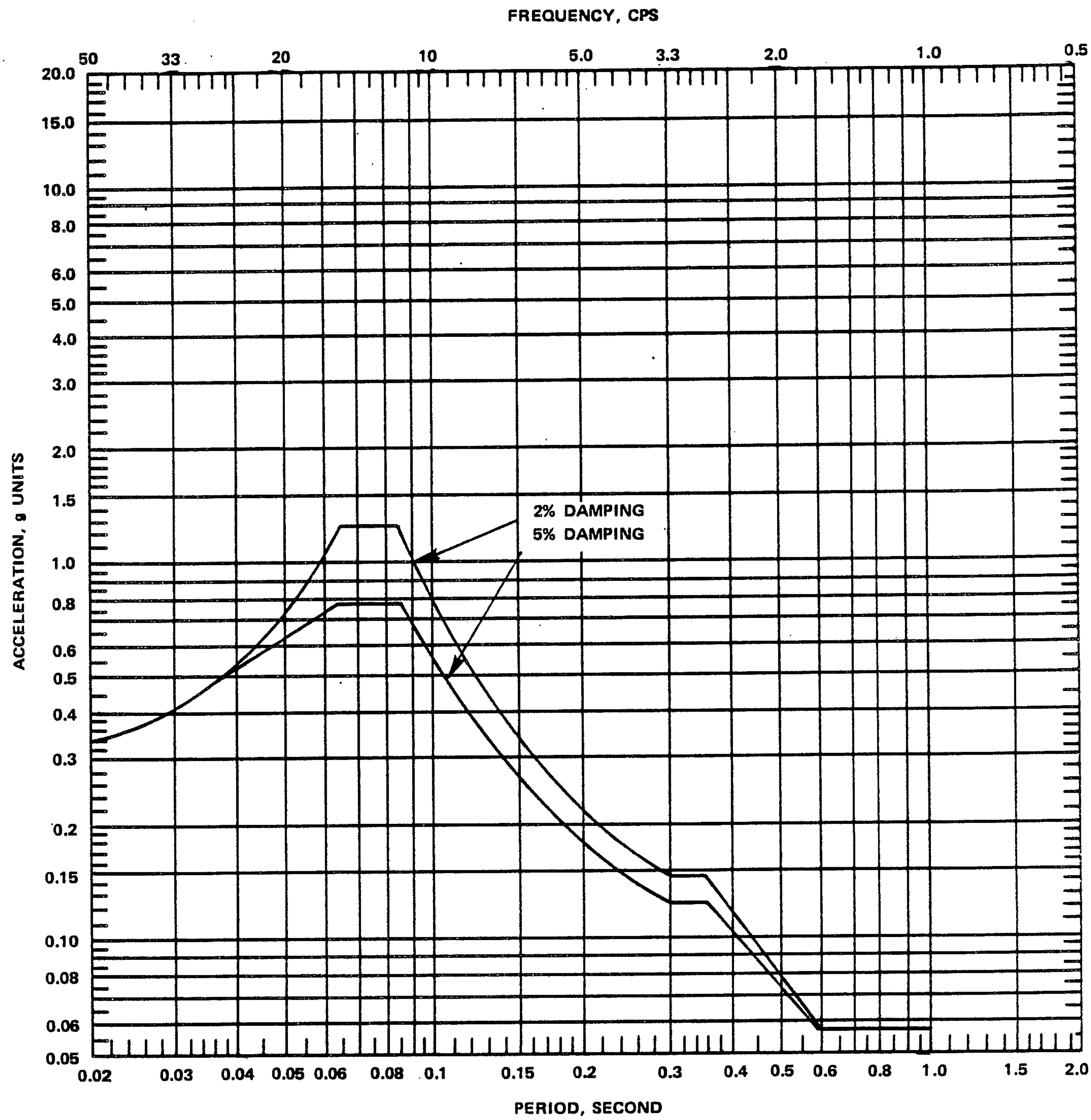

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FIGURE 3.7-88

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE
REACTOR/AUXILIARY BUILDING
CRANE RAIL ELEVATION

SARGENT & LUNDY REPORT NO. SL-2682

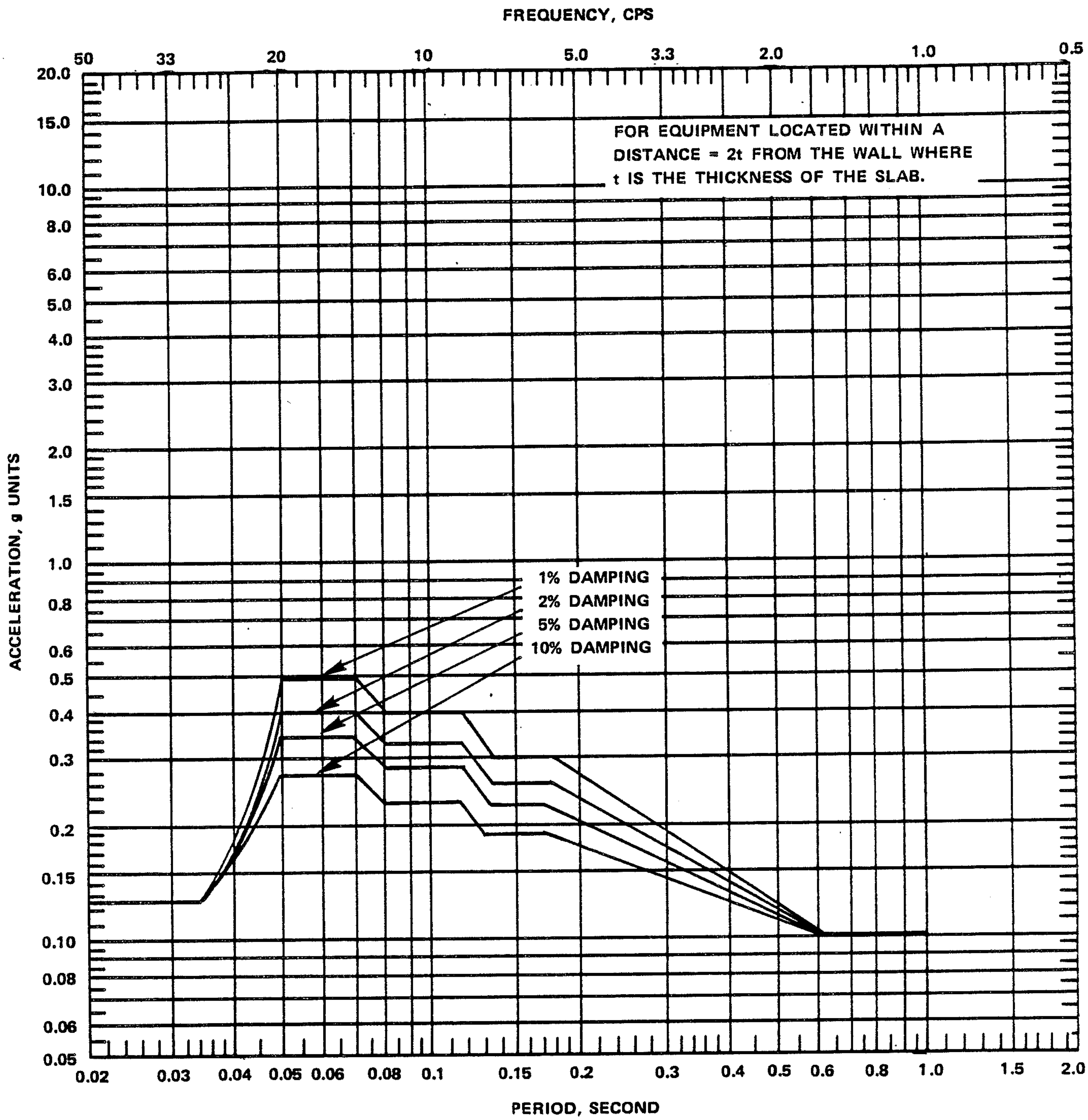

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FIGURE 3.7-89

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
REACTOR CONTAINMENT SHIELD
ELEVATIONS 583.5 FT AND 613.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

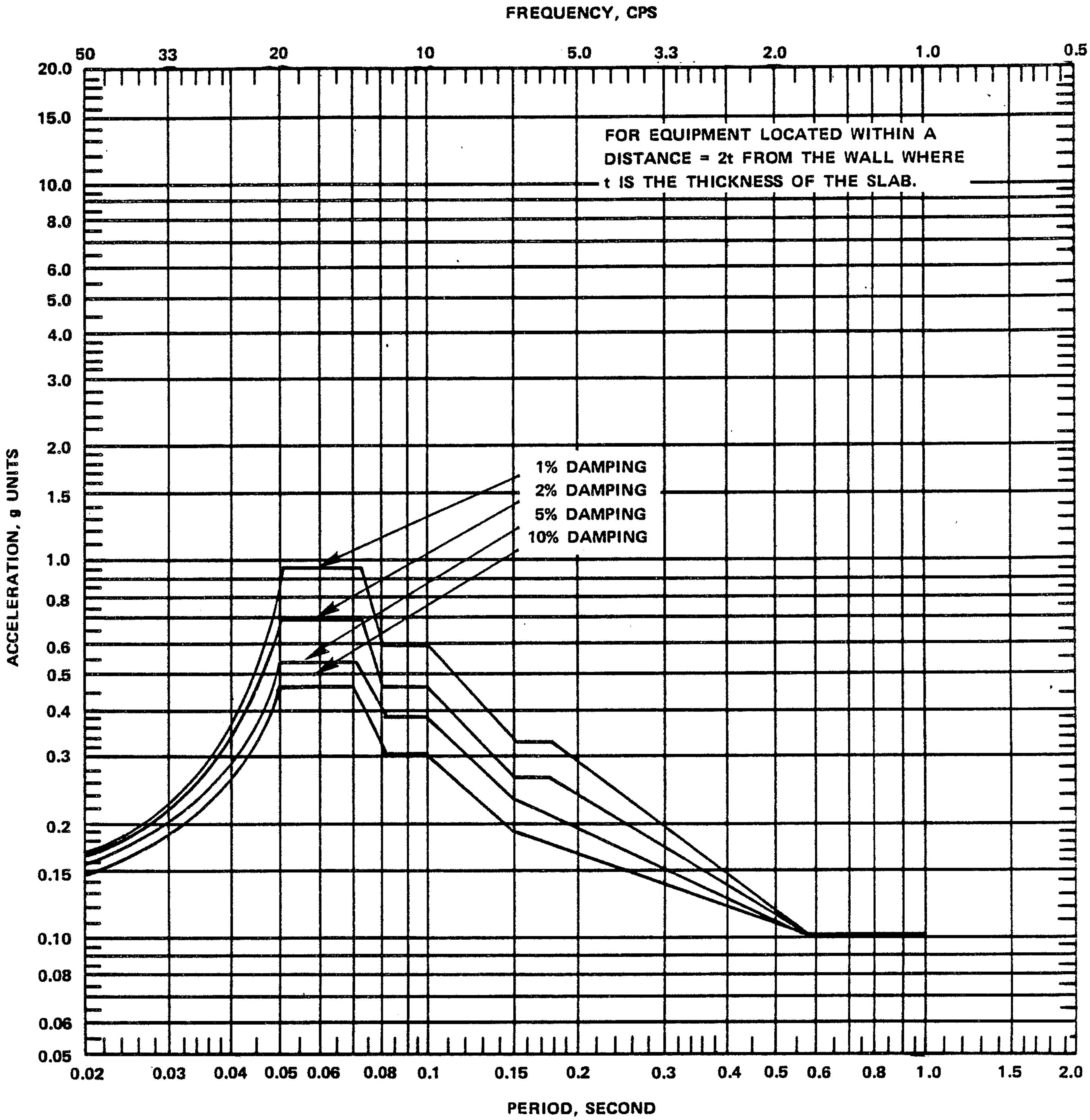

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FIGURE 3.7-90

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
REACTOR CONTAINMENT SHIELD
ELEVATIONS 643.5 FT, 659.5 FT, AND 684.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

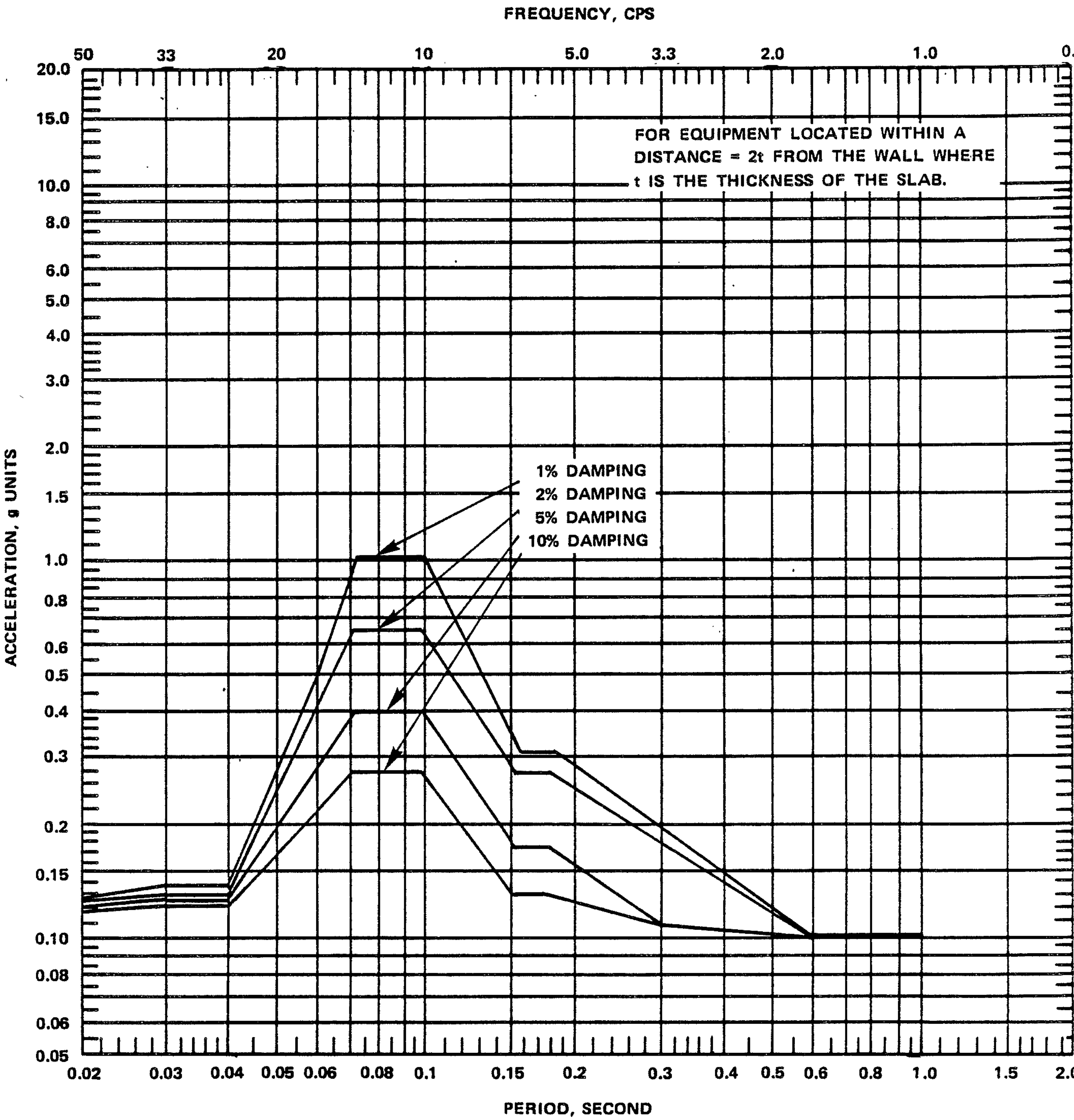

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FIGURE 3.7-91

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
REACTOR/AUXILIARY BUILDING WALL
ELEVATIONS 583.5 FT AND 613.5 FT

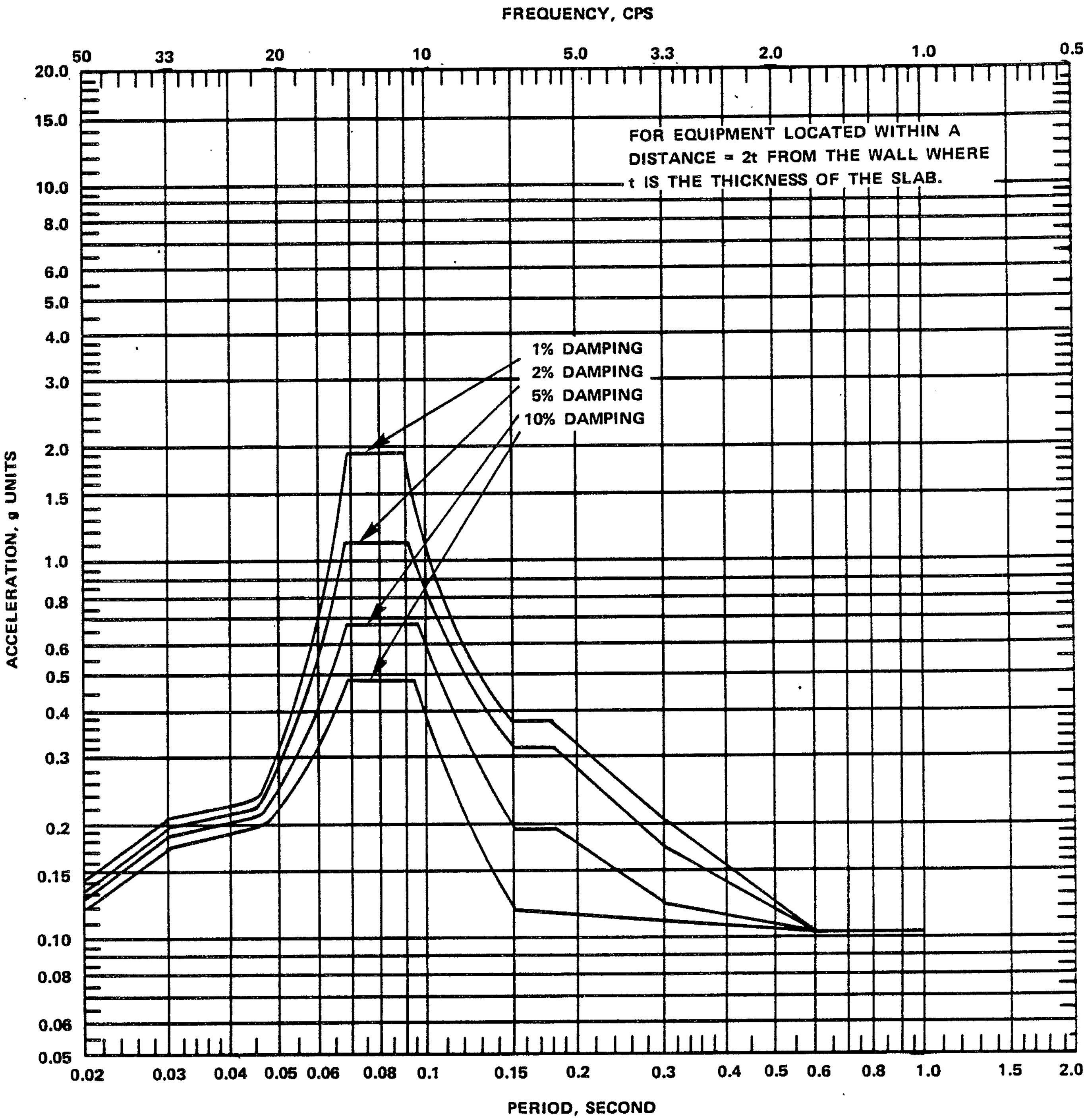

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FIGURE 3.7-92

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
REACTOR/AUXILIARY BUILDING WALL
ELEVATIONS 641.5 FT, 659.5 FT, AND 684.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

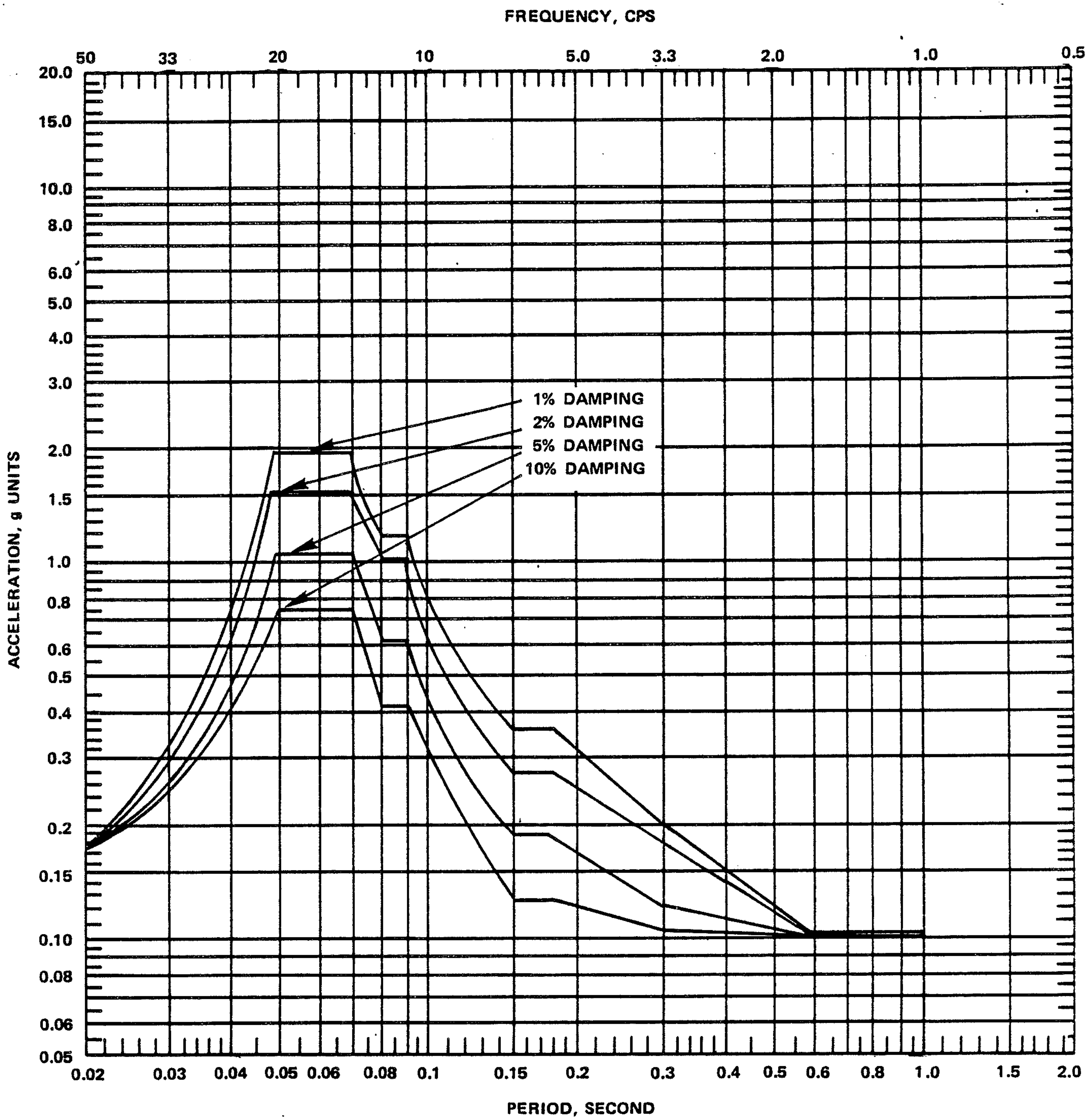

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FIGURE 3.7-93

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
REACTOR BUILDING SLAB
ELEVATIONS 583.5 FT AND 613.5 FT

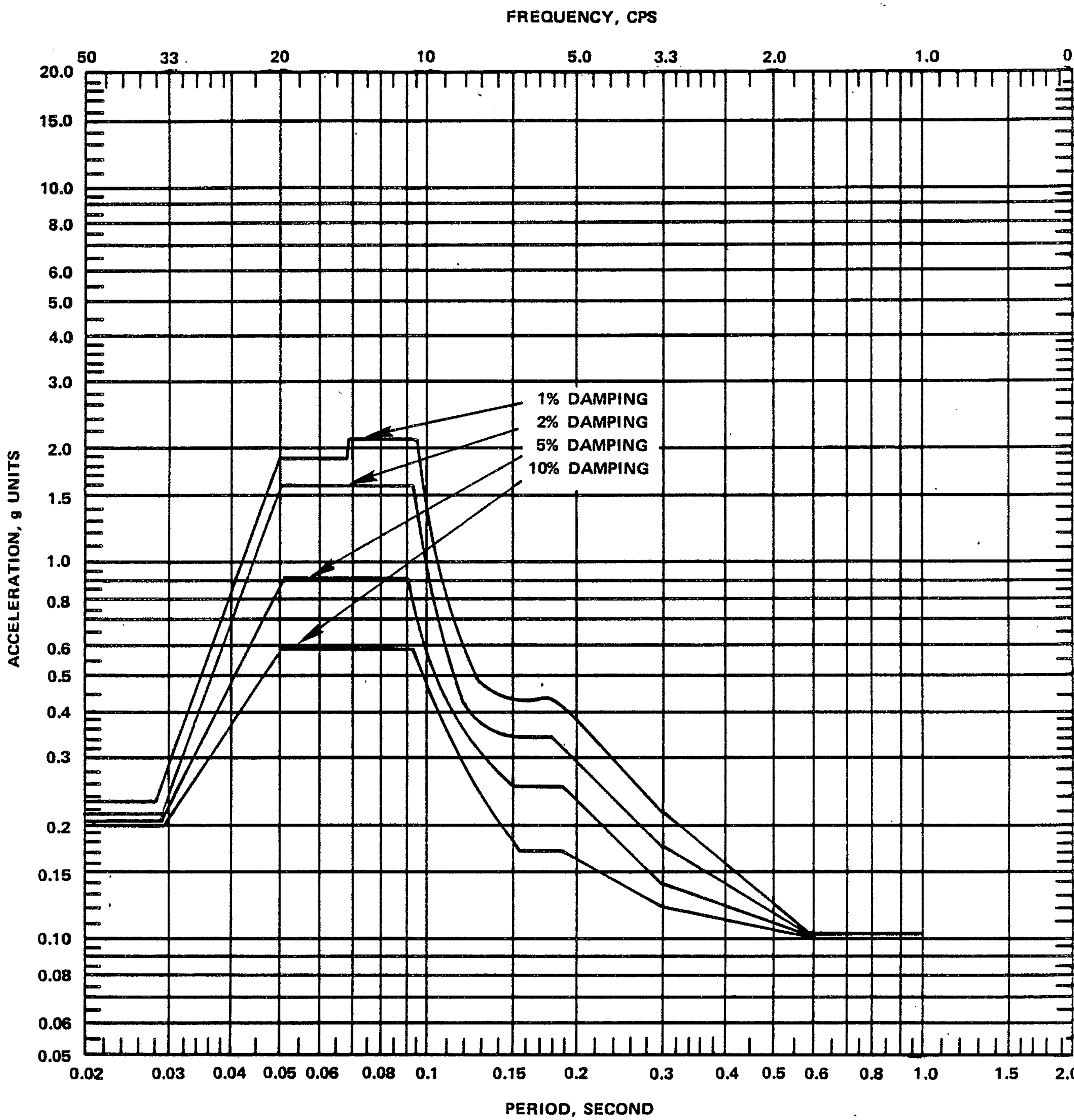

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FIGURE 3.7-94

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
REACTOR BUILDING SLAB
ELEVATIONS 641.5 FT, 659.5 FT, AND 684.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

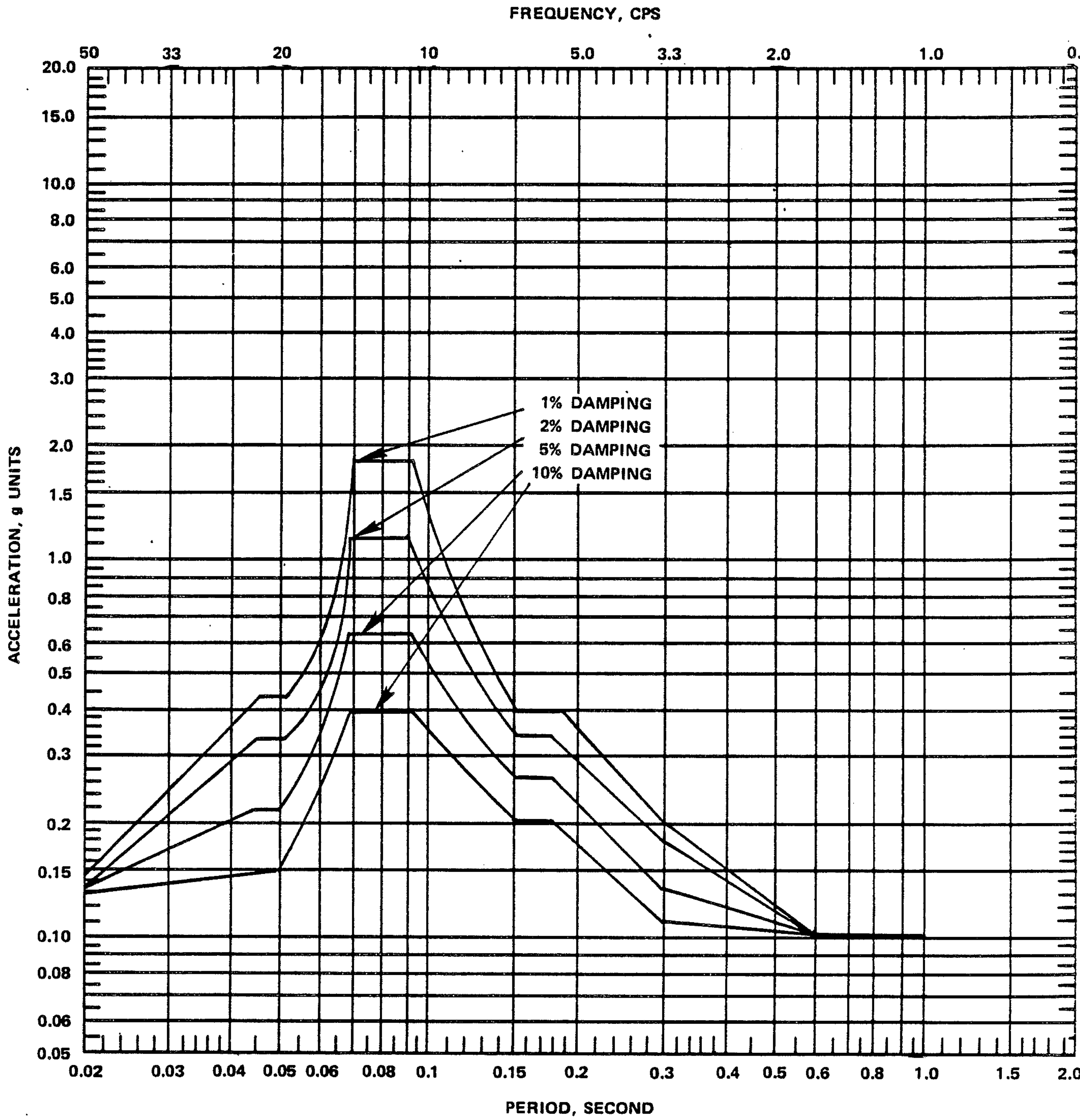

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FIGURE 3.7-95

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
AUXILIARY BUILDING SLAB
ELEVATIONS 583.5 FT, 613.5 FT, AND 659.5 FT

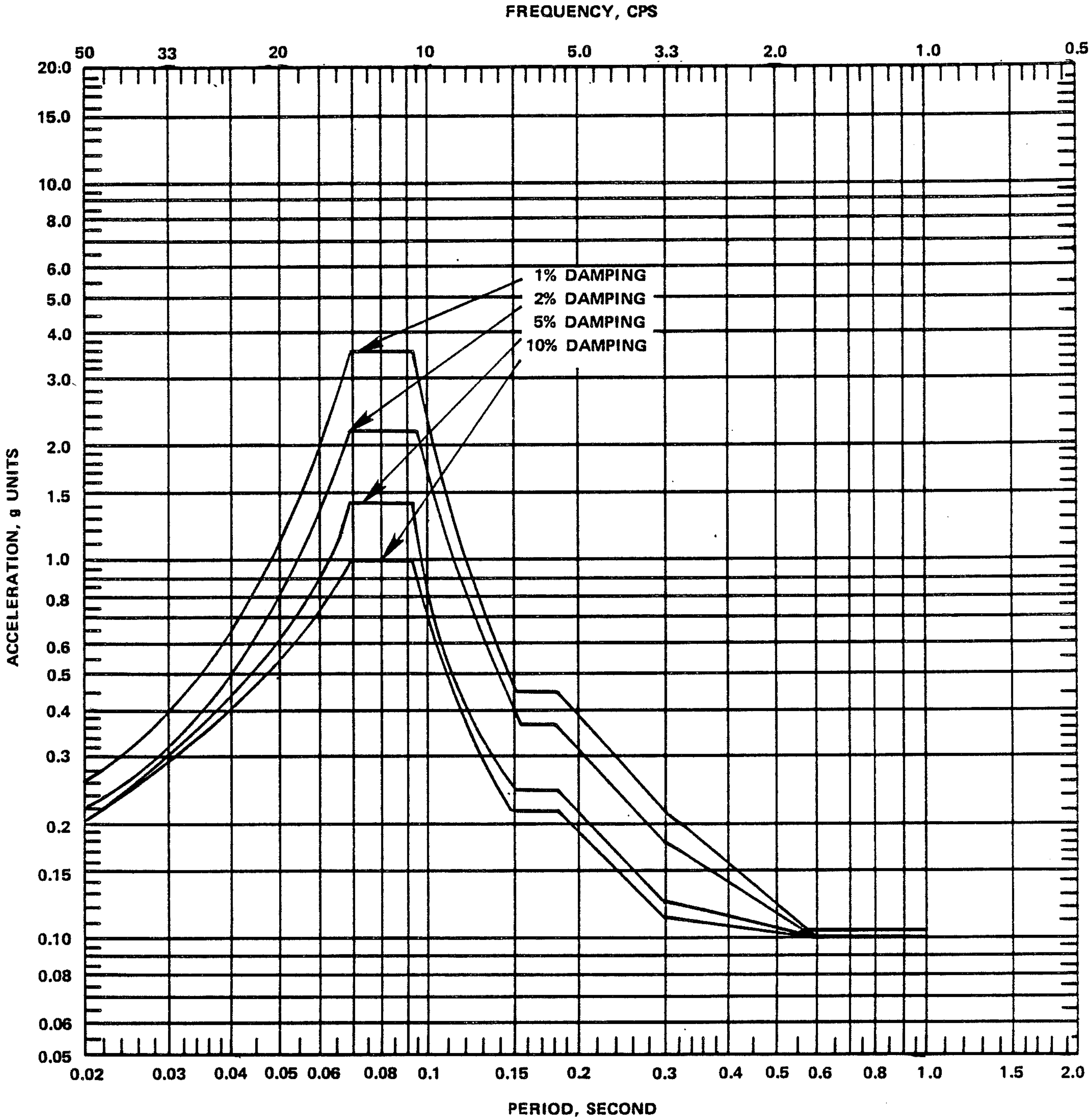

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FIGURE 3.7-96

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
AUXILIARY BUILDING SLAB
ELEVATIONS 643.5 FT AND 677.5 FT

SARGENT & LUNDY REPORT NO. SL-2682

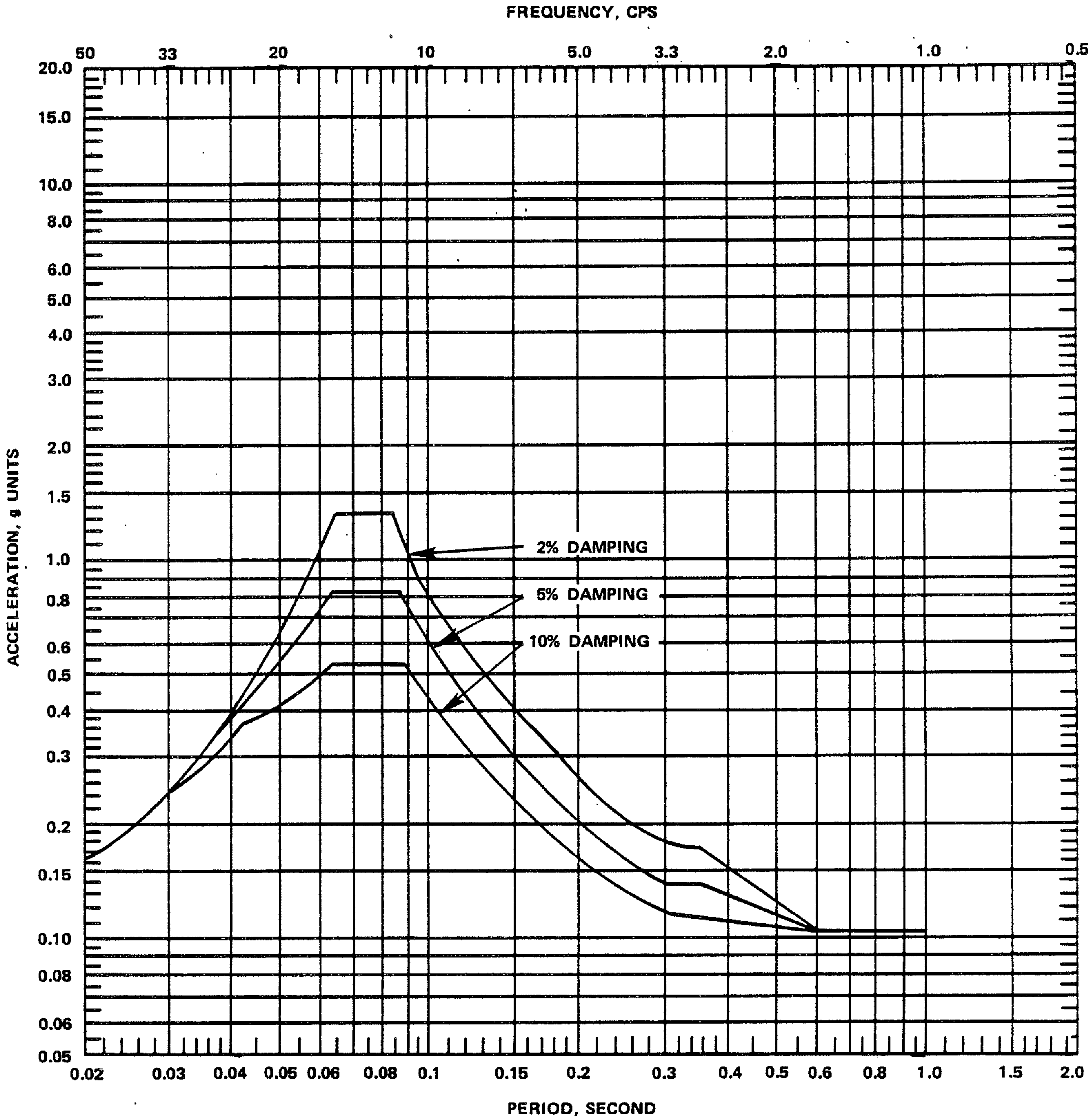

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FIGURE 3.7-97

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE
REACTOR/AUXILIARY BUILDING
CRANE RAIL ELEVATION

SARGENT & LUNDY REPORT NO. SL-2682

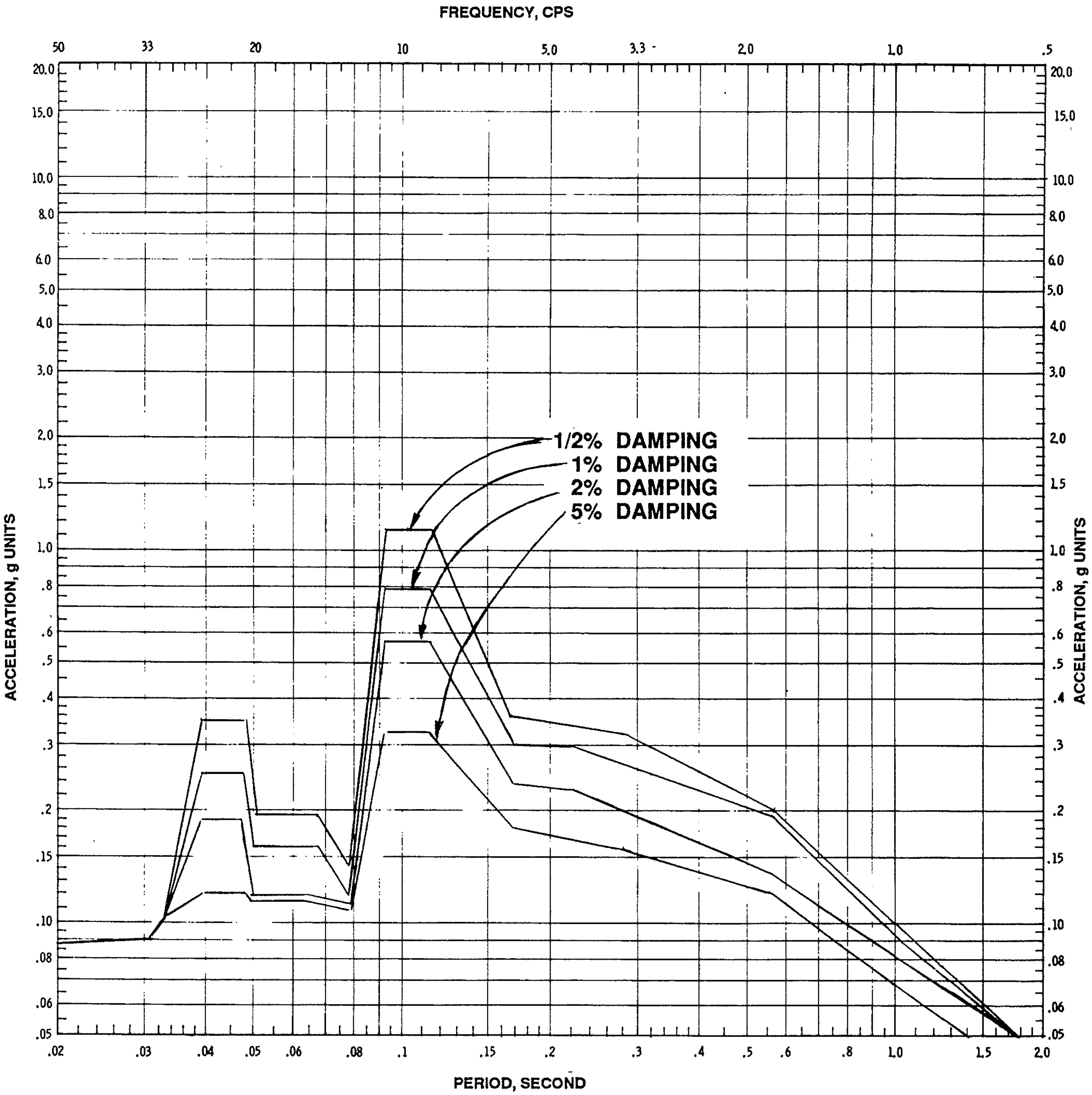

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FIGURE 3.7-98

HORIZONTAL FLOOR RESPONSE SPECTRA OPERATING-BASIS EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX – NORTH-SOUTH COMPONENT – SLAB 1 – ELEVATION 590.0 FT

SARGENT & LUNDY REPORT NO. SL-3147

REV 6 3/93

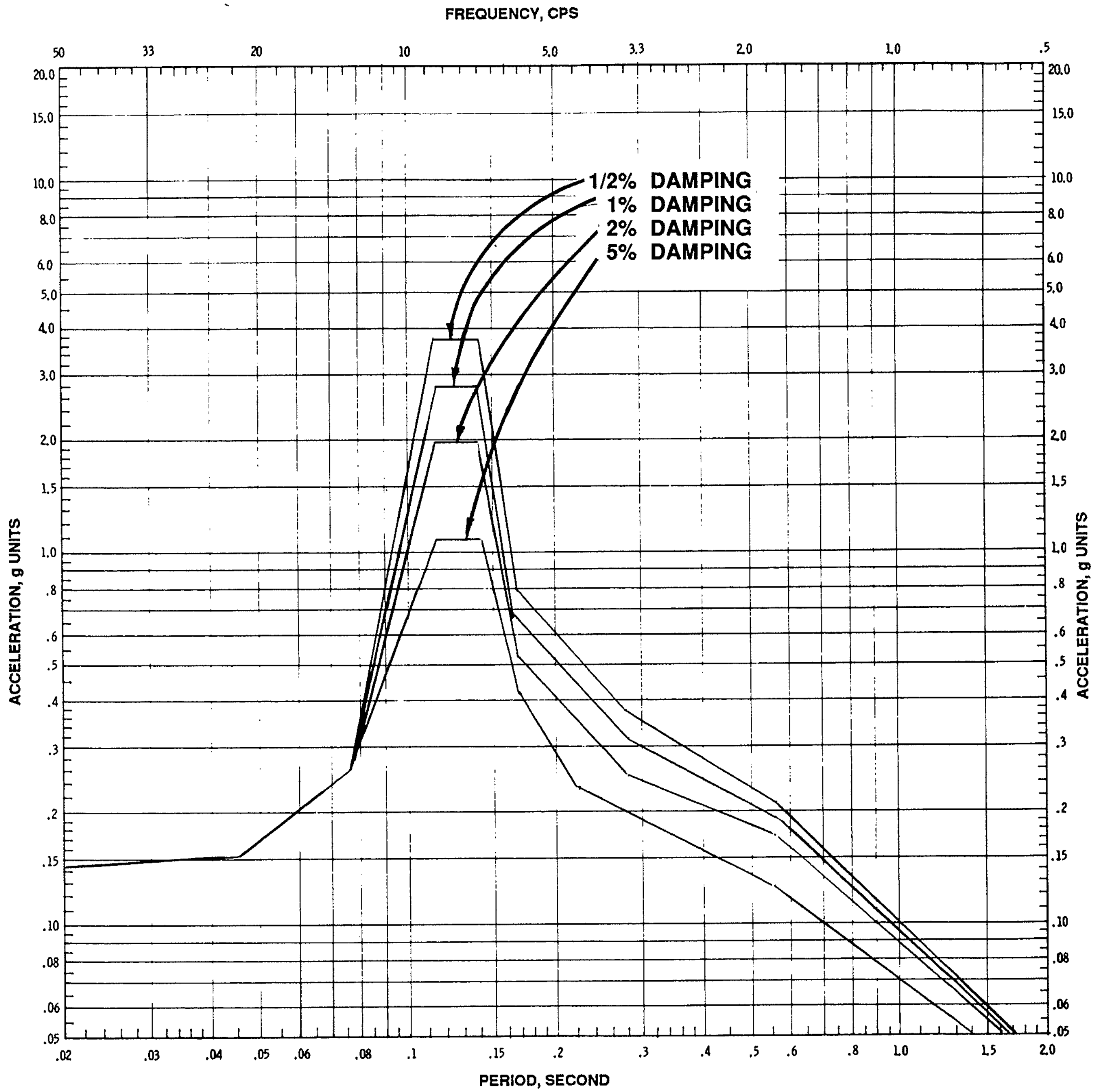

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FIGURE 3.7-99

HORIZONTAL FLOOR RESPONSE SPECTRA OPERATING-BASIS EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX – EAST-WEST COMPONENT – SLAB 1 – ELEVATION 590.0 FT

SARGENT & LUNDY REPORT NO. SL-3147

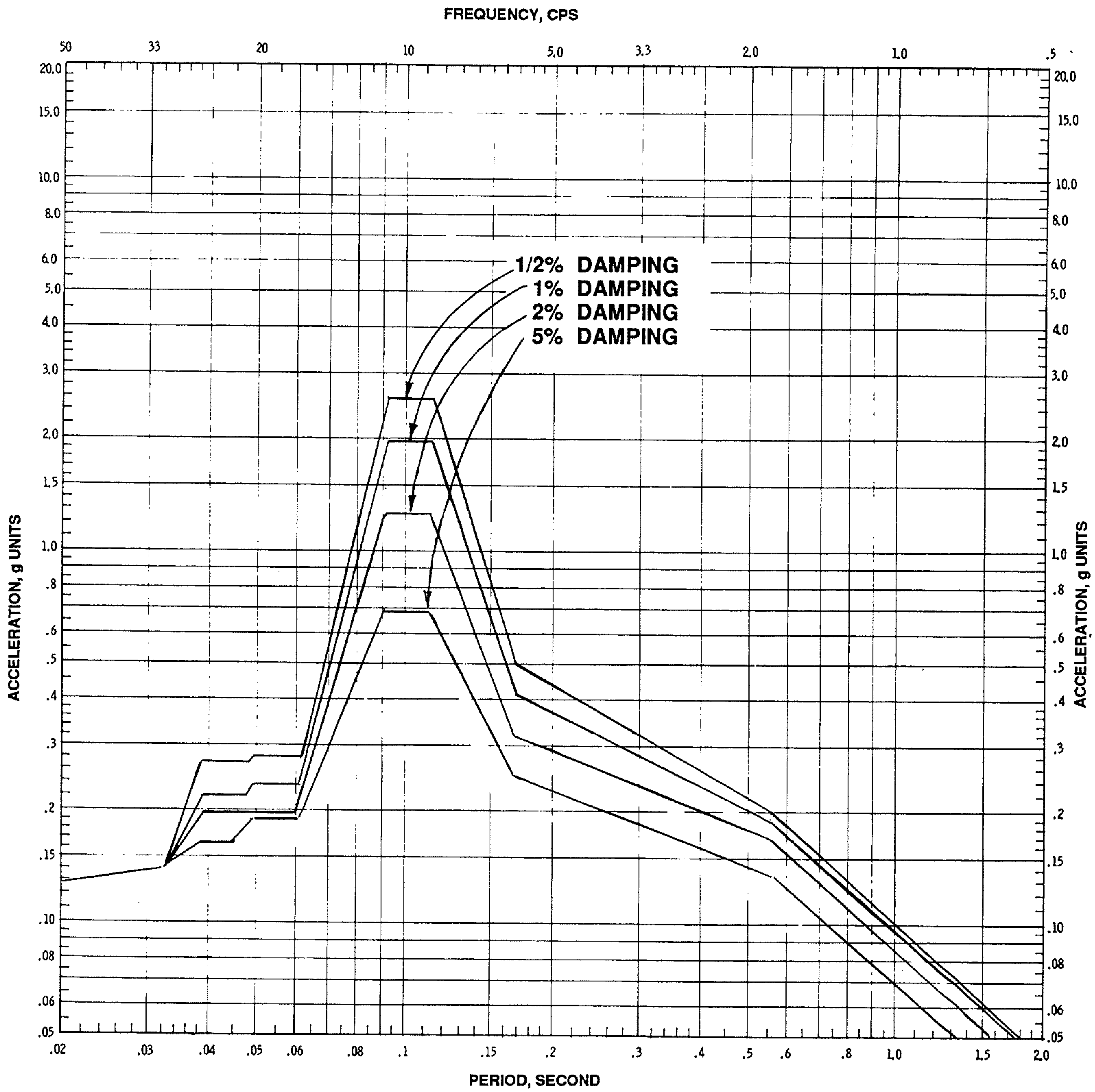

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FIGURE 3.7-100

HORIZONTAL FLOOR RESPONSE SPECTRA OPERATING-BASIS EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX – NORTH-SOUTH COMPONENT – SLAB 2 – ELEVATION 617.0 FT

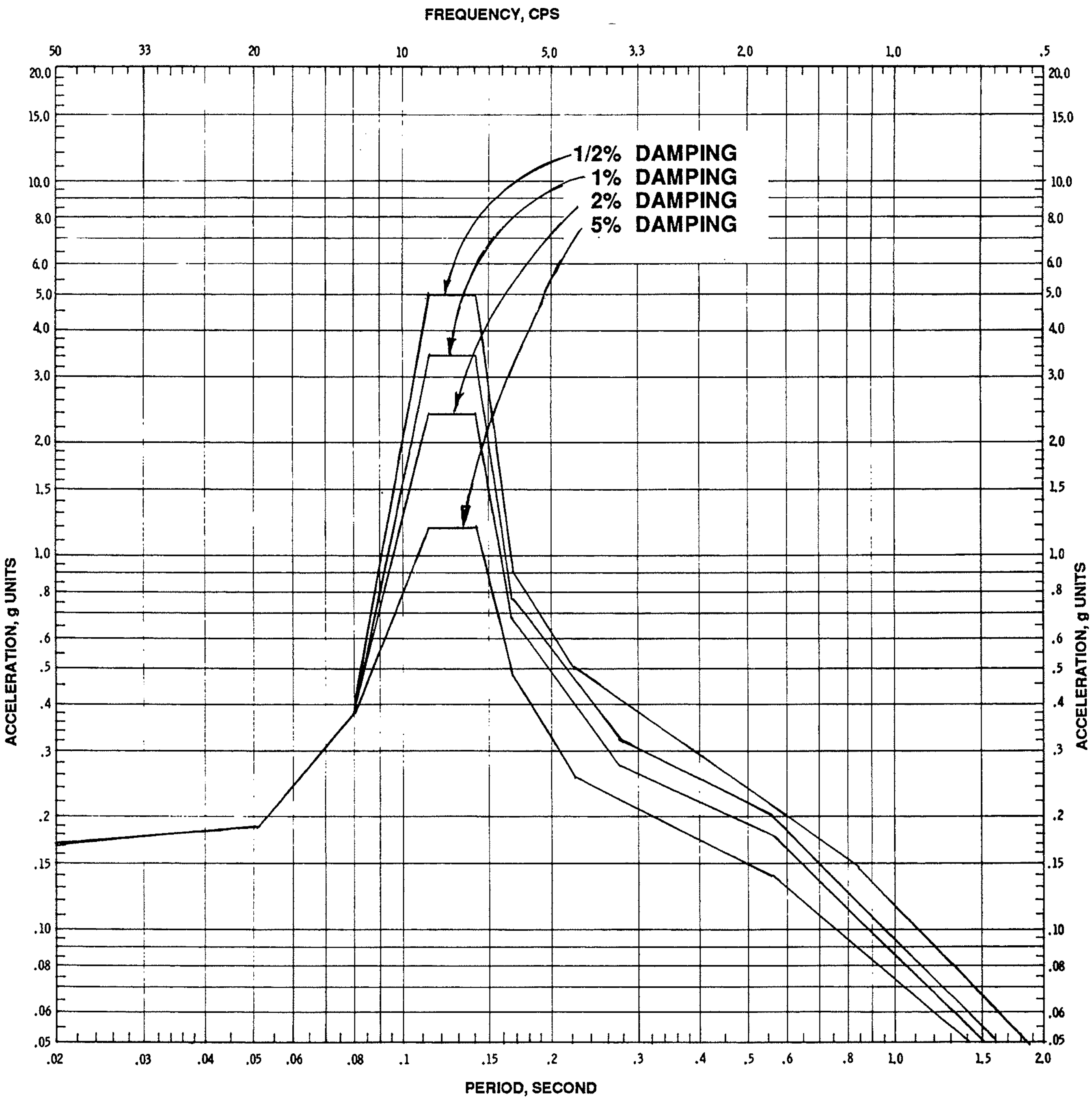

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FIGURE 3.7-101

HORIZONTAL FLOOR RESPONSE SPECTRA OPERATING-BASIS EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX – EAST-WEST COMPONENT – SLAB 2 – ELEVATION 617.0 FT

SARGENT & LUNDY REPORT NO. SL-3147

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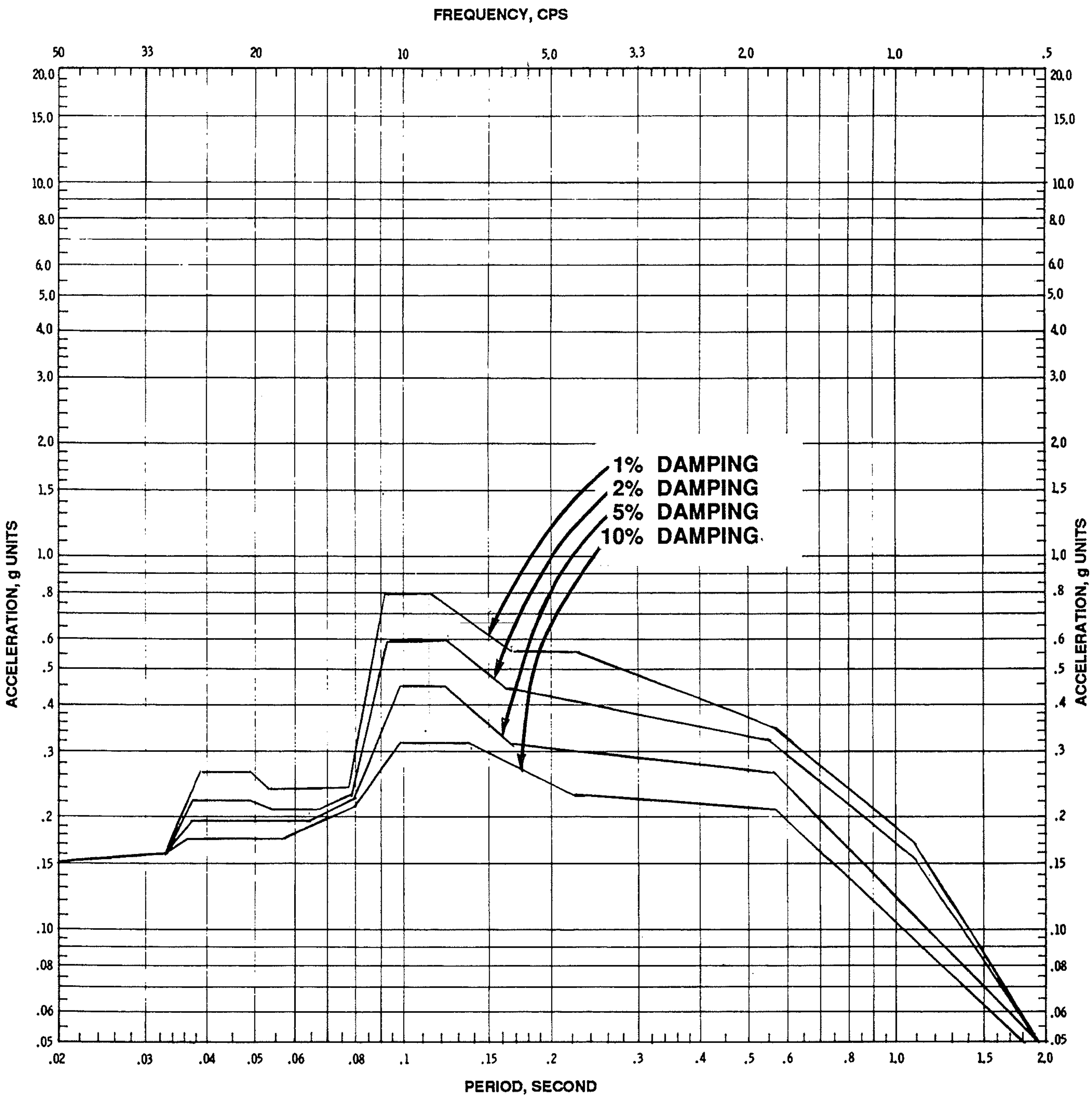

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FIGURE 3.7-102

HORIZONTAL FLOOR RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – RESIDUAL
HEAT REMOVAL COMPLEX – NORTH-SOUTH
COMPONENT – SLAB 1 – ELEVATION 590.0 FT

SARGENT & LUNDY REPORT NO. SL-3147

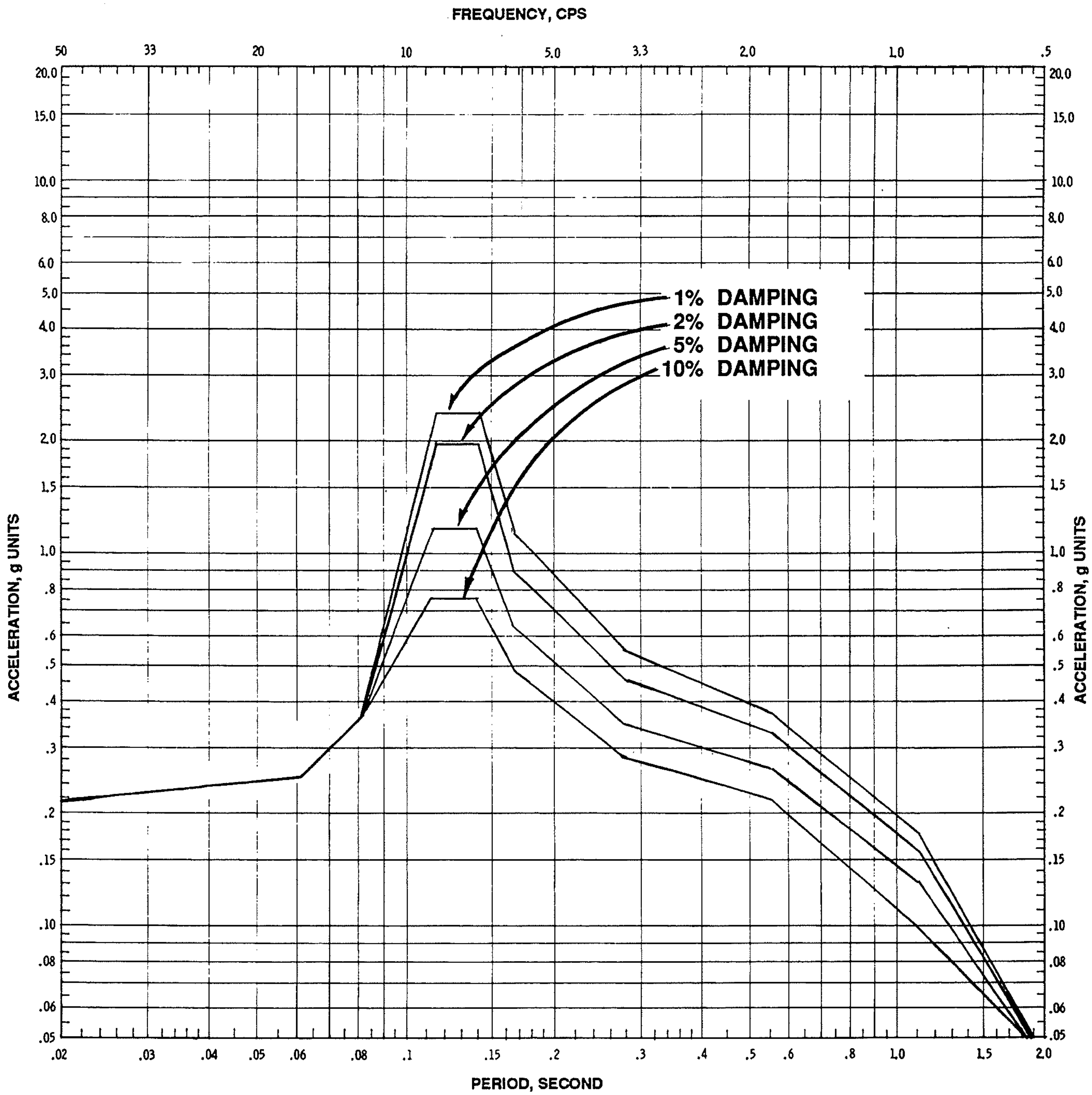

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FIGURE 3.7-103

HORIZONTAL FLOOR RESPONSE SPECTRA SAFE-SHUTDOWN EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX – EAST-WEST COMPONENT – SLAB 1 – ELEVATION 590.0 FT

SARGENT & LUNDY REPORT NO. SL-3147

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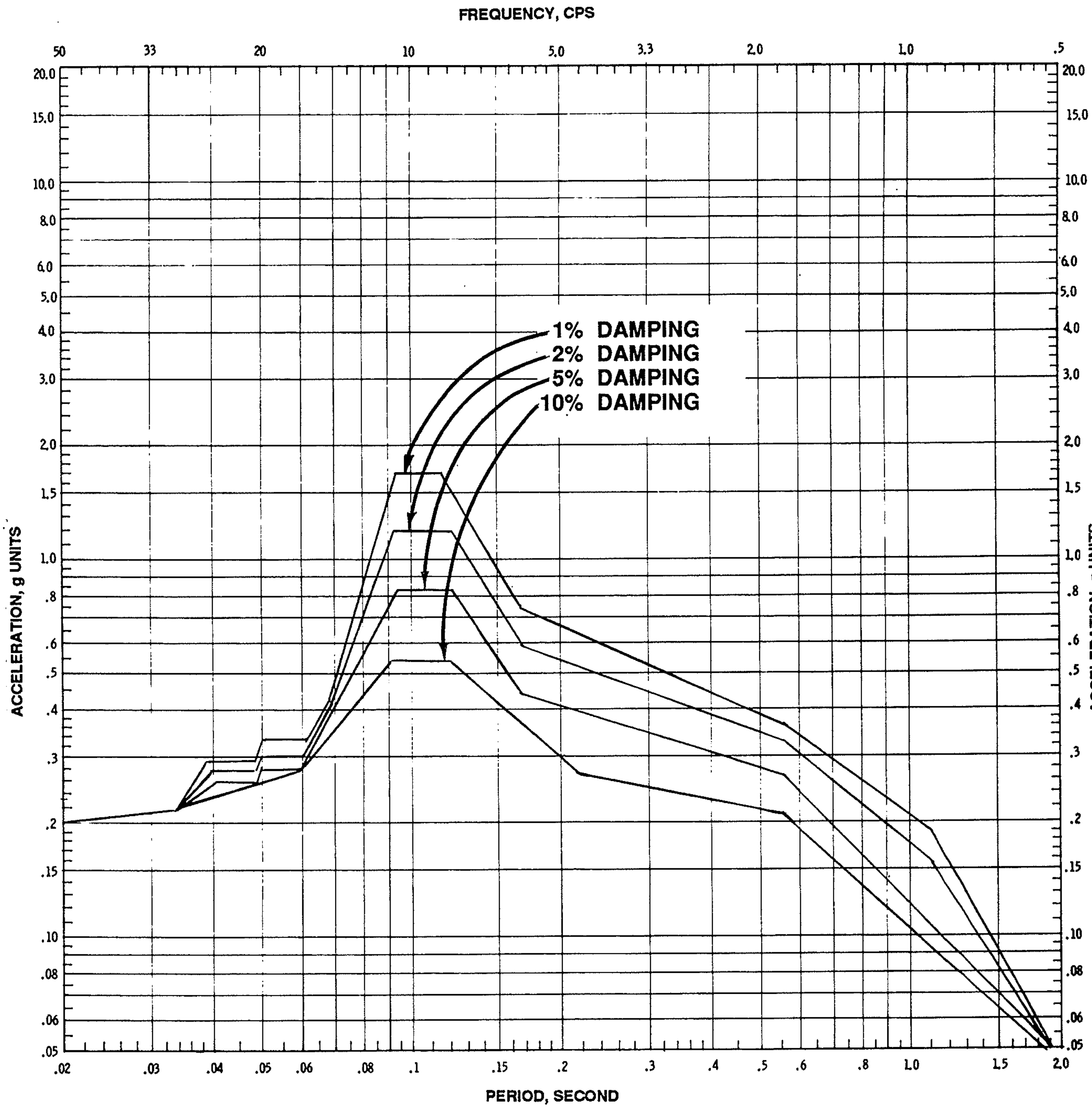

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FIGURE 3.7-104

HORIZONTAL FLOOR RESPONSE SPECTRA SAFE-SHUTDOWN EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX – NORTH-SOUTH COMPONENT – SLAB 2 – ELEVATION 617.0 FT

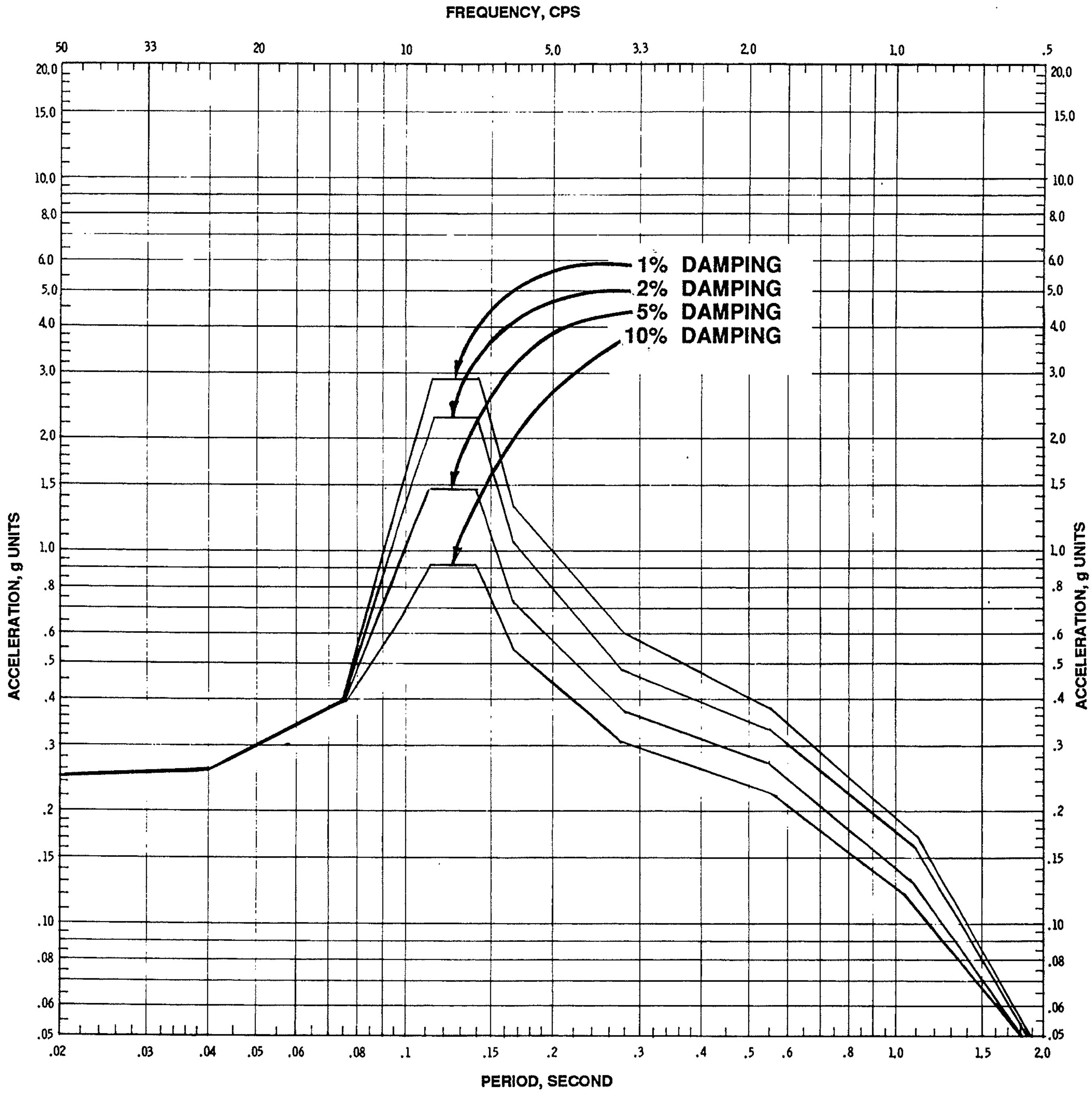

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FIGURE 3.7-105

HORIZONTAL FLOOR RESPONSE SPECTRA SAFE-SHUTDOWN EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX – EAST-WEST COMPONENT – SLAB 2 – ELEVATION 617.0 FT

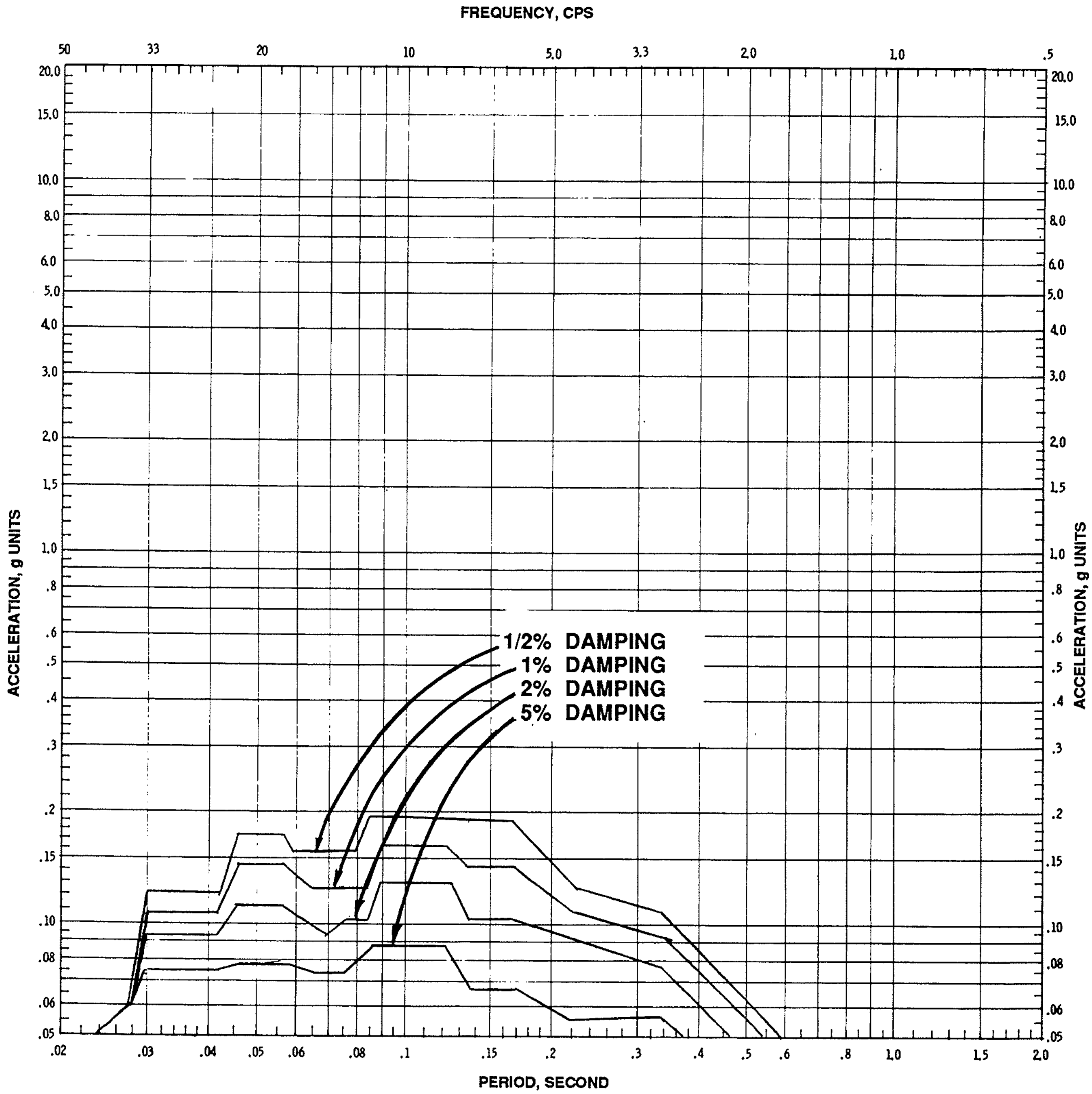

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FIGURE 3.7-106

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX WALLS AND SLAB 1A
ELEVATION 590.0 FT

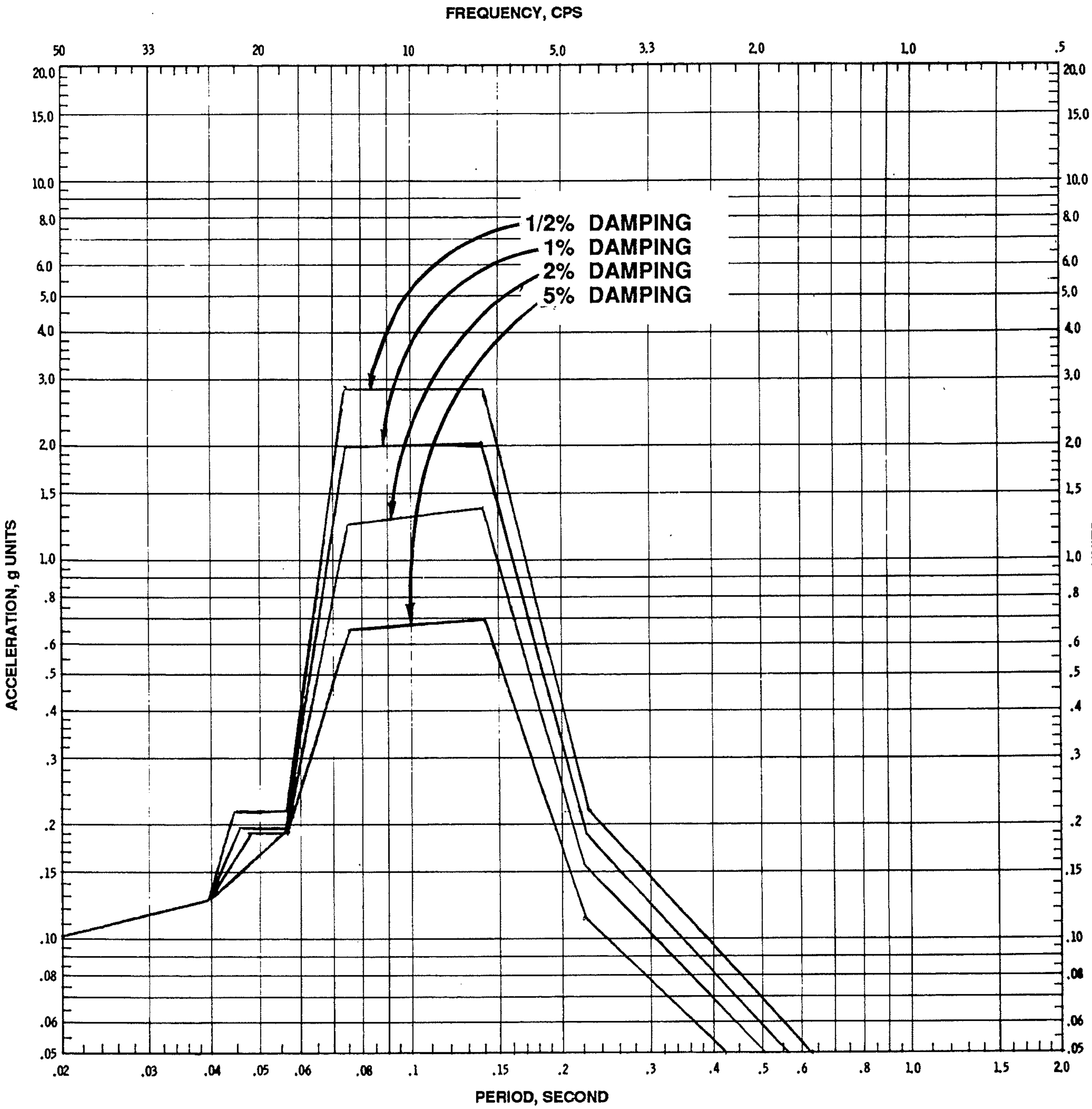

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FIGURE 3.7-107

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE – RESIDUAL
HEAT REMOVAL COMPLEX – SLAB 1B
ELEVATION 590.0 FT

SARGENT & LUNDY REPORT NO. SL-3147

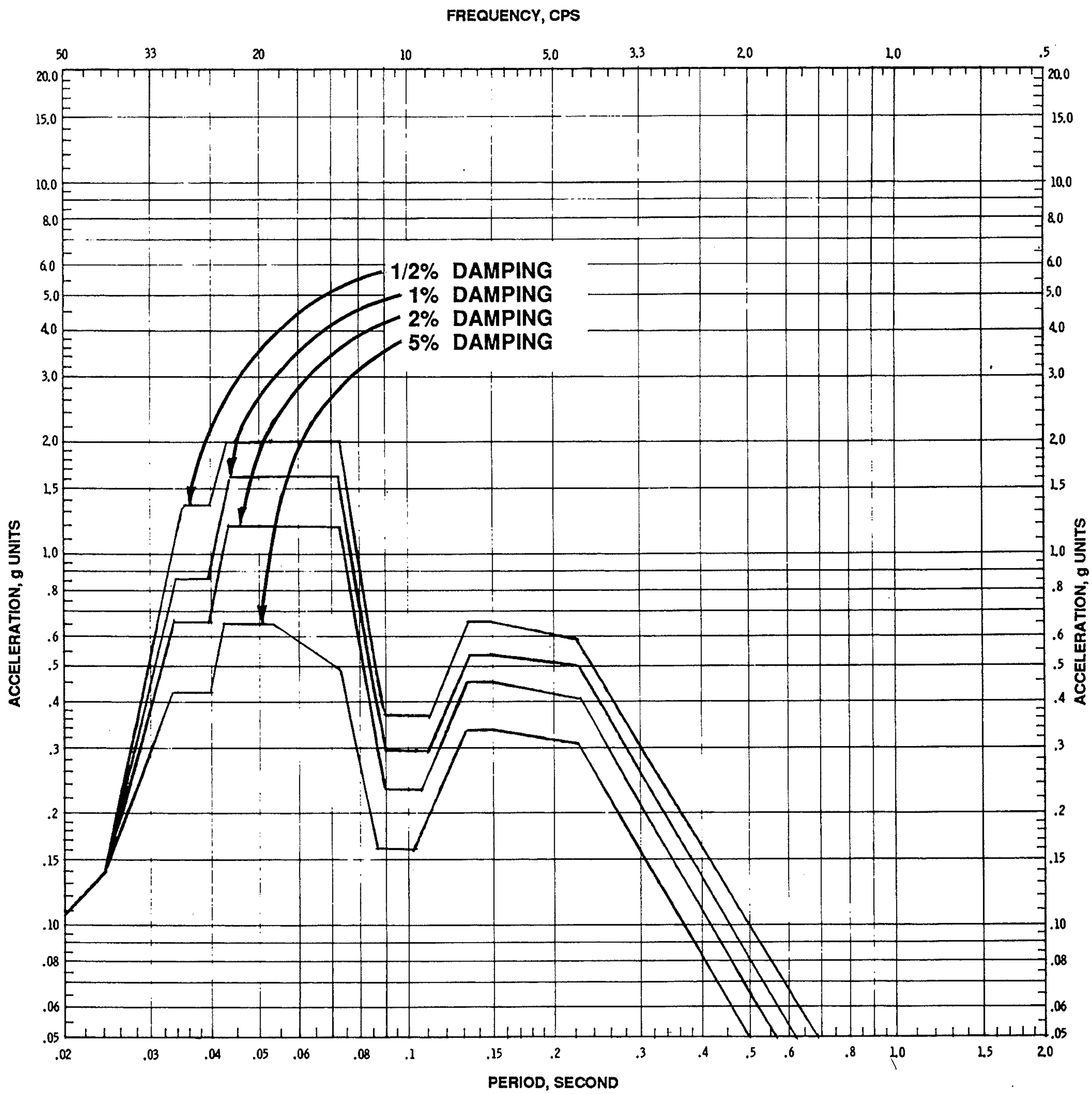

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FIGURE 3.7-108

VERTICAL RESPONSE SPECTRA OPERATING-BASIS EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX – SLAB 1C ELEVATION 590.0 FT

SARGENT & LUNDY REPORT NO. SL-3147

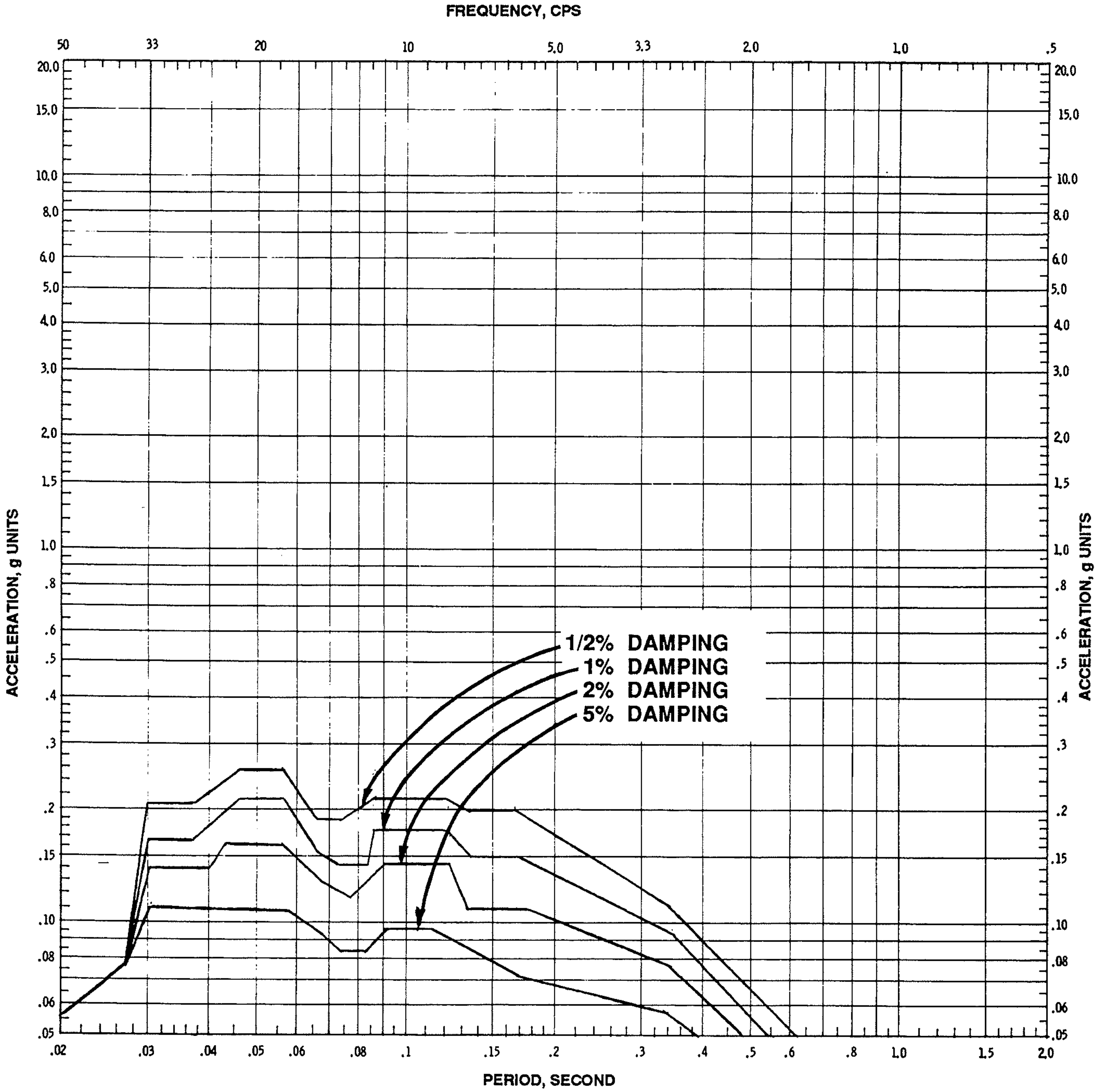

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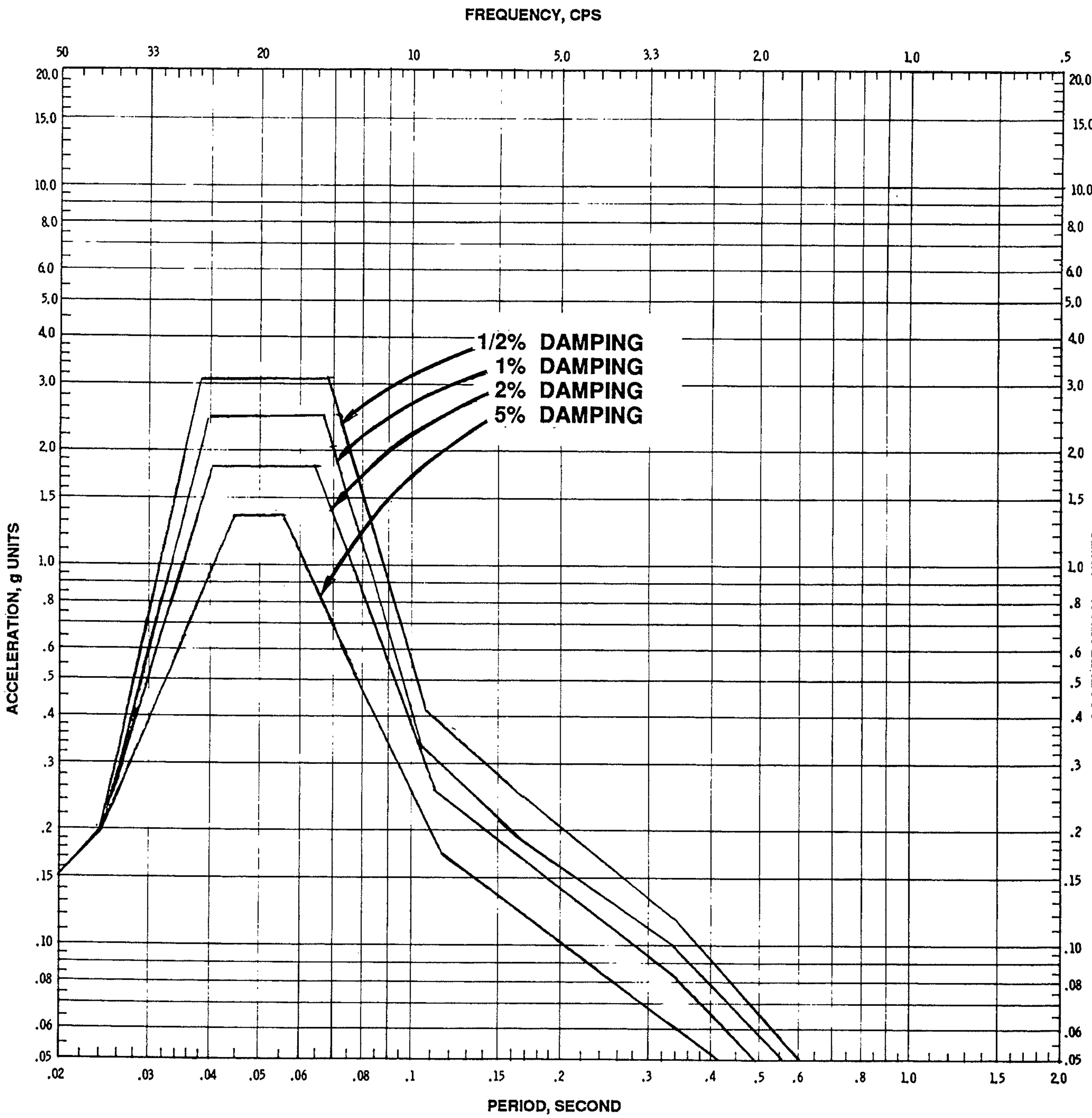
FIGURE 3.7-109

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE – RESIDUAL
HEAT REMOVAL COMPLEX – WALLS
ELEVATION 617.0 FT

SARGENT & LUNDY REPORT NO. SL-3147



SARGENT & LUNDY REPORT NO. SL-3147

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FIGURE 3.7-110

VERTICAL RESPONSE SPECTRA
OPERATING-BASIS EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX – SLAB 2
ELEVATION 617.0 FT

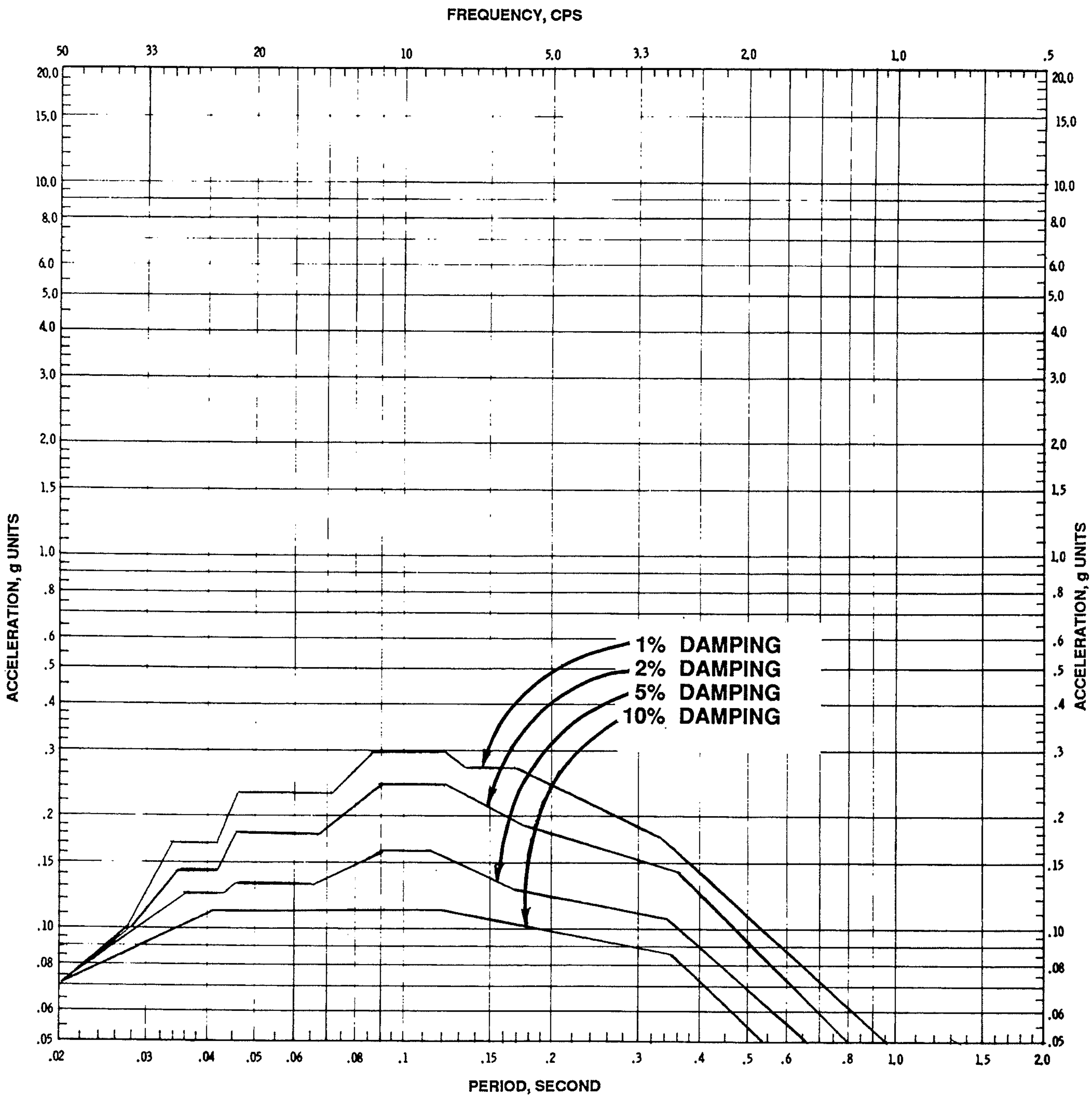

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FIGURE 3.7-111

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – RESIDUAL
HEAT REMOVAL COMPLEX – WALLS AND SLAB 1A
ELEVATION 590.0 FT

SARGENT & LUNDY REPORT NO. SL-3147

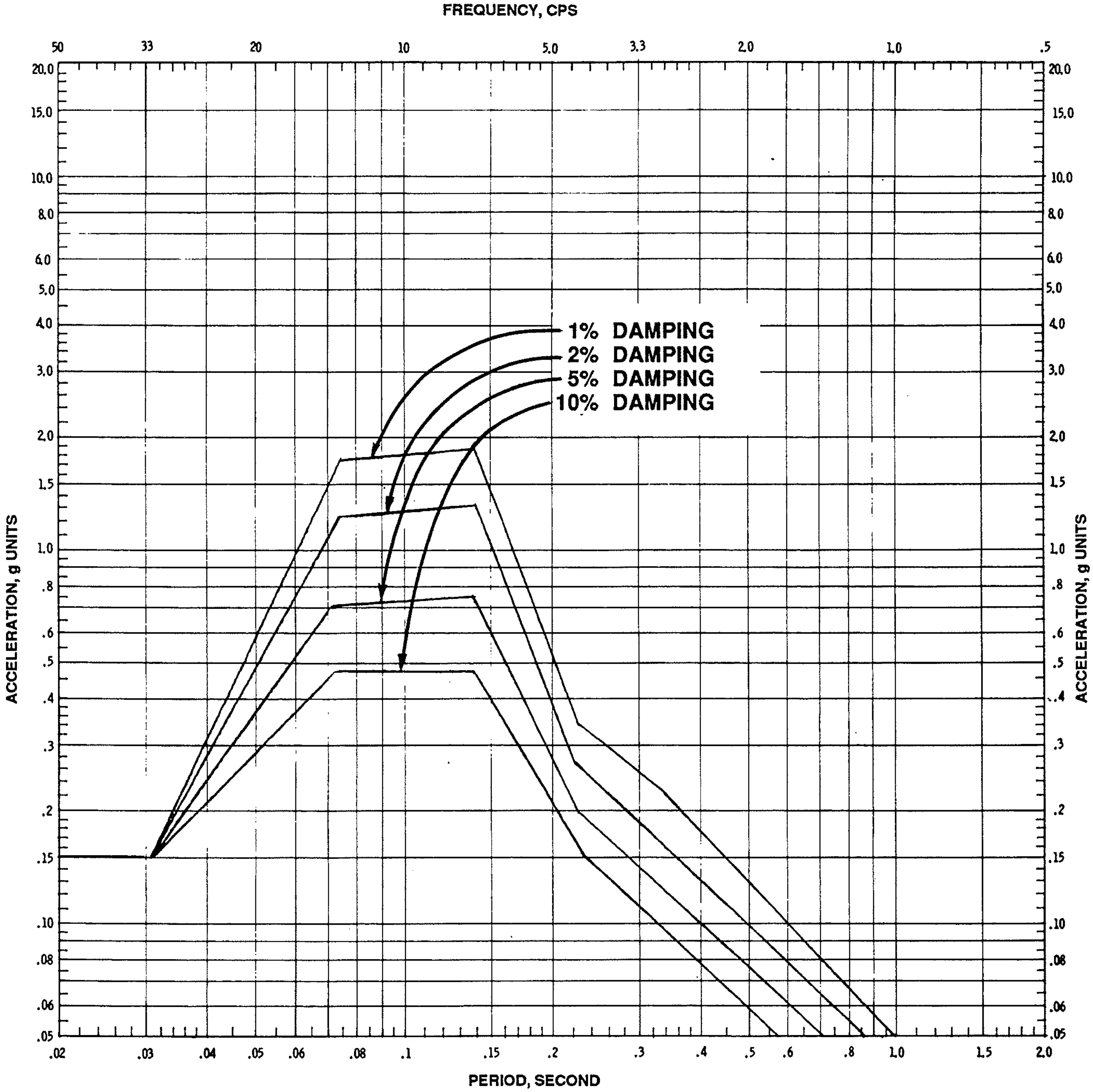

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FIGURE 3.7-112

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX – SLAB 1B
ELEVATION 590.0 FT

SARGENT & LUNDY REPORT NO. SL-3147

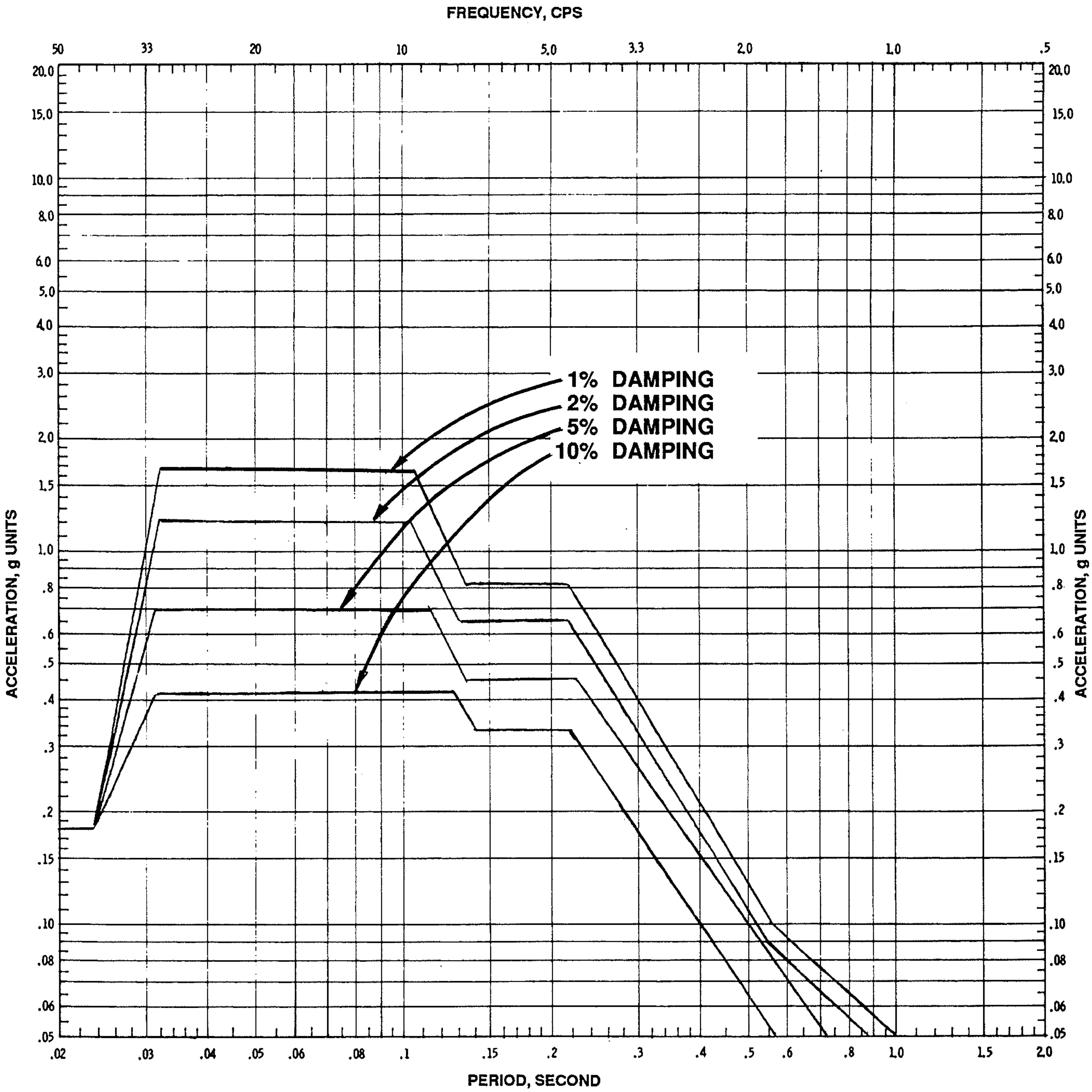

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FIGURE 3.7-113

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – RESIDUAL
HEAT REMOVAL COMPLEX – SLAB 1C
ELEVATION 590.0 FT

SARGENT & LUNDY REPORT NO. SL-3147

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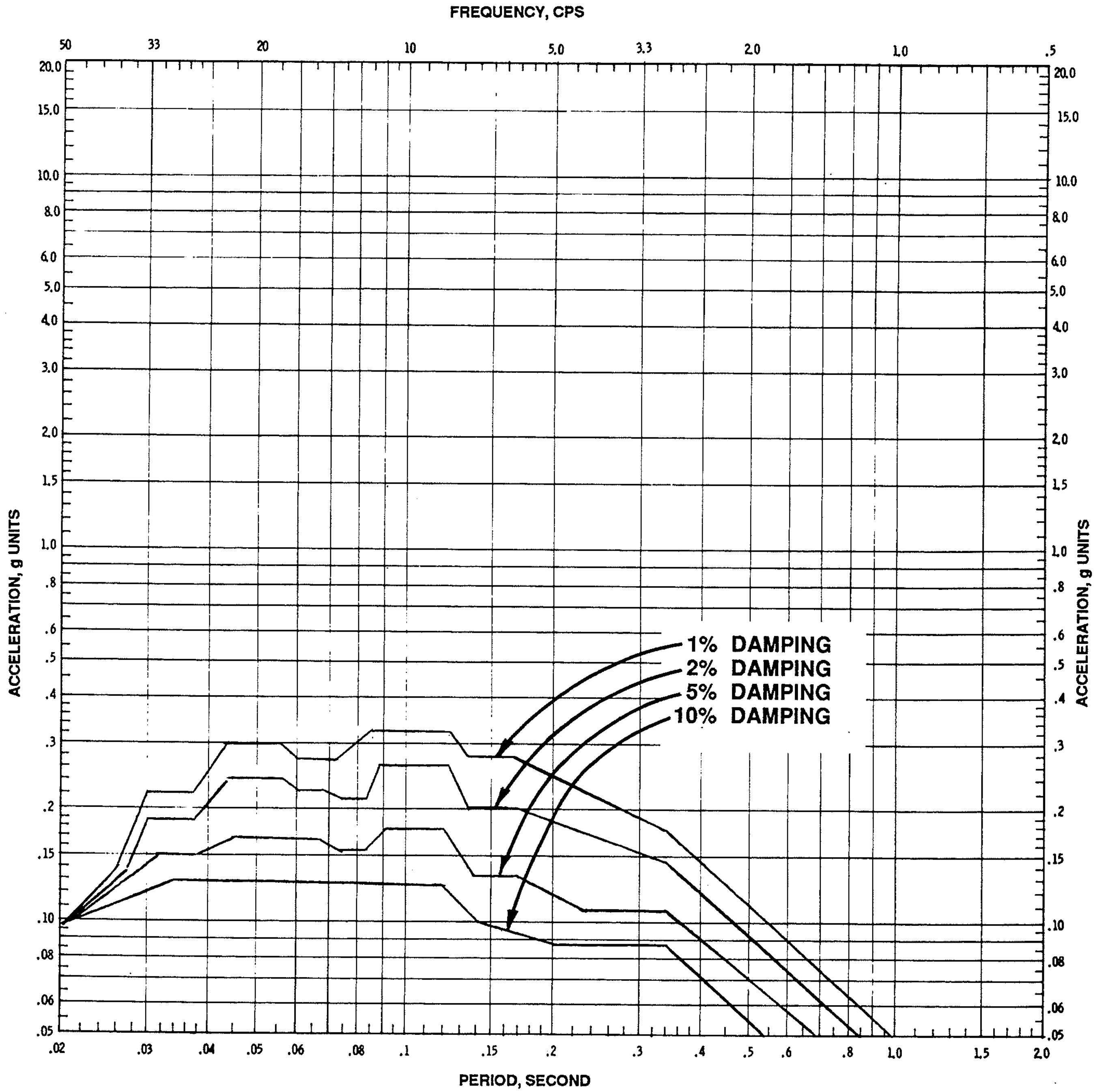

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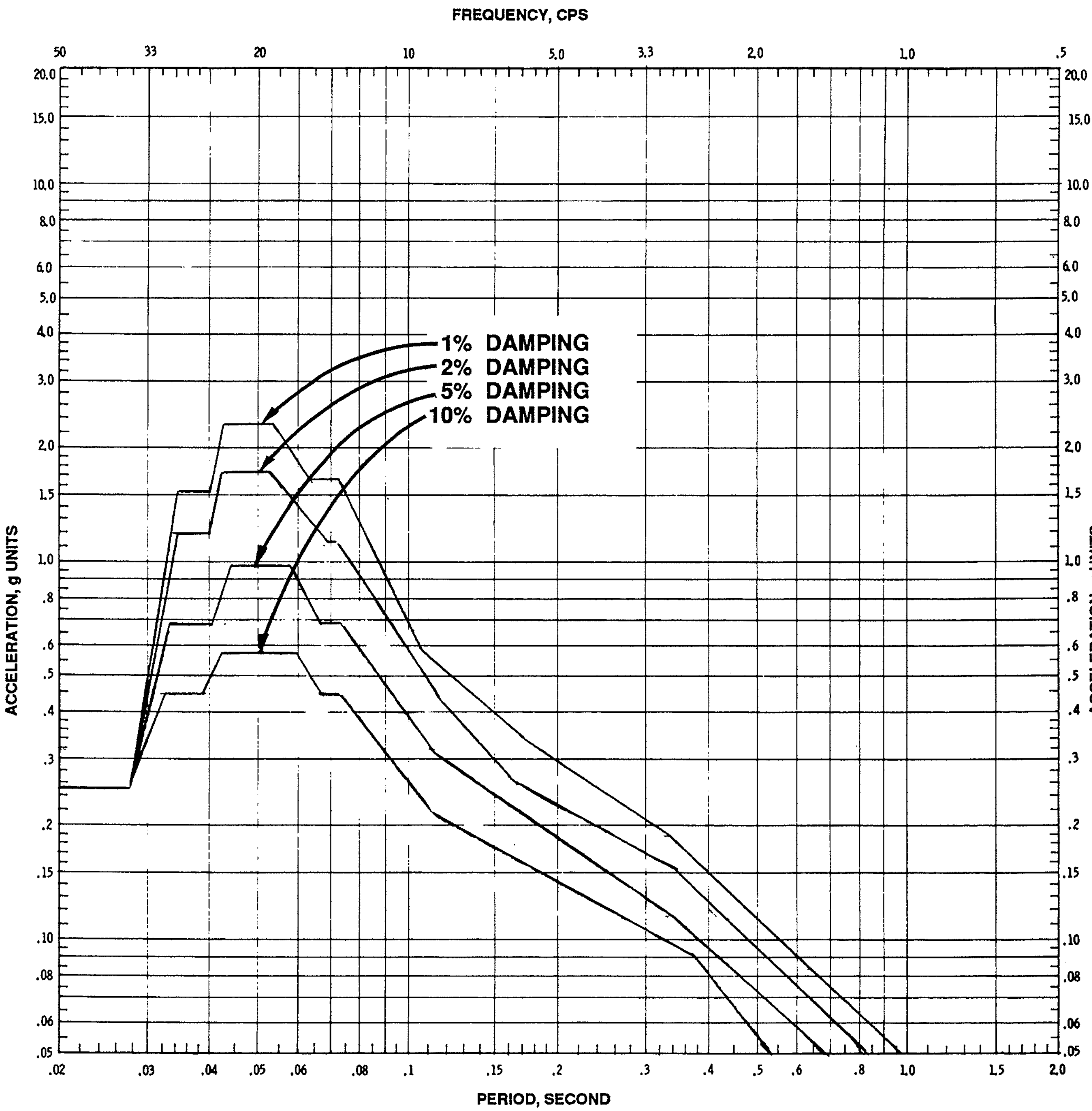
FIGURE 3.7-114

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – RESIDUAL HEAT REMOVAL COMPLEX – WALLS
ELEVATION 617.0 FT

SARGENT & LUNDY REPORT NO. SL-3147



SARGENT & LUNDY REPORT NO. SL-3147

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FIGURE 3.7-115

VERTICAL RESPONSE SPECTRA
SAFE-SHUTDOWN EARTHQUAKE – RESIDUAL
HEAT REMOVAL COMPLEX – SLAB 2
ELEVATION 617.0 FT

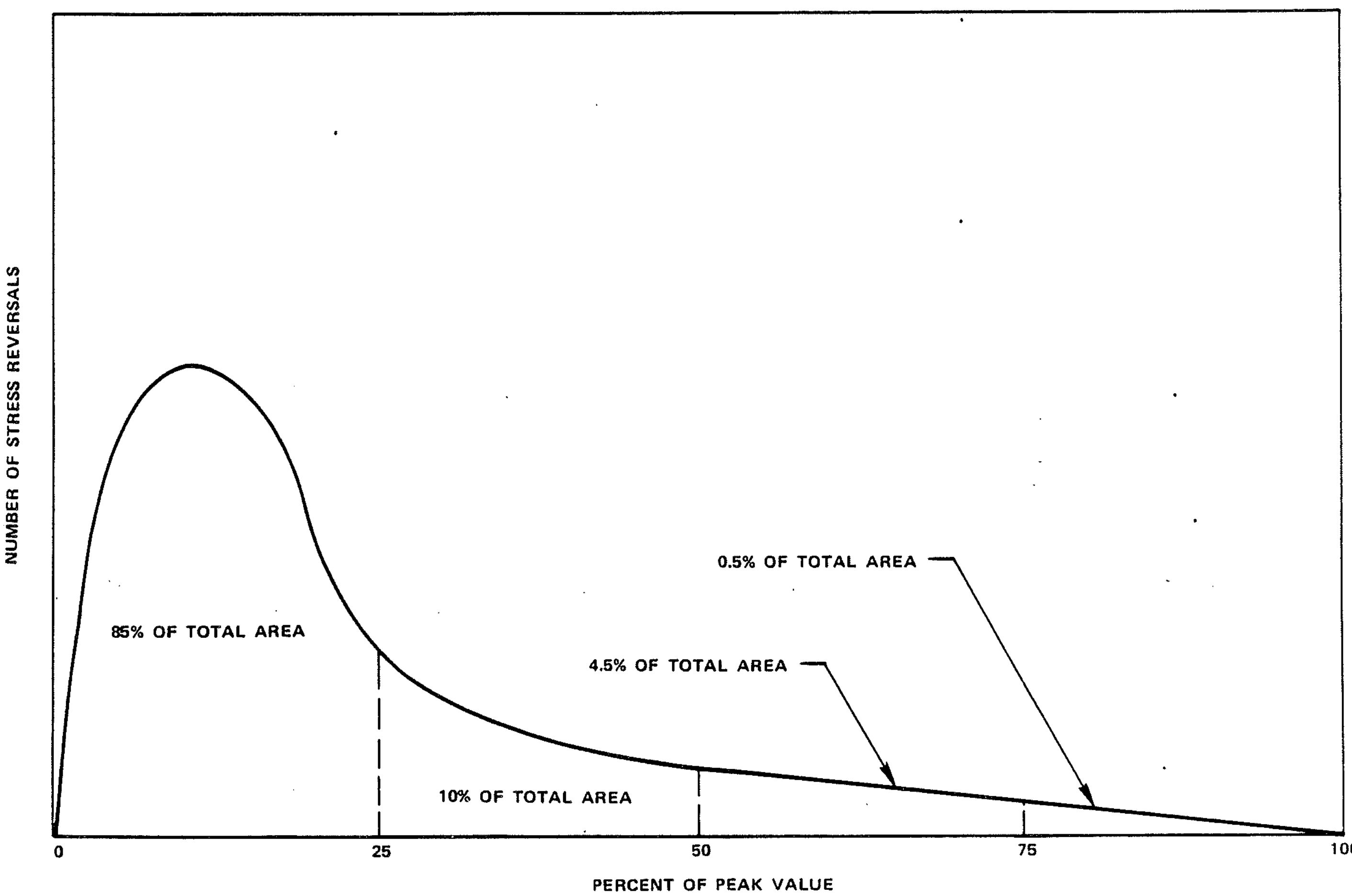

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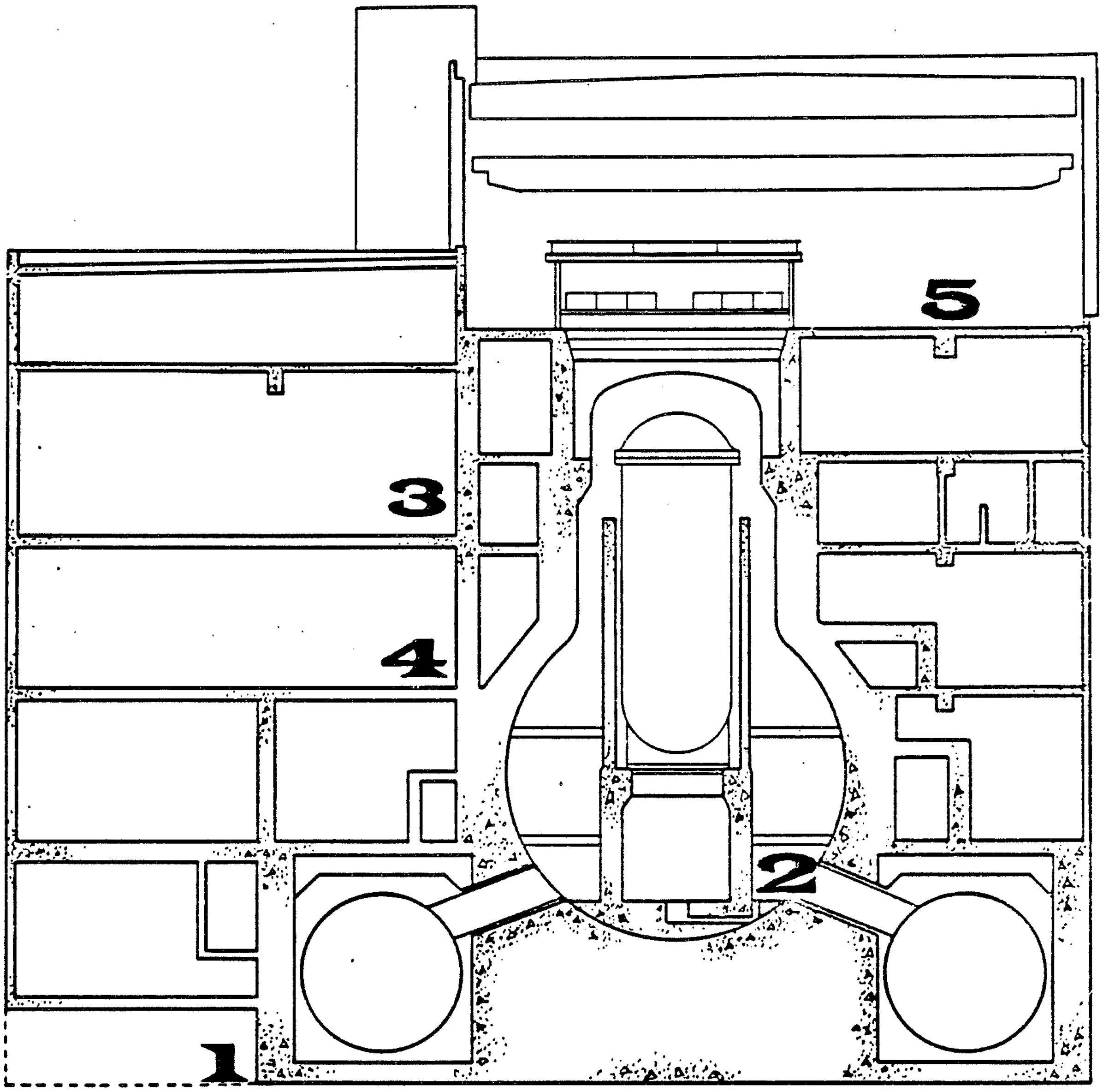
FIGURE 3.7-116

DENSITY OF STRESS REVERSALS



SECTION

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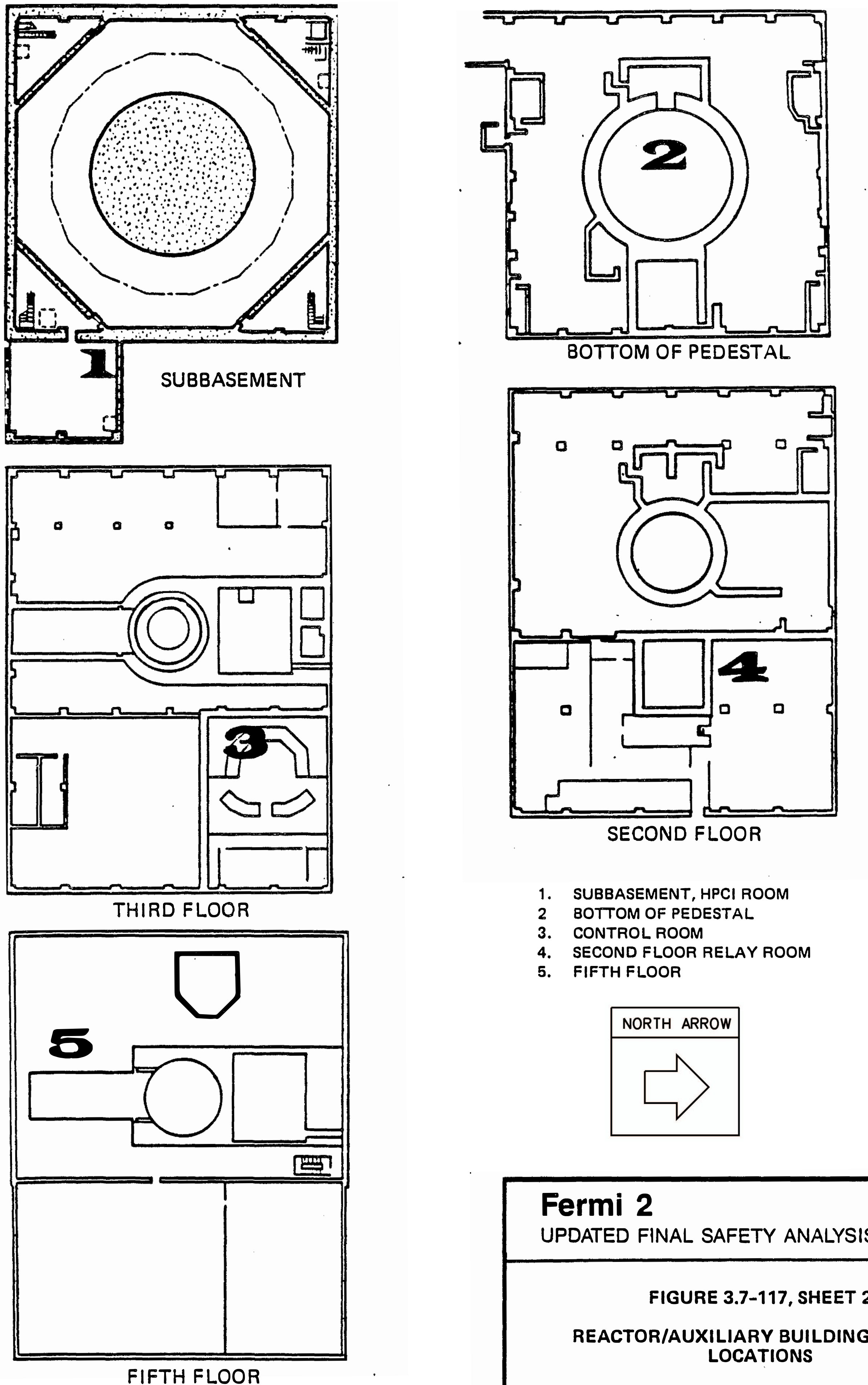
FIGURE 3.7-117, SHEET 1

REACTOR/AUXILIARY BUILDING SENSOR LOCATIONS

RALPH M. PARSONS COMPANY DRAWING



RALPH M. PARSONS COMPANY DRAWING

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FIGURE 3.7-117, SHEET 2

REACTOR/AUXILIARY BUILDING SENSOR LOCATIONS

FIGURE 3.7-118, SHEETS 1 AND 2 HAVE BEEN DELETED

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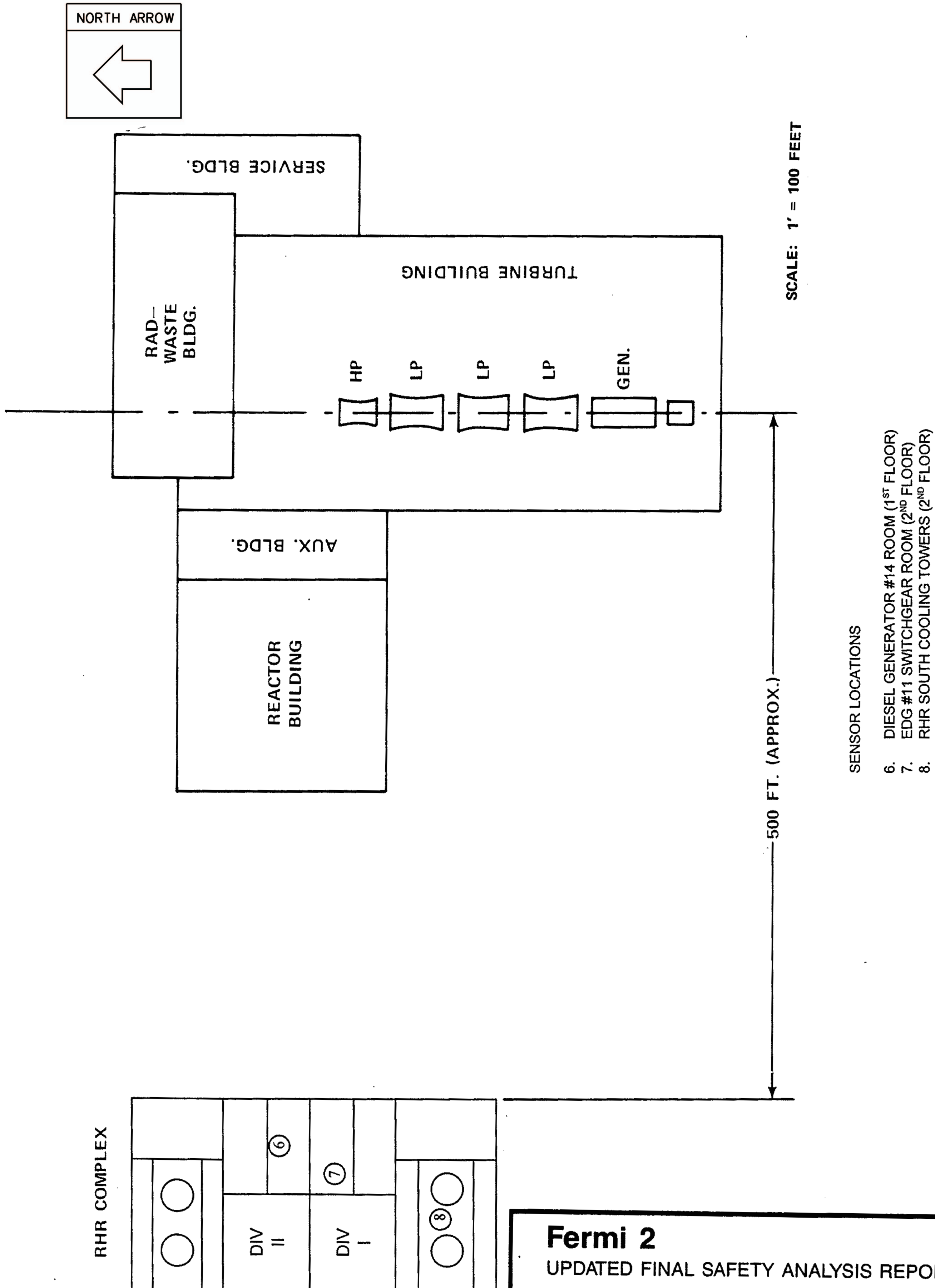

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FIGURE 3.7-119

GENERAL ARRANGEMENT PLAN
MAJOR STRUCTURES

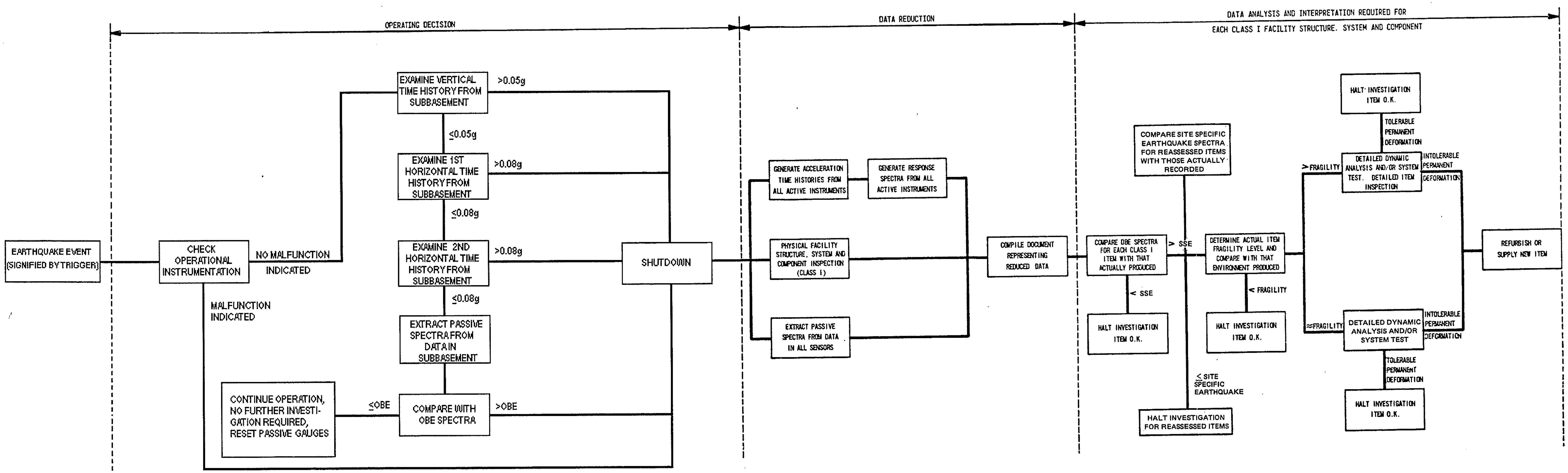

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FIGURE 3.7-120

DATA REDUCTION/ANALYSIS FLOW CHART

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3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 Concrete Containment

Fermi 2 uses a steel primary containment. Subsection 3.8.2 and Reference 1 discuss the steel containment, and Subsection 3.8.4.1.1 discusses the concrete reactor building surrounding the primary steel containment and used as a secondary containment. Information about the foundation supporting these structures can be found in Subsection 3.8.5.

3.8.2 Steel Containment System (ASME Class B Components)

3.8.2.1 Description of the Containment

3.8.2.1.1 Introduction

The primary containment (known as the Mark I containment) is a leaktight steel-plate containment vessel consisting of a light- bulb-shaped drywell and a torus-shaped suppression chamber. The primary containment was designed, erected, and pressure-tested by the Chicago Bridge & Iron Company.

The basic objective of the primary containment system is to provide the capability, in the event of the postulated design- basis accident (DBA), that is, LOCA, to limit the release of fission products to the plant site environs so that offsite doses do not exceed the values specified in 10 CFR 50.67 or 10 CFR 100. The reactor building, in conjunction with the steel containment, is designed as a secondary containment. A standby gas treatment system (SGTS) is installed to exhaust (automatically or manually) the reactor building atmosphere by way of its filter system to a vent on the auxiliary building roof, thereby causing the reactor building internal pressure to be lower than the external pressure, so that leakage is into the reactor building.

To meet the basic safety objective, the following subsidiary objectives are achieved by the system or one or more of its components:

a. The primary containment system is capable of withstanding the conditions that could result from any of the postulated accidents for which the primary containment system is assumed to be functional, including the largest amount of energy release and mass flow associated with the DBA. The criteria set forth in the NRC's Safety Evaluation Report on the Mark I containment program, NUREG-0661 (Reference 2), have been applied as the basis for acceptance of the analysis methods and the primary containment system design

b. The primary containment system has a design margin for metal/water reactions and other chemical reactions subsequent to any postulated DBA for which the primary containment system is assumed to be functional, consistent with the performance objectives of the nuclear safety systems and engineered safety feature (ESF) systems

c. The primary containment system has the capability to maintain its functional integrity during any postulated design event, including protection against missiles from internal or external sources, excessive motion of pipes, and jet forces associated with the flow from the postulated rupture of any pipe within the containment

d. The primary containment system is capable of being filled with water as an accident recovery method for any postulated DBA in which a breach of the nuclear system primary barrier cannot be sealed

e. The primary containment system, in conjunction with other nuclear safety systems and ESF systems, is capable of limiting leakage during any of the postulated DBAs for which it is assumed to be functional such that offsite doses do not exceed guideline values

f. The primary containment system has the means of rapidly condensing the steam portion of the flow from the postulated design-basis rupture of a recirculation line

g. The primary containment system has the means to

    1. Conduct the flow from postulated pipe ruptures to the suppression chamber

    2. Distribute such flow uniformly throughout the pool

    3. Limit pressure differentials between the drywell and the suppression chamber during the various postaccident cooling modes

    4. Effectively quench the steam flow from safety/relief valve (SRV) discharges.

h. The primary containment system has the capability to rapidly close or isolate all pipes or ducts that penetrate the primary containment, thereby maintaining leakage within permissible limits

i. The primary containment system is capable of being periodically leak tested to confirm the integrity of the containment at pressure

j. The primary containment system is capable of storing sufficient water to supply the emergency core cooling system (ECCS) requirements.

3.8.2.1.2 General Description

The steel primary containment consists of a drywell, vent pipes, and suppression chamber, and houses the reactor pressure vessel (RPV), recirculation system, and other primary components.

The primary containment is a steel structure composed of a series of vertical cylinders and truncated cones which form a drywell. This drywell is attached to a suppression chamber through a series of vents. The suppression chamber is a steel pressure vessel in the shape of a torus. The drywell total free air volume and torus minimum air and water volumes are referenced in Table 6.2-1.

In the event of a process system piping failure within the drywell, reactor water and steam are released into the drywell air space. The resulting increased drywell pressure then forces a mixture of air, steam, and water through the vents into a pool of water stored in the suppression chamber. The steam condenses rapidly and completely in the suppression chamber, resulting in a rapid pressure reduction in the drywell. Air that is transferred to the suppression chamber pressurizes the chamber and subsequently is vented to the drywell to equalize the pressure between the two vessels. The specific suppression chamber hydrodynamic events that result from a process system piping failure are detailed in Reference 1. A containment cooling spray system is provided to remove heat from the drywell and suppression chamber. Appropriate isolation valves are actuated during this period to ensure containment of radioactive materials that might otherwise be released from the primary containment during the course of an accident.

The primary containment system free volume is capable of being inerted with a nitrogen atmosphere during normal operation. The containment atmosphere control system is capable of reducing and maintaining the oxygen content of the atmosphere below 3.9 percent during normal operation to eliminate the possibility of a hydrogen/oxygen reaction.

3.8.2.1.2.1 Description of the Drywell

The drywell is a steel pressure vessel with a spherical lower portion 68 ft in diameter and a cylindrical upper portion 38 ft 10 in. in diameter. The overall height is approximately 114 ft 8 in. The design, fabrication, inspection, and testing of the drywell comply with the requirements of Section III, Sub-section B, of the ASME Boiler and Pressure Vessel (B&PV) Code, 1968 edition. The thickness of the cylindrical portion of the drywell and the lower spherical portion has been determined by the rules defined in Section UG-27 of ASME Section VIII. The drywell is enclosed in a reinforced-concrete biological shield (Subsection 3.8.4) and is supported by the drywell pedestal, as shown in Figure 3.8-1. The biological shield provides resistance to deformation and buckling in areas where it backs up the steel shell. Below the transition zone located at elevation 659.5 ft (New York Mean Tide, 1935), the drywell is separated from the shield by a gap of approximately 2 in.; this gap is filled with a compressible polyurethane material to allow for movement between the drywell and concrete. The polyurethane sheets are coated on both sides with an epoxy resin binder to prevent water leakage into the foam. The bottom portion of the shell (below elevation 572 ft 6 in.) is totally embedded in concrete and therefore is not subject to significant thermal stresses. The lower portion of the transition zone is backed by compacted sand to aid in condensation drainage. There are four 1-1/2-in. drain lines that can be used to remove moisture from the sand cushion in case of leakage into the gap between the drywell and shield. Shielding over the top of the drywell is provided by a removable, segmented, reinforced-concrete shield plug. See Figure 3.8-2 for a developed view of the drywell and the drywell penetration schedule.

In addition to the drywell head, one double-door air lock and two bolted equipment hatches are provided for access to the drywell (Subsection 3.8.2.1.3.4 and Figures 3.8-3 and 3.8-4). The locking mechanism on each air-lock door is designed to maintain a tight seal when the doors are subject to either external or internal pressure. The doors are mechanically interlocked so that neither door can be operated unless the other door is closed and locked. The drywell head and hatch cover are bolted in place and sealed with gaskets. Provisions have been made to permit leakage testing of the door and hatch cover seals.

The drywell will be entered during low power operation; however, access is nonroutine, infrequent, and rigidly controlled.

The exposed interior surfaces of the drywell pressure boundary are coated as described in Subsection 6.2.1.6 to protect steel surfaces from galvanic corrosion and to facilitate decontamination.

Drywell and equipment sumps are provided at the bottom of the drywell to collect and drain waste liquids. All waste liquids are then routed from the drywell sumps to the radwaste building with the aid of sump pumps.

The supporting structure for the drywell and biological shield is described in Subsection 3.8.5.

3.8.2.1.2.2 Description of the Suppression Chamber and Vent System

The suppression vent system, which connects the drywell and the suppression chamber, conducts flow from the drywell to the suppression chamber without excessive resistance and distributes this flow effectively and uniformly in the pool following a postulated DBA, intermediate-break accident (IBA), or small-break accident (SBA) in the drywell. The suppression chamber receives this flow, the steam portion is condensed, and the noncondensible gases are released to the suppression chamber air space. The suppression chamber and vent system response due to a postulated design-basis pipe rupture or main steam relief valve operation is further discussed in Reference 1. The suppression-chamber-to- drywell vacuum breakers limit the pressure differential between the drywell and the suppression chamber during postaccident primary containment system cooling.

A total of eight circular vent pipes, 6 ft in diameter, form the connection between the drywell and the suppression chamber. Jet deflectors (Figure 3.8-5) are provided in the drywell at the inlet end of each vent pipe to prevent possible damage to the vent pipes from jet forces accompanying a pipe break in the drywell. The pipes are enclosed in sleeves and provided with expansion joints to accommodate differential motion between the drywell and the suppression chamber.

The suppression chamber is a torus-shaped, continuous, leaktight steel pressure vessel with a major diameter of 112 ft 6 in. situated below and encircling the drywell. The inside diameter of the mitered cylinders that make up the suppression chamber is 30 ft 6 in. The suppression chamber shell thickness is typically 0.587 in. above the horizontal centerline and 0.658 in. below the horizontal centerline, except at penetration locations where it is locally thicker.

The suppression chamber shell is reinforced at each mitered joint location by a T-shaped ring beam. The ring beam is braced laterally with stiffeners connecting the ring beam web to the suppression chamber shell.

The suppression chamber is supported vertically at each mitered joint location by inside and outside columns and by a saddle support that spans the inside and outside columns (Figure 3.8-6). The columns, associated column connection plates, and saddle support are located parallel to the mitered joint in the plane of the ring beam web. Space has been provided outside the chamber for inspection and maintenance.

The anchorage of the suppression chamber to the basemat is achieved by a system of base plates, stiffeners, and anchor bolts located at each column and at two locations on each saddle support.

The design, fabrication, inspection, and testing of the suppression chamber comply with the requirements of ASME B&PV Code Section III, Class B. The thickness of the torus-shaped pressure vessel has been determined by means of the rules defined in Sections UG-27 and UG-28 of ASME Section VIII.

The suppression chamber shell, supports, internals, and attachments have also been reevaluated (References 1, 3, and 4) to include the hydrodynamic loading events and analysis methods defined by Topical Report NEDO-21888, Mark I Containment Program Load Definition Report (Reference 5), and NUREG-0661 (Reference 2). The appropriate service limits and edition of Section III of the ASME Code, specified in NUREG-0661, have been applied to the reevaluation.

The chamber has a total volume of approximately 251,980 ft^3. The center of the torus lies slightly below the bottom of the drywell (see Figure 3.8-7 for a plan view of the suppression chamber).

The drywell vents are connected to a torus-shaped ring header, 4 ft 3 in. in diameter, placed within the air space of the suppression chamber. Eighty 24-in.-diameter downcomer pipes project from the ring header and terminate below the water surface in the chamber pool. The pool water level is maintained to ensure a 3.00- to 3.33-ft submergence of the downcomer pipes.

A vent from the primary containment system is provided and is normally closed, but permits the vent discharge to be routed to the plant SGTS to control the release of gases from the primary containment.

The physical parameters of the primary containment are summarized in Table 3.8-1.

The total water and steam volume of the reactor vessel and recirculation system are referenced in Table 3.8-1.

3.8.2.1.3 Primary Containment Penetrations

Penetrations carry piping, mechanical systems, and electrical wiring through the biological shield and primary containment vessel. These penetrations can be classified as follows:

a. Piping penetrations (sleeved and unsleeved)

b. Electrical service penetrations

c. Mechanical system penetrations (traversing in-core probe penetrations)

d. Access openings.

To maintain design containment integrity, containment penetrations have the following design characteristics:

a. Capability to withstand peak transient pressures

b. Capability to withstand without failure the forces caused by impingement of the fluid from the rupture of the largest local pipe or connection

c. Capability to accommodate without failure the thermal and mechanical stresses that may be encountered during all modes of operation.

The number and sizes of the drywell penetrations are shown in Table 3.8-2. The corresponding details of these penetrations are shown in Figure 3.8-8. Penetrations for the suppression chamber are listed in Table 3.8-3.

3.8.2.1.3.1 Pipe Penetrations

The two general types of pipe penetrations provided are (1) those that must accommodate thermal movement (sleeved), shown in Figure 3.8-9, and (2) those that experience relatively little thermal stress (unsleeved), as shown in Figures 3.8-10 and 3.8-11.

Sleeved Penetrations

Relative or thermal movement is accommodated wherever required by using bellows-type expansion joints. For this type of joint, the penetration sleeve passes through concrete and is welded to the primary containment vessel reinforcement plate. The process line that passes through the penetration is free to move axially, and a bellows expansion joint accommodates the movement. A guard pipe surrounds the process line and is designed to protect the bellows and maintain the penetration. Insulation and air gaps reduce thermal stresses and limit the radial heat flow resulting from convection and radiation from the pipe penetration, and keep the temperature of the concrete adjacent to the sleeve below 150°F. Also, penetrations accommodating hot pipes feature cooling coils on the guard pipe.

Where necessary, the penetration lines are anchored outside the containment to limit the movement of the lines relative to the containment. The bellows accommodates the relative movement between the pipe and the containment shell. This design ensures the integrity of the flexing penetration during plant operation. The configuration of the sleeved penetrations is shown in Figure 3.8-9.

Figure 3.8-12 shows a main steam line penetration assembly, its associated inboard and outboard isolation valves, the penetration flued head anchor structure, outboard pipe whip restraint structure, and the inboard pipe whip restraint/seismic guide. The configuration is typical of those cases where high-energy line penetrations are required to resist pipe whip or jet impingement loads due to postulated pipe breaks. Design details and criteria of the various components that make up the containment penetration system shown in Figure 3.8-12 are discussed below.

Penetration Assembly

The penetration assembly consists of the process pipe, guard pipe, penetration sleeve bellows, and flued head. The process pipe is mounted concentrically within the penetration. It is connected at the outboard side to the penetration flued head, and is considered part of the piping inside containment. The process pipe is constructed of ASME Type SA-l06, Grade B, or SA-333, Grade 6 material, and is designed in accordance with ASME III, Class 1 requirements.

For all normal and upset conditions specified, design criteria limits are provided in Subsection 3.6.2.1.2.2. It should be noted that for Fermi 2 the upset condition includes the operating-basis earthquake (OBE).

The guard pipe is designed to protect the containment penetration sleeve and the containment sleeve bellows from damage due to pressurization or jet impingement loads in the event of a break in the process pipe within the penetration assembly. A jet deflector ring mounted on the inboard end of the guard pipe protects the containment sleeve from damage due to jet impingements emanating from other sources inside the containment. The guard pipe connection is hinged to the flued head to prevent excessive bending loads from being transferred from the guard pipe to the flued head, in the event of a break in the process pipe. The hinged connection is provided with a bellows to ensure the pressure integrity of the guard pipe. The guard pipe is constructed from American Society for Testing and Materials (ASTM) A-106 Grade B, or A-155 Grade KCF 70 material. Design criteria limit the stress in the guard pipe and hinge bellows to 0.90 of the yield strength, when the guard pipe is subjected to the design pressure and temperature of the process pipe.

The flued head serves as an extension of the process pipe and the process pipe anchorage point, and as a part of the primary containment pressure boundary. The flued head is constructed from a one-piece forging of ASTM A-l05 Grade II, or ASTM A-182 type-304 or 316 material, as required for compatibility with the process piping. Design of the flued head is in accordance with ASME III Class 1 requirements. The flued head forging is ultrasonically examined and radiographed in accordance with ASME III requirements. Attachment of the flued head to the anchor structure is by mechanical means; there is no welding involved.

The containment sleeve bellows allows relative movement between the containment sleeve attached to the primary containment shell and the flued head anchored to the biological shield wall. The bellows is constructed of ASTM A-240 material. Design calculations are made per Expansion Joint Manufacturers Association (EJMA) standards. Design pressure and temperature for the bellows under the various operating modes (normal, upset, emergency, and faulted) are identical with those of the primary containment.

<u>Flued Head Anchor Structure</u>

The flued-head anchor structure is provided as a structural support between the primary containment penetration flued head and the biological shield wall. The structure is designed to accept normal and upset condition loads as well as piping system reactions as a result of emergency (safe-shutdown earthquake [SSE]) and faulted condition loads (pipe whip and/or jet thrust). The structure is fabricated from a series of built-up structural tubes made from ASTM A-588 material. Design criteria under the various loading conditions limit allowable stresses to the following:

a. Normal and upset (OBE) conditions - American Institute of Steel Construction (AISC) allowable stresses

b. Emergency conditions - 0.9 x yield strength

c. Faulted conditions - 0.9 x ultimate strength.

For anchor structures that support more than one flued head, only one pipe line is assumed to be in the faulted condition at a given time.

<u>Piping Between the Containment Penetration and the Outboard Isolation Valve</u>

Piping between the primary containment flued head and the outboard isolation valve is designed to ASME III Class 1 requirements. Maximum stresses, considering all normal and upset conditions, may not exceed the limits provided in Subsection 3.6.2.1.2.2.

Outboard Pipe Whip Restraint Structure

A pipe whip restraint structure is provided at the outboard side of the outboard isolation valve. The structure is designed to limit the bending and downward thrust loads associated with pipe whips resulting from postulated breaks downstream of the isolation valve. Torsional loads on the valve are controlled by a U-bolt-type restraint system on the riser at the top of the main steam tunnel. A complete description of the outboard restraint structures, including the relevant design criteria, is given in Subsection 3.6.2.2.1.2.

Inboard Pipe Whip Restraint and Seismic Guide

The inboard pipe whip restraint and seismic guide is a dual purpose structure. During seismic events, the guide serves to support the piping system and limit its deflections to acceptable limits. During pipe-break events, a series of crushable stainless-steel tubes in the annular space between the pipe and guide intercept the pipe and absorb its kinetic energy. A more complete description of the inboard pipe-whip restraint/seismic guide is given in Subsection 3.6.1.5.1.5 and in Reference 1 to Section 3.6.

Unsleeved Penetrations

Low-temperature pipelines that contain fluids whose temperature is 150°F or less, and that do not require anchorage to the biological shield, are routed through unsleeved penetration assemblies of the type shown in Figure 3.8-11. Design criteria and analyses for those unsleeved penetrations serving ASME III Class 1 piping systems are the same as those previously described for the sleeved penetrations.

Piping penetrations serving ASME III Class 2 and 3 piping systems are classified ASME III Class 2.

The primary containment piping penetration arrangement for Class 2 systems is typically made up of three major components. They are the piping from the inboard isolation valve to the flued head, the flued head proper, and the piping from the flued head to the outboard isolation valve.

The inboard and outboard process piping between the isolation valves is designed to meet the criteria defined in Subsection 3.6.2.1.2.2.

The fabrication and materials used for the construction of the unsleeved flued heads are similar to those described previously for the sleeved flued heads. Analyses are performed in accordance with the requirements of ASME III, Subsection NC-3000.

Penetration LOCA Thermal Overpressure

NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions", raises the concern that during a postulated LOCA, piping inside containment will be heated beyond its normal operating temperature. The temperature increase would cause water trapped in piping (isolated by closed valves) to expand and the resulting pressurization could challenge piping integrity. Nonessential penetrations with piping susceptible to LOCA thermal overpressure have been evaluated in accordance with the criteria of the ASME Boiler and Pressure Vessel Code, Section III, Appendix F. Alternatively, some susceptible penetrations will relieve the overpressure condition before the limits of Appendix F are exceeded.

3.8.2.1.3.2 Electrical Penetrations

Figure 3.8-13 shows an electrical penetration and associated radiation shields of the general type that is used for power, control, and instrumentation circuits. Electrical conductors penetrating the biological shield and primary containment pass through the penetrations that are mounted in steel pipe sleeves. The sleeves are welded to the primary containment vessel.

Electrical termination cabinets are mounted on each end of the penetration canisters, which are attached by bolted flange connections and with O-ring seals. Each primary containment electrical penetration has provisions for continual testing for leaktightness with a pressure gage.

3.8.2.1.3.3 Traversing In-Core Probe Penetration

A total of seven traversing in-core probe (TIP) penetrations (five for guide tubes, and two spares) pass from the reactor building through the primary containment. (See Figure 3.8-14.) Penetrations of the insertion guide tubes through the primary containment are sealed by brazing and meet the requirements of ASME Section VIII. These seals also meet the intent of ASME Section III, even though the ASME Code has no provisions for qualifying the procedures or performance.

3.8.2.1.3.4 Personnel and Equipment Access Lock

One personnel access lock is provided for access to the drywell (see Figure 3.8-4). The lock has two gasketed doors in series and is designed to withstand the drywell design pressure. The doors are mechanically interlocked to ensure that at least one door is locked at all times. The locking mechanisms are designed so that a tight seal is maintained when the doors are subjected to either internal or external pressure. The seals on this access opening are capable of being tested for leakage. Both doors are furnished with a pressure-equalizing connection.

Two equipment access hatches and a control rod drive (CRD) removal hatch are provided and welded in the spherical portion, thus permitting extensive maintenance of the drive mechanism. These hatches have double testable seals and are bolted in place. (Figures 3.8-3 and 3.8-4 show hatch details.) The double seals are provided with a leakage test tap with which the space enclosed between the seals is pressurized to containment design pressure to test for leakage through the seal when the cover or door is locked in place.

3.8.2.1.3.5 Access To the Suppression Chamber

Access from the reactor building to the suppression chamber is provided at two locations. Each is a 4-ft-diameter manhole entrance with a double-gasketed bolted cover connected to the chamber by a 4-ft-diameter steel pipe. These access ports are bolted closed when the primary containment is required, and are opened only when the primary system temperature falls below 212°F and the pressure suppression system is not required to be operational.

The double seals are provided with a leakage test tap by which the enclosed space between the seals is pressurized to containment design pressure to test for leakage through the seal when the cover is bolted in place.

Externally, access to the suppression chamber is provided by maintenance platforms and walkways.

3.8.2.1.3.6 Access for Refueling Operations

The drywell head is removed during refueling operations. This head is held in place by bolts and is sealed with a double seal. It is bolted closed when the primary containment is required and is opened only when the primary coolant temperature falls below 212°F and the pressure suppression system is not required to be operational.

The double seals are provided with a leakage test tap by which the enclosed space between the seals is pressurized to containment design pressure to test for leakage through the seal when the cover is bolted in place.

3.8.2.2 Applicable Codes, Standards, and Specifications

Table 3.8-4 contains a comprehensive listing of all applicable codes, standards, and specifications for Fermi 2.

3.8.2.2.1 Primary Containment Vessel and Suppression Chamber

a. ASME Codes - The ASME B&PV Code, 1968 edition up to and including summer l969 Addenda, including the following sections:

    1. Section II, "Material Specifications," Part A, "Ferrous" - All steel material used in the primary containment and the suppression chamber conforms to the requirements of this section

    2. Section III, Class B, including Code Cases 1330-2, 1177-6, 1431, and 1443 - This section is used for the design, fabrication, examination, testing, inspection, and material specification for the primary containment vessel (Subsection 3.8.2.1.2.1) and the suppression chamber and vent system (Subsection 3.8.2.1.2.2)

    3. Section VIII.

b. AISC Steel Construction Manual - The "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," Sixth Edition, 1963, of the AISC is used in the design of non-pressure-retaining components

c. Code of Federal Regulations - The primary containment system leakage rate test is performed in accordance with the requirements of Appendix J, 10 CFR 50, "Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The details of the type of testing performed are addressed in Subsection 6.2.4

d. ACI Specification ACI 318-63 - This American Concrete Institute (ACI) specification, titled "Building Code Requirements for Reinforced Concrete" and dated June 1963, is used in the design of the primary containment system

e. Steiger Occupation Safety and Health Act of l970

f. NUREG-0661 - "Safety Evaluation Report, Mark I Containment Long-Term Program," July 1980 (Reference 2), which establishes requirements affecting the design and operation of the primary containment system.

3.8.2.2.2 Penetrations

a. ASME Codes - The ASME B&PV Code, 1971 edition, including the following sections:

   1. Section II, "Material Specifications," Part A, "Ferrous" - All steel material used in the penetration conforms to the requirements of this section

   2. Section III, Class 1 and 2

   3. Section XI, for inservice inspection and baseline data accumulation, is used for examination and inspection

   4. The bellows used for the piping penetrations are designed in accordance with ASME Code Case 1177-6 (Subsection 3.8.2.3.2.2)

   5. Section VIII

   6. Section III, Subsection NE, is used for the design, fabrication, and testing of primary electrical penetrations (penetrations are class MC)

   7. Section IX is used for welding.

b. EJMA Specification - The design of all expansion joints conforms to the specifications of the EJMA

c. AISC Steel Construction Manual - The "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," Sixth Edition, 1963, of the AISC is used in the design of non-pressure-retaining components

d. Code of Federal Regulations - The penetration leak- detection and leakage rate test is performed in accordance with the requirements of Appendix J, 10 CFR 50, "Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The details of the type of testing performed are addressed in Subsection 6.2.4

e. IEEE Standard 317-1972 - This standard of the Institute of Electrical and Electronics Engineers (IEEE), titled "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," is used as a guide for the design, construction, testing, and installation of electrical penetrations

f. Steiger Occupation Safety and Health Act of 1970

g. NUREG-0661 - "Safety Evaluation Report, Mark I Containment Long-Term Program," July 1980, (Reference 2) which establishes requirements affecting the design and operation of the attachments to the suppression chamber.

3.8.2.2.3 Access Opening

a. ASME Codes - The ASME B&PV Code, l968 edition up to and including summer l969 addenda, including the following sections:

   1. Section II, "Material Specifications," Part A, "Ferrous" - All steel material used in the access opening conforms to the requirements of this section

   2. Section III, Class B, including Code Cases 1330-2, 1177-6, 1431, and 1443 - This section is used for the design, fabrication, examination, testing, inspection, and material specification for all access openings described in Subsections 3.8.2.1.3.4 through 3.8.2.1.3.6

   3. Section VIII.

b. AISC Steel Construction Manual - The "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," Sixth Edition, 1963, of the AISC is used in the design of non-pressure-retaining components

c. Code of Federal Regulations - The access openings leak detection and leakage rate test is performed in accordance with the requirements of Appendix J, 10 CFR 50, "Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The details of the type of testing performed are addressed in Subsection 6.2.4

d. Steiger Occupation Safety and Health Act of 1970.

3.8.2.2.4 Special Precautions

Special precautions in addition to those required by codes are taken in the fabrication of the drywell shell. The steel plate is preheated to a minimum temperature of 200°F before welding whenever seam thickness exceeds 1 in., regardless of the surrounding air temperature. Furthermore, the plate is preheated to a minimum temperature of l00°F before the welding of all seams 1 in. or less in thickness if the ambient temperature falls below 40°F.

3.8.2.3 Loads and Loading Combinations

3.8.2.3.1 General Description

The loads and loading combinations given in Tables 3.8-5 through 3.8-17 were applied in the design of the primary containment.

The suppression chamber, vent system, and piping penetrations have also been analyzed for load combinations, including seismic and hydrodynamic loads resulting from LOCA-related and safety/relief valve discharge events. These loads and load combinations are described in Fermi 2 Mark I containment long-term program plant unique analysis reports (References 1 and 3).

Following is a general description of the loads that are normally associated with containment vessel design:

a. Seismic load - Horizontal and vertical accelerations for both the OBE and the SSE are considered. The following maximum accelerations are used to determine the seismic loads on the structure

1. OBE

Horizontal	0.08g
Vertical	0.053g

2. SSE

Horizontal	0.15g
Vertical	0.10g

b. Pipe break loads

c. Bellows loads

d. Gallery floor loads

e. Hydrostatic load - The containment may be flooded to the operating floor level during fuel-retrieving operations after an accident

f. Construction loads

g. Jet impingement loads

h. Dead load

i. Selected design temperatures and pressures

1. Suppression chamber

Internal design pressure	56 psig
External design pressure minus internal pressure	2 psid
Maximum external pressure	2 psig
Internal design temperature	281°F

2. Drywell

Internal design pressure	56 psig
External design pressure minus internal pressure	2 psid
Maximum external pressure	2 psig

Internal design temperature	340°F
3. Vent pipes and vent header	
Internal design pressure	56 psig
External design pressure minus internal pressure	2 psid
Internal design temperature	281°F

3.8.2.3.2 Loading Combinations

3.8.2.3.2.1 Drywell

a. Cylindrical and spherical portion (general shell loads) -These parts of the drywell are designed for the loads and loading combinations described in Table 3.8-5. Their design is in accordance with ASME Section III for Class B Vessels and the AISC Specification for non-pressure-retaining parts. The drywell is protected from pipe rupture jet and reaction forces as described in Subsection 3.6.1. The drywell is also protected from concentrated missile loads as described in Section 3.5. Flooding of the drywell to an elevation of 684 ft 6 in. may be necessary for postaccident recovery and is considered in Table 3.8-5. The allowable stress consideration for these loading combinations is presented in Figure 3.8-15

b. Drywell vent penetrations (accident loads) - The pressure-retaining parts of the drywell vent penetrations are designed for the loads and loading combinations described in Table 3.8-7. These parts are designed in accordance with ASME Section III for Class B Vessels

c. Spherical embedment (accident loads) - The spherical embedment section of the drywell is designed for the loads and loading combinations described in Table 3.8-8. It is designed in accordance with ASME Section III for Class B Vessels

d. Drywell knuckle region (accident loads) - The knuckle region of the drywell is designed for the loads and loading combinations described in Table 3.8-9. These parts are designed in accordance with ASME Section III for Class B Vessels

e. Drywell cone and top head (accident loads) - The cone and top head region of the drywell is designed for the loads and loading combinations described in Table 3.8-10. These parts are designed in accordance with ASME Section III for Class B Vessels

f. Drywell top flange - The drywell top flange is designed for the loads and loading combinations described in Table 3.8-11. Since the flanges are attached to, and are considered part of, the pressure boundary, their design is in accordance with ASME Section III for Class B Vessels. The water seal is designed in accordance with the AISC Specification

g. Equipment hatches - The equipment hatch doors and other pressure-retaining parts are designed for the loads and loading combinations listed in Table 3.8-12. The design of these parts is in accordance with ASME Section III for Class B Vessels

Those parts that do not form part of the pressure boundary (i.e., support bracket, pin, etc.) are designed for the loads and loading combinations listed in Table 3.8-12. The design of these parts is in accordance with the AISC Specification

h. Personnel lock - The loads and loading combinations for the personnel locks are the same as those given for the equipment hatch in Table 3.8-12

In addition, the personnel lock attachment to the drywell shell is designed for the seismic loading condition of the SSE applied at the lock center of gravity. The design of the attachment to the drywell is in accordance with ASME Section III for Class B Vessels, as shown in Table 3.8-12

i. Beam seats - These parts of the drywell are designed for the loads and loading combinations described in Table 3.8-13

Since they are not pressure-retaining parts, their design is in accordance with the AISC Specification. The part of the drywell to which the supports are attached is a pressure-retaining part and is designed for the loads and loading combinations listed in Table 3.8-13, in accordance with ASME Section III for Class B Vessels

j. Spray header - These parts of the drywell are designed for the loads and loading combinations described in Table 3.8-14

Since they are not pressure-retaining parts, their design is in accordance with the AISC Specification. The part of the drywell to which the supports are attached is a pressure-retaining part and is designed for the loads and loading combinations listed in Table 3.8-14, in accordance with ASME Section III for Class B Vessels

k. Vent jet deflectors - These parts of the drywell are designed for the loads and loading combinations described in Table 3.8-14

Since they are not pressure-retaining parts, their design is in accordance with the AISC Specification. The part of the drywell to which the supports are attached is a pressure-retaining part and is designed for the loads and loading combinations listed in Table 3.8-14, in accordance with ASME Section III for Class B Vessels

l. Stabilizer connection - The stabilizer connection is designed for the loads and loading combinations described in Table 3.8-15

Since these parts are not pressure-retaining parts, their design is in accordance with the AISC Specification. The part of the drywell to which the stabilizer plates are attached is a pressure-retaining part and is designed using the allowable limit specified in ASME Section III for Class B Vessels

m. Skirt - The skirt is designed for the loads and loading combinations described in Table 3.8-16

Since these parts are not pressure-retaining parts, their design is in accordance with the AISC Specification. The part of the drywell to which the skirt is attached is a pressure-retaining part and is designed using the allowable limit specified in ASME Section III for Class B vessels

n. Penetrations - All penetrations are designed in accordance with ASME Section III for Class B Vessels. Pressure area replacement has been completed on each penetration in addition to the design for piping loads completed on those having significant loading. The loads and loading combinations for those penetrations having significant piping reactions are described in Table 3.8-17.

3.8.2.3.2.2 Suppression Chamber

a. Cylindrical torus - These parts of the suppression chamber are designed for the loads and loading combinations described in Table 3.8-6 and Reference 1. Their design is in accordance with ASME Section III for Class B vessels and the AISC Specification for non-pressure- retaining parts. Flooding of the suppression chamber during accident recovery is considered in Table 3.8-6. The allowable stress considerations for the loading conditions applied in the original design are presented in Figure 3.8-15. The allowable stresses for the load and load combinations resulting from the subsequently identified LOCA-related and safety/relief valve discharge events are addressed in References 1 and 3.

b. Torus support system - These parts of the suppression chamber are designed for the loads and loading combinations described in Table 3.8-6 (conditions 5 through l0) and Reference 1. The allowable stress limitations are presented in Reference 1

c. Penetrations

1. Bellows/vent - The vent penetrations in the suppression chamber have been provided with a bellows expansion joint to limit stresses in the suppression chamber below those allowed by ASME Section III for Class B Vessels. The design of the bellows is in accordance with Code Case 1177-6 for the design movement specified below:

Axial (compression)	0.875 in.
Axial (tension)	0.375 in.
Lateral (positive or negative)	0.625 in.

2. General - All penetrations are designed in accordance with ASME Section III for Class B Vessels. Pressure area replacement has been completed on each penetration. The loads and loading combinations for the penetrations are presented in Reference 3.

3.8.2.4 Design and Analysis Procedures

The primary containment vessel was designed and has been analyzed in accordance with the ASME B&PV Code, 1968 edition including the summer 1969 addenda, Section III for Class B Vessels. The suppression chamber shell, supports, internals, and attachments have also been reevaluated (References 1 and 3) to include the hydrodynamic loading events and analysis methods defined by Topical Report NEDO-21888, "Mark I Containment Program Load Definition Report" (Reference 5), and NUREG-0661 (Reference 2). The appropriate service limits and editions of Section III of the ASME Code, specified in NUREG-0661, have been applied in the reevaluation. The NRC reviewed the Fermi 2 Plant Unique Analysis Report (PUAR) for the Mark I containment long-term program and concluded that the PUAR analysis verified that the completed containment modifications had restored the original design safety margin to the Fermi 2 Mark I containment (Reference 6).

3.8.2.4.1 Drywell

In general, the drywell has been analyzed and designed as an axisymmetrically loaded thin shell of revolution. The drywell has complete freedom of movement, except at its base, where it is rigidly attached to the drywell pedestal, and at its top, where it is restrained tangentially by the earthquake-stabilizer truss system (see Subsection 3.8.3 for a description of the earthquake-stabilizer truss system).

The primary shell membrane stresses have been computed for each of the load combinations specified in Subsection 3.8.2.3.2 by using the general equations for an axisymmetrically loaded shell of revolution. The derivation of these equations can be found in Chapter 14 of Reference 7. A CBI computer program, No. 7-78 (Section 3.13), which uses these equations to solve for the membrane forces, deflections, stresses, and strains, was used. The membrane stresses obtained from this analysis have been compared with the ASME allowables, and the compressive membrane stresses have been compared with the critical buckling stresses.

Shear and moment diagrams for both OBE and SSE accelerations have been calculated as outlined in Section 3.7 and are shown in Figures 3.8-16 through 3.8-21. These shears and moments are applied as static loads to determine the stresses in the drywell shell.

Included in the analysis of the drywell are the effects of dead and live loads, internal and external pressures, temperatures, earthquake loads, and the hydrostatic load of water in the drywell during an accident or refueling. The effects from penetrations, access openings, and beam seats are local in nature and are not considered to affect the overall analysis. These localized effects are analyzed individually as described in the following paragraphs. The drywell is reinforced around penetrations and access openings to minimize the effects from localized loads. The effects of significant nonaxisymmetric and transient loads are considered in all analyses.

During pressurization of the drywell, the vent pipes exert radial and vertical thrusts on the drywell shell. Because the vent pipes are equally spaced around the drywell circumference, the radial thrusts cancel each other. The upward lift of the vent pipes is conservatively neglected in the drywell analysis, because it opposes the shell weight. However, local membrane and secondary bending stresses are found at the local shell region of the vent penetration for the various vent thrusts specified in Subsection 3.8.2.3.2. This local shell analysis was completed by the method outlined in Welding Research Council Bulletin No. l07 (Reference 8), with the resulting stresses being compared with those allowed in ASME Section III for Class B Vessels. These penetrations have also been evaluated for the stress conditions resulting from the LOCA-related and safety/relief valve discharge events defined in NUREG-0661. The analytical-model and stress results are presented in Reference 1.

During erection and pressure testing, the drywell was supported by a temporary construction skirt anchored to the drywell pedestal. Openings in the skirt permit proper placing of concrete fill between the structural concrete pedestal and the drywell bottom.

The skirt is designed to provide for forces due to vent pipe thrust during the pressure test, wind load, and the dead load of the drywell vessel. On completion of the pressure tests, the skirt was embedded into the concrete slab. The local discontinuity region of the spherical shell to concrete embedment was analyzed by using the KALSHEL computer code developed by A. Kalnins of Yale University (see Section 3.13). This program performs the analysis of shells of revolution that are subject to symmetrical and nonsymmetrical loadings.

Included in the model loading are the restraining effects of the concrete surrounding the steel plates that make up the concrete transition section, as well as the effects of dead and live loads, internal and external pressures, temperatures, earthquake loads, and the hydrostatic load of water in the suppression pool (see Subsection 3.8.2.3.2). The boundary conditions for the transition section were taken as being fixed at the concrete junction. The stresses in those parts of the skirt that are not pressure-retaining were analyzed considering acting forces and moments, and were compared with the allowable limit of the AISC Specification. Refer to Subsection 3.8.3 for a discussion of the anchorage for the drywell floor to the drywell support pedestal.

The drywell shell was analyzed in the region of the knuckle for the accident condition to determine its discontinuity stresses. The knuckle was subjected to pressure loads acting normal to the shell, and to vertical loads resulting from dead, live, and seismic loads applied by the cylindrical shell to the knuckle.

The analysis was performed using the KALSHEL program. The boundary conditions were taken from the general shell analysis performed by CBI Program 7-78. Maximum primary stresses were calculated and compared with those allowed in ASME Section III for Class B Vessels.

The drywell shell was analyzed in the regions of the cone section and top head for the accident condition to determine the discontinuity stresses. The shell was subjected to an internal pressure of 56 psig. The boundary conditions were taken from general equilibrium equations.

The drywell head region is separated from the rest of the drywell by the bulkhead plate (Subsection 5.4.6.3.6.). During normal operation, atmospheres in the two regions communicate via eight 12-in. holes in the bulkhead plate. A study has been made on the head region pressure transient caused by the rupture in the head spray line.

The calculation was in two parts:

a. Mass flow out of the break

b. Pressure differential across the bulkhead plate for that mass flow rate.

Equations and physical parameters were obtained from standard engineering references and handbooks. Mass flow out of the break is based on choke flow in the 3-in. inside diameter pipe in the nozzle. Empirical equations for mass flow rate of steam under choke-flow conditions give a mass flow of 10^5 lb/sec.

Given this mass flow, the pressure drop across the bulkhead plate was calculated. The equation used was for flow rate through an orifice for cases other than choke flow. The equation includes the parameters of gas constant, R, ratio of specific heats, k, and discharge factor for the orifice, c. The values used for the parameters were selected to be representative of saturated steam, i.e., 65 lbf ft/lbm °R, 1.3 and 0.6 respectively.

The study showed the pressure in the head region would be 2 lb/in^2 greater than the drywell when all eight holes are open, and 8 lb/in^2 if half of the holes are blocked. These pressure differentials are far below design criteria on the drywell head and the bulkhead plate.

The design is in accordance with the allowable stress limits of ASME Section III for pressure-retaining parts and the AISC Specification for non-pressure-retaining elements. The loadings specified in Subsection 3.8.2.3.2 were used in the design calculations.

The drywell equipment hatches were analyzed using standard hand formulas taken from References 9, 10, and 11.

Their design is in accordance with the allowable stress limits of ASME Section III for pressure-retaining parts and the AISC Specification for non-pressure-retaining elements. The loadings specified in Subsection 3.8.2.3.2 were used in the design calculations and consist of jet forces, bolt loads, pressure (plus or minus), and earthquake forces. The local area between the equipment hatch and the drywell shell is designed to meet the area reinforcement requirement shown in Paragraph N-454 of ASME Section III.

The design evaluation of the personnel lock was completed by the same methods and loading conditions as those described for the equipment hatch, with the following exceptions:

a. A finite element study has been completed for the effect of jet forces on the rectangular door

b. Additional calculations were made for the overhang of the personnel lock with relation to local drywell shell stresses. Local stresses in the drywell were calculated by the methods outlined in Reference 8 for the loading conditions of dead weight and earthquake forces.

The beam seats, spray header, and jet deflectors were analyzed using standard hand formulas taken from References 7, 9, 10, 12, 13, and 14.

Their design is in accordance with the allowable stress limits of ASME Section III for pressure-retaining parts and the AISC Specification for non-pressure-retaining elements. The loadings specified in Subsection 3.8.2.3.2 were used in the design calculations and consist of dead and live loads, pressure, and jet forces. In addition, maximum compressive stresses were evaluated to the allowable limits specified in the buckling formulas prescribed in Welding Research Council (WRC) Standard 69.

The stabilizer mechanism is designed to transfer into the building the reaction due to seismic loads or seismic plus jet loads acting on the drywell, reactor, and shield. The stabilizer mechanism is composed of four components: (1) the connection between the reactor stabilizer and the drywell shell, (2) the male lug, (3) the female lug, and (4) the concrete shear connectors. The geometry of the stabilizer mechanism allows for radial and vertical movements due to pressure and temperature. Computed stresses in the stabilizer mechanism were found by standard elastic hand formulas taken from References 9 and 10.

The design of the stabilizer mechanism is in accordance with the allowable stress limits of ASME Section III for pressure- retaining parts and the AISC Specification for non-pressure-retaining elements. Anchorage to concrete structure was checked for allowable bearing and shear stresses in accordance with ACI 318-63. The loads and loading combinations are specified in Subsection 3.8.2.3.2.

All penetrations are designed for area replacement using the reinforcing requirements of ASME Section III. In addition, penetrations with significant nozzle loadings have been evaluated for those loadings by the methods presented in Reference 8. These loads and loading combinations are described in Subsection 3.8.2.3.2.

There are no pipe restraints attached to the drywell. However, in the event of a LOCA, pipes that penetrate the drywell may impart in-plane membrane forces to the shell.

In the areas where the drywell shell is not backed up by concrete (e.g., at the drywell head), primary stresses from all loads, including LOCA jet and piping reaction forces, are held within 0.90 times the yield strength of the material at the indicated temperature, as specified in Table N-424 of ASME Section III. The combined primary and secondary stresses are limited, in accordance with Paragraph N-414.4 of ASME Section III, to three times the allowable stress intensity values given in Table N-421 of ASME Section III.

In the areas where the drywell shell is backed by concrete, LOCA jet loadings and piping reaction forces were evaluated by conducting physical load-deflection tests. These tests were completed by CBI using a spherical shell segment of the same geometric configuration as that of the drywell sphere. Three tests were performed and consist of

a. The evaluation of the spherical shell deflection under the loading of a representative LOCA jet

b. The evaluation of the spherical shell deflection at an integrally reinforced penetration under the loading of a representative LOCA piping reaction

c. The evaluation of the spherical shell deflection at a pad reinforced penetration under the loading of a representative LOCA piping reaction.

In each test above, it has been shown that the steel shell can deflect up to 3 in. locally without failure. Considering the 2-in. gap between the drywell shell and the shielding concrete, this 3-in. deformation criterion ensures a conservative design. Permanent deformations are acceptable, providing that failure does not occur, as indicated by the above tests. The cylindrical drywell area was justified by a comparison of its rigidity to the sphere rigidity.

3.8.2.4.2 Suppression Chamber

The torus-shaped suppression chamber is designed as an axisymmetric shell of revolution. Analysis techniques similar to those used for the drywell were applied in the original design of the suppression chamber. The suppression chamber design has subsequently been reevaluated and modified for the effects of the LOCA-related loads and SRV discharge-related loads defined by NUREG-0661 (Reference 2) and the GE Report NEDO-21888, "Mark I Containment Program Load Definition Report" (Reference 5). The loads, load application methods, and structural analysis techniques applied in the suppression chamber reevaluation are described in References 1 and 3. The criteria set forth in NUREG-0661 and the original containment design specifications have been applied as a basis for acceptance of the analysis methods and the suppression chamber design.

3.8.2.5 Structural Acceptance Criteria

The structural acceptance criteria for stress and strain are specified in the codes. The following is a general listing of these criteria that for the suppression chamber have been supplemented or modified by the criteria set forth in NUREG-0661 (Reference 2):

a. The design of the primary containment is such that the stress intensities do not exceed the limits prescribed in Subarticle N-1320 of ASME Section III for Class B Vessels

b. The primary containment design details conform to the rules specified in Subarticle N-414 of ASME Section III

c. For configurations where compressive stresses occur, the critical buckling stresses were calculated, and the ratio of compressive stress to critical buckling stress was ascertained to be less than 1.0

d. Pneumatic testing is used for all pressure tests of the primary containment and is conducted in accordance with the requirements of Subarticle N-713 of ASME Section III

e. The ASME B&PV Code does not specifically address itself to deformation limits. However, the deformations have been limited by keeping the stresses within the elastic range of allowable stress requirements of ASME Section III. For local conditions, the biological shield, which is spaced 2 in. away from the primary containment, provides an ultimate limit for all local deformations

f. All non-pressure-retaining parts are designed such that no stresses exceed the limitation of the AISC Specification, Sixth Edition, 1963

g. All concrete bearing stresses are limited to the allowable stresses stated in ACI 318-63.

3.8.2.6 Design Loading Combination Stress Limits

The design loading combinations are categorized in Subsection 3.8.2.3.2. The design stress limits for these combinations are given in Subsection 3.8.2.5.

3.8.3 Concrete and Structural Steel Internal Structures of the Steel Containment

3.8.3.1 Description of the Internal Structures

The containment internal structures are Category I structures. They are mostly heavily reinforced-concrete walls and slabs, with the exception of structural steel flooring or truss systems. They are designed to support the principal nuclear steam supply equipment and the several floor levels within the containment. They are also designed for DBA condition and radiation shielding. The radiation will not adversely affect these structures. The containment internal structures include the following major components:

a. Sacrificial shield
b. Reactor pedestal
c. Drywell floor
d. Gallery floor levels
e. Earthquake-stabilizer truss system
f. Pipe-break-support truss system.

3.8.3.1.1 Sacrificial Shield

The sacrificial shield (Figure 3.8-22) is a composite structural steel and plain concrete open-ended cylindrical shell placed concentric to the reactor pressure vessel (RPV) vertical centerline (see Reference 15). It functions as a radiation and heat barrier between the RPV and the primary steel containment wall. Because of its proximity to the piping, it provides support for pipe whip restraints either directly or indirectly through a pipe-break- support truss system. The geometry of the shield is as follows.

a. Outside diameter: 29 ft 1 in.
b. Height: 48 ft 11-3/4 in.
c. Wall thickness: 1 ft 9-1/4 in.

The shield has 3/8-in.-minimum-thick steel plates on its exterior and interior surfaces and is stiffened meridionally by vertical steel columns. The steel plates are welded to the flanges of the columns, and the annular space between the plates is filled with grout.

Openings are provided in the shield for the passage of lines from the RPV to the drywell. Those openings which lie within an area 9 ft above and 16 ft below the centerline of the core are required to be shielded and are equipped with shielding doors. These doors are locked and will not open during a pipe break within the annulus. The openings above and below this band have no shielding requirements; they are covered with a light-weight rupture diaphragm designed to help relieve the annulus pressure should a break occur.

The exterior surfaces of the shield are sandblasted and coated as described in Subsection 6.2.1.6.

The shield is rigidly attached at the bottom to the reactor support pedestal; the top is free to displace in all directions, except tangential, which is restrained by an earthquake- stabilizer truss system.

3.8.3.1.2 Reactor Pedestal

The reactor pedestal concentric to the RPV vertical centerline (Figure 3.8-23) supports the RPV, sacrificial shield, and pipe whip restraints, which are attached to the pedestal, either directly or indirectly through a pipe-break-support truss system. The pedestal is a reinforced-concrete cylindrical shell with an outer radius of 14 ft 6-1/2 in. and a height of approximately 26 ft. The thickness of the shell varies from 4 ft at its base to 5 ft 6-1/2 in. at its top. The shell is reinforced on both faces by hoop and meridional steel and is integral with the drywell floor.

The RPV ring girder is bolted to a ring plate and then anchored to the top of the reactor pedestal with anchor bolts (Figure 3.8-24). Shear bars welded to the ring plate and embedded in the pedestal transfer tangential shear loads from the RPV to the pedestal; the anchor bolts resist vertical reactions and radial shear.

The inside and outside surfaces of the RPV support pedestal are coated with Nu-klad surfacer 110AA and one finish coat of Ameron polyamide epoxy No. 66.

This coating system protects the pedestal surfaces against attack by either demineralized (aggressive) water or radiation contamination and facilitates washdown.

3.8.3.1.3 Drywell Floor

The drywell floor is a reinforced-concrete pad poured on the bottom of the containment. It is connected to the basemat by special shear keys that transfer lateral forces to the mat (Figure 3.8-1). The shear lugs have anchors attached to them to transfer the uplift forces to the basemat. The main function of the drywell floor is to act as a foundation for the reactor support pedestal within the containment as well as to support the drywell vessel itself.

3.8.3.1.4 Gallery Floor Levels

There are two gallery floor levels within the containment; these serve as a means of access to the internals of the primary steel containment. The gallery levels consist of radial steel beams; the lower gallery is supported by the reactor pedestal, and the upper by the sacrificial shield.

3.8.3.1.5 Earthquake-Stabilizer Truss System

The earthquake-stabilizer truss system (Figure 3.8-25) is a structural steel truss constructed at the top elevation of the sacrificial shield. This system stabilizes the RPV and sacrificial shield under earthquake excitation by transferring the earthquake-induced forces to the concrete biological shield. The RPV is connected to the sacrificial shield, and the sacrificial shield, in turn, is connected to the primary steel containment by a steel truss arrangement. A special "shear lug" connection attaches the truss gusset plates to the containment wall. Similarly, a shear lug connection attaches the primary containment wall to the biological shield. Briefly, the shear lug connection permits radial movement and restrains tangential movement; this type of connection allows the primary steel containment to expand and contract freely under all service conditions.

3.8.3.1.6 Pipe-Break-Support Truss System

The primary steel containment is not designed to withstand loads imposed by pipe break restraints. Therefore, a structural steel pipe-break-support truss system is designed to carry those pipe restraints that cannot be carried by the steel containment (Reference 16 and

Section 3.6). The truss system is supported by the sacrificial shield, reactor pedestal, drywell floor, or any combination thereof (Figure 3.8-26).

3.8.3.2 Applicable Codes, Standards, and Specifications

This subsection lists the codes, specifications, standards of practice, regulatory guides, and other accepted industry guidelines that have been adopted to the extent applicable in the design and construction of the structures internal to the containment. To eliminate repetitious listing for each structure, the codes, standards, and specifications are listed and discussed in Table 3.8-4 and are given a specification reference number.

For each structure internal to the containment, the applicable specification reference numbers are as follows:

Structure	Specification Reference Numbers
Sacrificial shield	2 through 5, 8 through 11, 13 through 17, 20, 21, 23, 28, 34, 39, and 41
Reactor pedestal	1 through 9, 11, 13 through 17, 19, 20, 28, 34, 35, 38, 39, and 41
Drywell floor	Same as for the reactor pedestal
Gallery floor levels	20, 21, 23, 34, 39, and 41
Earthquake-stabilizer truss system	Same as for the gallery floor levels
Pipe-break-support truss system	Same as for the gallery floor levels

3.8.3.3 Loads and Loading Combinations

3.8.3.3.1 Sacrificial Shield

The sacrificial shield is designed for the following loads, in addition to its own dead and live loads (Reference 15):

a. Accident pressures caused by postulated pipe breaks at the nozzles of pipe lines, such as at the recirculation line

b. Thermal and pressure loads under normal operating and accident conditions

c. Pipe rupture loads transmitted by pipe whip restraints connected directly or indirectly through the pipe- break-support trusses to the sacrificial shield

d. Forces induced in either OBE or SSE.

The effects of shrinkage are minimized by designing the grout mix for minimal shrinkage (Subsection 3.8.4.6) and by prescribing construction techniques to minimize differential shrinkage. Where areas of critical shrinkage were defined in the design phase, appropriate shrinkage strains were input as loads in the analysis procedure.

The loading combinations and load factors shown in Tables 3.8-18 and 3.8-19 were applied in the design of the sacrificial shield. A project specification specifies the load combinations for which the sacrificial shield doors were designed.

3.8.3.3.2 Reactor Support Pedestal

The reactor support pedestal is designed to resist the following loads, in addition to its own dead load and live loads:

a. Dead and live loads from the RPV, sacrificial shield, gallery floor levels, and pipe-break-support trusses

b. Thermal and pressure loads under normal operating and accident conditions.

c. Pipe rupture loads transmitted by pipe whip restraints connected directly or indirectly through the pipe- break-support trusses to the reactor support pedestal

d. Forces induced in either OBE or SSE

e. Thermal, pressure, earthquake, and pipe rupture loads that act on the RPV and sacrificial shield and are transmitted to the reactor support pedestal via the support reactions.

The effects of shrinkage are minimized by designing the concrete mix for minimal shrinkage (Subsection 3.8.4.6) and by prescribing construction techniques to minimize differential shrinkage. Where areas of critical shrinkage were defined in the design phase, appropriate shrinkage strains were input as loads in the analysis procedure.

The loading combinations and load factors shown in Tables 3.8-19 and 3.8-20 were applied in the design of the reactor support pedestal.

3.8.3.3.3 Drywell Floor

The drywell floor is designed for the following loads in addition to its own dead and live loads:

a. The reactor pedestal support reactions (vertical, base shear, and overturning moment)

b. The reactions imposed by the pipe-break-support truss system

c. Thermal and pressure loads imposed during normal operating and accident conditions

d. Forces induced during an OBE or SSE.

The loading combinations and load factors shown in Tables 3.8-19 and 3.8-20 were applied in the design of the drywell floor.

3.8.3.3.4 Gallery Floor Levels

The gallery floor levels are designed for the following loads in addition to their own dead load:

a. A uniform platform load of 100 lb/ft^2

b. Miscellaneous loads from pipe hangers, ventilation ducts, and electrical cable trays

c. Forces induced during an OBE or SSE.

The loading combinations and load factors shown in Tables 3.8-l8 and 3.8-19 were applied in the design of the gallery floor levels.

3.8.3.3.5 Earthquake-Stabilizer Truss System

The earthquake-stabilizer truss system is designed for the following loads in addition to its own dead load (Reference 16):

a. Reactions imposed by the RPV overturning moment

b. Thermal and pressure loads imposed during normal operating and accident conditions

c. Forces induced during an OBE or SSE.

The loading combinations and load factors shown in Tables 3.8-18 and 3.8-19 were applied in the design of the earthquake- stabilizer truss system.

3.8.3.3.6 Pipe-Break-Support Truss System

The pipe-break-support truss system is designed for the following loads in addition to its own dead load:

a. Pipe whip restraint forces due to a rupture of the supported pipes (see Reference 16)

b. Miscellaneous loads from pipe hangers, ventilation ducts, and electrical cable trays as applicable

c. Temperature and pressure effects during normal operating and accident conditions.

The loading combinations and load factors shown in Tables 3.8-18 and 3.8-19 were applied in the design of the pipe-break-support truss system.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Sacrificial Shield

The sacrificial shield resists loads in the same manner as a meridionally stiffened cylindrical shell (Reference 15). The pipe whip restraints (Reference 16) are attached directly or through secondary members to column flanges, enabling the pipe whip forces to be rapidly distributed due to shell action. Buckling of the plates is prevented by welding studs to the plates and embedding the studs in the grout. The grout acts mainly as a radiation shield and is not reinforced to carry any direct or flexural stresses. However, because it is in an enclosed space, the grout has been designed to transfer shear forces between the exterior and interior plates.

The sacrificial shield is designed as an anisotropic, asymmetric, cylindrical shell. Asymmetry is due to the presence of openings in the shell. A Sargent & Lundy (S&L) three-dimensional finite element program, SLSAP, has been used to analyze the shield (Section 3.13).

The sacrificial shield columns have been modeled as beam elements; the plates and grout in between have been modeled as plane stress elements. The base of the shield was considered fixed in all directions against rotation and translation. The top of the shield was considered free to rotate and translate in all directions, except the tangential, which is fixed against translation by the earthquake-stabilizer truss system.

For both normal operating and accident conditions, the temperature gradients across the shield and their corresponding axial temperatures caused by radiation-generated heat were calculated by applying the principles of heat transfer. The temperature gradients and axial temperatures were input to the SLSAP model loading conditions (Subsection 3.8.3.3.1).

Loads were combined as appropriate, taking account of the postulated failure locations and types. It was concluded, because of the dynamic characteristics of the sacrificial shield, that peak restraint impact loads are local impulsive loads on the shield wall (Reference 15). These loads occur in the first milliseconds after rupture and are not combined with other loads. The shield wall is allowed to yield locally at regions of impact loads, provided

a. The overall capacity of the shield wall to resist elastically to the other forces listed is not affected

b. The local yielding does not produce effects that jeopardize the safety of other components.

The shield wall design is presently based on the maximum steady-state jet thrust of 1.25 p x A (where p is the pressure and A is the pipe area) at each postulated restraint location. This is conservative, since jet thrust loads decay, depending on break location proximity to feeding volumes.

For each loading condition, all the individual element stresses were output by SLSAP. A maximum stress envelope was then obtained for all the various load combinations specified in Subsection 3.8.3.3.1.

3.8.3.4.2 Reactor Support Pedestal

The reactor support pedestal is designed as a variable-thickness, axisymmetric cylindrical shell fixed at its base and free at its top. Two S&L shell structural analysis programs, SOR-III and KALSHEL (Section 3.13), were used to analyze the support. Geometry, thickness, boundary conditions, elastic properties, and loads are the inputs to both programs; stresses and force resultants at specified cross sections are the outputs. Thermal gradients and their corresponding axial temperatures caused by radiation-generated heat were calculated by applying the principles of heat transfer. The temperature gradients and axial temperatures were input as loads to SOR-III and KALSHEL.

The use of two independent analytical techniques, SOR-III and KALSHEL, provides a means of checking the analysis. Using the force-resultant outputs from SOR-III and KALSHEL, critical cross sections were chosen for detailed analysis by TEMCO-III (Section 3.13). The geometry of the concrete section and the force resultants acting on that section were inputs to TEMCO-III, and the reinforcing steel and concrete stresses are outputs. For sections that are critical in terms of allowable stresses, the capacity of a section under combined loads was verified by plotting an interaction diagram with the aid of the computer program INDIA (Section 3.13).

The top portion of the reactor support pedestal is designed to resist all seismic and pipe rupture forces transmitted through the RPV skirt and also the base of the shield wall. Pipe rupture forces and discontinuity forces at the base of the shield wall, resulting from pressurization of the annulus between the RPV and primary shield wall during a recirculation line break, were used to analyze and design the pedestal in combination with the seismic forces determined from the dynamic analysis of the reactor building.

In addition, the overturning moment and shear associated with a main steam line rupture in combination with seismic overturning moments and shears from the RPV and shield wall were used for the analysis and design of the pedestal.

The seismic and pipe rupture forces on the pedestal, discussed above, were used in combination with other loads as outlined in Subsection 3.8.3.3.2.

3.8.3.4.3 Drywell Floor

The drywell floor was analyzed using conventional elastic methods and designed in accordance with ACI 318-63 and/or ACI 318-71.

3.8.3.4.4 Gallery Floor Levels

The gallery floor levels were analyzed using conventional elastic methods and designed in accordance with the AISC Specification, 1969 Edition.

3.8.3.4.5 Earthquake-Stabilizer Truss System

The earthquake-stabilizer truss system was analyzed as a statically indeterminate truss by conventional elastic methods and designed in accordance with the AISC Specification, 1969 Edition. Applicable computer programs listed in Section 3.13 were used in part or totally for the structural analysis.

3.8.3.4.6 Pipe-Break-Support Truss System

The pipe-break-support truss system was analyzed by conventional elastic methods, as stated in the AISC Specification, 1969 Edition. Applicable computer programs listed in Section 3.13 were used in part or totally for the structural analysis.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Sacrificial Shield

The stresses in the sacrificial shield steel plates are limited to those specified in the AISC Specification, 1969 Edition, Part I, when the steel plates were being designed for the loading combinations listed in Tables 3.8-18 and 3.8-19 (see Reference 15).

The appropriate factors of safety against yield used are those discussed in the Commentary to the 1969 AISC Specifications. The allowable steel stresses were increased to 1.6 times those specified above, subject to an upper limit of 0.95 f_y (yield stress), when designing for loading conditions 5, 10, and 11 in Table 3.8-18 and corresponding stresses in Table 3.8-19. In this situation a minimum design factor of safety of 1.0/0.95 = 1.05 against yield is ensured. In both cases, deformation of the steel plates is limited because the steel stresses are kept within the elastic range.

The stresses and strains in the plain concrete between the steel plates are limited to those specified by the Strength Design Method of ACI 318-71. The factors of safety against material strength are contained in the load factors listed in Tables 3.8-19 and 3.8-20, and the undercapacity factors (ϕ) are specified by ACI 318-71.

Earthquake-induced stresses and strains are limited to the aforementioned allowables; no increases are permitted.

3.8.3.5.2 Reactor Support Pedestal

The strain in the reinforcing steel and concrete is determined in accordance with ACI 318-63 and/or ACI 318-71.

The load combinations given in Tables 3.8-19 and 3.8-20 are designed for using the yield limit criteria. The yield limit strength of the structure was defined for this design as the upper limit of elastic behavior of the effective load-carrying material. The allowable stresses for this limit are in accordance with ACI 318, with the following limitations and clarifications:

a. Concrete

1. Compression

(a) Membrane stress = $0.6\ f_c'$

(b) Membrane plus flexural stress = $0.75\ f_c'$

(c) Local compression = $0.9\ f_c'$

2. Tangential shear

The principal stresses resulting from the tangential shear stresses and membrane stresses were computed for all load combinations. If principal tension greater than $3\sqrt{f_c'}$ developed in localized areas, the reinforcing steel was designed to carry the total tensile force.

b. Reinforcing Steel

1. Maximum tensile stress = $0.9\ f_y$

2. Maximum compressive stress = $0.9\ f_y$ (load carrying).

Deformations of the reactor support pedestal are limited by specifying a maximum allowable concrete strain of 0.002 in. per in. and by keeping the stresses in the reinforcing steel below yield. Redistribution of loads caused by plastic deformations is not permitted. The factors of safety against material strength are contained in the load factors listed in Table 3.8-20 and the under capacity factors (ϕ) specified in ACI 318. Serviceability checks in accordance with ACI 318 were made to ensure adequate crack control and to limit deformations.

As in the sacrificial shield, no increases in the allowable stresses or strains specified above were permitted when designing for earthquake-induced forces.

3.8.3.5.3 Drywell Floor

The stresses and strains in the reinforced-concrete floor are limited to those specified in ACI 318-63 and/or ACI 318-71. The factors of safety against material strength are contained in the load factors listed in Table 3.8-20 and in the under capacity factors (ϕ) of ACI 318. Serviceability checks are made in accordance with ACI 318 to limit cracking of the floor.

3.8.3.5.4 Gallery Floor Levels

The allowable steel stresses and strains for the gallery floor levels are as specified in Subsection 3.8.3.5.1. Steel member deflections were calculated and kept below the allowable AISC limits or below manufacturers' recommendations for equipment supported by the steel.

3.8.3.5.5 Earthquake-Stabilizer Truss System

The allowable steel stresses and strains in the earthquake-stabilizer truss system are specified in Subsection 3.8.3.5.1. No increases in the allowable stresses and strains were permitted when designing for the earthquake-induced forces.

3.8.3.5.6 Pipe-Break-Support Truss System

The allowable steel stresses and strains for the pipe-break-support truss system are specified in Subsection 3.8.3.5.l. Steel deflections were calculated and kept below allowable AISC limits or below manufacturers' recommendations for equipment supported by steel. For a discussion of the design criteria for the pipe break restraints, see Section 3.6.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

3.8.3.6.1 Sacrificial Shield

The construction materials and quality control (QC) procedures for the sacrificial shield conform to the standards set forth in Subsection 3.8.4.6.

Radiation damage to steel is caused by a neutron flux with neutrons of energy greater than l MeV. It has been ascertained that a neutron flux incident on the inside face of the sacrificial shield steel plate is 1.6×10^7 neutrons per square centimeter per second. This will result in a neutron fluence of 2.0×10^{16} n/cm^2 in the 40-year operating life of the plant (See Appendix B, Section B, for discussion of operation beyond the original design plant life).

The first indication of neutron damage to steel is a decrease in the brittle fracture transition temperature. This occurs at a fluence of about 10^{19} n/cm^{2,} which is three orders of magnitude greater than the inside steel plates of the sacrificial shield will experience. Therefore, there is no danger of radiation damage to the sacrificial shield plates.

3.8.3.6.2 Reactor Support Pedestal, Drywell Floor, Gallery Floor Levels, Earthquake-Stabilizer Truss System, and Pipe- Break-Support Truss System

The construction materials and QC procedures for these structures conform to the standards set forth in Subsection 3.8.4.6. These structures are not located in a region of high-energy neutron flux; thus, radiation damage to these structures is not expected.

3.8.3.7 Testing and Surveillance Requirements

3.8.3.7.1 Testing and Surveillance Requirements During Plant Construction Phase

The structures specified in Subsection 3.8.3.1 are visually inspected as part of the Quality Control program. Structural steel members are examined for corrosion, excessive deformation, and warpage; their bolted or welded connections are examined for tightness and soundness. The structural integrity of reinforced concrete members is evaluated by mapping cracks in critical areas identifiable by design and by checking for spalling and excessive deformations. Anchor bolts are inspected for tightness.

Rigorous inspection is carried out during construction and in conjunction with the quality control (QC) assurance procedures for structural materials outlined in Subsection 3.8.4.6.

3.8.3.7.2 Testing and Inservice Surveillance Requirements

No inservice structural integrity and/or performance tests are conducted for containment internal structures.

3.8.4 Other Category I Structures

3.8.4.1 Description of the Structures

All structures that contain or support safety-related systems and/or equipment are designed to withstand both seismic and tornado loads, including tornado-generated missiles. Seismic loads are not considered to act simultaneously with tornado loads. Subsection 3.3.2 identifies the Category I equipment and structures that are protected against tornadoes.

No unique materials or new features are used in the design or construction of the structures described in this section.

No concrete block masonry walls have been used as load-bearing walls in Category I structures. Piping or equipment is not supported on masonry walls. The walls are basically non-load- bearing partitions. However, minor attachments of weight totaling less than about 2 percent of the weight of the wall, e.g., junction boxes or key card readers, are permitted. In cases where the weight of items attached to the wall is significant compared to the weight of the wall, the actual weight of the attachment is considered in the design.

Masonry walls, with exception of seismic Category 1 control center pressure boundary walls, are classified as seismic Category II/I structures, and are, therefore, required to maintain structural integrity during a safe shutdown earthquake (SSE). Control center boundary walls are classified as seismic Category I, since they are required to maintain pressure boundary integrity.

The walls are analyzed for dead load plus SSE Load. External supporting steel is installed, where required, to limit tension stresses in the mortar joints to allowable levels.

The block walls are modeled as plate elements with boundary conditions reflecting actual field installations. The provisions of IEEE Standard 344-1975 are used in the seismic analysis of the walls, i.e., a multi-frequency excitation and multi-mode response factor of 1.5, or any other justified factor, is used as a multiplier to the corresponding spectral acceleration. For those walls proved to be rigid by dynamic analysis, with no resonances in the response spectrum amplification range, a zero period acceleration (ZPA) is used in the seismic analysis.

Following are the remaining Category I structures not discussed above or in Subsections 3.8.1, 3.8.2, 3.8.3, or 3.8.5:

a. Reactor/auxiliary building

b. Residual heat removal (RHR) complex

c. Category I Ductbanks.

3.8.4.1.1 Reactor/Auxiliary Building

The reactor/auxiliary building is a single structure that houses both the reactor and auxiliary portions of the building. In the following subsections, the reactor portion of the reactor/auxiliary building will be referred to as the reactor building, and the auxiliary portion will be referred to as the auxiliary building.

3.8.4.1.1.1 Reactor Building

The reactor building, in conjunction with the reactor building heating and ventilating system and the SGTS, constitutes the secondary containment. The primary purposes of the secondary containment are

a. To minimize ground-level release of airborne radioactive materials

b. To provide means for a controlled release of the building atmosphere.

See Section 1.2 for general arrangement drawings of the reactor building.

The reactor building completely encloses the drywell and the suppression chamber and is supported on the reactor building foundation mat. The structure provides secondary containment when the primary containment is closed and in service, and it provides primary containment during reactor refueling and maintenance operations when the primary containment is open. The reactor building houses the refueling and reactor servicing equipment, biological shield, new- and spent-fuel storage facilities, and other reactor auxiliary or service equipment, including the reactor core cooling (RCIC) system, reactor water cleanup isolation system (RWCUS), standby liquid control system (SLCS), equipment for the CRD system, the reactor core and containment cooling system, and components of the electrical equipment.

The approximate overall dimensions of the reactor building are 116 ft by 162 ft in plan and 200 ft in height measured from the subbasement floor to the top of the parapet. The substructures and exterior walls of the building up to the refueling floor consist of poured-in-place reinforced concrete. Above the level of the refueling floor, the building structure is steel-framed with insulated metal siding with sealed joints. The reactor building has a built-up roof over insulated metal deck. The reactor building has access openings from the auxiliary building and the outside for personnel and equipment. The access openings from the outside are provided with interlocked doors that have weather-strip-type seals. Interconnecting services between the reactor building (Category I) and other nonseismic structures have the flexibility to allow for all relative movement between the structures.

The reactor building has two ventilation exhaust systems. During normal power operation, shutdown, or refueling, the normal ventilation system provides outside filtered air to all levels and equipment rooms within the building. Air is exhausted through a vent extending above the reactor building roof level. During emergencies, the normal ventilation system shuts down, and the reactor building is ventilated through the SGTS. This system causes the building internal pressure to be lower than the external pressure to ensure inleakage rather than outleakage. For a complete discussion of the heating, ventilating, and air conditioning (HVAC) system, see Section 9.4.

The biological shield is a major structure enclosed by the reactor building. This shield is a reinforced-concrete structure with a thickness of 4 to 7 ft; it extends from the bottom of the drywell to the top of the refueling floor, completely encasing the drywell structure (Figure 3.8-27). The top of the shield consists of a removable, segmented reinforced-concrete plug.

The main function of the biological shield is to serve as a radiation shield around the drywell; however, it also functions as a major mechanical barrier for the protection of the containment and reactor system against missiles that may be generated external to the primary containment. The shield resists deformation and buckling of the drywell walls over areas where the shield is in contact with the drywell. Above the transition zone, the drywell is separated from the reinforced concrete by a gap of approximately 2 in.; this gap is filled with a compressible material.

In addition to the above functions, the biological shield supports the various reactor building floor elevations that frame into it, and it resists the earthquake-induced forces that act on the RPV and sacrificial shield transferred to it through the earthquake-stabilizer truss system.

The spent-fuel storage pool, dryer-separator pool, and reactor refueling pool are reinforced-concrete structures completely lined with seam-welded stainless steel plate. The stainless steel liners prevent leakage. There are no connections that would allow the fuel storage pool to be drained below the pool grade between the reactor well and the fuel storage pool. Channels are located in the concrete directly behind the welded seams of the pool liners, and these are monitored to detect leakage from the pools. (Figures 3.8-28 through 3.8-31.)

The reactor building crane runway and supporting structure are designed as an integral part of the building superstructure to withstand earthquake accelerations at the level of the crane runway. See Figure 3.8-32 and Subsection 9.1.4.2 for details of the crane seismic safety features.

3.8.4.1.1.2 Auxiliary Building

The auxiliary building is a reinforced-concrete structure supported on a reinforced-concrete mat foundation. The exterior walls provide tornado missile protection. The main steam tunnel passes through this building. Other piping and electrical cables pass through this building in separate tunnels and connect with adjacent buildings. The reinforced-concrete steam tunnel walls, floor, and roof protect the equipment outside the tunnel from the effects of a postulated steam line break within the tunnel.

The approximate dimensions of the auxiliary building are 88 ft by 160 ft in plan and 161 ft in height, measured from the subbasement floor to the top of the parapet. See Section 1.2 for general arrangement drawings of the auxiliary building.

The auxiliary building walls, floors, and roof are constructed mainly of cast-in-place reinforced concrete. A seismic category II/I steel frame penthouse, approximately 51 ft by 20 ft in plan and 48 ft in height, with steel siding walls, is constructed on the auxiliary building roof to house the exhaust stack for the ventilation equipment located in the auxiliary building. For a complete description of the HVAC equipment in the auxiliary building, refer to Section 9.4. The auxiliary building is integrally connected to the reactor building by the common east wall of the reactor building, but separated from the turbine building by a 4-in. seismic rattle space. Services interconnecting the auxiliary and turbine buildings have the flexibility to allow for all relative movement between the two structures.

The auxiliary building houses the following major plant and safety-related systems and components:

a. Main control room

b. High-pressure coolant injection (HPCI) pumps and turbines

c. CRD pumps

d. Emergency equipment cooling water (EECW) heat exchanger and pumps

e. Main battery room

f. SGTS rooms

g. Main ventilation room

h. Main power distribution center for the reactor building

i. Switchgear rooms

j. Relay room.

3.8.4.1.2 <u>Residual Heat Removal Complex</u>

The RHR complex is a reinforced-concrete structure designed to serve as the ultimate heat sink for the reactor during normal shutdowns and postulated accident conditions. The structure is approximately 280 by 127 ft in plan and is located west of the reactor/auxiliary building. The complex consists of two divisions: Division I and Division II. Each division is comprised of a water reservoir, a pump house, a two-cell mechanical draft cooling tower, and two emergency diesel generators. Division I is in the south side, and Division II is in the north side of the complex. With the two reservoirs cross-connected to permit access to the entire ultimate heat sink inventory, each division has the capacity to safely and orderly shut down the reactor during normal and/or accident conditions completely independent of the other. See Section 1.2 for general arrangement drawings of the RHR complex.

The RHR complex houses the RHR service water (RHRSW), emergency equipment service water (EESW), and the diesel generator service water (DGSW) systems. During normal and/or accident shutdown conditions, the function of the RHRSW and EESW systems is to remove decay heat from the RHR heat exchangers and the EECW heat exchangers, respectively. The function of the DGSW system is to remove the heat from the emergency diesel generator heat exchangers during operation of the generators.

Adequate protection from potential postulated missiles has been provided, as described in Section 3.5.

Penetrations are provided for the RHRSW and EESW systems. All penetrations below Elevation 590.0 ft are watertight, as described in Subsection 2.4.2.

3.8.4.1.3 Category I Electrical Ductbank Concrete Structures

There are two sets of Category I concrete ductbanks and manholes located between the RHR complex and the Reactor/Auxiliary Building, with a Division I and Division II ductbank in each set. The first set was designed and installed during plant construction. The essential I&C and Control cables will remain in these ductbanks and the 4160-V essential power circuits are abandoned and new cables routed in the second set.

The second set of Category I ductbanks and associated, manholes and above ground cable vaults were installed to house the 4160-V essential power cables that replaced the abandoned cables in the original ductbanks due to water intrusion issues. These ductbanks also have spare conduits should the need arise to replace other essential cables in the original ductbanks.

Both set of ductbanks are cast-in-place rectangular shaped reinforced concrete ducts with each 4160-V circuit separately house in its own conduit.

3.8.4.2 Applicable Codes, Standards, and Specifications

This section lists the codes, specifications, standards of practice, regulatory guides, and other industry-accepted guidelines that have been adopted to the extent applicable in the design and construction of all Category I structures. To eliminate repetitious listing for each structure, the codes, standards, and specifications are described and discussed in Table 3.8-4 and given a specification reference number. For each Category I structure, the applicable specification reference numbers are as follows:

Reactor/auxiliary building	1 through 9, 11 through 17, 19, 20, 21, 23, 25, 26, 28, 30, and 32 through 42
RHR Complex	1A, 2B through 9, 11 through 17, 19, 20, 21, 23, 29 through 36, and 38 through 42
Category I Ductbanks	
First Set	1A, 2B through 9, 11 through 17, 19, 20, 38, 39, 41
Second Set	43 through 46

During the construction period many of the industry codes, specifications, and standards have been revised. Project specifications have been revised to incorporate later editions, as considered appropriate, than those listed in Table 3.8-4.

3.8.4.3 Loads and Loading Combinations

3.8.4.3.1 Reactor/Auxiliary Building

The load factors and loading combinations given in Tables 3.8-l8 and 3.8-20 for structural steel members and for reinforced-concrete members, respectively, and the corresponding allowable stress values given in Tables 3.8-19 and 3.8-20 have been applied in the design of the reactor/auxiliary building floor slabs, walls, roof, reactor building crane, equipment foundations, biological shield, spent-fuel pool and dryer-separator storage pool, and all other structures integral with the reactor/ auxiliary building, as outlined in Subsection 3.8.4.1.1. Following is a general discussion of the loads for which the aforementioned structures are designed.

3.8.4.3.1.1 Reactor Building Crane

The reactor building crane rails and columns are designed to carry loads transmitted from the crane for the loading combinations listed in Tables 3.8-18 and 3.8-19. The lateral force on the crane runway is 20 percent of the sum of the weights for the lift load and of the crane trolley applied at the top of each rail, one-half on each side of the runway, acting in either direction normal to the runway. The longitudinal force is 10 percent of the maximum wheel loads of the crane. An induced impact of 25 percent of the wheel load was included in the design of the support structure.

3.8.4.3.1.2 Reactor/Auxiliary Building Roof

In addition to its dead load, the reactor/auxiliary building roof is designed for a normal live load of 30 lb/ft^2. The roof purlins and decking are designed to withstand a suction pressure of 33 lb/ft^2 induced by a 90-mph wind (Subsection 3.3.1) and to blow off before a suction pressure of 72 lb/ft^2 induced by a 200-mph wind is reached. The roof decking is assumed to blow away when the wind velocity exceeds 200 mph. The structural steel frames are designed to withstand the effects of the tornado specified in Subsection 3.3.2.

3.8.4.3.1.3 Reactor/Auxiliary Building Walls

The reactor/auxiliary building walls, in addition to their own dead load, are designed for external and internal missiles and transient thermal gradients caused by the temperature differential between the exterior and interior environs (see Table 3.8-21 for the specified temperature ranges). The walls are designed to carry all members, equipment, and floor elevations framing into them.

The concrete walls up to the refueling floor elevation are designed to withstand the effects of the tornado (Subsection 3.3.2 and Reference 17). However, the metal siding walls above that elevation are designed to withstand a 90-mph wind, but are designed to blow away before a wind velocity of 200 mph is reached. Where blowout panels are not provided in walls that form totally enclosed compartments, the walls are designed for a tornado-induced internal pressure of 3 psi, as specified in Section 3.3.

Walls below grade are designed for lateral soil pressure, hydrostatic pressure from ground water level at elevation 576 ft and a surcharge of 500 lb/ft^2 under normal condition. In addition, these walls are designed for lateral soil pressure and maximum flood level specified in Section 3.4 under extreme environmental condition (similar to tornado case).

The reactor/auxiliary building walls, interacting with the reactor/auxiliary building floor slabs, are designed to resist the reactor/auxiliary building seismically induced base shears.

3.8.4.3.1.4 Reactor/Auxiliary Building Equipment Supports

To account for the effects of impact, machinery support reactions have been increased by the following percentages:

a. For elevator supports - 100 percent

b. For supports of light machinery (shaft or motor driven) -20 percent

c. For supports of reciprocating machinery or power-driven units - 50 percent

3.8.4.3.1.5 Reactor/Auxiliary Building Floor Slabs

In addition to the slab and equipment dead loads, conservative live loads have been selected for each slab. Pattern live loads have been applied to determine the maximum shears and moments in the slab. In addition to floor live and dead loads, slabs are designed for internal missiles, temperature gradients, and pressure differentials caused by operating or accident conditions as applicable.

Additionally, the reactor building slabs are loaded during ISFSI campaigns to transfer nuclear fuel from the spent fuel pool to the outdoor long-term Independent Spent Fuel Storage Installation (ISFSI) location. A HI-TRAC transfer cask with a loaded multi-purpose canister (MPC) is moved from the spent fuel pool to the Dryer-Separator Storage Pool for processing prior to movement to a low profile transport on the first floor to be moved outside the Reactor Building. Horizontal seismic loads on the HI-TRAC are reduced by an engineered Teflon friction reducing pad that is placed between the HI-TRAC and the floor of the Dryer Separator Pool and low profile transport. Horizontal seismic loads were reduced, thus reducing moments that tend to overturn the HI-TRAC such that it will not tip and induce additional vertical loads on RB slabs.

3.8.4.3.1.6 Biological Shield

In addition to its own dead load, the biological shield is designed for the temperature gradients T_a and T_o (Table 3.8-21) between the containment and exterior face of the shield, seismic loads, pipe break loads, missile loads (Section 3.5), and the dead and live load reactions of the floor elevations that frame into it.

3.8.4.3.1.7 Spent-Fuel Pool and Dryer-Separator Storage Pool

The spent-fuel pool and dryer-separator storage pool are designed for the following loads:

a. Dead load

b. Water load (including the hydrodynamic forces associated with the water set in motion by seismic accelerations)

c. Mechanical equipment loads

d. Temperature gradient caused by a maximum water temperature of 150°F

e. Accident and operating temperature differential between the containment and exterior walls for both summer and winter extremes (Table 3.8-21).

f. ISFSI HI-TRAC with a fuel-loaded multi-purpose canister (MPC).

All of the reactor/auxiliary building Category I structures and structural components are designed for the vertical and horizontal accelerations of both OBE and SSE.

3.8.4.3.2 Residual Heat Removal Complex

The load factors and loading combinations given in Table 3.8-20 for reinforced-concrete members and in Table 3.8-18 for structural steel members and the corresponding allowable stress values given in Table 3.8-19 have been applied in the design of the floor slabs, walls, equipment foundations, roof, and other structures integral with the RHR complex, as outlined in Subsection 3.8.4.3.

The discussion on the design loads for the roof, floor slabs, walls, and equipment supports found in Subsection 3.8.4.3.1 applies to the RHR complex (Reference 18). The roof of the RHR complex is designed for a total live load of 70 lb/ft^2. In addition, the RHR complex reservoir walls are designed for the hydrodynamic forces of the water in the reservoir set in motion by seismic accelerations.

3.8.4.4 Design and Analysis Procedures

3.8.4.4.1 Reactor/Auxiliary Building

The reactor/auxiliary building floor slabs, roof, walls, and miscellaneous structures integral with the reactor/auxiliary building have been analyzed and designed using conventional elastic techniques. All significant openings and discontinuities in structural members were included in the structural model. The boundary conditions selected for all structural models were determined by evaluating the stiffness (flexural, torsional, and axial) of all the members connected at a boundary point, and those conditions represent, to the extent practicable, the actual restraint conditions.

The reactor/auxiliary building walls, interacting with the floor slabs, are proportioned to resist the combination of seismically induced overturning moments, vertical loads, and shears in accordance with the applicable provisions of ACI 3l8. Adequate provisions are made to transfer wall moments, vertical loads, and shears to the mat foundation.

The computer programs used in the analysis of walls, floor slabs, beams, roof, reactor building crane, and all other structures are listed in Section 3.13.

3.8.4.4.1.1 Biological Shield

The biological shield was originally analyzed by two methods. The first analysis was based on elastic shell theory using the KALSHEL computer program. The second analysis was based on finite element theory using the computer program DYNAX (Section 3.13). The biological shield was considered to be fixed at its base and restrained by the fuel pools at its top. The results of the two independent analyses were compared, and the more conservative of the two was used for design. To determine the local effects at larger penetrations, the areas around those penetrations were modeled by finite element programs such as PLFEM-II or SLSAP (Section 3.13). The element nodes lie along the centerline of the shield, thus modeling the curvature of the wall. The size of the model was chosen such that the boundary conditions are compatible with those obtained from KALSHEL. The final load verification calculation of the Biological Shield Wall addressing additional loads was performed using ANSYS.

3.8.4.4.1.2 Spent-Fuel Pool and Dryer-Separator Storage Pool

The pools were originally analyzed as a beam simply supported at both ends by the reactor building exterior walls and rigidly supported at the middle by the biological shield. Two independent structural models were used in the analysis. First, the structure was modeled as beam elements using the appropriate stiffness and the STRESS program (Section 3.13). The STRESS output consists of the moments and shears in the pool walls for all loading conditions. Second, a finite element model was made using the PLFEM-II program. PLFEM-II output gives localized moments in the pool walls caused by hydrostatic and temperature loads. The design of the pool walls is in accordance with ACI 318 and is based mainly on the PLFEM-II output with reference being made to STRESS. The temperature gradient loads were analyzed by hand to verify the results from PLFEM-II.

In the case of the Spent Fuel Pool the analysis has been updated to incorporate final loads. The new analysis uses ANSYS to analyze the design of the Spent Fuel Pool.

During an Independent Spent Fuel Storage Installation (ISFSI) campaign or storage cask unloading, a HI-TRAC with multi-purpose canister (MPC) containing spent fuel is temporarily placed in the Dryer-Separator Storage Pool for processing. The Dryer-Separator Storage Pool structures were analyzed using the STAAD.Pro program by an equivalent frame method similar to that of the original calculation. The potential tipping and sliding motion of the cask in the Dryer Separator Storage Pool has been analyzed for OBE and SSE vertical and horizontal accelerations.

3.8.4.4.2 Residual Heat Removal Complex

The RHR complex structure was designed and analyzed using conventional elastic techniques as described for the reactor building. The computer programs used in the design and analysis process for the RHR complex are listed in Section 3.13.

3.8.4.4.3 Second Set of Category I Ductbanks and Associated Manholes and Cable Vaults

The Category I underground ductbanks, manholes and above ground cable vaults at the RHR complex have been constructed and analyzed to meet all the requirements of Category I structures as provided in ACI 349-01 and RG 1.142 & RG 1.76.

The load factors and loading combinations given in Table 3.8-20 for reinforced concrete structures have been applied in the design of these Category I structures.

3.8.4.5 Structural Acceptance Criteria

3.8.4.5.1 Reactor/Auxiliary Building

The stresses and strains in the reinforced-concrete walls, floor slabs, beams, and equipment supports in the reactor/auxiliary building are limited to those specified in ACI 318-63 and/or ACI 318-71. Serviceability checks are made in accordance with ACI 318-63 and/or ACI 318-71 to ensure crack control and to keep deflections below the limits prescribed by the manufacturers' recommendations for equipment supported by reinforced concrete.

The basic criterion for strength design is expressed as required strength versus calculated strength.

All members and all sections of members are proportioned to meet this criterion. The required strength is expressed in terms of design loads or their related internal moments and forces. Design loads are defined as loads that are multiplied by their appropriate load factor (safety factors), as given in Tables 3.8-19 and 3.8-20.

Calculated strength is that computed by the provisions of ACI 318-63 and/or ACI 318-71.

Stresses and strains in the structural steel used for the reactor/auxiliary building are limited to those specified in the 1969 AISC Specifications, Part I, when the loading combinations listed in Tables 3.8-18 and 3.8-19 were being designed for. The appropriate factors of safety against yield are those discussed in the Commentary to the 1969 AISC Specifications. The allowable steel stresses have been increased to 1.6 times those specified above, subject to an upper limit of 0.95 f_y (yield stress), when loading combinations 10 and 11 of Table 3.8-18 were being designed for. In this situation, a minimum factor of safety of 1.05 against yield has been ensured. In either case, deformations of structural steel members are limited because the stresses are kept within the elastic range, and redistribution of loads due to plastic deformations is not permitted. In addition, the deflections of all critical steel members were calculated and kept below the limits prescribed by the 1969 AISC Specifications or manufacturers' recommendations for equipment supported by steel.

The biological shield was designed using the yield limit criteria defined for the reactor support pedestal in Subsection 3.8.3.5.2.

3.8.4.5.2 Residual Heat Removal Complex

The structural acceptance criteria for the RHR complex are in accordance with the 1969 AISC Specification and ACI 318-71 and are similar in method to those described in Subsection 3.8.4.5.1.

3.8.4.5.3 Category I Ductbanks

There are two sets of Category I concrete ductbanks and manholes between the RHR complex and the Reactor/Auxiliary building, with a Division I and Division II ductbank in each set. The structural acceptance criteria for the first set of concrete ductbanks and associated manholes are in accordance with the Specifications and ACI 318-71 and are consistent with criteria described in Subsection 3.8.4.5.1 for concrete structures.

The design and construction acceptance criteria for the second set of Category I 4160-V ductbanks and associated, manholes and cable vaults is in accordance with ACI 349-01 Code, Reg. Guide 1.142 and RG 1.76.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

Noncombustible and fire-resistant materials are used wherever necessary throughout the facilities, particularly in areas containing critical portions, such as the containment, main control room, and components of ESF systems.

The construction materials for the reactor/auxiliary building and RHR complex structure conform to the standards set forth in the following discussion.

3.8.4.6.1 Concrete

"Specifications for Structural Concrete for Buildings," ACI 301, together with ACI 347, "Recommended Practice for Concrete Formwork," and ACI 318, "Building Code Requirements for Reinforced Concrete," form the general basis for the concrete specifications.

The requirements of ACI 301 have been supplemented as necessary with mandatory requirements relating to types and strengths of concrete, proportioning of ingredients, reinforcing steel, joint treatments, and testing.

Admixtures, types of cement, bonding of joints, embedded items, concrete curing, additional test specimens, additional testing services, cement and reinforcing-steel mill test report requirements, and additional concrete test requirements are specified in detail.

Specifications ACI 349-01 “Code Requirements for Nuclear Safety Related Concrete Structures” and Regulatory Guide 1.142 “Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)” provide the general basis for the design and construction of the second set of Category I 4160-V ductbanks, manholes and cable vaults.

3.8.4.6.1.1 Materials

All cement conforms to either ASTM C150, "Specification for Portland Cement Types I, II, and V," or Canadian Standards Association (CSA) Standard A5, "Portland Cements." The cement meets the requirements of the edition of the standard or specification that was current at the time the cement was manufactured.

Certified copies of mill tests, showing that the cement met or exceeded the ASTM requirements for portland cement, are furnished by the manufacturer.

Aggregates conform to the Michigan Department of State Highways Standard Specifications for Road and Bridge Construction, Article 8.02. Fine aggregates are of the natural sand designation 2NS. Coarse aggregates are of the designation 6AA; these requirements equal or exceed those of ASTM Specification C33. Where a larger size aggregate is specified for use in mass concrete portions of the work, it conforms in all respects, except size, to designation 6AA. Aggregates are free from any materials that would be deleteriously reactive in any amount sufficient to cause excessive expansion of mortar or concrete.

Mixing water is clean and free from injurious amounts of oils, acids, alkalies, salts, organic materials, or other substances deleterious to concrete or steel. Water used, as required, for concrete produced at the onsite batch plant is supplied from the Frenchtown Township Water Treatment Plant. This water is tested as processed and meets the Michigan Department of Public Health Drinking Water Standards.

An air-entraining agent is used in concrete subject to weathering. This agent conforms to the requirements of the Standard Specification for Air-Entraining Admixtures for Concrete, ASTM C260. The solution is batched by means of a mechanical dispenser capable of accurate measurement and in such a manner as to ensure uniform distribution of the agent throughout the batch during the specified mixing period. Air-entrained cement is not used.

Fly ash is obtained from the Trenton Channel Power Plant, which is also owned by the applicant, The Detroit Edison Company; it conforms to ASTM Specification C6l8. The quantity of fly ash used is determined by making laboratory tests on trial batches containing various amounts of fly ash. The mix selected is that with the maximum fly-ash-to-cement ratio that consistently yielded the specified concrete strength and provided workability.

Other admixtures to control the rate of set, reduce the water content, or improve the workability and cohesiveness of concrete are used in specific instances and conform to ASTM C494. Such admixtures are used only after tests have been made in combination with the cement and aggregates being used and specifically approved. Calcium chloride is not used under any circumstances.

3.8.4.6.1.2 Mixing

The concrete used is normal-weight concrete, with an average density of 145 lb/ft^3. Concrete or grout used for neutron shielding contains boron frits.

The proportioning of structural concrete conforms to ACI 301. In general, concrete mixes have a 28-day specified strength of 4000 psi.

Proportions of ingredients are determined and tests are conducted in accordance with the methods of ACI 301 for combinations of materials to be established by trial mixes.

Batching and mixing conform to ACI 301 and ASTM C94. Concrete ingredients are batched in an onsite central batch plant and transported to the point of placement in truck mixers, operating at agitating speed. In the event of a malfunction of the onsite plant, concrete may be batched at an offsite backup plant and truck mixed.

Concrete protection for reinforcement, preparation and cleaning of construction joints, concrete mixing, delivering, placing, and curing, with the following exceptions, is equal to or exceeds the requirements of ACI 301. The slump is varied as part of the mix design within a range of a maximum of 5 in. and a minimum of 1 in. to suit the portion of the work being placed. The minimum slump is waived on concrete used in ramps or other sloping construction. The samples for the slump tests are taken at the end of the last conveyor, chute, or pipeline before the concrete is placed in the forms.

3.8.4.6.1.3 Placement

Placing of concrete is by bottom dump buckets, chuting, concrete pump, or conveyor belt. The rate of placing concrete is controlled so that concrete is effectively placed and compacted by vibrating, with particular attention given around embedded items and near the forms.

Vertical drops greater than 6 ft are not permitted for any concrete, except where suitable equipment is provided to prevent segregation.

Cold and hot weather placing temperatures are as follows:

a. Cold weather - The ingredients are heated whenever necessary to produce concrete having a temperature of not less than 45°F. When the concrete ingredients are heated, the maximum temperature of the concrete is 80°F. Heated concrete is obtained by heating the water or aggregates, or both

b. Hot weather - Concrete deposited in hot weather has a placing temperature that does not cause difficulty from loss of slump, flash set, or cold joints. In addition, the following maximum temperatures are adhered to unless noted otherwise on the drawings:

    1. 75°F - Sections 6 ft or less but greater than 2 ft 6 in. in least dimension

    2. 65°F - Sections greater than 6 ft in least dimension.

    3. 85°F - Sections 2 ft 6 in. or less in least dimension and all electrical duct or pipe encasements.

3.8.4.6.1.4 Curing

Curing and protection of freshly deposited concrete conform to ACI 301, with the following supplementary provisions:

a. Concrete cured with water is kept wet by covering with an approved water-saturated material, by a system of perforated pipes or mechanical sprinklers, and by other approved methods that keep surfaces continuously wet. Water used for curing is clean and free from any elements that might cause objectionable effects. Curing compounds are also used

b. When curing compounds are used on surfaces on which additional concrete is to be bonded, the curing compound manufacturer provides documentary evidence that the curing compound will not prevent bond. In the event the manufacturer is unable to prove that the curing compound does not prevent bond, the curing compound is completely removed from the joint surface prior to bonding the next layer of concrete.

3.8.4.6.2 Concrete Testing

The concrete mix is designed in accordance with ACI 301-72, using method 1. Revisions of approved mix designs will be in accordance with method 2. The trial mixes are tested in accordance with the ASTM standards listed below:

Test	ASTM Designation
Making and curing of the test specimen	C192
Air content	C231
Slump	C143
Compressive	C39

Compressive strength tests are made at 7 and 28 days. A minimum of two cylinders are used for each test.

Concrete strength tests are evaluated in accordance with ACI 301 and ACI 214.

Strength of concrete is considered satisfactory if the averages of all sets of strength test results of the laboratory cured specimens at 28 days' age are equal to or greater than the specified compressive strength (f_c') of the concrete.

The Edison computer code Concrete Quality Assurance is used to evaluate the concrete compression strength tests. This program uses as input the 7- and/or 28-day test results, from individual or multiple concrete mixes, and plots the average strength as well as the moving averages to provide a means of forecasting the longterm trend of compression testing. Statistical means are used to find the concrete quality assurance variables, test averages, cumulated averages, moving averages, as the required average strength (RAS). The RAS value is calculated using the following formula:

$$\text{RAS} = \frac{\text{design strength}}{1 - (\text{ACI constant})\ (\text{coefficient of variation})}$$

where the ACI constant depends on the allowable number of tests with results falling below the design strength specified in ACI 2l4.

The field tests for slump of portland cement concrete are in accordance with ASTM C43. Any batch not meeting specified requirements is rejected. Slump tests are made frequently during concrete placement and each time concrete test specimens are made.

If cylinders should fail to meet the concrete strength requirements at 28 days, strength development and design strength requirements are reviewed. Where necessary, nondestructive tests and core tests are conducted in accordance with ASTM C42, "Method of Obtaining and Testing Drilled Cores and Sawed Beams of Concrete."

3.8.4.6.3 Reinforcing Steel

All reinforcing conforms to Grade 60 of the Standard Specification for Deformed Billet-Steel Bars for Concrete Reinforcement, ASTM A6l5.

Mill test reports showing actual chemical and physical properties, including bend tests, are furnished for each heat of steel used in making all reinforcing steel furnished.

Placing of reinforcing steel conforms to the requirements of Chapter 5 of ACI 30l, "Structural Concrete for Buildings," and Chapter 7 of ACI 3l8, "Building Code Requirements for Reinforced Concrete."

Typical reinforcing steel details are shown in Figures 3.8-33 through 3.8-38.

In addition to ASTM A615, Grade 60, reinforcing steel for the second set of Category I 4160-V ductbanks, manholes and cable vaults conforms to the requirements of ACI 301, “Structural Concrete for Buildings”, ACI 349-01, “Code Requirements for Nuclear Safety Related Concrete Structures” and RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)”.

3.8.4.6.4 Reinforcing-Steel Inspection and Testing

The testing of reinforcing bars was generally in accordance with Regulatory Guide 1.15, Testing of Reinforcing Bars for Concrete Structures, as modified below. (Regulatory Guide 1.15 was withdrawn by the NRC in July 1981.)

At least one full-diameter specimen of each bar size from every heat is control tested in accordance with ASTM A615.

Tests are performed in the field test laboratory under the jurisdiction of Edison under the direct supervision of qualified personnel.

Three test samples of each bar size and heat are obtained from the fabricator upon his receipt of an acceptable shipment from the mill. Tensile and bend tests are performed, and, if acceptable, the fabricator is authorized to proceed with fabrication.

Reinforcing concrete steel is fabricated from certified material that has been accepted by Edison. Bending conforms to ACI 318 or ACI 349-01.

3.8.4.6.5 Reinforcing-Steel Splices

Where required by space limitations or by design requirements, splices in reinforcing bars are made by cadwelding. Cadwelding is done according to written field procedures that conform to the intent of Regulatory Guide 1.10 (withdrawn in July 1981).

In order to qualify operators for making cadweld process joints, each operator is required to prepare two qualification splices for each of the splice positions to be used. The joints are tensile tested, simulating field conditions and using the same materials as those to be used in the structure.

The ends of the reinforcing-steel bars to be joined by the Cadweld process are saw cut or flame cut. The ends of the bars are thoroughly cleaned of all rust, scale, grease, oil, water, or other foreign matter before the joints are made.

3.8.4.6.6 Cadweld Testing and Inspection

Cadweld process splices are visually inspected in accordance with Regulatory Guide 1.10 (withdrawn in July 1981). Visual inspection includes random inspection of the ends of the bars for dryness and cleanliness prior to fitting the sleeve over the ends.

Completed splices are accepted or rejected according to the criteria described in the following:

Accept

a. Sound metal visible at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve

b. Filler metal may be recessed 1/8 in. to as much as 1/2 in. from end of sleeve. Recessing is due to "bulging" of packing material

c. Presence of a single shrinkage bubble below riser

d. Radial pencil lines with "stars" or dendritic gas "pipes" when not combined with other indications

e. Splice sleeve not concentric and/or rebars not in axial alignment

f. Compression-only splices with solid metal in the tap hole may have voids to a maximum of 1 in. either as spot voids or complete circumferential low fill.

Reject

a. Presence of slag and the absence of solid metal in tap hole

b. Absence of filler at one end of horizontal splice

c. Absence of filler metal at top end of vertical splice

d. Porous metal in tap hole (general porosity).

Randomly selected cadweld splices based on position, size of rebar, and operator are removed from the structure and tensile tested, or a combination of production and companion splices is tested. Testing is in accordance with the following schedule if only production splices are tested:

a. One production splice of the first 10 splices

b. One production splice of the next 90 splices

c. Two production splices of the next and subsequent groups of 100 splices.

If combinations of production and companion splices are tested, the sample frequency is as follows:

a. One production splice of the first group of 10 production splices

b. One production and three companion for the next 90 production splices

c. Three splices, either production or companion splices, for the next and subsequent groups of l00 splices. At least one-fourth of the total number of splices tested are to be production splices.

When companion splice only is required, the following schedule

a. One companion splice of the first group of 10 splices (the companion splice is included in the group count)

b. Three companion splices of the next group of 90

c. Subsequent testing to be done at the rate of three companion splices included in each 100 splices made in accordance with the following schedule:

    1. One of the first group of 30 splices

    2. One of the last group of 30 splices

    3. One of the middle group of 40 splices.

The tensile strength of each sample tested equals or exceeds 125 percent of the specified minimum yield strength for the grade of reinforcing bar used. Failure of any splice to achieve 125 percent of the specified minimum yield strength is evaluated in accordance with Section 5 of the Procedure for Sub-Standard Tensile Test Results as given in Regulatory Guide l.l0, Mechanical (Cadweld) Splices in Reinforcing Bars of Concrete Containments.

3.8.4.6.7 Structural Steel

Structural steel material, erection, and fabrication tolerances are in accordance with the l969 AISC Specification. In general, steel used for structural framing conforms to ASTM A36.

Certified copies of mill test reports showing actual chemical and physical properties are furnished for each heat of steel in accordance with ASTM A6.

Welding of structural steel is in accordance with AWS Dl.0-69, AWS D1.1-72, and AWS D1.1 later issues as well.

The material installation and inspection of high-strength bolts, in general, conform to the requirements of the specification for structural joints using ASTM A325 or A490 bolts.

3.8.4.6.8 Summary of Quality Assurance for Construction and Construction Materials

The Quality Assurance Program, implemented with a full and complete field quality control system, has provided documented assurance that the structural work at the site, including all concrete, reinforcing steel, miscellaneous steel, structural steel, and all ingredients and special processes used in producing the aforementioned items, is in accordance with the project specification requirements and the applicable ACI, ASTM, and AISC standards.

The results of the continuous concrete and reinforcing bar testing program, carried out at the site laboratory, have confirmed the effectiveness of the controls. All reinforcing has met or exceeded the design tensile strength requirements, and the evaluation program for monitoring the concrete cylinder compression test results shows a continuous average strength well above the project specification requirements.

3.8.4.7 Testing and Inservice Surveillance Requirements

Secondary containment leak-rate testing is discussed in Subsections 6.2.1.4.2, 6.2.3.3.2, and 14.1.3.2.51.

Some cracking of the reactor building exterior walls may occur during an SSE, but large, predominantly open cracks are not expected. Therefore, the leakage rate from the reactor building will not change significantly subsequent to an SSE.

No other preliminary structural integrity or performance tests are conducted on the reactor/auxiliary building or RHR complex structures. However, rigorous inspection techniques and QC procedures are adopted throughout their construction, as indicated in Subsection 3.8.4.6.

3.8.5 Foundations and Concrete Supports

3.8.5.1 Description of Foundations and Supports

The reactor/auxiliary building is supported by a reinforced-concrete basemat (Figure 3.8-39), approximately 4 ft thick. A 77-ft-diameter by l9-ft-high reinforced-concrete pad, integral with the base and centered under the RPV, supports the biological shield, drywell, reactor support pedestal, and all other structures internal to the containment (Figures 3.8-40 and 3.8-41). The RHR complex is supported by a reinforced-concrete basemat approximately 4 ft thick.

The RHR complex and the reactor/auxiliary building foundation mats bear on bedrock at approximately Elevations 551.0 and 536.0 ft, respectively.

The dead weight of the RHR complex is designed to offset the remote and unlikely occurrence of building flotation. Therefore, anchoring of the reservoir bottom is not necessary.

Category I equipment is adequately anchored to and/or supported by concrete supports. The mass of the concrete supports is generally a minimum of 2-1/2 times the mass of the supported equipment. The concrete supports and anchorages for the following Category I machinery and equipment are discussed:

a. HPCI pump and turbine

b. RCIC pump turbine and barometric condenser

c. RHR pumps

d. Core spray pumps.

The HPCI pump and turbine foundation (Figure 3.8-42) is located in the subbasement of the auxiliary building at Elevation 540.0 ft. The pump and turbine foundations consist of reinforced-concrete pads poured monolithically with each other and connected integrally with dowels to the auxiliary building basemat. The HPCI turbine concrete pad is approximately 13 ft 1-1/8 in. by 6 ft 2-1/2 in. in plan and 2 ft 8 in. high; the HPCI pump concrete pad is approximately 16 ft 5 in. by 6 ft 2-1/2 in. in plan and 3 ft 11 in. high.

The RCIC pump and turbine and barometric condenser foundations (Figure 3.8-43) are located in the subbasement of the auxiliary building at Elevation 540.0 ft. The foundations consist of reinforced-concrete pads poured monolithically with each other and integrally connected with dowels to the auxiliary building basemat. The RCIC pump pad is approximately 5 ft 5 in. by 4 ft 8 in. in plan and 2 ft 1 in. high; the RCIC turbine pad is 7 ft 1 in. by 4 ft 8 in. in plan and 2 ft 11/l6 in. high; and the barometric condenser pad is 3 ft 4 in. by 4 ft 4 in. in plan and 6 in. high.

Foundations for four RHR pumps (Figure 3.8-44) are provided at the west end of the reactor building subbasement floor at Elevation 540.0 ft. Each pump is supported by a circular steel sole plate 2-1/2 in. thick and 7 ft 6 in. in diameter; this plate is directly anchored to the reactor building base mat by 24 anchor bolts (2 in. in diameter and 2 ft 3 in. long) equally spaced along a 7-ft-diameter bolt circle. The underside of the sole plate contains a rectangular grid pattern of grout grooves 1 in. wide and 3/8-in. deep. The sole plates rest on a 1-1/2-in. grout pad; the final elevation of the top of the sole plate is 540.0 ft. Leveling screws are provided in the plates to facilitate leveling before the plates are grouted in. The sole plates are drilled and tapped for 16 bolts, 1-3/4 in. in diameter, equally spaced along a bolt circle, about 2 ft 8 in. in diameter, to receive the RHR pumps.

Foundations for four core spray pumps (Figure 3.8-45) are provided in the auxiliary building subbasement floor at Elevation 540.0 ft. Each pump is supported by a steel sole plate that is 2 in. thick and 5 ft 5 in. in diameter; this plate is anchored directly to the auxiliary building basemat by 16 anchor bolts (1-3/4 in. in diameter and 2 ft long) equally spaced along a bolt circle that is 4 ft 10 in. in diameter. The undersides of the sole plates contain a rectangular grid pattern of grout grooves 1 in. wide by 3/8 in. deep. Leveling screws are provided in the sole plate to facilitate leveling prior to placing a 2-in. grout pad under the sole plates. The sole plates are drilled and tapped for 16 bolts, 1-5/8 in. in diameter, equally spaced along a bolt circle, 2 ft 8 in. in diameter, to receive the core spray pumps.

Figure 3.8-46 shows typical reinforcing patterns at the junction of reinforced-concrete walls and the foundation basemat. Typical anchor bolt details for Category I equipment are shown in Figure 3.8-47.

3.8.5.2 Applicable Codes, Standards, and Specifications

This section lists the codes, specifications, standards of practice, regulatory guides, and other accepted industry guidelines that have been adopted to the extent applicable in the design and construction of foundations and anchorages for Category I structures and equipment. To eliminate repetitious listing, the codes, standards, and specifications are described in Table 3.8-4 and given a specification reference number. The following are the specification reference numbers for the foundations:

a. 1 through 9 inclusive

b. 11 through 21 inclusive

c. 23 and 28

d. 33 through 35 inclusive

e. 38, 39, and 41.

3.8.5.3 Loads and Loading Combinations

The load combinations and load factors given in Tables 3.8-19 and 3.8-20 have been applied in the design of reinforced-concrete foundations and supports for Category I structures and equipment. The following is a brief description of the loads for which the RHR complex and reactor/auxiliary building basemats and foundation walls have been designed:

a. Dead load

b. Live load (the live load on the reactor building basemat is 350 lb/ft^2)

c. Equipment load

d. Wind load (Subsection 3.3.1)

e. Tornado load (Subsection 3.3.2)

f. Seismic load - Horizontal and vertical accelerations are applied for both OBE and SSE

g. Lateral pressure on subsurface walls - The lateral pressures due to soil and water under static and dynamic conditions are as shown in Figures 3.8-48 and 3.8-49

h. Hydrostatic loads - Foundation walls and basemats are designed for the following water levels:

   1. Design water levels at Elevation 576 ft

   2. Maximum design flood level at Elevation 588 ft (for wave forces, see Subsection 2.4.5).

i. Hydrodynamic loads - Foundation walls are designed for the hydrodynamic forces associated with ground water in motion under both OBE and SSE. For a complete discussion of this effect, refer to Section 3.7

   The RHR complex reservoir walls are designed for the hydrodynamic forces of water in the reservoir due to ground motions during both OBE and SSE

   The reactor building basemat has been designed for torus uplift loads that occur during earthquake, accident and safety/relief valve loading conditions (Reference 19)

j. Surcharge loads - The surcharge load of 500 lb/ft^2 was investigated

k. Thermal loads - The following thermal gradients were applied to the foundation walls:

   1. A 70°F ambient inside temperature under normal operating conditions and a 50°F ambient rock/soil temperature outside

   2. A 170°F ambient inside temperature under accident conditions and a 50°F ambient rock temperature outside. This applies to the reactor building subbasement.

All loads interior and exterior to the building are transferred to the basemat through elastic deformation of the slabs, supporting walls, and columns. Differential settlements of the mat foundations are not anticipated, because they are supported by rigid bedrock.

The foundation mats are properly sized and reinforced to accommodate the total overturning moments caused by winds and tornadoes without exceeding the allowable rock bearing stress at the toe of the mat while keeping the resultant upward soil reaction within the middle third of the mat area. Passive resistance of the soil acting against the foundation walls was neglected in computing the resisting overturning moments. Moreover, any uplift resistance that may be provided by bond of the concrete to the bedrock was neglected.

Horizontal translation of the mat foundations caused by wind loading is resisted by the frictional force between the concrete mat and the bedrock. Passive resistance of the soil acting against the subgrade walls was neglected. In computing the frictional resistance, the resultant uplift force caused by the hydrostatic pressure at the base of the mat was deducted from the building dead load.

The ability of the buildings to resist torsional rotation when engulfed by a tornado is provided by the adhesive forces between the building subgrade walls and soil, and the frictional resistance between the concrete basemats and bedrock.

In general, Category I concrete equipment supports and anchorages are designed for the following loads:

a. Dead load of the equipment

b. Seismic loads - Horizontal and vertical accelerations for both OBE and SSE

c. Operating live loads - This includes overturning moments and base shears caused by rotating or reciprocating type equipment, including short circuit or seizure moments and reactions from piping connected to the machinery

d. Impact loads - To account for the effects of impact and vibration, all centrifugal and rotating equipment support reactions were increased by 20 percent.

All equipment supports and anchorages are designed to behave elastically.

3.8.5.4 Design and Analysis Procedures

The design and analysis of the mat foundations and concrete supports for all Category I structures and equipment are in accordance with conventional elastic techniques. The mat foundations have been analyzed as a "mat on a rigid foundation." The boundary conditions selected for all structural models are determined by evaluating the stiffness (flexural, torsional, and axial) of all the members connected at a boundary point and represent (to the extent practicable) the actual restraint conditions. Loads are transferred from the foundation mats to the bedrock by direct bearing contact pressure. Because the rock provides a rigid support for the basemats, concentrated loads acting on the mats are not uniformly distributed over the area of the mat, and this effect is accounted for in the design of the mat. The analysis procedures for the reactor/auxiliary building and RHR complex mats neglect any uplift resistance (negative bearing pressure) that may be afforded by the bonding of the concrete to the bedrock.

To determine the seismic forces acting on the mat, the supported structure was analyzed by means of the computer programs DSASS and DYNAS (Section 3.13). To analyze the mat foundations for hydrostatic uplift pressures and thermal loads, the computer programs SOR-III and TEMCO-III, respectively, were used (Section 3.13).

The drywell pedestal (77 ft in diameter and 23 ft high, including the 4-ft-thick mat) for the support of the RPV, drywell, and biological shield was analyzed and designed in connection with the biological shield. The horizontal base shears and overturning moments from these structures induced by OBEs and SSEs and normal and operating accident conditions (including jet impingement forces from the complete and instantaneous severance of one of the largest connecting pipes) were applied at the top of the concrete pad. The critical section for the pad is at its base; the pad was designed taking special precautions to consider the net overturning moment at the base. Figure 3.8-41 shows the reinforcing plan at the top of the drywell pedestal.

3.8.5.5 Structural Acceptance Criteria

The allowable stresses and strains for the reinforced-concrete mat foundations and supports are in accordance with the provisions of ACI 318-71 for the RHR complex and ACI 318-63 and/or ACI 318-71 for the reactor/auxiliary building.

Serviceability checks are made in accordance with the above codes to ensure adequate crack control for the mat foundations and to limit deformations of the concrete supports within the limits prescribed by ACI 318-71 (for the RHR complex) and ACI 318-63 (for the reactor/auxiliary building) or the manufacturers' recommendations for equipment supported by the concrete supports.

A study by Dames & Moore (D&M) for the RHR complex foundation (see Reference 18) showed that the bedrock is permeable because of its fragmented nature and the presence of interconnected solution cavities (vugs). An evaluation of the rock quality based on measurements from core recovery indicated that the upper 15 to 20 ft is fractured. Based on results of compression tests on core samples and applying a reduction factor to account for the rock fractures, it was estimated that the ultimate bearing capacity of the rock is on the order of 300 ksf. In the design of the mat foundations, an allowable bearing capacity of 25 ksf was adopted, thereby providing a safety factor of 12 against bearing failure.

Furthermore, as specified in the Uniform Building Code, the minimum safety factor to be adopted against overturning is 1.5. In determining the safety factor (ratio of resisting moments to overturning moments), the resisting moment of the passive soil pressure against the subgrade walls was neglected. Also, the resultant of the base bearing pressure was kept within the middle third of the basemat.

The safety factor against base sliding (ratio of the resisting forces to driving forces) was taken as a minimum of 1.5. Moreover, the passive pressure of the soil against the subgrade walls was neglected in determining the resisting forces.

Differential settlements of the mat foundations are not expected, because they rest on essentially rigid bedrock. The load and load combinations, and the resulting factors of safety, are shown in Table 3.8-22.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The construction materials used for the mat foundations, concrete supports, and machinery and equipment anchors conform to the standards set forth in Subsection 3.8.4.6, which contains a discussion of the QC procedures adopted (including the frequency and location of sampling and test requirements for the materials). Cadwelding is also described in detail.

A description of the construction procedures for the RHR complex mat is included here. A small amount of bedrock was removed prior to the placement of the structure's foundation. This rock was removed by a controlled and monitored program of blasting.

The blasting criteria and monitoring program ensured a minimal impact on the environment, including nearby residents. The excavation was dewatered by sump pumps, which included backup pumps in case of pump failure or other system malfunctions. Observation wells both on and off the site were used to monitor the ground-water level during construction to ensure that an unacceptable lowering of the adjacent ground-water level did not occur. Dewatering in the area of the RHR complex has been discontinued.

Pressure grouting of 15 to 20 ft of the upper rock layers was carried out to provide assurance that no zones of excessive fracturing or highly vugged material are horizontally continuous across the site. In consideration of the high sulfate content of the natural ground water, sulfate-resistant cement (Type V) is used for all cement grout and subsurface concrete that is in contact with the ground water. Grouting was accomplished in two stages, extending to depths of about 6 and 20 ft below the foundation level, respectively. Initial or primary holes within each zone were spaced 30 ft on centers, and final closure was achieved by grouting intermediate holes as required. The grout holes were drilled under qualified engineering supervision. The drilling methods permitted any zones of excessive fractures, vugs, or soil seams to be detected, and particular attention was given to these zones in subsequent grouting operations.

The foundations of the RHR complex are installed on the Bass Islands dolomite and are designed to limit the bearing pressures to values much less than the safe bearing capacity of 50,000 lb/ft^2.

As the foundations of the reservoirs are below the natural ground-water level, they are subject to uplift pressures when the reservoir is empty. The reservoir could be totally empty for possible maintenance. Flotation of the reservoirs has been prevented by using a 4-ft basemat.

Site fill is crusher-run rock material, predominantly dolomite 1-1/2 in. and smaller in diameter. It is placed in loose horizontal lifts approximately 12 in. deep. Each lift is compacted with a vibration roller similar to that used in the compaction test area. Dames & Moore conducted a seismic investigation of the compacted crushed rock (see Reference 20) and measured both compression and shear waves. These data were incorporated into the design of the Category I buildings. The RHR complex, being a Category I facility, was designed by applying the seismic design criteria used for the reactor/auxiliary building.

3.8.5.7 Testing and Inservice Surveillance Techniques

Preliminary field explorations were conducted to evaluate the soil and rock conditions at the Fermi 2 site. The field investigation consisted of the following:

a. Geologic test boring program - All geologic borings were logged in detail, and a general description of the soils and rocks encountered at the site was recorded (Subsection 2.5.1.2)

b. Water pressure tests - Pressure tests were performed during the drilling of representative borings to evaluate the bedrock (Subsection 2.5.1.2)

c. Piezometer observations - Piezometers were installed in several borings to study the seasonal fluctuations of the ground-water table; periodic ground-water level measurements were taken (Subsection 2.4.13.2)

d. Geologic reconnaissance - A geologic reconnaissance of the quarry that serves as a source of fill material for the site area was carried out to assist in the interpretation of subsurface conditions (Subsection 2.5.1.2).

In addition to the field explorations, the following laboratory tests were performed on undisturbed soil samples extracted from the borings to evaluate the physical properties of the soil and fill materials at the site (Subsection 2.5.1.2):

a. Unconfined compression tests - The purpose of this test was to determine the stress-strain characteristics of the soil. In addition, laboratory unconfined compression tests were performed on representative rock samples to determine the strength of the rock

b. Pulsating triaxial load tests - The pulsating triaxial load tests yield the dynamic moduli of elasticity and the shear moduli for the soils. The shear moduli of the bedrock were computed using the elastic relationships between the shear modulus, the modulus of elasticity, and Poisson's ratio. The moduli of subgrade reaction for the bedrock were computed by using the relationship between the subgrade modulus, the modulus of elasticity, Poisson's ratio, and the size of the loaded area used for the pulsating triaxial load specimen

c. Consolidation tests - The consolidation tests indicate the load-settlement properties of the soils

d. Moisture and density tests.

Routine observations are made of the mat foundations and concrete supports to determine the existence, if any, location, and extent of cracking. Representative equipment anchor bolts are tested periodically for tightness.

Rigorous inspection was carried out during construction in conjunction with the QC procedures outlined in Subsection 3.8.4.6 for the structural materials.

3.8 DESIGN OF CATEGORY I STRUCTURES

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18. Dames & Moore, "Foundation Investigation Residual Heat Removal Complex, Enrico Fermi Unit II, Final Report for The Detroit Edison Company," August 28, 1972.

19. "Enrico Fermi Atomic Power Plant Unit 2, Evaluation of Site-Specific Earthquake Base Mat Uplift Loads," DET-04-050, Nutech Engineers, May 1982.

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TABLE 3.8-1 SUMMARY OF THE PRIMARY CONTAINMENT PHYSICAL PARAMETERS

Primary system volume:

Volume water in vessel, ft^3	11,744
Volume steam in vessel, ft^3	9470
Volume water in recirc. loops	1168
Total, ft^3	22,382

Containment heat removal capacity per loop, using 90°F service water and 170°F pool temperature; one LPCI and two service water pumps, Btu/hr	66.5×10^6
Drywell free volume, including vent system, ft^3	163,730
Suppression chamber total volume, excluding vent System, ft^3	251,980
Submergence of vent pipe below suppression pool surface, ft, minimum	3.0
Submergence of vent pipe below suppression pool surface, ft, maximum	3.33

TABLE 3.8-2 DRYWELL PENETRATIONS

Type of Service	Penetration Number[a]	Type of Penetration[b]	Line Size (in.)	Sleeve Diameter (in.)	Number Provided
Equipment hatch	X-1A	-	-	156 I.D.	1
Equipment hatch	X-1B	-	-	144 I.D.	1
Personnel air lock	X-2	-	-	122 I.D.	1
Vent line	X-5A through X-5H	-	72	85 I.D.	8
CRD removal hatch	X-6	-	-	24 I.D.	1
Main steam	X-7A through X-7D	1	26	42	4
Steam drain	X-8	1	3	16	1
Reactor feedwater	X-9A, X-9B	1	24	40	2
Steam to RCIC turbine	X-10	1	4	18	1
Steam to HPCI turbine	X-11	1	10	28	1
RHR supply	X-12	1	20	36	1
RHR return	X-13A, X-13B	1	24	34	2
Spare	X-14	1	6	20	1
H2 control, Div. I	X-15	3	4	20	1
Core spray system	X-16A, X-16B	1	12	28	2
RHR RPV head spray[e]	X-17	1	6	20	1
DFDS[c] discharge	X-18	2	3	6	1
DEDS[d] discharge	X-19	2	3	6	1
Service water	X-20	2	6	8	1
Service air (Plugged)	X-21	2	1	3	1
Nitrogen supply	X-22	2	1	3	1
RBCCW supply	X-23	2	10	14	1
RBCCW return	X-24	2	10	14	1
Vent from drywell	X-25	3	24	24	1
Vent to drywell	X-26	3	24	24	1

TABLE 3.8-2 DRYWELL PENETRATIONS

Type of Service	Penetration Number[a]	Type of Penetration[b]	Line Size (in.)	Sleeve Diameter (in.)	Number Provided
Containment atmosphere sample and containment water level instrumentation	X-27	5	1	10	1
Jet pump instrumentation	X-28A, X-28C, X-28D	4	1	10	3
Spare	X-28B, X-28E, X-28F, X-28G	4	1	10	4
RPV instrumentation	X-29A, X-29B	4	1	10	2
Recirculating pump instrumentation	X-30A, X-30B	4	1	10	2
Spare	X-31A	4	1	10	1
Drywell on-line pressure control	X-31B	4	1	10	1
Recirculating flow to RPV	X-32A, X-32B	4	1	10	2
RPV instrumentation	X-33A, X-33B	4	1	10	2
EECW supply and return	X-34A, X-34B	2	10	14	2
TIP drive system	X-35B through X-35F	7	3/8	1 1/2	5
Spare	X-35A, X-35G	7	3/8	1 1/2	2
Nitrogen to drywell	X-36	2	4	10	1
Control rod drive insert	X-37A through X-37D	6	1	1	193
Control rod drive withdraw	X-38A through X-38D	6	3/4	1	193
Containment spray supply	X-39A, X-39B	3	12	12	2
RPV instrumentation	X-40A through X-40D	4	1	10	4
Spare	X-41	2	1	6	1
Standby liquid control	X-42	2	2	6	1
RWCU supply	X-43	1	6	30	1
H2 control, Div. II	X-44	3	4	26	1
Spare	X-45	1	20	34	1
Main steam flow	X-46A, X-46B	4	1	10	2

TABLE 3.8-2 DRYWELL PENETRATIONS

Type of Service	Penetration Number[a]	Type of Penetration[b]	Line Size (in.)	Sleeve Diameter (in.)	Number Provided
Reactor protection system	X-47	4	1	10	1
Containment atmosphere sample	X-48	4	1	10	1
Recirculating pump seal purge	X-49	4	1	10	1
Spare	X-50	4	1	10	1
Recirculating pump seal purge	X-51	4	1	10	1
Main steam flow	X-52	4	1	10	1
RPV instrumentation	X-53	4	1	10	1
Reactor level pressure	X-54A, X-54B	1	4	10	2
Reactor level pressure	X-55A, X-55B	1	4	10	2
Neutron monitor	X-100A, X-100G	8	-	12	2
Spare	X-100C, X-100E, X-100F	8	-	12	3
Low level signal vibration test	X-100D	8	-	12	1
Low voltage switching	X-100B	8	-	12	1
Recirculating pump power, 5 kV	X-101A through X-101F	8	-	12	6
Neutron monitor	X-102A	8	-	12	1
Low-voltage switching/RPS	X-102B	8	-	12	1
Thermocouples and misc. sign	X-103A	8	-	12	1
Neutron monitor	X-103B	8	-	12	1
CRD position indicators	X-104A through X-104F	8	-	12	6
Low voltage power (480 V)	X-105A	8	-	12	1
Low voltage switching/RPS	X-105B	8	-	12	1
Low voltage switching/RPS	X-105C	8	-	12	1

TABLE 3.8-2 DRYWELL PENETRATIONS

Type of Service	Penetration Number[a]	Type of Penetration[b]	Line Size (in.)	Sleeve Diameter (in.)	Number Provided
Low voltage power (480 V)	X-105D	8	-	12	1
Spare	X-106A	8	-	12	1
Low-level signal vibration test	X-106B	8	-	12	1
Spare	X-107A	8	-	12	1
Thermocouple	X-107B	8	-	12	1

[a] See Detroit Edison drawing 6C721-2304.

[b] See Figure 3.8-8.

[c] Drywell floor drain sump.

[d] Drywell equipment drain sump.

[e] RHR RPV head spray piping is no longer attached to RPV. Portion of head spray piping between RPV and bulkhead penetration is removed. The remaining head spray piping within the drywell is blind flanged.

TABLE 3.8-3 SUPPRESSION CHAMBER PENETRATIONS

Type of Service	Penetration Number[a]	Type of Penetration	Line Size (in.)	Sleeve Diameter (in.)	Number Provided
Access hatch	X-200A, X-200B	-	-	48 I.D.	2
Vent line	X-201A through X-201H	-	72	80-1/8 I.D.	8
Vacuum breaker	X-202A through X-202M	-	18	18	12
Vacuum breaker air	X-204A through X-204M	14	1	1	12
Purge penetrations	X-205A through X-205D	5	20	20	4
Liquid level indicator	X-206A through X-206D	6	1	1	4
Vent line drain	X-207A through X-207H	12	1	1	8
Electromatic relief valve discharge	X-208A through X-208P	-	12	12	15
Spares	X-209A, X-209C	8	1	1	2
Thermocouples	X-209B, X-209D	8	1	1	2
RHRS test line	X-210A, X-210B	6	18	18	2
RHRS to spray header	X-211A, X-211B	6	6	6	2
RCIC turbine exhaust	X-212	8	8	10	1
Torus water management discharge supply	X-213A, X-213B	-	8	8	2
RCIC and HPCI steam return vacuum Breaker	X-214	5	4	4	1
Post-LOCA H_2 continuous suction, Div. I	X-215	5	4	4	1
Spares	X-216A, X-216B	5	1/2	2	2
Grab sample	X-217	5	1/2	2	1
Post-LOCA H_2 continuous return, Div. I	X-218	5	4	10	1
Post-LOCA H_2 continuous suction, Div. II	X-219	5	10	10	1
HPCI turbine exhaust	X-220	8	24	24	1

TABLE 3.8-3 SUPPRESSION CHAMBER PENETRATIONS

Type of Service	Penetration Number[a]	Type of Penetration	Line Size (in.)	Sleeve Diameter (in.)	Number Provided
Condensate from HPCI turbine drain pot	X-221	8	2	2	1
RCIC vacuum pump discharge	X-222	8	2	2	1
Shutdown and RHRS pump suction	X-223A through X-223D	-	24	24	4
Core spray pump suction	X-224A, X-224B	-	20	20	2
HPCI pump suction	X-225	-	24	24	1
RCIC pump suction	X-226	-	8	8	1
Core spray test line	X-227A, X-227B	6	10	10	2
Vacuum breaker solenoids	X-228A through X-228D	6	-	10	4
Spare	X-229	15	1	1	1
PCMS suction, Div. I	X-230	15	1 1/2	1 1/2	1
PCMS suction, Div. II	X-231	15	1 1/2	1 1/2	1

[a] See Detroit Edison drawing 6C721-2305.

TABLE 3.8-4 LIST OF SPECIFICATIONS, CODES, AND STANDARDS FOR FERMI 2[a]

Specification Reference number	Specification or Standard Designation	Title	Edition
1A	ACI 318-71	Building Code Requirements for Reinforced Concrete[b]	Feb. 9, 1971
1B	ACI 318-63	Building Code Requirements for Reinforced Concrete[b]	June 1963
2A	ACI 301-72	Specifications for Structural Concrete for Buildings	1972
2B	ACI 301-66	Specifications for Structural Concrete for Buildings	1966
3	ACI 347-68	Recommended Practice for Concrete Formwork	1968
4	ACI 305-72	Recommended Practice for Hot Weather Concreting	1972
5A	ACI 211.1-74	Recommended Practice for Selecting Proportions for Normal Weight Concrete	1974
5B	ACI 211.1-70	Recommended Practice for Selecting Proportions for Normal Weight Concrete	1970
7	ACI 315-65	Manual of Standard Practice for Detailing Reinforced Concrete Structures	1965
8	ACI 306-66	Recommended Practice for Cold Weather Concreting	1966
9	ACI 309-72	Recommended Practice for Consolidation of Concrete	1972
10	ACI 322-72	Building Code Requirements for Structural Plain Concrete	1972
11	ACI 308-71	Recommended Practice for Curing Concrete	1971
12	ACI 212	Guide for Use of Admixtures in Concrete	ACI Journal, Sept. 1971

TABLE 3.8-4 LIST OF SPECIFICATIONS, CODES, AND STANDARDS FOR FERMI 2[a]

Specification Reference number	Specification or Standard Designation	Title	Edition
13	ACI 214-65	Recommended Practice for Evaluation of Compression Test Results in Field Concrete	1965
14	ACI 311-64	Recommended Practice for Concrete Inspection	1964
15	ACI SP-2	Manual of Concrete Inspection	1963
16	ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting and Placing Concrete	1973
17	ACI Committee Report 304	Placing Concrete by Pumping Methods	ACI Journal, May 1971
18	ACI Committee Report 437 Subcommittee	Strength Evaluation of Existing Concrete Structure	Nov. 1967
19	CRSI	Manual of Standard Practice	19th Edition
20	UBC	Uniform Building Code[c]	1970
21A	AISC-69	Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings	Feb. 12, 1969
21B	AISC-63	Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings	1963
22	AISI	Specification for the Design of Light Gage Cold-Formed Steel Structural Members	1968
23A	AWS D1.1-72	Structural Welding Code	1972
23B	AWS D1.0-69	Structural Welding Code	1969
24	AWS D12.1-61	Recommended Practice for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction	1961

TABLE 3.8-4 LIST OF SPECIFICATIONS, CODES, AND STANDARDS FOR FERMI 2[a]

Specification Reference number	Specification or Standard Designation	Title	Edition
25	ASME	1971 ASME Boiler and Pressure Vessel Code, Subsection NE of Section III	Summer of 1972 Addenda
26	ASME	ASME Boiler and Pressure Vessel Code Material Specifications, Part A - Ferrous	1972
27	ASME	ASME Boiler and Pressure Vessel Code, Section XI, "In Service Inspection of Nuclear Reactor Coolant Systems"	
28	ASTM	Annual Books of ASTM Standards	1972
29	ANSI B31.1.0	Standard Code for Pressure Piping, Power Piping	
30	API 620	Specifications for Welded Steel Storage Tanks	Feb. 1970
31	CTI	Standards for the Cooling Tower Institute	
32	NEC	National Electric Code	
33		U.S. Army Corps of Engineers - Regulations with Respect to Dredging and Construction	
34		Steiger Occupation Safety and Health Act	1970
35	Regulatory Guide. 1.10	Mechanical Cadweld Splices in Reinforcing Bars of Concrete Containments	Feb. 1, 1971 (withdrawn July 1981)
36	Regulatory Guide 1.12	Instrumentation for Earthquakes	Feb. 1, 1971
37	Regulatory Guide 1.13	Fuel Storage Facility Design Basis	Oct. 27, 1971
38	Regulatory Guide 1.15	Testing of Reinforcing Bars for Concrete Structures	Oct. 27, 1971 (withdrawn July 1981)

TABLE 3.8-4 LIST OF SPECIFICATIONS, CODES, AND STANDARDS FOR FERMI 2[a]

Specification Reference number	Specification or Standard Designation	Title	Edition
39	Regulatory Guide 1.26	Quality Group Classification and Standards	Mar. 23, 1972
40	Regulatory Guide 1.27	Ultimate Heat Sink	Mar. 23, 1972
41	Regulatory Guide 1.29	Seismic Design Classification	Rev. 3, September 1978
42	Regulatory Guide 1.31	Control of Stainless Steel Welding	Aug. 11, 1972
43	ACI 349-01	Code Requirements for Nuclear Safety Related Concrete Structures	2001
44	Regulatory Guide 1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)	Rev. 2, Nov. 2001
45	ACI 318-05	Building Code Requirements for Reinforced Concrete	2005
46	ACI 318-77 Handbook	Handbook for Building Requirements for Reinforced Concrete	1977

[a] In design and operation, inspection, etc. of structures and other components for Fermi 2, it has been the practice to specify the use of code(s) and the related design guides applicable at the initiation of the design activity. In the course of design process, the use of later editions of the code and/or any supplements issued thereto has been allowed.

[b] Appendix A adopted for seismic design.

[c] Official building code of Frenchtown Township.

TABLE 3.8.5 DRYWELL LOADING

LOADS	Construction OBE	Construction SSE	Overload Test	Initial Leak Rate Test	Operating OBE	Operating SSE	Accident OBE	Accident SSE	Flooded OBE	Flooded SSE	Refueling OBE	Refueling SSE
Loading case no.	1	2	3	4	5	6	7	8	9	10	11	12
Dead load, vessel and attachment	x	x	x	x	x	x	x	x	x	x	x	x
Pressure, positive and negative		x	x	x	x	x	x	x				
Contained air			x	x								
Wind load	x	x	x	x								
Seismic	x	x			x	x	x	x	x	x	x	x
Vent thrusts			x	x	x	x	x	x				
Weld pads:												
Dead load	x	x	x	x	x	x	x	x	x	x	x	x
Live load					x	x					x	x
Jet forces							x	x				
Temporary pressure or unrelieved deflection due to concrete load					x	x	x	x				
Equipment support loads				x	x	x	x	x	x	x	x	x
Weight and/or restraint of compressible material					x	x	x	x	x	x	x	x
Personnel lock:												
Dead load	x	x	x	x	x	x	x	x	x	x	x	x
Live load					x	x					x	x
Equipment hatch:												
Dead load	x	x	x	x	x	x	x	x	x	x	x	x
Live load					x	x					x	x
Refueling seal loads					x	x					x	x
Water on refueling seals											x	x
Hydrostatic pressure due to flooding									x	x		

TABLE 3.8-6 SUPPRESSION CHAMBER LOADING

LOADS	Construction OBE	Construction SSE	Overload Test	Initial Leak Rate Test	Operating OBE	Operating SSE	Accident OBE	Accident SSE	Flooded OBE	Flooded SSE
Loading case no.	1	2	3	4	5	6	7	8	9	10
Dead load, vessel and attachments	x	x	x	x	x	x	x	x	x	x
Suppression pool water			x	x	x	x	x	x	x	x
Pressure:										
Positive										
Negative			x	x	x	x	x	x	x	x
Seismic	x	x			x	x	x	x	x	x
Vent thrusts			x	x			x	x		
Contained air			x	x						
Temporary concrete loads	x									
Suppression chamber spray header full of water	x	x	x	x	x	x	x	x		
Jet forces on downcomer pipes							x	x		
Live load on catwalks and platforms	x				x	x	x	x	x	x
Weld pads:										
Dead load	x	x	x	x	x	x	x	x	x	x
Live load					x	x				

Note: The operating and accident loads have been supplemented and/or modified according to References 1 and 3.

TABLE 3.8-7 LOADS AND LOADING COMBINATIONS FOR THE DRYWELL VENT PENETRATIONS

Design Category	Load Combination	Stress Comparison[a]
I	E_s	$P_m \leq S_m$ @ T_d
		$P_1 + P_b \leq 1.5\ S_m$ @ T_d
II	$R_p + R_t$	$P_m \leq S_m$ @ T_a
		$P_1 + P_b \leq 1.5\ S_m$ @ T_a
III	$R_p + P_t + E_s$ + Design	$P_1 \leq 1.1\ S_m$ @ T_a
		$P_1 + P_b \leq 1.5\ S_m$ @ T_a

E_s = Maximum seismic reaction due to SSE "g" loads on vent header

R_p = Piping reaction due to maximum accident pressure expansion between drywell and suppression pool.

R_t = Piping reaction due to maximum accident thermal expansion between drywell and suppression pool

Design = Maximum general stresses calculated at vent penetration in Table 3.8-5

T_d = Design temperature

T_a = Temperature associated with design accident

[a] Stress nomenclature in accordance with the ASME B&PV Code Section III.

TABLE 3.8-8 LOADS AND LOADING COMBINATIONS FOR THE DRYWELL SPHERICAL EMBEDMENT

Design Category	Loading Combination	Stress Comparison[a]
I	Design + P_a + T_a	$P_1 \leq 1.5\ S_m$ @ T_a
		$P_1 + P_b + Q \leq 3.0\ S_m$ @ T_a

P_a = Pressure loading due to design accident

T_a = Temperature corresponding to design accident

Design = Maximum general shell stress due to loading in Table 3.8-5 for the specific location of the spherical embedment

[a] Stress nomenclature in accordance with the ASME B&PV Code Section III.

TABLE 3.8-9 LOADS AND LOADING COMBINATIONS FOR THE DRYWELL KNUCKLE REGION (ACCIDENT LOADS)

Design Category	Loading Combinations	Stress Comparison[a]
I	Design + P_a	$P_1 \leq S_m$ @ T_a
		$P_1 + P_b \leq 1.5\ S_m$ @ T_a

P_a = Operating pressure associated with design accident condition

Design = Maximum general shell stress in the knuckle region calculated from Table 3.8-5 for condition 8

T_a = Temperature associated with design accident

[a] Stress nomenclature in accordance with the ASME B&PV Code Section III.

TABLE 3.8-10 LOADS AND LOADING COMBINATIONS FOR THE DRYWELL CONE AND TOP HEAD

Design Category	Loading Combinations	Stress Comparison[a]
I	Design P_a	$P_1 \leq S_m$ @ T_a
		$P_1 + P_b \leq 1.5\ S_m$ @ T_a

P_a = Operating pressure associated with design accident condition

T_a = Temperature associated with design accident

[a] Stress nomenclature in accordance with the ASME B&PV Code Section III.

TABLE 3.8-11 LOADS AND LOADING COMBINATIONS FOR THE TOP DRYWELL FLANGE

Design Category	Loading Combination	Stress Comparison[a]
Operating		
I	$D_b + P_o + W_s$	$P_m \leq S_m$ @ T_d
II	$D_b + P_v + W_s$	$P_1 + P_b \leq 1.5\ S_m$ @ T_d
Refueling		
I	$D_b + W_s$	$P_m \leq S_m$ @ T_d
II	Ws	$P_1 + P_b \leq 1.5\ S_m$ @ T_d
Accident		
I	$D_b + P_a$	$P_m \leq S_m$ @ T_a
II	$D_b + P_v$	$P_1 + P_b \leq 1.5\ S_m$ @ T_a

D_b = Design bolting load calculated for gasket seating and for internal pressure

P_o = Design pressure during normal operation

P_v = Design vacuum pressure

P_a = Design accident pressure

W_s = Design loads due to the weight of the water seal, together with loads imposed by the expansion bellows

T_d = Design temperature

T_a = Temperature associated with design accident

[a] Stress nomenclature in accordance with the ASME B&PV Code Section III.

TABLE 3.8-12 LOADS AND LOADING COMBINATIONS FOR THE EQUIPMENT HATCH

PRESSURE-RETAINING PARTS

Design Category	Loading Combination	Stress Comparison[a]
I	P_a	$P_m \le S_m$ @ T_a
		$P_l + P_b \le 1.5\ S_m$ @ T_a
II	$P_a + R_a$	$P_m \le 0.9\ S_y$ @ T_{jet}

P_a = Accident pressure load

R_a = Jet force associated with pipe rupture

T_a = Temperature associated with design accident

T_{jet} = Temperature at the jet

STRUCTURAL PARTS

Design Category	Loading Combination	Stress Comparison[a]								
		Tension		Bending		Bearing		Shear		Weld
		P_L	Bolt	P_L	Bolt	P_L	Bolt	P_L	Bolt	
I	$D + L + P_a$	S_m	S_m	1 1/2 S_m	1 1/2 S_m	1.6 S_m	1.6 S_m	.4 F_y	.8 S_m	.8 S_m
		(17.5)	(25)	(26.25)	(37.5)	(28)	(40)	(14.5)	(16)	(14)
II	$D + L + P_a + R_j$	.9 F_y	.9 F_y	.9 F_y	[b]1.5 F_y	1.33 x .9 F_y	.9 F_y	.8 F_y	.8 F_y	.8 F_y
		(30.3)	(90)	(30.3)	(150)	(40.4)	(90)	(29)	(80)	(27)

D = Dead load

L = Live load

P_a = Accident pressure load

R_j = Reaction forces due to jet force R_a on cover

[a] Stress nomenclature in accordance with the ASME B&PV Code Section III.

[b] Maintain less than F_y in combination with shear.

TABLE 3.8-13 LOADS AND LOADING COMBINATIONS FOR THE DRYWELL BEAM SEATS

STRUCTURAL PARTS

Design Category	Loading Combination	Stress Comparison[a]
I	$D + L + E_o$	Per AISC Specification
II	$D + L + E_s$	1.33 Allowable Increase for Design II

PRESSURE–RETAINING PARTS

Design Category	Loading Combination	Stress Comparison[a] Shell	Weld
I	$D + L + E_o + P_a$	$P_1 \leq 1.5\ S_m$ @ T_a	
		$P_1 + P_b \leq 1.5\ S_m$ @ T_a	
II	$D + L + E_o + P_o$	Buckling Allowable per WRC 69[b]	$0.55\ S_m$ @ T_a
III	$D + L + E_s + P_a$	$P_1 \leq 1.5\ S_m$ @ T_a	
		$P_1 + P_b \leq 1.5\ S_m$ @ T_a	
IV	$D + L + E_s + P_o$	Buckling Allowable per WRC 69[b]	$0.55\ S_m$ @ T_a

D = Dead load

L = Live load

E_o = Operating–basis earthquake

E_s = Safe–shutdown earthquake

P_a = Operating pressure associated with design accident condition

P_o = Operating pressure associated with normal operating

T_a = Temperature associated with design accident

[a] Stress nomenclature in accordance with ASME B&PV Code Section III.

[b] Welding Research Council Standard 69.

TABLE 3.8-14 LOADS AND LOADING COMBINATIONS FOR THE SPRAY HEADER AND VENT JET DEFLECTOR

STRUCTURAL (NON-PRESSURE RETAINING) PARTS

Design Category	Load Combination	Stress Comparison[a]
I	R_a	Per AISC Specification with 1.33 Allowable Increase

PRESSURE–RETAINING PARTS

Design Category	Load Combination	Stress Comparison[a]
I	R_a	$P_1 \leq 1.1\ S_m$ @ T_a
		$P_1 + P_b \leq 1.5\ S_m$ @ T_a

R_a = Jet force associated with pipe rupture

T_a = Temperature associated with design accident

[a] Stress nomenclature in accordance with the ASME B&PV Code Section III.

TABLE 3.8-15 LOADS AND LOADING COMBINATIONS FOR THE DRYWELL STABILIZER CONNECTION

Design Category	Loading Combination	AISC (Membrane) Plate	AISC (Membrane) Weld	Stress Comparison[a] AISC (Bending) Plate	ASME (Membrane) Plate	ASME (Membrane) Weld	ACI
I	$D + L + E_o$	$P_m \leq .5\ F_y$	$P_m \leq 15800$	$P_1 + P_b \leq .5\ F_y$	$P_m \leq .5\ S_y$	13,600	Concrete Bearing Stresses in Accordance with ACI 318-71
II	$D + L + R_j + E_o$	$P_m \leq F_y$	$P_m \leq .8\ F_y$	$P_1 + P_b \leq F_y$	$P_m \leq S_y$	$.8\ S_y$	
III	$D + L + R_j + E_s$	$P_m \leq F_y$	$P_m \leq .8\ F_y$	$P_1 + P_b \leq F_y$	$P_m \leq S_y$	$.8\ S_y$	
IV	$D + L + R_f + E_s$	$P_m \leq F_y$	$P_m \leq .8\ F_y$	$P_1 + P_b \leq F_y$	$P_m \leq S_y$	$.8\ S_y$	

D = Dead load

L = Live load

E_o = Operating–basis earthquake

E_s = Safe–shutdown earthquake

R_j = Reaction force due to jet force R_a

R_f = Reaction force due to flooding

[a] Stress nomenclature in accordance with the ASME B&PV Code Section III.

TABLE 3.8-16 LOADS AND LOADING COMBINATIONS FOR THE DRYWELL SKIRT

Design Category	Loading Combinations	Stress Comparison
I	D + W	Per AISC Specification
II	D + E_s	Per AISC Specification
III	D + T	Per AISC Specification

D = Dead load

E_s = Safe–shutdown earthquake

T = Test condition

W = Design loads due to wind

TABLE 3.8-17 LOADS AND LOADING COMBINATIONS FOR PENETRATIONS

Design Category	Loading Combination	Stress Comparison[a]
I	$P_d + R_o$	$P_1 \leq 1.1\ S_m$ @ T_d
		$P_1 + P_b \leq 1.5\ S_m$ @ T_d

P_d = Pressure associated with design condition

R_o = Maximum piping reaction due to operating, accident, test, or flooding

T_d = Design temperature

[a] Stress nomenclature in accordance with the ASME B&PV Code Section III.

TABLE 3.8-18 LOADING COMBINATIONS FOR STEEL STRUCTURES, ELASTIC DESIGN[a]

Load Combination Category		Load Condition No.	Overall Loading Equation[b,c,d]
I	Construction	1	$F = 1.0(D + L + C + W + T_o)$
		2	$F = 1.0(D + L + S + C + T_o)$
II	Test	3	$F = 1.0(D + L + S + C + R_o + T_o)$
III	Normal	4	$F = 1.0(D + L + S + C + R_o + T_o)$
IV	Severe environmental	5	$F = 1.0(D + L + C + R_o + E_o + T_o)$
		6	Deleted
		7	$F = 1.0(D + L + C + R_o + W + T_o)$
V	Abnormal	8	$F = 1.0(D + L + S + C + R_a + T_a + P_a)$
		9	$F = 1.0(D + L + S + C + R_o + T_o + M)$
VI	Extreme environmental	10	$F = 1.0(D + L + C + R_o + E_s + T_o)$
		11	$F = 1.0(D + L + R_o + W_t + T_o)$
		12	$F = 1.0(D + L + C + R_o + T_o + H)$
VII	Abnormal/severe environmental	13	$F = 1.0(D + L + C + R_a + E_o + T_a + P_a + Y_r + Y_j + Y_m)$
VIII	Abnormal/extreme environmental	14	$F = 1.0(D + L + C + R_a + E_s + T_a + P_a + Y_r + Y_j + Y_m)$

[a] Loads not applicable to a particular system under consideration may be deleted. If for any load combination the effect of any load other than D reduces the load, it will be deleted from the combination. For both E_s and E_o, the resultant effects (resultant stresses) at both horizontal and vertical earthquake components shall be determined by combining the individual effects by the square root of the sum of the squares. This procedure also applies when combining the dynamic effects of W_t, M, R_a, and P_a.

[b] F = Working load.

[c] See Table 3.8-21 for definition of terms.

[d] For allowable stresses, see Table 3.8-19.

TABLE 3.8-19 ALLOWABLE DESIGN STRESSES

	Loading Condition	Structural Steel F_y = 36,000 psi: Tension (F_t) and Bending (F_b)	Structural Steel F_y = 36,000 psi: Compression (F_a)	Concrete* f'_c = 4000 psi, n = 8: Basic Design Stress	Concrete* f'_c = 4000 psi, n = 8: Compression Stress	Reinforcing Bars F_y = 60,000 psi: % F_y	Reinforcing Bars F_y = 60,000 psi: Tensile Stress F_s
A	Dead load (D.L.) + live load (L.L.)	F_t and F_b from Section 1.5, Appendix A, 1969 AISC Specification	F_a from Table 1-36 of 1969 AISC Specification	From ACI 318-63	$f_c = 0.45\ f'_c$	0.4 F_y	24,000 psi
B	D.L. + L.L. + OBE	F_t and F_b from Section 1.5, Appendix A, 1969 AISC Specification	F_a from Table 1-36 of 1969 AISC Specification	From ACI 318-63	$f_c = 0.45\ f'_c$	0.4 F_y	24,000 psi
C	D.L. + 0.50 L.L. + OBE + forces due to thermal expansion and snubber loads	F_t and F_b from Section 1.5, Appendix A, 1969 AISC Specification	F_a from Table 1-36 of 1969 AISC Specification	From ACI 318-63	$f_c = 0.45\ f'_c$	0.4 F_y	24,000 psi
D	Case B, except SSE instead of OBE	$F_t = F_b = F_y$	1.67 x Case A	1.67 x Case A	$f_c = 0.85\ f'_c$ maximum	1.67 x Case A	0.9 F_y maximum = 54,000 psi
E	Case C, except SSE instead of OBE	$F_t = F_b = F_y$	1.67 x Case A	1.67 x Case A	$f_c = 0.85\ f'_c$	1.67 x Case A	0.9 F_y maximum = 54,000 psi
F	D.L. + L.L. + basic design wind	1.33 x Case A	1.33 x Case A	1.33 x Case A	$f_c = 0.60\ f'_c$	1.33 x Case A = 0.53 F_y	31,800 psi
G	D.L. + L.L. + tornado wind design or maximum	$F_t = F_b = F_y$	1.67 x Case A	1.67 x Case A	$f_c = 0.85\ f'_c$ maximum	1.67 x Case A	0.9 F_y maximum = 54,000 psi

D.L. = Dead load of structure and equipment plus any other permanent loads, such as soil or hydrostatic loads or operating pressure.

*Concrete with 4000 psi specified compressive strength (f'_c) is generally used in Fermi 2. The use of other grades of concrete or use of higher strength for the same grade of concrete based on the time-strength relationship has been noted in corresponding design document package.

TABLE 3.8-20 LOADING COMBINATIONS FOR REINFORCED CONCRETE STRUCTURES, ULTIMATE STRENGTH DESIGN[a]

Load Combination Category		Load Condition No.	Overall Loading Equation[b,c,d,e]
I	Construction	1	$u = 1.3(D + L + C + W + T_o)$
II	Normal	2	$u = 1.4(D + R_o) + 1.7(L + C) + 1.3T_o$
III	Test	3	$u = 1.1(D + R_o) + 1.3(L + C + T_o)$
IV	Severe environmental	4	$u = 1.4(D + R_o) + 1.7(L + C + W) + 1.3T_o$
		5	$u = 1.2(D + R_o) + 1.7W + 1.3T_o$
		6	$u = 1.4(D + R_o) + 1.7(L + C) + 1.9E_o + 1.3T_o$
		7	$u = 1.2(D + R_o) + 1.9E_o + 1.3T_o$
		8	Deleted
V	Abnormal	9	$u = 1.0(D + L + C + R_a + T_a) + 1.5P_a$
		10	$u = 1.0(D + L + C + R_o + T_o + M)$
VI	Extreme environmental	11	$u = 1.0(D + L + C + R_o + Es + T_o)$
		12	$u = 1.0(D + L + R_o + W_t + T_o)$
		13	$u = 1.0(D + L + C + R_o + T_o + H)$
VII	Abnormal/severe environmental	14	$u = 1.0(D + L + C + R_a + T_a + Y_r + Y_j + Y_m) + 1.25(E_o + P_a)$
VIII	Abnormal/extreme environmental	15	$u = 1.0(D + L + C + R_a + E_s + T_a + P_a + Y_r + Y_j + Y_m)$

[a] Loads not applicable to a particular system under consideration may be deleted. If for any load combination the effect of any load other than D reduces the load, it will be deleted from the combination. For both E_s and E_o, the resultant effects (resultant stresses) at both horizontal and vertical earthquake components shall be determined by combining the individual effects by the square root of the sum of the squares. This procedure also applies when combining the dynamic effects of W_t, M, R_a, and P_a.

[b] u = Ultimate load.

[c] See Table 3.8-21 for definition of terms.

[d] Allowable stresses shall be according to ACI 318-71.

[e] Allowable loads shall be according to ACI 349-01 & RG 1.142, Rev. 2 for the second set of Category I underground ductbanks, manholes and above ground cable vaults.

TABLE 3.8-21 NOTES FOR TABLE 3.8-18 AND TABLE 3.8-20

C = Crane-lifted load.

D = Dead load of the structure plus any other permanent load except prestressing forces, including vertical and lateral pressures of liquids, piping, cable pan, self weight of crane, and weight of permanent equipment and its normal contents under operating and test conditions.

E_o = Operating-basis earthquake (OBE) including dynamic lateral soil pressure and hydrodynamic ground-water pressure

Horizontal ground acceleration = 0.08g

Vertical ground acceleration = 66-2/3 percent of the horizontal acceleration where g = 32.2 ft/sec^2.

E_s = Safe-shutdown earthquake (SSE) including dynamic lateral soil pressure and hydrodynamic ground-water pressure

Horizontal ground acceleration = 0.15g

Vertical ground acceleration = 66-2/3 percent of the horizontal ground acceleration where g = 32.2 ft/sec^2.

H = Forces associated with the maximum probable flood or seiches (see Section 3.4).

L = Conventional floor and roof live loads, movable equipment loads, and other loads that vary in intensity, such as lateral soil pressure. Live load intensities may vary from zero to their maximum values to determine the most critical effect upon the structure for the load combination under consideration.

Note: Reduced intensities of live loads such as conventional floor loads may be associated with accident or extreme environmental conditions.

TABLE 3.8-21 NOTES FOR TABLE 3.8-18 AND TABLE 3.8-20

M = Loads associated with both internal and external missiles (see Section 3.5).

P_a[(a)] = Pressure load caused by a postulated pipe break accident. Containment design accident pressures based upon peak calculated pressure with appropriate margin provided for uncertainties are:

Internal pressure	56 psig
External pressure	2 psig

P_o = Design pressure during normal operating condition. Containment design normal pressures are:

Internal pressure	2 psig
External pressure	atmospheric

P_t = Containment test pressure:

Internal pressure	70 psig
External pressure	atmospheric

R_a = Pipe reactions due to postulated break accident including R_o.

R_o= Normal operating reactions of piping at supports or anchor points.

S = Stability load.

T_a[(a)] = Thermal loads generated by postulated break accident including T_o. Containment temperatures associated with a design accident are:

Internal temperature of the suppression chamber	281°F
Internal temperature of the drywell	340°F
Minimum external temperature	50°F

TABLE 3.8-21 NOTES FOR TABLE 3.8-18 AND TABLE 3.8-20

T_o= Thermal effects associated with normal, construction, and test conditions:

(a) Climatic temperature ranges:

Maximum outside temperature	102°F
Minimum outside temperature	-18°F

(b) Operating temperature ranges:

Ambient temperature inside the reactor/auxiliary building and RHR complex	70°F

u = Ultimate load capacity as defined by ACI 318-71 [b]

W = Design wind load (see Subsection 3.3.1)

W_t = Tornado load (see Subsection 3.3.2)

Y_j = Jet impingement equivalent static load.

Y_m = Missile impact equivalent static load.

Y_r = Equivalent static reaction load from high-energy line break

---

[a] Since these loads are time dependent, their effects will be superimposed accordingly.

[b] Use ACI 349-01 for the second set of Category I underground ductbanks, manholes and above ground cable vaults.

TABLE 3.8-22 FACTORS OF SAFETY FOR CATEGORY I FOUNDATIONS

Category I Structure	Load Combinations				
	D+H+E	D+H+W	D+H+E’	D+H+W_t	D+F'
Reactor and aux. bldg.	1.92	26.70	1.32	2.85	1.76
RHR complex	1.78	22.20	1.43	2.43	1.10*

*The factor of safety for flotation of the RHR complex is computed based on the reservoirs totally empty

where

D = dead loads or their related internal moments and forces, including any permanent equipment loads and hydrostatic loads

E = loads generated by the operating-basis earthquake

W = loads generated by the design wind specified for the plant

E’ = loads generated by the safe-shutdown earthquake

W_t = loads generated by the design tornado specified for the plant; tornado loads include loads due to the tornado wind pressure, the tornado-created differential pressure, and tornado-generated missiles

H = lateral earth pressure

F’ = buoyant force of the probable maximum flood (PMF)

Figure Intentionally Removed
Refer to Plant Drawing C-2407

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FIGURE 3.8-1

TYPICAL DETAIL OF THE DRYWELL FLOOR CONNECTION TO THE DRYWELL SUPPORT PEDESTAL

Figure Intentionally Removed
Refer to Plant Drawing C-2304

Fermi 2

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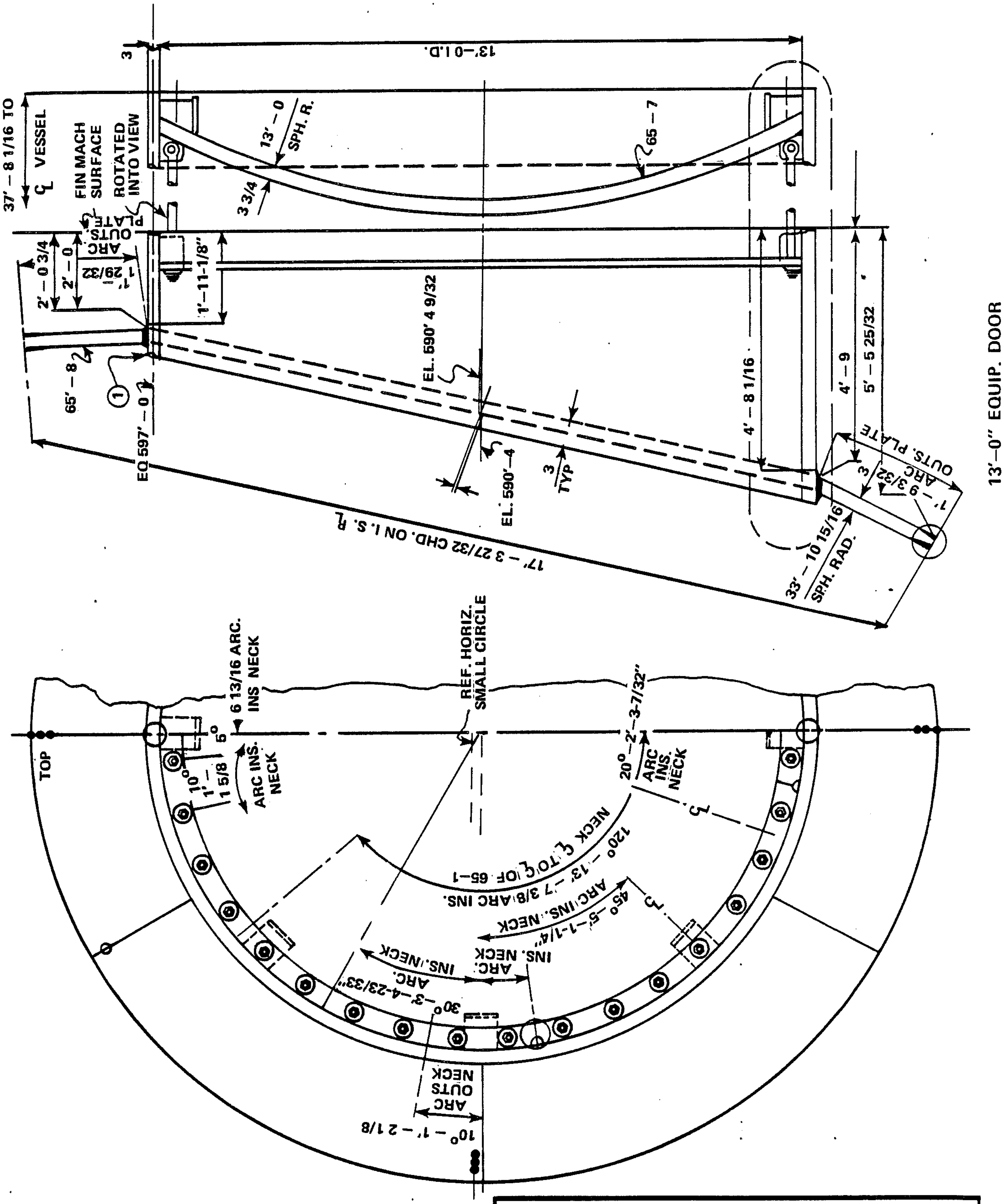
FIGURE 3.8-2

DEVELOPED VIEW OF THE DRYWELL



CHICAGO BRIDGE & IRON COMPANY
DRAWING NO. 64, REV. G

Fermi 2

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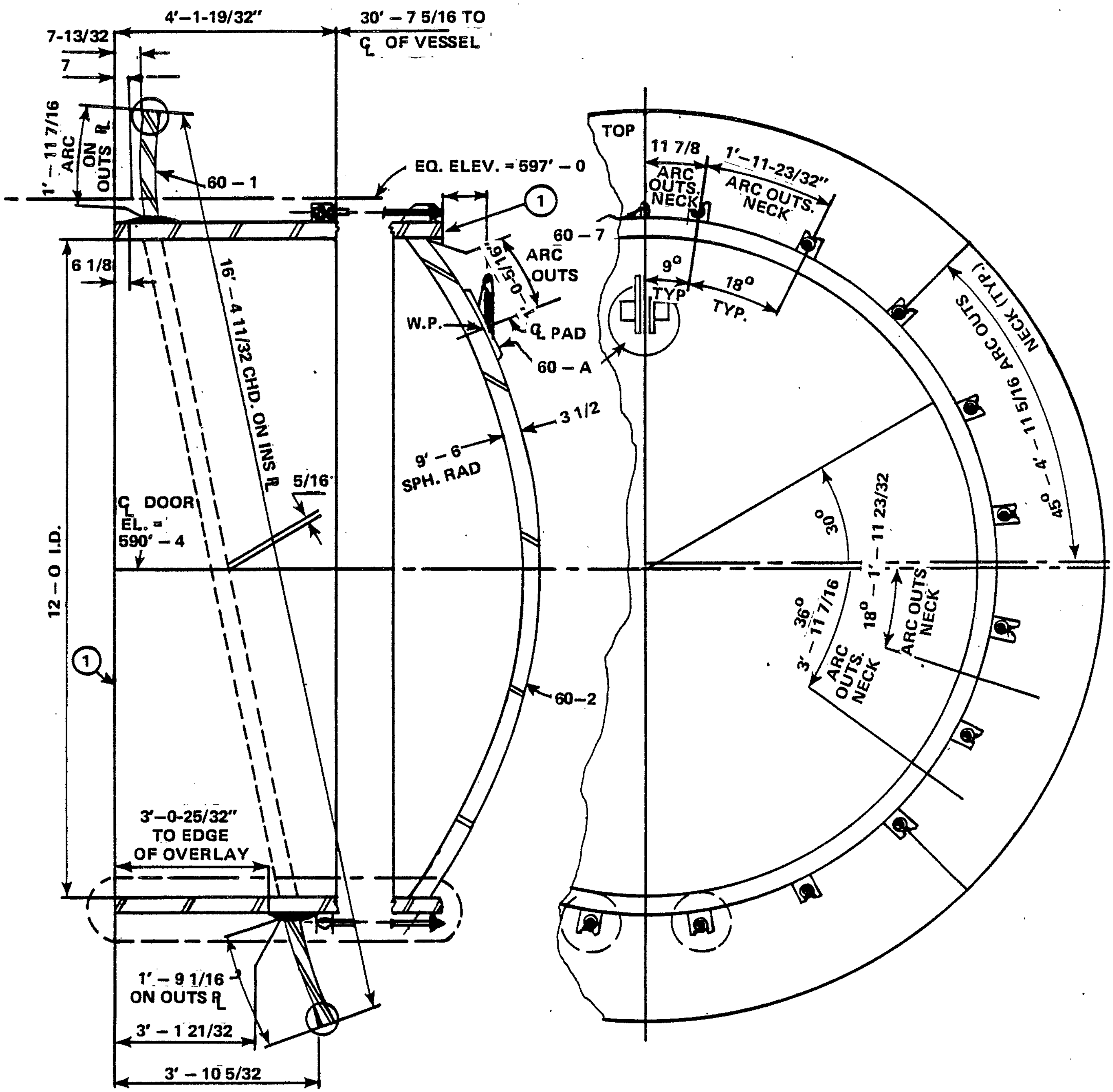
FIGURE 3.8-3, SHEET 1

PRIMARY CONTAINMENT EQUIPMENT HATCH



12'-0" EQUIP. DOOR

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FIGURE 3.8-3, SHEET 2

PRIMARY CONTAINMENT EQUIPMENT HATCH

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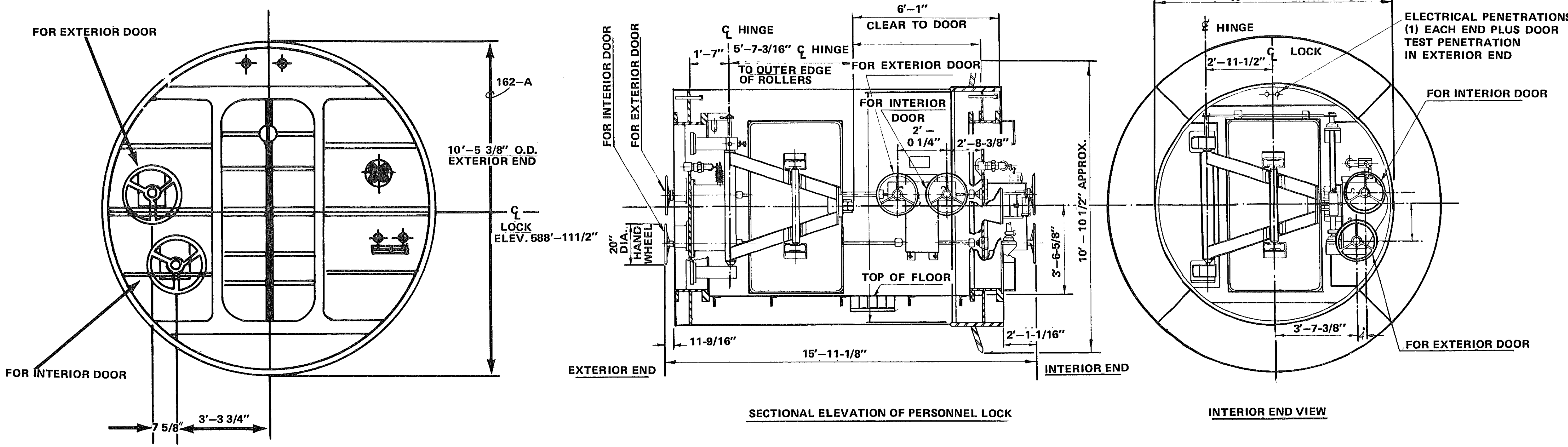
DRAWING NO. 59, REV. F



CHICAGO BRIDGE & IRON COMPANY DRAWING NO. 100, REV. B

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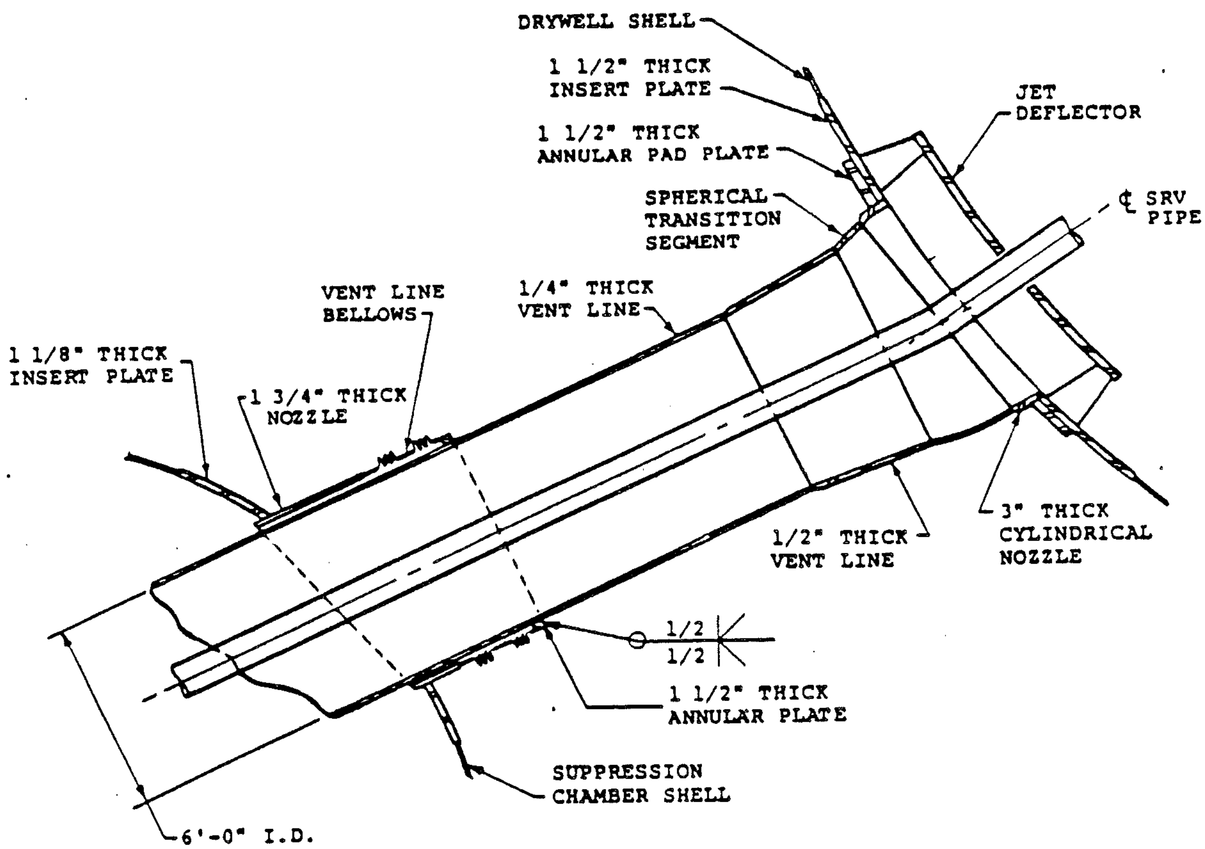
FIGURE 3.8-4

PRIMARY CONTAINMENT PERSONNEL HATCH



NUTECH DRAWING NO. DET-04-028-3. REV. 0
PUAR FIGURE 3-2.1-7

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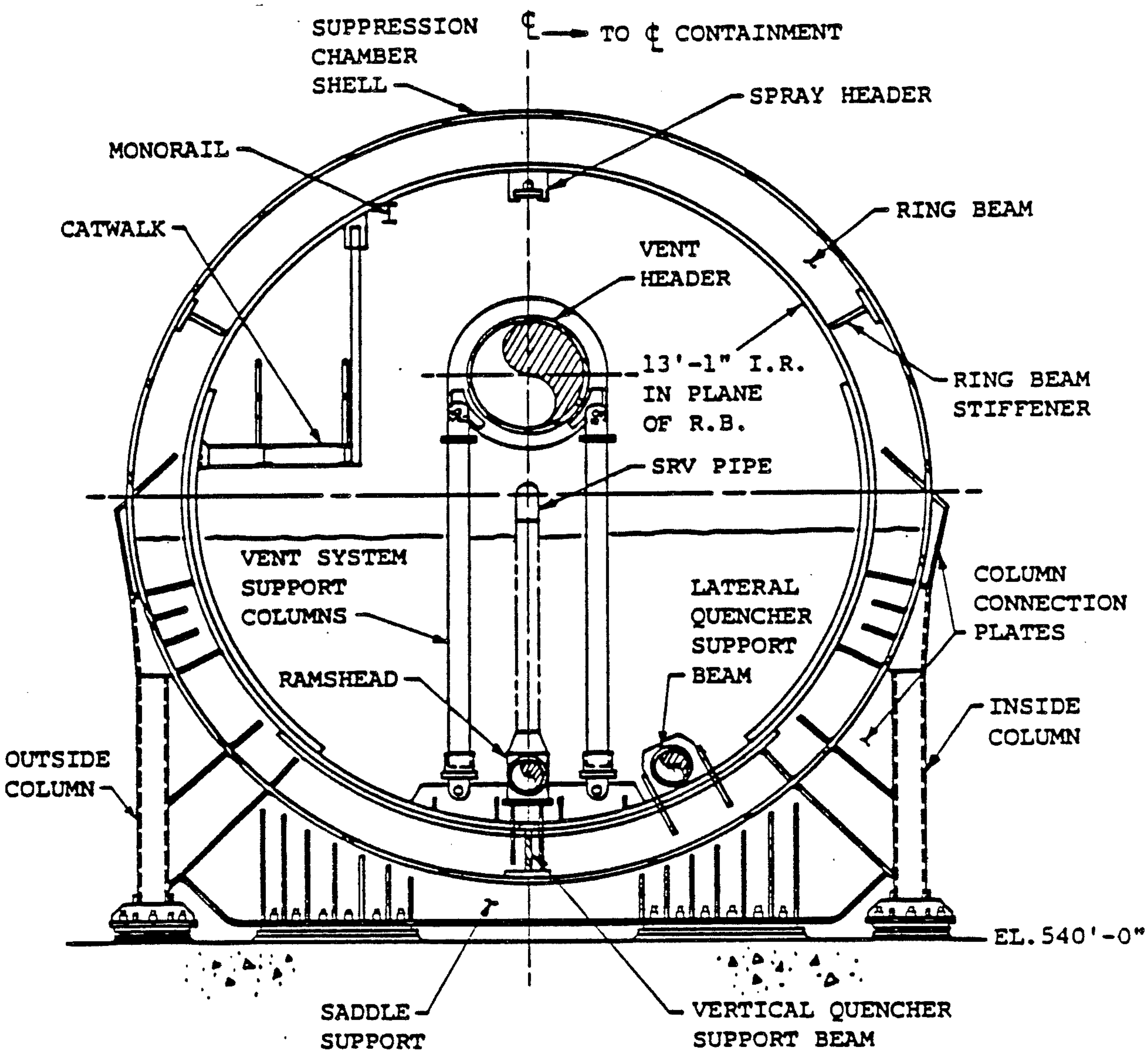
FIGURE 3.8-5

VENT LINE DETAILS – UPPER END



NUTECH DRAWING NO. DET-04-028-1, REV. 0
PUAR FIGURE 5-1-2.1-4

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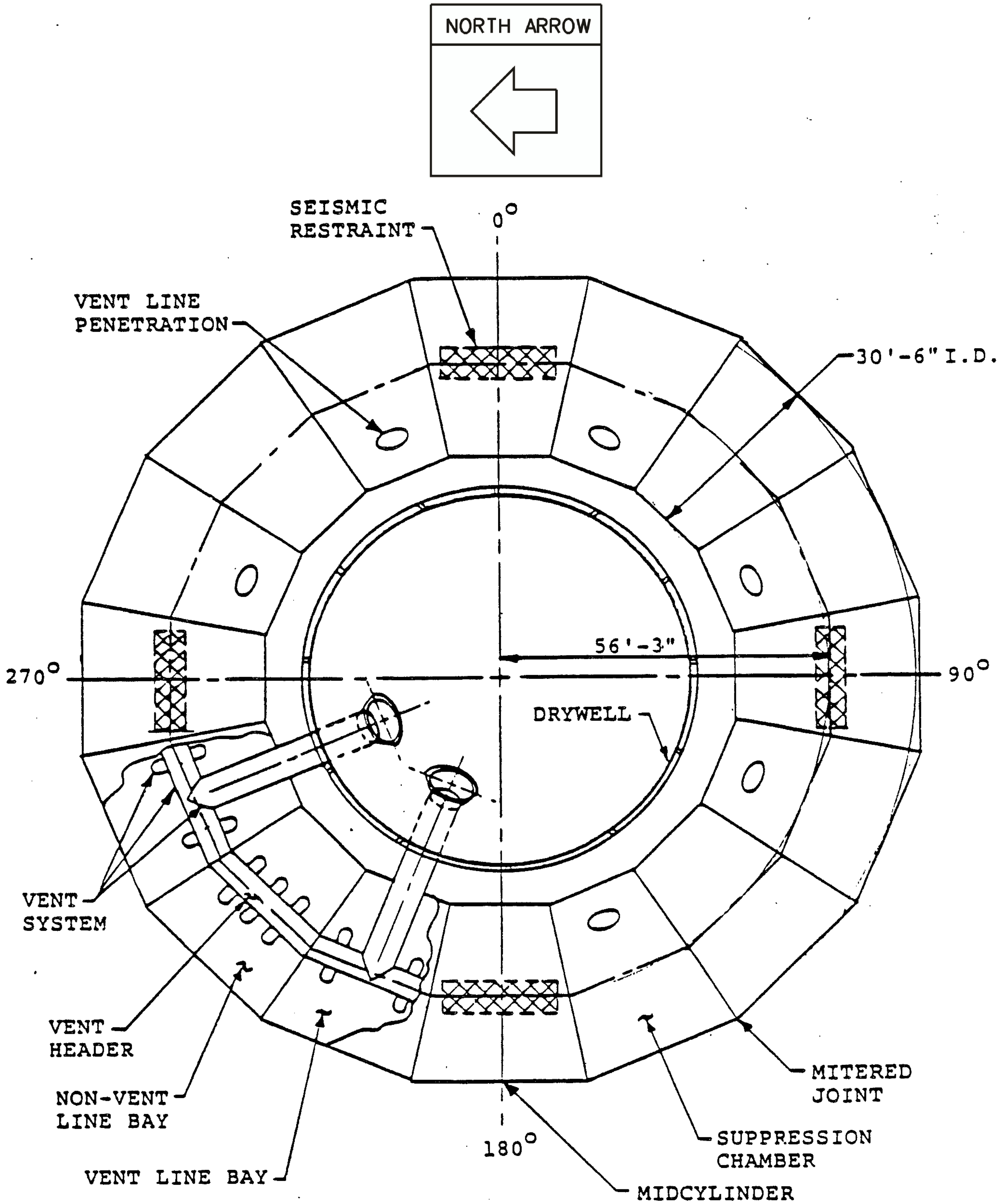
FIGURE 3.8-6

SUPPRESSION CHAMBER SUPPORT DETAILS



NUTECH DRAWING NO. DET-04-028-2, REV. 0
PUAR FIGURE 2-2.1-1

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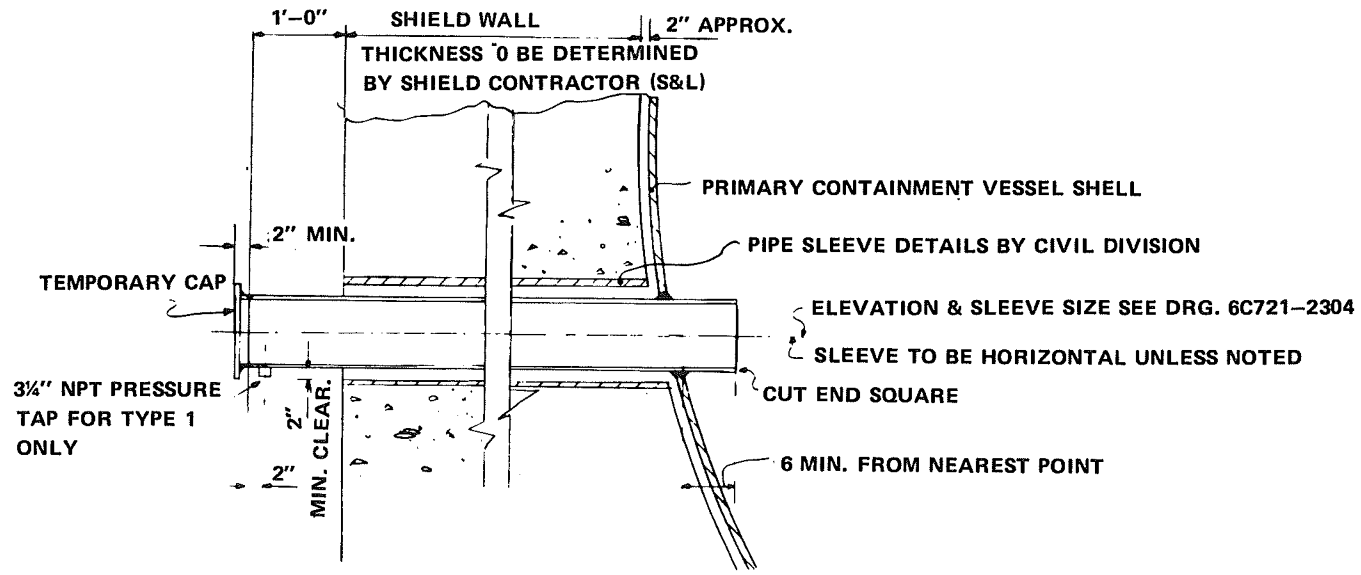
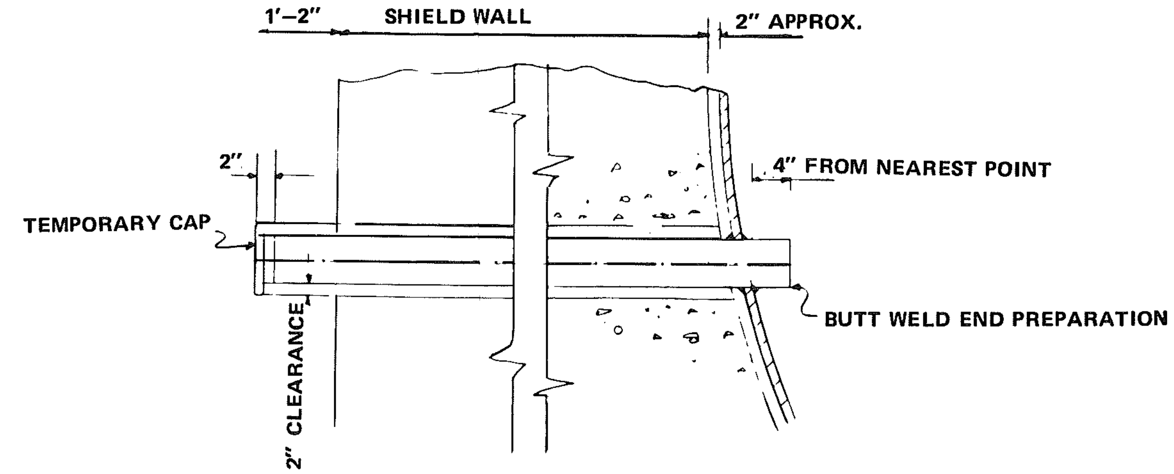
FIGURE 3.8-7

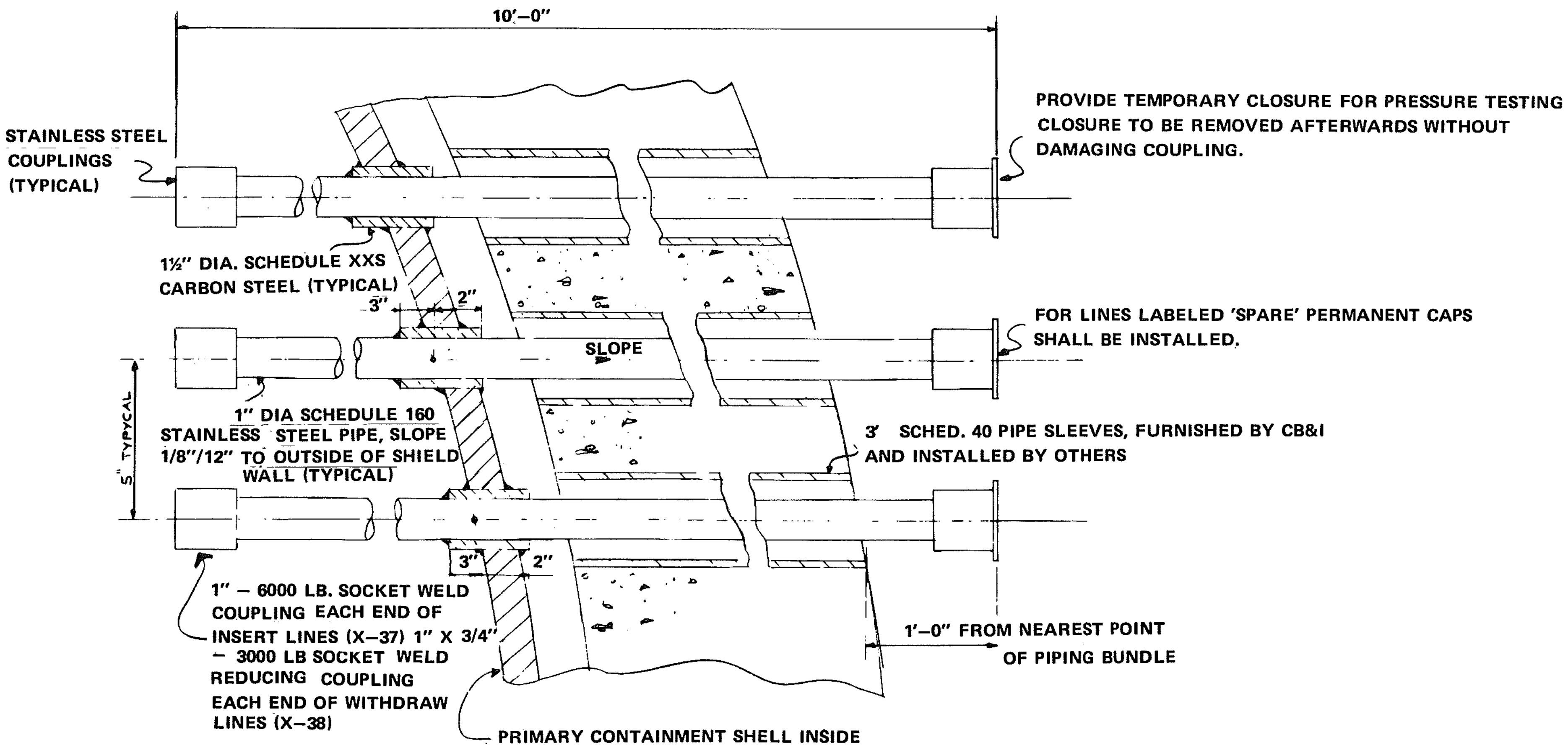
SUPPRESSION CHAMBER PLAN



PENETRATION NOZZLE DETAIL
TYPE 1 & 2



PENETRATION NOZZLE DETAIL
TYPE 3



PENETRATION NOZZLE DETAIL
TYPE 6

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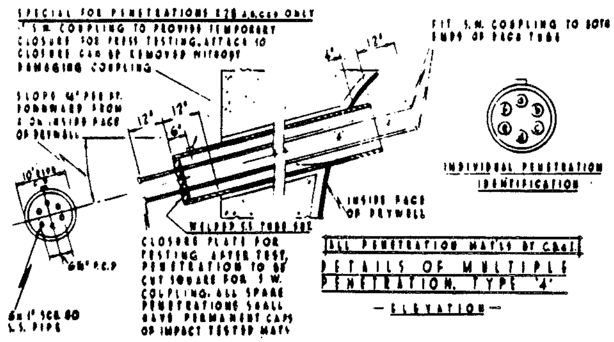
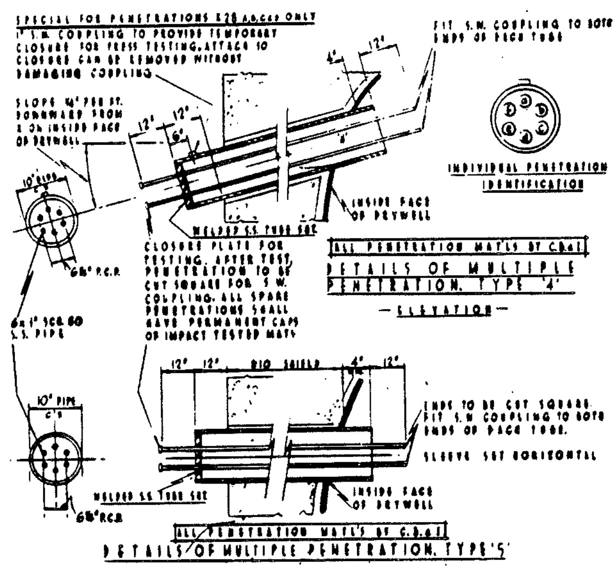
FIGURE 3.8-8, SHEET 1

DRYWELL PENETRATION TYPES

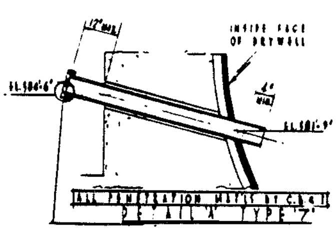


TYPE 4

NOTE: MATERIALS & WELD DETAILS SHALL MEET THE REQUIREMENTS OF ASME–3PVC SECTION II PARAGRAPH N–1333 OF LATEST ISSUE



TYPE 5



TYPE 7

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FIGURE 3.8–8, SHEET 2

DRYWELL PENETRATION TYPES

Figure Intentionally Removed
Refer to Plant Drawing M-2501

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FIGURE 3.8-9

PRIMARY CONTAINMENT SYSTEM PROCESS LINE SLEEVED PENETRATIONS

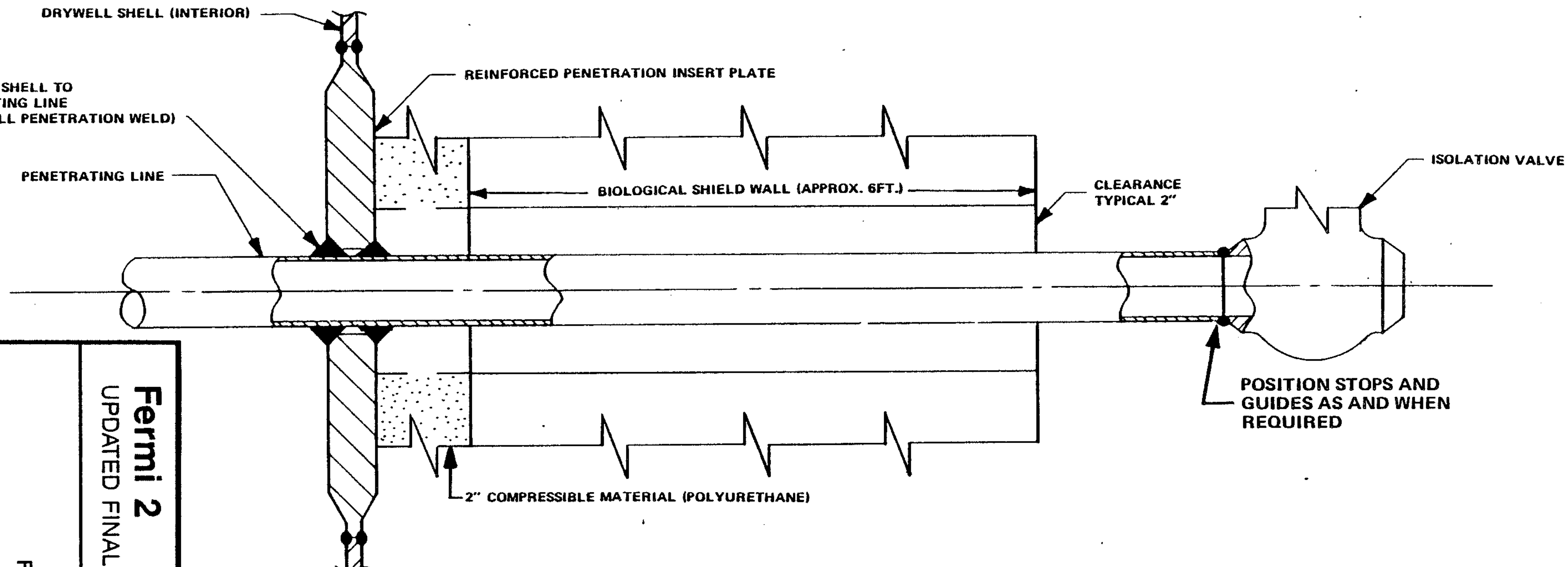

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FIGURE 3.8-10

PRIMARY CONTAINMENT SYSTEM PROCESS LINE UNSLEEVED PENETRATION ASSEMBLY

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Refer to Plant Drawing M-2502

FIGURE 3.8-11

PRIMARY CONTAINMENT SYSTEM PROCESS LINE UNSLEEVED PENETRATIONS

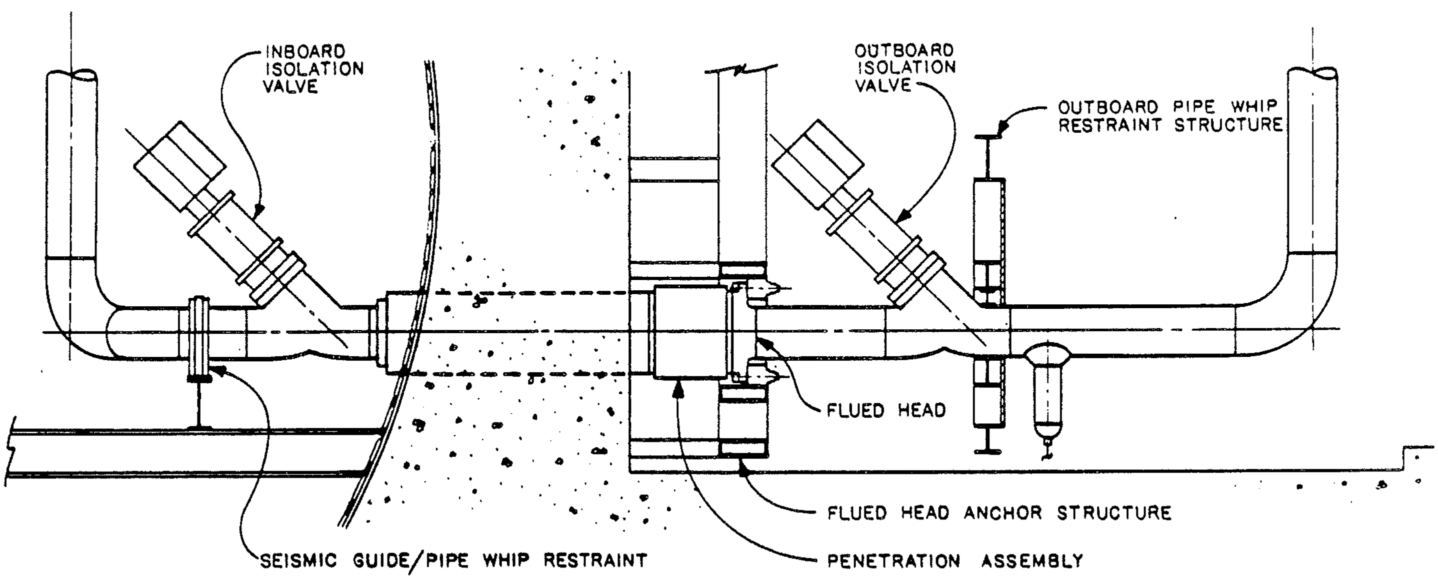

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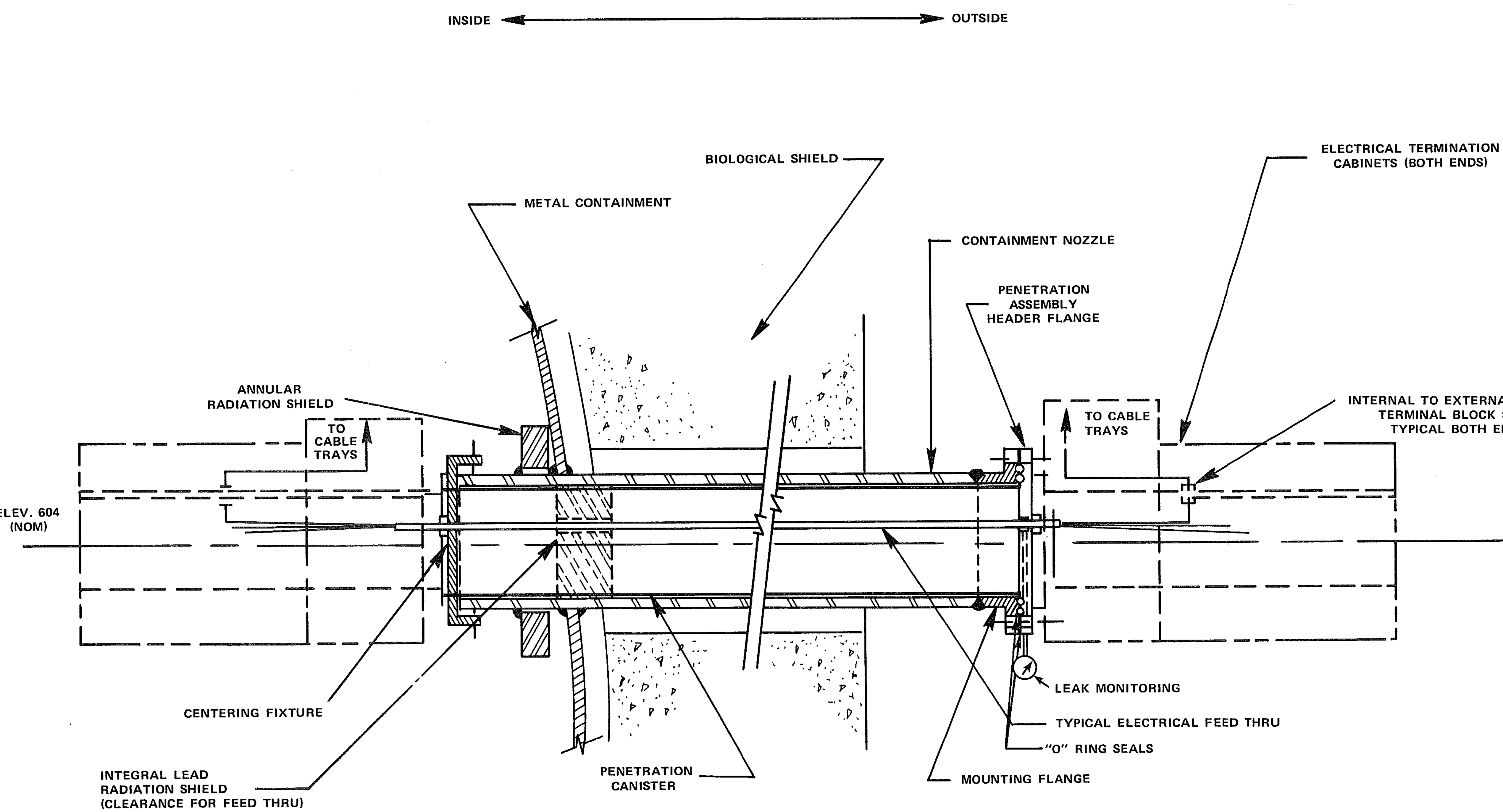
FIGURE 3.8-12

TYPICAL PRIMARY CONTAINMENT PENETRATION ARRANGEMENT



TYPE 8 PENETRATION

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FIGURE 3.8-13

PRIMARY CONTAINMENT SYSTEM ELECTRICAL PENETRATION

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Refer to Plant Drawing I-2146-02

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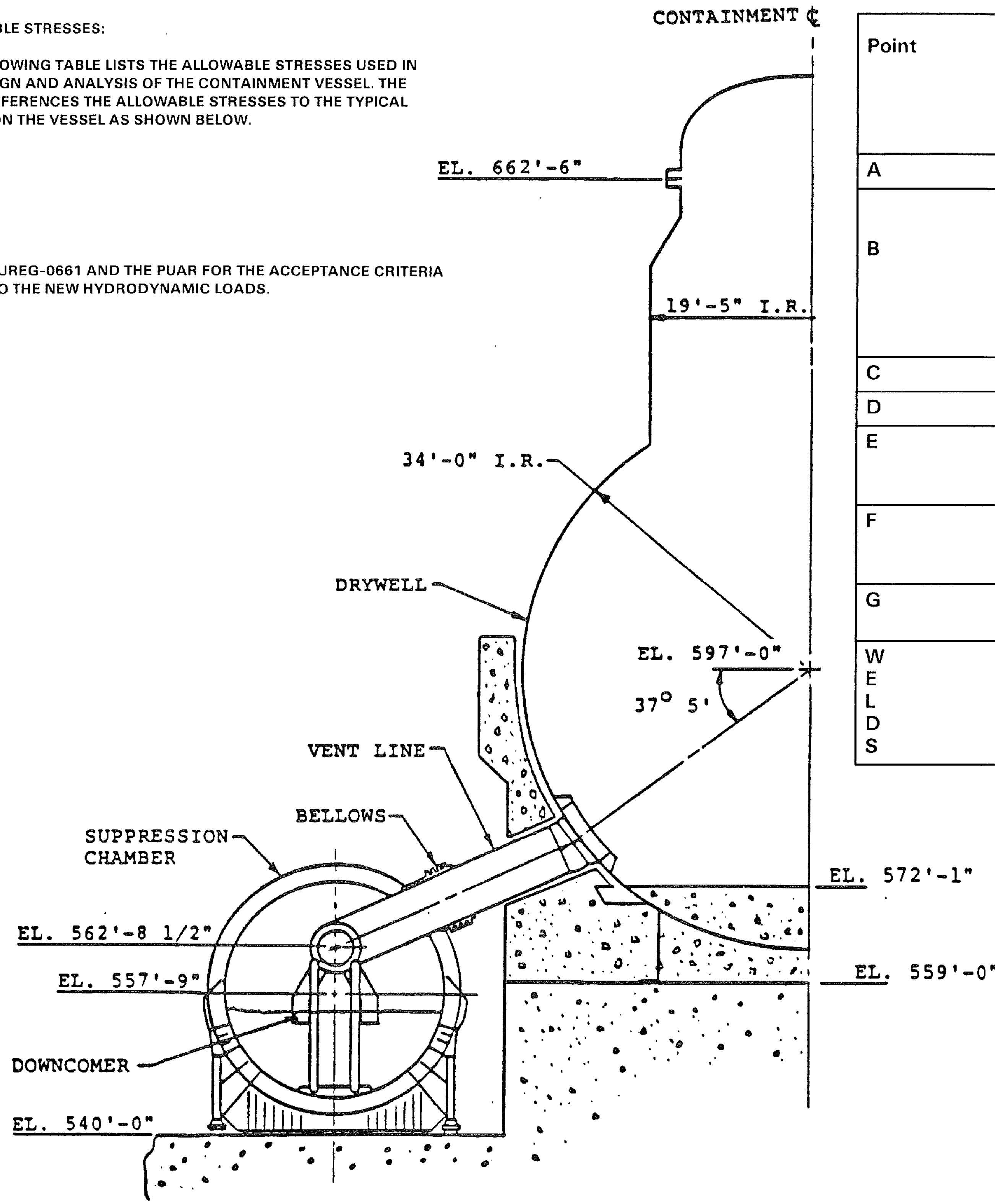
FIGURE 3.8-14

NUCLEAR INSTRUMENTATION SYSTEM POWER RANGE MONITORING SYSTEM TIP DRIVE SYSTEM

ALLOWABLE STRESSES:

THE FOLLOWING TABLE LISTS THE ALLOWABLE STRESSES USED IN THE DESIGN AND ANALYSIS OF THE CONTAINMENT VESSEL. THE TABLE REFERENCES THE ALLOWABLE STRESSES TO THE TYPICAL POINTS ON THE VESSEL AS SHOWN BELOW.

SEE NUREG-0661 AND THE PUAR FOR THE ACCEPTANCE CRITERIA DUE TO THE NEW HYDRODYNAMIC LOADS.



Point	Applicable Code	Type of Stress	Operating Condition		Accident Condition		Flooded Condition	
			Operating Basis Earthquake	Safe Shutdown Earthquake	Operating Basis Earthquake	Safe Shutdown Earthquake	Operating Basis Earthquake	Safe Shutdown Earthquake
A	ASME III-B	Membrane	S_m	S_y	S_m	S_y	S_y	S_y
B	AISC (Plate)	Bending	(.6 F_y)	F_y	(.6 F_y)	F_y	1.5 F_y	1.5 F_y
		Bearing	(.9 F_y)	1.33 (.9 F_y)	(.9 F_y)	1.33 (.9 F_y)	1.33 (.9 F_y)	1.33 (.9 F_y)
		Shear	(.4 F_y)	1.33 (.4 F_y)	(.4 F_y)	1.33 (.4 F_y)	.8 F_y	.8 F_y
		Compression	Code	1.33 Code	Code	1.33 Code	1.33 Code	1.33 Code
C	ASME III-B	Membrane	S_m	S_y	S_m	S_y	S_y	S_y
D	ASME III-B	Membrane	S_m	S_y	S_m	S_y	S_y	S_y
E	ASME III-B	Membrane	S_m	S_y	S_m	S_y	S_y	S_y
		Bending	(.9 F_y)	F_y	(.9 F_y)	1.33 F_y	1.5 F_y	1.5 F_y
F	AISC (Pin)	Bearing	(.9 F_y)	1.33 (.9 F_y)	(.9 F_y)	1.33 (.9 F_y)	1.33 (.9 F_y)	1.33 (.9 F_y)
		Shear	(.4 F_y)	1.33 (.4 F_y)	(.4 F_y)	1.33 (.4 F_y)	.8 F_y	.8 F_y
G	ACI	Bearing	.375 f'c .250 f'c	.626 f'c .418 f'c	.375 f'c .250 f'c	.626 f'c .418 f'c	.8 f'c	.8 f'c
WELDS	AISC (Fillet & Groove)	Shear	15,800 psi	1.33 (.4 F_y)	15,800 psi	1.33 (.4 F_y)	.8 F_y	.8 F_y

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PUAR FIGURE 1-2.1-1

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FIGURE 3.8-15

CONTAINMENT VESSEL STRESS LIMITS

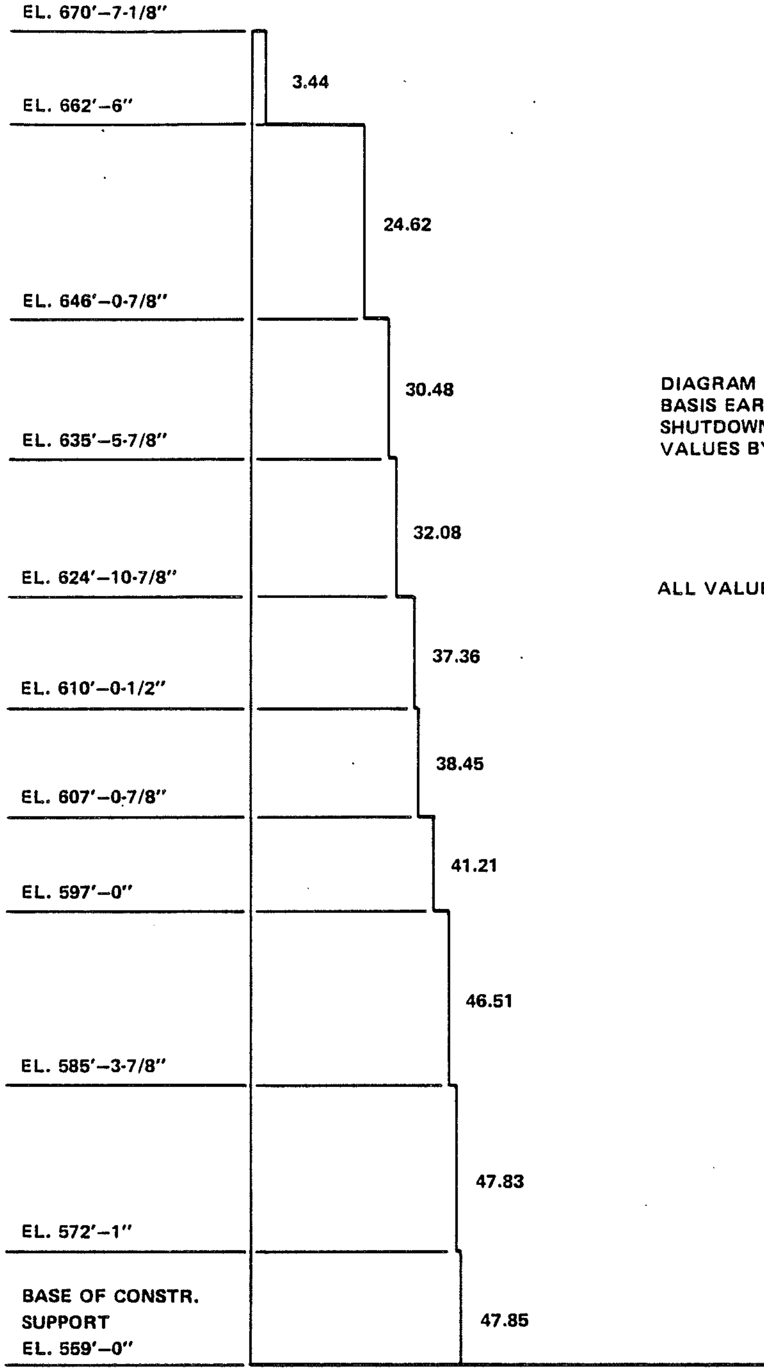

DIAGRAM BASED ON OPERATING BASIS EARTHQUAKE. FOR SAFE SHUTDOWN EARTHQUAKE INCREASE VALUES BY A FACTOR OF 1.875.

ALL VALUES SHOWN ARE IN KIPS.

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FIGURE 3.8-16

SHEAR DIAGRAM CONSTRUCTION MODEL DRYWELL

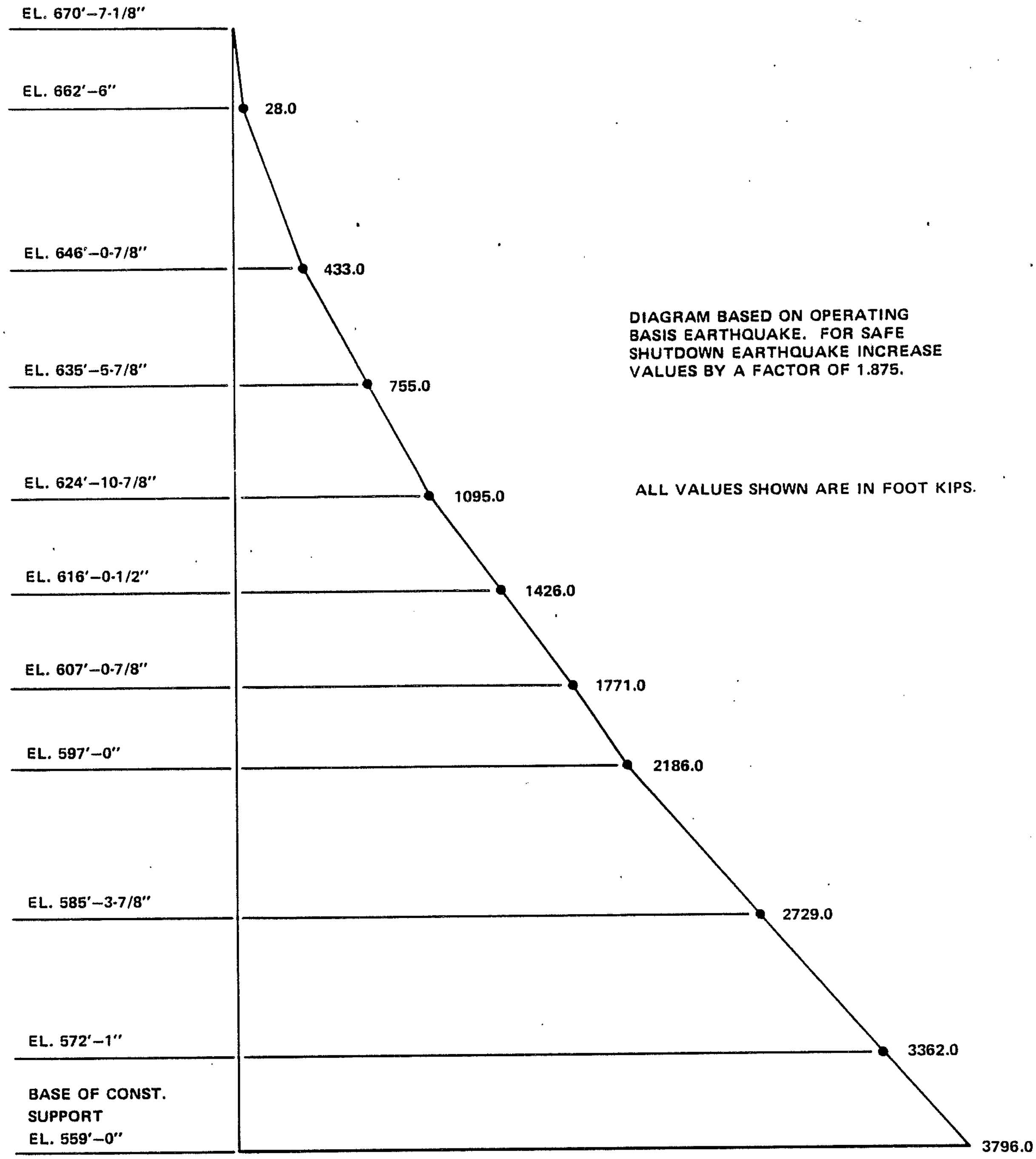

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FIGURE 3.8-17

MOMENT DIAGRAM CONSTRUCTION MODEL DRYWELL

DIAGRAM BASED ON OPERATING BASIS EARTHQUAKE. FOR SAFE SHUTDOWN EARTHQUAKE INCREASE VALUES BY A FACTOR OF 1.875.

ALL VALUES SHOWN ARE IN KIPS.

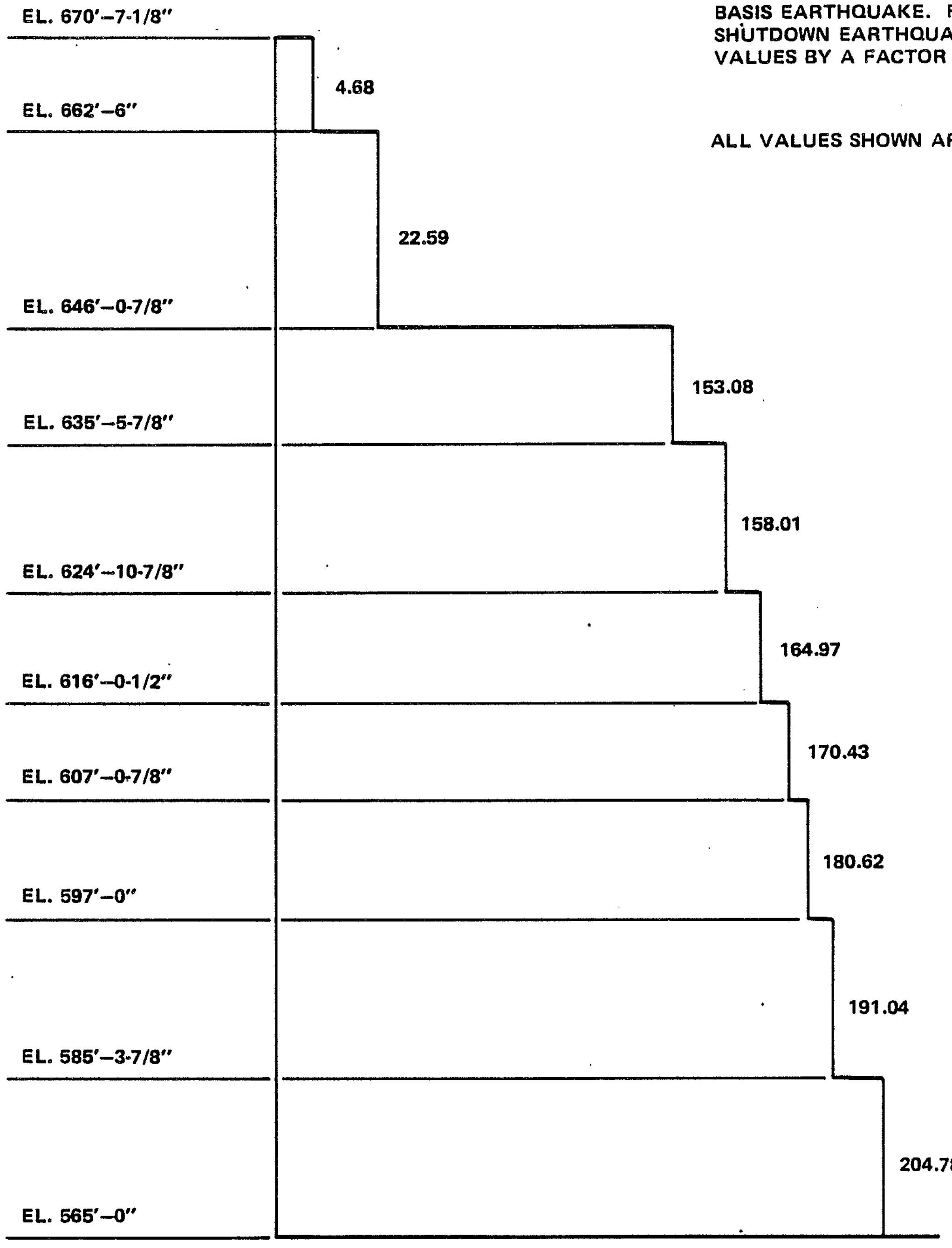

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FIGURE 3.8-18

SHEAR DIAGRAM – DRYWELL EMPTY

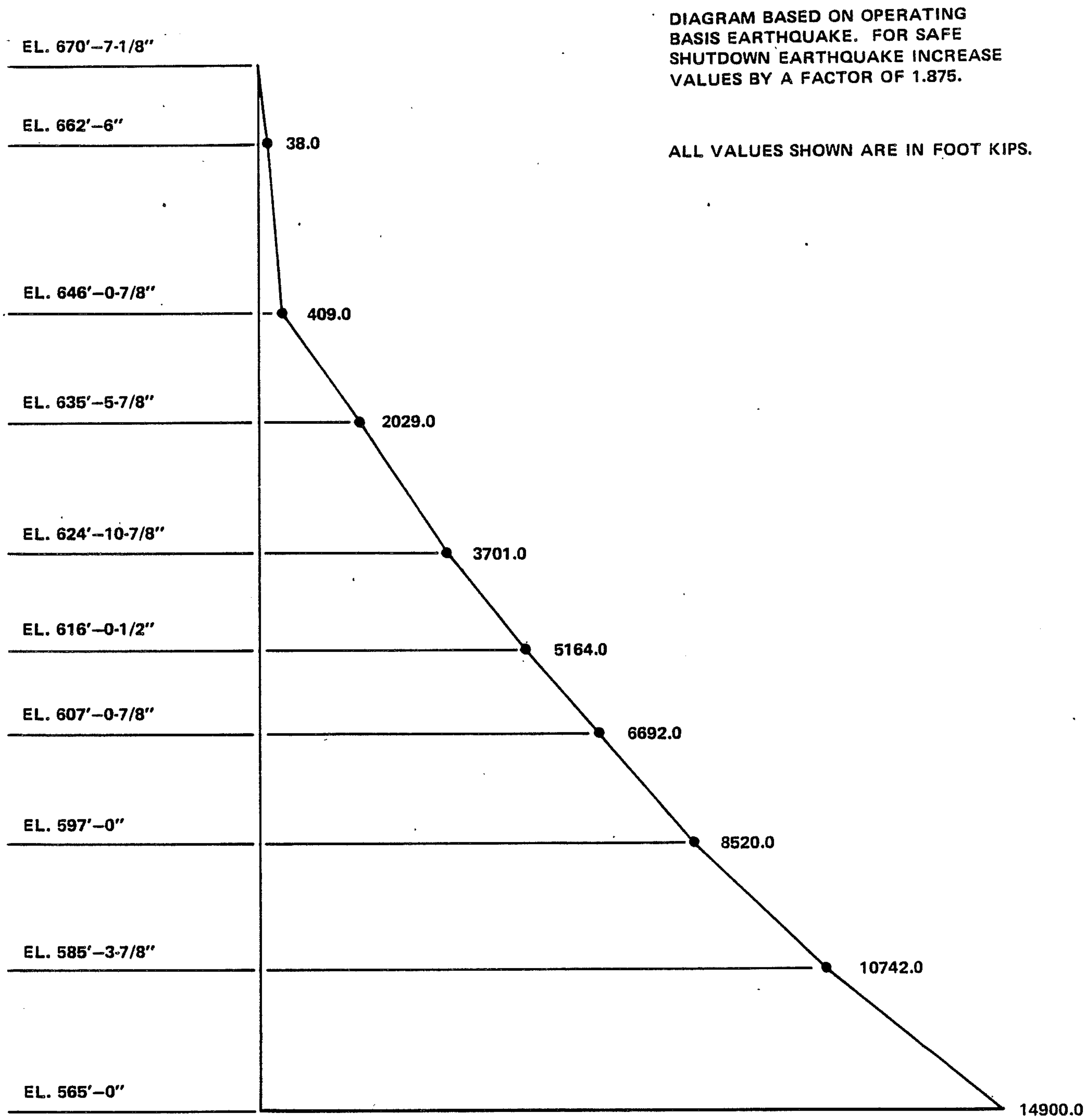

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FIGURE 3.8–19

MOMENT DIAGRAM – DRYWELL EMPTY

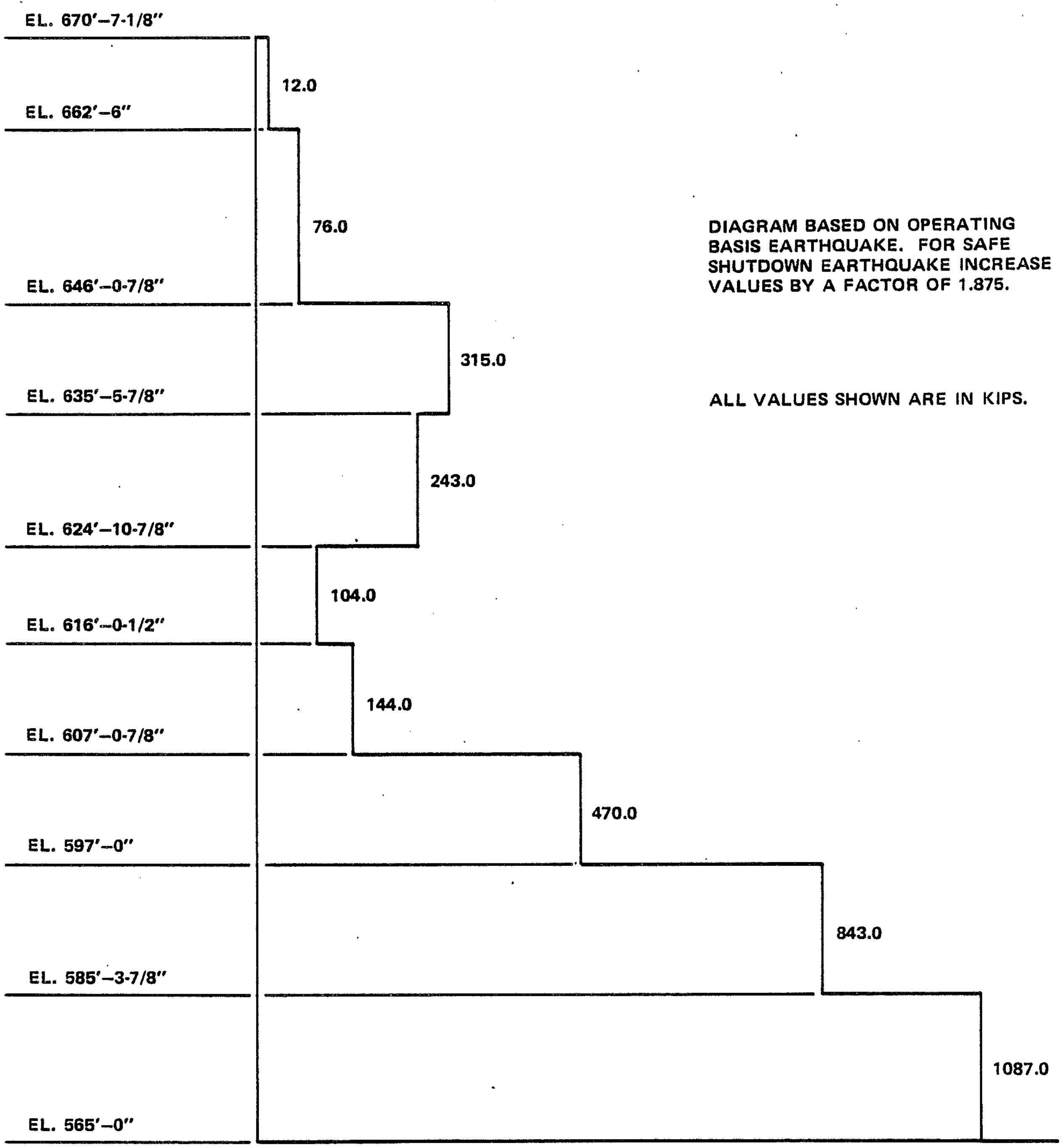

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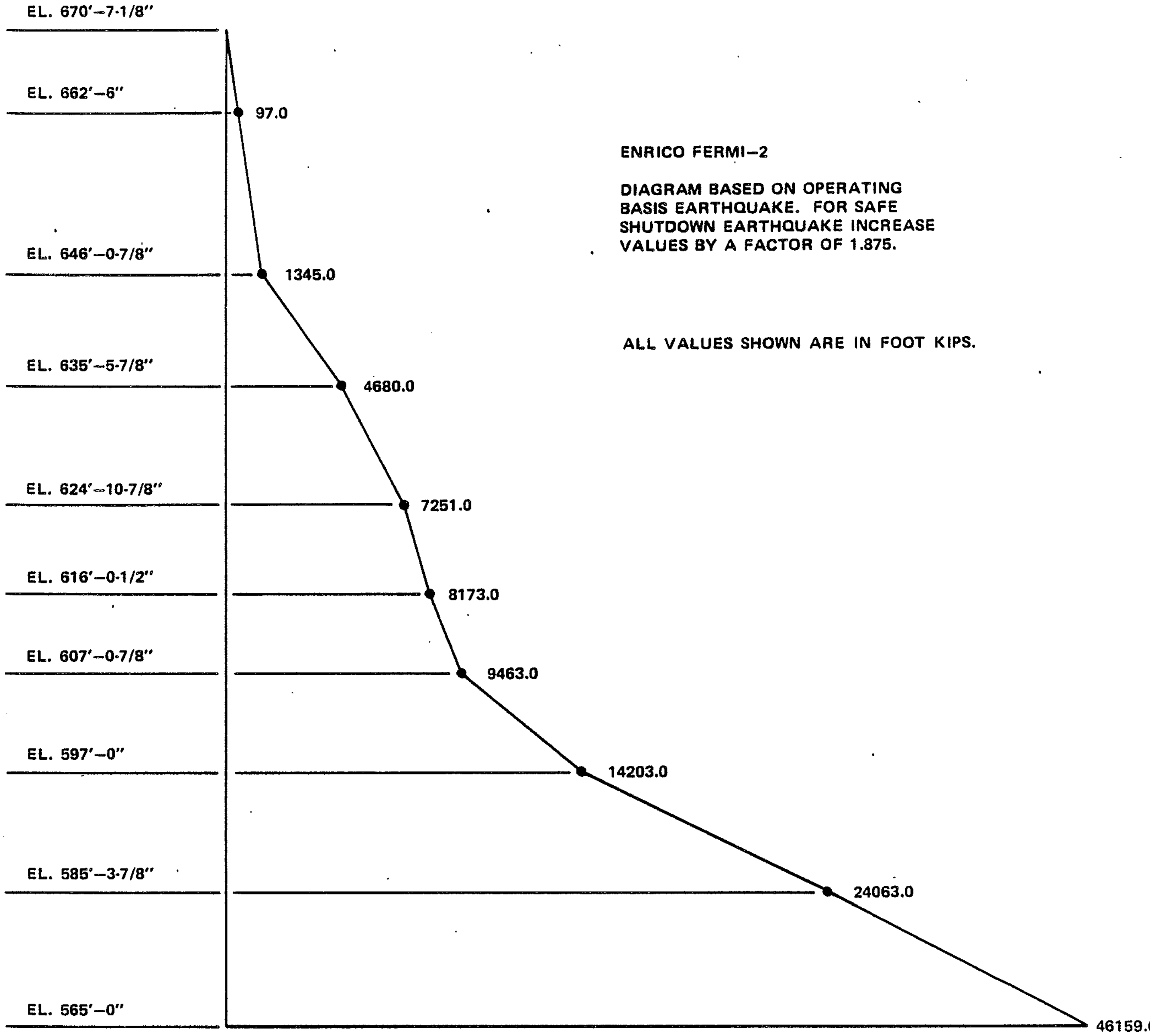
FIGURE 3.8-20

SHEAR DIAGRAM – DRYWELL FILLED WITH WATER



ENRICO FERMI–2

DIAGRAM BASED ON OPERATING BASIS EARTHQUAKE. FOR SAFE SHUTDOWN EARTHQUAKE INCREASE VALUES BY A FACTOR OF 1.875.

ALL VALUES SHOWN ARE IN FOOT KIPS.

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FIGURE 3.8–21

MOMENT DIAGRAM – DRYWELL FILLED WITH WATER

Figure Intentionally Removed
Refer to Plant Drawing C-2431

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FIGURE 3.8-22

SACRIFICIAL SHIELD DETAILS

Figure Intentionally Removed
Refer to Plant Drawing C-2431

FIGURE 3.8-23

SECTION THROUGH THE REACTOR PRESSURE VESSEL SUPPORT PEDESTAL

Figure Intentionally Removed
Refer to Plant Drawing C-2431

FIGURE 3.8-24

DETAIL OF REACTOR PRESSURE VESSEL
CONNECTION TO REACTOR SUPPORT PEDESTAL

Figure Intentionally Removed
Refer to Plant Drawing C-2441

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FIGURE 3.8-25

TYPICAL PART PLAN OF THE
EARTHQUAKE-STABILIZER TRUSS SYSTEM

Figure Intentionally Removed
Refer to Plant Drawing C-2446

Figure Intentionally Removed
Refer to Plant Drawing C-2358

Fermi 2

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FIGURE 3.8-27

TYPICAL SECTION THROUGH THE BIOLOGICAL SHIELD SHOWING REINFORCING LAYOUT

Figure Intentionally Removed
Refer to Plant Drawing C-2360

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FIGURE 3.8-28, SHEET 1

LONGITUDINAL SECTION THROUGH THE SPENT FUEL STORAGE POOL, REACTOR REFUELING POOL, AND DRYER SEPARATOR STORAGE POOL SHOWING REINFORCING LAYOUT

Figure Intentionally Removed
Refer to Plant Drawing C-2361

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FIGURE 3.8-28, SHEET 2

LONGITUDINAL SECTION THROUGH THE SPENT FUEL STORAGE POOL, REACTOR REFUELING POOL, AND DRYER SEPARATOR STORAGE POOL SHOWING REINFORCING LAYOUT

Figure Intentionally Removed
Refer to Plant Drawing C-2400

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FIGURE 3.8-29

TRANSVERSE SECTION THROUGH THE SPENT FUEL STORAGE POOL SHOWING REINFORCING LAYOUT

Figure Intentionally Removed
Refer to Plant Drawing C-2372

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FIGURE 3.8-30

TRANSVERSE SECTION THROUGH THE DRYER SEPARATOR POOL

Figure Intentionally Removed
Refer to Plant Drawing C-2348

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FIGURE 3.8-31, SHEET 1

PLAN VIEW OF THE STORAGE POOLS

Figure Intentionally Removed
Refer to Plant Drawing C-2349

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FIGURE 3.8-31, SHEET 2

PLAN VIEW OF THE STORAGE POOLS

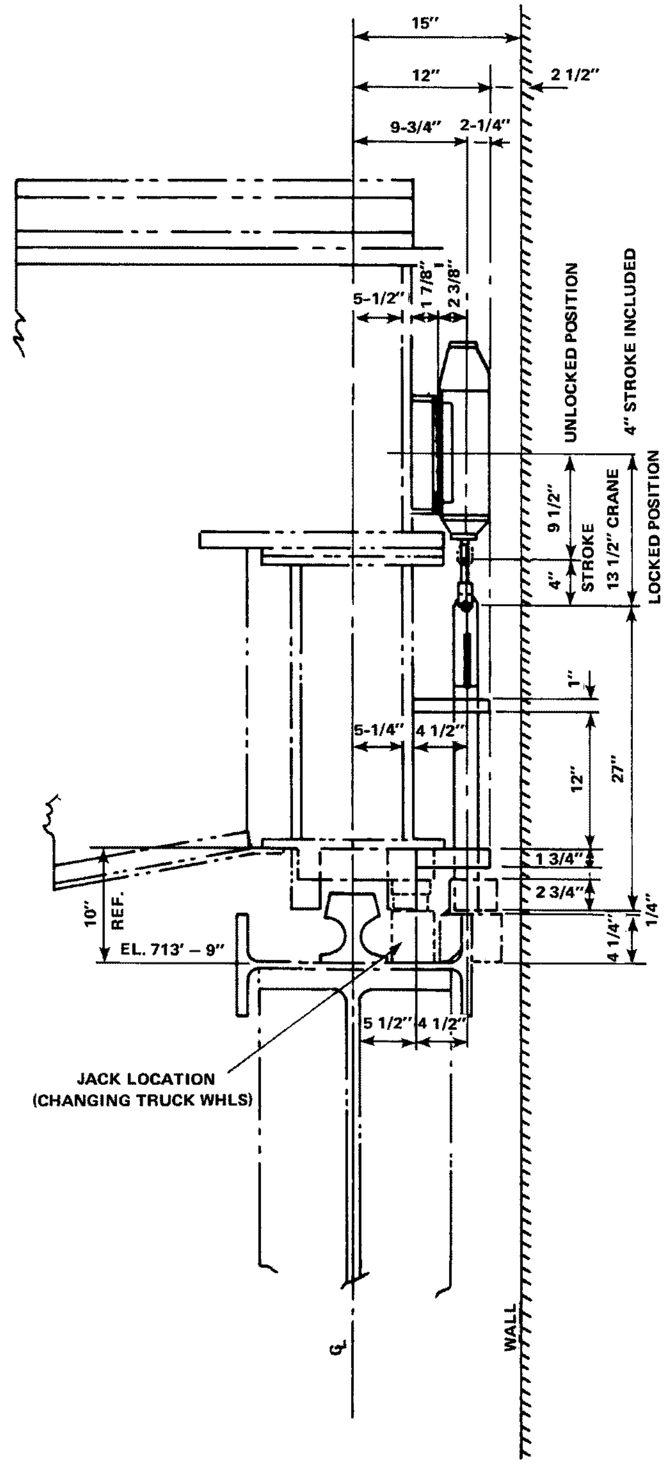

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FIGURE 3.8-32

REACTOR BUILDING CRANE SEISMIC AND TORNADO SAFETY FEATURES

HARNISHPHAGER CRANE MANUAL DRAWING NO. 106A64

Figure Intentionally Removed
Refer to Plant Drawing C-2415

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FIGURE 3.8-33

TYPICAL COLUMN REINFORCEMENT AND TIE SPACING

Figure Intentionally Removed
Refer to Plant Drawing C-2412

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FIGURE 3.8-34

TYPICAL WALL REINFORCING SPLICE DETAIL

Figure Intentionally Removed
Refer to Plant Drawing C-2413

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FIGURE 3.8-35

TYPICAL BEAM REINFORCING DETAILS

Figure Intentionally Removed
Refer to Plant Drawing C-2412

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FIGURE 3.8-36

TYPICAL ADDITIONAL SLAB REINFORCING AT RECTANGULAR OPENINGS

Figure Intentionally Removed
Refer to Plant Drawing C-2412

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FIGURE 3.8-37

TYPICAL SLAB REINFORCING DETAILS

Figure Intentionally Removed
Refer to Plant Drawing C-2412

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FIGURE 3.8-38

TYPICAL CONSTRUCTION JOINT DETAILS

Figure Intentionally Removed
Refer to Plant Drawing C-2313

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FIGURE 3.8-39

PLAN VIEW OF THE REACTOR/AUXILIARY BUILDING BASE MAT – TYPICAL REINFORCING DETAIL

Figure Intentionally Removed
Refer to Plant Drawing C-2315

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FIGURE 3.8-40

SECTION THROUGH THE DRYWELL PEDESTAL AND SUPPRESSION CHAMBER BASE SLAB TYPICAL REINFORCING DETAIL

Figure Intentionally Removed
Refer to Plant Drawing C-2313

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FIGURE 3.8-41

TOP OF DRYWELL PEDESTAL
TYPICAL REINFORCING DETAIL

Figure Intentionally Removed
Refer to Plant Drawing M-2836

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FIGURE 3.8-42

HIGH PRESSURE COOLANT INJECTION PUMP AND TURBINE FOUNDATIONS - TYPICAL REINFORCING DETAILS

Figure Intentionally Removed
Refer to Plant Drawing M-2840

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FIGURE 3.8-43

REACTOR CORE ISOLATION COOLING TURBINE PUMP AND BAROMETRIC CONDENSER FOUNDATIONS – TYPICAL REINFORCING DETAILS

Figure Intentionally Removed
Refer to Plant Drawing M-2734

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FIGURE 3.8-44

TYPICAL DETAILS OF THE RESIDUAL HEAT REMOVAL PUMP FOUNDATIONS

Figure Intentionally Removed
Refer to Plant Drawing M-2679

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FIGURE 3.8-45

TYPICAL DETAILS OF THE CORE SPRAY PUMP FOUNDATIONS

Figure Intentionally Removed
Refer to Plant Drawing C-2308

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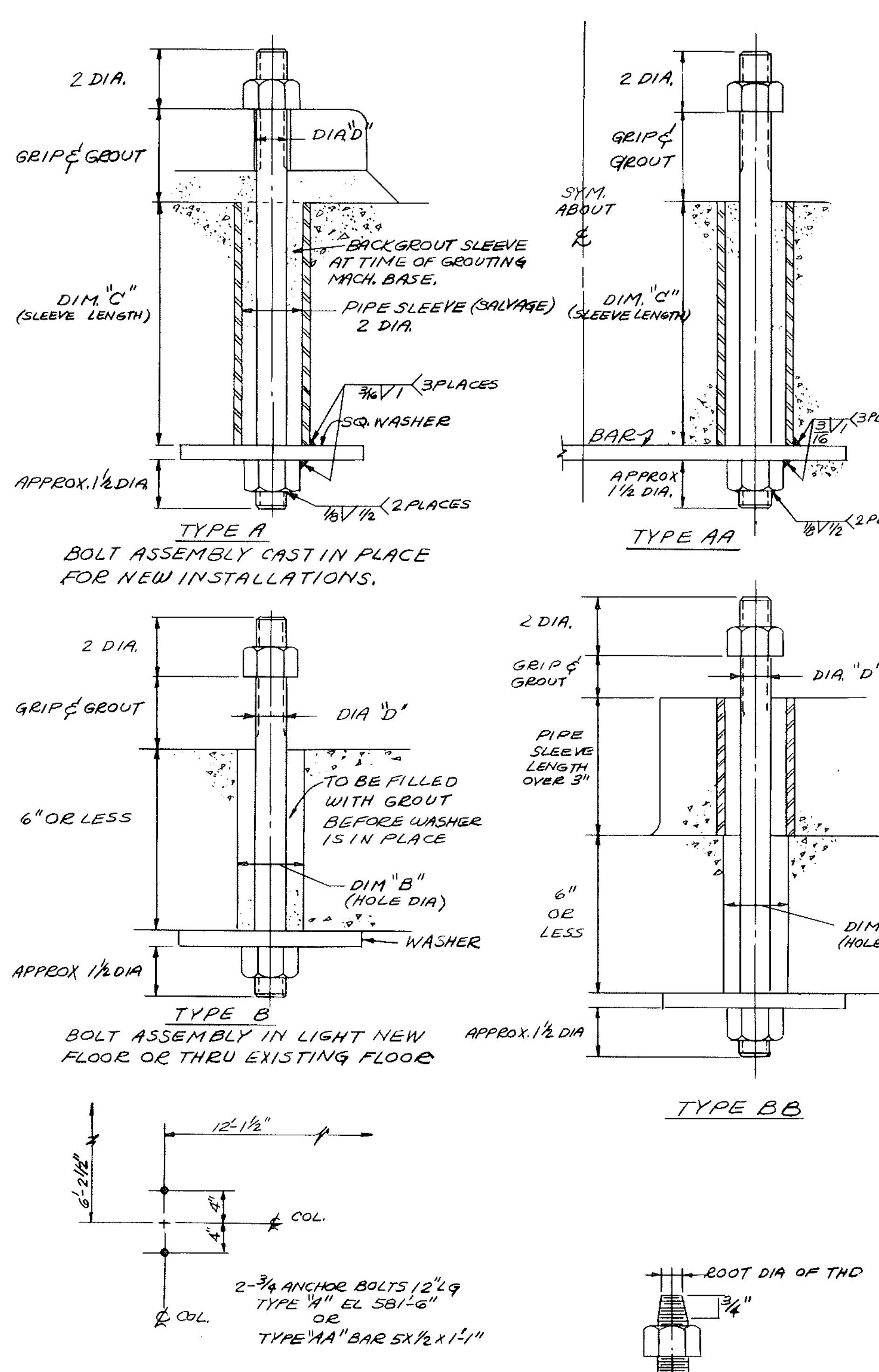
FIGURE 3.8-46

TYPICAL REINFORCING PATTERNS AT THE JUNCTION OF CONCRETE WALLS AND THE FOUNDATION MATS



TYPE A
BOLT ASSEMBLY CAST IN PLACE FOR NEW INSTALLATIONS.

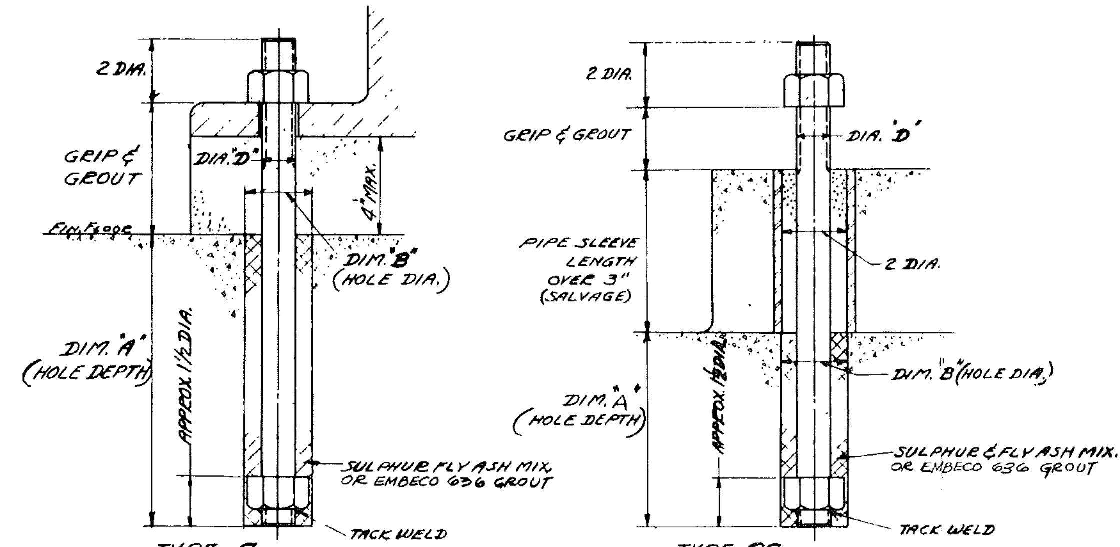
TYPE AA

TYPE B
BOLT ASSEMBLY IN LIGHT NEW FLOOR OR THRU EXISTING FLOOR

TYPE BB

TYPICAL EXAMPLE FOR COLUMN ANCHORS AS IT SHOULD APPEAR ON THE DESIGN DRAWING.

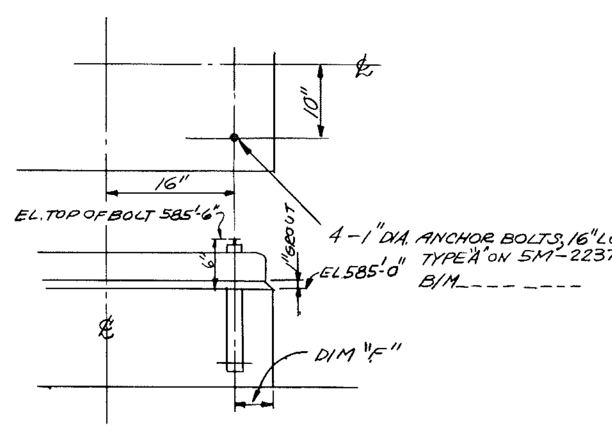
"TAPER TOP"
(TYPICAL FOR ALL COLUMN ANCHORS.)



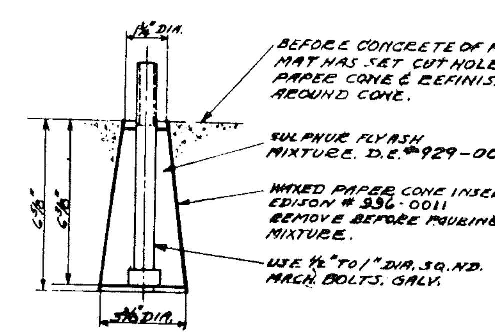
TYPE C
IN EXISTING CONCRETE FLOOR

TYPE CC
IN EXISTING CONCRETE FLOOR

BOLT & SLEEVE SIZES ALL DIMENSIONS IN INCHES					WASHER SIZES			*SULPHUR & FLY ASH MIXTURE	
					WASHERS				
BOLT SIZE "D" DIM.	HOLE DEPTH "A" DIM.	DRILL SIZE "B" DIM.	SLEEVE LENGTH "C" DIM.	EDGE CLEAR. "F" DIM.	INCHES SQ.	INCHES THK.	HOLE DIA.	LBS MIX PER HOLE	Nº OF BOLTS PER 10 LBS
1/2	4	1 1/2	8	4 1/2	4	1/4	9/16	.44	22.6
5/8	5	1 5/8	8	4 1/2	4	3/8	11/16	.82	12.13
3/4	6	1 3/4	8	4 1/2	4	3/8	13/16	1.35	7.4
7/8	7	1 7/8	8	5 1/2	6	1/2	15/16	2.10	4.78
1	8	2	8	5 1/2	6	1/2	1 1/16	2.61	3.83
1 1/4	10	2 1/2	9	5 1/2	6	1/2	1 5/16	5.11	1.95
1 1/2	12	3	11	7	8	5/8	1 9/16	8.81	1.13
1 3/4	14	3 1/2	13	7	8	5/8	1 13/16	13.8	.72
2	16	4	15	7	8	5/8	2 1/16	16.2	.67



TYPICAL EXAMPLE FOR MACHINE ANCHORS AS IT SHOULD APPEAR ON THE DESIGN DRAWING.



TYPE D
ANCHORING EQUIPMENT TO OUTDOOR MATS.

NOTES:

GENERAL DESIGN INSTRUCTIONS—(SEE SKETCH).

1. INDICATE ON DESIGN DRAWING THE LOCATION, ELEVATION OF TOP OF BOLT, DIAMETER LENGTH & TYPE OF ANCHOR (ALSO SLEEVE LENGTH FOR TYPE "CC" & BAR SIZE FOR TYPE "AA").
2. REFER TO THIS STANDARD DRAWING NO.
3. INDICATE ANY SPECIAL CONSIDERATIONS, IF OTHER THAN SHOWN IN TABLE, SUCH AS SLEEVE LENGTH, THREAD LENGTH EACH END, ETC.
4. KEEP LENGTH OF BOLT TO NEAREST ½" WHERE POSSIBLE.
5. THREAD TOP END OF ALL BOLTS TO A LENGTH EQUAL TO 4 x DIAMETER AND OTHER END TO A LENGTH EQUAL TO 1½ x DIAMETER—U.S. STANDARD COARSE THREAD — CLASS 3 (MEDIUM FIT).
6. USE HEAVY HEX NUTS, SEMI FINISHED, WASHER FACED.—CLASS 3 (MEDIUM FIT).
7. USE STANDARD HEX HEAD OR SQ. HD. BOLTS WHEREVER POSSIBLE.
8. ANCHOR BOLTS WILL BE ORDERED BY THE USING DIVISION.

SULPHUR & FLY ASH MIXTURE—
D.E. CO. STOCK #989-0099, IN 10# CANS. MAKES 1¼ QUARTS WHEN MELTED.
PREPARE HOLES IN CONCRETE AS INDICATED ABOVE & SET BOLTS IN PROPER POSITION. MELT PREMIXED MATERIAL IN MELTING POT SET AT 240°-250° F. STIR MELTED MATERIAL IN POT IMMEDIATLY BEFORE PURING AROUND BOLT. AVOID OVERHEATING, AT 270# F. MIX BEGINS TO THICKEN AND BECOMES DIFFICULT TO POUR TOO HIGH A TEMPERATURE WILL CAUSE MIX TO FUME EXCESSIVELY & TO CATCH FIRE. THIS MIXTURE HAS A HOLDING STRENGTH SEVERAL TIMES THAT OF LEAD. (D.E. CO. RESEARCH DEPT. REPORT 51H87.)

* THEORETICAL QUANTITIES, ADD APPROX. 10% WHEN ORDERING SULPHUR MIX.

DETROIT EDISON COMPANY DRAWING NO. 5M-2237, REV. E

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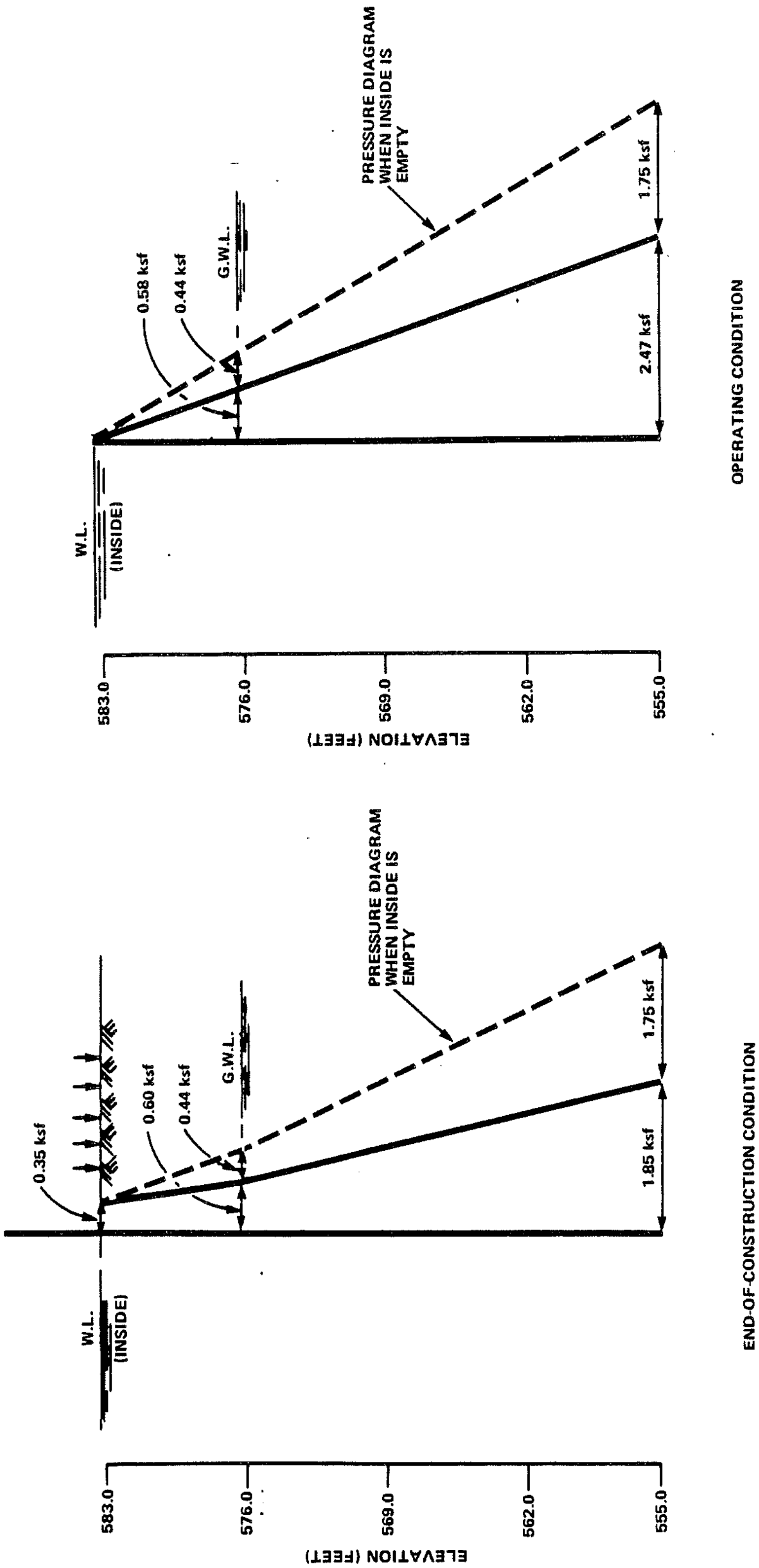
FIGURE 3.8-47

TYPICAL ANCHOR BOLT DETAILS FOR CATEGORY I EQUIPMENT



NOTE:
A SURCHARGE OF 0.500 ksf ON THE GRADE IS CONSIDERED FOR END-OF-CONSTRUCTION CONDITION ONLY.

FOR DYNAMIC LATERAL PRESSURE COMPUTATION TECHNIQUE; SEE TABLE 2.5-17.

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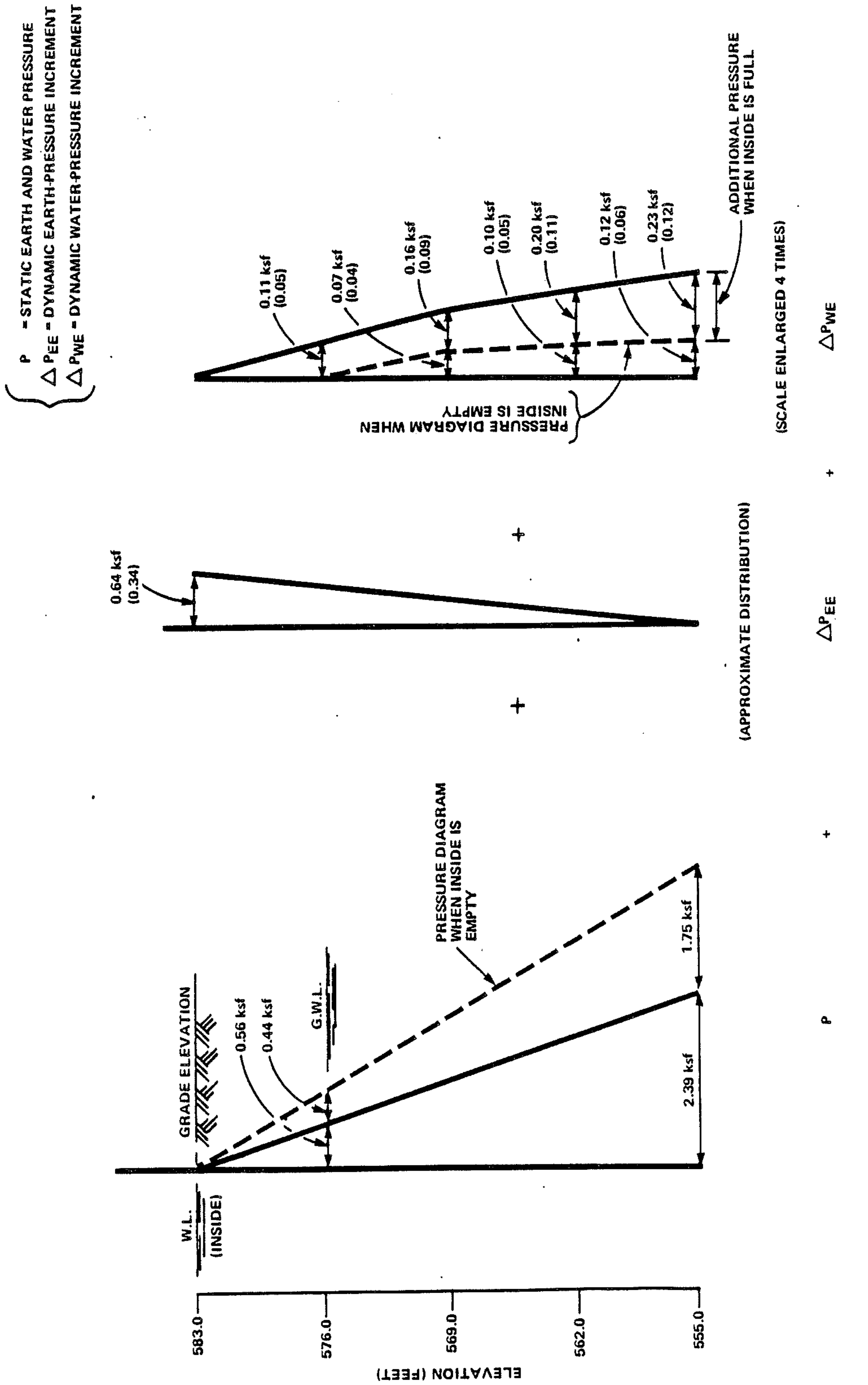
FIGURE 3.8-48, SHEET 1

EARTHQUAKE CONDITION – STATIC LATERAL PRESSURE ON SUBSURFACE WALL OF RESIDUAL HEAT REMOVAL BUILDING



NOTE:
FIGURES IN THE PARENTHESES ARE VALUES CORRESPONDING TO THE OBE (OPERATING BASIS EARTHQUAKE) CONDITION WITH k_h = 0.08 g; OUTSIDE FIGURES CORRESPOND TO THE DBE (DESIGN BASIS EARTHQUAKE) CONDITION WITH k_h = 0.15 g.

FOR DYNAMIC LATERAL PRESSURE COMPUTATION TECHNIQUE, SEE TABLE 2.5-17.

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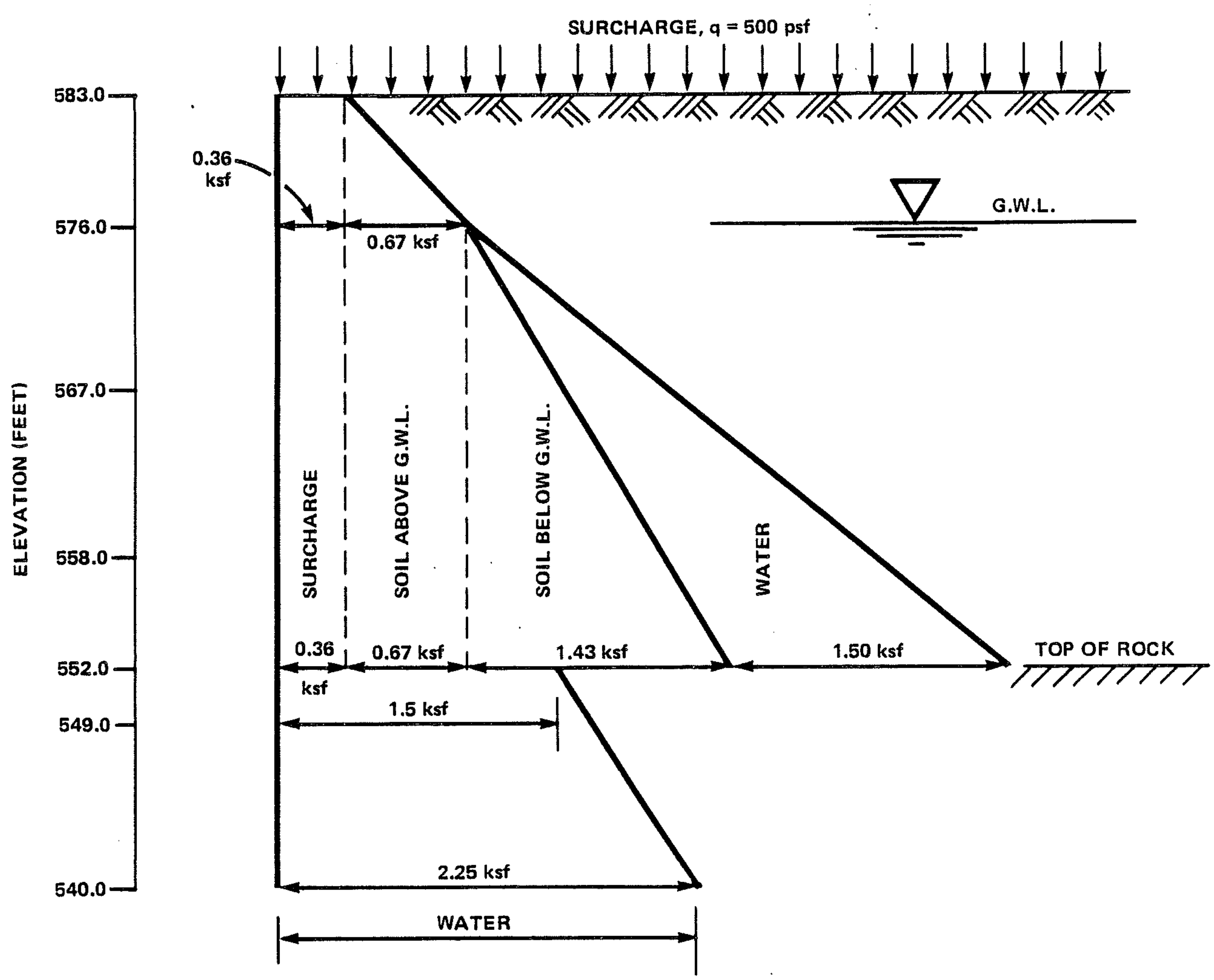
FIGURE 3.8–48, SHEET 2

EARTHQUAKE CONDITION – STATIC LATERAL PRESSURE ON SUBSURFACE WALL OF RESIDUAL HEAT REMOVAL BUILDING



CASE 1. END OF CONSTRUCTION CONDITION

NOTE:
FOR DYNAMIC LATERAL PRESSURE COMPUTATION TECHNIQUE, SEE TABLE 2.5-17.

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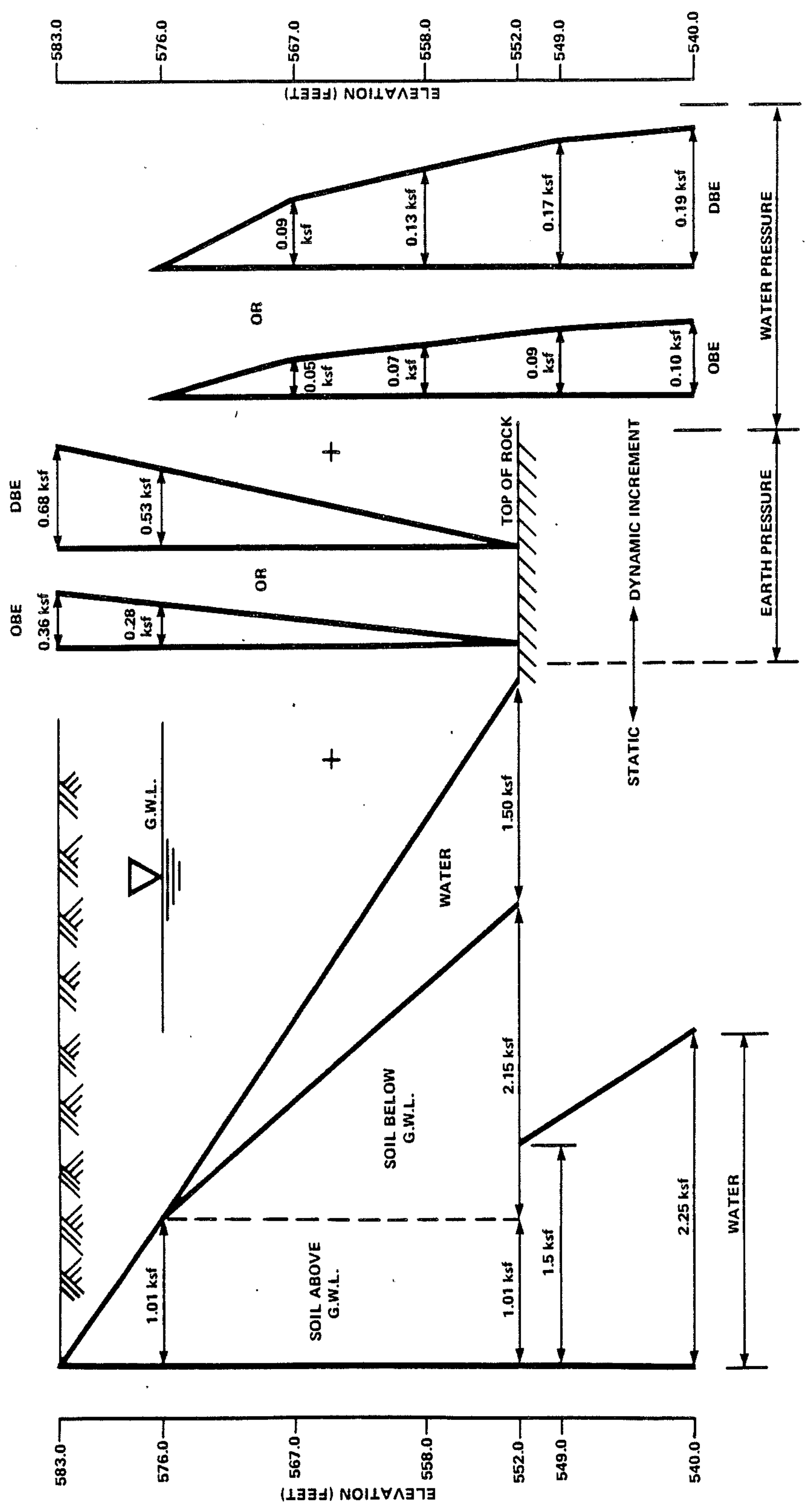
FIGURE 3.8-49, SHEET 1

REACTOR/AUXILIARY BUILDING STATIC AND DYNAMIC LATERAL PRESSURES – OUTSIDE WALL



NOTE:
FOR DYNAMIC LATERAL PRESSURE COMPUTATION TECHNIQUE, SEE TABLE 2.5-17.

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FIGURE 3.8-49, SHEET 2

REACTOR/AUXILIARY BUILDING STATIC AND DYNAMIC LATERAL PRESSURES – OUTSIDE WALL

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 Dynamic System Analysis and Testing

3.9.1.1 Piping and Rotating Equipment Test Program

3.9.1.1.1 Test Objectives

The piping of Fermi 2 is designed in conformance with the vibration requirements of the USAS B31.7-1969 Code for Pressure Piping and/or ASME Boiler and Pressure Vessel (B&PV) Code Section III, 1971 issue (including winter addenda). In accordance with the general objectives of this code, preoperational and startup phase vibration inspections will be conducted on piping systems and rotating equipment. These surveys will be performed with the following objectives in mind.

a. Observe that the vibration of the tested piping system is within acceptable limits

b. Provide baseline vibration signatures of rotating equipment which will serve as data for future comparison

c. Reveal potentially significant, equipment-induced resonances or pressure pulsations within the system and the operating modes during which they occur

d. Provide data to verify compliance with manufacturer's standards and tests, or existing Edison standards

e. Verify that the piping and support systems perform properly over the operating temperature range.

Table 3.9-1 lists the piping systems included in this test program and indicates the extent to which each system will be tested. Using the data collected during this testing and the acceptance criteria outlined in Subsection 3.9.1.1.5, each system will be evaluated for compliance with the original intent of the piping design criteria.

If the test results exceed the acceptance criteria for a given piping system, further evaluation will be performed to determine if it is necessary to modify the system. If the system is modified, additional testing of the modified system will be conducted if the modification significantly changes the vibratory behavior of the piping system.

Final design evaluation of safety-related piping systems has been performed in accordance with B&PV Code Section III after all modifications were completed.

Emergency and faulted-type transients, including such events as pump seizure, pipe rupture, etc., are not part of this testing program because these transients cannot be tested or simulated.

3.9.1.1.2 Rotating Equipment Vibration Testing

Vibration testing of the Fermi 2 rotating equipment was conducted during the preoperational and startup phases. The equipment tested is as follows.

Component	Quantity
Residual heat removal (RHR) and core spray pumps and motors (four each)	8
High pressure coolant injection pump and turbine	1
Reactor core isolation cooling pump and turbine	1
Reactor recirculation pumps and motors	2

The specific conditions for which vibration data has been obtained on each piece of equipment are as follows:

a. Design flow rate

b. Minimum normal flow rate

c. Maximum normal flow rate

d. Startup

e. Shutdown

To obtain vibration data for the rotating equipment, instrumentation has been installed temporarily on the various pumps and motors at locations that, based on the experience of Edison and the manufacturers, provide significant data. This instrumentation is used to measure bearing vibration or relative motion of the shaft, with respect to the case, in the axial and radial directions. Where possible, the vibration instrumentation is fastened to the machines using magnetic bases or some other temporary means. Instruments of an adequate type, number, and location already installed on machines can be used for testing.

Data is recorded on a multichannel, magnetic tape recorder. Narrow-band frequency spectrum analysis has been performed on these data to permit comparison of each frequency component with applicable criteria. The magnetic tape recording and its analysis is retained as permanent baseline data to permit identification of the deterioration of equipment performance.

In I&E Bulletin 79-15, the NRC identified concerns over the long-term operability of deep-draft pumps. In response to this, Edison described the steps being taken to ensure long-term operability of the pumps. These steps are

a. Quality verification of the pump and motor assembly during manufacture

b. Construction verification of the foundation and sump

c. Verification of proper installation and alignment of the pump assembly

d. Startup testing sufficient to verify pump capability and condition for long-term operability

e. Inservice surveillance testing using sophisticated vibration-measuring techniques to determine any degradation of internal components.

Inservice surveillance testing takes two forms: operational monitoring and diagnostic testing. Operational monitoring is performed by the operating instrumentation installed locally or in the control room. Thrust bearing temperature, on-off-auto control switches, and pump running status are all operationally monitored.

Surveillance testing consists of the following:

a. Determination of the total head developed by the pump

b. Measuring the flow from the pump

c. Measuring vibration of the pump-motor assembly using readings in velocity units.

These tests are aimed at providing the earliest possible detection of pump problems which exhibit the following symptoms:

a. Degradation of capacity or developed head

b. Excessive vibration

c. Excessive thrust bearing temperature.

Following repair and/or reassembly of a pump/motor unit, and/or during the Section XI inservice test, vibration base data will be taken prior to returning the unit to service.

The only safety-related deep-draft pumps used at Fermi 2 are the service water pumps.

3.9.1.1.3 Piping System Vibration Testing

Vibration surveys will be conducted on the piping systems listed in Table 3.9-1 during the preoperational and startup phases. For the majority of these systems, the system and plant conditions existing during the preoperational phase will be adequate to obtain vibration data representative of the piping vibration experienced during normal system operation.

Table 3.9-1 indicates the extent of vibration testing that will be performed on the piping systems and their supports. Normally, vibration data will be taken during steady-state operation of the system. Data will be taken also on portions of selected systems during specific transient events. These systems and the events are

a. Feedwater system piping from the feedwater pump discharge to the containment penetration, following a trip of a feedwater pump

b. High-pressure coolant injection (HPCI) system piping from the HPCI pump discharge to the feedwater system tee connection, after a rapid start of the HPCI turbine

c. Main steam piping from the turbine stop valve to the reactor vessel, after a turbine stop valve and control valve fast closure

d. Selected main steam safety/relief valve (SRV) discharge piping during SRV operation

e. Recirculation piping for a pump trip at 100 percent rated flow.

The vibration surveys will entail monitoring the overall system for indications of unacceptable vibratory response. Where appropriate, deflections, pressure pulsations, restraint forces, or accelerations will be monitored. The points that will be selected for monitoring will be, typically, those points which are predicted, either by experience or analysis, to undergo the highest deflections, pressure surges, operating stresses, or vibrations during system operation. Using the criteria of Subsection 3.9.1.1.5, the vibration data will be evaluated to determine the acceptability of the piping system design.

In addition to the piping systems listed in Table 3.9-1, safety- related small-bore piping and instrument lines will be included in the vibration surveys. Test, branch, bypass, and instrument lines attached to the piping systems in the areas selected for monitoring will be observed to ascertain that there will be no danger to personnel and no potential damage to the system under investigation.

Special attention will be given to ensure that these small lines are not in resonance with operating equipment or flow-induced vibrations of the attached large-bore lines.

Based on those observations, instrumentation and other lines 2 in. and smaller attached to the test system piping and to the system components in the piping areas selected for monitoring will be inspected visually for the following specific reasons:

a. To eliminate danger to personnel

b. To ensure that there will be no damage (such as fatigue failure) to the primary system at junctions with large piping and equipment.

Therefore, safety-related small-bore piping and instrument lines will be included in the test program, subject to the above considerations and, in general, covering only the junction points (taps, tees, etc.) with the main system piping under test as listed in Table 3.9-1. Such junction points are assumed to be the worst case for fatigue failure of the small-bore piping instrument lines. If the piping system itself is small-bore piping, as in the case of the control rod drive (CRD) lines, then the test program covers small-bore piping and instrument lines in its entirety (as limited by accessibility and personnel safety) at the system level.

For instrumentation lines that because of accessibility or personnel safety cannot be inspected during system operation, an inspection of the lines' routing and supports was completed prior to operation. The inspection verifies that the instrumentation lines have been adequately supported to resist vibrations caused by the header piping or equipment to which the lines are attached (as there is no flow in instrumentation lines, vibrations from header piping or equipment are the source of excitation). If it was determined that the lines were not adequately supported or routed, the routings were modified or supports were added to obtain an appropriate design.

3.9.1.1.4 Thermal Expansion Testing

The thermal expansion movements of piping systems identified in Table 3.9-1 will be monitored when these systems are heated initially to their normal operating temperatures. This testing will normally take place during the startup phase.

Prior to the heatup of a system, points of potential contact with other equipment will be identified. During heatup of the system, these points will be monitored to verify that free movement of the piping is not hindered.

The thermal expansion deflections of selected points of the piping systems will be monitored either visually or with test instrumentation. Normally, the points to be monitored will be those points that are predicted by the stress analyses to exhibit relatively large, thermally induced deflections. During heatup, data will be taken at temperature intervals that will allow abnormal conditions to be identified before specified limits are exceeded. An additional set of data will be taken when the monitored system returns to ambient temperature to verify that piping is free to contract during cooldown. The criteria of Subsection 3.9.1.1.5 will be used to evaluate the thermal expansion performance of the tested systems.

3.9.1.1.5 Acceptance Criteria for Piping Vibration and Thermal Expansion Testing

These vibration criteria apply only to the systems being monitored as part of the Vibration and Dynamic Effects Test Program (See Table 3.9-1). All piping systems are subject to various dynamic forces caused by fluid flow, some transient, some steady-state. Each piping system, because of its unique configuration, will vibrate at its own fundamental frequencies. These criteria are developed to detect any vibratory deflections of sufficient amplitude to cause the intent of the original design criteria to be violated.

Thermal expansion occurs as a result of any system heatup. When a piping system is designed, a flexibility analysis is performed that verifies analytically that the system configuration is not overstressed while undergoing the thermal growth expected to result from the change in system temperatures. The purpose of the thermal expansion testing is to verify that the actual thermal growth is reasonable and unrestricted and is within the parameters of the acceptance criteria contained herein.

3.9.1.1.5.1 Level 1 and Level 2 Criteria

When applicable, Level 1 and Level 2 acceptance criteria will be established. Violation of Level 1 acceptance criteria for those systems and locations being monitored indicates that the design limits of the piping may be exceeded during the tests. Further operation of the system in the offending mode of operation will be avoided. The system response will be evaluated and the violation will be resolved by analysis and/or corrective action.

Violation of Level 2 criteria indicates that stress levels exceed long-term operating criteria but that a short-term threat to the piping system integrity does not exist. Violations of Level 2 criteria will not require the halting of the test but will require post-test evaluation to be performed to ascertain if the apparent violation was of significance and to determine what, if any, system modifications may be necessary to bring the system into acceptable limits.

3.9.1.1.5.2 Steady-State Vibration Acceptance Criteria

The following allowable stress amplitude, S_a, will be used for steady-state piping vibration:

S_a = 7690 psi for carbon steels with UTS $<$80 ksi

S_a = 12,000 psi for stainless steels

These stress amplitudes represent values based on 80 percent of the alternating stress intensity at 10^6 cycles for carbon steels and 60 percent 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 Figures I-9.1 and I-9.2 of Appendix I of ASME B&PV Code Section III.

3.9.1.1.5.3 Transient Vibration Acceptance Criteria

Analyses have been completed for the piping systems that are expected to experience significant operational transients (main steam piping, main steam SRV discharge piping, and feedwater piping). For these systems the calculated responses are the basis of the acceptance criteria for the measured transient response.

For other systems which transient testing is to be completed, the piping will be instrumented and/or visually inspected during the transient. If the acceptance criteria are exceeded, the source of the transient will be eliminated, the piping or restraints will be modified, or it will be proved by detailed measurement or analysis that the stresses are acceptable.

3.9.1.1.5.4 Thermal Expansion Acceptance Criteria

The piping and its appurtenances will not be constrained from expanding or contracting. All interferences will be resolved. During heatup, actual expansion movements will be within the greater of a specified tolerance of the calculated values or $\pm$0.25 in. Calculated or actual displacements of $\pm$0.25 in. or less will be ignored. At steady-state operating temperatures, the actual movements will be within a specified tolerance of calculated values. Discrepancies from these criteria will be resolved.

3.9.1.2 Dynamic Testing Procedures

A description of the tests or analyses used in the design of safety-related mechanical equipment (e.g., pumps, valves, and heat exchangers) to withstand seismic loadings is given in Subsections 3.7.2 and 3.7.3.

Most of the safety-related mechanical equipment is situated in the secondary containment and isolated from the reactor coolant pressure boundary (RCPB) by two or more isolation valves. Fluid dynamics and associated vibrations generated in the RCPB cannot propagate beyond closed isolation valves and their rigid anchorages at the point of containment penetration. Consideration of dynamic hydraulic transients generated within an emergency core cooling system (ECCS) subsystem is provided by establishing the following design criteria.

a. Piping and components not designed to withstand the dynamic effects of pipe whip, must be part of redundant, physically separated subsystems so that single failure of one subsystem does not affect the operability of the redundant subsystem

b. Where systems are subjected to potential vibratory loadings due to the dynamic effects of fluid momentum changes (i.e., water hammer), the following measures are taken to avoid the causes of such changes:

    1. Motor-operated valves in the ECCS are not capable of closing or opening at speeds greater than 1.0 in./ sec. Catastrophic failure is improbable for motor- operated valves. Where exception to the above is probable (such

as a break occurring in the feedwater line and the instant flow reversal causing the check valves to slam closed), a detailed analysis was made, and the valves and piping have been designed to withstand such an event

2. The ECCS and feedwater pumps are not capable of fast starts under normal operating conditions, because the lines are filled with fluid. Seizure of the prime mover (motor or turbine) is considered a single failure in the ECCS and renders the complete subsystem inoperative. Pressures and fluid velocities in the ECCS systems, except HPCI and reactor core isolation cooling (RCIC), are such that a water hammer stemming from pump motor seizure can be tolerated within the ASME Code faulted limits. For the feedwater system, the circumferential pipe rupture is identified to be the controlling event. The feedwater flow reversal and check-valve-closure transient resulting from this event were analyzed and the pressure surges, or peak pressure, for this transient were calculated to be less than 2.8 times the system design pressure. Accordingly, a faulted design pressure transient of 2.8 times the system design pressure is used in the ASME B&PV Section III NB-3656 analysis of the Class 1 portion of the feedwater systems

   Transient pressure surges, or peak pressures, associated with pump seizure are less than those associated with pipe rupture and are, therefore, not limiting. Similarly, the calculated transient pressure surges, or peak pressures, associated with pump seizure in the HPCI and RCIC systems are less than 2.5 times the system design pressure. Thus, a faulted design pressure transient of 2.5 times the system design pressure is used in the NB-3656 analysis of these systems

3. Air and steam voids that may develop in a stagnant system due to leakage are prevented in the RHR and core spray systems by providing pump discharge check valves and automatic condensate or demineralized water charging on the pump discharge piping. The HPCI pump discharge piping, up to the normally closed injection valve, is kept charged with condensate water. See section 6.3.2.2.5 for further discussion of the HPCI keep fill system. The RCIC pump lines do not need a charging system because the condensate storage tank provides the same function. The pump suction piping is pressurized by the condensate storage tank. RCIC discharges to the feedwater line from the pump. Thus, the water in the discharge piping cannot leak into the higher pressure feedwater line

   Although system vents are located at the piping high points, air pockets resulting from poor or inadequate system drainage, filling and venting during and after maintenance or prior to startup, could result in severe water hammer. To preclude this, Edison has included appropriate cautions in the applicable system operating procedures

4. The dynamic effects of rapid check valve closure in the feedwater piping due to feedwater line break have been analyzed. The frequency of this transient is so much higher than the natural frequency of the system that vibratory amplification of the equipment responses will not occur.

c. The piping systems have been designed and analyzed to accommodate thermal expansion due to system operational transients. Procedures will be instituted during the preoperational testing phase to verify the validity of the analytical predictions of pipe displacements by measuring pipe movement and comparing the field data to analytical predictions (see Subsection 3.9.1.1.4). It will also be verified that pipe supports and restraints are loaded within their design range.

3.9.1.3 Dynamic System Analysis Methods for Reactor Internals

3.9.1.3.1 Forcing Functions and Dynamic Response of Reactor Internals

The major reactor internal components are subjected to extensive testing, coupled with dynamic system analyses, to properly describe the resulting flow-induced vibration phenomena incurred from normal reactor operation and from anticipated operational transients.

In general, the vibration-forcing functions for operational flow transients and steady-state conditions are not predetermined by detailed analysis. Special analysis of the response signals measured from reactor internals of similar designs are performed to predict amplitude and model contributions. Parameter studies useful for extrapolating the results from tests of internals and components of similar designs are performed. This vibration prediction method is appropriate where standard hydrodynamic theory cannot be applied because of the complexity of the structure and flow conditions. Elements of the vibration prediction method are outlined as follows.

a. Dynamic analysis of major components and subassemblies is performed to identify natural vibration modes and frequencies. The analysis models used for Category I structures are similar to those outlined in Subsection 3.7.2, Seismic System Analysis

b. Data from previous plant vibration measurements are assembled and examined to identify predominant vibration response modes of major components. In general, response modes are similar but response amplitudes vary among BWRs of differing size and design

c. Parameters are identified that are expected to influence vibration response amplitudes among the several reference plants. These include hydraulic parameters such as velocity and steam-flow rates, and structural parameters such as natural frequency and significant dimensions

d. Correlation functions of the variable parameters are developed that, when multiplied by response amplitudes, tend to minimize the statistical variability between plants. A correlation function is obtained for each major component and response mode

e. Predicted vibration amplitudes for components for the prototype plant are obtained from these correlation functions based on applicable values of the parameters for the prototype plant. The predicted amplitude for each dominant response mode is stated in terms of a range, taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency are obtained from the dynamic analysis of Item a. in this listing.

The dynamic model analyses also form the basis for interpretation of the prototype plant preoperational and initial startup test results (Subsection 3.9.1.3.2). Model stresses are calculated and relationships are obtained between sensor-response amplitudes and peak-component stresses for each of the lower normal modes. The allowable amplitude in each mode is that which produces a peak stress amplitude of $\pm$10,000 psi.

3.9.1.3.2 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Fermi 2 reactor internals were tested in accordance with provisions of Regulatory Guide 1.20, Revision 2, for nonprototype Category I plants. The test procedure requires operation of the recirculation system at rated flow with internals installed (less fuel), followed by inspection for evidence of vibration, wear, or loose parts. The test duration was sufficient to subject critical components to at least 10^6 cycles of vibration during two- loop and single-loop operation of the recirculation system. At the completion of the flow test, the vessel head and shroud head were removed; the vessel was drained and major components were inspected on a selected basis. The inspection covered all components that were examined on the prototype design, including the shroud, shroud head, core support structures, the jet pumps, and the peripheral control rod drive and in-core guide tubes. Access was provided to the reactor lower plenum.

Reactor internals for Fermi 2 are substantially the same as the internals design configuration that was tested in prototype BWR/4 plants. Results of the prototype tests are presented in Reference 1. This report also contains additional information on the confirmatory inspection program.

3.9.1.4 Correlations of Reactor Internals Vibration Tests With the Analytical Results

Prior to initiation of the instrumented vibration test program for the prototype plant, extensive dynamic analyses of the reactor and internals were performed. The results of these analyses were used to generate the allowable vibration levels during the test. The results of the data analysis, vibration amplitudes, natural frequencies, and mode shapes were then compared to those obtained from the theoretical analysis.

Such comparisons provided the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained was used in the generation of the dynamic models for seismic and LOCA analyses for Fermi 2. The models used for Fermi 2 were the same as those used for the vibration analysis of the prototype plant.

The vibration test data were supplemented by data from forced- oscillation tests of reactor internal components to provide additional information concerning the dynamic behavior of the reactor internals.

3.9.1.5 Analysis Method Under Loss-of-Coolant Accident Loading

Annulus pressurization refers to the loading on the sacrificial shield wall, reactor vessel, reactor vessel supports, and reactor internals caused by a postulated pipe rupture at the nozzle safe ends. The assumed break is an instantaneous guillotine rupture that allows mass and energy release into the drywell and annular region between the shield wall and the reactor pressure vessel (RPV).

The mass and energy released during this postulated pipe rupture results in the following:

a. An acoustic asymmetric loading on reactor internals, due to a rapid decompression of the annular region between the vessel and the shroud

b. A transient asymmetric pressurization of the annular region between the shield wall and the RPV

c. A jet stream release of the RPV and pipe inventory

d. A force against the restraint attached to the shield wall due to the impact and constraint of the ruptured pipe

The study was broken into four tasks:

a. Calculation of mass energy release

b. Calculation of annulus pressure distribution history

c. Structural design assessment of the sacrificial shield and pedestal

d. Structural design assessment of the reactor components.

The first three tasks are described in detail in Subsection 6.2.1.3.11. This section is a brief description of these tasks and the results of the assessment of reactor components.

3.9.1.5.1 Mass and Energy Release

The postulated pipe rupture at the weld between recirculation or feedwater piping and the reactor nozzle safe end leads to a high flow rate of water and steam mixture into the annulus between the RPV and the shield wall. Figure 3.9-1 illustrates the location of this break. Calculation of the mass and energy release is performed using the generic method for short-term mass releases. This method is described in Subsection 6.2.1.3.11. As mentioned previously, this mass energy release results in acoustic loads, pressure loads, and jet loads.

3.9.1.5.2 Acoustic Loads

The recirculation suction line break is the most limiting break relative to the generation of asymmetric pressure loads on the shroud. The following pressure loads are used for input to the reactor internals stress analysis. There are two types:

a. Acoustic decompression wave loads that last for less than 5 msec; the method of the modeling of this load is consistent with NEDO-24048, "Evaluation of Acoustic Pressure Loads on BWR/6 Internal Components," September 1978 (Reference 2)

b. Flow-induced pressure loads due to the flow out of the break; these are analyzed by a potential flow theory analysis of the reactor downcomer region.

   Because the BWR is a two-phase system that operates at or close to saturation pressure, the differential pressure across the reactor shroud is of short duration and the structural supports of the system are not subjected to a significant shock-type load. This short-duration acoustic load is confined to a bending moment and shear force on the reactor shroud and reactor shroud support. Typical results of the integrated force acting on the reactor vessel shroud are given in Table 3.9-2. (These typical results apply to the Fermi 2 reactor.)

3.9.1.5.3 Pressure Loads

The pressure responses of the RPV-shield wall annulus for a recirculation suction line and a feedwater line were investigated using the COMPARE computer code. The pressure histories generated by COMPARE were in turn used to calculate the loads on the sacrificial shield wall and the RPV. Time-force histories representing the resultant loads on the RPV in the structural model were generated by taking the product of the pressure in each pressure node and its effective area, and summing these to give the force of the geometric center of each structural node (See Figure 3.9-3).

3.9.1.5.4 Jet Loads

To completely address structural loads on the vessel and internals, jet thrust, jet impingement, and pipe whip restraint loads must be considered in conjunction with the pressure loads. Jet thrust refers to the vessel reaction force that results as the jet stream of liquid is released from the break. Jet impingement refers to the jet stream force that leaves the broken pipe and impacts the vessel. Jet impingement and jet thrust forces are modeled as suddenly applied constant forces rising from a value of zero at time zero to its full value in one time step (0.001 sec). The pipe whip restraint load is the force that results when the energy-absorbing pipe whip restraint restricts the pipe separation to less than one full pipe diameter. These jet loads are calculated as described in ANSI 176 (draft), "Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Ruptures," January 1977 (Reference 3).

The jet-load forces used for the recirculation suction-line-break analysis are shown in Figure 3.9-2 and Table 3.9-3. These values were also used for the feedwater load evaluation. This is conservative because the calculation of these jet effects depends largely on the area of the break, and because the recirculation line is about 2.5 times larger in area.

3.9.1.5.5 Structural Dynamic Analysis

The pressure loads and jet loads described in Subsections 3.9.1.5.3 and 3.9.1.5.4 are combined to perform a structural dynamic analysis. Both of these loads are distributed along the horizontal beam model shown in Figure 3.9-3. The force-time- histories are then applied to a composite lump mass model of the pedestal, shield wall, and a detailed representation of the RPV and internals.

In each analysis, a multiple force input time-history is performed. The DYSEA computer program, described in response to La Salle NRC Item 111.61 (La Salle County Stations, NRC Docket No. 50-373-374), was used for the analysis. The maximum forces and moments at each end of the element were calculated for evaluating the RPV and internals at uprated power conditions. Acceleration time histories and the broaden response spectra at all nodes were generated and used for subsystem analysis. Only the horizontal excitations were generated for this analysis, since the AP loads are all horizontal.

The peak loading on the major components used to establish the adequacy of the component design is shown in Tables 3.9-5 and 3.9-6. A new set of time histories from the 12" recirculation discharge line break was provided for uprated power. The new AP analysis was done by using the same model with combined AP and jet loads. The maximum forces and moments at the RPV, shield wall, and pedestal location were obtained. The loads on major components are shown in Tables 3.9-5 and 3.9-6 for power uprate conditions.

3.9.1.5.6 Annulus Pressurization With a Safe-Shutdown Earthquake

A design analysis has been performed to evaluate the effect of such loading on the Fermi 2 RPV and internals. This evaluation accounted for the load combination of normal loads (NL) and annulus pressurization (AP) with a safe-shutdown earthquake (SSE). The AP and SSE are combined by the square-root-sum-of-squares (SRSS) method and added directly to normal loads and internal pressure differentials due to the line break.

The following safety-related RPV components were evaluated:

a. Top guide
b. Shroud
c. Core support
d. Jet pumps
e. Core ΔP line
f. RPV support (ring girder)
g. RPV stabilizer
h. Shroud support
i. Vessel skirt
j. Vessel stabilizer bracket
k. CRD housing
l. Control rod guide tube
m. Fuel assembly.

A comparison of loads resulting from this evaluation and allowable loads and stresses appears in Tables 3.9-7, 3.9-8, and 3.9-9. No allowable stresses are exceeded.

The critical buckling stress in the support skirt is equal to the allowable compressive stress determined in accordance with Article I-1150 of ASME B&PV Code Section III, 1968 Edition. The skirt is treated as a cylinder of radius equal to the largest skirt radius, and thickness equal to the thickness of one support skirt plate.

L_1 = Radius of skirt

t_n = Plate thickness

$$\frac{L_1}{100\ t_n} = 0.415 \qquad (3.9\text{-}1)$$

From Figure 1-1100 (B), factor B for SA-516-Gr70 material at design temperature of 575°F:

B = $S_{critical}$ = maximum allowable compressive stress for design conditions = -11.5 ksi.

According to paragraph N-417-10 of ASME B&PV Code Section III, the allowable compressive stress for emergency and faulted condition is increased by the same ratio as for other conditions.

$$S_{critical}\ (\text{emergency and faulted}) = B\left(\frac{S_y}{S_m}\right) \qquad (3.9\text{-}2)$$

$$= -11.5 * \frac{28.76}{19.15}$$

$$= -17.25\ \text{ksi}$$

The axial stresses in the skirt for original loads are

$$0_x = -\left(v_1 + v_2 + \frac{2M}{L_1}\right)\frac{1}{2\pi L_1 t_n \cos^u} \qquad (3.9\text{-}3)$$

= -8.8 ksi < $S_{critical}$ = -11.5 ksi (design)

= -14.7 ksi < $S_{critical}$ = -17.25 ksi (emergency and faulted)

To show design adequacy of the RPV support skirt, the resultant loads from the combination of responses due to LOCA and SSE are applied to the highest stressed point on the skirt. The skirt knee is the highest stressed portion of the RPV support. In comparing the loads due to the combination of plant-unique LOCA and SSE responses, it was found that the loads calculated for the original design of the vessel skirt are not exceeded. The calculated and allowable stresses for the support skirt are shown in Table 3.9-10.

The load combinations and maximum tensile forces in the RPV pedestal bolts are given in SL-3647 (Reference 4). Table 15 from SL-3647 gives the maximum forces in the RPV anchor bolts.

For new-loads evaluation, effective vertical load on the support skirt ($v_1 + v_2 + 2M/L_1$) is compared with the original load. The faulted-condition values are as follows:

	Original Load	New Load
Load $v_1 + v_2$	5213 kips	5196.5 kips
Moment M	1,344,000 in-kips	530,553 in-kips
Effective vertical load $v_1 + v_2 + \frac{2M}{L_1}$	30,262 kips	15,085 kips

The axial stresses for the new-load faulted condition are less than half the value of design faulted loads.

The efficiency of the dome segment in the penetration region is 43.8 percent.

The fuel assembly is modeled by seven axial nodes. The maximum acceleration occurring at each node is separately determined for the pressure reaction (PR), jet reaction (JR), and SSE loading. This results in three acceleration profiles. The acceleration profiles are then combined by taking the SRSS of the individual PR, JR, and SSE profiles at each axial position along the length of the fuel assembly.

The resulting profile is then compared to the design-basis profile. If the resultant acceleration profile is less than the design-basis profile, the resultant loads, moments, stresses, and deflection will be less than those for the design-basis case and therefore are acceptable. The acceleration components for Fermi 2 are shown in Table 3.9-11. (None of the accelerations exceed the design-basis profile.)

In addition, GE licensing topical report, "BWR Fuel Assembly Evaluation of Combined SSE and LOCA Loadings," NEDE-21175-P (Amendment 3), July 1982 (Reference 5), provides a bounding evaluation of potential fuel assembly liftoff during such loadings. The evaluation shows the fuel response is within acceptable limits as shown in Table 3.9-12 for the methodology used in the topical report.

Loads on the RPV shroud were studied for asymmetric pressure in addition to SSE, normal loads, and concurrent symmetric pressure differentials. The shroud buckling stresses for this loading combination are approximately 13 percent higher. These are still well below allowable stress limits.

At the request of the NRC, some piping was reanalyzed. These reanalyses confirmed that the piping stress evaluations for the large-bore RCPB piping systems considered faulted condition loadings, including annulus pressurization. This report further indicated that these analyses adequately represent the as-built configurations of these piping systems.

The as-built analysis of the recirculation and drywell RHR piping for combined annulus pressurization and DBE loadings showed that all piping stresses are within code allowable values (3 S_m), and all support component loads are within their Level D component ratings (with one exception where the rating was exceeded by a negligible [4.4 percent] amount). However, Edison has made minor modification (weld size increase) to structural steel for three supports to bring all supports into compliance with code allowable weld stress limits.

In addition, Edison has reviewed the annulus pressurization analysis of all other large-bore (NPS ≥4 in.) RCPB piping systems, comparing the analysis input to the as-built configuration. For these piping systems, the existing analyses were found to adequately represent the as-built configurations.

3.9.1.5.7 Steam Line Break With a Safe-Shutdown Earthquake

The simultaneous occurrence of a steam line break and an SSE also was analyzed. The analysis for core support structures was performed conservatively using the symmetric internal pressure differentials for a steam line break, which are higher than the symmetric internal pressure differentials for a recirculation or feedwater line break. The results are presented in Table 3.9-7.

3.9.1.5.8 Conclusion

In conclusion, a dynamic analysis of the RPV and internals has been performed considering loads due to a LOCA with an SSE. The results of this evaluation show that no RPV and internals allowable stresses are exceeded.

The current practice for such an analysis is to use a high degree of conservatism for each of the key parameters. As these parameters are combined during the evaluation, the degree of conservatism becomes magnified and the final results of the evaluation contain very high total conservatism. The following are some conservatisms for key parameters.

a. Dynamic loads are very conservatively defined in terms of amplitudes, frequencies, and phasing

b. To reduce the number of analysis cases, multiple-load cases are frequently combined into one by enveloping the input response spectra, which are more critical than the worst of individual cases

c. Damping values are conservatively specified

d. Response spectra peaks are broadened by ±15 percent

e. Linear analyses are performed in cases where nonlinear analyses are justifiable. Note that the nonlinear analysis generally results in significantly lower responses because of stress redistribution and higher energy dissipation

f. Allowable stresses used, as specified in the ASME Code, are based on static reserve margins, while for dynamic loads there are considerable additional reserve margins. Instantaneous or brief excursions into the inelastic range are of no or little structural consequences.

In addition to the above, the Fermi 2 evaluation made the following conservative assumptions.

a. The RPV/internals loads and equipment response spectra for the SSE were assumed to be 1.875 times the operating-basis earthquake (OBE) loads, which in effect ignores the higher damping allowed for SSE

b. Dampings used in the AP analysis were the same as those for OBE, although SSE dampings are more appropriate for loads associated with the faulted conditions. The RPV, shroud, and support skirt used 2 percent for OBE, SSE, and AP. Regulatory Guide 1.61 suggests dampings of 2 percent for the OBE and 4 percent for the SSE. Furthermore, in transmittal of structural properties for the reactor building, containment, pedestal, and shield wall, the architect-engineer suggests use of 2 percent damping for the OBE and 5 percent damping for the SSE. General Electric used 2 percent damping for the architect-engineer structures for OBE, SSE, and AP

c. A further conservatism exists in the use of recirculating jet loads in combination with all of the postulated subcompartment histories.

3.9.1.6 Analytical Methods for ASME Code Class l Components

Both elastic and inelastic stress analysis techniques were used in the design of the reactor vessel core support and reactor internal structures to show that stress limits are not exceeded. If an inelastic stress analysis was performed on these components, the elastic (linear) system analysis was checked to see whether the analysis requires modification. The procedure is first to determine the equivalent element stiffness including the inelastic component. The equivalent linear element stiffness is determined by using the method of equivalent linearization of Krylov and Bogoliubov (Reference 6). In this method, the nonlinear differential equation is replaced by an equivalent linear differential equation such that the solutions of the two equations differ from each other by an error of the order of the square of the nonlinear parameter. An alternative method is to determine the equivalent linear system by means of orthogonal polynomials (Reference 7). In either case, the fundamental frequency of the equivalent linear system is then determined. If the fundamental natural frequency of the equivalent linear system deviates less than 15 percent from that of the original linear system, the original linear analysis is considered adequate. A nonlinear dynamic analysis or an equivalent nonlinear dynamic analysis is performed if the natural frequency of the system with reduced stiffness deviates by more than l5 percent from that of the original system.

The l5 percent deviation criterion is applied to the system response for the particular component of interest. This is a realistic approach, since it is difficult, if not impossible, to discuss localized frequencies in dynamic analysis. The whole system must be considered when determining eigenvalues and eigenvectors. The 15 percent deviation criterion was selected in view of the uncertainties in the analytical models of structures and systems. Such uncertainties are normally accounted for in design by introducing conservatisms in the whole analytical process. For example, in the seismic analysis of the structure, floor spectra are generally broadened by 10 to 15 percent to account for uncertainties in the structural models, in the soil structure, and the system modeling. Because of the uncertainties in the dynamic model of the structure and the equipment, it is pointless to refine the analysis beyond the input uncertainty range.

Results for selected RPV internals and associated equipment analyses are provided in Table 3.9-10 and in Tables 3.9-13 through 3.9-15.

3.9.1.6.1 Method of Load Combinations for Class 1 Piping

ASME Code Section III, Class 1 piping systems and components are analyzed by elastic stress analysis techniques. The main computer programs used are PIPSYS or AutoPIPE, described in Subsections 3.13.1.26 and 3.13.3.18, respectively, and the GE in-house verified computer programs PISYS and ANSI7.

Forces and moments in the three normal orthogonal directions are determined for each individual load condition. Forces and moments are then combined in accordance with the rules of the ASME B&PV Code Section III, NB-3652 and NB-3653, in each orthogonal direction. The stress is then calculated on the basis of a single moment after combining the three orthogonal moments by the SRSS such that

$$M_i = \left(M_x^2 + M_y^2 + M_z^2\right)^{1/2} \quad (3.9\text{-}4)$$

The individual load and operating conditions used in the analysis are as follows:

Condition	Load	ASME Section III Criteria
Operating	Dead weight Design pressure	Primary stress intensity limit, Equation (9), NB-3652 $\leq$ 1.5 S_m
Upset	Dead weight Design pressure OBE	Primary stress intensity limit, Equation (9), NB-3652 $\leq$ 1.5 S_m (or 1.8 S_m depending on the code issue used)
	Thermal expansion Thermal displacement OBE	Primary plus secondary stress intensity range, Equation (10), NB-3653.1, or Equations (12) and (13), NB-3653.6
	Other upset occasional loads	Peak stress intensity range, Equation (11), NB-3653.2
	Operating and upset transients	Alternating stress intensity, Equation (14), NB-3653.6
Emergency	Dead Weight Design pressure SSE Other emergency occasional loads	Primary stress intensity limit, Equation (9), NB-3652 $\leq$ 2.25 S_m
Faulted	Dead weight SSE Other faulted occasional loads Design pressure	Primary stress intensity limit, Equation (9), NB-3652 $\leq$ 3.0 S_m

A fatigue analysis is performed in accordance with NB-3653.4 considering all cyclic load conditions, including pressure, hydrostatic testing, operating and upset transients, and OBE stress reversals. For some components, NB-3200 analysis is performed instead of the NB-3600 analysis described above. A typical listing of transients applied is given in Table 5.2-2.

The recirculation and main steam piping systems meet the requirements of the ANSI B31.7 Nuclear Power Piping Code Class 1. Other Class 1 piping systems meet the requirements of ASME Section III.

The analysis performed by GE consists of the recirculation loop and those portions of the main steam piping, the steam supply piping to the HPCI and RCIC turbines, the RHR supply and return piping, and reactor water cleanup (RWCU) piping inside the drywell. The analyses performed by GE include the optimization of suspension devices.

Other large bore (NPS ≥ 1-1/4 in.) Class 1 piping systems include the RCPB portions of the following:

a. Core spray system
b. Feedwater system (including HPCI, RCIC, and RWCU lines)
c. Main steam drains system
d. Standby liquid control system
e. RPV vent line
f. Outside containment portions of HPCI and RCIC steam lines, RHR supply and return lines, and RWCU line.

All of the above systems were analyzed to determine the forces and moments acting on each component as a result of thermal expansion, dead weight, and earthquake. In addition to the above, the main steam system was analyzed for relief valve lift and turbine stop valve closure. The moments obtained from these analyses were then used in conjunction with information obtained from an analysis of temperature gradients to determine the stress intensities and fatigue life for each component in the system.

3.9.1.6.2 Combination of Earthquake Response - Piping Systems

Modal responses and spatial components in seismic response analysis are combined using the methods described in Regulatory Guide 1.92, Revision 1.

3.9.1.6.3 Combination of Earthquake Loads With Other Occasional Mechanical Loads

Earthquake loads are combined with other occasional mechanical loads using the SRSS method.

3.9.1.6.4 Valves and Equipment

The requirements of the draft ASME Code for Pumps and Valves or ASME Section III were adhered to in the design of active Code Class 1 valves. Stress intensities were limited to 1.0 S_m for general membrane and 1.5 S_m for general membrane plus bending. These limits ensure that the valve stresses will remain within elastic limits and that no plastic deformation will occur. Representative analyses of Code Class 1 valves are summarized in Tables 3.9-17, 3.9-18, and 3.9-19.

The requirements of Section III of the ASME B&PV Code are adhered to in the design of Code Class 1 manually operated globe valves and check valves 2 in. in size and smaller.

Additional discussion relative to Code Class 1 equipment is provided in Section 5.2. Representative analyses of Code Class 1 pumps are summarized in Table 3.9-20, Recirculation Pumps.

3.9.1.6.5 Structural Supports

Structural supports for Class 1 components were generally designed using the same criteria as for Class 2 and 3 components as described in Subsection 3.9.2.2.4.1.

In the GE design of supplied supports where jet reactions from a break are included with SSE and normal loads, the calculated stress is less than the following:

a. Bending 0.9 yield

b. Tension 0.85 yield

c. Shear 0.50 yield

d. Compression 1.5 times the allowable stress from the AISC Specification Part 1, Paragraph 1.5.1.3.1.

Items that support components such as CRD housing supports and RPV stabilizers were designed using these acceptability criteria.

3.9.1.6.6 Stress Levels for Class 1 Piping Systems

The methods used to analyze Class 1 piping systems are discussed in Subsections 3.9.1.6.1 through 3.9.1.6.3. Typical results are presented here. As-built system data were used as input to the final stress analyses of these systems to ensure that the code- specified allowable stresses are not exceeded.

Piping isometrics, stress levels, and usage factors for the major Category I, Class l, systems are given in the following figures and tables:

System	Figure Numbers	Table Numbers
Main Steam	3.9-6 to 8	3.9-21 and 22
Recirculation – RHR	3.9-9 and 10	3.9-23 and 24
Feedwater	3.9-14	3.9-25
Core Spray	3.9-15	3.9-26

These isometrics and tables show the piping arrangement, stress levels, and usage factors at the high stress points, as well as at the locations of changes of flexibility. They are representative of the analyses and results for all Class 1 systems. The remaining Class 1 systems are not as critical as the previously listed systems, since failure or pipe rupture in these systems does not result in a design-basis LOCA. The Usage values listed in Tables 3.9-21 through 26 are based on original plant design. See FP FERM 310 (Ref. 19) for the fatigue usage accumulated for all monitored locations based on Fermi 2 operating history.

Any detailed information of specific results may be obtained from the certified stress reports for these systems.

3.9.2 ASME Code Class 2 and 3 Components

For Fermi 2 this refers to either ASME Code Class 2 and 3 components or similar non-RCPB safety-related pressure-retaining components designed to earlier codes.

3.9.2.1 Plant Conditions and Design Loading Combinations

These active and inactive components are identified and listed in Table 3.9-27. American Society of Mechanical Engineers Code Class 2 and 3 components of fluid systems were constructed in accordance with Section III of the ASME B&PV Code. Most components (piping, pumps, and valves) were ordered prior to July 1971 and were designed to other industry codes (see Table 3.2-1) when the effective Section III was not applicable. The specific quality group classification for each principal component is provided in Table 3.2-1. Tables 3.9-28 through 3.9-36 list the design loading combinations for the major components of representative safety-related systems.

Definitions of symbols used in the equations in these tables are contained in the applicable code referenced by the table.

3.9.2.2 Design Loading Combinations

The combination of design loadings for the components are categorized with respect to plant conditions identified as normal, upset, emergency, or faulted in Tables 3.9-28 through 3.9-36. The design stress limits associated with each of the plant conditions are specified in the subsections that follow.

3.9.2.2.1 Fluid System Components (Vessels Including Heat Exchangers and Pumps) and Piping Systems

ASME Code Class 2 and 3 safety-related fluid system components were designed considering the following load combinations:

Category	Loads	Pressure Boundary Stress Limits
Normal	Design pressure Design temperature	S = Allowable stress ASME Section III
Upset	Design pressure Design temperature	General membrane = 1.0 S
	Operating-basis earthquake Including nozzle loads	Local membrane/bending = 1.5 S
Emergency*	Design pressure Design temperature	General membrane = 1.2 S
	Safe-shutdown earthquake Including nozzle loads	Local membrane/bending = 1.8 S

* Inactive components may use the smaller of 0.7 S_u or 2.4 S

ASME Section III, Class 2 and 3, piping systems and components are generally analyzed by computerized elastic analysis techniques. The main computer programs used are PIPSYS, AutoPIPE, and NUPIPE, described in Subsections 3.13.1.26, 3.13.3.18, and 3.13.4.1, respectively.

Analysis is based on a single equivalent moment, evaluated as the SRSS combination of the three orthogonal moments generated in the pipe by the various loads, as described in Subsection 3.9.l.6.l. Earthquake response is calculated as described in Subsection 3.9.1.6.2. Earthquake loads are combined with other occasional mechanical loads as described in Subsection 3.9.1.6.3. The combined moments and resulting stresses are evaluated in accordance with the equations and allowable stress criteria of ASME III, Subsection NC-3652, for the various operating categories listed below.

Category	Loads	ASME III Criteria
Normal	Design pressure Dead weight Sustained mechanical loads	NC-3652.1 EQ (8) $\leq$ 1.0 S_h
Upset	Design pressure Dead weight Sustained mechanical loads Operating-basis earthquake OBE displacements* Occasional mechanical loads**	NC-3652.1 EQ (9) $\leq$ 1.2 S_h
Emergency	Design pressure Dead weight Sustained mechanical loads	NC-3652.2
	Safe-shutdown earthquake*** SSE displacements*, *** Occasional mechanical loads**	EQ (9) $\leq$ 1.8 S_h

* The earthquake terminal displacements may be neglected in EQ (9) if they are considered in Equation (10) or (11) as permitted by the Code.

** Such as relief valve blowdown loads.

*** As an alternative to using the SSE response in evaluating stresses under this category, 1.875 times the OBE response may be used.

In addition to the above primary loads, thermal expansion effects are considered by evaluation of Equation (10) or (11) of NC3652.3. The acceptance criteria for allowable stresses are as listed in the code.

Transients, that is, time-varying temperature or pressure changes, are not evaluated for Class 2 or 3 piping as they are for Class l piping. Transient phenomena, such as relief valve blowdown loads on the piping, are considered in the design when these events are specified in the System Design Specification.

The structural analyses for large- and small-bore torus-attached piping, piping supports, and related equipment are described in the Plant Unique Analysis Report and DC-6003 Vol I "Evaluation of New ECCS Suction Strainers on Existing TAP Analysis" for Torus-Attached Piping (Reference 8 & 20). Similarly, the structural analyses for the safety/relief valve discharge piping are described in Volume 5 of the Plant Unique Analysis Report (Reference 9) and in the piping stress reports. The criteria set forth in NUREG-0661 (Reference 10) have been used in the analysis methods and in the evaluation of the results for these systems.

3.9.2.2.2 Containment

Refer to Section 3.7 and Chapter 6.

3.9.2.2.3 Valves

The valve pressure-retaining parts designed to ASME III, Class 2 and 3, were designed to withstand seismic forces and pipe reactions of the SSE. If seismic consideration is necessary for other parts, the following applies:

Operating Condition	Loads
Upset	1. Normal operating 2. OBE
Emergency	1. Normal operating 2. SSE*

* As an alternative to using the SSE response, 1.875 times the OBE response may be used. Maximum horizontal ground acceleration for the SSE is 0.15g; for the OBE it is 0.08g. (see Subsection 3.7.1.1.)

The original design of ASME III, Class 2 and 3, valves is in accordance with MSS-SP-66 or ANSI-Bl6.5. Allowable stress limits are defined by ASME Section I. When more than one allowable stress value was listed in ASME Section I for an austenitic stainless steel material at a temperature, the lower value was used. The pressure-temperature ratings used for the design of valves are either the standard primary service pressure ratings of 150, 300, 400, 600, 900, 1500, or 2500 lb covered by ANSI-B16.5, or are determined by the following formula in compliance with the requirements of MSS-SP-66:

$$P_1 = \frac{PS_1}{S - P(y_1 - y)} \qquad (3.9\text{-}5)$$

where

P_1 = maximum allowable pressure at desired temperature, psi

P = maximum allowable pressure at design temperature, psi

S_1 = allowable stress at desired temperature (ASME Section I)

S = allowable stress at design temperature (ASME Section I)

Y_1 = plastic stress distribution factor at desired temperature

Y = plastic stress distribution factor at design temperature

In no case do the pressure-temperature ratings used for a weld- end valve exceed those given for weld-end valves in ASME III, articles NB-3530 and NB-3541. Valves purchased in accordance with more recent editions or Addenda of ASME Section III are designed in accordance with ANSI B16.34.

3.9.2.2.4 Nonpressure Parts

Parts that are not pressure boundary meet the requirements herein for supports and structures (Subsection 3.9.2.2.4.1), provision for anchor bolts (Subsection 3.9.2.2.4.2) and pressure boundary bolting subject to external loads (Subsection 3.9.2.2.4.3) as applicable.

3.9.2.2.4.1 Supports and Structures

Piping and equipment supports are designed for stress levels less than shown below for the loading condition defined for the pressure boundary:

a. Normal and upset

   1. Plate and shells - Primary membrane. 1.0 S; primary membrane plus bending, 1.5 S, where S is the allowable stress limit of the applicable code

   2. Linear supports and bolts - Stress less than the allowable limits of Part I, Section 1.5 through Section 1.10, of the AISC Specifications for the design, fabrication, and erection of structural steel for buildings

   3. Standard support components - Manufacturers' normal and upset condition rated capacity

   4. Concrete expansion anchor bolts - The average ultimate tensile and shear loads established by test divided by the following factors of safety.

      a. Four for wedge-type anchor bolts

      b. Five for self-drilling-type anchor bolts in pipe supports

      c. Four for self-drilling-type anchor bolts in applications other than pipe supports.

b. Emergency

   1. Plates and shells - Stress less than 1.2 times the allowable stress limit values for normal and upset above

2. Linear supports and bolts - Stress less than 1.33 times the allowable stress limit values for normal and upset above

3. Standard support components - Manufacturers' emergency condition rated capacity

4. Concrete expansion anchor bolts - Same as normal and upset above.

c. Load combinations

The load combinations used in the analysis of structural support and anchor bolts for safety-related components are as follows:

1.	Normal and upset conditions active components	Passive components	
	D + E + H + O	D + E + H	(3.9-6)
2	Emergency conditions active components	Passive components	
	$D + E^1 + O + H$	$D + E^1 + H$	(3.9-7)

where

D = dead load (flooded)

H = operating thermal effects

E = operating-basis earthquake

E^1 = safe-shutdown earthquake

O = operational loads (nozzle reactions, pressure, motor torque, pump inertia, etc., as applicable)

In the design of structures and structural components for Fermi 2, it has been Edison's practice to specify the use of codes and related design guides applicable at the initiation of the design activity. In the course of the design process, the use of later editions of the codes and/or any supplements issued thereto has been allowed.

3.9.2.2.4.2 Provision for Anchor Bolts

Equipment mounted on concrete support structures is fastened with anchor bolts (in drilled and grouted holes) or with expansion anchors. Sufficient holes for anchor bolts are provided to limit anchor bolt stress to those allowable per the AISC Code. Equipment anchored to a steel foundation and equipment mounted using expansion anchors follow the provisions of supports and structures (see Subsection 3.9.2.2.4.1).

3.9.2.2.4.3 Pressure Bolting for Component Flanges Subject To External Load

Where appreciable loads can occur on equipment-gasketed pressure joints, the external loads are considered in calculations to determine required bolt area. Code allowable stresses are maintained consistent with the operating conditions associated with the external loads.

3.9.2.2.5 Pipe and Equipment Supports

3.9.2.2.5.1 Equipment Supports

Refer to Subsection 3.9.2.2.4.2 above.

3.9.2.2.5.2 Seismic and Dynamic Effects - Shock Suppressors

Shock suppressors (snubbers) are provided on Category I piping systems, where necessary, to prevent shock forces from causing damaging motion and concurrently to allow for the normal thermal motion of the piping. In general, the snubbers for piping located inside the primary containment (drywell) and inside the steam tunnel, as well as the snubbers for the field run piping described in Subsection 3.9.2.7, are of the mechanical type. The mechanical shock suppressors conform with the requirements of the ASME Code Section III, Subsection NF, 1974 issue up to and including the winter 1976 addendum. Snubbers for the balance of the plant piping are of the hydraulic type and are designed in accordance with the requirements of ANSI B31.7, 1969, and/or ANSI B31.1, 1967, as appropriate for the class of piping being restrained.

As a result of concern about the reliability of the hydraulic and mechanical snubbers used for piping, identified as 10 CFR 50(e), Item 69, Edison reviewed the use of such snubbers. The results of the study included the elimination of about 29 percent of the snubbers, either by replacement with rigid supports or by proving that no restraint was required.

3.9.2.2.6 Relief Valve Operation - ASME III Components

If, during relief valve operation, the pressure exceeds the design pressure, it shall be considered to be an emergency condition and 110 percent of the design stress limit is permitted for Class 2 and 3 components.

3.9.2.2.7 Structural Cast Iron

The following are acceptable allowable stress limits for cast iron used for structures (e.g., bearing housings):

	Unidentified Gray Cast Iron	ASTM Class 20
Tension	3.5 ksi	5 ksi
Shear	3.5	5
Bending	5.25	7.5
Compression	7.0	10

3.9.2.2.8 Nozzle Loads

Nozzles withstand the pipe reactions from dead weight, thermal expansion, earthquake, and relief valve operation.

3.9.2.3 Design Stress Limits

For safety-related non-RCPB pressure retaining components, representative design stress limits are listed in Tables 3.9-28 through 3.9-36. Inelastic methods of analysis are not used for these components.

3.9.2.4 Analytical and Empirical Methods for Design of Pumps and Valves

3.9.2.4.1 General

To ensure the functional performance of Class 2 and 3 active pumps and valves, the design requirements of Subsection 3.9.2.2 were applied. Operability will be further demonstrated by the Operability Assurance Program described in Subsection 3.9.4.

The design methods were a combination of analysis and past testing and operating experience. These methods are the responsibility of the vendor, who is responsible for meeting the requirements of the applicable codes and standards identified in the component specification.

3.9.2.4.2 Valves

Class 2 and 3 (1971 code language) active valves were designed as described in Section 3.9.2.2.3. In addition, an analysis of the extended structure was performed, generally using statically applied acceleration loads from the piping stress analysis, for valves that are required to function during or after an SSE. For this analysis, stresses were limited to values that restrict the maximum stress in the extended structure to within upset condition code stress limits. Deflections of the extended structure will thus be small and operability of the valves will not be impaired.

3.9.2.4.3 Pumps

Active pumps were designed in accordance with the ASME Code for Pumps and Valves or the ASME B&PV Code for Nuclear Power Plants, depending on which code was in effect at the time the purchase order was issued. Forces resulting from seismic accelerations in the horizontal and vertical directions are included in the analysis of the pumps and their supports.

3.9.2.5 Design and Installation Criteria, Pressure-Relieving Devices

3.9.2.5.1 General

All pressure vessels are protected by pressure-relieving devices to meet applicable code requirements, such as ASME Code Sections III and VIII, and ANSI B31.1.

A discussion of the design and installation criteria for Class 1 pressure-relieving devices is given in Chapter 5. The derivation of the forcing functions that govern the fluid thrust during valve operation is presented in Subsection 3.9.2.5.2.

All ASME Code Class 2 and 3 overpressure relief valves and their connecting piping are designed to withstand the maximum load due to the discharge reaction force calculated by the following formula, regardless of the arrangement of the discharge piping:

$$F = 2PA \quad (3.9\text{-}8)$$

where

F = reaction force, lbf

P = valve setpoint pressure, lbf/in.^{2}

A = cross-sectional area of the valve inlet nozzle, in.^{2}

The discharge thrust loads so calculated were applied simultaneously with the loads due to internal pressure, dead weight, and seismic (SSE or OBE as applicable). When more than one relief valve is attached to a piping system, the loads due to all relief valves discharging simultaneously were applied to the system along with the above-mentioned primary loads. In addition, the loads from the most critical combination of valves discharging were applied.

3.9.2.5.2 Forcing Functions

The analytical basis for the forcing functions used in the dynamic and static analysis of the relief valves and connected piping is given in the following subsections.

3.9.2.5.2.1 Basic Fluid Flow Equations

One-dimensional flow is assumed in every straight pipe section. The conservation equations used are the following.

Mass

$$\frac{\partial \rho}{\partial t} + \frac{\partial}{\partial t}(\rho v) = 0 \quad (3.9\text{-}9)$$

Momentum

$$\frac{\partial v}{\partial t} + v\frac{\partial v}{\partial Z} = \frac{-g_c}{\rho}\left(\frac{\partial P}{\partial Z} + F'''\right) \quad (3.9\text{-}10)$$

Energy

$$\frac{\partial S}{\partial t} + v\frac{\partial S}{\partial Z} = \frac{v}{\rho T}F''' \quad (3.9\text{-}11)$$

where

ρ = fluid density, lbm/sec

t = time, sec

v = velocity, fps

Z = displacement, ft

g_c = 32.2, lbm x ft/(lbf x sec^2)

P = pressure, lbf/ft^2

S = entropy, ft-lbf/(lbm x °F)

T = temperature, °F

and

$$F''' = \frac{f}{D}\frac{\rho}{2g_c} v\,|v| \qquad (3.9\text{-}12)$$

where

f = Darcy friction factor

D = hydraulic diameter, ft

3.9.2.5.2.2 Reaction Forces

Reaction forces are considered to act longitudinally on each straight section of pipe rather than at each bend or turn, as shown in Figure 3.9-16(a). Consider a general straight section of pipe bounded by two other sections, Figure 3.9-16(b), and consider three fluid volumes from the pipes, Figure 3.9-16(c). Equation 3.9-10 is integrated over the pipe length:

$$\int_0^L \rho\left(\frac{\partial v}{\partial t} + v\frac{\partial v}{\partial Z}\right)\partial Z = -g_c \int_0^L \left(\frac{\partial P}{\partial Z} + F'''\right)\partial Z \qquad (3.9\text{-}13)$$

From Equation 3.9-9,

$$\frac{\partial v}{\partial t} + v\frac{\partial v}{\partial Z} = \frac{\partial}{\partial t}(\rho v) + \frac{\partial}{\partial t}(\rho v^2) \qquad (3.9\text{-}14)$$

so that

$$P_a - P_b - \frac{F_D}{A} = \frac{1}{g_c}\int_0^L \frac{\partial}{\partial t}(\rho v)\,\partial Z + \frac{(\rho v^2)b}{g_c} - \frac{(\rho v^2)a}{g_c} \qquad (3.9\text{-}15)$$

$$\frac{F_D}{A} = \int_0^L F'''\partial Z \qquad (3.9\text{-}16)$$

It is assumed that the turn sections are small in volume (compared to the straight pipe sections), so that no storage terms apply. Furthermore, lossless flow occurs (no friction or other irreversibilities).

Reactions R_1 and R_2 are parallel to pipe section L; R_L and R_R are parallel to the adjoining pipes at the left and right ends. It follows that

$$\rho_a A_a v_a^2 - \rho_1 A_1 v_1^2 \cos\alpha_1 = g_c(R_1 - R_L\cos\alpha_1 + P_1A_1\cos\alpha_1 - P_aA_a) \qquad (3.9\text{-}17)$$

$$\rho_2 A_2 v_2^2 \cos\alpha_2 - \rho_b A_b v_b^2 = g_c(P_bA + R_R\cos\alpha_2 - R_2 - P_2A_2\cos\alpha_2) \qquad (3.9\text{-}18)$$

Equations 3.9-17 and 3.9-18 give momentum conservation at left and right ends parallel to pipe L. Moreover, momentum conservation for the left and right ends in a direction normal to L is

$$-\rho_1 A_1 v_1^2 \sin\alpha_1 = g_c(-R_L\sin\alpha_1 + P_1A_1\sin\alpha_1) \qquad (3.9\text{-}19)$$

$$-\rho_2 A_2 v_2^2 \sin\alpha_2 = g_c(-R_R\sin\alpha_2 + P_2A_2\sin\alpha_2) \qquad (3.9\text{-}20)$$

Equations 3.9-15 and 3.9-17 through 3.9-20 combine to give the net longitudinal force on fluid in section L as

$$R_1 - R_2 - F_D = \frac{A}{g_c}\int_0^L \frac{\partial}{\partial t}(\rho v)\,\partial Z \qquad (3.9\text{-}21)$$

Reactions R_L and R_R are included with longitudinal loads on the adjoining pipe sections. Equation 3.9-21 gives the net force of the pipe walls on the fluid. The pipe load is equal and opposite, or

$$R_{bounded} = -\frac{A}{g_c}\int_0^L \frac{\partial}{\partial t}(\rho v)\,\partial Z \qquad (3.9\text{-}22)$$

Equation 3.9-22 gives the reaction load on a straight pipe section that is bounded at each end by two other adjoining pipes. However, if pipe L is open at the end designated "2," it follows that Equations 3.9-18 and 3.9-20 do not apply, and the pipe reaction load is

$$R_{open} = -AP_b + \frac{\rho_b v_b^2}{g_c} + \frac{1}{g_c}\int_0^L \frac{\partial}{\partial t}(\rho v)\,\partial Z \qquad (3.9\text{-}23)$$

For steady-state flows, Equation 3.9-22 gives zero reaction, whereas Equation 3.9-23 gives

$$R_{open,steady} = -AP_b + \frac{\rho_b v_b^2}{g_c} \qquad (3.9\text{-}24)$$

3.9.2.6 Stress Levels for Class 2 and 3 Piping

Piping isometric sketches, stress levels, and allowable stress limits for selected portions of the below-listed Category I, ASME III, Subsection 2 and 3 subsystems, are given in the following figures and tables.

System	Figure No.	Table No.
EECW system pump suction from heat exchanger	3.9-17	3.9-37
RHR service water return from heat exchanger	3.9-18	3.9-38
RHR containment spray from return header to drywell	3.9-19	3.9-39

These figures and tables indicate the piping system arrangement and stress levels at terminal ends and locations of high change in flexibility. The stress levels given are based upon final analyses and are typical of the stress levels predicted for all Category I, Class 2 and 3 systems. Any detailed information or specific results should be obtained from the specific stress analysis design calculations.

3.9.2.7 Field Run Piping Systems

Piping classified under ASME Code Section III, Classes 2 and 3, size 4 in. and under, with design temperatures of 575°F or less, is analyzed using the computerized stress analysis techniques described in Subsection 3.9.2.2.1, or is analyzed in the field using the simplified approach described in this section. For the field-designed piping, simplified analysis techniques were used for thermal, weight, and dynamic analyses and to determine restraint locations and design loads. These techniques are based on the following criteria.

a. Extreme conservatism is economically practical

b. Uncertainties in manufacturing are present so that a more precise analysis would not be useful

c. The pipe is flexible so that thermal expansion and nozzle movements can be easily accommodated

d. Similarities in the systems allow for use of standard components and design characteristics.

Based on these criteria, simplified analysis techniques pre-sented in Subsections 3.9.2.7.1 through 3.9.2.7.3 are used in the seismic, deadweight, and thermal analyses of field run piping systems. In field run piping systems, the stresses determined by these methods do not exceed the following values.

Stress	Maximum Value (psi)
Thermal expansion	15,000
Anchor movements	15,000
Thermal expansion plus anchor movements	15,000
Weight	3,000
Seismic (OBE)	7,000

Full consideration is given to seismic, weight, and thermal loadings imposed upon equipment and header nozzles to ensure that the imposed loads are within allowable limits.

3.9.2.7.1 Seismic Analyses

Simplified seismic analysis and design procedures are used, treating piping spans between rigid supports and/or restraints as independent simple beams. The span period, maximum mid-span deflection, allowable mid-span deflection, and end restraint forces are determined for a series of span lengths for each pipe size. No restraint credit is taken for hangers or restraints not offering stiffness in the direction of the seismic excitation. The maximum mid-span deflection and restraint forces are a function of the floor response spectra of the building structure in the vicinity of the piping. The spectra used are for the OBE. To predict the effects of the SSE, the responses are doubled.

The resulting data were developed into a set of design curves that are used to

a. Ensure that seismic stresses are not greater than the allowable

b. Ensure that seismic deflections are not large enough to cause damaging contact between pipe and surroundings

c. Provide seismic restraint design loads.

To ensure that seismic stresses are not greater than the allowable, and that seismic deflections are not large enough to cause damaging contact between pipe and surroundings, seismic deflection versus span curves similar to Figure 3.9-20 are used. These curves show the first mode seismic deflection of a simply supported beam representing the pipe span. The response is based upon the most energetic response spectrum expected in the building of interest.

A significant feature of the seismic deflection versus span curves is a line showing the deflection needed to produce a bending stress of 7000 psi in the pipe span. To ensure that code allowable stresses will be met when seismic loading is combined with other appropriate loadings, the seismic deflection due to the OBE is never allowed to exceed that shown by the 7000-psi curve. In addition, to protect against damaging contact between pipe and surroundings, seismic deflections greater than 2 in. are not allowed.

To provide seismic restraint design loads, seismic restraint load versus span curves similar to Figure 3.9-21 are used. These curves give the seismic restraint reactions for a span. Again, the model is a simply supported beam and the response is based on the response spectra curve as used in determining the seismic deflection curves. To account for the continuity of the piping across a restraint attachment point, the reactions from all piping spans supported from a restraint are added.

The following factors are considered in applying the design curves to actual piping systems. The major excitation due to earthquake will be horizontal. However, since it is not known from which horizontal direction the loading will come, all horizontal spans are restrained in the lateral and axial directions. Application of the above criteria requires that the maximum span distance between seismic restraints in the reactor building for the various size pipes does not exceed those spans given below:

	Maximum Span (ft)	
Pipe Size (in.)	Vertical	Horizontal
½	6	9
¾	7	11
1	8	12
1½	10	15
2	12	17
3	14	21
4	16	24

The simplified design curves are based on the accelerations associated with the OBE. The piping design criteria require that Class 2 and those Class 3 piping systems that are designated Category I satisfy normal code stress requirements during an earthquake of this intensity. The 7000-psi seismic stress limit was selected to ensure that normal code stress requirements can always be satisfied. To provide a seismic restraint design that is compatible with the piping design, the Class 2 and 3 system seismic restraints are designed in accordance with the AISC Manual.

A further requirement of the piping design criteria is that the designer must make an assessment of the effect of an earthquake of twice the intensity of the OBE or equal in magnitude to the SSE. The design goal for the SSE is to maintain a safe-shutdown capability for the nuclear energy system. Since the design curves are based on the OBE, all deflections, stresses, and reactions, as determined from the curves, are doubled to obtain SSE values. The allowable seismic stress for the SSE is obviously two times the OBE allowable, or 14,000 psi. The maximum allowable seismic deflection, however, remained at 2 in. The 14,000-psi seismic stress limit ensures that code stress requirements can be met. To provide a seismic restraint design that is compatible with the piping design, the Class 2 and 3 system seismic restraints are designed in accordance with the AISC Manual, except that the restraint stress shall not exceed the AISC allowable by more than 33 percent.

3.9.2.7.2 Weight Analysis

The standard procedure for designing a weight support system involves the use of recommended span lengths, to limit the weight-induced bending stress. The requirements of ASME Code Section III are satisfied by this approach even though the governing equations of NC-3652 do not directly indicate an allowable weight stress level.

In using this method, it is only necessary to determine the fraction of the allowable stress that the weight load should contribute. Recommended spans based on this allowable stress are then calculated by elementary bending theory. The effect of the weight of thermal insulation is also considered. The recommended span length is given by

$$\text{span(ft)} = \sqrt{\frac{2000Z}{W}} - 1 \qquad (3.9\text{-}25)$$

where

Z = section modulus, cubic in.

W = linear weight density, lb/ft

The span recommendations are listed in Table 3.9-40 for pipes filled with water and for gas-filled pipes. This formula is based on the assumption that the pipe element may be represented as a simply supported beam. The maximum bending moment for a continuous beam or for a beam with other end conditions cannot exceed the maximum for the chosen model. Therefore, although the analytical model may not always accurately represent the actual piping, it does establish an upper limit for the bending stress.

To accommodate concentrated weights, the spans are shortened to ensure that the allowable bending stress is not exceeded.

The following rules are used to determine span lengths.

a. A half-span of plain pipe must have a vertical support at one end

b. A half-span that includes an elbow must have a vertical support at both ends

c. The length of a half-span that includes a concentrated weight of less than 10 percent of the normal span load (Table 3.9-40) should not exceed 40 percent of the normal span length. It must have a vertical support on one end

d. A half-span that includes a concentrated weight of from 10 percent to 40 percent of the normal span load must have a vertical support on both ends

e. The length of a half-span that includes a concentrated weight of more than 40 percent but less than 250 percent of the normal span load should not exceed 10 percent of the normal span length and should have a vertical support on both ends. This length does not include the actual length of the concentrated weight

f. A concentrated weight that exceeds 250 percent of the normal span load should be supported directly rather than by the piping to which it is attached

g. The length of a half-span with one free end should not exceed 50 percent of the normal span length

h. A half-span with a concentrated weight should not have a free end

i. Supports not required by these rules should not be used.

3.9.2.7.3 Thermal Analysis

The object of the thermal expansion analysis was to ensure adequate flexibility so that nozzle movements and pipe expansion would not cause stresses in excess of the allowable. This allowable was determined by allocating a percentage of the allowable stress indicated by Equation (10) or (11) of NC-3652, to thermal expansion and anchor displacements.

Flexibility in a given direction is dependent upon the amount of pipe which is perpendicular to that direction. For instance, a component of nozzle movement in the X direction can be accommodated if the nozzle is separated from the first X direction restraint by enough piping in the Y and Z directions (Figure 3.9-22).

The bending stress in the perpendicular legs, B and D, must therefore be examined. These legs are conservatively modeled as a beam with guided ends subjected to a displacement (Figure 3.9-23). This model is conservative since it ignores the flexibility of the elbows and imposes more rigid end conditions than the actual supports.

The allowable stress is limited to 15,000 psi. Therefore the length of perpendicular pipe required to accommodate the component of nozzle movement is

$$\ell = 12\sqrt{\delta r_o} \qquad (3.9\text{-}26)$$

where

ℓ = length of perpendicular pipe required, ft

r_o = outside radius, in.

δ = deflection, in.

The recommended lengths for various values of δ and r are listed in Table 3.9-41. Also, a graph of r versus δ for various values of ℓ is given in Figure 3.9-24. In this analysis, nozzle movements are checked in three orthogonal directions.

The problem of pipe expansion can be handled in a similar manner, except that the movement, δ, is imposed by the expansion of a section of pipe.

As an example, the length of low-carbon steel (SA-106 grade B or equivalent) pipe required to accommodate the expansion of the length, ℓ', at 300°F is

$$\ell = 1.232\sqrt{\ell' r_o} \qquad (3.9\text{-}27)$$

where

ℓ' = expanding length of pipe, ft

ℓ = required offset, ft

r_o = outside radius of pipe, in.

Similar equations for both austenitic and ferritic steel at various design temperatures were developed. Values for different lengths and radii at several design temperatures for both austenitic and ferritic steels are listed in Table 3.9-42.

Where nozzle movement and pipe expansion can occur simultaneously, the sum of the lengths required for each is used as the total length of perpendicular pipe required.

3.9.3 Components Not Covered by the ASME Code

3.9.3.1 General

Safety-related mechanical components not covered by the ASME B&PV Code are identified in Table 3.9-43. The design codes for each principal component are identified and qualification methods for such equipment are summarized herein. This subsection specifically addresses (1) the details of the mechanical design and analytical procedures for the design of the fuel; (2) the methods and procedures used to determine the operability of the control rod drives and control rod insertability under LOCA and seismic loadings; and (3) mechanical and structural loading criteria for motors, the RCIC turbine, and active instrumentation designed to manufacturer's standards and design calculations.

3.9.3.2 Fuel Mechanical Design and Analytical Procedures

The fuel bundle performance history is specified by the cycle specific design reference fuel cycle as defined in Subsection 4.2.l. Performance of individual fuel rods is then determined from the fuel bundle performance history coupled with the exposure-dependent design, local and axial power, and exposure peaking factors. The most limiting fuel rods within the peak performance fuel bundle, with respect to power and exposure combination, are then analyzed to determine thermal and mechanical performance characteristics.

The performance of all fuel rods satisfies the requirements identified in Subsection 4.2.1. Satisfaction of these requirements for all fuel rods is demonstrated by analysis of the performance of the most limiting fuel rods, with respect to power and exposure level identified in the design reference fuel cycle.

Thermal design analyses performed include, but are not limited to, the determination of cladding (Zircaloy-2) and fuel (UO_2) temperatures, cladding and fuel thermal expansion, fuel irradiation swelling, and fuel fission gas generation and release as a function of time. Using these thermal analysis results, the mechanical design analyses are then performed to determine the most limiting cladding stress and/or strain due to such loadings as

a. Internal fuel rod pressure from gaseous fission product release to the fuel rod plenum plus initial fill gas

b. Differential fuel-cladding expansions

c. External coolant pressure

d. Flow-induced rod vibrations.

Finally, the limiting combinations of cladding stress in the categories summarized in Subsection 4.2.1 are identified and compared to the cladding design stress limits. All stresses are below the defined limits.

3.9.3.3 Control Rod Drive Operability and Control Rod Insertability Under LOCA and Seismic Loadings

In the event of a significant seismic disturbance and/or LOCA, only the rapid insertion mode (scram) is essential. Descriptions of the CRD and the CRD system operation during scram are presented in Subsection 4.5.2.2.

The hydraulic nature of the CRDs and their location relative to the reactor vessel provide scram operability that is insensitive to LOCA and seismic loadings. In addition, insertability of the control rods during seismic events is ensured by the generous control-rod-to-channel and control-rod-to-guide tube clearances. However, LOCA produces larger than normal pressure differentials across the reactor vessel internals, tending to reduce these clearances. These pressure differentials are considered in determining the insertability of the control rods.

The highest pressure differentials across the RPV internals occur as a result of a postulated steam line break. To ensure adequate control-rod-to-guide tube clearance, the guide tube must be capable of resisting the external to internal pressure difference without collapse. In addition, any increase of friction force due to channel bulging is shown to be small compared to the total force available to insert the control rods. The above are addressed in Subsections 4.2.2 and 4.2.3. The adequacy of the design margins of the control rod guides to prevent control tube collapse in the event of a main steam line break or recirculation line break (LOCA) was noted as a concern by the AEC in its Safety Evaluation Report on the Fermi 2 Construction Permit (Reference 11). The concern was identified as Post-Construction Permit Open Item No. 9. The Edison position on this open item was submitted to the AEC in May 1973 (Reference 12). The position was based on information supplied by GE and concluded that design margins of the control rod guide tubes were adequate and that no collapse under normal or abnormal conditions was expected. The AEC, after reviewing Reference 12, requested additional information in the form of five specific questions (Reference 13). Edison responded to these five questions on February 14, 1974 (Reference 14), and received AEC provisional approval to start construction by AEC letter dated June 25, 1974 (R. DeYoung to H. Tauber).

3.9.3.4 Mechanical and Structural Loading Criteria for Equipment Not Covered by ASME Code

For nonpressure-retaining equipment important to safety (i.e., motors, the RCIC turbine, and active instrumentation), the following criteria apply.

3.9.3.4.1 Reactor Core Isolation Cooling Turbine

The turbine mechanical and structural loading criteria are given in Table 3.9-35.

3.9.3.4.2 Motors, Motor Control Centers, Switchgear, and Diesel Generators

These components are designed to meet the support and structures criteria (Subsection 3.9.2.2.4.1) and provision for anchor bolts (Subsection 3.9.2.2.4.2) for the following.

Operating Conditions	Loads
Normal	Normal operating + dead weight
Upset	Normal operating + dead weight + OBE
Emergency and Faulted	Normal operating + dead weight + SSE

3.9.3.4.3 Air-Handling Equipment (Safety Related)

The following air-handling systems require equipment satisfying the requirements of this section:

a. Standby gas treatment train
   1. Exhaust fans
   2. Carbon dioxide tanks
   3. Decay heat removal fans
   4. Room cooling units.

b. Emergency equipment area cooling units
   1. ECCS pump room cooling units
   2. Switchgear room cooling units
   3. Emergency equipment cooling water (EECW) pump area cooling units
   4. Thermal recombiner area cooling units.

c. Control center air conditioning system (CCACS)
   1. Multizone unit
   2. Return air fans
   3. Chillers
   4. Chilled water pumps
   5. Equipment room cooling units
   6. Emergency makeup air filter
   7. Emergency recirculation air filter
   8. Emergency recirculation air filter fans.

The above equipment includes fans, housings, ducts, coils, dampers, drives, motors, and plenums (coils are ASME Section III items except for control center equipment room cooling units, and cooling coil in multizone unit). All associated safety-related components and accessories have been designed to Category I requirements.

These components have provisions for installation to meet support and structure criteria of Subsection 3.9.2.2.4.1 and provision for anchor bolts per Subsection 3.9.2.2.4.2. These components are located in Category I buildings and are supplied with electrical power sources and utility services (control air, water, and drains) of Category I classifications. In addition, the buildings provide flood, tornado, wind, missile, and dynamic effects of ruptured piping protection to the air-handling equipment.

3.9.4 Operability Assurance Program

3.9.4.1 General

Active mechanical equipment classified as Category I is designed to perform its function during the life of the plant under postulated plant conditions (See Appendix B, Section B, for discussion of operation beyond the original design plant life). Equipment with faulted-condition functional requirements includes active pumps and valves in fluid systems, such as the RHR system, core spray system, and the HPCI and RCIC systems. Operability has been ensured by a series of comprehensive preoperational tests.

Certain Category I valves and pumps were procured before Branch Technical Position MEB position papers concerning operability assurance (References 15 through 18) were available. The codes that were used in the procurement of these components are given in Tables 3.2-2 and 3.2-3. Table 3.9-44 provides a comparison of the Fermi 2 operability assurance program criteria to those provided in NRC Standard Review Plan 3.9.3.

3.9.4.2 ASME Code Class Valves

Safety-related active valves perform their mechanical motion in times of an accident. Assurance is therefore required that these valves will operate during a seismic event. Qualification tests accompanied by analyses have been conducted for all active valves in the GE scope-of-supply.

All other safety-related code Class 1, 2, and 3 active valves equipped with motor operators have been operationally qualified by a combination of test and analysis. Prototype tests have been performed for motor operators situated inside the primary containment and the steam tunnel and subjected to faulted environmental conditions associated with a LOCA. These tests are essentially consistent with the guidelines of IEEE-382, l972. The specific conditions are as follows.

Conditions	Test Results
Seismic operational capability	Up to 5.0g (two planes)
Radiation environment	2 x 10^8 rad
Pressure-temperature environment	IEEE-382, BWR profile
Humidity	100 percent steam atmosphere

Motor-operated active valves located outside the primary containment are equipped with identical motor operators, except that motor insulation is Class B NEMA rated for 130°C service and 100 percent humidity. The operability of the motor valve assembly is ensured by analytical methods.

Each valve type and size has been analyzed, as described in sections 3.9.1 and 3.9.2, to ensure that design loads do not render the valve inoperative. In addition, the below-described preservice and inservice testing is conducted.

The safety-related valves are subject to a series of stringent tests prior to service and during the plant life (See Appendix B, Section B, for discussion of operation beyond the original design plant life). Prior to installation, the following tests are performed: shell hydrostatic test to code requirements, backseat and main seat leakage tests, disk hydrostatic test, functional tests to verify that the valve will open and close within the specified time limits when subjected to the design differential pressure, and operability qualification of valve actuators.

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.

Valves that are safety related, but can be classified as not having an overhanging structure, such as check valves and safety/ relief valves, are considered separately.

Because of the particularly simple characteristics of the check valves, they are qualified by a combination of the following tests and analysis:

a. Stress analysis including the seismic loads where applicable

b. In-shop hydrostatic test

c. In-shop seat leakage test

d. Periodic in-situ valve exercising and inspection, as applicable, to ensure the functional capability of the valve.

Safety/relief valves are also subjected to tests and analyses similar to check valves. These consist of 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 (Technical Specifications).

During a seismic event, it is anticipated that the seismic accelerations imposed upon the valve may cause it to open momentarily and discharge under system conditions which otherwise would not result in valve opening.

Using the methods described, all the safety-related valves in the systems are qualified for operability during a seismic event. These analytical methods conservatively simulate the seismic event and ensure that the active valves will perform their safety-related function when necessary.

3.9.4.3 ASME Code Class Pumps

No active pumps are located inside the primary containment. Those active pumps located in the secondary containment and subject to adverse environmental conditions as a result of high- energy and moderate-energy pipe breaks outside the primary containment are discussed in Section 3.6.

All active pumps are qualified for operability by first being subjected to extensive tests, both prior to installation in the plant and after installation in the plant. The in-shop tests include the following:

a. Hydrostatic tests of pressure-retaining parts to 1.25 times the design pressure times the ratio of material allowable stress at the test temperature to the allowable stress value at the design temperature

b. Seal leakage tests

c. 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 operational tests. These tests demonstrate reliability of the pump for the design life of the plant (See Appendix B, Section B, for discussion of operation beyond the original design plant life).

In addition to these tests, the safety-related active pumps have been analyzed for operability during a seismic condition by ensuring that the pump will not be damaged during the seismic event, and the pump will continue operating despite the seismic loads. Performing these analyses, with the conservative loads stated and with the restrictive stress limits discussed in Subsection 3.9.2 as allowables, will ensure that critical parts of the pump will not be damaged during the seismic condition. Therefore, the reliability of the pump for postseismic-condition operation will not be impaired by the seismic event.

The pump/motor rotor combination is designed to rotate at a constant speed under all conditions. Because of the high rotary inertia in the operating pump rotor, and the nature of the random short duration loading characteristics of the seismic event, the seismic loading will cause only a slight increase in the torque necessary to drive the pump at the constant design speed.

Furthermore, a generic analysis was performed for motor-driven, vertically mounted RHR and core spray pump motor assemblies to determine shaft and rotor deflections associated with the SSE forces, and to assess the operability of rotating equipment during a seismic event. The results show negligible effect for perpendicular and axial rotor loads equivalent to l.5g static acceleration, which is significantly higher than the resonance equipment response peak of the applicable Fermi 2 floor response spectrum.

The HPCI pump is also analyzed, but because of its rigidity, the analysis of deflections is limited to alignment with the driver.

The functional ability of active pumps after a seismic condition is ensured, 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 postseismic-condition operating loads will be no worse than the normal plant operating limits. This is ensured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and postseismic conditions are limited by the magnitudes of the normal condition nozzle loads.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

REFERENCES

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14. Edison Letter EF2-21952, dated February 19, 1974, C.M. Heidel to A. Giambusso, Enrico Fermi Atomic Power Plant, Unit 2, AEC Docket No. 50-341, Control Rod Guide Tube Collapse.
15. MEB Position - Class I Valve Operability Assurance Program, USAEC, May 24, 1973.
16. MEB Position - Class 2 and 3 Valve Operability Assurance Program, USAEC, May 24, 1973.
17. MEB Position - Class 1 Pump Operability Assurance Program, USAEC, May 24, 1973.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

REFERENCES

18. MEB Position - Class 2 and 3 Pump Operability Assurance Program, USAEC, May 24, 1973.

19. FP FERM 310, Fermi FP4-FERM-BUILD (1501462).

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TABLE 3.9-1 VIBRATION AND DYNAMIC EFFECTS TEST PROGRAM SYSTEMS TEST LEVEL MATRIX

	System	Code Class (Note a)	Test Levels (Note b) Type A	Type B	Type C	Type D
1.0	Nuclear boiler system					
1.1	Selected main steam SRV discharge piping	III-3	--	X	C.2	--
1.2	Main steam piping from reactor to primary containment outboard isolation valves	B31.7-1	--	X	C.1, C.3	--
1.3	Feedwater piping within outermost isolation valves	III-1	--	X	C.1	--
2.0	Reactor recirculation system					
2.1	Piping	B31.7-1	--	X	C.1, C.3	X
3.0	CRD hydraulic system					
3.1	Scram discharge volume and header	III-2	A.1	--	--	--
3.2	Insert and withdraw lines	III-2	A.1	--	--	--
3.3	Water supply piping	B31.1.0	A.1	--	--	--
4.0	Standby liquid control system					
4.1	Piping within isolation valves	III-1	A.1	--	--	--
4.2	Pump discharge piping beyond isolation valves	III-3	A.1	--	--	--
4.3	Pump suction piping	III-3	A.1	--	--	--
5.0	Residual heat removal system					
5.1	Other piping within outermost isolation valves	III-1	A.1	X	--	--
5.2	Piping beyond outermost isolation valves	III-2	A.1	X	--	X
6.0	Core spray system					
6.1	Piping within outermost isolation valves	III-1	A.1	X	--	--
6.2	Piping beyond outermost isolation valves	III-2	A.1	--	--	X
7.0	High-pressure coolant injection system					
7.1	Turbine steam supply piping within outermost isolation valves	III-1	--	X	C.1	--
7.2	Turbine steam supply beyond outermost isolation valve and exhaust piping	III-2	A.2	X	--	X
7.3	Suction line from condensate storage tank	III-2	A.2	--	--	--
7.4	Return line to condensate storage tank	B31.1.0	A.2	--	--	--
7.5	HPCI pump discharge piping	III-2	A.2	--	C.3	X
8.0	Reactor core isolation cooling system					

TABLE 3.9-1 VIBRATION AND DYNAMIC EFFECTS TEST PROGRAM SYSTEMS TEST LEVEL MATRIX

	System	Code Class (Note a)	Test Levels (Note b) Type A	Type B	Type C	Type D
8.1	Turbine steam supply piping within outermost isolation valves	III-1	--	X	C.1	--
8.2	Turbine steam supply beyond outermost isolation valve and discharge piping	III-2	A.2	X	--	X
8.3	Suppression pool suction and pump discharge piping	III-2	A.2	--	--	X
8.4	Suction line from condensate storage tank	III-2	A.2	--	--	--
8.5	Return line to condensate storage tank	B31.1.0	A.2	--	--	--
9.0	Reactor water cleanup system					
9.1	Piping within outermost isolation valves	III-1	A.1	X	--	--
9.2	Piping from containment penetration to the heat exchangers	B31.1.0	A.1	X	--	--
10.0	Fuel pool cooling and cleanup system					
10.1	Cooling loop piping	III-3	A.1	--	--	--
11.0	RHR service water system			--	--	--
11.1	Piping	III-3	A.1			
12.0	Plant service and cooling water systems					
12.1	Emergency equipment cooling water system	III-3 (Note c)	A.1	--	--	--
12.2	Emergency equipment service water system	III-3	A.1	--	--	--
13.0	Emergency diesel generator systems					
13.1	Fuel oil system piping	III-3	A.1	--	--	--
13.2	Service water system piping	III-3	A.1	--	--	--
14.0	Power conversion system					
14.1	Main steam piping from outboard MSIV to turbine stop valve	B31.1.0	--	X	C.1, C.4	--
14.2	Main steam piping to RFP turbine	B31.1.0	A.2	X	--	--
14.3	Main steam dump line	B31.1.0	A.2	X	--	--
14.4	Feedwater piping from reactor feed pumps to outboard isolation valves	B31.1.0	A.2	X	C.3	--
14.5	Main steam drains	B31.1.0	A.2	X	--	--
15.0	Radwaste system					

TABLE 3.9-1 VIBRATION AND DYNAMIC EFFECTS TEST PROGRAM SYSTEMS TEST LEVEL MATRIX

	System	Code Class (Note a)	Test Levels (Note b) Type A	Type B	Type C	Type D
15.1	Drywell and reactor building sump pumps discharge piping	III-2, B31.1.0	A.1	--	--	--
16.0	Offgas system					
16.1	Piping	B31.1.0	A.1	--	--	--
17.0	Control air system					
17.1	Piping	III-3	A.1	--	--	--
18.0	Control center air conditioning system					
18.1	Condenser piping	III-3	A.1	--	--	--
18.2	Chilled water piping	B31.1.0	A.1	--	--	--

NOTES:

a. System Code Class

Notations for principal construction codes are

III-1, 2, 3 - ASME Boiler and Pressure Code Section III, Class 1, 2, or 3

B31.7-1 - ANSI Nuclear Power Piping Code Class I

B31.1.0 - ANSI B31.1.0 Standard Code for Pressure Piping, Power Piping.

b. Levels of Testing - The designations in this table refer to the following specific paragraphs:

Type A: Visual Monitoring – The vibration surveys conducted will visually monitor deflections of selected points. Acceptable vibratory response of the overall system will be verified also. The vibration testing will be performed during:

A.1 – Preoperational test phase

A.2 – Startup test phase

Type B: Thermal Expansion

X – Observation or recording of the thermal expansion movements of key points on the piping will be made during startup test phase. Testing will be conducted during both heatup and cooldown phases of system operation.

c. That portion of EECWS piping between the outboard isolation valves and components inside primary containment is ASME Section III, Class 2.

Type C: Vibration Measurement - Acceptable overall vibratory response of the system will be verified. The vibration surveys conducted will entail the following:

TABLE 3.9-1 VIBRATION AND DYNAMIC EFFECTS TEST PROGRAM SYSTEMS TEST LEVEL MATRIX

C.1 – Steady-state vibration measurement, using mechanical devices, of maximum deflection at selected points during the startup test phase

C.2 – Measurement, using mechanical devices and remote recording devices, of vibration and deflection at selected main steam SRV discharge line

C.3 – Measurement of the piping system vibration and structural deflection, and piping system transient pressure levels and forces at selected points on the piping system, will be conducted during startup test phase transient tests

C.4 – Measurement of the piping system transient vibration and structural deflection and piping system transient pressure levels and forces at selected points on the piping system will be conducted during an inadvertent turbine trip after the startup test program is completed.

Type D: Rotating Equipment Vibration Testing

X – Baseline vibration data will be obtained for the rotating equipment associated with this piping. See Subsection 3.9.1.1.2 for an inclusive list of the rotating equipment that will be tested.

TABLE 3.9-2 ACOUSTIC LOADING ON REACTOR PRESSURE VESSEL SHROUD

Time (msec)	Acoustic Load (Kips)
0	0
1.2	0
1.6	150
2.0	320
2.5	650
2.8	250
3.0	100
3.2	0

TABLE 3.9-3 JET LOAD FORCE DATA

Characteristics	Measurements
Effective clearance (in.)	1.000
Pipe bending strain limit (in./in.)	0.08207
Impact velocity (fps)	15.42
Number of bars composing the restraint	2
Force on restraint in direction of thrust (lb)	765,924
Total energy absorbed by the restraint (ft-lb)	266,301
Energy absorbed by the top hinge (ft-lb)	0
Length from restraint to break (ft)	4.020
Pipe rotation stability limit (deg.)	7.0530
Deflection of structure in direction of thrust (in.)	0.7659
Force on structure in direction of thrust (lb)	765,924
Energy absorbed by the structure (ft-lb)	24,443
Restraint load (peak) components (lb)	
PD1	765,924
PD2	0
Restraint loading direction (deg.)	0
Maximum allowable bending moment (ft-lb)	1,943,235
Impact time (sec)	0.0098
Deflection of restraint in direction of thrust (in.)	5.1548
Time at peak dynamic load (sec)	0.0559
Energy absorbed by the bottom hinge (ft-lb)	10,195

TABLE 3.9-3 JET LOAD FORCE DATA

Restraint load (static) components (lb)	
PS1	200,266
PS2	0
Relative deflection of pipe end in the direction of the thrust (in.)	3.8487
Deflection time for pipe end after impact (sec)	0.0330
Energy absorbed by the restraint hinge (ft-lb)	158,535
Pipe deflection at restraint components (in.)	
XR1	6.9207
XR2	0
Total deflection of the pipe end in the direction of thrust (in.)	11.5563
Total time of movement (sec)	0.0559
Total absorbed energy (ft-lb)	459,474
Pipe deflection at the break components (in.)	
XP1	11.5563
XP2	0

TABLE 3.9-4 JET LOAD FORCES USED FOR RECIRCULATION SUCTION LINE BREAK ANALYSIS[a]

Time (sec)	Pipe Displacement at Restraint (in.)	Pipe Velocity at Restraint (fps)	Pipe Acceleration at Restraint (fps)	Relative Displacement of End (in.)	Total Displacement of End (in.)	Restraint Load Component PD1 (lb)	Restraint Load Component PD2 (lb)	Blowdown Force (lb)
0.00255	0.1000	5.102	1,458.	0	0.1114	0	0	476,820
0.00390	0.2000	7.050	1,437.	0	0.2227	0	0	476,820
0.00496	0.3000	8.564	1,428.	0	0.3341	0	0	476,820
0.00586	0.4000	9.845	1,423.	0	0.4455	0	0	476,820
0.00665	0.5000	10.98	1,419.	0	0.5569	0	0	476,820
0.00737	0.6000	12.00	1,416.	0	0.6682	0	0	476,820
0.00804	0.7000	12.94	1,414.	0	0.7796	0	0	476,820
0.00866	0.8000	13.82	1,412.	0	0.8910	0	0	476,820
0.00924	0.9000	14.64	1,410.	0	1.002	0	0	476,820
0.00980	1.000	15.42	1,409.	0	1.114	0	0	476,820
0.01080	1.184	15.24	-243.1	0.02330	1.342	151,025	0	476,820
0.01180	1.365	14.87	-427.8	0.08668	1.607	277,941	0	476,820
0.01280	1.541	14.43	-424.8	0.1848	1.900	346,631	0	476,820
0.01380	1.711	14.03	-372.2	0.3101	2.216	393,033	0	476,820
0.01480	1.878	13.70	-306.9	0.4583	2.550	427,999	0	476,820
0.01580	2.040	13.43	-242.3	0.6238	2.896	456,146	0	476,820
0.01680	2.200	13.22	-183.9	0.8021	3.252	479,808	0	476,820
0.01780	2.358	13.06	-133.5	0.9891	2.615	500,312	0	476,820
0.01880	2.514	12.95	-917.4	1.182	3.981	518,484	0	476,820
0.01980	2.669	12.88	-585.5	1.376	4.349	534,865	0	476,820
0.02080	2.823	12.84	-334.1	1.571	4.715	549,830	0	476,820
0.02180	2.977	12.81	-156.6	1.764	5.079	563,644	0	476,820
0.02280	3.131	12.80	-4.582	1.952	5.439	576,503	0	476,820
0.02380	3.284	12.80	-0.5496	2.135	5.793	588,552	0	476,820
0.02480	3.438	12.80	-0.4494	2.311	6.140	599,902	0	476,820
0.02580	3.592	12.80	-4.203	2.480	6.480	610,637	0	476,820
0.02680	3.745	12.79	-12.78	2.640	6.811	620,826	0	476,820
0.02780	3.899	12.77	-24.71	2.790	7.132	630,521	0	476,820
0.02880	4.052	12.74	-39.46	2.932	7.444	639,763	0	476,820
0.02980	4.204	12.69	-56.57	3.063	7.745	648,587	0	476,820
0.03080	4.356	12.63	-75.60	3.184	8.035	657,020	0	476,820
0.03180	4.507	12.54	-96.20	3.294	8.314	665,086	0	476,820

TABLE 3.9-4 JET LOAD FORCES USED FOR RECIRCULATION SUCTION LINE BREAK ANALYSIS[a]

Time (sec)	Pipe Displacement at Restraint (in.)	Pipe Velocity at Restraint (fps)	Pipe Acceleration at Restraint (fps)	Relative Displacement of End (in.)	Total Displacement of End (in.)	Restraint Load Component PD1 (lb)	Restraint Load Component PD2 (lb)	Blowdown Force (lb)
0.03280	4.657	12.43	-118.0	3.394	8.581	672,804	0	476,820
0.03380	4.805	12.30	-140.8	3.484	8.836	680,188	0	476,820
0.03480	4.952	12.15	-164.3	3.563	9.079	687,253	0	476,820
0.03580	5.097	11.97	-188.2	3.633	9.309	694,010	0	476,820
0.03680	5.239	11.77	-212.5	3.692	9.527	700,468	0	476,820
0.03780	5.379	11.54	-236.9	3.741	9.732	706,636	0	476,820
0.03880	5.516	11.29	-261.4	3.781	9.924	712,521	0	476,820
0.03980	5.650	11.02	-285.8	3.811	10.10	718,130	0	476,820
0.04080	5.780	10.72	-310.0	3.832	10.27	723,467	0	476,820
0.04180	5.907	10.39	-334.0	3.845	10.42	728,538	0	476,820
0.04280	6.030	10.05	-357.8	3.849	10.56	733,346	0	476,820
0.04688	6.454	7.501	-638.7	3.849	11.04	748,107	0	476,820
0.05403	6.878	2.924	-615.0	3.849	11.51	763,044	0	476,820

[a] Except for the restraint load components PD1 and PD2, all variables are in a direction parallel to the blowdown force.

TABLE 3.9-5 FERMI 2 MAXIMUM MEMBER FORCES DUE TO ANNULUS PRESSURIZATION

Component Description	Element Number[c]	28 Inch Recirculation	12 Inch Recirculation[d]	Feedwater	Jet Reaction
Top guide[a]	1	22.58	16.06	24.00	30.98
Core plate[a]	6	24.15	13.97	22.20	36.86
Fuel assembly[a]	7	14.49	61.00	47.00	20.48
Fuel assembly[b]	7	0.56	2.31	1.78	0.78
CRD housing[a]	59	15.68	16.51	9.23	21.32
CRD housing[b]	59	0.74	0.45	0.37	0.93
Shroud[a]	18	81.17	98.48	52.90	89.15
Shroud[b]	18	10.27	13.68	6.80	15.12
Shroud support[a]	27	140.01	255.90	183.90	340.20
Shroud support[a]	27	21.95	17.84	27.40	13.86
Vessel skirt[a]	52	737.52	1867.00	1467.40	1303.47
Vessel skirt[b]	52	98.60	203.90	283.50	124.53
Pedestal containment[a]	55	2213.30	1196.00	792.00	1382.43
Pedestal containment[b]	55	588.42	326.80	422.60	312.69
Stabilizer[a]	II	728.04	1171.54	1877.10	694.47
CRD restraint beam[a]	V	25.30	23.70	16.50	74.97

Notes:

(a) Load = 10^3 x lb.

(b) Moment = 10^6 x in-lb.

(c) Refer to Figure 3.9-3

(d) Combine pressure and jet load from 12” recirculation line break

TABLE 3.9-6 FERMI 2 MAXIMUM ACCELERATION DUE TO ANNULUS PRESSURIZATION (in./sec^2)

Component Description	Node Number[a]	28 Inch Recirculation	12 Inch Recirculation[b]	Feedwater	Jet Reaction
ΔP line	8	99.12	153.85	101.60	214.52
CRD guide tube	13	70.25	341.33	234.20	90.62
Feedwater sparger	43	157.29	262.29	179.80	192.89
Jet pump	45	165.90	249.95	169.80	220.61
RPV	51	97.02	181.21	120.20	271.43
RPV	55	149.10	135.13	79.30	257.57
RPV (bottom)	18	162.12	131.38	71.50	259.04
Shield Wall	63	363.09	1093.00	262.40	547.89
Top of shield wall	64	92.82	194.09	64.30	110.67

NOTES:

(a) Refer to Figure 3.9-3 for node number.

(b) Combine pressure and jet load from 12” recirculation line break.

TABLE 3.9-7 RPV INTERNALS ANALYSIS SUMMARY

	Stress (ksi)	
	Calculated	Allowable
Core support weld	11.414	20.28
Shroud buckling	4.82	12.497
Top guide	12.30	20.28
Jet pump	53.38	60.84
Head spray nozzle[b]	18.73	63.0
Core ΔP line	42.82	50.70
Fuel assembly (acceleration)	1.04g	3.12g[a]

[a] This is the design-basis acceleration rather than the allowable limit.

[b] Head spray piping is no longer attached to the reactor pressure vessel. Calculated stress value in this table is conservative.

TABLE 3.9-8 RPV EQUIPMENT ANALYSIS SUMMARY

Component		Stress (ksi) Calculated[a]	Stress (ksi) Allowable
RPV support (ring girder)	tension	94.0	125.0
	shear	18.01	33.4
RPV stabilizer	bending	21.16	36.0
	shear	6.3	21.5
CRD housing		13.15	20.0
Control rod guide tube		5.7	25.4

[a] The stress reported here is the highest of the dynamic load evaluation or the original design basis.

TABLE 3.9-9 RPV SUPPORT EQUIPMENT ANALYSIS SUMMARY

	Design	Calculated	Allowable
Shroud support (primary local plus bending)	16.2 ksi	53.9 ksi	55.9 ksi
Vessel skirt	23.6 ksi	14.5 ksi	a
Vessel stabilizer bracket	45.6 ksi	<45.6 ksi[b]	63.45 ksi

[a] Not calculated because the original design load produces a stress that is lower than the emergency allowable (28.7 ksi).

[b] Actual stress was not calculated because the calculated new load is lower than the original design load.

TABLE 3.9-10 RESULTS FOR REACTOR PRESSURE VESSEL AND INTERNALS

Shroud Support Gusseted Plate and Cylinder[a,b]

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
ASME B&PV Code Sec. III Primary Stress Limit for SB-168:				
For design condition: S_m = 23,300 psi 1.5 S_m = 35,000 psi	Design mechanical loads ≥ following: Normal and upset condition load 1. Dead weight 2. Design earthquake (operating-basis earthquake) 3. Design pressure differential	General membrane $P_L + P_B$	23,300 35,000	2900 23,500
For emergency condition: 1.2 S_m= 28,000 psi	Emergency-condition loads 1. Dead weight 2. Maximum credible earthquake (design-basis earthquake) 3. Normal pressure differential	General membrane $P_L + P_B$	28,000 42,100	4300 33,900
For faulted condition:[c]	Faulted-condition loads 1. Dead weight 2. Maximum credible earthquake 3. Pressure drop across core support plate due to LOCA blowdown	General membrane	c	4800[b] 39,600[b]

TABLE 3.9-10 RESULTS FOR REACTOR PRESSURE VESSEL AND INTERNALS

Vessel Support Skirt

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
ASME B&PVC Code Sec. III Primary Stress Limit for SA-516 Grade 70:				
For design condition: S_m = 19,150 psi	Design mechanical loads ≥ following: Normal- and upset-condition loads 1. Dead weight 2. Design earthquake (operating-basis earthquake)	General membrane	19,150	12,500
For emergency condition: S_y = 30,750 psi	Emergency-condition loads 1. Dead weight 2. Maximum credible earth- quake (design-basis earthquake)	General membrane	30,750	20,900
For faulted condition[c]	Faulted condition loads 1. Dead weight 2. Maximum credible earthquake 3. Jet reaction forces	General membrane	c	23,700[c]

TABLE 3.9-10 RESULTS FOR REACTOR PRESSURE VESSEL AND INTERNALS

Stabilizer Bracket-Adjacent Shell

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
ASME B&PV Code Sec. III Primary Local Membrane Plus Primary Bending Limit for SA-533 Grade B, Class I:				
For design condition: S_m = 26,700 1.5 x 26,700 = 40,050	Normal and upset condition load	General membrane Local membrane plus bending	26,700 40,050	26,500 28,400
	1. Design earthquake (operating-basis earthquake) 2. Design pressure			
For emergency condition: S_y = 42,600 1.5 S_y = 64,000	Emergency condition load	General membrane Local membrane plus bending	42,600 64,000	28,600 46,000
	1. Maximum credible earthquake (design earthquake) 2. Design pressure			
For faulted condition[c]:	Faulted-condition load	Local membrane plus bending	c c	26,500[c] 24,600[c]
	1. Maximum credible earthquake (design-basis earthquake) 2. Jet reaction forces 3. Design pressure			

[a] Gusseted support plate segments are sufficiently stiff that stability (buckling) would not be a predicted failure mode with increasing overturning (seismic) moment.

[b] Symbols are as defined in the ASME B&PV Code.

[c] Since the calculated stress for the faulted condition is less than the allowable stress for the emergency condition, and the allowable stress for the faulted condition is greater than the allowable stress for the emergency condition, the faulted allowable was not calculated.

TABLE 3.9-11 ACCELERATION (g) FOR FUEL ASSEMBLY

Node*	PR	JR	SSE	$(PR^2 + SSE^2 + JR^2)^{1/2}$	Design Basis
1 (top)	0.39	0.46	0.44	0.75	1.30
3	0.33	0.38	0.50	0.71	1.90
4	0.23	0.32	0.70	0.80	2.70
5	0.37	0.21	0.78	0.89	3.12
6	0.22	0.37	0.67	0.80	2.54
7	0.42	0.57	0.58	0.92	1.68
8	0.42	0.86	0.46	1.06	1.08

* See Figure 3.9-3.

TABLE 3.9-12 FUEL ASSEMBLY (INCLUDING CHANNEL)[a]

Acceptance Criteria	Loading	Primary Load Type	Calculated Peak Acceleration	Evaluation Basis Acceleration
Acceleration envelope	Horizontal direction	Horizontal acceleration profile	1.5 g	b
	1. Peak pressure 2. Safe-shutdown earthquake 3. Annulus pressurization			
	Vertical direction	Vertical accelerations	1.4 g	b
	1. Peak pressure 2. Safe-shutdown earthquake 3. Annulus pressurization			

[a] From an assessment comparing bounding limits (net holddown forces) to those for other BWR-4 plants already analyzed, a screening calculation was performed for Fermi 2. According to this analysis, Fermi 2 would experience virtually no fuel movement.

[b] Evaluation-basis accelerations and evaluations are contained in NPDE-21175-3-P.

TABLE 3.9-13 REACTOR PRESSURE VESSEL SUPPORT EQUIPMENT

Criteria	Loading	Location	Allowable Stress (psi)	Calculated Stress (psi)
RPV stabilizer Primary stress limit				
AISC specification for the construction, fabrication, and erection of structural steel for buildings	Upset condition 1. Spring preload	Rod	90,000	f_{b+t} = 82,000[a]
	2. Operating-basis earthquake	Bracket	22,000 14,000	f_b = 9200 f_v = 2730
For normal and upset conditions AISC allowable stresses, but without the usual increase for earthquake loads				
For emergency conditions 1.5 x AISC allowable stresses	Emergency condition 1. Spring preload 2. Design-basis earthquake	Bracket	33,000 21,000	f_b = 18,400 f_v = 5460
For faulted conditions Material yield strength	Faulted condition 1. Spring preload 2. Design-basis earthquake 3. Jet reaction load	Bracket	36,000 21,500	f_b = 21,160 f_v = 6300
RPV support (ring girder) Primary stress limit				
AISC specification for the design, fabrication, and erection of structural steel for buildings	Normal and upset condition 1. Dead loads	Top flange	22,000	f_b = 10,000
	2. Operating-basis earthquake	Bottom flange	22,000	f_b = 10,000
	3. Loads due to scram	Vessel to girder bolts	54,000 20,000	f_t = 35,200 f_v = 4450
For normal and upset conditions AISC allowable stresses, but without the usual increase for earthquake loads				
For emergency conditions 1.5 x AISC allowable stresses	Emergency condition 1. Dead loads	Top flange	33,000	f_b = 22,000
	2. Design-basis earthquake	Bottom flange	33,000	f_b = 20,000
	3. Loads due to scram	Vessel to girder bolts	81,000 30,000	f_t = 70,400 f_v = 8900
For faulted conditions 1.67 x AISC allowable stresses for structural steel members. Yield strength bolts (vessel to ring girder)	Faulted condition 1. Dead loads	Top flange	36,800	f_b = 28,000
	2. Design-basis earthquake	Bottom flange	36,800	f_b = 23,400
	3. Jet reaction load	Vessel to girder bolts	125,000 33,400	f_t = 94,000 f_v = 18,010

TABLE 3.9-13 REACTOR PRESSURE VESSEL SUPPORT EQUIPMENT

Criteria	Loading	Location	Allowable Stress (psi)	Calculated Stress (psi)
CRD housing support Primary stress limit				
AISC specification for the design, fabrication, and erection of structural steel for buildings	Faulted condition loads	Beams (top cord)	33,000 33,000	f_a = 12,200 f_b = 16,500
	1. Dead weight	Beams (bottom cord)	33,000 33,000	f_a = 10,300 f_b = 11,700
	2. Impact force from failure of a CRD housing	Grid structure	41,500 27,500	f_b = 40,700 f_v = 12,500
For normal and upset conditions $f_a = 0.60\ f_y$ (tension) $f_b = 0.60\ f_y$ (bending) $f_v = 0.40\ f_y$ (shear) For faulted conditions f_a limit = 1.5 f_a (tension) f_b limit = 1.5 f_b (bending) f_v limit = 1.5 f_v (shear) f_y = Material yield strength	(Dead weights and earthquake loads are very small as compared to jet force)			

[a] The ratio max. stress/stress limit is highest for upset loading conditions.

TABLE 3.9-14 REACTOR PRESSURE VESSEL INTERNALS AND ASSOCIATED EQUIPMENT

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Top guide - highest stressed beam				
Primary stress limit - The allowable primary membrane stress plus bending stress is based on ASME B&PV Code Sec. III for type 304 stainless steel plate				
For normal and upset condition Stress Intensity SA = 1.5 S_m = 1.5 x 16,900 psi = 25,350 psi	Normal and upset - condition loads 1. Operating-basis earthquake 2. Weight of structure	General membrane plus bending	25,350	12,820
For emergency condition: S_{limit} = 1.5 S_A = 1.5 x 25,350 = 38,025 psi	Emergency condition loads 1. Design-basis earthquake 2. Weight of structure	General membrane plus bending	38,025	12,220
For faulted condition: S_{limit} = 2 S_A = 2 x 25,350 = 50,700 psi	Faulted-condition loads (same as emergency condition)	General membrane plus bending	50,700	20,250
Top guide beam end connections				
Primary stress limit - ASME B&PV Code Sec. III, defines material stress limit for type 304 stainless steel				
For normal and upset condition Stress Intensity S_A = 0.6 S_m = 0.6 x 16,900 psi = 10,140 psi	Normal and upset-condition loads 1. Operating-basis earthquake 2. Weight of structure	Pure shear	10,140	4,500
For emergency condition: S_{limit} = 1.5 S_A = 1.5 x 10,140 psi = 15,210 psi	Emergency-condition loads 1. Design-basis earthquake 2. Weight of structure	Pure shear	15,210	4,400
For faulted condition: S_{limit} = 2 S_A = 2 x 10,140 psi = 20,280 psi	Faulted-condition loads (same as emergency condition)	Pure shear	20,280	12,300

TABLE 3.9-14 REACTOR PRESSURE VESSEL INTERNALS AND ASSOCIATED EQUIPMENT

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Top guide aligners				
Primary stress limit - The allowable primary membrane stress plus bending stress is based on ASME B&PV Code Sec. III for type 304 stainless steel plate				
For normal and upset condition Stress Intensity $S_A = 1.5\ S_m = 1.5 \times 16{,}900$ psi = 25,350 psi	Normal and upset-condition loads 1. Operating-basis earthquake 2. Weight of structure	General membrane plus bending	25,350	0[a]
For emergency condition: $S_{limit} = 1.5\ S_A = 1.5 \times 25{,}350 = 38{,}025$	Emergency-condition loads 1. Design-basis earthquake 2. Weight of structure	General membrane plus bending	38,025	0[a]
For faulted condition: $S_{limit} = 2\ S_A = 2 \times 25{,}350 = 50{,}700$ psi	Faulted-condition loads (same as emergency condition)	General membrane plus bending	50,700	0[a]
Core support			Allowable ΔP	Calculated ΔP
	Normal and upset-condition loads 1. Normal operation pressure drop 2. Operating-basis earthquake		27	18.9
For allowable stresses see top guide, longest beam, above	Emergency condition loads 1. Normal operation pressure drop 2. Design-basis earthquake		40.5	20.6

TABLE 3.9-14 REACTOR PRESSURE VESSEL INTERNALS AND ASSOCIATED EQUIPMENT

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
	Faulted-condition loads 1. Pressure drop after recirculation line rupture 2. Design-basis earthquake		54	27.7
Core support aligners				
Primary stress limit - ASME B&PV Code Sec. III, defines material stress limit for type 304 stainless steel	Normal and upset-condition loads 1. Operating-basis earthquake 2. Normal operation pressure drop	Pure Shear	10,155	0[b]
For allowable shear stresses, see top guide beam end connections, above	Emergency condition load 1. Design-basis earthquake 2. Normal operation pressure drop	Pure shear	15,232	0[b]
	Faulted condition load 1. Design-basis earthquake 2. Steam line rupture	Pure Shear	20,310	0[b]
Control rod drive housing				
Primary stress limit - The allowable primary membrane stress is based on ASME B&PV Code Sec. III, for Class I vessels, for type 304 stainless steel	Normal and upset-condition loads 1. Design pressure 2. Stuck rod scram loads 3. Operating-basis earthquake with housing lateral support installed	Maximum membrane stress intensity occurs at the tube-to-tube weld near the center of the housing for normal, upset and emergency conditions	16,660	13,150
For normal and upset condition S_m = 16,600 psi at 575°F				

TABLE 3.9-14 REACTOR PRESSURE VESSEL INTERNALS AND ASSOCIATED EQUIPMENT

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
For emergency conditions: S_{limit} = 1.2 S_m = 1.2 x 16,660 = 20,000 psi	Emergency condition loads 1. Design pressure 2. Stuck rod scram loads 3. Design-basis earthquake, with support installed		20,000	13,150
Control rod drive Primary stress limit - The allowable primary membrane stress plus bending stress is based on ASME B&PV Code	Normal and upset condition loads. Maximum hydraulic pressure from the control rod drive supply pump[c]	Maximum stress intensity occurs at a point on the Y-Y axis of the indicator tube	25,860	20,790
For normal and upset condition S_A = 1.5 S_m = 1.5 x 17,238 = 25,860 psi				
Control rod guide tube Primary stress limit- The allowable primary membrane stress plus bending stress is based on ASME B&PV Code Sec. III for type 304 stainless steel tubing				
For normal and upset conditions S_m = 16,925 psi				
For faulted condition: S_{limit} = 1.5 S_m = 1.5 x 16,295 = 25,400 psi	Faulted condition loads 1. Dead weight 2. Pressure drop across guide tube due to failure of steam line	The maximum bending stress under faulted loading conditions occurs at the center of the guide tube	25,400	5,701
In-core housing Primary stress limit- The allowable primary membrane stress is based on ASME B&PV Code Sec. III for Class 1 vessels for type 304 stainless steel				
For normal and upset conditions: S_m = 16,660 psi at 575°F				

TABLE 3.9-14 REACTOR PRESSURE VESSEL INTERNALS AND ASSOCIATED EQUIPMENT

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
For emergency condition: S_{limit} = 1.2 S_m = 1.2 x 16,600 = 20,000 psi	Emergency condition loads 1. Design pressure 2. Design-basis earthquake	Maximum membrane stress intensity occurs at the outer surface of the vessel penetration	20,000	15,290

[a] Thirty-two wedges that will resist the horizontal seismic top guide shear load are installed in the annulus between the top guide and shroud. Therefore there is no load on the top guide aligners

[b] The friction force between core support and core support flange due to the preload of the studs is greater than the shear load induced by the specified earthquake.

[c] Accident conditions do not increase this loading. Earthquake loads are negligible.

TABLE 3.9-15 INITIAL FUEL LOAD 100 MIL FUEL CHANNELS

Criteria	Loading	Primary Stress Type	Moment Limit Accounting for Pressure Loads (in.-lb)	Maximum Moment (in.-lb)
Fuel channels				
Primary stress limit – Design stress intensity S_m for zircaloy determined according to methods recommended by ASME B&PV Code Sec. III. Allowable moment determined by calculating limit moment, then applying SFmin for applicable loading conditions	Normal and upset condition load 1. Operating-basis earthquake 2. Normal pressure load	Membrane and bending	35,000	9550
(S = 9,000 psi; 1.5 S_m = 13,500 psi) (1.5 S_m = Allowable Stress)	Faulted condition load 1. Design-basis earthquake 2. Loss-of-coolant accident pressure	Membrane and bending	68,000	15,850

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TABLE 3.9-17 MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress (psi), Minimum Thickness (in.), or Minimum Area (in.^{2})	Calculated Stress (psi), Actual Thickness (in.) or Minimum Area (in.^{2})
Design of Pressure-Retaining Parts	All references are made to ASME Code for Pumps and Valves for Nuclear Power, dated November 1968. Reference the same code for explanation of the symbols used.		
Body Minimum Wall Thickness	Reference Article 452.1b(2), Nonstandard Pressure - Rated Valve, Table NB 451.4 For design condition of 1,250 psig and 575°F The primary service rating = 655 lb, based on a core diameter of 23 in. t_m= 1.925 in. (including a corrosion allowance of 0.12 in.)	1.925 in.	1.9375 in.
Body Shape Rules	Reference Article 452.2, Body Shape Rules		
Radius of Crotch	Reference Article 452.2a(1), Radius of Crotch Criterion: $r_2 > 0.3$ tm; $r_2 = 1.0$ in., $t_m = 1.925$ in., 0.3 x 1.925 = 0.578 <1.0; criterion satisfied	0.578 in.	1.0 in.
Out-of-Roundness	Reference Article 452.2e. Since no ovality was built into the valve body, the requirements of this article are satisfied.	Not applicable	Not applicable
Flat Wall Limitation	Reference Article 452.2g, Flat Wall Limitation. Since no flat sections were built into the valve body design, the requirements of this article are satisfied.	Not applicable	Not applicable
Primary Crotch Stress Due to Internal Pressure	Reference Article 452.3 Criterion: $P_m = \left(\frac{A_f}{A_m} + 0.5\right) P_s < S_m$ where A_f = 504 in.^{2}, A_m = 58 in.^{2}, P_s = 1.375 psig, P_m = 12,650 psi, S_m = 19,400 psi; since $S_m > P_m$, criterion satisfied	19,400 psi	12,650 psi
Valve Body Secondary Stress	Reference Article 452.4		
Primary Plus Secondary Stress Due to Internal Pressure	Reference Article 452.4a $Q_p = C_p \left(\frac{r_i}{t_e} + 0.5\right) P_s C_a$ where $C_p = 3$, r_i = 11.625 in., P_s = 1,375 psi, t_e = 2.75 in. for wye-type valve, $C_a = 1.33 \rightarrow Q_p$ = 25,965 psi		
Secondary Stress Due to Pipe Reaction	Reference Article 452.4b, Figures 452.4b(3), 452.4b(4), 452.4b(5)		
Direct or Axial Load Effect	$P_{ed} = \frac{F_d S}{G_d}$, where S = 30,750 psi, F_d = 30 in.^{2}, G_d = 183 in.$^2 \rightarrow P_{ed}$ = 5,040 psi	19,400 psi	5,040 psi
Bending Load Effect	$P_{eb} = C_b \frac{F_b S}{G_b}$ where S = 30,750 psi, F_b = 340 in.^{3}, i.d. = 23.25 in., r_i = 11.625 in., t_e = 2.75 in., $\bar{r}$ = 13.90 in. as $\frac{t_e}{\bar{r}} = 0.197 > 0.19 \rightarrow C_b = 1$ $G_b = \frac{I}{r_i + t_e}$ where I = 15,028 in.^{4}, r_i = 11.625 in., t_e = 2.75 in. $\rightarrow G_b$ = 1,052 in.$^3 \rightarrow P_{eb}$ = 9,940 psi	19,400 psi	9,940 psi
Torsion Load Effect	Reference Article 452.4b $P_{et} = 2 \frac{F_b S}{G_t}$ where F_b = 340 in.^{3}, S = 30,750 psi, G_t = 2,162 in.^{3}, P_{et} = 9,670 psi	19,400 psi	9,670 psi

TABLE 3.9-17 MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress (psi), Minimum Thickness (in.), or Minimum Area (in.^{2})	Calculated Stress (psi), Actual Thickness (in.) or Minimum Area (in.^{2})
Thermal Secondary Stress at Crotch Region	Reference Article 452.4C, Figures 452.4C(4), 452.4C(3), 452.4C(5) $Q_T = Q_{T_1} + Q_{T_2}$ where T_{e_1} = 3 in., Q_{T_1} = 1,100 psi, $Q_{T_2} = C_6C_2\Delta T_2$ where C_2 = 0.21, C_6 = 220, and ΔT_2 = 5.6 Q_{T_2} = 260 psi, Q_T = 1,360 psi Criterion: $S_N = Q_p + P_e = 2\,Q_{T_2} \le 3\,S_m$ where Q_p = 25,965, P_e = 9,940, Q_T = 1,360 as 38,625 ≤ 58,200, criterion satisfied	58,200 psi	38,625 psi
Normal Duty Valve Fatigue Requirements	Reference Article 452.5, Figure 452.5(a) Criterion $N_a \ge$ 2,000 cycles. $S_{p_1} = \frac{2}{3}\,Q_p + \frac{P_{eb}}{2} + Q_{T_2} + 1.3Q_{T_1}, Q_{T_1} = 1{,}100$ psi $S_{P_2} = 0.4Q_p + \frac{K}{2}\,(P_{eb} + 2Q_{T_2})$ where Q_p = 25,965, P_{eb} = 9,940, Q_{T_1}= 1,160, Q_{T_2}= 260 psi, K = 2 → S_{p_1}= 23,970 psi, S_{P_2}= 20,845 psi, S_a equal to the larger of $S_{p_1}$and S_{P_2} → S_a = 23,970 psi → N_a = 55,000 ± 2,000, criterion satisfied	2,000 cycles	55,000 cycles
Cyclic Loading Requirements at Valve Crotch	Reference Article 454 Thermal Transients Not Excluded by Code Criterion: $\sum \frac{N_{ri}}{N_i} < 1$ Calculate the fatigue usage factor (I_t) as follows: $S_n\,Max = Q_p + P_{eb} + C_6(C_3 + C_4)\Delta T_f\,max$ $S_n$max = 105,810 psi for ΔT_{fi} = 90, N_{ri} = 120, N_i = 2,700 $\frac{N_{ri}}{N_i}$ = 0.044 ΔT_{fi} = 122, N_{ri} = 10, N_i = 1,600 $\frac{N_{ri}}{N_i}$ = 0.006 ΔT_{fi} = 342, N_{ri} = 8, N_i = 42 $\frac{N_{ri}}{N_i}$ = 0.19 as $I_t = \sum \frac{N_{ri}}{N_i}$ = 0.240 < 1, criterion satisfied	1	0.240[a]
Disk Design Calculation	From Roark's Formulas for Stress and Strain, third addition Disk design conditions, P_s = 1,250 psi at 575°F, S_m = 17,800 psi at 600°F Case No. 13: $S_t = \frac{3W}{4mt^2\,(a^2-b^2)}\,[a^4\,(3m+1) + b^4\,(m-1) - 4_m a^2b^2 - 4\,(m+1)a^2b^2\,\left(\ln\left(\frac{a}{b}\right)\right)]$ where W = 1,250 psi, m = 10/3, t = 5.625 in., a = 10.75 in., b = 1.75 in., $S_{t_{13}}$ = 10,354 psi Case No. 14: $S = \frac{3\,W}{2\pi mt^2}\left[\frac{2a^2\,(m+1)}{a^2 - b^2}\,\ln\left(\frac{a}{b}\right) + (m-1)\right]$ where W = 59,044 lb_f, t = 5.625 in., m = 10/3 a = 10.75 in., b = 1.75 in., $S_{t_{14}}$= 4,943 psi Total stress = $S_{t_{13}}$+ $S_{t_{14}}$ = 15,297 psi, allowable stress = 17,800 psi	17,800 psi	15,297 psi

TABLE 3.9-17 MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress (psi), Minimum Thickness (in.), or Minimum Area (in.^{2})	Calculated Stress (psi), Actual Thickness (in.) or Minimum Area (in.^{2})
	Case No. 21: $S_r = \frac{3W}{4t^2}\left[\frac{4a^4(m+1)\ln\left(\frac{a}{b}\right)a^4(m+3)+b^4(m-1)+4a^2b^2}{a^2(m+1)+b^2(m-1)}\right]$ where W = 1,250 psi, m = 10/3, t = 3.125 in., a = 10.75 in., b = 7.25 in. → $S_{r_{21}}$ = 5760 psi Case No. 22: $Sr = \frac{3W}{2^h t^2}\left[\frac{2a^2(m+1)\ln\left(\frac{a}{b}\right)+a^2(m-1)-b^2(m-1)}{a^2(m+1)+b^2(m-1)}\right]$ where W = 1,250 psi, m = 10/3, t = 3.125 in., a = 10.75 in., b = 7.25 in. → $S_{r_{22}}$ = 10,740 psi Total stress = $S_{r_{21}}$+ $S_{r_{22}}$ = 16,500 psi, allowable stress = 17,800 psi	17,800 psi	16,500 psi
Tensile Stress at Thread Relief Valve Stem	Valve open $S_A = \frac{F}{A_t}$ where F = 31,586 lb, A_t = 1.956 in.^{2}, S_{max} = 16,148 psi Valve closed F = 46,342 lb, S_{max} = 23,692 psi	30,600 psi	23,692 psi
Bonnet Design Calculations Including Seismic Accelerations for SSE	Paragraph UG – 34c(2) of ASME Code Section VIII		
Minimum Thickness	$P_{fd} = P + P_{eg}$, $P_{eg} = \frac{16M}{\pi G^3} + \frac{4F}{\pi G^2}$ where M = 335,253 in.-lb, F = 46,342 lb, G = 24.75 in., P_{eg} = 204 psi, P_{fd} = 1,459 psi $t = d\sqrt{\frac{CP_{fd}}{S} + \frac{(1.78W)(hg)}{Sd^3}}$ where C = 0.3, P_{fd} =1,459 psi, S = 17,800 psi, hg = 2.625 in., W = 910,144 lb, d = 24.75 in. → t = 4.975 in., t = 4.975 + 0.120 = 5.095 in. (corrosion allowance is 0.120 in.)	5.095 in.	5.344 in.
Reinforcement	Reference Paragraph I-704.41(c) of USAS B31-7 To account for the opening for stem in the bonnet Required reinforcement d x t x 0.5 = $(d_3t_3 + d_4t_4)/2$ d_3 = 1.875, t_4 = 2.223, t_3 = 2.875, d_4 = 3 Reinforcement = 6.030 in.^{2} required 6.6126 in.^{2} available	6.030 in.^{2}	6.6126 in.^{2}
Bonnet Studs Design Calculation	Reference Article E-1000 Bolt used 20 pieces of 2.652 in.^{2}/bolts Total bolt area = 53.04 in^2		
Normal Operation	1. Pressure stress at operating condition $S_1 = \frac{W_{m1}}{A_b} =$ 17,160 psi where W_{m1}= 910,144 lb A_b = 53.04 in.^{2}	27,700 psi	17,160 psi
	2. Gasket load at ambient condition with no internal pressure $S_2 = \frac{W_{m2}}{A_b} =$ 2,019 psi where W_{m2}= 107,065 lb_f A_b = 53.04 in.^{2} Maximum tensile stress = 17,160 psi Thermal stress is assumed negligible because the coefficients of thermal expansion of bonnet place and stud are the same.	35,000 psi	2,019 psi
Longitudinal Hub Stress	$S_H = \frac{fM_o}{Lg_1{}^2B} + \frac{PB}{4g_o}$ = 21,773 psi < 1.5 S_{fo} = 26,700 psi	26,700 psi	21,773 psi

TABLE 3.9-17 MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress (psi), Minimum Thickness (in.), or Minimum Area (in.^{2})	Calculated Stress (psi), Actual Thickness (in.) or Minimum Area (in.^{2})
Radial Stress	Reference UA-51 (1), Equation (7) of Section VIII of ASME B&PV Code, 1971 Edition $S_R = \frac{(1.33t_e + 1)M_o}{Lt^2B}$ = 12,288 psi < 1.5 S_{fo} = 26,700 psi	26,700 psi	12,288 psi
Tangential	$S_T = \left(\frac{YM_o}{t^2B}\right) - ZS_R$ = 7,117 psi < 1.5 S_{fo} = 26,700 psi where Y = 4.5, t = 4.125 in., Z = 2.4, B = 21.75 in.	26,700 psi	7,117 psi
Body Flange Design Calculations	Reference Paragraph I-704.5.1 of USAS B31-7 Total flange moment under operating conditions $M_o = M_D + M_G + M_T$ $M_D = H_D h_D$, $H_D = 0.785\ B^2P$, $h_D = R + 0.5g$, where B = 21.75 in., P = 1,459 psi → H_D = 542,080 lb_f h_D = 2.813 in., M_D = 1,524,871 in.-lb $M_G = H_G h_G$, $H_G = W_{m1} - H$, $h_G = \frac{C-G}{2}$ where W is the higher of W_{m1} and W_{m2} W_{m1} = 910,144 lb W_{m2} = 107,065 lb H_G = 208,210 lb, h_G = 2.625 in. → M_G = 546,531 in.-lb $M_T = H_T h_T$ H_T = 159,854 lb, h_T = 3.375 in., M_T = 539,507 in.-lb M_o = 2,610,929 in.-lb Total flange moment under gasket seating condition $M_o = W\frac{(C-G)}{2}$, $W = \frac{(A_m + A_b)S_a}{2}$ where C = 30 in., A_b = 53.04 in.^{2}, G = 24.75 in., A_m = 32.857 in.^{2}, S_a = 35,000 psi at 100°F → W = 1,503,193 lb → M_o = 3,010,718 in.-lb Where w = design pressure, 1250 psi m = inverse of Poisson ratio, 3.3333 t = disk thickness, 5.875 in. a = outside radius of poppet, 10.75 in. b = inside radius of pilot hole, 1.75 in. St = Maximum stress at inner edge, 9,489 psi For a plate with a hole in the center, outer edge supported and uniformly loaded along the inner edge $St = \frac{3W}{2\pi mt^2}\left[\frac{2a^2(m+1)}{a^2 - b^2}\ln\left(\frac{a}{b}\right) + (m-1)\right]$ where W = operator, spring and internal pressure acting on pilot poppet, 59,044 < B S_t = Maximum stress at inner edge, 4531 psi Total stress = $St_1 + St_2$ = 14,020 as 17,800 > 14,020 criterion satisfied	17,800	14,020
	3. Disk Flexibility Roark's Formula for stress and strain, third edition, case 21 Max. Stress $\sigma_1 = \frac{3W}{4t^2}\left[\frac{4a^4(m+1)\ln\left(\frac{a}{b}\right) - a^4(m+3) + b^4(m-1) + 4a^2b^2}{a^2(m+1) + b^2(m-1)}\right]$ Deflection $\Delta_1 = \frac{3W(m^2-1)}{16m^2E_t{}^3}\left[\begin{matrix} a^6(7m+3) + b^6(m-1) - a^4b^2(m+7) - a^2b^4(7m-5) \\ -4a^2b^2[a^2(5m-1) + b^2(m+1)]\ln\frac{a}{b} \\ \frac{-16a^4b^2(m+1)(\ln\frac{a}{b})^2}{a^2(m+1)+b^2(m-1)} \end{matrix}\right]$ where E = modulus of elasticity, 25.7 x 10^6 psi at 600°F		

TABLE 3.9-17 MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress (psi), Minimum Thickness (in.), or Minimum Area (in.[2])	Calculated Stress (psi), Actual Thickness (in.) or Minimum Area (in.[2])

σ_1 = 5760 psi
Δ_1 = 0.00035 in.
Roark's Formulas for Stress and Strain, third edition, Case 22

$$\sigma_2 = \frac{3W_2}{2\pi t^2}\left[\frac{2a^2(m+1)\ln\left(\frac{a}{b}\right) + a^2(m-1) - b^2(m-1)}{a^2(m+1) + b^2(m-1)}\right]$$

$$\Delta_2 = \frac{3W_2}{4\pi m^2 E t^3}\left[a^4(3m+1) - b^4(m-1) - 2a^2b^2(m+1)\;\frac{-8ma^2b^2\ln\left(\frac{a}{b}\right) - 4a^2b^2(m+1)\left(\ln\left(\frac{a}{b}\right)\right)^2}{a^2(m+1) + b^2(m-1)}\right]$$

where W_2 = Operator, spring, and internal pressure acting on main disc, 252,755 lb
σ_2 = 10,740 psi
Δ_2 = 0.00086 in.
Total stress $\sigma_2 = \sigma_1 + \sigma_2$ = 16,500 psi
Total deflect $\Delta_t = \Delta_1 + \Delta_2$ = 0.0012 in.

Criteria	Method of Analysis	Allowable	Calculated
	as 17,800 > 16,500	17,800	16,500

For the above calculation:

W_1 = total applied load, 59,044 lb for St
W_2 = total applied load, 252,755 for St_2

w = design pressure, 1250 psi
a = large disc radius, 10.75 in.
b = smaller disc radius, 7.25 in.
t = larger disk thickness, 3.125 in
m = inverse of Poisson Ratio, 3.3333
E = Young's Modulus, 25.7 x 10^6 psi at 600 °F

TABLE 3.9-17 MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress (psi) of Minimum Thickness Required	Calculated Stress, Actual Thickness
Stem Analysis	1. Valve open Tension at undercut at back seat $S = \frac{W}{A} = 15{,}230$ psi Criterion satisfied	30,600	15,230
	Where W = total open force, 31,586 lb A = cross sectional area, 2.074 $in.^2$		
	Tension at undercut at thread $S = \frac{W}{A} = 16{,}148$ psi Criterion satisfied	30,600	16,148
	Where W = total open force, 31,586 lb A = cross sectional area, 1.956 $in.^2$		
	Tension at thread at root area $S = \frac{W}{A} = 15{,}953$ psi Criterion satisfied	30,600	15,953
	Where W = total opening load, 31,586 lb A = cross sectional area, 1.98 $in.^2$		
	Stress at thread $S = \frac{W}{A} = 5561$ psi Criterion satisfied	18,360	5,561
	Where W = total opening load, 31,586 lb A = cross sectional area, 5.74 $in.^2$		
	2. Valve closed Compression at undercut at back seat $S = \frac{W}{A} = 22{,}344$ psi Criterion satisfied	30,600	22,344
	Where W = total closed load, 46,345 lb A = cross sectional area, 2.074 $in.^2$		
	Compression at undercut at thread $S = \frac{W}{A} = 23{,}692$ psi Criterion satisfied	30,600	23,682
	Where W = total closed load, 46,342 lb A = cross sectional area, 1.956 $in.^2$		
	Compression at thread root area $S = \frac{W}{A} = 23{,}405$ psi Criterion satisfied	30,600	23,405
	Where W = total closed load, 46,342 lb A = cross sectional area, 1.98 $in.^2$		
	Shear at thread $S = \frac{W}{A} = 8{,}141$ psi Criterion satisfied	18,360	8,141
	Where W = total closed load, 46,342 lb A = root area of thread, 4.75 $in.^2$		

TABLE 3.9-17 MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress (psi) of Minimum Thickness Required	Calculated Stress, Actual Thickness
Cyclic Rating	Based on Article 454 1. Instantaneous fluid temperature $\Delta T_{fo} = \frac{3Sm - Qp - Peb}{C_6(C_3 - C_4)} > 150$ °F Where C_3 = 0.625 from Figure 454.3b C_4 = 0.0105 from Figures 454.3a and 454.4c(4) C_6 = 220 from Figure 454.3 T_{fo} = 158 °F > 150 °F Criterion satisfied 2. Fatigue stress intensity resulting in step change at 300 °F Salt = 84,140 psi N_{300} = 900 cycles at 500 °F Salt = 155,540 psi N_{500} = 170 cycles at 158 °F Salt = 40,540 psi N_{158} = 8000 cycles applied the above to Figure 454.2, these points are above the thermal cyclic rating curve and therefore qualified for cyclic rating per article 454.3. 3. Thermal cyclic index (article 454.2) $It = \sum \frac{Nri}{Ni} < 1$ Where It = cyclic rating index Nri = Required number of fluid step changes at ΔTi Ni = Permissible number of fluid step changes at ΔTi It = 0.240 < 1 Criterion satisfied		
Special Requirement with Pipe Rupture	Based on Article 452.4b Secondary stresses due to pipe reaction, crotch secondary effect due to bending load; and crotch secondary effect due to pipe torsion. Reference item 4, part 2, except in this case the stress from connecting pipe is raised to 41,000 psi Ped = 6,722psi Peb = 13,251 psi Pet = 12,896 psi These are all below 1.5 Sm = 29,100 Criterion satisfied	24,100	13,251
	Valve Body Secondary Stresses Also Sn = Q_p + Pe + $2Q_T$ Sn= 41,936 psi < 3 Sm (= 58,200) Criterion satisfied	58,200	41,936
	So even at the high pipe connection load the crotch area maximum stress is still within code allowance		

[a] See FP FERM 310 (Ref. 19) for maximum Cumulative Usage Factor values using event number of cycles based on Fermi 2 operating history.

TABLE 3.9-18 STRUCTURAL AND MECHANICAL LOADING CRITERIA

Reactor Recirculation Gate Valve, 28-In. Discharge

Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
Body and Bonnet			
Loads: Design pressure, design temp., pipe reaction, thermal effects			
Pressure rating, psi	Used Tables 451.4 & 451.5 of NPVC	P_r = 799 psi	P_r = 799 psi
Minimum wall thickness, in.	Used Table 452.1 of NPVC, dm = 22	$t_m \geq 2.205$ in.	$t_m = 2.205$ min.
Primary membrane stress, psi	Used Paragraph 452.3 of NPVC	$P_m \leq S_{m(500°F)}$ = 19,600 psi	P_m = 9512 psi
Secondary stress due to pipe reaction	Used Paragraph 452.4b of NPVC (S = 16,600 psi)	P_e = greatest value of P_{ed} P_{eb} and $P_{et} \leq 1.5\ S_{m(500°F)}$ 1.5 (19,600) = 29,400 psi	P_{ed} = 5502 psi P_{eb} = 12,550 psi P_{et} = 12,080 psi $P_e = P_{eb}$ = 12,550 psi
Primary plus secondary stress due to internal pressure	Used Paragraph 452.4a of NPVC	$S_n \leq 3\ S_{m(500°F)}$ = 58,800 psi	Q_p = 24,255 psi
Thermal secondary stress	Used Paragraph 452.4c of NPVC	$S_n \leq 3\ S_{m(500°F)}$ = 58,800 psi	Q_T = 6560 psi
Sum of primary plus secondary stress	Used Paragraph 452.4 of NPVC	$S_n \leq 3\ S_{m(500°F)}$ = 58,800 psi	$S_n = Q_p + P_e + 2Q_T$ S_n = 49,925psi
Fatigue requirements	Used Paragraph 452.4 of NPVC	$N_a \geq 2000$ cycles	$N_a = 3.0 \times 10^5$ cycles
Cyclic rating	Used Paragraph 454 of NPVC	$I_t \leq 1$	I_t = 0.006 (normal duty)[a]

TABLE 3.9-18 STRUCTURAL AND MECHANICAL LOADING CRITERIA

Reactor Recirculation Gate Valve, 28-In. Discharge

Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
Body and Bonnet Bolting			
Loads: Design pressure & temp., gasket loads, stem operational load, seismic load (design-basis earthquake)	USAS B31.7 Paragraph 1-704.5.1 Used ASME Section VIII, 1968 Paragraph UA-47 to UA-51 as required by Paragraph 453.1 of NPVC		
Bolt area	USAS B31.7 Paragraph 1-704.5.1 Used ASME Section VIII, 1968 Paragraph UA-47 to 51 as required by Paragraph 453.1 of NPVC	$A_b \geq 42.46$ in.^{2}	$A_b = 55.86$ in.^{2}
Body flange stresses	USAS B31.7 Paragraph 1-704.5.1 Used ASME Section VIII, 1968 Paragraph UA-47 to UA-51 as required by Paragraph 453.1 of NPVC	$S_b \leq 27{,}975$ psi	$S_b = 21{,}628$ psi
Operating condition	USAS B31.7 Paragraph 1-704.5.1 Used ASME Section VIII, 1968 Paragraph UA-47 to UA-51 as required by Paragraph 453.1 of NPVC	$S_H \leq 1.5\ S_{m(575°F)} = 28{,}837$ psi $S_R \leq 1.5\ S_{m(575°F)} = 28{,}837$ psi $S_T \leq 1.5\ S_{m(575°F)} = 28{,}837$ psi	$S_H = 25{,}970$ psi $S_R = 7909$ psi $S_T = 7909$ psi
Gasket seating condition	USAS B31.7 Paragraph 1-704.5.1 Used ASME Section VIII, 1968 Paragraph UA-47 to UA-51 as required by Paragraph 453.1 of NPVC	$S_H \leq 1.5\ S_{m(150°F)} = 30{,}000$ psi $S_R \leq 1.5\ S_{m(150°F)} = 30{,}000$ psi $S_T \leq 1.5\ S_{m(150°F)} = 30{,}000$ psi	$S_H = 29{,}225$ psi $S_R = 11{,}727$ psi $S_T = 11{,}918$ psi

TABLE 3.9-18 STRUCTURAL AND MECHANICAL LOADING CRITERIA

Reactor Recirculation Gate Valve, 28-In. Discharge

Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
Bonnet flange			
Operating condition	Calculate bonnet flange thickness according to rules of ASME Section VIII, Art. UA-6, Fig. UA-6c	$S_{max} \leq 1.5\ S_{m(575°F)}$ = 19,600 psi	S = 5863 psi
Stresses in Stem Loads: Operator thrust and torque			
Stem thrust stress	Calculate stress due to operator thrust in critical cross section	S_T , C ≤ S_m = 44,100 psi	S_T , C = 28,512 psi
Stem torque stress	Calculate shear stress due to operator torque in critical cross section	$S_S \leq 0.8\ S_m$ = 35,280 psi	S_S = 23,011 psi
Disk Analysis			
Loads: Maximum differential pressure			
Maximum stress in the disk	Calculate maximum according to Table10 of Roark's "Formula for Stress and Strain"	$S_{max} \leq 1.5\ S_{m(575°F)}$ = 28,500 psi	Max. stress = 22,885 psi
Yoke and Yoke Connections			
Maximum stress in yoke	Calculate stresses in the yoke to acceptable structural analysis methods	$S_{max} \leq S_m$ = 19,400 psi	Max. stress = 8488 psi
Yoke - bonnet bolt stress	Calculate stresses in the yoke bolts	$S_{max} \leq S_m$ = 28,800 psi	Max. stress = 7940 psi

TABLE 3.9-18 STRUCTURAL AND MECHANICAL LOADING CRITERIA

Reactor Recirculation Gate Valve, 28-In. Discharge

Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
Body and Bonnet			
Loads: Design pressure, design temp., pipe reaction, thermal effects			
Pressure rating, psi	Used Tables 451.4 & 451.5 of NPVC	P_r = 655 psi	P_r = 655 psi
Minimum wall thickness, in.	Used Table 452.1 of NPVC, dm = 22	$t_m \geq 1.70$ in.	$t_m = 1.70$ min.
Primary membrane stress, psi	Used Paragraph 452.3 of NPVC	$P_m \leq S_{m(500°F)} = 19{,}600$	P_m = 8797 psi
Secondary stress due to pipe reaction	Used Paragraph 452.4b of NPVC (S = 23,700 psi)	P_e = greatest value of P_{ed} P_{eb} and $P_{et} \leq 1.5$ $S_{m(500°F)}$ 1.5 (19,600) = 29,400 psi	P_{ed} = 5253 psi P_{eb} = 11,917 psi P_{et} = 11,573 psi $P_e = P_{eb}$ = 11,917 psi
Primary plus secondary stress due to internal pressure	Used Paragraph 452.4a of NPVC	$S_n \leq 3\ S_{m(500°F)}$ = 58,800 psi	Q_p = 20,580 psi
Thermal secondary stress	Used Paragraph 452.4c of NPVC	$S_n \leq 3\ S_{m(500°F)}$ = 58,800 psi	Q_T = 5815 psi
Sum of primary plus secondary stress	Used Paragraph 452.4 of NPVC	$S_n \leq 3\ S_{m(500°F)}$ = 58,800 psi	$S_n = Q_p + P_e + 2Q_T$ S_n = 44,127 psi
Fatigue requirements	Used Paragraph 452.4 of NPVC	$N_a \geq 2000$ cycles	$N_a > 10^6$ cycles
Cyclic rating	Used Paragraph 454 of NPVC	$I_t \leq 1$	I_t = 0.131 (normal duty)[a]

TABLE 3.9-18 STRUCTURAL AND MECHANICAL LOADING CRITERIA

Reactor Recirculation Gate Valve, 28-In. Discharge

Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
Body and Bonnet Bolting Loads: Design pressure and temp., gasket loads, stem operational load, seismic load (design-basis earthquake)	USAS B31.7 Paragraph 1-704.5.1 Used ASME Section VIII, 1968 Paragraph UA-47 to UA-51 as required by Paragraph 453.1 of NPVC		
Bolt area	USAS B31.7 Paragraph 1-704.5.1 Used ASME Section VIII, 1968 Paragraph UA-47 to UA-51 as required by Paragraph 453.1 of NPVC	$A_b \geq 36.8$ in.^{2} $S_b \leq 27{,}975$ psi	$A_b = 55.86$ in.^{2} $S_b = 17{,}326$ psi
Body flange stresses	USAS B31.7 Paragraph 1-704.5.1 Used ASME Section VIII, 1968 Paragraph UA-47 to UA-51 as required by Paragraph 453.1 of NPVC		
Operating condition	Same as above	$S_H \leq 1.5\ S_{m(575°F)} = 28{,}837$ psi $S_R \leq 1.5\ S_{m(575°F)} = 28{,}837$ psi $S_T \leq 1.5\ S_{m(575°F)} = 28{,}837$ psi	$S_H = 20{,}891$ psi $S_R = 6336$ psi $S_T = 6336$ psi
Gasket seating condition	USAS B31.7 Paragraph 1-704.5.1 Used ASME Section VIII, 1968 Paragraph UA-47 to UA-51 as required by Paragraph 453.1 of NPVC	$S_H \leq 1.5\ S_{m(150°F)} = 30{,}000$ psi $S_R \leq 1.5\ S_{m(150°F)} = 30{,}000$ psi $S_T \leq 1.5\ S_{m(150°F)} = 30{,}000$ psi	$S_H = 27{,}887$ psi $S_R = 11{,}366$ psi $S_T = 11{,}647$ psi
Bonnet flange	Same as above		

TABLE 3.9-18 STRUCTURAL AND MECHANICAL LOADING CRITERIA

Reactor Recirculation Gate Valve, 28-In. Discharge

Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
Operating condition	Calculate bonnet flange thickness according to rules of ASME Section VIII, Art. UA-6, Fig. UA-6(c)	$S_{max} \leq S_{m(575°F)}$ = 19,600 psi	S = 5960 psi
Stresses in Stem			
Loads: Operator thrust and torque			
Stem thrust stress	Calculate stress due to operator thrust in critical cross section	S_T , C $\leq S_m$ = 44,100 psi	S_T , C = 24,343 psi
Stem torque stress	Calculate shear stress due to operator torque in critical cross section	$S_S \leq 0.8\ S_m$ = 35,280 psi	S_S = 19,185 psi
Disk Analysis			
Loads: Maximum differential pressure			
Maximum stress in the disk	Calculate maximum according to Table10 of Roark's "Formula for Stress and Strain"	$S_{max} \leq 1.5\ S_{m(575°F)}$ = 28,500 psi	Max. stress = 19,432 psi
Yoke and Yoke Connections			
Maximum stress in yoke	Calculate stresses in the yoke to acceptable structural analysis methods	$S_{max} \leq S_m$ = 19,400 psi	Max. stress = 5552 psi
Yoke - bonnet bolt stress	Calculate stresses in the yoke bolts	$S_{max} \leq S_m$ = 28,800 psi	Max. stress = 4008 psi

[a] See FP FERM 310 (Ref. 19) for maximum Cumulative Usage Factor values using event number of cycles based on Fermi 2 operating history.

TABLE 3.9-19 MAIN STEAM SAFETY/RELIEF VALVES (PILOT OPERATED), FERMI 2
(ASME Code, Section III, 1968, Including Addenda through Summer 1970)

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
1. BODY INLET AND OUTLET FLANGE STRESSES Note, Topics 1 and 2: Design Pressures: P_d = 1375 psig (inlet) P_b = 625 psig (outlet) These are the equivalent maximum anticipated pressures under all operating conditions. Analyses include applied moments of M = 400,000 in.-lb (inlet) and M = 300,000 in.-lb (outlet) Actual tested capability (including accelerations and moments) is as described in Topic 11. The analyses also include consideration of seismic, operational, and flow reaction forces. Allowable vs. tested capabilities are provided in Topic 12.	$S_H = \frac{fMo}{Lg_1{}^2B} + \frac{PB}{4g_o} < 1.5\ Sm$ $S_R = \frac{(\frac{4te}{3}+1)Mo}{Lt^2B} < 1.5\ Sm$ $S_T = \frac{YM_o}{t^2B} - Z\,S_R < 1.5\ Sm$ where S_H = Longitudinal "hub" wall stress, psi. S_R = Radial "flange" stress, psi. S_T = Tangential "flange" stress, psi.	P_D(Target Rock) = P (codes) Body Material: A105 Gr. II Sm = 19,400 psi (500°F, equivalent inlet and outlet temperature)	1.5 Sm = 29,100 psi	Inlet: S_H = 1.2 Sm = 0.77(allowable) S_R = 0.52 Sm = 0.35 (allowable) S_T = 1.2 Sm = 0.76(allowable) Outlet: S_H = 0.36 Sm = 0.24(allowable) S_R = 0.5 Sm = 0.33 (allowable) S_T = 1.36 Sm = 0.91(allowable)
2. INLET AND OUTLET STUD AREA REQUIREMENTS	Total cross-sectional area shall exceed the greater of: $Am_1 = \frac{Wm_1}{Sb}$, or $Am_2 = \frac{Wm_2}{Sa}$ where Am_1 = total required bolt (stud) area for operating conditions Am_2 = total required bolt (stud) area for gasket seating	$\left.\begin{array}{l} Am_1 = \frac{Wm_1}{Sb} \\ Am_2 = \frac{Wm_2}{Sa} \end{array}\right\}$ # Bolting Material: SA193 GR#B7 # Where Am (required minimum) is the greater of Am_1 and Am_2; and A_b (actual bolt area) must exceed Am.	Inlet: ($Am_1 > Am_2$) = 8.02 in.^{2} Outlet: Am = 4.73	Inlet: A_b (actual area) = 1.72 Am (required min.) Outlet: A_b = 2.04 Am
3. BODY WALL THICKNESS	1. Valve Wall Thickness Criterion: $t_{min} = t_A$ where t_{min} = minimum calculated thickness requirement, including corrosion allowance. t_A = Actual wall thickness. (Note: This t_{min} is t_m per notation of the codes.)	Section at inlet: $t_{RQD} < t_{ACT}$ Section at middle of body $t_{RQD} < t_{ACT_C}$	 t_{RQD} = 0.67 in. Actual thickness greater than t_m at the section under consideration.	t_{ACT} = 1.67 (t_{RQD}) t_{ACT_C} = 1.28 (t_{RQD})

TABLE 3.9-19 MAIN STEAM SAFETY/RELIEF VALVES (PILOT OPERATED), FERMI 2
(ASME Code, Section III, 1968, Including Addenda through Summer 1970)

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
	2. Cyclic Rating:	$It = \sum \frac{Nri}{Ni}$ (i = 1, 2 & 3)	It (max) = 1.0	It = 0.33[a] = 0.33 (allowable)
	Thermal $It = \sum \frac{Nri}{Ni}$ Fatigue Na ≥ 2,000 cycles, as based on Sa, where Sa is defined as the larger of	$N_a \geq$ 2,000 cycles, as based on $S_A = S_{P_2} (> S_{P_1})$, where S_A (Target Rock) = Sa (codes)	Na ≥ 2,000 cycles	Na (based on S_{P_2}) = 1.8 x 10^5 cycles: ∴ satisfies criteria
	$S_{P_1} = \left(\frac{2}{3}\right) Q_P + \frac{P_{eb}}{2} + Q_{T_2} + 1.3Q_{T_1}$ or $S_{P_2} = 0.4Q_P + \frac{K}{2}(P_{eb} + 2Q_{T_3})$ where S_{P_1} = Fatigue stress intensity at inside surface of crotch, psi. S_{P_2} = Fatigue stress intensity at outside surface of crotch, psi.	{(Uses same notation as codes)}		
4. BONNET FLANGE STRESS (BODY SIDE)	$S_{H_1} = \frac{PB_1}{4g_1} \mp \frac{6M_H}{\pi B_1 g_1^2}$ (longitude hub stress adjacent to flange) $S_{H_2} = \left(\frac{Q}{\pi B_1 t} + P\right)(Z + Y) + \frac{Et\theta_B}{B_1} + \frac{0.075PB_1}{g_1} \pm \frac{1.8M_H}{\pi B_1 g_1^2}$ (circumferential stress in hub adjacent to flange) $S_R = \frac{6(M_P + M_S)}{t^2(\pi C - nD)}$ (@ Bolt circle) $S_R = \left(\frac{Q}{\pi B_1 t} + P\right) \pm \frac{6M_S}{\pi B_1 t^2}$ (adjacent to hub) $ST = \left(\frac{Q}{\pi B_1 t} + P\right) Z \pm \left(\frac{Et\theta_B}{B_1} + \frac{1.8M_S}{\pi B_1 t^2}\right)$	$S_H < 1.5\ S_m$ $S_R < 1.5\ S_m$ $S_T < 1.5\ S_m$ P_{FD} (Target Rock) = P (codes) Material: A105 Gr. II. S_m = 19,400 psi (@500°F)	1.5 Sm = 29,100 psi	S_H = 0.82 Sm = 0.55 (allowable) S_R = 0.5 Sm = 0.33 (allowable) S_T = 0.27 Sm = 0.18 (allowable)

TABLE 3.9-19 MAIN STEAM SAFETY/RELIEF VALVES (PILOT OPERATED), FERMI 2
(ASME Code, Section III, 1968, Including Addenda through Summer 1970)

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
5. BONNET FLANGE STRESS (BONNET SIDE)	Using Roark's formula for stress and strain, Table X, 4th. Edition, superposition of case 2 and 3 $S_R = S_T = \frac{-3W}{2\pi m t^2}\left[m + (m+1)\log\frac{a}{ro} - (m-1)\frac{r_o^2}{4a^2}\right]$ $S_R = S_T = \frac{-3W}{2\pi m t^2}\left[\frac{1}{2}(m-1) + (m+1)\log\frac{a}{r_o} - (m-1)\frac{r_o^2}{2a^2}\right]$	S_R < 1.5 Sm S_T < 1.5 Sm Material: A105 Gr II Sm = 19,400 (@500°F)	1.5 Sm = 29,100 psi	$S_R = S_T$ =1.27 = 0.85 (available)
6. BONNET STUD AREA REQUIREMENTS	Total cross-sectional area shall exceed: $Am_1 = \frac{Wm_1}{Sb}$ where Am_1 = total required bolt (stud) area	$Am_1 = \frac{Wm_1}{Sb}$ Bolting Material: SA 193 Gr B7	Am_1 = 9.839	Am (actual) = 1,044 (required minimum)
7. BONNET WALL THICKNESS	Using Roark's formula for stress and strain, Table XIII, case 35, considering the circumferential stress, S_2 (the governing stress), and setting equal to Sm. $S_2 = P\frac{b^2+a^2}{b^2-a^2}$ where P = design pressure a = inside diameter b = outside diameter	$t_m < t_a$	tm = 0.119 in.	ta = 3.75 t_m
8. PILOT VALVE HOUSING FLANGE	$S_H = \frac{fM_o}{Lg_1^2B}$ $S_R = \frac{(\frac{4te}{3}+1)M_o}{Lt^2B}$ $S_T = \frac{YM_o}{t^2B} - Z\,S_R$ where S_H = Longitudinal "hub" wall stress, psi S_R = Radial "flange" stress, psi S_T = Tangential "flange" stress, psi	S_H < 1.5 Sm S_R < 1.5 Sm S_T < 1.5 Sm Material A105 Gr II Sm = 19,400 psi (@500°F)	1.5 Sm = 29,100 psi	S_H = 0.54 Sm = 0.36 (allowable) S_R = 0.36 Sm = 0.24 (allowable) S_T = 0.30 Sm = 0.20 (allowable)

TABLE 3.9-19 MAIN STEAM SAFETY/RELIEF VALVES (PILOT OPERATED), FERMI 2
(ASME Code, Section III, 1968, Including Addenda through Summer 1970)

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
9. PILOT VALVE BODY FLANGE STRESS	Using Roark's formulas for stress and strain, 4th. edition, Table X case 2	$S_R = S_T < Sm$ Material: A105 Gr II Sm = 19,400 psi (@500°F)	Sm = 19,400 psi	$S_R = S_T$ =0.34 (available)

$$S_R = S_T = \frac{-3W}{2\pi m t^2}\left[m + (m+1)\log\frac{a}{r_o} - (m-1)\frac{r_o{}^2}{4a^2}\right]$$

where

W = applied load

m = reciprocal of Poisson's ratio

a = radius of flange

r_o = radius of applied load

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
10. MAIN DISC STRESS	Using Roark's formulas for stress and strain, 4th edition, page 250	$S_{max} < Sm$ Material: SA182 Sm = 13,600 psi (@ 500°F)	Sm = 13,600 psi	S_{max} = 0.68 (allowable)

$$S_{max} = \frac{\beta W a^2}{t_o^2}$$

where

β = 1.63

W = applied load

a = radius of disc

t_o = thickness at center

11. SEISMIC CAPABILITY: Stress analysis uses $F_{vertical}$ = (mass of valve) x (2.0g) and $F_{horizontal}$ = (mass of valve) x (3.0g), with concurrent 400,000 in.-lb and 300,000 in.-lb applied at the inlet and outlet, respectively. Valve operability has been verified by test, with applied moments of 800,000 in.-lb and 600,000 in.-lb at the inlet and outlet, respectively, and at actual acceleration levels of $a_{vertical}$ = 6g and $a_{horizontal}$ = 8g. Tests were per IEEE-344 (1975).

12. VALVE LOADS: For a comparison of calculated loadings and seismic capability see Tables 3.9-24 and 3.9-25.

[a] See FP FERM 310 (Ref. 19) for maximum Cumulative Usage Factor values using event number of cycles based on Fermi 2 operating history.

TABLE 3.9-20 RECIRCULATION PUMPS

Criteria	Method of Analysis	Analytical Results	Allowable Stress or Actual Thickness
1. Casing Minimum Wall Thickness Loads:	$t = \frac{PR}{SE-0.6P} + C$	t = 2.72 in.	$S_{allow.}$ = 15,114 psi $t_{act.}$ = 2.750 in.
	where		
Normal and Upset Condition	t = min. req'd. thickness, in.		
Design pressure and temperature	P = design pressure, psig R = max. internal radius, in. S = allowable working stress, psi E = joint efficiency C = corrosion allowance, in.		
Primary membrane stress limit:			
Allowable working stress per ASME Sec. III, Class C			
2. Casing Cover Minimum Thickness Loads:	$S_S = \frac{F}{A}$ F = force A = area at shear point	S_S = 3440 psi	$S_{alow.}$ = 8775 psi $t_{act.}$ = 3.5 in.
Normal and Upset Condition	$S_b = \frac{Kqa^2}{h^2}$		
Design pressure and temperature			
Primary Bending and Shear		S_b = 6050 psi	$S_{allow.}$ = 15,114 psi
Stress limit:	q = pressure load		$t_{act.}$ = 7in.
1.5 S_m per ASME Code for pumps and valves for Nuclear Power Class I	a = radius of O.D. b = radius of I.D. h = plate thickness		
3. Cover and Seal Flange Bolt Loads:	Bolting loads, areas, and stresses shall be calculated in accordance with "Rules for Bolted Flange Connections"	Cover Flange Bolts	

TABLE 3.9-20 RECIRCULATION PUMPS

Criteria	Method of Analysis	Analytical Results	Allowable Stress or Actual Thickness
		$S_{act.}$ = 18,178 psi	$S_{allow.}$ = 20,000 psi
Normal and Upset Condition	ASME Sec. VIII, Appendix II	A_m = 91.8 in.^{2}	$A_{allow.}$ = 101.0 in.^{2}
Design pressure and temperature			
Design gasket load		Seal Flange Bolts	
		$S_{act.}$ = 18,050 psi	$S_{allow.}$ = 20,000 psi
Bolting Stress Limit:		A_m = 10.0 in.^{2}	$A_{allow.}$ = 11.1 in.^{2}
Allowable working stress per ASME Sec. III, Class C			
4. Cover Clamp Flange Thickness Loads:	Flange thickness and stress shall be calculated in accordance with "Rules for Bolted Flange Connections" - ASME Sec. VIII, Appendix II	Flange Thickness	
		t = 9.05 in.	$t_{act.}$ = 9.25 in.
Normal and Upset Condition		$S_{act.}$ = 16,870	$S_{allow.}$ = 17,500 psi
Design pressure and temperature			
Design gasket load			
Design bolting load			
Tangential Flange Stress Limit			
Allowable working stress per ASME Sec. III, Class C			
5. Seal Cover Loads:	$S_r = \frac{3w}{4t^2}\left[a^2 - 2b^2 + \frac{b^4(m-1)-4b^4(m+1)\ln\left(\frac{a}{b}\right)+a^2b^2(m+1)}{a^2(m-1)+b^2(m+1)}\right] + \frac{3W}{2\pi t^2}\left[1 - \frac{2mb^2-2b^2(m+1)\ln\frac{a}{b}}{a^2(m-1)+b^2(m+1)}\right]$	S_r = 2540 psi	S_m = 18,750 psi
Normal and Upset Condition			
Design pressure and temperature			
Design gasket load			

TABLE 3.9-20 RECIRCULATION PUMPS

Criteria	Method of Analysis	Analytical Results	Allowable Stress or Actual Thickness
	where S_r = radial stress at outer edge, psi w = pressure load, psi t = disk thickness, in. m = reciprocal of Poisson's ratio a = radius of disk, in. b = radius of disk hole, in. W = force, lb		
6. Seal Chamber Minimum Wall Thickness Loads	$t = \frac{PR}{SE-0.6P} + C$	t = 0.753 in.	$S_{allow.}$ = 15,114 psi $t_{act.}$ = 1.375 in.
Normal and Upset Condition Design pressure and temperature Piping reactions during normal operation	where t = min. required thickness, in. P = design pressure, psig R = max. internal radius, in. S = allowable working stress, psi E = joint efficiency C = corrosion allowance, in.		
Combined Stress Limit: 1.5 S_m per ASME Code for pumps and valves for Nuclear Power Class I			
7. Mounting Bracket Combined Stress Loads: Flooded weight SSE horizontal seismic force = 1.76 g	Bracket vertical loads shall be determined by summing the equipment and fluid weights and vertical seismic forces. Bracket horizontal loads shall be determined by applying the specified seismic force at mass center of pumping motor assembly (flooded)	Combined stress (shear plus tensile) Lug no.1 S_C = 6505 psi Lug no.2 S_C = 7976 psi Lug no.3 S_C = 10,762 psi	S_m = 15,150 psi S_y = 30,000 psi
SSE vertical seismic force = 0.67g			
Combined Stress Limit			
Yield stress	Horizontal and vertical loads shall be applied simultaneously to determine tensile, shear, and bending stresses in the brackets. Tensile, shear, and bending stresses shall be combined to determine max. combined stresses		

TABLE 3.9-20 RECIRCULATION PUMPS

Criteria	Method of Analysis	Analytical Results	Allowable Stress or Actual Thickness
8. Stresses Due to Seismic Loads Loads: Operation pressure and temperature SSE horizontal seismic force = 1.76g SSE vertical seismic force = 0.67g Combined Stress Limit: Yield stress	The flooded pump-motor assembly shall be analyzed as a free body supported by constant support hangers from the pump brackets. Horizontal and vertical seismic forces shall be applied at mass center of assembly and equilibrium reactions shall be determined for the motor and pump brackets. Loads, shear, and moment diagrams shall be constructed using live loads, dead loads, and calculated snubber reactions. Combined bending, tension shear stresses shall be determined for each major component of the assembly including motor support barrel, bolting and pump casing. The maximum combined tensile stress in the cover bolting shall be calculated using tensile stresses determined from loading diagram plus tensile stress from operating pressure	Motor Bolt Tensile Stresses $S_{act.}$= 10,703 psi Pump Cover Bolt Tensile Stress $S_{act.}$= 20,611 psi Motor Support Barrel Combined $S_{act.}$= 1606 psi	$S_{allow.}$ = 30,800 psi $S_{allow.}$ = 32,000 psi $S_{allow.}$ = 22,400 psi

TABLE 3.9-21 STRESS SUMMARY - HIGH-PRESSURE COOLANT INJECTION STEAM LINE AND MAIN STEAM LINE "A" (CODE USED FOR ANALYSIS: ASME III, CLASS 1, 1983 EDITION INCLUDING WINTER 1984 ADDENDA)

HPCI BRANCH LINE

	Equation 9			Equation 10	Usage[b]	
Node	Normal ($S<1.5S_m$)	Upset ($S<1.8S_m$)	Emergency ($S<2.25S_m$)	($S_n<3S_m$[a])	($U<1.0$)	(Equation 12 $S<3S_m$) / (Equation 13 $S<3S_m$)
029	8880	23278	28291	64649	0.09	11366/33486
402N	4516	11427	14087	21822	0.00	
402F	4508	7404	8941	15476	0.00	
408N	4460	8548	10408	16822	0.00	
408F	4350	8019	9598	15719	0.00	
418N	1711	7623	10235	21589	0.00	
418F	1837	7502	10330	21550	0.00	
424N	4347	6396	7404	13576	0.00	
424F	4267	6072	6936	13447	0.00	
426N	1505	6278	8719	24171	0.00	
426F	1777	6793	9373	23470	0.00	
430N	4421	7169	8437	15628	0.00	
430F	4411	7037	8358	15149	0.00	
434N	4383	7174	8611	15216	0.00	
434F	4324	7992	9989	16320	0.00	
440	4424	8852	11343	18189	0.00	
442	5204	9424	12021	18327	0.00	
448	5154	9616	12247	21501	0.00	
MAIN STEAM LINE A						
003	7715	11109	11329	26040	0.00	
004F	371	3536	4726	50897	0.03	

TABLE 3.9-21 STRESS SUMMARY - HIGH-PRESSURE COOLANT INJECTION STEAM LINE AND MAIN STEAM LINE "A" (CODE USED FOR ANALYSIS: ASME III, CLASS 1, 1983 EDITION INCLUDING WINTER 1984 ADDENDA)

	Equation 9			Equation 10	Usage[b]	
Node	Normal ($S<1.5S_m$)	Upset ($S<1.8S_m$)	Emergency ($S<2.25S_m$)	($S_n<3S_m$[a])	($U<1.0$)	(Equation 12 $S<3S_m$) / (Equation 13 $S<3S_m$)
011N	4203	5449	5778	24160	0.00	
011F	4235	5249	5519	25638	0.00	
014N	787	2559	3064	52201	0.03	
014F	737	3728	4679	47827	0.02	
017	7870	9817	10296	22607	0.00	
019	8084	10197	10709	21839	0.00	
021	8423	10154	10312	20581	0.00	
025	8348	10066	10473	20581	0.00	
030F	892	5944	8005	42402	0.02	
040N	681	5504	6674	34421	0.01	
040F	581	3963	5403	36438	0.01	
043	7795	9945	10052	26135	0.00	
051	8013	9878	9944	25238	0.00	
063	8112	9789	9833	24703	0.00	
100	8327	21309	23106	57569	0.64	15198/29490
200	8769	17618	19126	47047	0.16	
300	9174	15854	17623	42145	0.03	

[a] Per ASME Code Section III, NB-3653.6, If Equation 10 stress > $3S_m$, then Equation 12 stress must be < $3S_m$ and Equation 13 stress must be < $3S_m$

[b] See FP FERM 310 (Ref. 19) for maximum Cumulative Usage Factor values using event number of cycles based on Fermi 2 operating history.

TABLE 3.9-22 STRESS SUMMARY - REACTOR CORE ISOLATION COOLING STEAM LINE AND MAIN STEAM LINE "B" (CODE USED FOR ANALYSIS: ASME III, CLASS 1, 1983 EDITION INCLUDING WINTER 1984 ADDENDA)

	Equation 9			Equation 10	Usage[b]	Max. of Equation 12 & 13
Node	Normal ($S<1.5S_m$)	Upset ($S<1.8S_m$)	Emergency ($S<2.25S_m$)	($S_n<3.0S_m$)[a]	($U<1.0$)	($S<3S_m$)
RCIC line						
039	8704	13494	15217	51968	0.05	30112
605N	4621	6625	7643	18560	0.00	
605F	4132	5656	6299	22119	0.00	
611N	4132	6461	7182	15348	0.00	
611F	4554	7196	8482	18632	0.00	
617	5593	16871	20904	42782	0.02	
635N	1883	5557	7672	26045	0.00	
635F	1700	5526	7689	25774	0.00	
649N	1551	7211	10592	31150	0.01	
649F	1260	7057	10579	31573	0.01	
661	4514	8642	12459	22124	0.00	
663	3938	8408	12454	22469	0.00	
669	5654	9891	14072	28659	0.01	
Main Steam Line B						
003	7500	11540	11612	27372	0.01	
004F	547	4106	4547	40314	0.01	
009	16073	17389	17603	26460	0.03	
011N	4374	6560	6989	22753	0.00	
011F	4513	6397	6653	24335	0.00	
014N	1295	3898	4145	44176	0.02	
014F	636	3820	4139	44524	0.02	
019	7894	9444	9639	21963	0.00	

TABLE 3.9-22 STRESS SUMMARY - REACTOR CORE ISOLATION COOLING STEAM LINE AND MAIN STEAM LINE "B" (CODE USED FOR ANALYSIS: ASME III, CLASS 1, 1983 EDITION INCLUDING WINTER 1984 ADDENDA)

	Equation 9			Equation 10	Usage [b]	Max. of Equation 12 & 13
Node	Normal ($S<1.5S_m$)	Upset ($S<1.8S_m$)	Emergency ($S<2.25S_m$)	($S_n<3.0S_m$)[a]	($U<1.0$)	($S<3S_m$)
023	7762	9483	9667	21108	0.00	
025	7778	9417	9549	20792	0.00	
029	7772	8737	8864	20792	0.00	
030	7772	8656	8765	21006	0.00	
033	7865	8800	8883	21691	0.00	
040N	1073	3383	3646	43898	0.02	
040F	1118	3556	3836	42233	0.02	
050N	804	2499	2718	35175	0.01	
050F	1592	3360	3573	36374	0.01	
052	8005	8573	8646	25304	0.01	
059	7751	8247	8321	24636	0.00	
063	7898	8274	8315	24004	0.00	
100	9060	20770	21591	54219	0.65	31120
200	9104	20756	21718	61634	0.67	31340
300	7858	17399	17963	47002	0.26	28806
400	8160	14851	16447	48990	0.09	29426
500	8740	14811	15704	47700	0.05	30540

[a] Per ASME Code Section III, Subsection NB 3653.6, if Equation 10 stress > $3S_m$, then Equation 12 stress must be < $3S_m$, and Equation 13 stress must be < $3S_m$.

[b] See FP FERM 310 (Ref. 19) for maximum Cumulative Usage Factor values using event number of cycles based on Fermi 2 operating history.

TABLE 3.9-23 STRESS SUMMARY - RECIRCULATION LOOP "A" AND RESIDUAL HEAT REMOVAL SYSTEM RETURN (CODE USED FOR ANALYSIS: ASME III, CLASS 1, 1983 EDITION INCLUDING WINTER 1984 ADDENDA)

	Equation 9			Equation 10	
Node	Design ($S<1.5S_m$)	Upset ($S<1.8S_m/1.5S_y$)	Emergency ($S<2.25S_m/1.8\ S_y$)	($S_n<3.0S_m$)	Usage[a] ($U<1.0$)
016	11642	13283	14489	18561	0.00
063	7441	9774	16247	30228	0.00
999	7532	10148	12882	27365	0.00
198	7497	14784	25057	33240	0.00
201	7496	13986	22721	32061	0.00
204	12244	15075	18306	19550	0.00
216	11888	13077	13113	27286	0.00
222	8057	17978	31134	44948	0.02
250	8378	14137	22401	41402	0.07
340	7766	13842	22286	47633	0.02
360	7699	12192	17891	43488	0.02
802	6118	9051	12553	31567	0.04
854	11735	15827	20864	27066	0.04

[a] See FP FERM 310 (Ref. 19) for maximum Cumulative Usage Factor values using event number of cycles based on Fermi 2 operating history.

TABLE 3.9-24 STRESS SUMMARY - RECIRCULATION LOOP "B" AND RESIDUAL HEAT REMOVAL SUPPLY AND RETURN (CODE USED FOR ANALYSIS: ASME III, CLASS 1, 1983 EDITION INCLUDING WINTER 1984 ADDENDA)

	Equation 9			Equation 10	
Node	Normal ($S < 1.5\ S_m$)	Upset ($S < 1.8\ S_m/1.5\ S_y$)	Emergency ($S < 2.25\ S_m/1.8\ S_y$)	($S_n < 3.0\ S_m$)	Usage[a] ($U < 1.0$)
016	15969	17698	18866	18224	0.00
018	8051	17355	26974	41148	0.02
204	12309	14089	15581	19432	0.00
216	11888	13084	13127	27286	0.00
222	8816	18800	30222	43070	0.02
340	7764	14019	21625	48809	0.02
250	8261	13444	19544	38727	0.07
360	7706	12398	18022	43664	0.02
508	2231	7395	12933	34571	0.00
558	19579	21191	22376	19803	0.03
516	8670	16573	24308	34188	0.09
546	8050	13238	19256	29529	0.09
602	6303	9232	12302	31696	0.04
656	11864	16541	22001	27298	0.04

[a] See FP FERM 310 (Ref. 19) for maximum Cumulative Usage Factor values using event number of cycles based on Fermi 2 operating history.

TABLE 3.9-25 STRESS SUMMARY - FEEDWATER SYSTEM INSIDE DRYWELL (FW01) (CODE USED FOR ANALYSIS: ASME III, CLASS 1, 1977 EDITION INCLUDING SUMMER 1979 ADDENDA)

	Equation 9			
Node	Normal and Upset ($S < 1.5\ S_m$)	Emergency ($S < 2.25\ S_m$)	Equation 10 ($S_n < 3.0\ S_m$)	Usage[a] ($U < 1.0$)
10	6017	6501	36969	0.02
15A	5804	6312	37725	0.04
25	6037	6509	46587	0.03
30	5950	6467	47054	0.03
40	9391	12387	44992	0.07
55	11769	16142	35201	0.05
60	13564	16534	35195	0.00
180B	8000	10531	37642	0.00
205A	7064	8792	49697	0.01
215B	9533	12997	48227	0.01

[a] See FP FERM 310 (Ref. 19) for maximum Cumulative Usage Factor values using event number of cycles based on Fermi 2 operating history.

TABLE 3.9-26 STRESS SUMMARY - CORE SPRAY SYSTEM INSIDE DRYWELL (CS-02) (CODE USED FOR ANALYSIS: ASME III, CLASS 1, 1977 EDITION INCLUDING SUMMER 1979 ADDENDA)

Node	Equation 9 Design ($S < 1.5\ S_m$)	Equation 9 Emergency ($S < 2.25\ S_m$)	Equation 10 ($S_n < 3.0\ S_m$)	Usage[a] ($U < 1.0$)
10B	7668	10,101	27,315	0.00
20A	7729	10,264	29,072	0.00
25B	8257	11,263	32,917	0.00
55	5919	6912	34,689	0.01
60A	9482	13,443	48,680	0.01
70	6499	7823	35,557	0.01
75B	7600	10,021	28,766	0.00
85A	7236	9366	29,113	0.00
90	10,831	12,000	41,422	0.03

[a] See FP FERM 310 (Ref. 19) for maximum Cumulative Usage Factor values using event number of cycles based on Fermi 2 operating history.

TABLE 3.9-27 ASME CODE CLASS 2 AND 3 COMPONENTS

		Code Class	Design Pressure(psi)	Design Temp. (°F)
I.	Deleted			
II.	Nuclear boiler system			
	Vessels, valve accumulators	3	150	340
	Piping, safety/relief valve discharge	2	570	575
III.	CRD hydraulic system			
	Valves, scram discharge volume lines, and portions of vent and drain lines	2	1250	280
	Valves, insert and withdraw lines	2	1750	150 (insert)
				280 (withdrawal)
	Piping, scram discharge volume lines	2	1250	280
	Piping, insert and withdraw lines	2	1750	150 (insert)
				280 (withdrawal)
IV.	Standby liquid control system			
	Pump	3	1400	150
	Valves, beyond isolation valves	3	1400	150
	Piping, beyond isolation valves	3	1400	150
	Valves, in test and flush lines	3	1400	150
	Piping, test and flush lines	3	1400	150
	Relief valve outlet line	3	150	150
	Storage and test tank outlets to pumps	3	150	150
V.	Deleted			
VI.	RHR system			
	Heat exchangers, primary side	2	450	470
	Heat exchangers, secondary side	3	450	470
	Pumps	2	450	360
	Pump discharge piping	2	480	335

TABLE 3.9-27 ASME CODE CLASS 2 AND 3 COMPONENTS

		Code Class	Design Pressure(psi)	Design Temp. (°F)
	Shutdown suction piping	2	150	335
	Test line and containment spray	2	480 and 150	335
	Pump suction piping	2	150	335
VII.	Core spray			
	Piping, beyond shutoff valves F004 A and B (pump discharge line, bypass line, and test line)	2	500	212
	(condensate and pump suction)	2	125	212
	Pumps	2	500	40-212
	Valves, beyond shutoff valves F004 A and B (pump discharge line, bypass line, and test line)	2	500	212
	(condensate and pump suction)	2	125	212
	Shutoff valves F004 A and B and piping between the shutoff valves and outboard isolation valves F005 A and B	2	1250	575
VIII.	High-pressure coolant injection			
	Piping, and valves, steam supply beyond outermost isolation valve, other	2	1250	575
	Main pump	2	1500	40-140
	Booster pump	2	450	40-140
	Piping and valves, steam exhaust	2	150	366
	Coolant supply to barometric condenser	2	460	170
	Coolant supply to barometric condenser	2	125	170
	Pump suction from condensate storage tank (including valves)	2	18	120
	Pump suction from suppression pool, piping and valves	2	125	170
	Pump discharge to feedwater, piping and valves	2	1330	170
	Pump discharge bypass line to suppression pool, piping and valves	2	125	340

TABLE 3.9-27 ASME CODE CLASS 2 AND 3 COMPONENTS

		Code Class	Design Pressure(psi)	Design Temp. (°F)
	Test line to condensate storage tank, piping and valves	2	1330	170
	Turbine exhaust vacuum breaker line	2	150	366
IX.	RCIC system			
	Pump	2	1500	40-140
	Piping and valves in steam line to turbine, outside isolation valve	2	1250	575
	Turbine exhaust to suppression pool, piping and valves	2	150	267
	Pump suction from condensate storage tank, piping and valves	2	18	120
	Pump suction from suppression pool, piping and valves	2	125	170
	Pump discharge to feedwater line, piping and valves	2	1280	170
	Pump minimum flow line, piping and valves	2	125 and 1280	212 and 170
	System test line, piping and valves	2	1280	170
	Turbine exhaust vacuum breaker line	2	150	267
X.	RPV service equipment			
	Refueling bellows	2*	12	140
	Drywell seal bellows	2*	12	140
XI.	Radwaste system			
	Valves, containment isolation	2	150	140
	Piping, containment isolation	2	150	140
	RWCU filter-demineralizer drains to phase separator	3	150	150
	Cleanup sludge pumps	3	150	150

* Belows were designed, fabricated, and installed as ASME Class 2 but were not N-Stamped.

TABLE 3.9-27 ASME CODE CLASS 2 AND 3 COMPONENTS

		Code Class	Design Pressure(psi)	Design Temp. (°F)
XII.	Reactor water cleanup system			
	Drain from filter-demineralizer unit	3	1300	150
	Line to chemical waste tank	3	150	150
XIII.	Fuel pool cooling and cleanup system			
	Vessels, filter-demineralizers	3	200	150
	Vessels, other	3	200	150 and 140
	Heat exchangers, tube side	3	200	150
	Heat exchangers, shell side	3	150	150
	Piping	3	200	150
	Pumps	3	200	150
	Valves	3	200	150
XIV.	Offgas system			
	None			
XV.	RHR service water system			
	Piping	3	175	125-155
	Pumps	3	150	40-100
	Valves	3	175	125-155
XVI.	Plant service and cooling water systems			
	Piping and valves forming part of primary containment boundary	2	150	150
	EECW system piping and valves	3	150	150
XVII.	Instrument air systems			
	Piping and valves in lines between above accumulators and safety-related systems	3	125	150
XVIII.	Diesel generator system			
	Day tanks	3	Atmospheric pressure	125

TABLE 3.9-27 ASME CODE CLASS 2 AND 3 COMPONENTS

		Code Class	Design Pressure(psi)	Design Temp. (°F)
	Piping and valves, Fuel oil system	3	75	125
	Diesel service water system	3	125	125
	Pumps, diesel service water system	3	75	100
XIX.	Primary containment	2	56	340
XX.	Primary containment atmospheric control system			
	Piping valves and other components	2	150	340
XXI.	Standby gas treatment system			
	None			
XXII.	Reactor building ventilation system			
	None			
XXIII.	Emergency equipment area cooling units			
	Fan-coil units (coils only)	3	150	150
	Drywell cooling coils	2	150	150

TABLE 3.9-28 STANDBY LIQUID CONTROL PUMP

	Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Calculation (psi)
1.	Closure Bolting Loads: Normal and Upset	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	Stuffing box bolts 25,000	18,150
	Design pressure and temperature.		Cylinder head bolts 25,000	19,600
	Design gasket load			
	Bolting Stress Limit			
	Allowable working stress per ASME Section VIII			
2.	Wall Thickness	Pressure Area Method. Maximum stress point on fluid cylinder	16,500	9000
	Loads: Normal and Upset			
	Design pressure and temperature			
	Stress limit			
	ASME Section VIII			
3.	Motor Mount Bolts Loads: Emergency	Seismic forces acting on motor bolts subject to tension and shear	Tension 16,500 Shear 10,000	860 1220
	Design-basis earthquake			
	Stress Limit			
	0.9 yield tension and twice allowable shear ASME VIII			
4.	Nozzle Loads	For the maximum moment due to pipe reaction, the maximum force shall not exceed the allowable	Force in lb, moment in ft-lb	

TABLE 3.9-28 STANDBY LIQUID CONTROL PUMP

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Calculation (psi)
Loads: Normal Plus Upset			
Design pressure and temperature, dead weight, thermal expansion and operating-basis earthquake	Total nozzle stress with this criterion does not exceed stress limits. Mount bolts do not exceed stress limits	Suction[a]	
		F = 730	F = 90
		M = 450	M = 75
		Discharge	
		F = 350	F = 220
		M = 108	M = 63
Loads: Emergency		Suction[a]	
Design pressure and temperature, dead weight, thermal expansion and design-basis earthquake		F = 875	F = 105
		M = 540	M = 80
		Discharge	
		F = 420	F = 284
		M = 130	M = 83
Stress Limit			
ASME Section VIII for normal and upset, 1.5 of allowable stress for emergency. Mount bolts 0.9 yield for tension and twice allowable shear for emergency	--	--	--

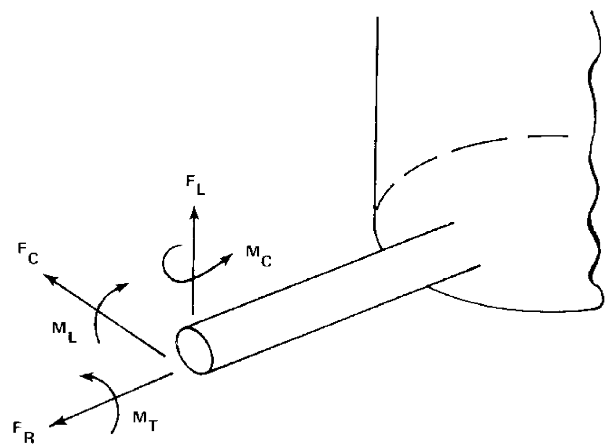
[a] Nozzle loads are the maximum allowable resultant loads applied simultaneously.

TABLE 3.9-29 STANDBY LIQUID CONTROL TANK

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Calculation (psi)
1. Shell Thickness	Minimum thickness	0.015 in.	3/16 in. (actual)
Loads: Normal and upset	$t = \frac{2.6D(H-1)G}{SE}$ in.		
Design pressure and temperature	D = Nom. I.D.		
	H = Tank height		
Stress Limit	G = Specif. gravity		
Allowable working stress per ASME Section VIII	S = Allowable stress		
	E = Joint efficiency		
	Not less than 3/16 in.		
2. Shell Stress	Loads will not produce excessive tensile or compressive (buckling) stresses	Tensile	
Loads: Emergency		18,750	9716
Design-basis earthquake nozzle load		Compressive	
		18,750	2895
Stress Limit			
ASME Section VIII Compression 1/3 yield			

TABLE 3.9-29 STANDBY LIQUID CONTROL TANK

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Calculation (psi)
3. Application of forces and moments by attaching pipe on outlet nozzle under combined maximum thermal expansion dead weight and design-basis earthquake loading reaction plus load due to internal pressure shall not produce an equivalent bending and torsional stress in the nozzles or shell in excess of the allowable stress as defined by the ASME B&PV Code Section III	Stresses will not be excessive if piping loads do not exceed the allowables	F_C = 235 lb F_L = 235 lb F_R = 105 lb M_C = 366 in.-lb M_L = 366 in.-lb M_T = 1050 in.-lb	F_C = 10 lb F_L = 50 lb F_R = 40 lb M_C = 160 in.-lb M_L = 1000 in.-lb[a] M_T = 75 in.-lb



[a] Equipment was requalified for the higher nozzle loadings.

TABLE 3.9-30 RESIDUAL HEAT REMOVAL PUMP

	Criteria	Method of Analysis	Allowable Stress psi	Calculation psi
1.	Closure Bolting Loads: Normal and Upset Design pressure and temperature Design gasket load Seismic acceleration, nozzle forces and/or moments, static mass forces Bolting Stress Limit Allowable working stress per ASME Section VIII	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	Maximum allowable stress 20,000	Maximum calculated 16,370
2.	Wall Thickness Loads: Normal and Upset Design pressure and temperature Stress Limit ASME Section VIII	Per rules of Part UG Section VIII	Maximum allowable stress main pump 17,500	Maximum calculated 14,960

TABLE 3.9-30 RESIDUAL HEAT REMOVAL PUMP

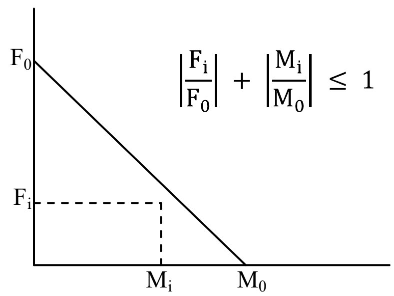
	Criteria	Method of Analysis	Allowable Nozzle Forces and Moments, Force in lb, Moment in ft – lb	Calculated Nozzle Forces and Moments
3.	Nozzle Loads Loads: Normal Plus Upset Design pressure and temperature Dead weight, force and/or moment, and operating-basis earthquake Loads: Emergency Design pressure and temperature Dead weight, force and/or moment, and design-basis earthquake Stress Limit ASME Section VIII primary local membrane stress 1.5 of allowable stress for normal and upset, 1.8 of allowable stress for emergency	For the maximum stresses due to the maximum loads	The following expression relates the allowable combination of forces and moments  $\left\lvert\frac{F_i}{F_0}\right\rvert + \left\lvert\frac{M_i}{M_0}\right\rvert \leq 1$ where Fi = Largest of the three actual external orthogonal forces (Fx, Fy, and Fz) that may be imposed by the pipe Mi = Largest of the three actual external orthogonal moments (Mx, My, and Mz) permitted from the pipe when they are combined simultaneously for any condition Fo = Allowable value of Fi when all moments are zero Mo = Allowable value of Mi when all forces are zero The values of Fo and Mo are given below Normal Plus Upset:	
			Suction: Fo = 10,440 Mo = 49,190	Force in lb, moment in ft–lb
			Discharge: Fo = 7030 Mo = 26,410	Emergency Loads[a]:
			Emergency:	Suction (pump D):

TABLE 3.9-30 RESIDUAL HEAT REMOVAL PUMP

Criteria	Method of Analysis	Allowable Nozzle Forces and Moments, Force in lb, Moment in ft – lb		Calculated Nozzle Forces and Moments
		Suction:	Fo = 12,520 Mo = 59,030	F_R = 25,000 M_R =82,800
		Discharge:	Fo = 8430 Mo = 31,700	Discharge (Pump B): F_R = 23,200 M_R = 56,000

[a] Equipment was requalified for higher nozzle loadings.

TABLE 3.9-31 RESIDUAL HEAT REMOVAL HEAT EXCHANGER

	Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Actual (psi)
1.	Closure Bolting Loads: Normal and Upset Design pressure and temperature Design gasket load	Bolting loads and stresses are calculated per “Rules for Bolted Flange Connections” ASME Section VIII, App. II		
	Bolting Stress Limit Allowable working stress per ASME Section VIII	Shell-channel bolted joint	25,000	24,675
2.	Wall Thickness Loads: Normal and Upset Design pressure and temperature Stress Limit	Shell side ASME Section III C, TEMA Class C Tube side ASME Section VIII and TEMA Class C		
	ASME Section VIII	a. Shell	0.830 in.	1.125 in.
		b. Shell cover	0.805 in.	1.00 in.
		c. Channel ring	0.832 in.	1.00 in.
		d. Tubes	0.044 in.	0.049/0.053
		e. Channel cover	6.627 in.	6.625 in.
		f. Tube sheet	6.697 in.	6.750 in.
3.	Nozzle Loads Design Pressure and Temperature	The maximum moments due to pipe reaction and the maximum forces shall not exceed the allowable limits	(See below)	(See below and next page)
	Dead weight, thermal expansion design-basis earthquake	Primary stress less than 1.5 ASME Section VIII allowable		

TABLE 3.9-31 RESIDUAL HEAT REMOVAL HEAT EXCHANGER

Allowable limits

	N1	N2	N3	N4
F_x	2975 lb	2975 lb	5310 lb	2975 lb
F_y	6690	6690	2360	6690
F_z	6690	6690	5310	6690
M_x	179,600 in.-lb	179,600 in.-lb	47,200 in.-lb	179,600 in.-lb
M_y	59,460	59,460	142,600	59,460
M_z	59,460	59,460	47,200	59,460

Actual Emergency Loads – Heat Exchanger E1101B001A[a b]

	N1	N2	N3	N4
F_x	2713 lb	2314 lb	541 lb	3171 lb
F_y	1235	1732	8176	2136
F_z	3841	3686	2245	6922
M_x	48,768 in.-lb	34,080 in.-lb	49,908 in.-lb	149,676 in.-lb
M_y	104,412	81,768	28,632	31,368
M_z	39,444	72,252	21,096	111,660

Actual Emergency Loads – Heat Exchanger E1101B001B[a]

	N1	N2	N3	N4
F_x	4520 lb	2060 lb	4460 lb	1790 lb
F_y	2180	5070	4290	1410
F_z	2690	1580	870	1630
M_x	119,680 in.-lb	130,660 in.-lb	25,640 in.-lb	45,260 in.-lb
M_y	60,200	115,320	50,000	21,780
M_z	59,980	154,550	145,520	126,770

[a] Equipment was requalified for the higher nozzle loadings.

[b] Reference: Stress Report DC-2966

TABLE 3.9-32 CORE SPRAY PUMP

	Criteria	Method of Analysis	Allowable Stress psi	Calculation psi
1.	Closure Bolting Loads: Normal and Upset Design pressure and temperature Design gasket load Seismic acceleration, nozzle forces and/or moments, static mass forces Bolting Stress Limit Allowable working stress per ASME Section VIII	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	Maximum allowable stress 20,000	Maximum calculated 18,000
2.	Wall Thickness Loads: Normal and Upset Design pressure and temperature Stress Limit ASME Section VIII	Per rules of Part UG Section VIII	Maximum allowable stress main pump 17,500	Maximum calculated 11,680

TABLE 3.9-32 CORE SPRAY PUMP

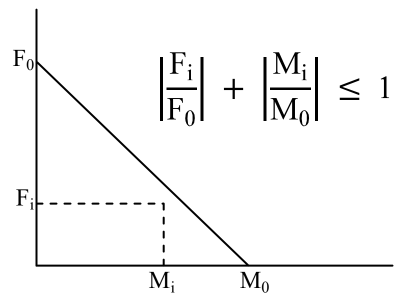
Criteria	Method of Analysis	Allowable Nozzle Forces and Moments, Force in lb, Moment in ft – lb	Calculated Nozzle Forces and Moments
3. Nozzle Loads Loads: Normal Plus Upset Design pressure and temperature Dead weight, force and/or moment, and operating-basis earthquake Loads: Emergency Design pressure and temperature Dead weight, force and/or moment, and design-basis earthquake Stress Limit ASME Section VIII Primary local membrane stress 1.5 of allowable stress for normal and upset, 1.8 of allowable stress for emergency	For the maximum stresses due to the maximum loads	The following expression relates the allowable combination of forces and moments  $\left\|\frac{F_i}{F_0}\right\| + \left\|\frac{M_i}{M_0}\right\| \leq 1$ where Fi = Largest of the three actual external orthogonal forces (F_x, F_y, and F_z) that may be imposed by the pipe Mi = Largest of the three actual external orthogonal moments (M_x, M_y, and M_z) permitted from the pipe when they are combined simultaneously for any condition Fo = Allowable value of Fi when all moments are zero Mo = Allowable value of Mi when all forces are zero	
		The values of Fo and Mo are given below Normal Plus Upset: Suction: Fo = 4540 Mo = 13,600 Discharge: Fo = 3550 Mo = 8800	Force in lb, moment in ft–lb Maximum Emergency Loads[a]: Suction (pump B): F_R = 21,000 M_R = 58,700

TABLE 3.9-32 CORE SPRAY PUMP

Criteria	Method of Analysis	Allowable Nozzle Forces and Moments, Force in lb, Moment in ft – lb		Calculated Nozzle Forces and Moments
		Emergency:		
		Suction:	Fo = 5450 Mo = 16,320	Discharge (Pump A): F_R = 7600 M_R = 25,200
		Discharge:	Fo = 4260 Mo = 10,570	

[a] Equipment was requalified for the higher nozzle loadings.

TABLE 3.9-33 HIGH-PRESSURE COOLANT INJECTION TURBINE

	Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Calculation (psi)
1.	Closure Bolting Loads: Normal and Upset Design pressure and temperature Design gasket load Bolting Stress Limit Allowable working stress per ASME Section VIII	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	Maximum allowable stress 20,000	Maximum calculated 18,290
2.	Casing Wall Thickness Loads: Normal and Upset Design pressure and temperature Stress Limit ASME Section VIII	Per rules of Part UG Section VIII	Maximum allowable stress 17,500	Maximum calculated 7200
3.	Nozzle Loads Loads: Normal Design pressure and temperature Dead weight and thermal expansion	For the resultant moment due to pipe reaction, the resultant force shall not exceed the allowable Detailed design analysis has demonstrated the acceptability of these values	Force in lb, moment in ft - lb Inlet F = (7570 – M)/3 Exhaust F = (9930 – M)/3	 F_R = 1320 M_R = 3370 F_R = 1090 M_R = 3560
	Loads: Normal plus Upset Design pressure and temperature Dead weight, thermal expansion, and operating-basis earthquake		Inlet F = (16,000 – M)/4 Exhaust F = (20,000 – M)/0.8	F_R = 1970 M_R = 4690 F_R = 2280 M_R = 9770
	Loads: Emergency Design pressure and temperature Dead weight, thermal expansion, and design-basis earthquake		Inlet F = (16,000 – M)/4 Exhaust F = (20,000 – M)/0.8	F_R = 1970 M_R = 4690 F_R = 2590 M_R = 11,600

TABLE 3.9-33 HIGH-PRESSURE COOLANT INJECTION TURBINE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Calculation (psi)
Stress limits Specified by vendor for normal, ASME Section VIII for upset, increased 20 percent for emergency			
4. Turbine Mounting Bolts (turbine to baseplate) Loads: Normal and Upset Operating-basis earthquake Nozzle loads for OBE, dead weight and thermal expansion Loads: Emergency Design-basis earthquake Nozzle loads for design-basis earthquake, dead weight, and thermal expansion Stress limits ASME Section VIII allowable for normal and upset. For emergency 0.9 yield and twice allowable shear	Vertical and horizontal forces on mounting bolts calculated as the sum of seismic accelerations on the turbine and the pipe reaction forces and moments on the nozzles	Tensile and shear stress for bolting materials as specified in ASME Section VIII	By meeting the nozzle load criteria of 3 above, the detailed seismic analysis indicates the mounting bolts satisfy the allowable stress requirements

TABLE 3.9-34 HIGH-PRESSURE COOLANT INJECTION PUMP

	Criteria	Method of Analysis	Allowable Stress Thickness Required (psi)	Calculation (psi)
1.	Closure Bolting	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	Main	Maximum calculated
	Loads: Normal and Upset		Maximum allowable stress 20,000	19,950
			Booster	
	Design pressure and temperature		20,000	17,400
	Design gasket load			
	Bolting Stress Limit			
	Allowable working stress per ASME Section VIII			
2.	Casing Wall Thickness	Per rules of Part UG Section VIII nozzle stress maximum case stress	Maximum allowable stress	Maximum calculated
			main pump 14,000	12,050
	Loads: Normal and Upset		booster pump 14,000	3650
	Design pressure and temperature			
	Stress Limit			
	ASME Section VIII			
3.	Nozzle Loads	For the maximum resultant moment due to pipe reaction, the maximum resultant force shall not exceed the allowable	Force in lb, moment in ft-lb	
	Loads: Normal Plus Upset			
	Design pressure and temperature	Total nozzle stress with this criterion does not exceed stress limits	Suction F = 33,000 – 0.79M	F_R = 6370 M_R = 22,240
	Dead weight, thermal expansion, and operating-basis earthquake		Discharge F = 32,000 – 1.54M	F_R = 4730 M_R = 15,630

TABLE 3.9-34 HIGH-PRESSURE COOLANT INJECTION PUMP

Criteria	Method of Analysis	Allowable Stress Thickness Required (psi)	Calculation (psi)
Loads: Emergency			
Design pressure and temperature Dead weight, thermal expansion, and design-basis earthquake		Suction F = 43,000 – 0.74M	F_R = 10,690 M_R = 28,890
		Discharge F = 47,000 – 1.23M	F_R = 7020 M_R = 24,220
Stress Limit			
ASME Section VIII for normal and upset, 1.5 of allowable stress for emergency			

TABLE 3.9-35 REACTOR CORE ISOLATION COOLING TURBINE

	Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Calculation (psi)
1.	Closure Bolting Loads: Normal and Upset Design pressure and temperature Design gasket load Bolting Stress Limit Allowable working stress per ASME Section VIII	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	Maximum allowable stress 20,000	Maximum calculated 6400
2.	Casing Wall Thickness Loads: Normal and Upset Design pressure and temperature Stress Limit ASME Section VIII	Per rules of Part UG Section VIII	Maximum allowable stress 17,500	Maximum calculated 12,700
3.	Nozzle Loads	For the resultant moment due to pipe reaction, the resultant force shall not exceed the allowable	Force in lb, moment in ft-lb	
	Loads: Normal Design pressure and temperature Dead weight and thermal expansion	Detailed design analysis has demonstrated the acceptability of these values	Inlet F = (2,620 – M)/3	F_R = 50 M_R = 250
			Exhaust F = (6,000 – M)/3	F_R = 790 M_R = 2550
	Loads: Normal plus Upset Design pressure and temperature		Inlet F = (7,000 – M)/4.7	F_R = 620 M_R = 530
	Dead weight, thermal expansion, and operating-basis earthquake		Exhaust F = 3(10,000 – M) but not to exceed 10,000 lb	F_R = 1710 M_R = 5350

TABLE 3.9-35 REACTOR CORE ISOLATION COOLING TURBINE

	Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Calculation (psi)
	Loads: Emergency Design pressure and temperature		Inlet F = (7,000 – M)/4.7	F_R = 630 M_R = 550
	Dead weight, thermal expansion, and design-basis earthquake		Exhaust F = 3(10,000 – M) but not to exceed 10,000 lb	F_R = 3420 M_R = 8470
	Stress limits Specified by vendor for normal, ASME Section VIII for upset, increased 20 percent for emergency			
4.	Turbine Mounting Bolts (turbine to baseplate) Loads: Normal and Upset	Vertical and horizontal forces on mounting bolts calculated as the sum of seismic accelerations on the turbine and the pipe reaction forces and moments on the nozzles		
	Operating-basis earthquake Nozzle loads for operating-basis earthquake, dead weight and thermal expansion		Tensile and shear stress for bolting materials as specified in ASME Section VIII	By meeting the nozzle load criteria of 3 above, the detailed seismic analysis indicates the mounting bolts satisfy allowable stress requirements
	Loads: Emergency Design-basis earthquake Nozzle loads for design-basis earthquake, dead weight, and thermal expansion		Tensile stress less than 0.9 yield and shear stress less than twice allowable of ASME Section VIII	

TABLE 3.9-35 REACTOR CORE ISOLATION COOLING TURBINE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Calculation (psi)
Stress Limits			
ASME Section VIII allowable for normal and upset. For emergency 0.9 yield and twice allowable shear			

TABLE 3.9-36 REACTOR CORE ISOLATION COOLING PUMP

	Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Calculation (psi)
1.	Closure Bolting Loads: Normal and Upset Design pressure and temperature Design gasket load	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	Maximum allowable stress 25,000	Maximum calculated 22,600
	Bolting Stress Limit Allowable working stress per ASME Section VIII			
2.	Casing Wall Thickness	Per rules of Part UG Section VIII	Maximum allowable stress	Maximum calculated
	Loads: Normal and Upset Design pressure and temperature	Nozzle Stress	Main pump 17,500	5350
	Stress Limit ASME Section III	Volute stress is calculated per Roark's "Formulas for Stress and Strain"	Main pump 17,500	9200
3.	Nozzle Loads	For the maximum moment due to pipe reaction, the maximum force shall not exceed the allowable	Force in lb, moment in ft-lb	
	Loads: Normal plus Upset Design pressure and temperature Dead weight, thermal expansion and operating-basis earthquake	Total nozzle stress with this criterion does not exceed stress limits	Suction $F = 9400 - 2.50M$ Discharge $F = 9400 - 4.33M$	$F_R = 420$ $M_R = 910$ $F_R = 980$ $M_R = 1420$
	Loads: Emergency Design pressure and temperature		Suction $F = 19,000 - 2.42M$	$F_R = 750$ $M_R = 1670$

TABLE 3.9-36 REACTOR CORE ISOLATION COOLING PUMP

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required (psi)	Calculation (psi)
Dead weight, thermal expansion, and design-basis earthquake		Discharge	$F_R = 1060$
		$F = 19{,}000 - 5.05M$	$M_R = 1890$
Stress limit			
ASME Section VIII for normal and upset, 1.5 of allowable stress for emergency			

TABLE 3.9-37 STRESS SUMMARY - EMERGENCY EQUIPMENT COOLING WATER SYSTEM[a,b,c]

(Code used for analysis: ASME III, 1977 Edition, including Winter 1978 Addenda, Class 3)

Node	Equation 8	Equation 9, Upset	Equation 9, Emergency
40	1110	1430	1450
45A	1310	1690	1720
45B	1090	1290	1310
65	1250	2120	2190
80	1380	1660	1680
85	1520	2550	2610
90	1060	1390	1410
130	990	1120	1130
135A	1000	1230	1250
135B	987	1140	1160
200	1570	2760	2810
201	1620	2650	2680
330	1150	1390	1390
335A	1360	2210	2260
335B	1330	2000	2020
340	1170	1410	1400
345A	1380	1810	1790
345B	1030	1490	1480
352	1610	2190	2200
353	1610	2150	2160
358	1050	1410	1420
360A	1110	1690	1710
360B	974	1290	1300
365B	951	1370	1390

TABLE 3.9-37 STRESS SUMMARY - EMERGENCY EQUIPMENT COOLING WATER SYSTEM[a,b,c]

(Code used for analysis: ASME III, 1977 Edition, including Winter 1978 Addenda, Class 3)

Node	Equation 8	Equation 9, Upset	Equation 9, Emergency
370	1040	1580	1610
375	1040	1390	1410
Allowable	15,000	18,000	27,000

Reference: Stress Report DC – 2955.

[a] See Figure 3.9-17.

[b] In accordance with our snubber reduction program criteria, systems with low design temperatures are not subjected to rigorous thermal expansion analysis.

[c] Stresses are in pounds per square inch.

TABLE 3.9-38 STRESS SUMMARY - RHR SERVICE WATER RETURN LINE[a,b]

(Code used for analysis: ASME III, 1977 Edition, including Summer 1979 Addenda, Class 3)

Node	Equation 8	Equation 9, Upset	Equation 9, Emergency	Equation 10
600A	2840	4943	4574	3752
600B	3315	5562	5134	1706
602B	3337	5702	5291	2147
615A	3601	7336	6623	5703
615B	3162	5138	4846	5049
630	4034	5203	5236	4464
640A	3540	7244	7040	7494
640B	2935	6308	6001	8405
655A	3067	5910	5511	3633
655B	2943	6705	6126	2013
656B	2901	7293	6637	2348
682A	3214	9629	8584	3667
682B	4017	7149	6677	3225
684	3168	8383	7529	927
710A	3191	5336	5072	1174
710B	3406	5287	5007	490
716A	3576	5658	5314	783
716B	2880	4183	3969	1087
718A	2929	4442	4203	1372
718B	3101	5546	5231	1411
730	2977	4948	4729	1021
Allowable	15,000	18,000	27,000	22,500

Reference: Stress Report DC-2965.

[a] See Figure 3.9-18.

[b] Stresses are in pounds per square inch.

TABLE 3.9-39 STRESS SUMMARY - RHR CONTAINMENT SPRAY SYSTEM[a,b]

(Code used for analysis: ASME III, 1977 Edition, including Winter 1978 Addenda, Class 2)

Node	Equation 8	Equation 9, Upset	Equation 9, Emergency	Equation 10	Equation 11
8	3920	5650	5410	1730	5650
10A	4060	5620	5410	1920	5980
10B	3790	4790	4650	1640	5430
15A	4250	6320	6020	3240	7490
15B	4200	6160	5860	4070	8270
20A	3870	4760	4590	5530	9400
20B	3810	4500	4430	5420	9230
40A	4320	6150	5790	5050	9370
40B	4330	5830	5530	5630	9960
42	4390	5830	5510	3920	8300
62	4420	7080	6630	13,500	17,900
75A	6400	9920	9110	13,100	19,500
75B	6180	9590	8750	11,300	17,500
80	8460	15,000	13,400	19,400	27,800
91	4740	7410	6870	5320	10,100
100A	5010	6770	6470	15,500	20,500
100B	5020	6970	6620	15,400	20,500
115	4260	7390	6950	11,500	15,800
Allowable	15,000	18,000	27,000	22,500	37,500

Reference: Stress Report DC – 2972 Vol IA DCD1 Rev B

[a] See Figure 3.9-19.

[b] Stresses are in pounds per square inch.

TABLE 3.9-40 RECOMMEDED SPAN FOR DEADWEIGHT SUPPORTS

		Water-Filled		Gas-Filled		
Nominal Diameter	Schedule	Span[a]	Load (lb)	Span[a]	Load (lb)	Rod Diameter
1/2	40	6 ft 5 in.	12	6 ft 5 in.	10	
	80	6 ft 6 in.	12	6 ft 10 in.	12	3/8
	160	6 ft 6 in.	14	6 ft 7 in.	14	
3/4	40	7 ft 7 in.	16	8 ft 4 in.	16	
	80	7 ft 11 in.	20	8 ft 4 in.	18	3/8
	160	7 ft 10 in.	22	8 ft 1 in.	22	
1	40	9 ft 0 in.	26	9 ft 10 in.	24	
	80	9 ft 2 in.	32	9 ft 10 in.	30	3/8
	160	9 ft 2 in.	38	9 ft 6 in.	38	
1-1/4	40	10 ft 5 in.	42	11 ft 7 in.	38	
	80	10 ft 9 in.	50	11 ft 6 in.	46	3/8
	160	10 ft 10 in.	58	11 ft 5 in.	56	
1-1/2	40	11 ft 4 in.	52	12 ft 10 in.	48	
	80	11 ft 9 in.	64	12 ft 10 in.	58	3/8
	160	11 ft 10 in.	80	12 ft 6 in.	76	
2	40	12 ft 9 in.	82	14 ft 9 in.	72	
	80	13 ft 2 in.	102	14 ft 9 in.	94	3/8
	160	13 ft 6 in.	136	14 ft 4 in.	128	
2-1/2	40	14 ft 6 in.	138	16 ft 9 in.	120	
	80	15 ft 0 in.	168	16 ft 7 in.	152	1/2
	160	15 ft 1 in.	202	16 ft 2 in.	190	

TABLE 3.9-40 RECOMMEDED SPAN FOR DEADWEIGHT SUPPORTS

		Water-Filled		Gas-Filled		
Diameter	Diameter	Diameter	Diameter	Diameter	Diameter	Diameter
3	40	15 ft 6 in.	210	18 ft 1 in.	180	
	80	16 ft 2 in.	248	18 ft 1 in.	232	1/2
	160	16 ft 7 in.	326	17 ft 10 in.	306	
3-1/2	40	16 ft 7 in.	272	19 ft 9 in.	232	
	80	17 ft 5 in.	340	19 ft 9 in.	302	5/8
	XXS	17 ft 11 in.	520	18 ft 11 in.	496	
4	40	17 ft 7 in.	346	21 ft 2 in.	290	
	80	18 ft 7 in.	436	21 ft 3 in.	384	5/8
	160	19 ft 2 in.	584	20 ft 6 in.	542	

[a] The actual span should not exceed the recommended value by more than 1 ft.

TABLE 3.9-41 MINIMUM OFFSET NEAR NOZZLES WITH EXPANSION MOVEMENT[a]

Nominal Diameter (in.)	Deflection (in.) 1/4	1/2	3/4	1	1-1/4	1-1/2	1-3/4	2	2-1/4	2-1/2
1/2	3ft 11in.	5ft 6in.	6ft 9in.	7ft 9in.	8ft 9in.	9ft 6in.	10ft 3in.	11ft 0in.	11ft 8in.	12ft 4in.
3/4	4ft 4in.	6ft 2in.	7ft 6in.	8ft 9in.	9ft 9in.	10ft 8in.	11ft 6in.	12ft 4in.	13ft 0in.	13ft 9in.
1	4ft 10in.	6ft 11in.	8ft 5in.	9ft 9in.	10ft 11in.	11ft 11in.	12ft 11in.	13ft 9in.	14ft 7in.	15ft 5in.
1-1/4	5ft 6in.	7ft 9in.	9ft 6in.	10ft 11in.	12ft 3in.	13ft 5in.	14ft 6in.	15ft 6in.	16ft 5in.	17ft 4in.
1-1/2	5ft 10in.	8ft 3in.	10ft 2in.	11ft 9in.	13ft 1in.	14ft 4in.	15ft 6in.	16ft 6in.	17ft 6in.	18ft 6in.
2	6ft 6in.	9ft 3in.	11ft 4in.	13ft 1in.	14ft 7in.	16ft 0in.	17ft 4in.	18ft 6in.	19ft 7in.	20ft 8in.
2-1/2	7ft 2in.	10ft 2in.	12ft 6in.	14ft 4in.	16ft 1in.	17ft 8in.	19ft 0in.	20ft 4in.	21ft 7in.	22ft 9in.
3	7ft 11in.	11ft 3in.	13ft 9in.	15ft 10in.	17ft 9in.	19ft 5in.	21ft 0in.	22ft 5in.	23ft 10in.	25ft 1in.
3-1/2	8ft 6in.	12ft 0in.	14ft 9in.	17ft 0in.	19ft 0in.	20ft 9in.	22ft 6in.	24ft 0in.	25ft 6in.	26ft 10in.
4	9ft 0in.	12ft 9in.	15ft 7in.	18ft 0in.	20ft 2in.	22ft 0in.	23ft 10in.	25ft 6in.	27ft 0in.	28ft 6in.

[a] This is the minimum length of pipe which is installed perpendicular to the direction of nozzle movement between the nozzle and the first restraint which acts in that direction. Movements in three orthogonal directions are considered.

TABLE 3.9-42 MINIMUM OFFSET REQUIRED TO ACCOMMODATE THERMAL EXPANSION OF PIPING

Nominal Diameter (in.)	Expanding Length (ft)	Design Temp. (°F)	Piping Material	Offset Required (ft – in.)
1/2	5	500	LCS[a]	2 ft 7 in.
3/4	10	325	LCS	3 ft 1 in.
1	15	150	LCS	2 ft 3 in.
1-1/4	20	575	LCS	7 ft 10 in.
1-1/2	25	475	LCS	8 ft 3 in.
2	30	450	LCS	9 ft 9 in.
2-1/2	35	275	LCS	8 ft 3 in.
3	40	175	LCS	6 ft 10 in.
3-1/2	45	225	LCS	9 ft 6 in.
4	50	300	LCS	13 ft 1 in.
1/2	5	500	AUS[b]	3 ft 0 in.
3/4	10	325	AUS	3 ft 6 in.
1	15	150	AUS	2 ft 9 in.
1-1/4	20	575	AUS	9 ft 3 in.
1-1/2	25	475	AUS	9 ft 7 in.
2	30	450	AUS	11 ft 4 in.
2-1/2	35	275	AUS	9 ft 11 in.
3	40	175	AUS	8 ft 4 in.
3-1/2	45	225	AUS	11 ft 6 in.
4	50	300	AUS	15 ft 8 in.

[a] LCS = Low carbon steel (SA-106 grade B or equivalent).

[b] AUS = Austenitic steel (SA-312 TP304L or 316L or equivalent).

TABLE 3.9-43 SAFETY-RELATED MECHANICAL COMPONENTS NOT COVERED BY ASME CODE

	Principal Component	Table or Subsection Number[a]	Design Code	Qualification Method
I.	Reactor system			
	CRD housing supports	-	AISC	Analytical
	Reactor internal structures, engineered safety features	4.5	NA	Analytical, empirical
	Control rods	4.5	NA	Prototype tests
	Control rod drives	4.5	NA	Analytical and prototype tests
	Core support structure	4.5	NA	Analytical
	Reactor vessel stabilizer	-	AISC	Analytical
	Fuel assemblies	4.5		
II.	Recirculation system			
	Pipe restraints, recirculation line	3.9.2.1 - 3.9.2.2	AISC	Analytical and tests
III.	CRD hydraulic system			
	Hydraulic control unit	4.5.2.3	ASME, ANSI	Analytical, prototype tests
IV.	Standby liquid control system[b]		API-620	Seismic analyses
	Atmospheric storage tank		API-650	
V.	High-pressure coolant injection		ASME Section VIII	Analytical
	Turbine			
VI.	RCIC System		ASME Section VIII	Analytical
	Turbine			

TABLE 3.9-43 SAFETY-RELATED MECHANICAL COMPONENTS NOT COVERED BY ASME CODE

	Principal Component	Table or Subsection Number[a]	Design Code	Qualification Method
VII.	RHR service water system		AISC, ACI	Analytical
	Mechanical draft cooling towers			
VIII.	Diesel generator systems		DEMA, ANSI, IEEE, NEMA	Analytical
	Diesel generators			
IX.	Standby gas treatment system		AMCA, SMACNA, ORNL-NSIC-65	Analytical, prototype tests
	All components with safety functions			
X.	Reactor building ventilation		AMCA, SMACNA	Analytical
	All components with safety function			
XI.	Emergency equipment area cooling units		AMCA, SMACNA	Analytical
XII.	Reactor building crane		CMAA, ASTM	Analytical, testing

[a] Location of summary of stress and dynamic calculations or experimental testing.

[b] SLCS was not originally intended, procured, designed or classified as safety-related, but it is maintained and tested as a safety-related system.

TABLE 3.9-44 OPERABILITY ASSURANCE PROGRAM CRITERIA COMPARISON

NRC Criteria (SRP 3.9.3.-II.2)	Fermi Criteria (UFSAR Section)	Comments, Subject
(2)(a)	3.9.4.2	SSE per IEEE 344, Environmental per IEEE 382-1972
(2)(b)i, ii	3.7.3.1(b)	Seismic stress cycling (valves)
	3.7.3.16.1	Seismic analysis, static-peak floor response or dynamic - actual eigen frequencies
	3.9.4.2	Valve seismic operability
	3.9.4.3	Pump seismic operability
	Tables 3.9-17 through 3.9-20, 3.9-28 through 3.9-36	Seismic and stress analysis details for pumps and valves
(2)(b)iii	3.9.1.2.b.4	Feedwater check valve disk analysis
(2)(b)iv	N/A	No essential primary coolant pump in BWRs
(2)(b)v	N/A See above	ECCS pumps are not LOCA-affected
(2)(b)vi	Tables 3.9-17 through 3.9-20, 3.9-28 through 3.9-36	Stress analysis details and wall thickness calculations
(2)(b)vii	3.7.3.16.1	See also above under (2)(b)i, ii
	3.9.2.2.1	Class 2 and 3 pumps, design criteria
	3.9.2.4	Analytical and empirical design methods for Class 2 and 3 pumps and valves
(3)(b)	3.9.2.2.4.1	Design analysis for supports and anchor bolts Based on AISC Criteria which is the basis for ASME Section III Subsection "NF."

TABLE 3.9-45 MAXIMUM CUMULATIVE USAGE FACTORS (CUF) BASED ON PLANT OPERATING HISTORY

Table 3.9-45 has been deleted.

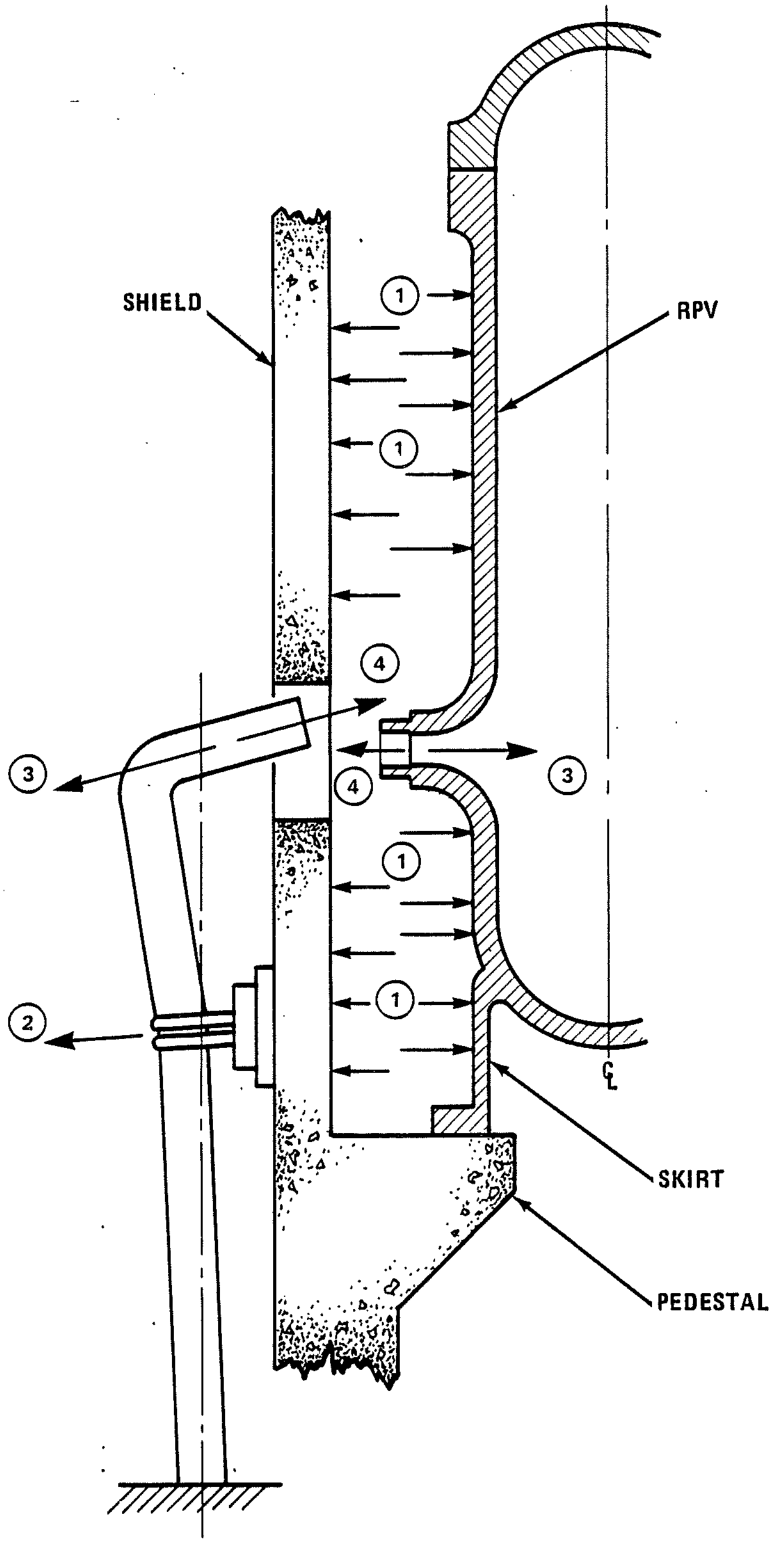


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FIGURE 3.9-1

LOADING DESCRIPTION

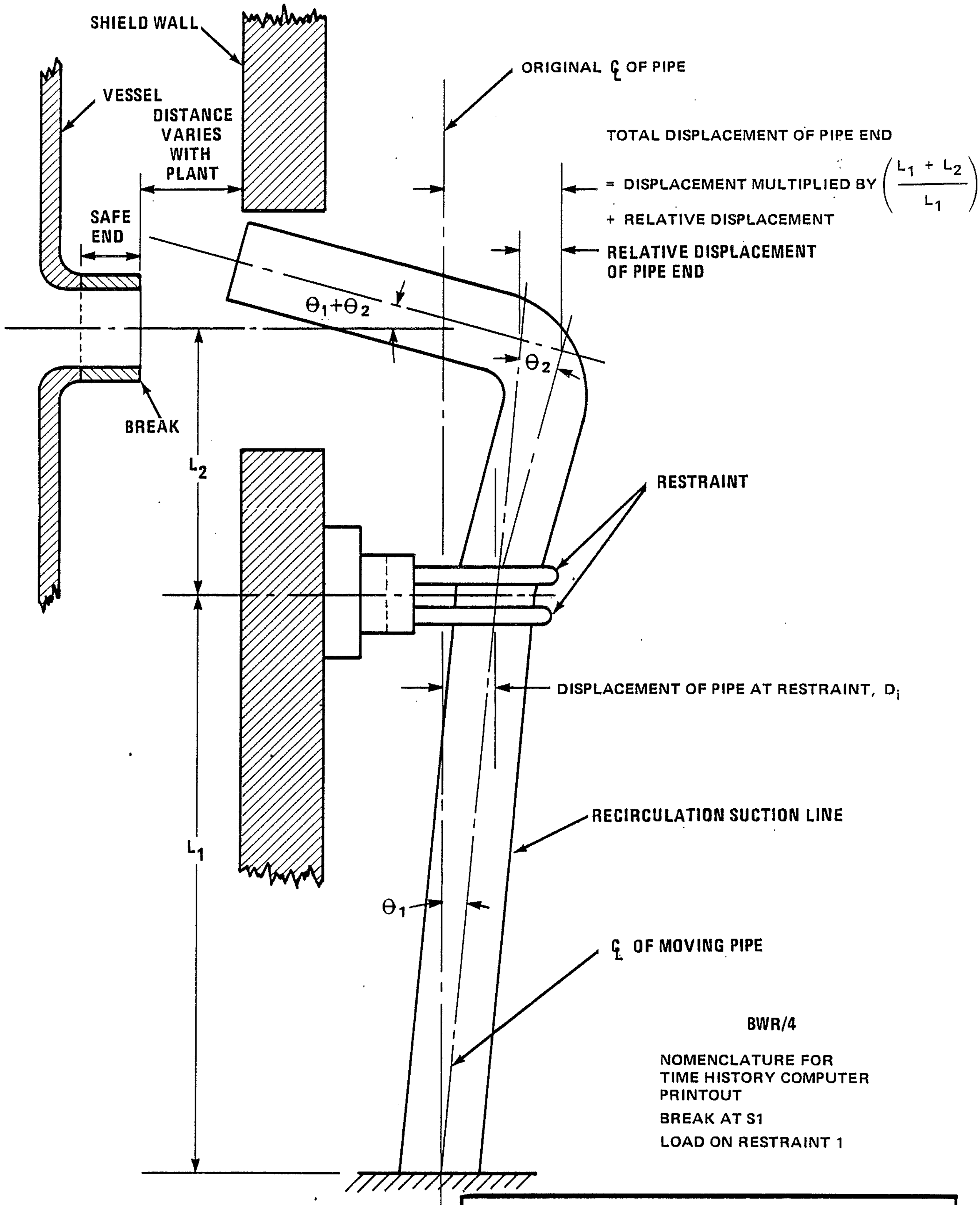

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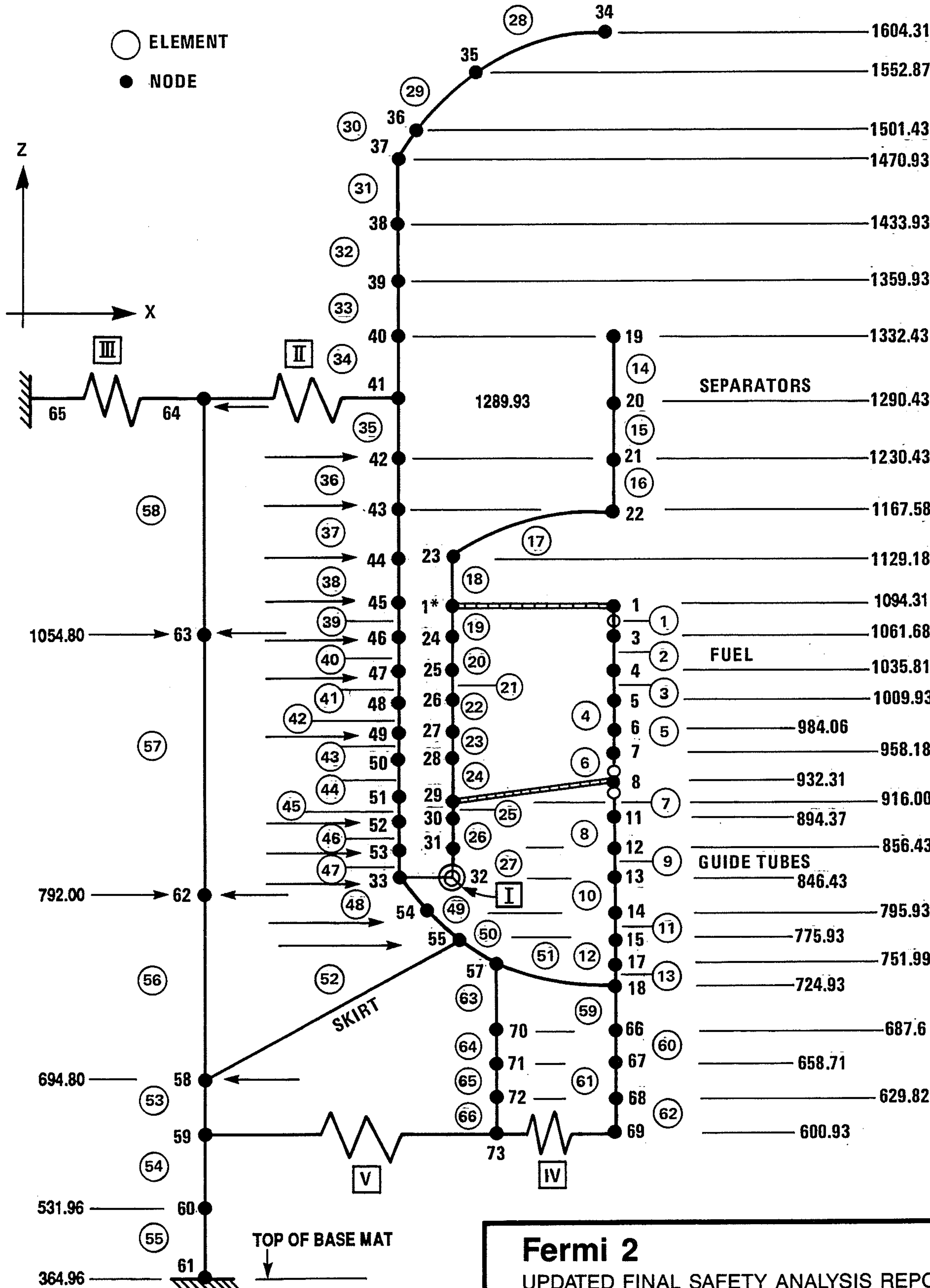
FIGURE 3.9-2

NOMENCLATURE FOR JET LOAD CALCULATION



NOTE: MODEL NODAL ELEVATION IS IN INCHES. ADD 6095.04 INCHES TO OBTAIN ACTUAL SITE ELEVATION.

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FIGURE 3.9-3

HORIZONTAL BEAM MODEL

FIGURES 3.9-4 AND 3.9-5 HAVE BEEN DELETED

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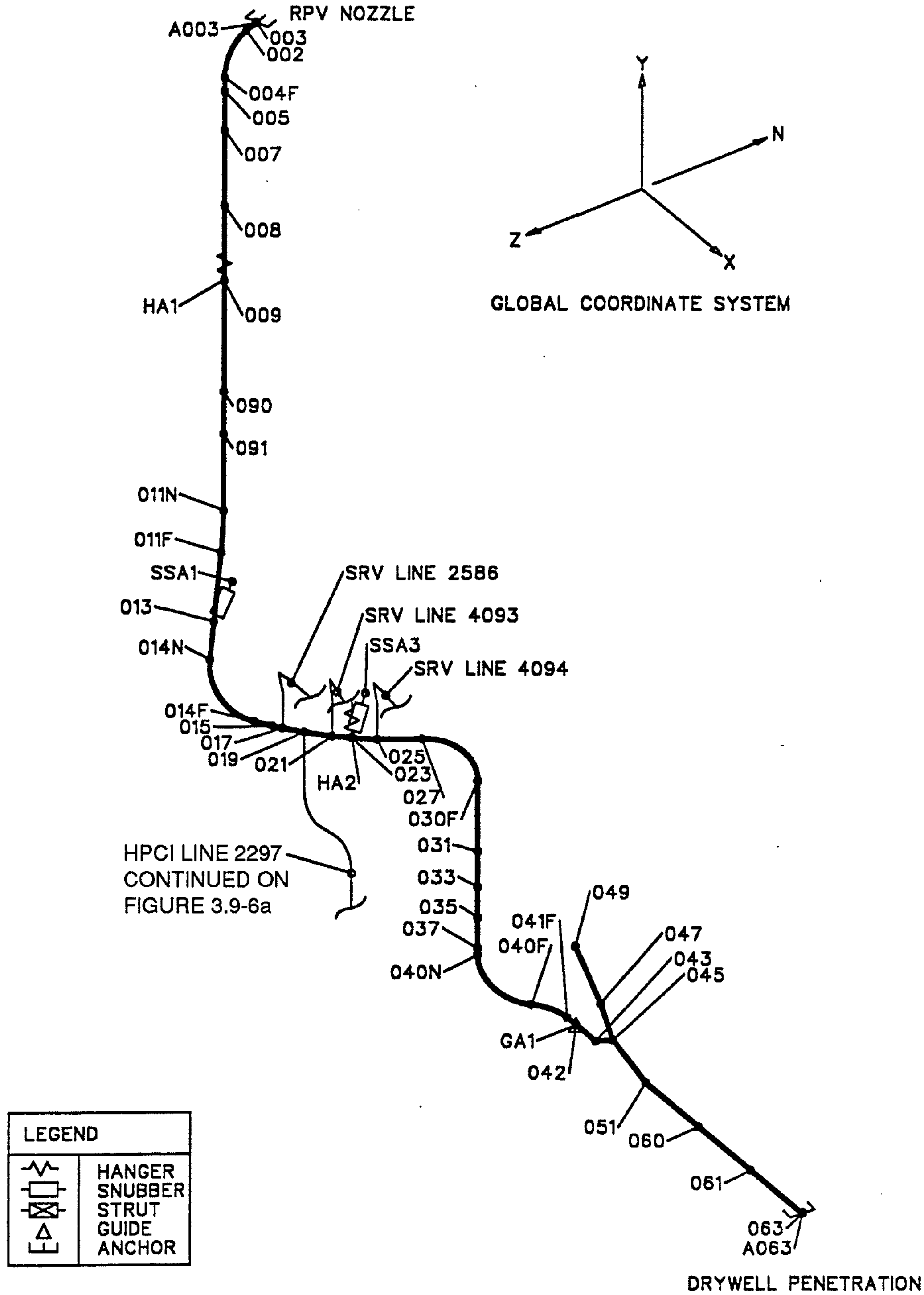

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FIGURE 3.9-6

MAIN STEAM LINE A
STRESS ISOMETRIC NODE DIAGRAM

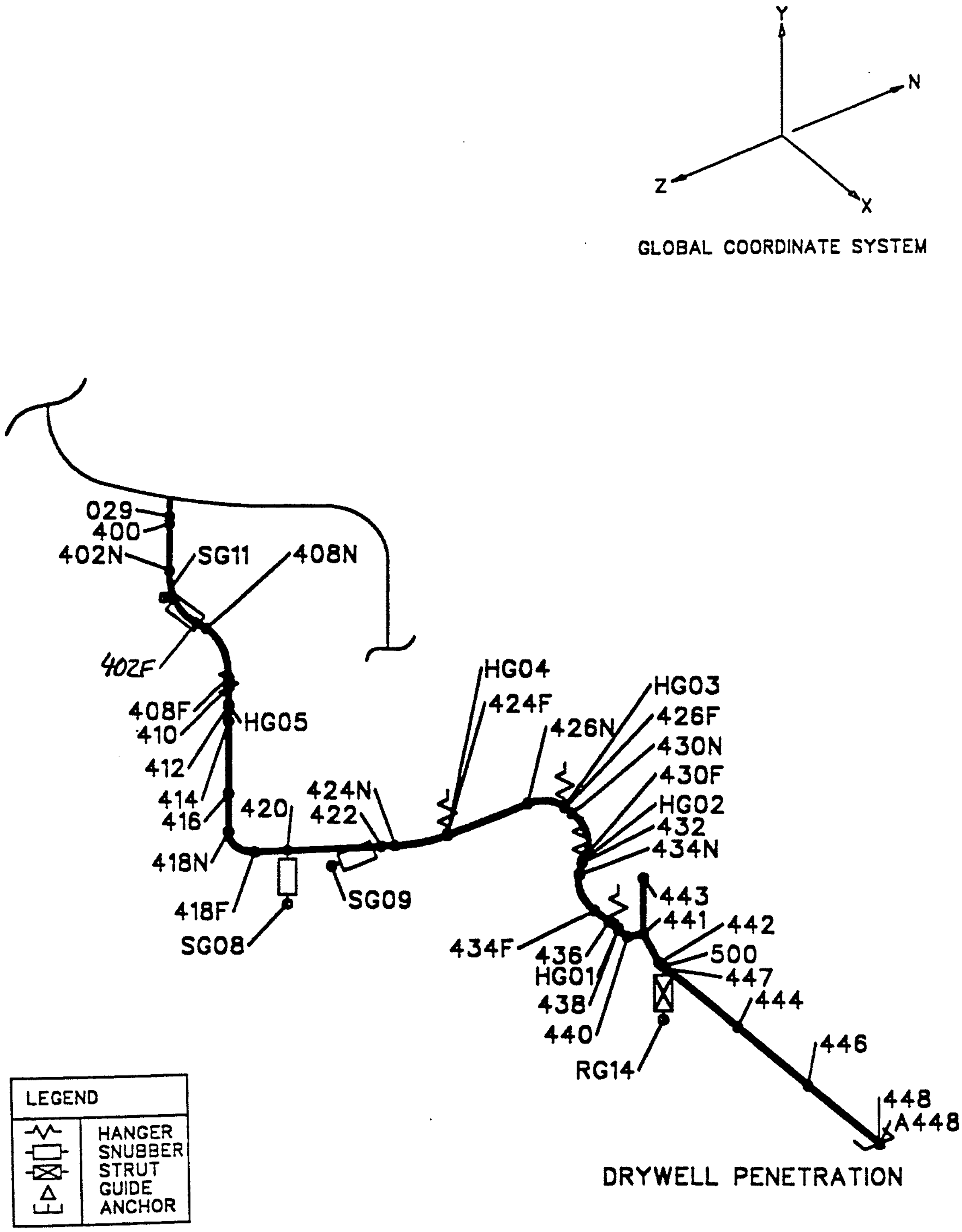

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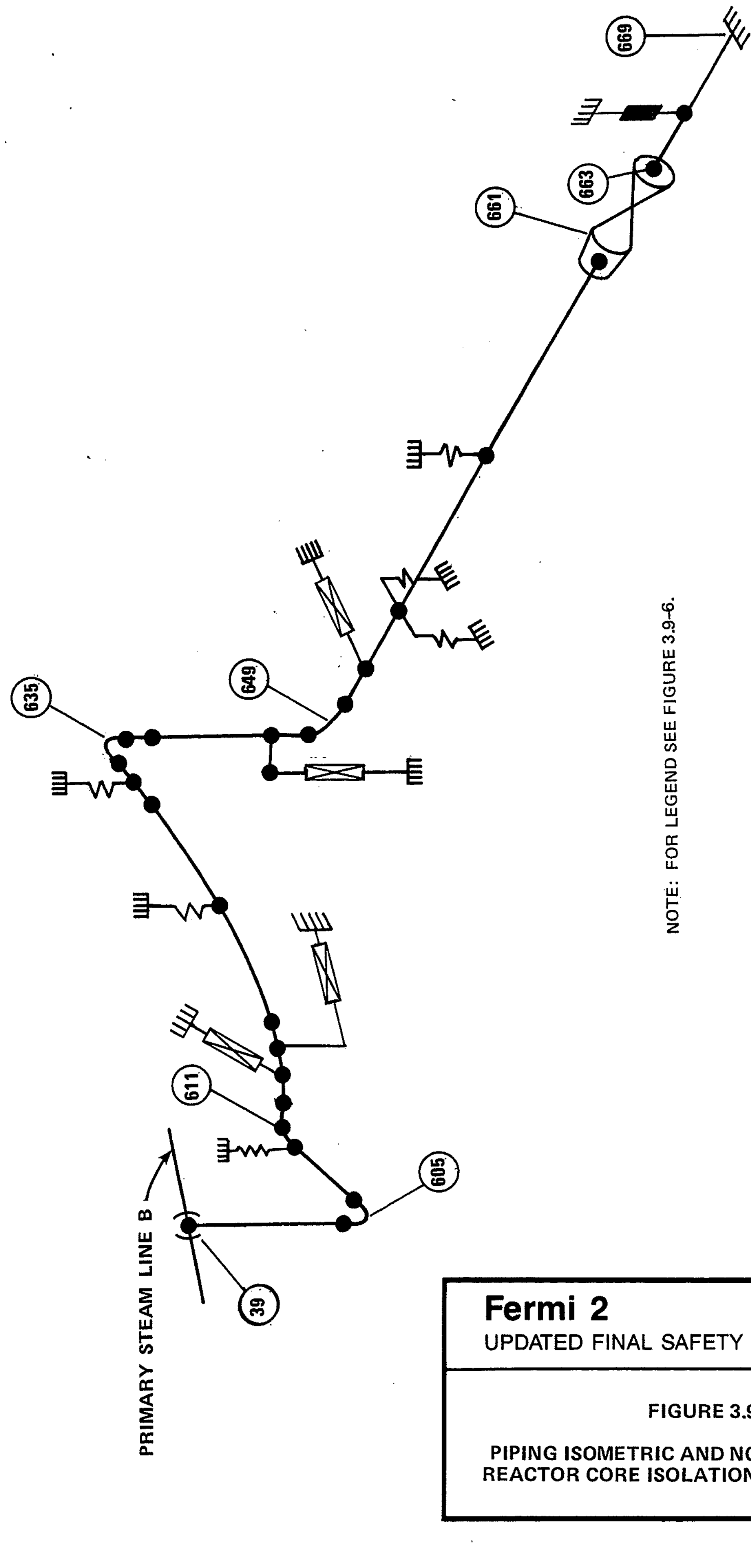
FIGURE 3.9-6a

HPCI LINE 2297
STRESS ISOMETRIC NODE DIAGRAM



NOTE: FOR LEGEND SEE FIGURE 3.9-6.

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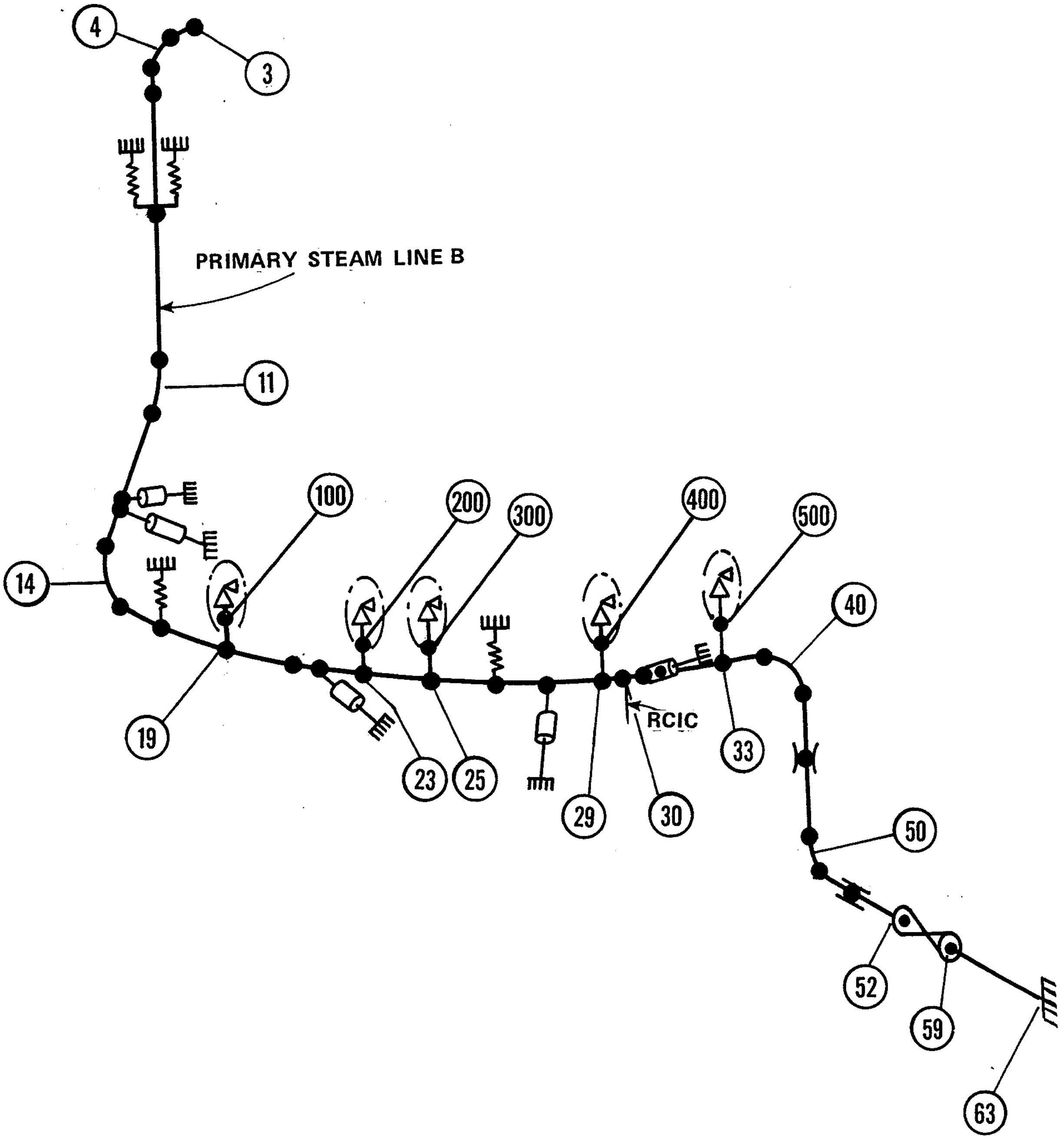
FIGURE 3.9-7

PIPING ISOMETRIC AND NODE DESIGNATION REACTOR CORE ISOLATION COOLING SYSTEM



NOTE: S/RVD LINES ARE INCLUDED IN THE ANALYSIS MODEL TO ACCOUNT FOR THEIR EFFECT ON THE MAIN STEAM LINE. FOR LEGEND SEE FIGURE 3.9-6.

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FIGURE 3.9-8

PIPING ISOMETRIC AND NODE DESIGNATION MAIN STEAM LINE "B"

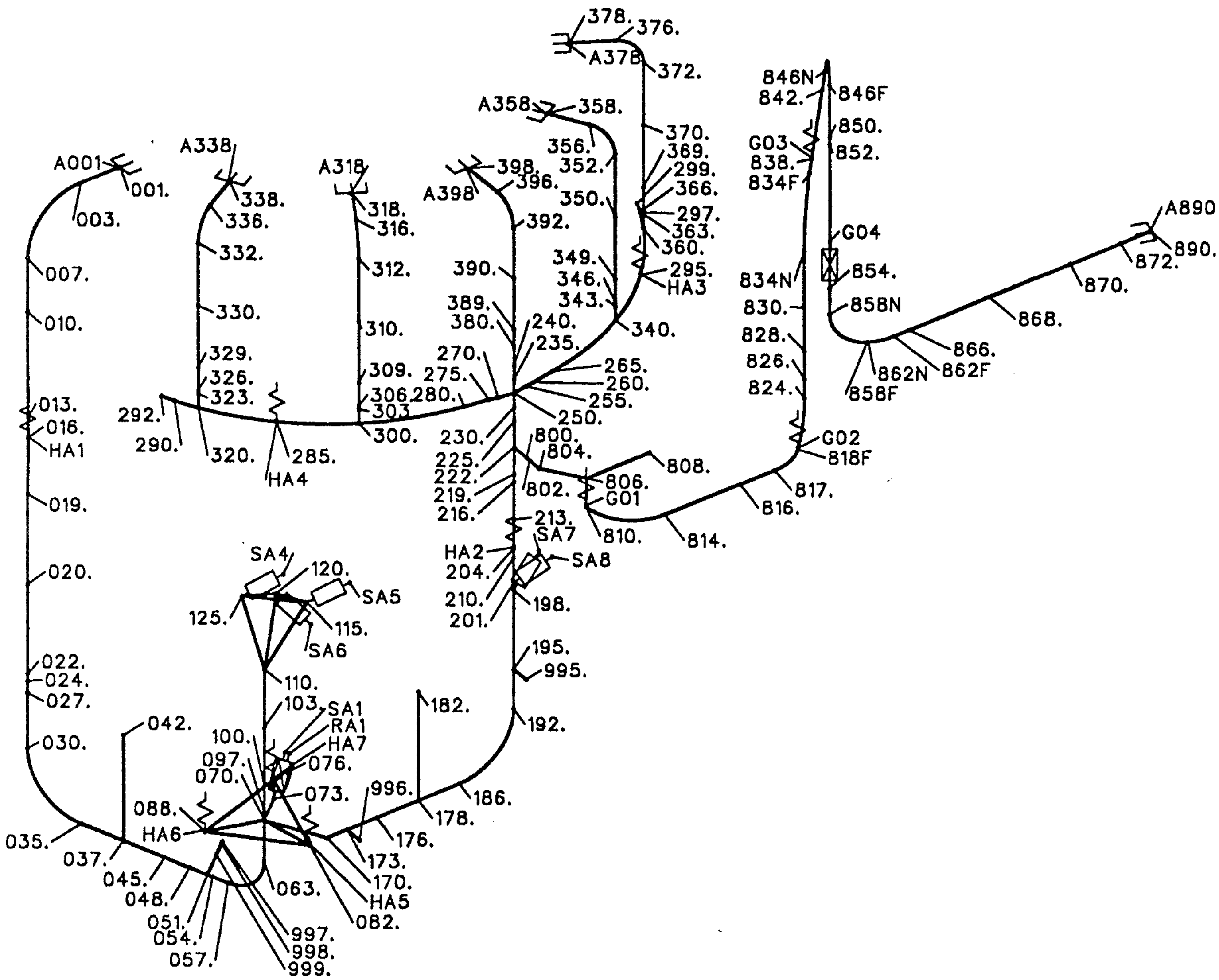

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FIGURE 3.9-9

PIPING ISOMETRIC AND NODE DESIGNATION
RECIRCULATION SYSTEM LOOP "A"

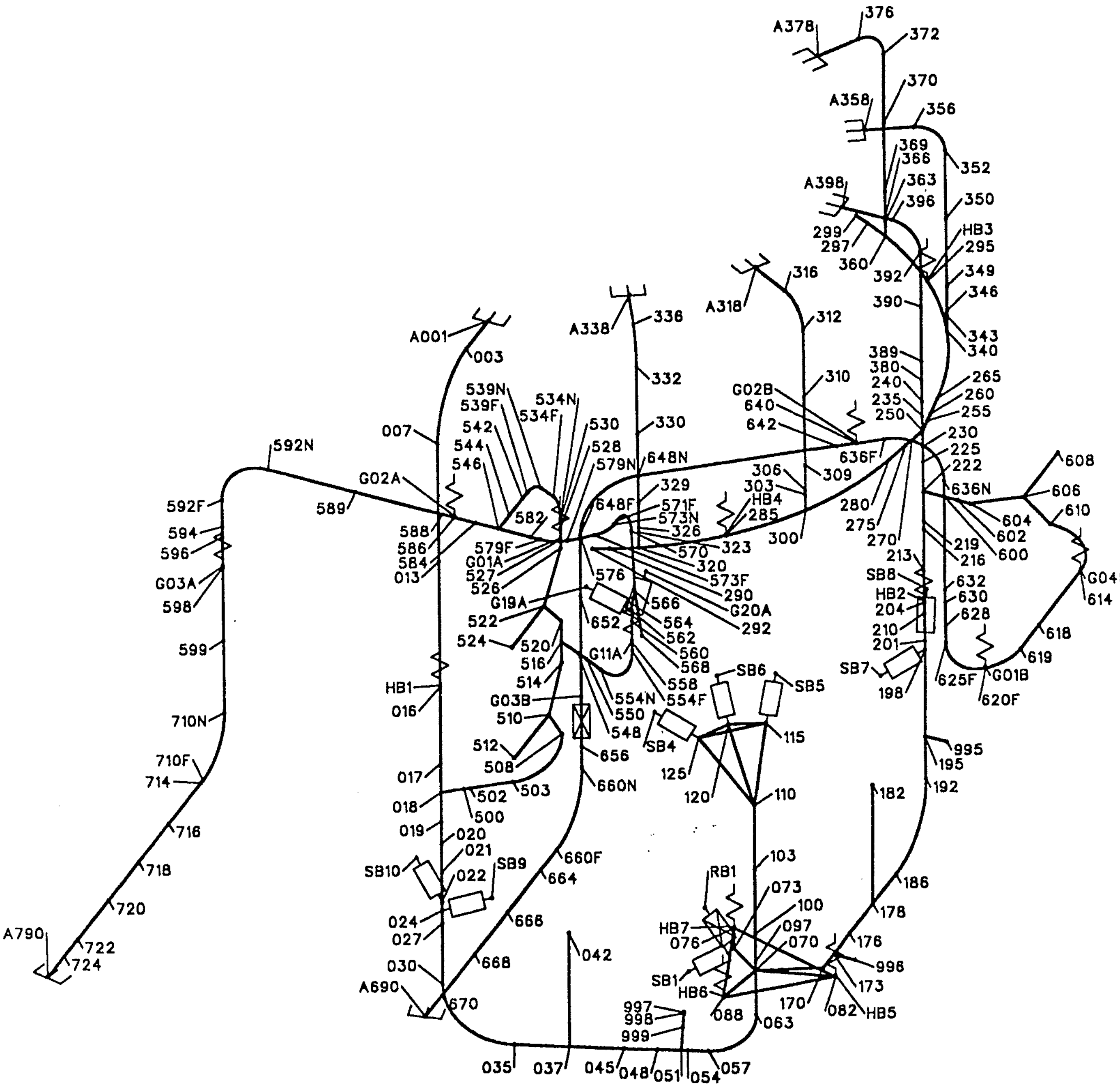

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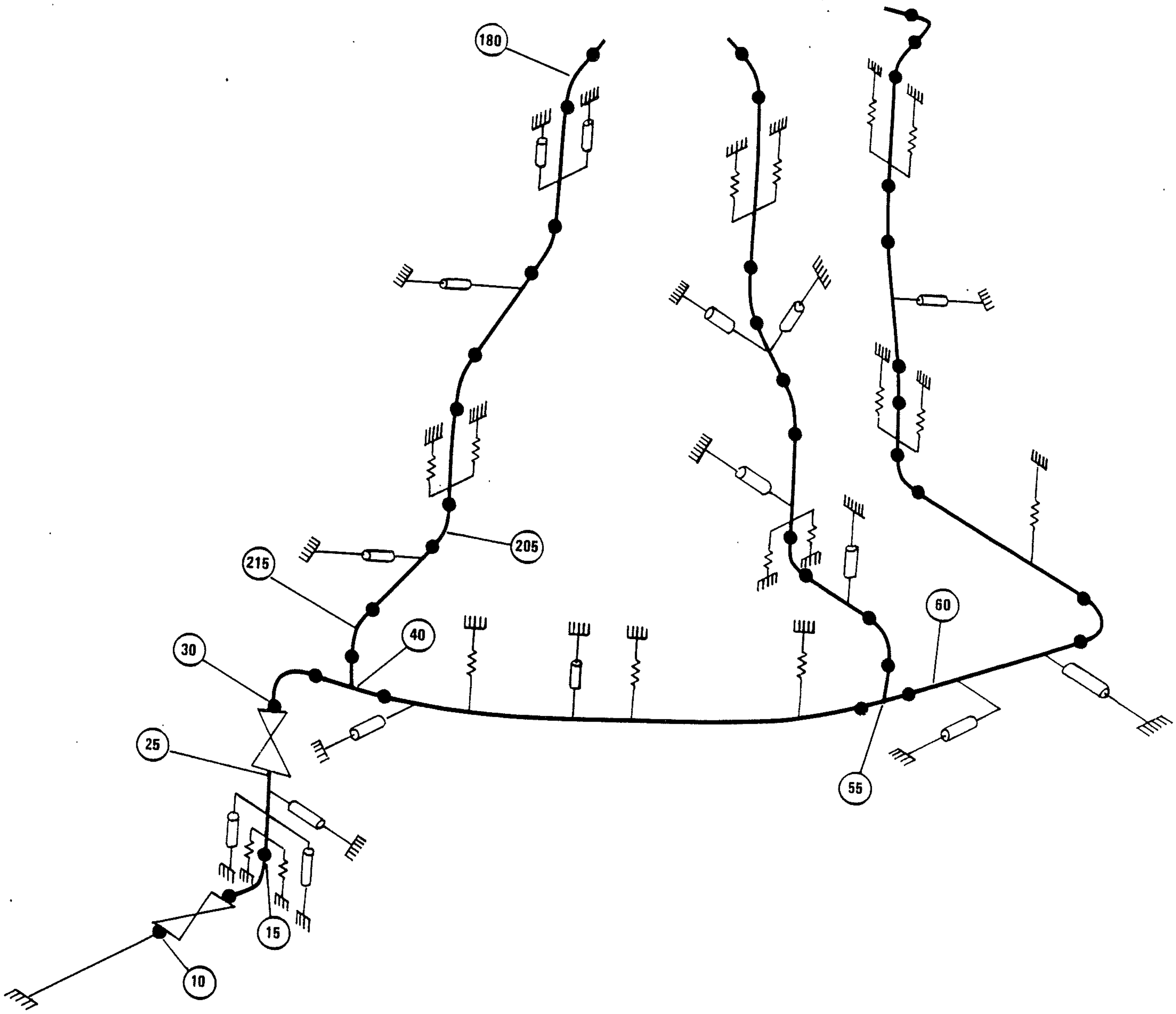
FIGURE 3.9-10

PIPING ISOMETRIC AND NODE DESIGNATION RECIRCULATION SYSTEM LOOP "B"

FIGURES 3.9-11 THROUGH 3.9-13

ARE INTENTIONALLY DELETED



NOTE: FOR LEGEND SEE FIGURE 3.9-6.

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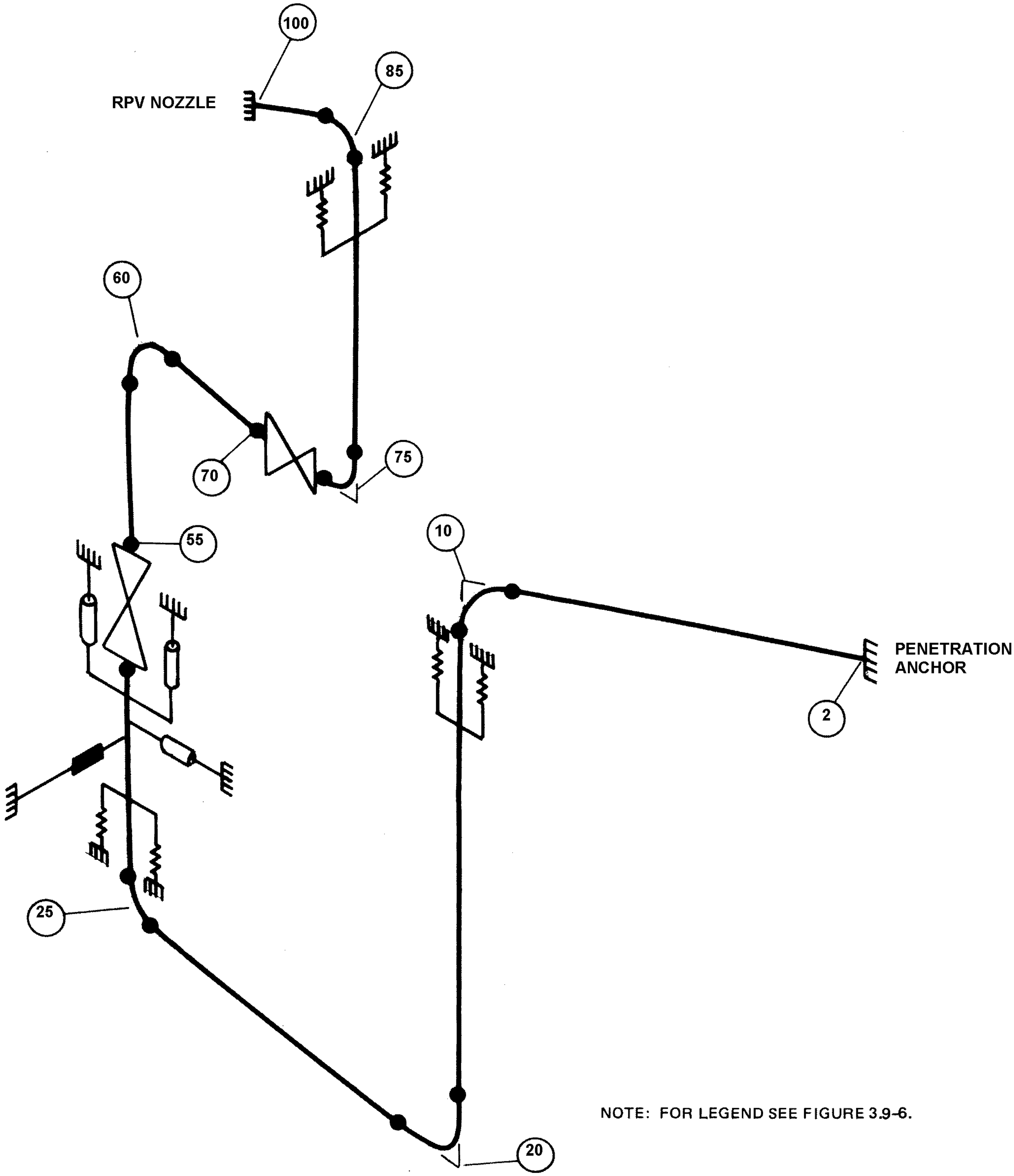
FIGURE 3.9-14

PIPING ISOMETRIC AND NODE DESIGNATION FEEDWATER SYSTEM FROM REACTOR PRESSURE VESSEL TO CONTAINMENT



NOTE: FOR LEGEND SEE FIGURE 3.9-6.

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FIGURE 3.9-15

PIPING ISOMETRIC AND NODE DESIGNATION CORE SPRAY SYSTEM INSIDE DRYWELL

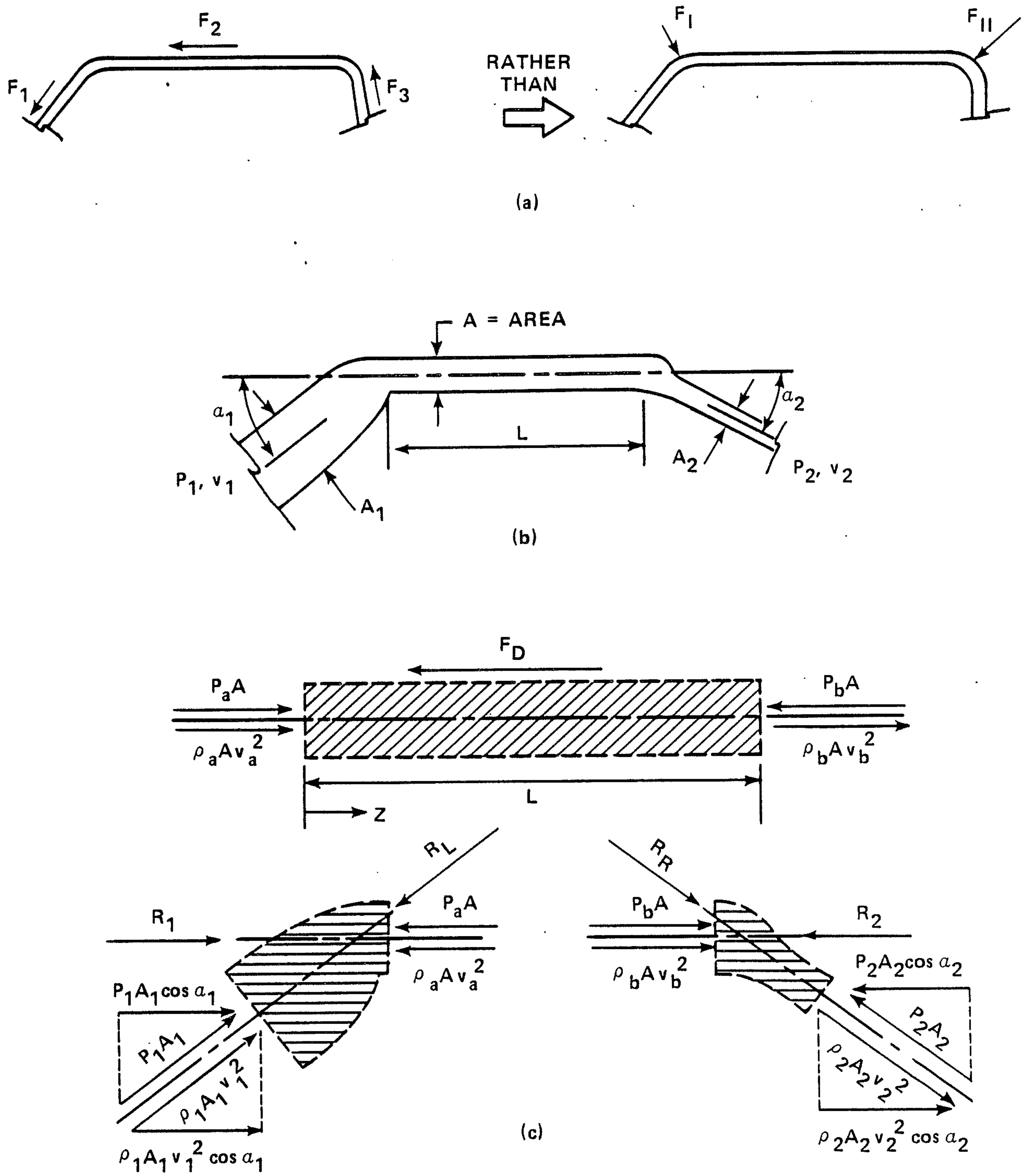

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FIGURE 3.9-16

RELIEF VALVE FORCING FUNCTIONS

Figure Intentionally Removed
Refer to Plant Drawing M-3084-2

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FIGURE 3.9-17

STRESS ANALYSIS DIAGRAM - EMERGENCY EQUIPMENT COOLING WATER SYSTEM PUMP SUCTION FROM HEAT EXCHANGER

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Refer to Plant Drawing M-3184-2

FIGURE 3.9-18

STRESS ANALYSIS DIAGRAM – RESIDUAL HEAT REMOVAL SERVICE WATER RETURN LINE FROM HEAT EXCHANGER

Figure Intentionally Removed
Refer to Plant Drawing M-3159-2

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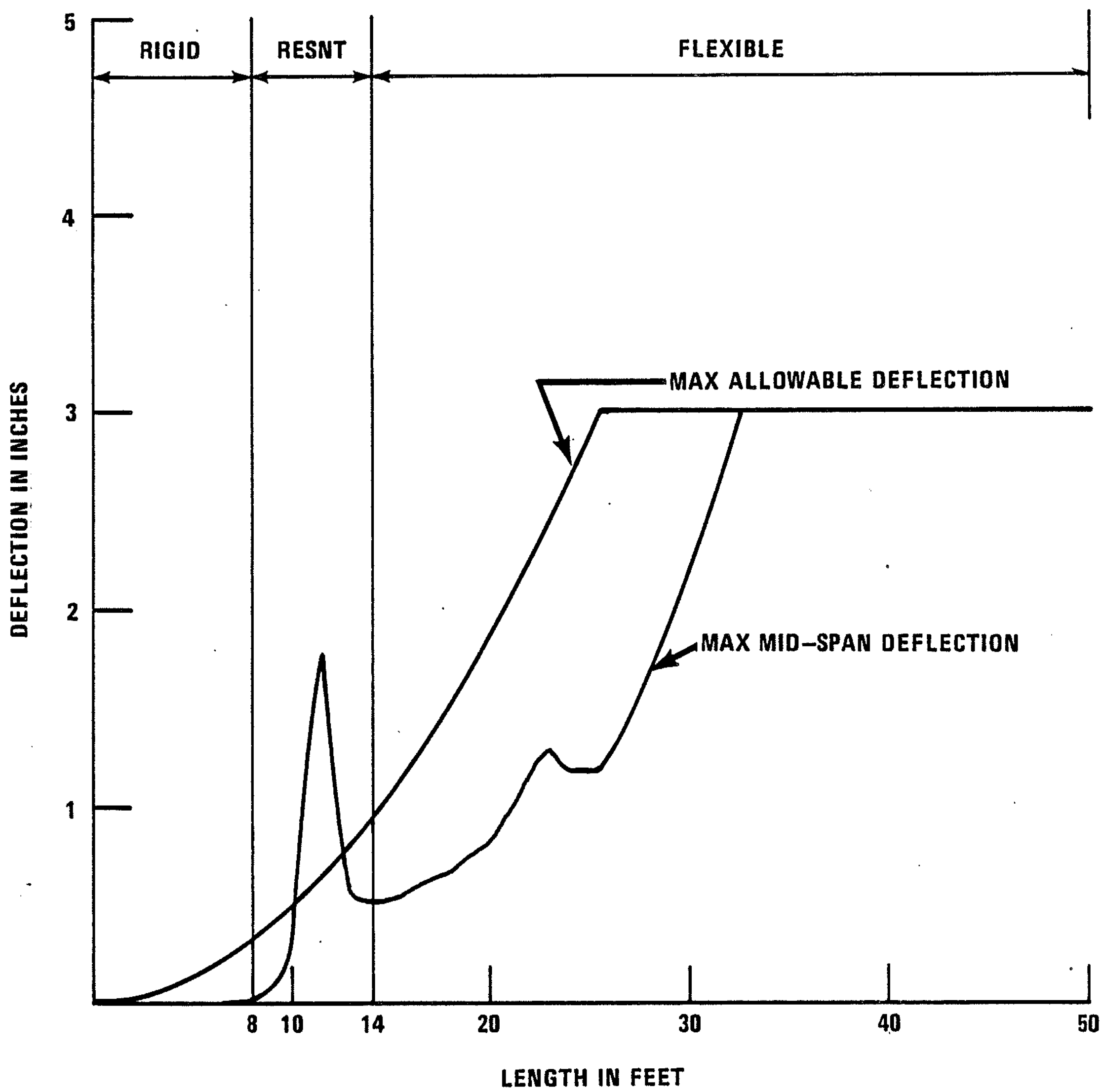
FIGURE 3.9-19

STRESS ANALYSIS DIAGRAM - RESIDUAL HEAT REMOVAL CONTAINMENT SPRAY RETURN TO DRYWELL



SARGENT & LUNDY DRAWING

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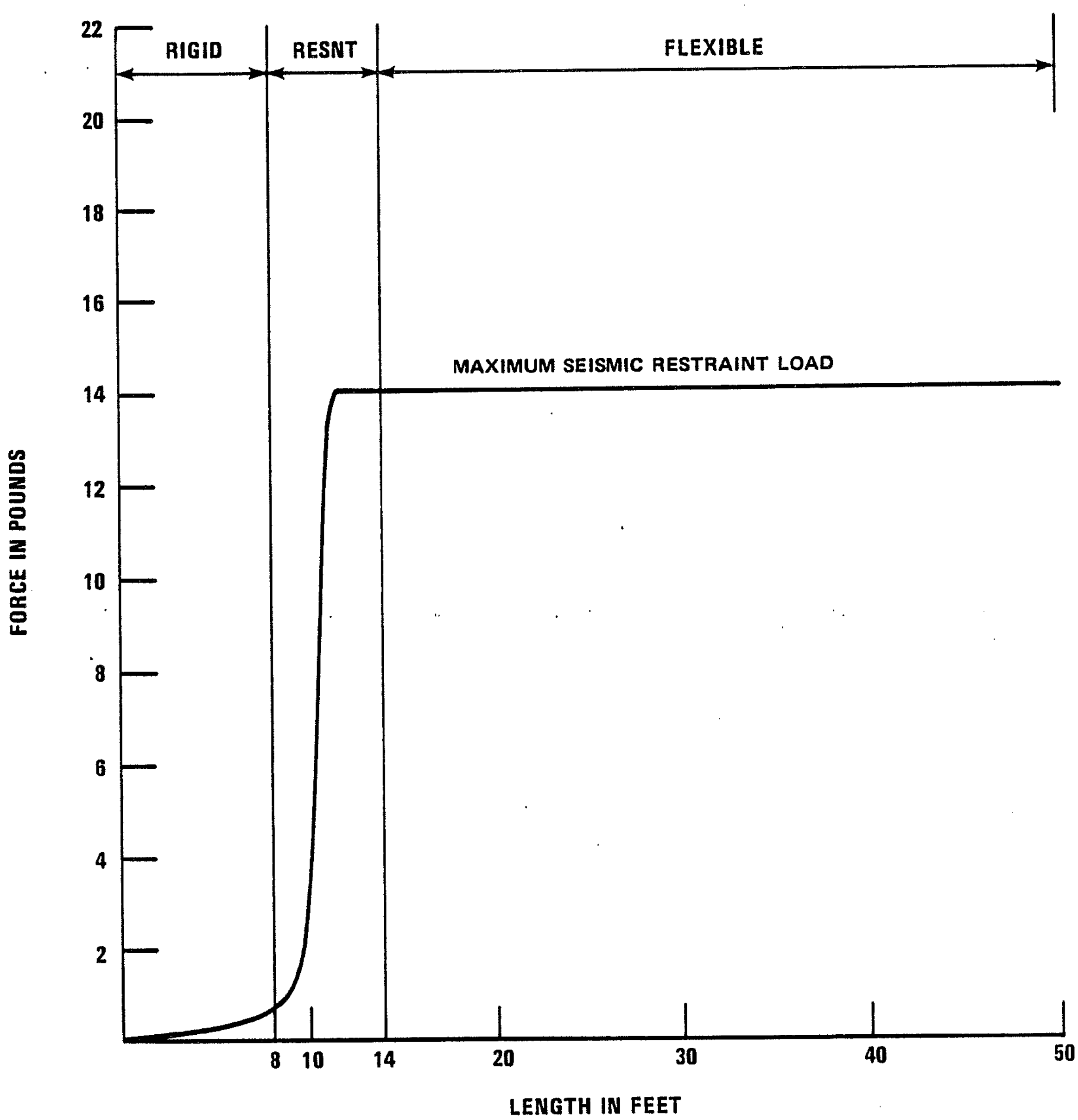
FIGURE 3.9-20

SEISMIC DESIGN CURVES – ½ IN. PIPE SEISMIC DEFLECTION VERSUS SPAN



Fermi 2

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FIGURE 3.9-21

SEISMIC DESIGN CURVES – ½ IN. PIPE SEISMIC RESTRAINT VERSUS SPAN

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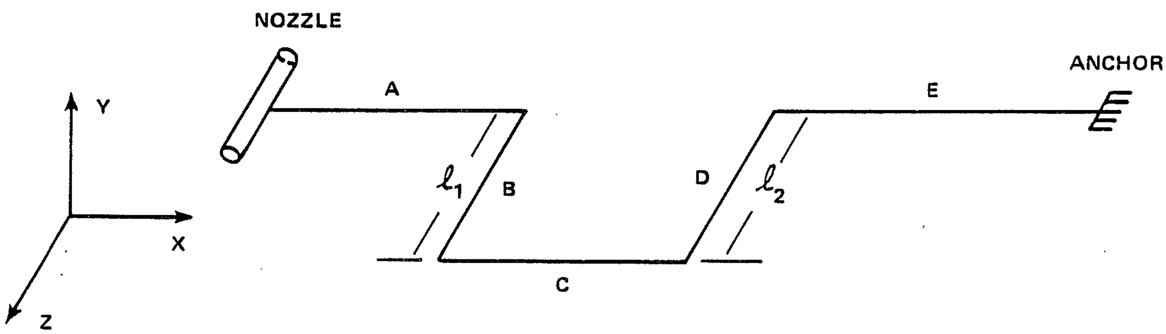


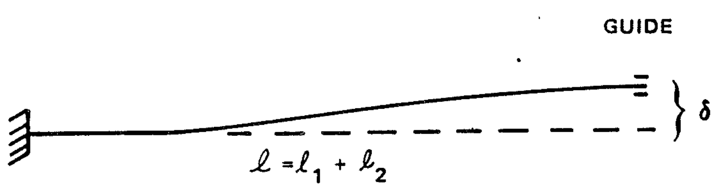
SARGENT & LUNDY DRAWING

Fermi 2

UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.9-22

FLEXIBILITY ANALYSIS OF PIPING SYSTEM SCHEMATIC



SARGENT & LUNDY DRAWING

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FIGURE 3.9-23

NOZZLE FLEXIBILITY MODEL SCHEMATIC

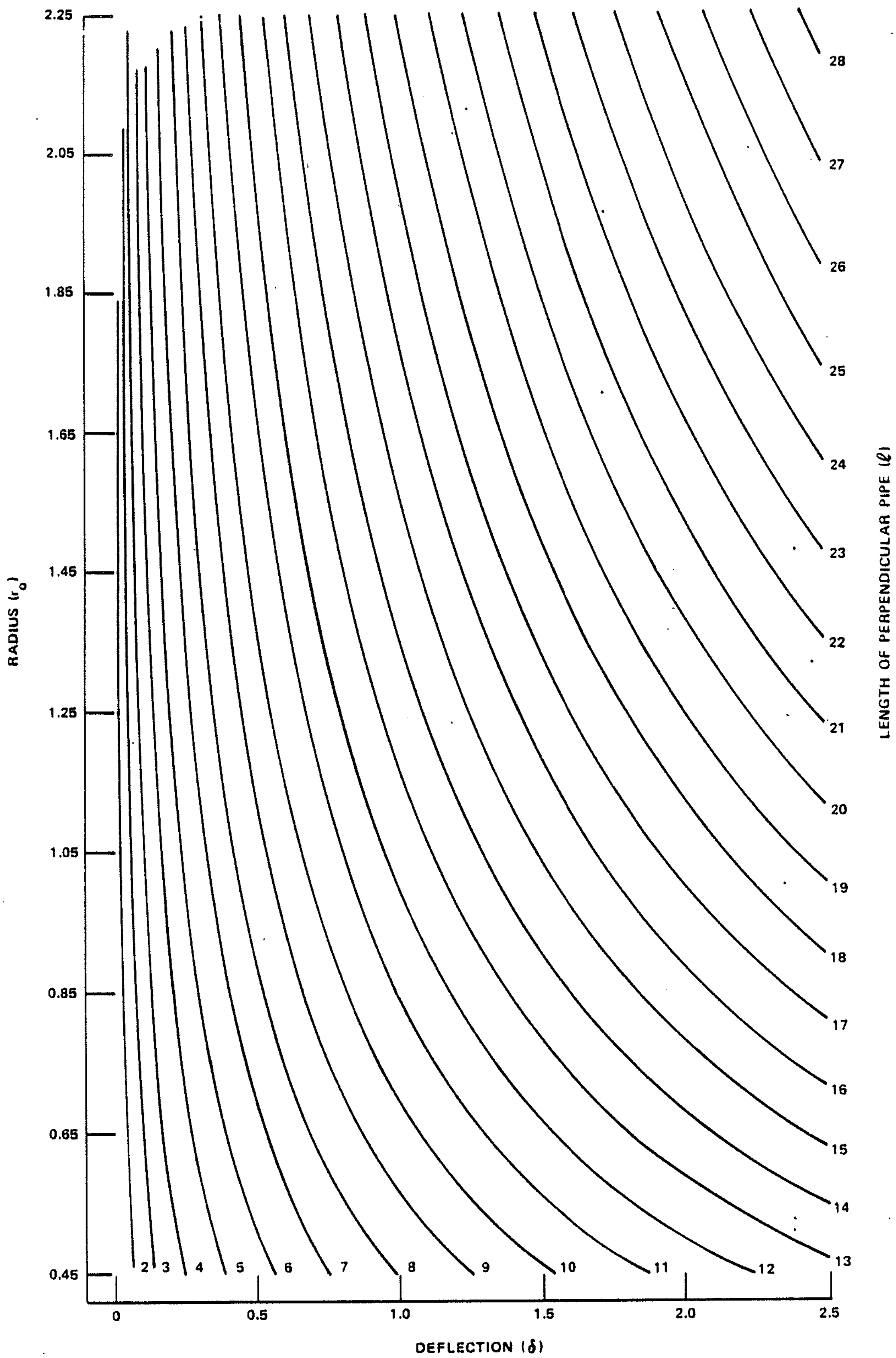

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FIGURE 3.9-24

LENGTH OF PERPENDICULAR PIPE REQUIRED TO ACCOMODATE THERMAL DEFLECTION

3.10 SEISMIC DESIGN OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.1 Seismic Design Criteria

3.10.1.1 Introduction

All Category I instrumentation and electrical equipment is designed to resist and withstand the effects of the two postulated Fermi 2 earthquakes, the safe-shutdown earthquake (SSE) and the operating-basis earthquake (OBE).

Category I instrumentation and electrical equipment is designed to withstand the effects of the SSE as defined in Section 3.7 without functional impairment. A list of major Category I instrumentation and electrical equipment is included in Table 3.2-1.

From the basic input ground motion data described in Sections 2.5 and 3.7, a series of response spectra at various floor elevations in both the vertical and horizontal directions was developed. After the dynamic analysis of the building was completed, the maximum seismic loadings derived from the appropriate spectra were included in the purchase specifications of Category I systems and equipment.

All vendors are required to qualify their equipment for both the SSE and the OBE using the response curves applicable to the particular building location of their equipment.

Suppliers of Category I equipment such as batteries, racks, local process-connected instrument panels, and control consoles are required to submit test data, operating experience, and/or calculations to substantiate that their components and systems would not suffer loss of function during and/or (as required) after seismic loadings as a result of the SSE. Before the equipment was accepted by Edison, proof of compliance with the accepted seismic qualifications procedures was provided to the Fermi 2 project for approval.

Since the construction permit application for Fermi 2 was docketed before October 27, 1972, the seismic qualification of Category I instrumentation, electrical equipment, and supports is required to meet the requirements of IEEE 344-1971 (Reference 1). The NRC staff conducted a review (Seismic Qualification Review Team (SQRT) audit to ensure that such components have an adequate margin to perform their intended design functions during the seismic event. During the review period, 1981 through 1984, Edison provided the SQRT with information regarding the seismic qualification of specific pieces of installed equipment and confirmed that all safety-related equipment identified to be installed at the time of fuel load was seismically qualified. Fermi 2 has a seismic qualification program to address design changes related to Category I equipment.

Category I components purchased after the issuance of IEEE 344-1975 (Reference 2) are specified to be qualified to the requirements of that standard.

As stated in Subsection 3.7.1.2, all structures, systems, and components required for cold shutdown were reaffirmed to be acceptable with respect to the Fermi 2 site-specific earthquake excitation.

Nuclear steam supply system (NSSS) items were qualified to acceleration levels that were selected to envelop potential facility excitation predictions at the time of initiation of the

Fermi 2 design. Although these acceleration levels do not envelop all facility accelerations at the time of fuel load, detailed evaluations of NSSS items have revealed significant excess aseismic design capabilities rendering the equipment quite satisfactory for Fermi 2 use. In addition, all NSSS item site-specific earthquake acceptability affirmations have been documented as satisfactory.

3.10.1.2 Reactor Protection System and Engineered Safety Feature Circuits

The Category I instrumentation and electrical equipment associated with the reactor protection system (RPS), engineered safety feature (ESF) circuits, nuclear safety systems circuits, and the emergency power system include instruments, sensory equipment, control equipment, power supplies, diesel generators, drywell penetrations, batteries, underground ducts, motor control centers (MCCs), switchgear, cable trays and conduits, consoles, local instrumentation panels, and anchorage systems. These systems maintain functional operability during and following any pre- or postaccident SSE excitation at the equipment location.

The design of the building and the electrical equipment support structures has used the applicable floor response spectra shown in Section 3.7 in determining the response spectra at the equipment locations in the RPS, nuclear safety features, and in the ESF circuits.

The seismic criterion used in the design and subsequent qualification of all Class 1E instrumentation and electrical equipment supplied by GE was as follows:

> The Class IE equipment shall be capable of performing all safety related functions during normal plant operation, during anticipated transients, during design-basis accidents, and during postaccident operation while being subjected to, and after the cessation of the accelerations resulting from, the SSE at the point of attachment of the equipment to the building or supporting structure.

The specific criteria for each of the many Class 1E systems are covered in Chapter 7. The criteria for each of the devices used in the many Class 1E systems depend on the use in a given system; for example, a relay in one system may have as its safety function to deenergize and open its contacts within a certain time, while in another system it must energize and close its contacts. Since GE supplies many devices for many applications, the approach taken was to test the device in all modes that might be used. In this way, the capability of protective action initiation and the proper operation of safety-feature circuits are ensured.

Non-GE Category I equipment will also maintain functional operability during and/or (as required) after the SSE, as dictated by the response spectra for the equipment location. Proof of this is shown by the vendor seismic qualification and confirmatory review of each item.

If a seismic disturbance occurs after a major accident, the emergency core cooling will not be interrupted. The control circuits, switchgear, and diesel-generator design are such that the system will not be shut down once it is initiated, except by operator-initiated signal or by some other protective device signal.

3.10.1.3 Extent of Compliance With IEEE 344-l97l

3.10.1.3.1 General Electric-Supplied Equipment

The compliance of GE-supplied Class 1E equipment with IEEE 344-1971 (Reference 1) can be summarized as follows:

a. Scope - Compliance not applicable

b. Definition - Compliance not applicable

c. Procedures - General Electric-supplied Class 1E equipment meets the requirement that the seismic qualification should demonstrate the capability to perform the required function during and after the SSE. In addition, those items necessary for shutdown after loss of offsite power were reaffirmed to be acceptable for the site-specific earthquake situation. Both analysis and testing were used, but most equipment was tested. Analysis was used to determine the adequacy of mechanical strength (mounting bolts, etc.) after operating capability was confirmed by testing

   1. Analysis - General Electric-supplied Class 1E equipment with primarily 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-l97l were used in such cases

   2. Testing - General Electric-supplied Class 1E equipment having primarily active electrical safety functions was tested in compliance with Section 3.2 of IEEE 344-1971.

d. Documentation - The documentation is that which verifies that the seismic qualification of GE-supplied Class 1E equipment is in accordance with the requirements of Section 4 of IEEE 344-1971.

3.10.1.3.2 Non-General Electric-Supplied Equipment

The qualification and documentation procedures used for non-GE-supplied Category I equipment and systems are specified in Subsection 3.10.1.3.3, which encompasses and amplifies the requirements of IEEE 344-1971.

3.10.1.3.3 Criteria for Seismic Qualification of Category I Equipment (Non-General Electric-Supplied)

The criteria for qualification of Category I instrumentation and electrical equipment (non-GE-supplied) are established in this subsection. The IEEE Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations is a basic part of these criteria except as specified and amended below. Paragraph numbers in parentheses conform to the paragraph numbers in IEEE 344-1971. Equipment purchased after 1971 conforms to this or later versions of this standard.

3.10.1.3.3.1 Scope (1)

All Category I instrumentation and electrical systems and equipment (assemblies and devices) supplied for this plant must withstand the postulated seismic occurrence specified herein. The equipment vendor is responsible for ensuring that the equipment and systems operate safely under the postulated seismic conditions, and he must verify that the equipment will meet the stated functional requirements for continued operation without any malfunction or loss of function during and/or (as required) after the specified events.

3.10.1.3.3.2 Category I Equipment (2.1)

Class I, as defined in Paragraph 2.1 of IEEE 344-1971, is synonymous with Category I equipment.

3.10.1.3.3.3 Safe-Shutdown Earthquake (2.2)

For Fermi 2, the design-basis earthquake, as defined in Paragraph 2.2 of IEEE 344-1971, is synonymous with SSE.

3.10.1.3.3.4 Malfunction (2.8 - Additional Definition)

Equipment malfunction or functional impairment is the failure of equipment to operate in the same manner in which it would have operated in the absence of a seismic disturbance. For protective systems, malfunction is the loss of capability to initiate a protective action, or the initiation of a spurious protective action.

3.10.1.3.3.5 Procedures (3)

When the malfunction of Category I equipment is considered, testing is the method recommended to verify the functional requirements. Table 3.10-1 summarizes seismic qualification testing of typical non-GE-supplied equipment.

3.10.1.3.3.6 Analysis (3.1)

The number of masses should be sufficient to define the dynamic behavior of the equipment. The mathematical model should be shown even for a single degree of freedom.

3.10.1.3.3.7 (Additional 3.1.6)

The analysis shall include the combined effect of gravity loads, and other loads included in the specification, combined with the appropriate seismic loads. The seismic stresses may be computed independently for the vertical and horizontal directions. The horizontal excitation may be based on an envelope encompassing the maximum acceleration levels of the N-S and E-W components of the horizontal spectra given in the Edison electrical equipment specifications. The vertical and horizontal responses are considered to act simultaneously in combining the stresses.

The normal operating primary stresses, combined with the SSE stresses, are not to exceed the minimum guaranteed American Society for Testing and Materials (ASTM) yield strength at the appropriate temperature. Combinations of primary, local, and self-limiting secondary stresses may exceed yield stress levels to the extent permitted by the appropriate codes as long as malfunction is prevented. Where biaxial or triaxial loads are involved, the principal stresses shall be calculated and kept within the allowable material stress levels.

3.10.1.3.3.8 Testing (3.2.2.4.2)

The test shall be conducted over a minimum frequency range of 1 to 33 Hz.

3.10.1.3.3.9 Test Data (4.3)

If proof of performance is obtained by testing, the test data shall contain the following information:

a. Equipment identification

b. Equipment specification

c. Test facility

   1. Location

   2. Test equipment.

d. Test method

e. Test data

f. Data analysis and evaluation (including the floor acceleration versus frequency spectra for the surface upon which the equipment was mounted when tested)

g. Summary and conclusions

h. Certifying signature of a registered professional engineer and date of signature, if the test is performed on ASME Code items

i. Calibration history of test equipment.

3.10.1.3.3.10 Certification of Compliance (4.4)

All test data submitted by the vendor to satisfy the requirements of this specification are witnessed and reviewed to determine that the data adequately demonstrate that the equipment satisfies the intent of these specifications.

3.10.2 Seismic Analyses, Testing Procedures, and Restraint Measures

3.10.2.1 Amplification of Floor Inputs by Supports

Response spectra for floors and walls where Category I equipment is located were supplied to the vendors. If the vendor chose to test or analyze a certain device or component not directly supported on the floor for which the spectra are applicable, account was taken of possible amplification through the support structure.

3.10.2.2 Cable Tray and Cable Support System

The cable trays and cable tray support system were verified to withstand forces caused by dead-load, live-load, and seismic conditions.

The following combinations of dead load, live load, and earthquake load were investigated and checked to determine the most severe condition:

a. Dead load of various components with allowable stresses according to American Iron and Steel Institute (AISI) Specifications

   The dead load on cable trays consists of cables plus tray. In the case of hangers, it includes the dead weight of hangers also. Originally, the cable tray loading within the relay room, cable spreading room, and directly below the relay room floor was 50 lb/ft^2. All other cable trays were designed to a dead load of 40 lb/ft^2. An on-going program was later established to monitor the actual weight of cables in the trays and to account for fire wrap, conduit and air drop loads. Cable tray design load is adjusted to reflect these actual loads

b. Dead load plus a concentrated live load of 200 lb at the mid-span to AISC allowable stresses for reactor, aux building and RHR complex. The concentrated live load is 250 lb for the drywell cable trays

c. Dead load plus seismic load.

The cable trays and the support system were modeled as a multidegree-of-freedom system with the mass of the cables plus tray lumped at the levels at which they are supported. Figure 3.10-1 shows typical models for a three-layer hanger with one, two, and three diagonal members for horizontal excitation.

For vertical excitation, the fundamental period of vibration was computed by using a simplified model of continuous beam with hinged ends. This approximation was found to be consistent with the numerous models studied for this purpose.

The response spectra obtained from the analysis of the building were used in determining the response of the cable tray support.

The horizontal and vertical seismic excitations were assumed to be acting simultaneously along the principal axis of the cable tray system. The seismic response was computed by taking the sum of the individual responses.

It was observed that contribution due to nonfundamental modes was negligible, and hence the effect of closely spaced modes was negligible also.

The design was based on the l968 edition of the "Specifications For the Design of Cold-Formed Steel Structural Members."

For the trays in the drywell, a concentrated live load of 250 lb was specified. As stated in Subsections 3.7.3.17.2 and 8.3.1.4.3, in the design specification for cable trays, dead-weight loading did not include the weight of fire wrapping material or any other attachments, such as top hat cover, which were subsequently added. Accordingly, hanger modifications were made where necessary, and the structural adequacy of the cable trays was verified.

3.10.2.3 Battery Racks, Battery Chargers, Instrument Racks, and Control Consoles

Response spectra for floors and walls where Category I equipment is located have been supplied to the vendor. The vendors were required to submit test data, operating experience, and/or calculations to verify that battery racks, battery chargers, instrument racks, and control consoles would not suffer any loss of function during or after the SSE. For equipment in the GE scope of supply, procedures are in accordance with GE Topical Report NEDO-10678 (Reference 3). For non-GE-supplied equipment, testing procedures are in accordance with IEEE 344-1971, as modified in Subsection 3.10.1.3.3, with the exception of the equipment purchased prior to the issuance of IEEE 344-1971. Equipment purchased prior to the issuance of IEEE 344-1971 is designed to withstand the SSE postulated by the response spectra for its location during and/or (as required) after such an event.

3.10.3 Seismic Analysis and Testing Procedures for General Electric-Supplied Equipment

3.10.3.1 Seismic Analysis

Very few of the GE-supplied Class 1E devices were completely qualified by analysis alone. A sample of such an analysis is shown in Appendix B of NEDO-10678 (Reference 3). Besides being used for passive mechanical devices, analysis was used in combination with testing for larger assemblies containing Class 1E devices. For instance, a test might have been run to determine whether there were natural frequencies in the equipment within the critical seismic frequency range (see Paragraph 3.2.2.3.1 of IEEE 344-1971). If the equipment was determined to be free of natural frequencies, then it was assumed to be rigid, and a static analysis was performed as shown in Appendix C of NEDO-10678 (see Paragraph 3.2.3.4 of IEEE Standard 344-1971. If it had natural frequencies in the critical frequency range, then calculations of transmissibility and responses to varying input accelerations were made to determine whether Class 1E devices mounted in the assembly would operate without malfunctioning.

3.10.3.2 Testing Procedures

Since the Class 1E equipment supplied by GE was and is used in many systems on many different plants under widely varying seismic requirements, the seismic qualification tests were performed using an expected worst-case envelope of 1.5g horizontal and 0.5g vertical at all frequencies from 5 to 33 Hz. (The actual qualification range was 0.25 to 33 Hz, but, since test facility capability usually limited the lower frequency test to 5 Hz, a combination of test and analysis was used to ensure that there were no untested resonances. A sample analysis is shown in Appendix B of NEDO-10678.) In general, Class 1E equipment was tested by the procedures described below.

3.10.3.2.1 Devices

The test procedure for devices required that the device be mounted on the table of the vibration machine in a manner similar to that in which it was to be installed. The device was tested in the operating states in which it was to be used while performing its Class 1E functions, and these states were monitored before, during, and after the test to ensure proper function and absence of spurious function. In the case of a relay, both energized and deenergized states and normally open and normally closed contact configurations were tested if the relay is used in those configurations in its Class 1E functions.

The seismic excitation was a single-frequency "continuous" test in which the applied vibration was a sinusoidal table motion at a fixed peak acceleration and a discrete frequency at any given time. Each frequency and acceleration combination was maintained for about 30 sec, except when a resonance search was made (see IEEE Paragraph 3.2.2.4.1 of Standard 344-1971). The vibratory excitation was applied in three orthogonal axes individually, with the axes chosen as those coincident with the most probable mounting configuration.

The first step was to search for resonances in each device. This was done since resonances cause amplification of the input vibration and are the most likely cause of malfunction. The resonance search was usually run at low acceleration levels (0.2g) to avoid destroying the test sample in case a severe resonance was encountered. The resonance search was run at frequencies from 5 to 33 Hz in accordance with IEEE 344-1971 in no less than 7 minutes; if the device was large enough, the vibrations were monitored by accelerometers placed at critical locations from which resonances were determined by comparing the acceleration level with that at the table of the vibration machine. Usually, the devices were either too small for an accelerometer, had their critical parts in an inaccessible location, or had critical parts that would be adversely affected by the mounting of an accelerometer. In these cases, the resonances were detected by visual (strobe light) or audible observation, or performance.

After the frequency scan and resonance determination, the devices were tested to determine their malfunction limit. This test was a necessary adjunct to the assembly test. The malfunction limit test was run at each resonant frequency as determined by the frequency scan. In this test, the acceleration level was gradually increased until either the device malfunctioned or the limit of the vibration machine was reached. If no resonances were detected (as was usually the case), the device was considered to be rigid (all parts move in unison) and the malfunction limit was therefore independent of frequency. To achieve maximum acceleration from the vibration machine, rigid devices were malfunction tested at the upper test frequency (33 Hz) since this procedure allowed the maximum acceleration to be obtained from deflection-limited machines.

Typical results of tests on the devices used in Class 1E applications are given in Table 3.2 of NEDO-10678 and include the malfunction limit and resonant frequencies for each device tested.

The above procedures were required of purchased devices as well as those made by GE. Vendor test results were reviewed, and if the results were unacceptable, the tests were repeated either by GE or the vendor. If the vendor tests were adequate, the device was considered to be qualified to the limits of the test.

3.10.3.2.2 Assemblies

Assemblies (e.g., control panels) containing devices whose seismic malfunction limits had been established were tested by mounting the assembly on the table of a vibration machine, in the manner it was to be mounted when in use, and vibration-testing it by running a low-level resonance search. Like the devices, the assemblies were tested in the three orthogonal axes. The resonance search was run in the same manner as described for devices. If resonances were present, the transmissibility between the input and the location of each Class 1E device was determined by measuring the accelerations at each device location and calculating the magnification between it and the input. Once known, the transmissibilities could be used analytically to determine the response at any Class 1E device location for any given input. (It was assumed that the transmissibilities were linear functions of acceleration even though they actually decrease as acceleration is increased. This assumption is therefore a conservative one.) If the input accelerations to the device were determined to be below its malfunction limit, the assembly was assumed to be qualified. If no resonances existed, the assembly was considered to be a rigid body with a transmissibility equal to 1, so that a device mounted on it would be limited directly by the assembly input acceleration.

Since control panels and racks constitute the majority of Class 1E electrical assemblies supplied by GE, seismic qualification testing of these will be discussed in more detail. There are four generic types, as shown in Table 3.10-2. One or more of each type was tested by the procedures described above.

Figures 3-1 through 3-4 of NEDO-10678 illustrate the panel types referenced in Table 3.10-2 and show typical accelerometer locations. Table 3.10-3 lists typical seismically tested panels supplied by GE.

The full-acceleration level tests described above disclosed that most of the panel types had more than adequate mechanical strength and that a given panel design acceptability was simply a function of its amplification factor and the malfunction levels of the devices mounted on it. Subsequent panels were, therefore, tested at lower acceleration levels and the transmissibilities to the various devices measured as described above. By dividing the devices' malfunction levels by the panel transmissibility between the device and the panel input, the panel seismic qualification level could be determined. Several high-level tests have been run on selected generic panel designs to ensure the conservatism in using the transmissibility analysis described.

3.10.3.2.3 Purchased Equipment

The seismic qualification of equipment supplied to GE by others was required to follow the same procedures as used by GE. The qualification data were supplied to and reviewed by GE for conformance with the required procedures.

3.10 SEISMIC DESIGN OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

REFERENCES

1. IEEE Standard 344-1971, "IEEE Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations."
2. IEEE Standard 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."
3. GE Topical Report, Seismic Qualification of Class I Electric Equipment, NEDO-10678, General Electric Company, November 1972.

TABLE 3.10-1 SEISMIC QUALIFICATION SUMMARY: TYPICAL AUXILIARY POWER SYSTEM CATEGORY I EQUIPMENT (NON-GENERAL ELECTRIC)

Part Identification System No.	Description	Qualification Method[a]
	4160-V Switchgear Buses	
R1400S001B	64B	Combination of seismic prototype test and supporting analysis
R1400S001C	64C	
R1400S001E	65E	
R1400S001F	65F	
R1400S002A	11EA	
R1400S002B	12EB	
R1400S002C	13EC	
R1400S002D	14ED	
Mounting Configuration of 4160-V Switchgear		Analysis
	480-V Switchgear Buses	
R1400S022	72B	Seismic prototype test
R1400S023	72C	
R1400S020	72E	
R1400S021	72F	
R1400S036	72EA	
R1400S037	72EB	
R1400S038	72EC	
R1400S039	72ED	
	480-V Unit Substation Transformers	
R1400S022A	72B	Seismic prototype test
R1400S023A	72C	
R1400S020A	72E	
R1400S021A	72F	
R1400S036A	72EA	
R1400S037A	72EB	
R1400S038A	72EC	
R1400S039A	72ED	
	480-V Unit Substation Voltage Regulators	
R1400S020B	72E	Fermi 2 equipment seismically tested
R1400S021B	72F	
R1400S038B	72EC	
R1400S039B	72ED	
Mounting Configuration of 480-V Switchgear, Transformers, and Voltage Regulators		Analysis

TABLE 3.10-1 SEISMIC QUALIFICATION SUMMARY: TYPICAL AUXILIARY POWER SYSTEM CATEGORY I EQUIPMENT (NON-GENERAL ELECTRIC)

Part Identification System No.	Description	Qualification Method[a]
	480-V Motor Control Centers	
R1600S002A	72B-2A	Seismic prototype test
R1600S002B	72B-3A	
R1600S003A	72C-2A	
R1600S003B	72C-3A	
R1600S003D	72C-F	
R1600S004B	72E-5A	
R1600S005A	72F-2A	
R1600S005C	72F-4A	
R1600S005D	72F-5A	
R1600S016A	72EA-2C	
R1600S017A	72EC-2D	
R1600S018A	72EC-2C	
R1600S019A	72ED-2D	
Mounting Configuration for Motor Control Centers		Analysis
	130/260-V-dc Power and Control Batteries	
R3200S003	Battery 2PA	Seismic prototype test
R3200S004	Battery 2PB	
	24/48-V-dc Instrument Batteries	
R3200S001	Battery 2IA	Seismic prototype test
R3200S002	Battery 2IB	
Battery Support Racks and Mounting Configuration		Analysis
R3200S020A-C	130-V Battery Chargers, Battery 2PA	Seismic prototype test
R3200S021A-C	130-V Battery Chargers, Battery 2PB	
R3200S023A, B	24-V Battery Charges, Battery 2IA	
R3200S024A, B	24-V Battery Charges, Battery 2IB	
R3200S025	Standby 24-V Battery Charger	
	130/260-V-dc Distribution Cabinets	
R3200S026	Main Distribution Cabinet 2PA-2	Analysis
R3200S027	Main Distribution Cabinet 2PB-2	
R3200S061A,B	Relay Room Distribution Panels	
R3200S064A, B	Relay Room Distribution Panels	
R3200S062, 65	Switchgear Room Distribution Panels	
R3200S063, 66	RHR Complex Distribution Panels	

TABLE 3.10-1 SEISMIC QUALIFICATION SUMMARY: TYPICAL AUXILIARY POWER SYSTEM CATEGORY I EQUIPMENT (NON-GENERAL ELECTRIC)

Part Identification System No.	Description	Qualification Method[a]
	24/48-V-dc Distribution Cabinets	
R3200S029	Relay Room Distribution Cabinet	Analysis
R3200S030	Relay Room Distribution Cabinet	
	260-V-dc Motor Control Centers	
R3200S015	DC Motor Control Center 2PA-1	Seismic prototype test
R3200S016	DC Motor Control Center 2PB-1	
	Battery Main Fuse Cabinets	
R3200S007A	Battery 2A-1 Dual Main Fuse Cabinet	Analysis
R3200S007B	Battery 2A-2 Dual Main Fuse Cabinet	
R3200S008A	Battery 2B-1 Dual Main Fuse Cabinet	
R3200S008B	Battery 2B-2 Dual Main Fuse Cabinet	
R3200S010	Battery 2PA Single Main Link Cabinet	
R3200S011	Battery 2PB Single Main Link Cabinet	
	Raceways	
	Conduit Supports	Analysis
	Underground Ducts	
	Primary Containment Penetrations	
	Cable Tray Hangers	
	120-V-ac I&C Power Supplies	
R3101S001	Division I Power Supply Unit	By analysis; some components by seismic prototype test
R3101S002	Division II Power Supply Unit	

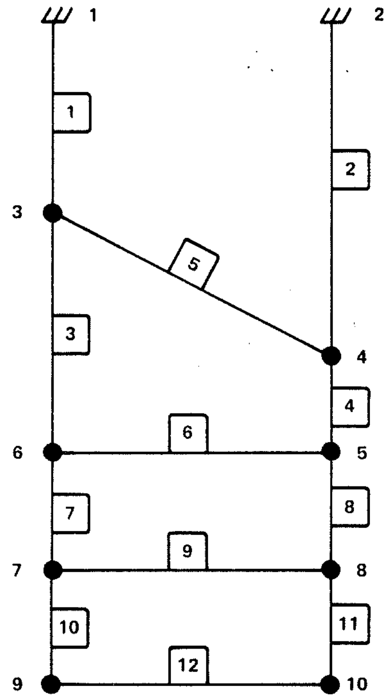
[a] Seismic prototype tests are tests of similar or identical equipment.

TABLE 3.10-2 PANEL TYPES

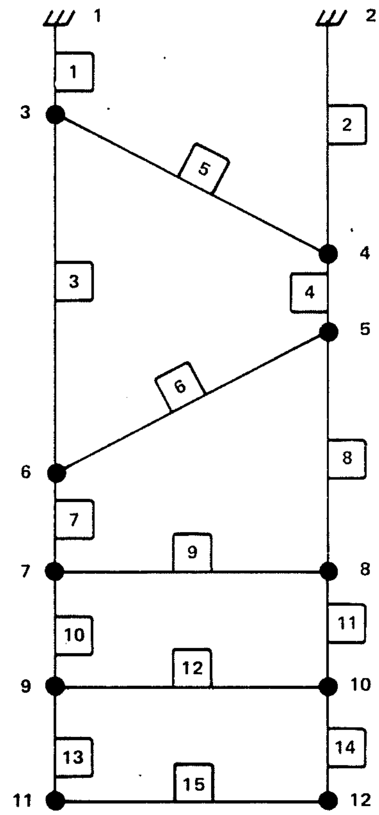
Panel Type	Use
Benchboards	Operating information and controls
Instrument and relay cabinets	Nuclear steam supply monitoring instrumentation
Local racks	Process instruments
NEMA enclosures	Miscellaneous

TABLE 3.10-3 TYPICAL SEISMICALLY QUALIFIED CONTROL PANELS, LOCAL PANELS AND RACKS (Supplied by General Electric)

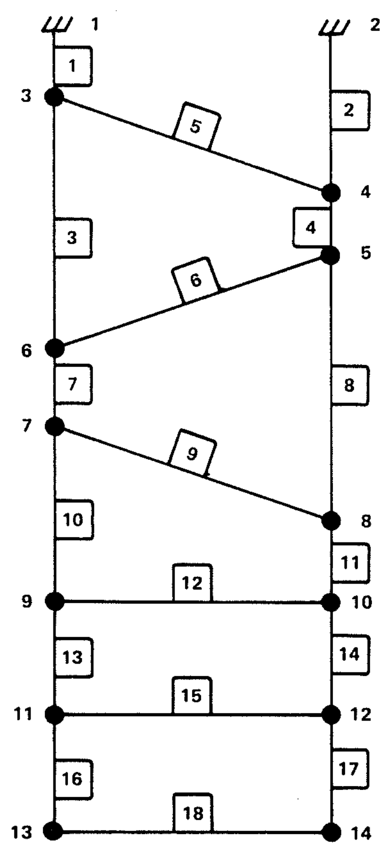
Control Panel	Description	Type	Class 1E Equipment Description
H11-P601	Reactor cooling and isolation (ECCS Div. I)	Benchboard	CR2940, MS ind. switches, CMC switches
H11-P602	Reactor water cleanup and recirculation (ECCS Div. II)	Benchboard	CR2940, MS ind. switches, CMC switches
H11-P603	Reactor control	Benchboard	Mode switch, IRM range switches, MS ind. switch CR 2940
H11-P606	Startup neutron monitor	Instrument cabinet	Trip auxiliary unit, indicator and trip unit, IRM, LRM
H11-P608	Power range neutron monitor	Instrument cabinet	APRM
H11-P609	Reactor protection system	Relay cabinet	HFA and HGA relays, magnetic contactor
H11-P611	Reactor protection system	Relay cabinet	HFA and HGA relays, magnetic contactor
H11-P612	Process instrumentation rack	Instrument cabinet	GEMAC instruments
H11-P613	Process instrumentation rack	Instrument cabinet	GEMAC instruments
H11-P614	Steam temperature recorders	Relay cabinet	CR2940 switches, HGA relay, timers, temperature monitor, inverter
H11-P617	RHR relays	Relay cabinet	HFA, HGA and HMA relays
H11-P618	RHR relays	Relay cabinet	HFA, HGA and HMA relays
H11-P620	HPCI relays	Relay cabinet	HFA and HGA relays
H11-P621	RCIC relays	Relay cabinet	HFA and HGA relays
H11-P622	Inboard isolation valve relays	Relay cabinet	HFA and HGA relays
H11-P623	Outboard isolation valve relays	Relay cabinet	HFA and HGA relays
H11-P626	Core spray	Relay cabinet	CR2940 switches, Agastat GP relays, HFA, HGA and HMA relays
H11-P627	Core spray	Relay cabinet	CR2940 switches, Agastat GP relays, HFA, HGA and HMA relays
H11-P628	Automatic depressurization relays	Relay cabinet	HFA and HGA relays
H21-P001	Core spray system A	Local rack	Barton 288, 289, and Barksdale pressure switch
H21-P002	Reactor water cleanup system	Local rack	Pressure transmitter
H21-P014	HPCI instruments	Local rack	Pressure transmitter, pressure switch, flow transmitter
H21-P015	Main steam flow	Local rack	Differential pressure switch, pressure transmitter
H21-P016	Core spray/HPCI leak detection	Local rack	Pressure switch, differential pressure switch
H21-P017	RCIC panel A	Local rack	Pressure transmitter, pressure switch, flow transmitter, flow switch
H21-P018	RHR – channel A	Local rack	Pressure switches, pressure transmitter, flow transmitter
H21-P019	Core spray channel B rack	Local rack	Pressure transmitter, pressure switch, flow transmitter
H21-P021	RHR – channel B	Local rack	Pressure switch, pressure transmitter, flow transmitter
H21-P025	Main steam flow	Local rack	Differential pressure switch, pressure transmitter
H21-P030 A thru D	SRM-IRM preamplifiers	Local NEMA enclosures	SRM-IRM preamplifiers
H21-P034	HPCI leak detection	Local rack	Pressure switch
H21-P035	RCIC leak detection	Local rack	Pressure switch, differential pressure switch
H21-P036	HPCI leak detection	Local rack	Differential pressure switch, pressure switch
H21-P037	RCIC instrument B	Local rack	Pressure switch
H21-P038	RCIC leak detection	Local rack	Pressure switch, differential pressure switch



1–DIAGONAL FRAME



2–DIAGONAL FRAME



3–DIAGONAL FRAME

Fermi 2

UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 3.10-1

TYPICAL MASS MODEL

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

3.11.1 Equipment Identification

Mechanical, electrical, and instrumentation portions of the engineered safety feature (ESF) systems, nuclear safety features, and reactor protection system (RPS) are designed to operate properly for the period required over a range of environmental conditions. The environmental conditions range from those of normal operation to those resulting from postulated accidents. The environmental design criteria established in Subsections 3.11.1 through 3.11.4 form the basis for the original Fermi 2 design. This information was derived from the parameters established by GE as part of their original design criteria. After performing their required functions, the systems can withstand these environmental conditions without functional impairment of the system involved, or other plant systems.

Table 3.11-1 lists the safety equipment and components inside the primary containment, which are designed to operate or be in a fail-safe condition during and following any accident up to the design-basis accidents (DBAs). Design environmental conditions and the associated duration of these conditions are also identified. The design environmental conditions envelop the maximum temperatures, pressures, humidity, and durations for the equipment over a wide variety of accident conditions up to a DBA, but the conditions will not necessarily occur coincidentally. Table 3.11-2 explains the significance of the design temperatures, pressures, and durations.

Safety-related equipment and components outside the primary containment that are designed to operate or fail into a safe condition during and following any accident, including the DBAs, are listed in Table 3.11-3. Design environmental conditions and associated durations of conditions are also given in this table. The design environmental and duration envelopes provided were used as guidelines for the selection of equipment and components used outside the primary containment. These conditions do not occur coincidentally during postulated accident conditions nor for all specific zone locations within the general areas listed.

Portions of the ESF systems, nuclear safety features, and RPS are located in a controlled environment which is considered an integral part of the ESF system, nuclear safety feature, or RPS. These areas and the controlled parameters are given in Table 3.11-4.

Fan-coil cooling units using water from the emergency equipment cooling water (EECW) system are used to control the environment to within the limits specified. These units are described in Section 9.4. Redundancy of units and equipment precludes the loss of the controlled environment as discussed in Subsection 3.11.4.

The RPS and ESF equipment is capable of functioning for the required design duration and, subsequently, remains in a fail-safe condition when subjected to the local environmental conditions (e.g., close proximity to the break) if the equipment is

a. Required to detect a steam line accident condition

b. Required to perform a steam line isolation function

c. Required to perform a water line isolation function and could be subjected to the environment, such as electrical cable or valve operator

d. Required for safety system operation, and is located so a steam line break in some other system exposes the safety system equipment to the local accident environment

e. Required to track the postaccident environment condition, such as pressure, temperature, hydrogen, oxygen, and radiation monitors.

Isolation valves and associated equipment required to perform the isolation function, per Items b. and c. above, will perform their required accident mitigation function in the local steam environment, and subsequently remain in a safe (closed) condition. Isolation valves inside the primary containment have been type- tested and have satisfied IEEE 382-1972.

Both equipment required for postaccident surveillance and ESF and RPS equipment, exposed to the local steam environment and required to be functional for the entire duration of the accident (items d. and e. above), will remain functional for a 100-day postaccident environment.

For equipment specified in Items a. through e. above, rotating machinery components such as pumps, motors, or operators in a safety system (i.e., emergency core cooling system [ECCS], reactor core isolation cooling [RCIC]) with a leak, are designed to function in the local environment caused by the leak.

3.11.2 Qualification Tests and Analysis

3.11.2.1 Environmental Criteria and Design Bases

The environmental conditions expected to exist during routine plant operations, both inside and outside the primary containment, are given in Table 3.11-5. Also included in Table 3.11-5 is the accident-basis radiation environment along with the DBA type. The accident-basis environmental conditions are defined as those which deviate from the routine plant operations environmental conditions given in Table 3.11-5. The accident-basis environment is specified as an envelope, which is not based upon one specific DBA, but on all postulated accidents relevant to an envelope. The worst-case environment was derived from Reference 1. The accident-basis environmental envelope is outlined in Table 3.11-2 for inside the primary containment. The ESF systems and RPS have been designed to remain operational or fail into a safe condition when subjected to the temperatures listed in Tables 3.11-2 and 3.11-5, unless such equipment is physically separated from the accident-basis environment.

The worst-case design environment for mechanical and Class 1E electrical equipment outside the primary containment is dependent on the location within the plant. Each location has been analyzed for different types of postulated accidents to define the maximum values of temperature, pressure, relative humidity, and radiation environment values. These values, which may differ between locations, are specified as the accident environment.

The radiation-accident-basis design environment has been calculated using conservative fission product inventories. The worst-case radiation-accident design environment is that resulting from the LOCA as derived from the AEC publication TID-14844, March 23, 1962.

In calculating doses on equipment and materials, fission products assumed to be in the recirculated water were 50 percent of the core halogen inventory and 1 percent of the core solid fission product inventory.

In calculating the range of radiation monitors and radiation doses to equipment in the containment atmosphere, fission products assumed to be in the primary containment atmosphere were 25 percent of the core halogen inventory, 1 percent of the core solid fission product inventory, and 100 percent of the noble gases.

With the implementation of the plant Hydrogen Water Chemistry program, normal radiation levels in those sections of the plant subject to main-steam environments will increase. Such increases have been taken into account in the overall environmental design of equipment and in the plant Environmental Qualification Program.

3.11.2.2 Qualification Tests

All Class 1E equipment and components were evaluated with respect to IEEE 323-1971. Since many of these items are used in several systems and in different plant locations, they were tested or analyzed for the worst-case situation. Wherever possible, the tests were performed to determine the malfunction limits for the critical parameters of the instruments for different applications. On the other hand, where the environmental conditions were known to have no effect on the equipment (i.e., reactor building pressure transients and radiation on solid-state electronic equipment), the tests were not performed. Class 1E equipment and components purchased after November 15, 1974, were evaluated with respect to IEEE 323-1974 (see also Subsection 3.11.5).

The Class 1E equipment supplied by GE was qualified by testing and was first described by equipment specifications that included or enveloped the intended application environment. Type tests were performed on pilot units to show conformance to the requirements of the equipment specifications. The test results were documented in a qualification test report.

In general, the Class 1E equipment supplied by GE was qualified by type tests; however, where the equipment's primary safety function is nonelectrical, such as forming a portion of a pressure boundary, calculations of the type contained in an ASME Boiler and Pressure Vessel (B&PV) Code stress analysis were used to establish qualification.

The four drywell cooler fans, which can also serve for post-LOCA atmosphere mixing, are the only continuous duty Class 1E motors inside the drywell. These motors have been tested beyond the requirements of IEEE 334-1971. Testing for these motors has included short-term transient testing at pressures up to 85 psig and temperatures up to 340°F in a saturated steam environment, and long-term testing at reduced pressure (20 psig) and temperature (250°F). Motor insulation has also been tested for radiation damage resistance.

General Electric-supplied mechanical equipment has either been qualification tested or analyzed for temperature effects to ensure that the material properties are not degraded by the environment of temperature, pressure, humidity, and radiation. Qualification testing has been done either by tests on that particular piece of mechanical equipment or on similar mechanical equipment.

The standby gas treatment system (SGTS) is designed to operate in the accident environment. Normal operation has a negligible effect on the SGTS, as indicated in Table 3.11-4. Periodic testing ensures operability of the system. Other non-GE-supplied equipment such as the control center heating, ventilation and air conditioning (HVAC) system, is located in areas not affected by the accident. The control center HVAC operation, including operation under accident conditions, is described in Subsection 9.4.1.

3.11.3 Qualification Test Results

Test results for GE-supplied Class 1E electric equipment are covered by GE Topical Report NEDO-10698, previously referenced, and in particular, Table 3.1 of that report.

The drywell cooler fans use two classes of insulation, "RN" for two-speed fans and "RH" for the single-speed fans. A motor with "RN" insulation has been tested in a saturated steam environment as indicated and has been shown to be suitable for the duty required. The tests show that the motors can withstand a temperature of at least 340°F for 3 hr, and a temperature of 320°F for an additional 4 hr, and at least 250°F indefinitely. A dosage of 10^9 rad of gamma radiation during the life of the motor can also be tolerated. Insulation breakdown occurs faster as conditions become more severe. Thermal endurance tests of 100 hr at 213°C indicate that the insulation will survive an insulation temperature (not ambient) of 105°C for 40 years.

3.11.4 Loss of Ventilation

3.11.4.1 Control Center

a. The control center is served by the control center air conditioning system (CCACS) as described in Subsection 9.4.1. The CCACS and the directly associated systems are designed to perform their intended functions during LOCA conditions, with the simultaneous occurrences of the safe-shutdown earthquake (SSE) as defined in Section 3.7, and the loss of all offsite power as described in Subsections 8.2.2.2 and 9.4.1.3. The CCACS is designed to provide fresh, filtered, and tempered ventilating air and/or air conditioning to all spaces within the control center. Space temperature inside the control center is maintained at a nominal temperature of 75°F [except for the mechanical equipment room (MER) and SGTS room which are discussed in Subsection 9.4.1.1], and the relative humidity is maintained at 50 percent on a year-round basis to ensure personnel comfort and satisfy safety-related control and electrical equipment requirements

   The reliability of the CCACS is achieved by providing two redundant air conditioning systems. The two systems separately supply air to the control center and, except for the common passive ductwork, are physically separated to preclude simultaneous loss of safety function that might occur as a consequence of a single accident. The return fans are used either to recirculate conditioned air or to discharge it to the outdoors. The supply fans in the multizone units provide the motive power to circulate the air to the various rooms. The two separate chilled water loops, each containing a liquid chiller and a pump, provide chilled water to the multizone units through two physically separated circuits

b. In the event of a failure of any major equipment component of the CCACS, the 100 percent standby system is available to preclude any adverse effect on the main control room and relay room environs. The standby air conditioning system is started manually from the main control room

c. The probability of losing both the 100 percent-capacity air conditioning systems, consisting of multizone units and liquid chillers at the same time, is remote. Only one multizone unit, liquid chiller, and chilled water pump is required for either the normal air conditioning or 100 percent recirculation modes. However, the CCACS is capable of providing fresh air from 100 percent outside air, which under certain outside temperature conditions (winter temperatures), could provide adequate cooling. The outside air can be supplied by either of the two 100 percent multizone units, and on occasions in conjunction with the two 100 percent return air fans, can recirculate the conditioned air in the rooms without outside air

d. The performance of the CCACS is verified while the system is in operation. The system ductwork and its components are subjected to leak and noise tests during manufacture and erection. Chillers, pumps, and piping systems are subjected to hydrostatic test during their manufacture and erection as well as being subjected to a manufacturing performance test

   Filters and filter housings are subjected to manufacturers' performance and production tests before installation as well as DOP and the appropriate tracer gas tests after installation. In addition, the complete air conditioning, heating, cooling, and ventilation systems are subjected to preoperational testing to demonstrate capability of maintaining the control center at 75°F and 50 percent relative humidity [except for the mechanical equipment room (MER) and SGTS room which are discussed in Subsection 9.4.1.1]

e. In the extremely unlikely event that the control center must be vacated, a remote shutdown panel located on the second floor of the auxiliary building provides remote control of the reactor systems needed to carry out the shutdown function. This panel is described in Section 7.5.

3.11.4.2 Engineered Safety Feature Switchgear Rooms

Two separate rooms are provided to house the Class IE electrical equipment. The Class 1E equipment provided in each ESF switchgear room is 100 percent redundant and satisfies IEEE-279-1971 and IEEE-308-1971 design criteria. The ESF switchgear heat-removal system is described in Subsection 9.4.2.

Each ESF switchgear room is provided with two 50 percent-capacity fan-coil units. Cooling water is supplied by the reactor building closed-cooling water/EECW (RBCCW/EECW) systems. These units are used to limit room temperature to less than 120°F, which is less than the maximum temperature for which equipment operation has been evaluated. Since the switchgear rooms are redundant, the two 50 percent heat-removal units in each room satisfy the single-failure criteria.

In the event of failure of a switchgear heat-removal unit, the ESF systems' function can be performed by the redundant equipment in the other essential switchgear room, and safe shutdown of the reactor is achievable.

3.11.4.3 Reactor/Auxiliary Building Safety-Related Ventilation Systems

During normal operation, the reactor/auxiliary building ventilation system provides ventilation for safety-related equipment in these buildings except for areas served by the CCACS. However, in the event the reactor building is isolated because of an abnormal condition, fan-coil cooling units provide the cooling for safety-related equipment. One unit of 100 percent capacity is furnished for each of the following:

a. Each division of residual heat removal (RHR) pumps

b. Each division of core spray pumps. The Division I unit also cools the RCIC pump

c. The high-pressure coolant injection (HPCI) pump room

d. Each division of the SGTS filter unit room

e. Each division of EECW pumps

f. Deleted

In addition, two units, each of 50 percent capacity, are furnished for each division switchgear room.

The fan-cooling units are physically separated and are located in Category I structures. Because of the separation, redundancy, and number of fan-coil cooling units supplied, it is extremely unlikely that cooling to both divisions of the same safety-related equipment would be lost.

The redundant battery rooms are ventilated by exhaust fans (one of two 100 percent capacity fans per room) which are required to operate during a DBA. Thus again, complete loss of battery room ventilation is unlikely. The fan-coil cooling units and exhaust fans are discussed further in Subsection 9.4.2.

3.11.4.4 Residual Heat Removal Complex Safety-Related Ventilation Systems

As described in Subsection 9.4.7, the RHR complex is composed of two identical divisions with the safety-related equipment in one division l00 percent redundant to that in the other division. Each division has two diesel generator rooms, two diesel-oil-storage rooms, two switchgear rooms, and a pump room.

To maintain conditions below the limits specified in Table 3.11-4, each diesel generator room, switchgear room, and pump room is ventilated with two 50 percent-capacity supply air fans. The intake air for the switchgear and pump rooms is filtered by medium-efficiency filters. These ventilation systems are of Category I design and are powered from the same ESF bus supplying equipment in the room being cooled. They are not required unless the equipment served is required, and are designed to start when the associated diesel generator starts, or a preset high room temperature is reached. Because a separate ventilation system is provided for each of the above rooms, the loss of a ventilation system does not affect safe shutdown of the plant. Each diesel-fuel-oil-storage room ventilation system purges air from a diesel generator room, a CO_2 storage room, and a ventilation equipment room to the outside. Each system is of Category I design and powered from the ESF bus corresponding to the diesel generator served. This system is designed to run continuously for all modes of operation. Again, as a system is supplied for each set of redundant rooms, loss of a system does not affect safe shutdown of the plant. With the redundancy and independence of ventilation systems described above, it is obvious that the probability of losing ventilation in both divisions of safety-related equipment is extremely small.

3.11.5 Environmental Qualification of Safety-Related Electrical Equipment Related To 10 CFR 50.49

All electrical equipment important to safety and exposed to a harsh environment has been reviewed to ensure that equipment required to perform necessary safety functions is capable of maintaining functional operability under all service conditions, including postulated accident conditions. This review was based on the criteria delineated for Category II plants as defined by NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment," and 10 CFR 50.49. Details of the Fermi 2 harsh environment qualification program are found in Reference 2. This document is maintained and updated periodically.

Environmental envelopes were developed specifically for this harsh-environment review, using NUREG-0588 as the source document for developing the environmental profiles. Areas inside and outside the containment containing equipment important to safety were divided into environmental zones, which included the drywell, all rooms and areas in the reactor building, the auxiliary building, and the RHR complex. The temperatures, pressures, humidities, and radiation levels were determined for each of these zones. The environments defined include the most limiting environments for the most severe postulated accident events in all applicable areas, as well as the environments expected during normal operation for the life of the plant (See Appendix B, Section B, for discussion of operation beyond the original design plant life).

The information established in Subsections 3.11.1 through 3.11.4 forms the basis for the original Fermi 2 EQ program. This information was derived from GE as part of their original design criteria. All environmental qualification activities performed for Fermi 2 related to 10 CFR 50.49 will incorporate the information contained in Reference 2.

3.11.6 DELETED IN PREVIOUS REVISION

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

REFERENCES

1. Dresden Nuclear Power Station, Unit 2, "Supplementary Information to Special Report of Incident of June 5, 1970," (submitted to the AEC, in response to its questions on the original report, by Commonwealth Edison Company).

2. Detroit Edison document, "Environmental Qualification of Safety-Related Electrical Equipment for Harsh Environment," (Identification, DTC: TEQSR; DSN: NE-1.16.9-EQE).

3. Deleted.

4. Detroit Edison document, “Summary of Environmental Parameters Used for the Fermi 2 EQ Program”, (Identification, DTC: TEGEN; DSN EQ0-EF2-018).

5. GE Specification, “BWR Equipment Environmental Requirements” (DTC: TSVEND; DSN: 22A3019).

TABLE 3.11-1 ACCIDENT ENVIRONMENT - INSIDE PRIMARY CONTAINMENT

NOTE: COMPONENTS ARE DESIGNED TO BE OPERABLE UNDER THE FOLLOWING CONDITIONS:

Component	Duration[a]	Temperature	Pressure[b]	Relative Humidity
1 Core spray injection check valve; LPCI/RHR injection check valve, reactor shutdown cooling suction valve, including operator and cable; relief valve, including operator and cable; RPV level indicator; structural components (e.g., loop restraints, RPV skirts, etc.)	45 Sec 3 hr 6 hr 1 day 100 days	340°F 340°F 320°F 250°F 200°F	-2 to 56 psig[c] -2 to 35 psig -2 to 35 psig 0 to 25 psig 0 to 20 psig	100% 100% 100% 100% 100%
2 Feedwater check valve; HPCI steam line isolation valve, including operator and cable; RCIC steam line isolation valve, including operator and cable; reactor water cleanup suction valve, including operator and cable. Lines 2 in. and smaller (isolation valves, operators, cabling); reactor vessel head spray isolation valve, including operator and cable	45 sec 3 hr 6 hr	340°F 340°F 320°F	-2 to 56 psig[c] -2 to 35 psig -2 to 35 psig	100% 100% 100%
3 Main steam isolation valves, including operator and cable; main steam drain isolation valves, including operator and cable; standby liquid control injection check valve	45 sec 1 hr	340°F 340°F	-2 to 56 psig -2 to 35 psig	100% 100%
4 Recirculation valves (main[d] valves, equalizer valve) including operators and cables	45 sec 30 minutes	310°F 285°F	-2 to 56 psig[c] -2 to 35 psig	100% 100%

NOTE: VALVES ARE DESIGNED NOT TO BE OPERABLE BUT MUST NOT FAIL OPEN UNDER THE FOLLOWING CONDITIONS:[e]

Component	Duration	Temperature	Pressure	Relative Humidity
5 Feedwater check valve; HPCI and steam line isolation valves, including operators and cables; recirculation valves (main valves, bypass valves, equalizer valves), including operator and cables; reactor vessel head spray isolation valve, including operator and cable; reactor water sample line valves, including operator and cable. Lines 2 inches and smaller (isolation valves, operators, cabling)	1 day 100 days	250°F 200°F	-2 to 25 psig -2 to 20 psig	100% 100%

TABLE 3.11-1 ACCIDENT ENVIRONMENT - INSIDE PRIMARY CONTAINMENT

Component	Duration[a]	Temperature	Pressure[b]	Relative Humidity
6 Main steam isolation valves, including operator and cable; main steam drain isolation valves, including operator and cable; standby liquid control injection check valve	3 hr	340°F	-2 to 35 psig	100%
	6 hr	340°F	-2 to 35 psig	100%
	1 day	250°F	-2 to 25 psig	100%
	100 days	200°F	-2 to 20 psig	100%
7 Drywell cooling system	45 Sec	340°F	-2 to 56 psig	100%
	3 hr	340°F	-2 to 35 psig	100%
	6hr	320°F	-2 to 35 psig	100%
	1 day	250°F	0 to 25 psig	100%
	100 days	200°F	0 to 20 psig	100%

[a] Durations shown are termination times measured from the initiation of the postulated accident; (i.e., Condition 1, the 3-hr duration, is the period from 45 sec through 3 hr, the 1-day duration is the period from 6 hr through 1 day (24 hr).

[b] The equipment inside the primary containment will be subjected to 56 psig and 135°F for a maximum of 3 days during periodic leak testing.

[c] 56 psig is 90 percent of maximum containment internal pressure of 62 psig, as allowed by ASME B&PV Code Section III, Article 13, Paragraph N-1312, Sub-Paragraph (2).

[d] For the recirculation valves to perform their safety function they must close following a recirculation line break, so that the core flooding can be carried out in the required time. For this safety requirement the environmental conditions will not exceed 310°F at 56 psig for ½ hr. The specified conditions in (4) above are to enable a normal vessel shutdown cooling procedure during a steam leak.

[e] Some of the equipment identified in Items 5, 6, and 7 is also required to operate at the beginning of the event. This equipment is therefore also shown in Items 2 through 4 above.

TABLE 3.11-2 DESIGN-BASIS ACCIDENT ENVIRONMENTAL ENVELOPE

Temperatures	
340°F	Upper boundary on maximum superheat temperature for a steam leak with the RPV at 400-500 psig, containment at 35 psig.
320°F	Maximum superheat temperature during shutdown cooling line flush after reactor has been depressurized to 150 psia.
250°F	Maximum long-term temperature in the containment during the first day following a postulated DBA.
200°F	Extended long-term temperature in the containment following a postulated DBA.
Pressures	
-2 psig	Assumed negative design pressure of the primary containment.
56 psig	Positive design pressure of the primary containment, coinciding with the 281°F design temperature.
35 psig	Containment pressure corresponding to all the noncondensables initially in the drywell being transferred to wetwell.
25 psig	Upper boundary on extended long-term pressure at one day and shorter following a postulated DBA.
20 psig	Upper boundary on extended long-term pressure at longer than one day following a postulated DBA.
62 psig	Assumed peak containment pressure.
Durations	
45 sec	Conservative time duration to cover positive design pressure.
1 hr	Time duration during which valves that must isolate automatically on low RPV level or high drywell pressure must be operable.
3 hr	Time duration to depressurize the RPV at a rate not exceeding 100°F/hr, down to 150 psia.
4.5 hr	Time at which shutdown cooling system flush is complete. Normal shutdown cooling necessitates closure of recirculation line valves.
6 hr	Time duration to complete RPV depressurization to approximate containment pressure. This time includes RPV depressurization to 150 psia not exceeding a rate 100°F/hr, flushing of system, and depressurization to approximate containment pressure.
100 days	Maximum postulated accident duration.

TABLE 3.11-3 DESIGN-BASIS ENVIRONMENT - OUTSIDE PRIMARY CONTAINMENT[a]

NOTE: COMPONENTS ARE DESIGNED TO BE OPERABLE UNDER THE FOLLOWING CONDITIONS:

Component	Duration	Temperature[b]	Relative Pressure	Humidity
1. HPCI pump, turbine, control, instrumentation and electrical equipment; RCIC pump, turbine, controls, instrumentation, and electrical equipment (other than in steam tunnel).	1 hr[c,g]	148°F[c]	7 in. H_2O gage[c]	100%[c]
2. RHR system isolation valves, including operators and cable; RHR pumps, heat exchanger, controls instrumentation and electrical equipment; core spray systems isolation valves, including operator and cable; core spray pumps, controls, instrumentation, and electrical equipment.	6 months[d] 1 hr	148°F[d,e] 148°F[d,e]	Zero in. H_2O gage 7 in. H_2O gage	90% 100%

NOTE: VALVES ARE DESIGNED NOT TO BE OPERABLE BUT MUST NOT FAIL OPEN UNDER THE FOLLOWING CONDITIONS:

Component	Duration	Temperature[b]	Relative Pressure	Humidity
3. HPCI system isolation valves, including operator and cable; RCIC system isolation valves, including operator and cable; main steam isolation valves in steam tunnel, including operators; feedwater isolation valves, including operator and cable; reactor water cleanup isolation valves, including operator and cable	13 sec[f] 1 hr	228°F 220°F	5.1 psig[f] 2.0 psig	100%[f] 100%

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[a] Design condition where operation is required. Note that these are design conditions and the actual conditions to which this equipment is environmentally qualified under the Fermi 2 EQ Program are documented in EQ0-EF2-018.

[b] Temperatures given do <u>not</u> take into account any temperature rise caused by direct steam impingement.

[c] 148°F, 100 percent R.H., and 7 in. static pressure may occur concurrently for the 1 hr as given, but R.H. and static pressure will decay after this period.

[d] Temperature based on RHR equipment operating. RHR pump basement and sub-basement quadrants: 153°F peak.

[e] Motors rated for continuous operation in an ambient temperature of 104°F will operate in a higher ambient temperature with decreased life expectancy. Space cooling may be required to limit the ambient to an acceptable level.

[f] Steam tunnel transient conditions due to main steam line rupture.

[g] These time frames are retained for historical purposes. HPCI is environmentally qualified to support a 3-hr mission time.

TABLE 3.11-4 ENVIRONMENTAL DESIGN OF AREAS CONTAINING SAFETY-RELATED EQUIPMENT AND COMPONENTS – OUTSIDE CONTAINMENT[b]

Location	Temperature	Relative Humidity
1. Control center[a]	75 °F	60% max.
2. ESF switchgear room	< 120°F max.	90% max.
3. Core spray, RCIC, RHR, HPCI emergency equipment rooms	148 °F max.[c]	90% max.
4. Standby gas treatment system room	104 °F max.	90% max.
5. Thermal recombiner area[d]	104 °F max.	90% max.
6. Emergency equipment cooling water pump room	104 °F max.	90% max.
7. Diesel Generator rooms (RHR complex)	65 °F min. 122 °F max.	-
8. Switchgear room (RHR complex)	65 °F min. 104 °F max.	-
9. Pump room (RHR complex)	104 °F max.	100% max.
10. Diesel-generator fuel-oil-storage room, and CO_2 storage room (RHR complex)	65 °F min. 125 °F max.	
11. Ventilation equipment rooms (RHR complex)	65 °F min. 104 °F max.	

Note a-Temperature for mechanical equipment room (MER) is 95°F.

Note b-These are design conditions and the actual conditions to which the equipment in this area is environmentally qualified under the Fermi 2 EQ Program are documented in EQ0-EF2-018.

Note c-RHR pump basement and sub-basement quadrants: equipment qualified to 153°F peak temperature.

Note d-The thermal recombiner units are retired in place, de-energized, and isolated from primary containment with redundant locked-closed isolation valves. The associated area coolers are retained and credited as a heat sink for post-accident environmental conditions.

TABLE 3.11-5 DESIGN ENVIRONMENTAL CONDITIONS (PLANT OPERATIONAL)[l,m]

Area	Pressure (as noted)	Temperature (°F)	Relative Humidity (Precent)	Radiation Type	Operating Dose Rate[a] Plant Operation	Operating Dose Rate[a] System Operation	Integrated Dose Normal[b]	Integrated Dose Accident[c]	DBA Type[d]	DBA Dose Rate
I. Primary containment[e]										
Drywell, with sacrificial shield	-0.5 to 2.0 psig	135° average[k] --- minimum	40-50 normal 90 maximum ----- minimum	Gamma neutron		--				
1. Above Core	Same as above	Same as above	Same as above	Same as above	25.0 $5x10^4$	--	$8.8\ x10^6$ $6.3x10^{13}$	$2.6x10^7$	LOCA	$1.3x10^6$
2. Core region	Same as above	Same as above	Same as above	Same as above	50.0 $1.4x10^5$	--	$1.8x10^7$ $1.8x10^{14}$			
3. Under reactor pressure vessel	Same as above	135° average[k] 100° minimum[f] 185° maximum[g]	Same as above	Same as above	7.2 <1	--	$2.5x10^6$ $<1.3x10^9$	$2.6x10^7$	LOCA	$1.3x10^6$
4. Vicinity recirculation pump motors	Same as above	128° average ---- minimum 135° maximum[k]	Same as above	Same as above	25.0 $2x10^3$	--	$8.8x10^6$ $2.5x10^{12}$	$2.6x10^7$	LOCA	$1.3x10^6$
5. 15 ft from recirculation pump motors	Same as above	135° average[k] ---- minimum 150° maximum	Same as above	Same as above	4.0 $2x10^3$	--	$1.4x10^6$ $2.5x10^{12}$	$2.6x10^7$	LOCA	$1.3x10^6$
6. Suppression pool	Same as above	Same as above	Same as above	Same as above	0.1 $2x10^2$	--	$3.5x10^4$	$2.6x10^7$	LOCA	
II. Secondary containment (reactor building)										
General floor area	-0.10 in. to -1.0 in. Water gage static pressure	70° normal 104° maximum 40° minimum	40 normal 90 maximum	Same as above	0.001	--	$3.5x10^2$	$1.7x10^5$	LOCA	$6.5x10^2$
HPCI & RCIC area	Same as above	70° normal 104° maximum[h] 60° minimum	Same as above[h]	Same as above	0.015	0.200	$5.3\ x10^3$	$4.5x10^4$	LOCA	$1.6x10^2$

TABLE 3.11-5 DESIGN ENVIRONMENTAL CONDITIONS (PLANT OPERATIONAL)[l,m]

Area	Pressure (as noted)	Temperature (°F)	Relative Humidity (Precent)	Radiation Type	Operating Dose Rate[a] Plant Operation	Operating Dose Rate[a] System Operation	Integrated Dose Normal[b]	Integrated Dose Accident[c]	DBA Type[d]	DBA Dose Rate
Core spray & RHR equipment area[i]	Same as above	70° normal 104° maximum 40° minimum	Same as above[h]	Same as above	0.015	0.030	5.3×10^3	4.5×10^4	LOCA	1.6×10^2
Steam Tunnel	-0.10 in. to 1.0in.	125° normal	40-50 normal	Gamma	5	--	1.8×10^6	4.5×10^4	LOCA	1.6×10^2
	Water gage static pressure	140° maximum 40° minimum	90-98 maximum					$>2.5\times10^2$	Rod drop	2.5×10^2
Standby liquid control area	Same as above	100° maximum 70° minimum	40 normal 90 maximum							
24-in. Pipe containing suppression pool H_20 (typical pipe)	Same as above	70° normal 104° maximum 40° minimum	Same as above	Gamma	0.0	--	0.0	7.9×10^5	LOCA	1.4×10^4
Cleanup systems	Same as above	Same as above	Same as above							
1. Heat exchangers				Gamma	15.0	--	5.4×10^6	1.7×10^5	LOCA	6.5×10^2
2. Pump room				Gamma	>0.05	--	1.8×10^4	1.7×10^5	LOCA	6.5×10^2
3. Filters & tanks				Gamma	10.0	--	3.6×10^6	1.7×10^5	LOCA	6.5×10^2
SGTS	Same as above	Same as above	Same as above	Gamma	0.001	--				
III. Turbine building[j]										
General areas protected by shields	0.0 in. to -0.25 in. H_20 gage static pressure	70° normal (winter) 104° maximum (elect) 40° minimum 90° normal (Summer) 120° maximum (non-elect)	40 normal 90 maximum	Gamma	0.001	--	4×10^3	--		
Operating floor, General	Same as above	Same as above	Same as above	Gamma	0.005-0.020	--	77.0×10^4	--		

TABLE 3.11-5 DESIGN ENVIRONMENTAL CONDITIONS (PLANT OPERATIONAL)[l,m]

Area	Pressure (as noted)	Temperature (°F)	Relative Humidity (Precent)	Radiation Type	Operating Dose Rate[a] Plant Operation	Operating Dose Rate[a] System Operation	Integrated Dose Normal[b]	Integrated Dose Accident[c]	DBA Type[d]	DBA Dose Rate
Contact high-pressure Turbine	Same as above	Same as above	Same as above	Gamma	0.5	--	$1.8x10^5$	--		
Contact low-pressure Turbine	Same as above	Same as above	Same as above	Gamma	0.1	--	$3.5x10^4$	--		
Equipment bay (htrs., condensers, etc)	Same as above	Same as above	Same as above	Gamma	0.05-5.0	--	$1.8x10^6$	--		
Steam-jet air ejector	Same as above	Same as above	Same as above	Gamma	15	--	$5.3x10^6$	--		
Condensate treatment	Same as above	Same as above	Same as above	Gamma	10	--	$3.5x10^6$	--		
IV. Radwaste building [j]										
Equipment cells (valve & pump rooms)	0.0 in. to -0.5 in. H_20 gage static pressure	70° normal 120° maximum 40° minimum	40 normal 90 maximum	Gamma	0.020	--	$7.0x10^3$	--		
Main control room	0.0 in. to -0.25 in. H_20 gage static pressure	75° normal 80° maximum 70° minimum	Same as above	Gamma	0.001	--	$3.5x10^2$	--		
Storage tanks (unprocessed) (unprocessed)	Same as above	Same as above	Same as above	Gamma	20.0		$7.0x10^6$	-- 0		
Centrifuge	Same as above	Same as above	Same as above	Gamma		100	$1x10^7$	--		
V. Main control room	0.10 in. to 0.5 in. H_20 gage static pressure	75° normal 95° maximum 60° minimum	50 normal 60 maximum	Gamma	0.0005	--	$1.75x10^2$	$3.0x10^0$		

TABLE 3.11-5 DESIGN ENVIRONMENTAL CONDITIONS (PLANT OPERATIONAL) [l,m]

[a] Gamma dose rate — Rads/hr
Neutron Flux — Neutrons/cm^2/sec

[b] Gamma dose — Rads
Neutron fluence — Neutrons/cm^2 (NVT)
Normal conditions — Integrated over 40years – 100% load factor @ rated power

[c] Gamma dose — Rads
Neutron fluence — Neutrons/cm^2 (NVT)
Accident conditions — Integrated over 6 months

[d] LOCA analysis was based upon the assumption that 100% of the noble gases, 50% of the halogens, and 1% of the solid fission products were released from the core.

[e] Primary containment atmosphere during normal operation may be inerted with nitrogen.

[f] Components located in the turbine building or radwaste building required to operate under normal conditions, if any, should be designed for equivalent conditions as shown for reactor building.

[g] The same minimum temperature (100°F), shall apply at the inside base of the shield wall. Air velocity over vessel insulation and exposed vessel parts shall be approximately 6 ft./sec.

[h] During the loss of offsite power, and emergencies, except during DBA, temperature of area underneath the RPV will be maintained at 185°F or lower for up to 30 minutes.

[i] Whenever the residual heat removal and core spray motor and emergency core cooling systems are running, during test periods, area space coolers may be required to maintain the ambient temperature listed.

[j] The maximum temperature and humidity will occur simultaneously in these spaces less than 1% of the time.

[k] The drywell volumetric average temperature may increase over 135°F and up to 145°F.

[l] These are design conditions and the actual conditions to which the equipment in this area is environmentally qualified under the Fermi 2 EQ Program are documented in EQ0-EF2-018.

[m] The environmental conditions documented in this table were established by GE as part of the original Fermi 2 design criteria as documented in GE specification 22A3019 (Reference 5).

3.12 SEPARATION CRITERIA FOR SAFETY-RELATED MECHANICAL AND ELECTRICAL EQUIPMENT

3.12.1 Introduction

This section defines separation criteria for safety-related mechanical and electrical equipment. Safety-related equipment to which the criteria apply is that equipment necessary to mitigate the effects of abnormal operational transients or accidents. The objective of the criteria is to delineate the separation requirements necessary to achieve true independence of safety-related functions compatible with the redundant equipment provided.

The sections to follow individually address mechanical and electrical equipment separation. The specific systems and equipment to which the criteria apply are listed, followed by the corresponding criteria.

3.12.2 Mechanical Systems and Equipment

3.12.2.1 Affected Systems and Equipment

The mechanical systems and related equipment (i.e., piping, valves, pumps, and heat exchangers) affected by the criteria of Subsection 3.12.2.2.1 are

a. Emergency core cooling system (ECCS)

   1. Low pressure coolant injection (LPCI) system
   2. Core spray system
   3. High pressure coolant injection (HPCI) system
   4. Automatic depressurization system (ADS).

b. Reactor core isolation cooling (RCIC) system

c. Deleted

d. Standby gas treatment system (SGTS)

e. Emergency equipment cooling water (EECW) system

f. Control center air conditioning system (CCACS)

g. Fan-coil unit ventilation systems

   1. ECCS equipment pump rooms
   2. SGTS filter unit rooms
   3. EECW pump area
   4. Hydrogen recombiner area
   5. Engineered safety feature (ESF) switchgear rooms
   6. CCACS equipment room

7. Residual heat removal (RHR) complex equipment rooms.

h. Nuclear pressure relief system

i. Main steam isolation valves (MSIVs)

j. Containment cooling mode of RHR system

k. Emergency equipment service water (EESW) system

l. Standby liquid control system (SLCS)

m. RHR service water system

n. Emergency diesel generator (EDG) and oil systems

o. Control air system.

3.12.2.2 Criteria

3.12.2.2.1 General

Separation of the affected mechanical systems and equipment is accomplished in such a manner that the substance and intent of 10 CFR 50 are fulfilled.

Consideration is given to the redundant and diverse requirements of the affected systems.

Consideration is given to the type, size, and orientation of possible breaks of the reactor coolant pressure boundary (RCPB) specified in Section 3.6.

The protection afforded by the ECCS network satisfies the single failure criterion. A single failure means an occurrence that results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered part of the single failure. Fluid systems are considered to be designed against an assumed single failure, if a single failure of any active component (assuming passive components function properly) does not result in a loss of capability of the system to perform its safety function.

The affected mechanical systems and equipment, along with their associated structures, are appropriately separated so that, by virtue of separation or other adequate provisions, systems important to safety are adequately protected against:

a. The LOCA dynamic effects outlined in Section 3.6

b. Missiles as defined in Section 3.5

c. Fires capable of damaging redundant mechanical safety equipment.

The need for and the adequacy of separation are determined in conjunction with the criteria specified in Sections 3.5 and 3.6.

3.12.2.2.2 System Separation

Piping for a redundant safety system is run independently of its counterpart. Supports, restraints, and mechanical components of redundant piping of the same system are not shared in common, unless it can be shown that such sharing does not significantly impair their ability to perform their safety functions.

Containment penetrations are separated so that damage to or failure of one branch of a system will not render its redundant counterpart(s) inoperable.

3.12.2.2.3 Physical Separation

Mechanical equipment and piping, including control system conduit and tubing for the ECCS, are separated so that no single credible event, such as a LOCA, is capable of disabling sufficient equipment to prevent reactor shutdown, removal of decay heat from the core, or isolation of the containment to the extent that an offsite dose in excess of 10 CFR 50.67 or 10 CFR 100 requirements results.

The ADS is separated from the HPCI system such that no portion of the HPCI influent line or HPCI steam supply line is located within jet impingement damage distance or pipe movement damage distance of any component considered essential to the operation of the ADS.

Provisions are made to ensure that no single failure could incapacitate both the HPCI and RCIC.

The RHR service water system, EESW system, and EDGs, all located in the RHR complex, are split into two divisions separated by a common wall that also serves as a missile barrier (see Section 3.5). The divisions are identical and each division is capable of performing the intended system safety function independent of the other division. The equipment of each system is housed in a Category I structure that also provides protection against natural phenomena such as tornadoes and floods. Piping between the RHR complex and the reactor/auxiliary building is provided for each division and is separated so that no single event is capable of damaging the piping in both divisions.

The CCACS likewise consists of two redundant, full-capacity systems, separated such that no single failure can incapacitate both divisions.

Independent fan-coil units are provided for each redundant piece of equipment and are separated in the same manner and provide the same protection as the equipment they serve.

3.12.3 Electrical Systems and Equipment

3.12.3.1 Affected Systems

The systems with electrical portions that might be affected by the criteria of Subsection 3.12.3.2.1 are those listed in Subsection 3.12.2.1 plus the reactor protection system (RPS) and other systems required for safe shutdown of the reactor. Affected equipment included in these systems are instrument channels, trip systems, trip actuators, standby power sources, average power range monitors (APRMs), and intermediate range monitors (IRMs).

3.12.3.2 Criteria

These systems have been fabricated in accordance with the intent of Institute of Electrical and Electronics Engineers (IEEE) 279-1971 and IEEE 308-1971 as applicable. Explicit criteria are given in Subsection 3.12.3.2.1.

3.12.3.2.1 General

As a consequence of the design of these systems and components and the separation provided, the single-failure criterion defined in accordance with Paragraph 4.2 of IEEE 279-1971 is satisfied. In addition, several potentially adverse effects are considered in the determination of the degree of separation. These are:

a. Electrical fires in wireways that could cause failure of unprotected insulation on other cables in the same wireway

b. Gross failure of electrical equipment in any single compartment of an instrument or control panel

c. Mechanical damage of equipment in a single location, the area of which is limited by the damaging potential of surrounding equipment

d. Damage caused by earthquakes of the safe-shutdown earthquake (SSE) magnitude

e. Single events that could disable an automatic protective function, i.e., reactor scram, containment isolation, or core cooling. Also, single failures that could incapacitate both the HPCI and RCIC systems, with initiation of the ADS and ECCS resulting during an abnormal operational transient.

Equipment associated with the RPS, safe shutdown systems (systems required for safe shutdown), and ESF systems are identified so that two facts are physically apparent to operating and maintenance personnel: first, that the equipment is part of the RPS, safe shutdown systems, or the ESF system engineered equipment; and second, the grouping (or division) of enforced segregation with which the equipment is associated, is identified. Identification and divisions conform to the following:

a. Panels and racks associated with the RPS, safe-shutdown systems, and ESF systems are labeled with marker plates that are conspicuously different in color from those for other panels or racks. The marker plates include identification of the proper division (I or II, for example). The equipment identification number and applicable segregation code, both numerical and color code, are applied to each piece of safety-related equipment

b. Junction and/or pull boxes enclosing wiring for the RPS, safe-shutdown systems, and ESF systems have identification similar to and compatible with the panel and racks considered above

c. Cables external to cabinets and/or panels for the RPS, safe shutdown systems, and ESF systems have color-coded jackets to distinguish them in color from other cables and to identify their separation division, as applicable. The color coding system is used throughout the plant for identification. For instance, Division I cable is orange, Division II cable is blue, and balance- of-plant (BOP) cable is black. The exceptions to cable color loading are described in Section 8.3.1.5.1. Reactor protection system cables are colored black since they are routed through their own exclusive, totally enclosed raceway system, as described in Subsection 3.12.3.2.2. The raceways are clearly identified with RPS channel numbers

d. Raceways that carry RPS wiring are identified at entrance points of each room they pass through (and exit points unless the room is small enough to facilitate convenient following of cable), and at intervals along the raceways, by markers indicating their separation division. The raceways have alpha-numeric fire resistant painted identification with color coding as described in Item c. above

e. Redundant sensory equipment is identified by suffix letters in accordance with Tables 3.12-1 and 3.12-2 for the RPS and Table 3.12-3 for the ESF systems. These tables also show the allocation of sensors to separated divisions. Allocations for safe shutdown systems sensors are given in Table 3.12-1 for the deenergize-to-operate type and in Table 3.12-3 for the energize-to-operate type.

3.12.3.2.2 System Separation

The following apply specifically to the RPS; however, the wiring guidelines also apply to the safe shutdown systems:

a. Wiring for the RPS, including the neutron monitoring system (NMS), outside the control system cabinets is run in enclosed raceway, with each of the four channels monitoring each variable being physically separated. Under-vessel neutron monitoring cables are exempted from this wireway requirement because of space limitations and the need for flexibility of IRM cables. The IRM and source range monitor (SRM) cables may be combined in the same wireway; however, the four- divisional separation is maintained.

b. Wiring to duplicate sensors on a common process tap is run in separate wireways to separate destinations

c. Wiring for sensors of more than one variable in the same trip channel can be, and is, run in the same wireway

d. Wires from both RPS trip system trip actuators to a single group of scram solenoids may be run in a single wireway. However, a single wireway does not contain wires to more than one group of scram solenoids. Wiring for two solenoids on the same control rod may be run in the same wireway

e. Cables through the containment penetrations are so grouped that failure of all cabling in a single penetration cannot prevent a scram. Conduits inside the dry-well are grouped so that failure of any one conduit will not result in disabling any APRM channel

f. Power supplies to systems that deenergize to operate require only that separation which is deemed prudent to give continuity of operation. Therefore, even though the load circuits go to separated panels, the protection system flywheel motor-generator sets and load circuit breakers are not required to comply with the separation criteria of this subsection for safety reasons

g. Even though the load circuits go to separated panels, the RPS wiring is run and/or protected in such a manner that no common source of potentially damaging energy (e.g., electrical fire in non-RPS wireways) could reasonably result in loss of ability to scram when required

h. The RPS has four independent input instrument channels for each measured variable. The four separate wireways for the four sensors for a specific variable are, in some cases, combined into two groupings or divisions for routing purposes by combining Divisions IA and IB as shown in Table 3.12-1 and Figure 3.12-1. However, under permitted bypass conditions, there is no case in which the total disabling of equipment within a single division is capable of preventing a required scram action.

3.12.3.2.3 Physical Separation

Electrical equipment and wiring for the ESF systems are segregated into separate divisions that are designated I and II, so that no single credible event is capable of disabling sufficient equipment to prevent reactor shutdown, removal of decay heat from the core, or isolation of the primary containment in the event of an accident. Separation requirements apply to control power and motive power for all systems concerned. In addition, the RCIC and HPCI systems are treated as functionally redundant counterparts and are divisionally separated, the RCIC system being in Division I, the HPCI system in Division II.

Arrangement and/or protective barriers are such that no locally generated force or missile can destroy both redundant safe shutdown and ESF system functions. In addition, because of treatment as functionally redundant systems, the same is true for the HPCI and RCIC systems. In the absence of confirming analysis to support less stringent requirements, the following rules apply:

a. In rooms or compartments having heavy rotating machinery, such as the main turbine generator, or the reactor feedwater pumps; or in rooms containing high-pressure feedwater piping or high-pressure steam lines such as those between the reactor and the turbine, at least one cable is run in metal (rigid or flexible) conduit if cables of different divisions are located in the room or compartment

b. Switchgear associated with redundant safety systems that are located in a potential mechanical damage zone such as that discussed above have a minimum horizontal separation of 20 ft or are separated by a protective wall equivalent to a 6-in.-thick reinforced-concrete wall

c. In any compartment containing an operating crane such as the turbine building, main floor, and the region above the reactor pressure vessel (RPV), there is a minimum horizontal separation of 20 ft or a 6-in.-thick reinforced-concrete wall between trays containing cables of the two divisions

d. Each RPS motor-generator set is housed in its own reinforced-concrete room with 12-in.-thick walls. The only path a missile such as a flywheel could take (to leave the room) would be through the door, but the position of the flywheel with respect to the door opening eliminates that possibility. Therefore, redundant safety-related systems cannot be failed due to such an event. The damage would be limited to the associated equipment located inside the room where the flywheel failure occurred. In addition, the RPS system cabling in this location will be contained in conduit

e. In the battery rooms, the only equipment is the batteries themselves and the only cabling is the main dc power cables to the main distribution cabinets located outside the battery rooms. The main power cables in the battery room, in addition to being fire retardant, are contained in conduit.

Arrangement of wiring and cabling ensures that fire will not propagate from one division to another. Cables have been tested and certified to be fire retardant (i.e., cable burning will stop when flame is removed). In addition, cables have been tested and certified to remain operating for 5 minutes during a fire. In addition, arrangement of wiring cabling of the HPCI and RCIC systems ensures that both systems are not disabled by a single failure. In the absence of confirming analysis to support less stringent requirements, the general guidelines used to determine the allocation of electrical wiring between segregated divisions of the safe shutdown and ESF systems are

a. Separation is such that no single failure can prevent operation of an ESF function (e.g., core cooling). Redundant (even dissimilar) systems are, in some cases, needed to perform the required function to satisfy the single-failure criteria. Table 3.12-4 illustrates the separation of subsystems of the nuclear safety and ESF systems valves. Figures 3.12-2 through 3.12-4 illustrate the ESF equipment separation into divisions and the allowable interconnections through isolating devices. Interconnecting wireways are assigned to the same division as the power for the contained circuits, and separation between divisions is maintained except at the immediate area of entrance to the cabinet of the other division, where steel barriers are provided

b. The inboard isolation system valve wiring between the control panel and the valve proper is separated from the outboard isolation valve wiring. (Figure 3.12-3 illustrates this requirement.) The manual controls for the isolation valves may be treated as an exception to this inboard division, if deemed necessary from an operational point of view, provided that no single failure can prevent the required automatic operation of at least one of an inboard/outboard pair of isolation valves

c. Routing of cables for RPS safe shutdown and ESF systems power through rooms or spaces where there is potential for accumulation of large quantities (gallons) of oil or other combustible fluids through leakage or rupture of lube oil or cooling systems is avoided. Where such routing is practically unavoidable, only one division of these cables is allowed in any such space

d. In any room or compartment in which the only source of fire is of an electrical nature, cable trays have a minimum horizontal separation of 3 ft, if no physical barrier exists between trays. If a horizontal separation of 3 ft is unattainable, a fire-resistant barrier is provided, extending at least 1 ft above (or to the ceiling) and 1 ft below (or to the floor) line of sight between the two trays. These trays are of the open-bottom type (ladder type) for power/control cable and solid-bottom-covered type for instrumentation

e. For subject cable trays, there is a minimum vertical separation of 5 ft between horizontal trays stacked one above the other; however, vertical stacking of trays is avoided wherever possible. In cases where trays must be run stacked one above the other, and where the trays meet the 5-ft vertical separation requirement, the lower tray has a solid-metal cover. Where the 5-ft separation can not be met, the upper tray also has a solid-metal bottom and a fire-resistant barrier is placed between the redundant trays

f. In the case of crossover of one tray over another (or over a panel), there is a minimum vertical separation of 18 in. (tray bottom to tray bottom), with the bottom tray covered with a metal cover, and the top tray provided with a metal bottom for a distance of 5 ft on each side of the tray crossover point

g. Any openings in floors for vertical runs of cables are sealed with fire-resistant material

h. There are two sets of Category I ductbanks between the RHR complex and the Reactor/Auxiliary building, with a Division I and Division II ductbank in each set. In each case, the buried cable ducts between the RHR complex and the Reactor/Auxiliary building provide adequate cable separation to maintain independence of redundant circuits.

   The first set of ductbanks was installed during plant construction. The buried cable runs between the RHR complex and reactor/auxiliary building are housed in reinforced- concrete ducts below grade and are physically separated by a distance of at least 20 ft. The separation is 30 ft at the point the cable ducts leave the reactor/ auxiliary building. The ducts make a sweeping bend with a minimum separation of 20 ft. The ducts then run parallel with a separation of 24 ft. This separation increases until the ducts enter (still below grade) the RHR complex. 4160-V essential power circuits are not routed within these ductbanks.

   The second set of ductbanks and associated manholes is installed above the maximum ground water elevation of 576.0 ft with ducts sloped to the manholes, such that circuits contained are not subject to continuous wetting. These are also cast-in-place, rectangular reinforced concrete ductbanks, but are located with the ductbank top approximately six inches below the surface and manhole covers at grade level. The Division I and Division II 4160-V ductbanks are separated by approximately 25 feet at the Auxiliary building entrance. The separation narrows to approximately 10’-6” at the closest point as they make a sweeping turn and widens to approximately 20 feet at the entrance to manholes 16946A and 16947A. The ductbank separation again narrows to approximately 7’-8” at a top elevation of approximately 580’-6” (three feet below grade) and runs underneath the ISFSI Transfer Pad to manholes 16946B and 16947B. The ductbanks exit manholes 16946B and 16947B with a separation of approximately 15 feet that increases to a separation of greater than 20 feet after approximately 30 feet from the manholes. The separation increases to approximately 115 feet during the run from manholes 16946B and 16947B to manholes 16946C and 16947C, located near the RHR building. Ductbank separation for the ductbank run between manholes 16946C and 16947C and the RHR Building cable vaults is greater than 80 feet. 4160-V essential power circuits are routed within these ductbanks.

The Division II 4160-V ductbank crosses above the original Division I ductbank at two locations:

1. Approximately 15 feet south of the Auxiliary building, with the Division I ductbank at a top elevation approximately 8'-9" below grade and a vertical separation between the ductbanks of approximately five feet, with an additional twenty inches of reinforced concrete separating the closest conduits in each ductbank.
2. Approximately forty feet north-west of manhole 16947B, with the Division I ductbank approximately five feet below grade and a vertical separation between the ductbanks of approximately eighteen inches, also with an additional twenty inches of concrete between the closest conduits in each ductbank.

The 4160-V RHR cable vaults and the manholes and ductbanks between these cable vaults and the Reactor/Auxiliary building cable vaults are designed as tornado missile barriers per the requirements of Regulatory Guide 1.76 Revision 1. Because of the tornado missile barrier design, the redundant cables will not be subject to a common mode failure from a tornado missile and, due to the separation provided, a redundant division cable will not cause a failure in the surviving divisional cable. (See Section 3.5 for a discussion of tornado missile protection.)

The minimum horizontal and vertical separation and/or barrier in the cable spreading room is

a. Where cables of different separation divisions approach the same or adjacent control panels with spacing less than the 3-ft minimum, at least one cable is run in metal (rigid or flexible) conduit to a point where 3 ft of separation exists
b. A minimum horizontal separation of 3 ft is provided between trays containing cables of different separation divisions if no physical barrier exists between trays. If a horizontal separation of less than 3 ft is not attainable, a fire-resistant barrier is provided extending at least 1 ft above (or to the ceiling) and 1 ft below (or to the floor) line-of-sight distance between the two trays. These trays may be of the open-bottom type (ladder type) or solid-metal-bottom type
c. Vertical stacking of trays carrying cables of different divisions is avoided wherever possible. There is a minimum vertical separation of 5 ft between horizontal trays running parallel one above the other. In situations where 5 ft of separation cannot be maintained, the top trays have solid metal bottoms and the bottom trays have solid covers with a fire-resistant barrier provided between the trays
d. In the case of crossing of a tray of one separation division over a tray of the other division, there is a minimum vertical separation of 18 in. (tray bottom to tray bottom), and the bottom tray is covered with a metal cover and the top tray

is provided with a metal bottom for a distance of 5 ft on each side of the intersection (identical to Item f. above).

No single control panel (or local panel or instrument rack) includes wiring essential to the protective function of two systems that are backups for each other, except as allowed by the applicable paragraphs below:

a. If two panels containing circuits of different separation divisions are less than 3 ft apart, there is a steel barrier between the two panels. Panel ends closed by steel end plates are considered acceptable barriers, provided that terminal boards and wireways are spaced a minimum of 1 in. from the end plate

b. Floor-to-top of panel fireproof barriers are provided between adjacent panels of different divisions

c. Penetrations of separation barriers within a subdivided panel where they occur are sealed so that an electrical fire could not reasonably propagate from one section to the other and destroy the protective function

d. For operational reasons, the mode switch, scram discharge volume (SDV) high-water-level-trip bypass switch, scram reset switch, and manual scram switch (all manual switches) are located on one panel. In this case, each device is mounted in a can with a sufficient number of barrier devices to maintain adequate separation. Also, conduit is provided from the cans to the logic cabinets

e. A specific set of separation criteria must be met by the internal wiring of individual operating panels, logic cabinets, or instrument racks that contain components (control devices and wiring) of both ESF divisions. Generally, the criteria specify the use of separate terminal boards and spacing of terminal boards and wiring to preclude the possibility of fire propagation from one division of wiring to another. Separation of control devices is accomplished by physical location or a suitable metallic barrier. Whenever possible, the redundant control devices are located on opposite sides of the barrier formed by the end enclosures of adjacent panels to effect the desired separation and immunity to fire damage. Alternatively, separation of a pair of redundant control devices that must be located in close proximity is achieved by totally enclosing the wiring to one of the devices within a fire-resistant material. In a few specific cases the criterion for separation within the metallic enclosure (cabinet or panel) is relaxed. This relaxation of the criterion is allowable since an analysis for the particular system shows that the complete failure of the equipment within the enclosure will not compromise the system's redundant counterpart or the redundant power supply (refer to the single-failure analysis in GE Report NEDO-10139, Compliance of Protection Systems to Industry Criteria: General Electric BWR Nuclear Steam Supply System)

f. Logic wiring associated with the plant annunciator and sequential recorders in some instances runs between divisional areas of a subdivided panel. An example would be the electrical connection of relay isolated contacts in each section of the RPS to provide an alarm function for the plant annunciator system. Interposing relays or equivalent isolation means are incorporated to effect the required degree of electrical separation. If practical design constraints tend to compromise the ability to provide the desired degree of separation, the design is analyzed to establish the existence of single-failure design adequacy

g. In response to an NRC concern where BOP cables tied electrically into Division II cables, Edison reviewed about 550 schematics where 1E and non-1E circuits interfaced electrically without the intrinsic separation provided by isolation devices as described in Regulatory Guide 1.75. As a result of this review, several cases where 1E and non-1E circuits interface electrically were identified, and the cases were categorized into representative samples for the purpose of analysis and documentation. The analysis of the representative samples of 1E and non-1E circuits showed that the ability of the 1E system to perform its assigned function was not impaired by the postulated electrical faults on the non-1E circuits that are associated with them, or the circuits were revised to provide additional protection or isolation. These analyses are maintained as a controlled design calculation. Future design changes must meet these conditions or additional analyses will be performed to the same criteria as established in these initial cases.

h. Single-fuse isolation between 1E and non-1E loads is acceptable if the following conditions are met:

   1. The fuse must be safety related and thus meet commensurate quality and qualification standards

   2. The fuse must be mounted in a safety-related enclosure

   3. It must be shown that the single-failure criterion is satisfactorily met assuming an accident and the single failure in the safety-related fuse; i.e., if an accident occurs and an assumed fault occurs in the non-1E load, it must be demonstrated that given a single failure of the safety-related fuse under the worst fault in the non-1E load and assuming all the potential cascading consequences of that fault/ failure, adequate safe shutdown may still be achieved by alternative safety-related means.

3.12.4 Comparison With Regulatory Guide 1.75

Fermi 2 design criteria were developed and electrical systems designed prior to issuance of Revision 1 of Regulatory Guide 1.75 in January 1975. The Fermi 2 design has, however, been reviewed, and the following differences have been identified:

a. Section 3 of IEEE Standard 384-1974 classifies associated circuits as non-Class 1E circuits that share power supplies, enclosures, or raceways with Class 1E circuits or are not physically separated from Class 1E circuits by acceptable separation distances or barriers. The Fermi 2 circuits are divided into three categories: Division I, Division II, and BOP. Divisional separation of redundant safety equipment is maintained throughout. However, no attempt is made to uniquely identify BOP cables that would fall into the "associated" category.

Fermi 2 separation criterion does state that, once a BOP cable comes in contact with a divisional tray, it cannot cross over to the other divisions. This is maintained by a computerized cable-routing program that does not allow a cable to be routed to the other division

The degradation of Class 1E circuits is avoided by the following design features:

1. The insulating materials and cable ratings are the same for BOP cables as Class 1E cables, the only exception being the fiber-optic cables. These cables are non-conducting cables, carrying light pulses, i.e., carry no fault energy and therefore cannot create shorts between circuits. The cable insulation is non-flame propagating, and certified to IEEE-383-1974, Paragraph 2.5.

2. The cable insulation is selected and tested not to propagate fire, thus eliminating the danger of a cable propagating a failure from one tray to another.

b. Balance-of-plant loads that are fed from Class 1E buses use breakers as a separation device. These breakers are fully qualified Class 1E devices. The cabling from the breakers to the load and to the control panel is BOP cabling. The breakers have full fault protection, but they are not opened on a LOCA signal. The incidence of reported false LOCA signals, notably due to high drywell pressure, indicates that this would cause unnecessary degradations in plant operational flexibility. As an added precaution, the large loads handled by 4160-V breakers have the external control circuit operated by a BOP battery, while the internal breaker control, including fault clearance, is operated by Class 1E battery power. The interfacing devices are Class 1E relays located in the switchgear (see Figure 3.12-5). The 480-V breakers feeding BOP loads from Class 1E buses are controlled entirely from the Class 1E battery. Since these are nonessential loads, the control cables between the switchgear and control room are treated as BOP cables. Control fuses in the switchgear protect the Class 1E battery. The Class 1E 480 volt distribution panel on each EDG, which feed BOP loads, is protected by 1E fuses located in the Class 1E MCC feeding the distribution panels. These Class 1E fuses provide isolation of the BOP load, assuming a failure of the distribution panels Class 1E overcurrent protective devices on faults on the non-1E circuit. The consequences of the loss of the Class 1E distribution panel have shown that EDG operability is not impacted.

c. Section 5.1.2 of IEEE 384 states that exposed Class 1E raceways be marked at intervals not to exceed 15 ft. Edison Specification 3071-128, standard EE, calls for markings "at point of entry into a room. . .." In addition, a standard note on all Fermi 2 cable tray identification drawings states that "tray numbers should occur at close intervals to enable any section to be readily and accurately identified"

d. Section 5.6.3 of IEEE 384 calls for Class 1E wire bundles or cables internal to control boards to be distinctly identified

These boards have already been manufactured, and no such marking has been provided. Division I and Division II circuits have been carefully isolated. Where a Division I circuit enters a Division II panel, it is run in metallic conduit, and the Division I device is canned. The same applies to Division II circuits entering a Division I panel. There is, however, no attempt to separate the BOP wiring or devices from the Class 1E wiring. The materials of the wiring are the same, which ensures that the reliability of the safety functions is not degraded.

TABLE 3.12-1 REACTOR PROTECTION SYSTEM AND DEENERGIZE-TO-OPERATE SENSOR SUFFIX LETTERS AND DIVISION ALLOCATION (INCLUDING PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM)

Total Sensors	Division IA	Division IB	Division IIA	Division IIB
	Trip Logic A1	Trip Logic B1	Trip Logic A2	Trip Logic B2
4	A	B	C	D
	Part of Trip System A	Part of Trip System B	Part of Trip System A	Part of Trip System B

TABLE 3.12-2 FOUR DIVISION GROUPING OF THE NEUTRON MONITORING SYSTEM UTILIZING FOUR DRYWELL PENETRATIONS

Penetration Designation[a]	F IRM A & E APRM 1	G IRM B & F APRM 2	A IRM C & G APRM 3	B IRM D & H APRM 4
Wireway	NA	NB	NC	ND
Neutron monitoring channel				
APRM channel[b]	1	2	3	4
APRM 2-out-of-4 Trip Voter[b]	1	2	3	4
IRM	A & E	B & F	C & G	D & H
RPS trip logic	A1	B1	A2	B2

[a] Penetrations across top of table for four penetrations grouping carry cables for neutron monitoring channels shown and each channel serves RPS trip logic directly below it.

[b] Each APRM channel provides inputs to all four 2-out-of-4 trip voters.

TABLE 3.12-3 ENGINEERED SAFETY FEATURES SYSTEM SENSOR SUFFIX LETTERS AND DIVISION ALLOCATION[a,b]

Total Sensors for Each Parameter	Division I Sensor Suffix Letters		Division II Sensor Suffix Letters	
4	A	C	B	D
	Operate system A directly, and system B through isolation devices		Operate system B directly, and system A through isolation devices	

[a] For systems required for safe shutdown energize-to-operate sensors, use this table. For systems required for safe shutdown deenergize-to-operate sensors (using RPS power), use Table 3.12-1.

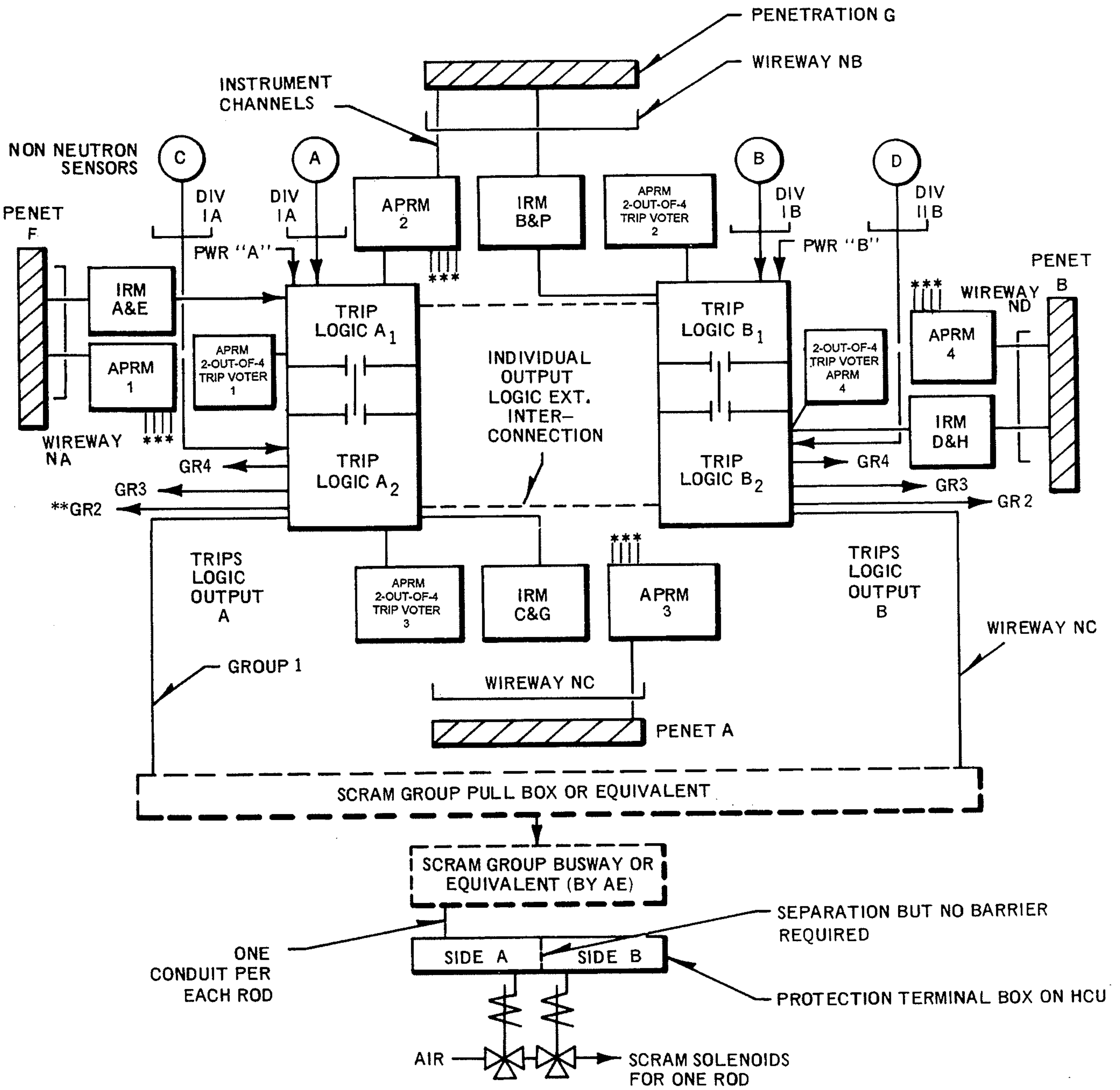
[b] ESF initiation is similar to RPS initiation, i.e., one of two times two (see Table 3.12-1 and Section 7.3).

TABLE 3.12-4 SYSTEM AND SUBSYSTEM SEPARATION

Division I	Division II
Core spray A	Core spray B
Automatic depressurization[a]	HPCI
RHR A (pumps A and C)	RHR B (pumps B and D)
Inboard safe shutdown system valves (except RCIC)[b]	Outboard safe shutdown system valves (except RCIC)[b]
Emergency equipment cooling water A	Emergency equipment cooling water B
RCIC	

[a] Wiring to each ADS valve inside the drywell is in a separate rigid conduit. All ADS valves wiring is separated as far as practical from HPCI piping inside the drywell.

[b] The inboard HPCI isolation valve control is independent of the outboard HPCI valve and of all RCIC isolation valve wiring.



*RPS SENSORS A&B OR C&D MAY BE CONNECTED TO A COMMON PROCESS TAP.
RPS SENSORS A&C OR B&D MUST NOT BE CONNECTED TO A COMMON PROCESS TAP.
**WIREWAYS NA, NB, ETC. MAY BE ASSIGNED TO SEPARATE DIVISIONS AS APPROPRIATE TO PLANT LAYOUT.
***EACH APRM CHANNEL PROVIDES INPUTS TO ALL FOUR 2-OUT-OF-4 TRIP VOTERS

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FIGURE 3.12-1

FOUR PENETRATION REACTOR PROTECTION SYSTEM SEPARATION CONCEPT

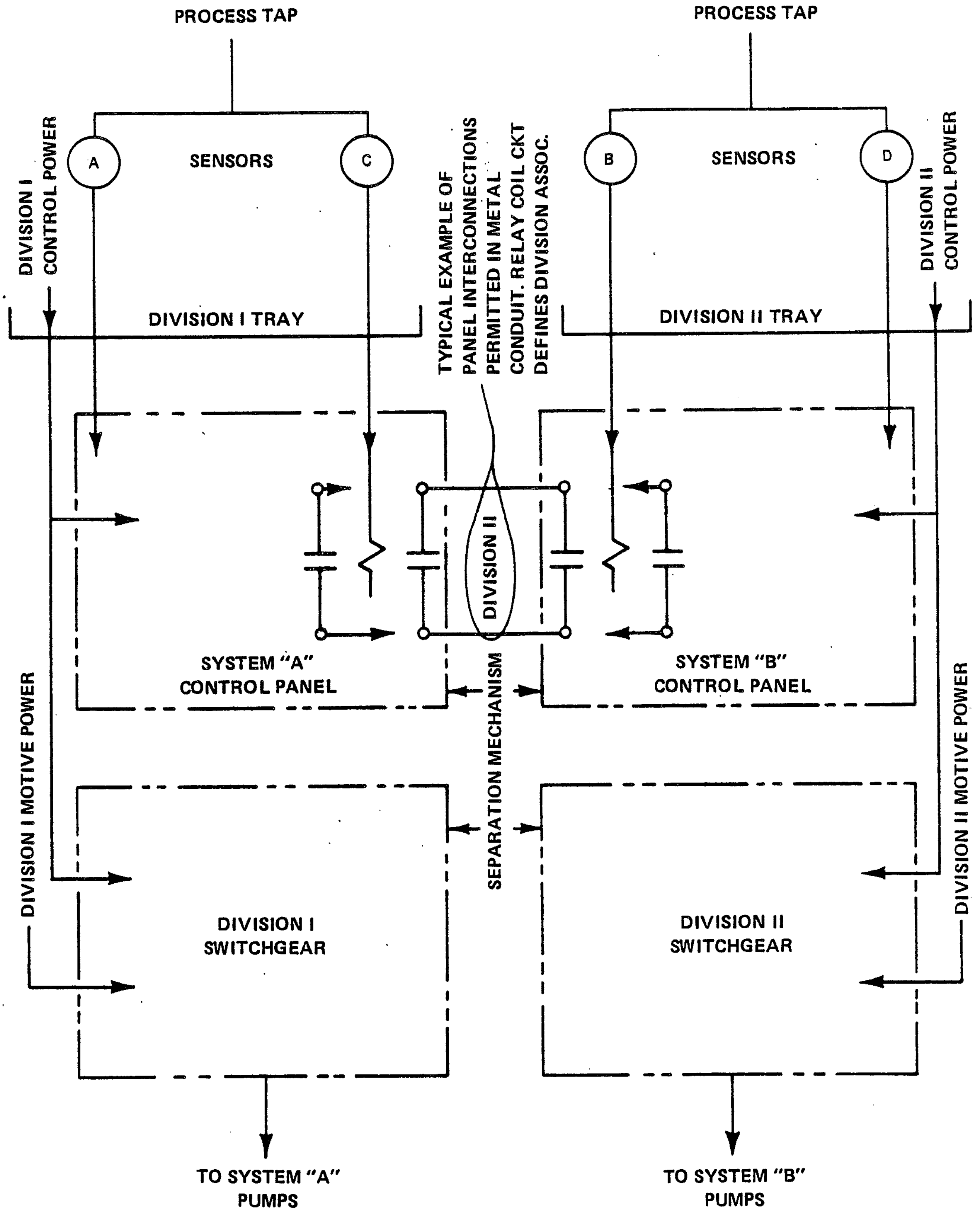

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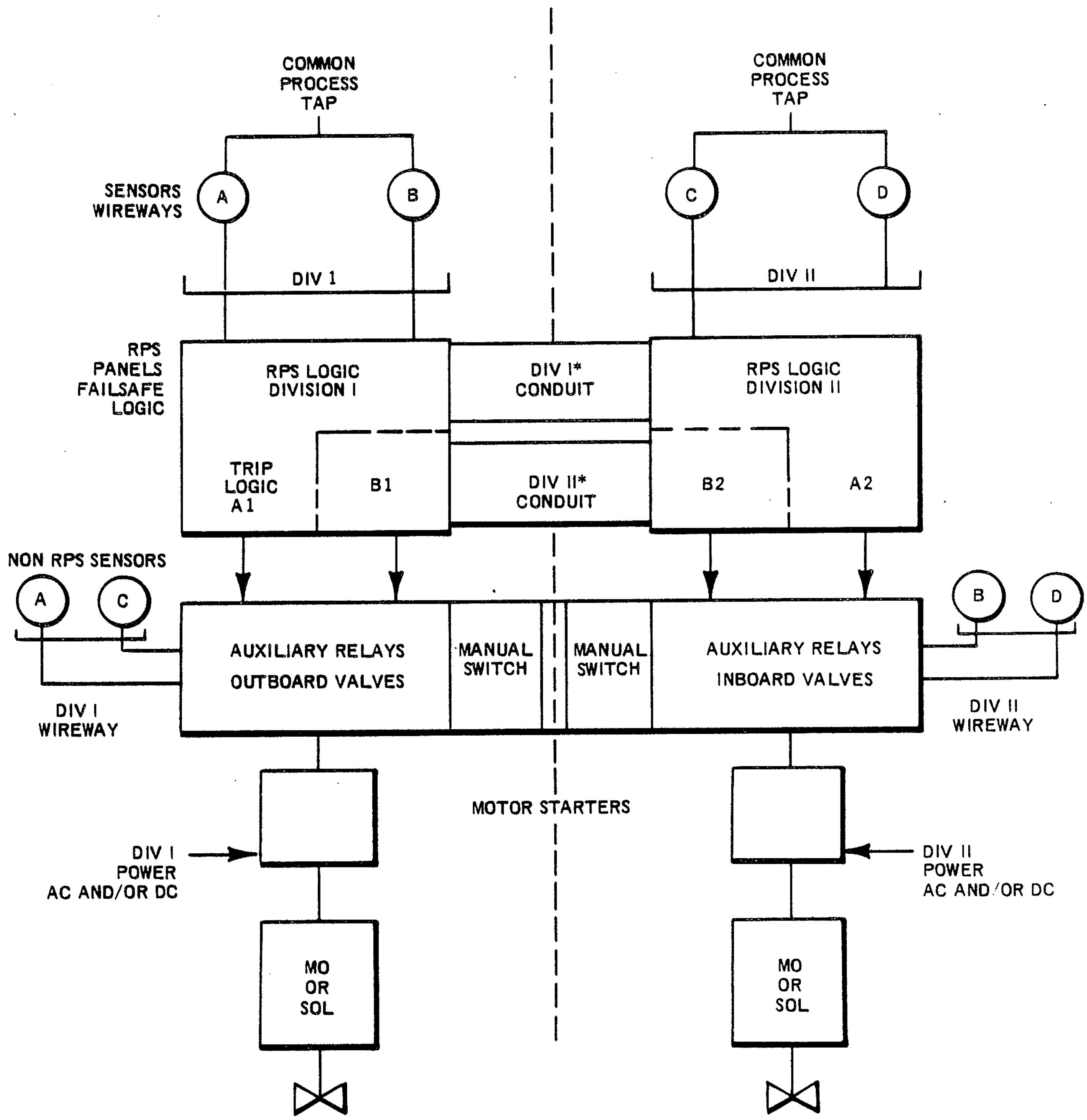
FIGURE 3.12-2

ENGINEERED SAFETY FEATURE SYSTEM SEPARATION SCHEME



*INTERCONNECTING CONDUITS USED FOR MAIN STEAM ISOLATION VALVE LOGIC ONLY

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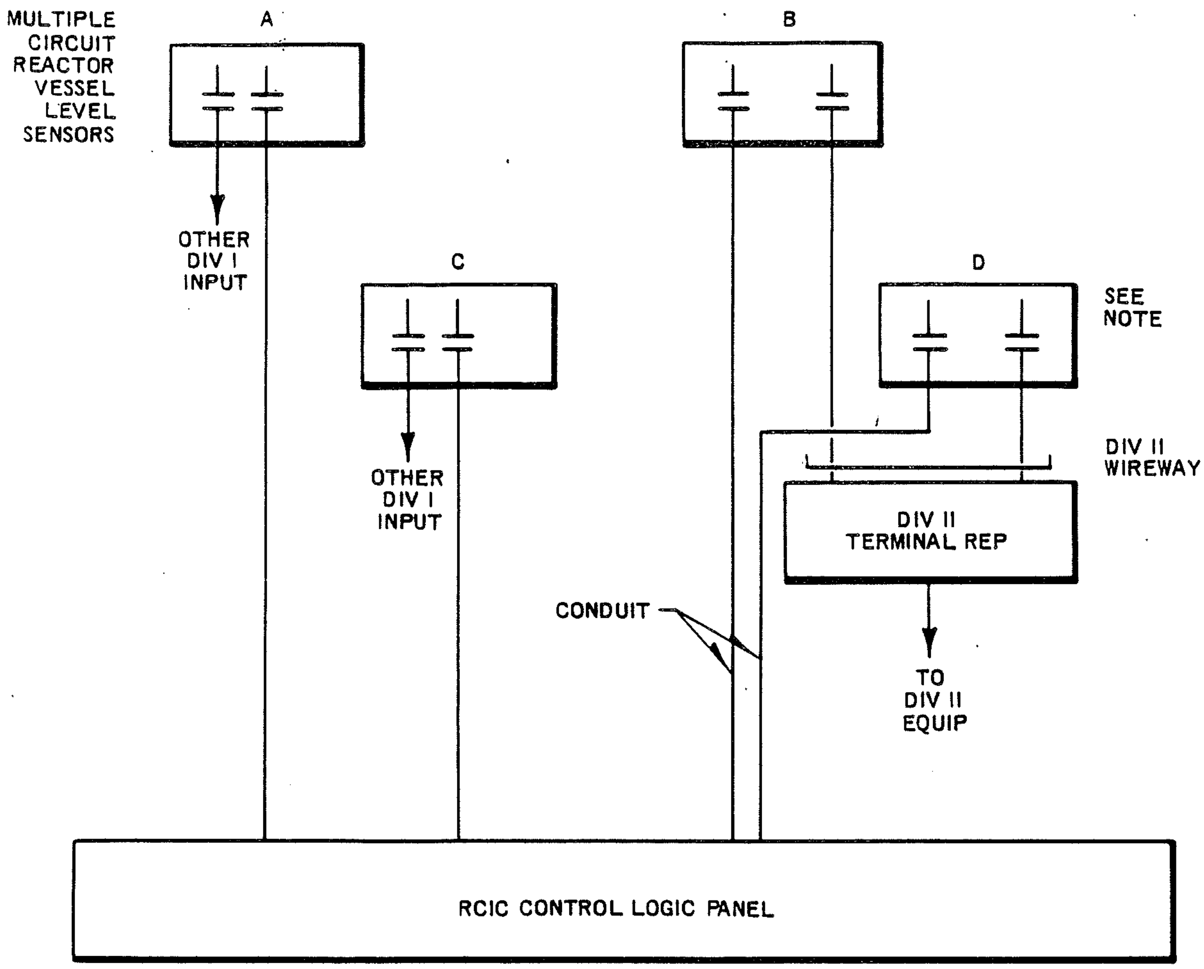
FIGURE 3.12-3

NUCLEAR SAFETY FEATURES
REACTOR PROTECTION SYSTEM AND SAFE
SHUTDOWN SYSTEM – SEPARATION CONCEPT



NOTE: CIRCUITS FOR RCIC INITIATIONS UTILIZE CONTACTS ELECTRICALLY SEPARATE FROM THOSE USED FOR DIV II INPUTS.

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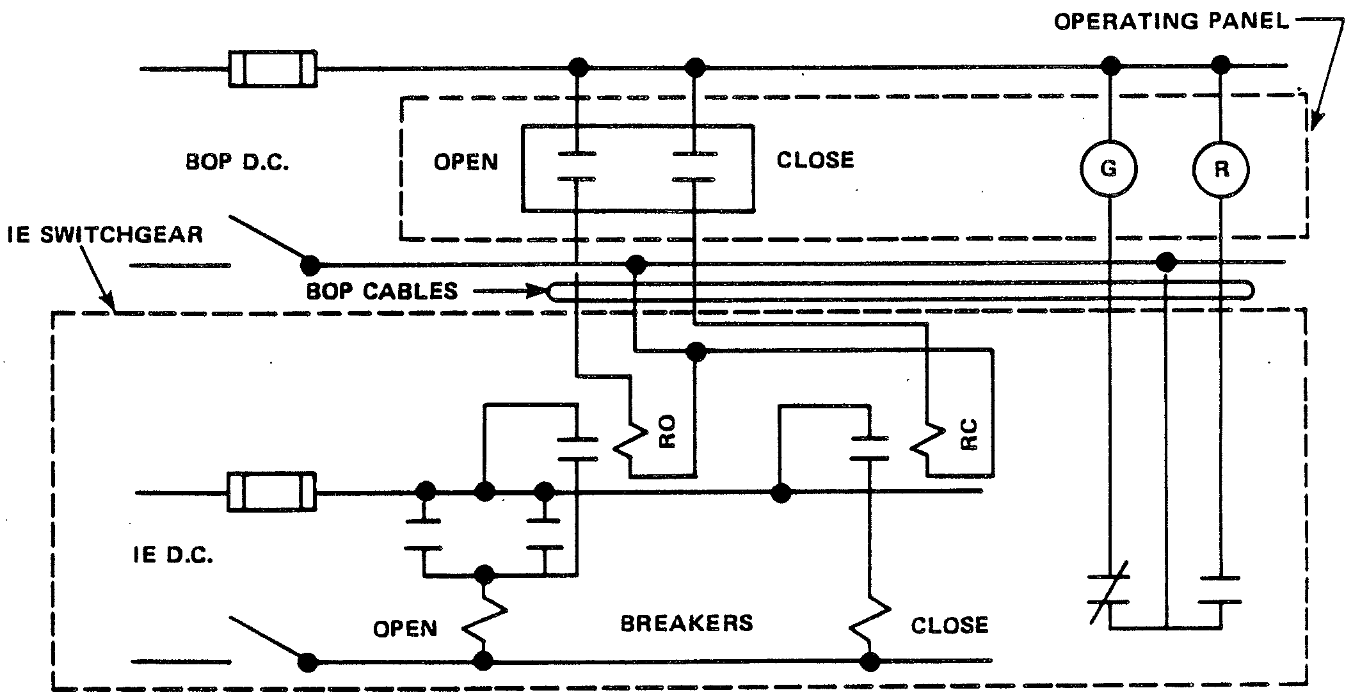
FIGURE 3.12-4

REACTOR CORE ISOLATION COOLING SENSOR SEPARATION SCHEME



Fermi 2

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FIGURE 3.12-5

CLASS 1E – BALANCE-OF-PLANT INTERFACE

3.13 COMPUTER PROGRAMS FOR STRUCTURAL ANALYSIS AND DESIGN

The computer programs referred to in Sections 3.7 and 3.8 are described herein. All programs have been verified, within the stated assumptions and limitations, for the correctness of the theory used and the validity of the results obtained for a wide variety of typical problems. Results have been checked against known solutions or solutions obtained from other programs using a different analytical approach. Furthermore, whenever applicable, internal checks, such as equilibrium and orthogonality checks, are printed out for each problem. Subsection 3.13.1 describes the computer programs used by Sargent & Lundy (S&L). Subsection 3.13.2 describes the computer programs used by Chicago Bridge & Iron (CBI). Subsection 3.13.4 describes the computer programs used by Stone & Webster, Michigan (S&W). Major computer programs used by others are described in Subsection 3.13.3.

3.13.1 Computer Programs Used by Sargent & Lundy

Subsections 3.13.1.1 through 3.13.1.32 describe computer programs used by S&L. The building structures to which each were applied are shown parenthetically following each program title.

3.13.1.1 AFEM - Axisymmetric Finite Element Method (Reactor/Auxiliary Building)

The Axisymmetric Finite Element Method (AFEM) is used for analysis of axisymmetric thick shells of revolution subjected to axisymmetric loads. The analysis is done using the finite element method with axisymmetric solid triangular elements. The analysis may be done for nodal loads, normal and shear pressures, and thermal loadings. For force or displacement-type boundary conditions, oblique or skewed restraints may be used.

The program output includes the displacements of each node, and the direct stresses, shear stresses, and principal stresses with their associated directions for each element. Boundary stresses are obtained through an extrapolation procedure, and the section stress resultants are obtained using a numerical integration procedure.

The Axisymmetric Finite Element Method is a modified version of the finite element program AMG032, developed by Rohm & Haas Company for the Redstone Arsenal Research Division, Huntsville, Alabama. It was obtained and modified by S&L in 1971. It is currently maintained on a UNIVAC 1106 operating under EXEC 8.

Three of the problems used to validate AFEM are presented here. Results obtained from AFEM are compared with hand calculations.

3.13.1.1.1 Problem 1

Problem 1 concerns the analysis of a uniformly loaded circular plate as shown in Figure 3.13-1. The solution from AFEM is compared to an evaluation of equations given by Timoshenko and Goodier in Reference 1.

Two computer runs using different grid sizes were used. As shown in Figures 3.13-2 through 3.13-5, the theoretical and computer solutions compare favorably.

3.13.1.1.2 Problem 2

Figure 3.13-6 shows the geometry for problem 2. A thick cylinder is loaded with its own weight and the body force is kept constant with the depth. For this problem the expressions for stresses are

$$\sigma_Z = \int_Z^h k \, d_Z = -kh \left[1 - \left(\frac{Z}{h}\right)\right] \tag{3.13-1}$$

and

$$\sigma_r = \sigma_\theta = \tau_{r\theta} = \tau_{rZ} = 0 \tag{3.13-2}$$

K = material density = 200 lb/in.^{3}

As shown in Figure 3.13-7, the results from AFEM are within 5 percent of the theoretical solution.

3.13.1.1.3 Problem 3

The third problem is a temperature distribution problem. The thick cylinder in Figure 3.l3-8 is subjected to a steady-state temperature gradient. The inside temperature of the cylinder is 10°F higher than the outside temperature. A steady state is assumed in this long cylinder with a concentric hole. If T_i is the temperature on the inner surface of the cylinder and the outer surface temperature is zero, the temperature T at any distance r from the center is represented by the expression

$$T = \frac{T_i}{\log\left({}^{b}/_{a}\right)} \log \frac{b}{r} \tag{3.13-3}$$

The expressions for stresses are given in Reference 1. Properties for this problem are

a. Radius of the cylinder a = 5 in.

b. Radius of the hole b = 1 in.

c. Modulus of elasticity E = 10^6 psi

d. Poisson's ratio υ = 0.2

e. Thermal coefficient α = 1/3000 in./in./°F

f. Inside temperature Ti = 10°F

As shown in Figures 3.13-9 and 3.13-10, the results compare favorably.

3.13.1.2 DSASS - Dynamic Seismic Analysis of Shear Structures (Reactor/Auxiliary Building)

Dynamic Seismic Analysis of Shear Structures (DSASS) is used for dynamic analysis of structures that could be modeled as slabs interconnected with springs. The masses are lumped at the slab levels and the springs offer resistance to relative displacements at their ends. The program considers the combined effects of translational, torsional, and rocking motion. The program uses either the response spectrum or time-history method of analysis. In the case of time-history analysis, the decoupled differential equations of motion are numerically integrated using Newmark's β-method (Reference 2). The program output includes modal responses, probable maximum time history of structural response, and response spectrum at any slab.

The DSASS program was developed by S&L in 1967. Version V is currently maintained on a UNIVAC 1106 operating under EXEC 8. This version has been used successfully since 1971.

To demonstrate the validity of the program, a three-story shear frame is analyzed and compared to a solution obtained by Biggs. The structure is represented by the closed-coupled system shown in Figure 3.13-11. The masses and stiffness values used are also given in the figure.

For the analysis, the following response spectrum was used:

Frequency (Hz)	Displacement (in.)
1.00	3.30
2.18	1.40
3.18	0.66

Table 3.13-1 represents a comparison of results obtained from DSASS and by Biggs. As demonstrated in this comparison, results obtained from DSASS are accurate.

3.13.1.3 DYNAS - Dynamic Analysis of Structures (Reactor/ Auxiliary Building and Residual Heat Removal Complex)

Dynamic Analysis of Structures (DYNAS) is designed for performing dynamic analysis of structures that can be idealized as three- dimensional space frames or rigid slabs connected by translational or torsional springs. The program considers the combined effects of translational, torsional, and rocking motions on the structure. The program uses either the response spectrum or time-history method of analysis, depending on the type of forcing function available. Both methods use the normal mode approach. In the case of time-history analysis, the decoupled differential equations of motion are numerically integrated using Newmark's β-method (Reference 2).

The program can be used for analyzing structures with parts having different associated dampings. The option is also available to analyze a large structural system using the modal synthesis technique. The system is divided into subsystems whose modal characteristics are computed separately, and then synthesized to obtain the response of the complete system. The base motion can be applied simultaneously in two orthogonal directions. Response spectra can be generated at specified slabs or joints. The program output includes modal responses, probable maximum responses, time history of structural response, and response spectrum at specified joints.

The DYNAS program, developed by S&L in 1970, is currently maintained on a UNIVAC 1106 operating under EXEC 8. Two examples of the problems used for validating the program are presented herein.

3.13.1.3.1 Problem 1

In the first problem, a three-story shear building is analyzed and compared to a solution obtained by Biggs (Reference 3). The structure is represented by the closed-coupled system shown in Figure 3.13-11, as are masses and stiffness values used. For the analysis, the following response spectrum was used:

Frequency (Hz)	Displacement (in.)
1.00	3.30
2.18	1.40
3.18	0.66

The results obtained by Biggs and from DYNAS are compared in Table 3.13-1.

3.13.1.3.2 Problem 2

In the second example, results of DYNAS are compared to those obtained by Wilson et al. (Reference 4) using the SAP IV program. At the fixed end of a cantilever beam (Figure 3.13-12), an acceleration is applied (Figure 3.13-13). The natural periods calculated by both SAP IV and DYNAS are shown in Table 3.13-2.

A comparison of the bending moment at the fixed end of the cantilever beam is shown in Figure 3.13-14. As demonstrated in both examples, DYNAS performs an accurate analysis.

3.13.1.4 DYNAX - Dynamic Analysis of Axisymmetric Structures (Reactor/Auxiliary Building and Residual Heat Removal Complex)

Dynamic Analysis of Axisymmetric Structures (DYNAX) is a finite element program for performing both static and dynamic analyses of axisymmetric structures. Its formulation is based on a small displacement theory.

Three types of finite elements are available: quadrilateral, triangular, and shell. The geometry of the structure can be general as long as it is axisymmetric. Both the isotropic and orthotropic elastic material properties can be modeled. Discrete and distributed springs are available for modeling elastic foundations.

For static analysis, input loads can be structure weight, nodal forces, nodal displacements, distributed loads, or temperatures. Loads can be axisymmetric or nonaxisymmetric. For the solids of revolution, the program outputs nodal displacements and element and nodal point stresses in the global system (radial, circumferential, and axial). In the case of shells of revolution, the output consists of nodal displacements, and element and nodal point shell forces in a shell coordinate system (meridional, circumferential, and normal).

For dynamic analysis, three methods are available: direct integration, modal superposition, and response spectrum. In the case of dynamic analysis by direct integration or modal superposition, a forcing function can be input as

a. Nodal force components versus time for any number of nodes

b. Vertical or horizontal ground acceleration versus time.

For nonaxisymmetric loads, the equivalent Fourier expansion is used. In the case of dynamic analysis by response spectrum method, spectral velocity versus natural frequency for up to four damping constants is input. The output of dynamic analysis is in terms of nodal displacements, element stresses, and resultant forces and moments at specified time steps. When the modal superposition method is used, and in the case of earthquake response analysis, the requested numbers of frequencies and mode shapes are computed and printed together with the cumulative response of all the specified modes, as computed by the root sum square method and the absolute sum method.

DYNAX was developed under the acronym ASHAD by S. Ghosh and E. L. Wilson of the University of California, Berkeley, in 1969 (Reference 5). It was acquired by S&L in 1972 and is operating under EXEC 8 on a UNIVAC 1106.

To validate the major analytical capabilities of DYNAX, documented results from six problems are compared with DYNAX results. As shown in these six problems, DYNAX is capable of producing accurate results for both static and dynamic analyses of shells.

3.13.1.4.1 Problem l

The first problem is taken from S. Timoshenko, Theory of Plates and Shells (Reference 6). A clamped shallow spherical shell (Figure 3.13-15) is analyzed for displacements and stresses produced by a uniform pressure applied on its outside surface. DYNAX and Timoshenko's solutions are compared in Figures 3.13-16 and 3.13-17.

3.13.1.4.2 Problem 2

The second problem, taken from Theory of Elasticity by Timoshenko and Goodier (Reference 1) is a plane strain analysis of a thickwalled cylinder subjected to external pressure. The finite element idealization and the loading system used for this case are shown in Figure 3.13-18. Results of the DYNAX analysis are compared with the exact solution in Figure 3.13-19. The agreement for both stresses and displacements is excellent.

3.13.1.4.3 Problem 3

The third problem was presented in an article by Budiansky and Radkowski in an August 1963 issue of the AIAA Journal (Reference 7). The structure (Figure 3.13-20), is a short, wide cylinder with a moderate thickness-to-radius ratio. The applied loads and the output stresses are pure uncoupled harmonics. For this finite element analysis, the cylinder is divided into 50 elements of equal size. This problem checks the harmonic deflections, element stresses, and forces. Figure 3.13-21, Sheets 1 and 2, compares DYNAX results with the results given in the article.

3.13.1.4.4 Problem 4

The fourth problem is taken from an article by Reismann and Padlog (Reference 8). A ring (line) load of magnitude P (500 lb) is suddenly applied to the center of a freely supported cylindrical shell. The dimensions of the shell and the time history of load are shown in Figure 3.13-22. Because of symmetry, only one-half of the cylinder is modeled, using 80 elements of equal size. The time history of radial deflection and meridional moments from DYNAX and from Reismann and Padlog are compared in Figures 3.13-23 and 3.13-24 respectively.

3.13.1.4.5 Problem 5

For the fifth problem, the method of mode superposition is used to solve a shallow spherical cap with clamped support under the action of suddenly applied, uniformly distributed load. The dimensions of the shell and the load time history are shown in Figure 3.13-25. The first 12 modes were considered to formulate the uncoupled equations of motion. Each of these equations was solved by the step-by-step integration method using a time step of 0.1×10^{-4} sec. The results are compared with those obtained by S. Klein (Reference 9); see Figures 3.13-26 and 3.13-27.

3.13.1.4.6 Problem 6

The sixth problem is a hyperbolic cooling tower (Figure 3.13-28). The tower is analyzed for horizontal earthquake motion. A response spectrum for 2 percent damping (Figure 3.13-29) was used for this analysis. The root mean square values of the meridional force are compared with those obtained by Abel et al. (Reference 10) in Figure 3.13-30.

3.13.1.5 EASE - Elastic Analysis for Structural Engineering (Reactor/Auxiliary Building)

The Elastic Analysis for Structural Engineering (EASE) was developed by Engineering Analysis Corporation, Redondo Beach, California. The program is maintained by Control Data Corporation and is in the public domain. It performs static analysis of two- and three-dimensional trusses and frames, plane elastic bodies, and plate-and-shell structures. The finite element approach is used with the standard linear or beam elements, the plane stress triangular element, and a triangular plate bending element.

The program accepts temperature loads, as well as pressure, gravity, or concentrated loads. A plot feature of the input is available.

The program output includes joint displacements, beam forces, and triangular element stresses and moments.

3.13.1.6 INDIA - Interaction Diagram for Reinforced Concrete Members (Reactor/Auxiliary Building and Residual Heat Removal Complex)

INDIA (Load-Moment Interaction Diagram) is a program used to compute the coordinates and to plot the bending moment-axial load interaction diagram for a rectangular, reinforced-concrete section. The program will plot interaction curves for ultimate strength, yield strength, and working stress methods. Both compression and tension axial loads are considered, as well as positive and negative moments for appropriate cross sections.

The procedures used for the working stress and yield stress methods are taken from American Concrete Institute (ACI) 3l8 Code. Equations used for the ultimate stress method are taken from a University of Illinois civil engineering study. INDIA was originally developed at S&L on the IBM 1130 in 1971. It was converted to a UNIVAC 1106, where it has been successfully operating under EXEC 8 since 1972.

To demonstrate the validity of the program, a sample problem, shown in Table 3.13-3 and Figure 3.13-31, was executed. Calculations were made by hand, and all results were found to be consistent with the theoretical approach.

3.13.1.7 KALSHEL - Kalnins' Shell of Revolution (Reactor/Auxiliary Building)

Kalnins' Shell of Revolution (KALSHEL) is a computer program used to analyze thin axisymmetric shells of revolution for arbitrary load conditions. The solution is obtained by transforming the H. Reissner-Neisser equations to eight first-order ordinary differential equations. An Adams method of numerical integration is used as a basis for the solution of transformed equations. Since the program is based on classical shell theory, it has the same limitations.

The shell wall may vary in thickness along the meridian. It consists of up to four layers of different isotropic or orthotropic materials. Branch shells may be connected to the main shell. Surface loads and line loads in the radial, tangential, or meridional directions, meridional moments, and temperature distributions may be considered in the analysis. The temperature distributions are assumed to vary linearly across the thickness. All loads may be asymmetric.

The program output includes shell displacements in the radial, tangential, and meridional directions, meridional rotations, meridional moment, hoop moment, meridional force, hoop force, transverse-shear force, and twist-shear force. In addition, outer fiber stresses calculated from the stress resultants may be obtained.

The program was originally developed by A. Kalnins of Lehigh University (Reference ll). It was acquired by S&L in l969. This version was modified by S&L to sum displacements and stress resultants of the individual Fourier harmonics along meridians at specified angles. The program is currently maintained on the S&L UNIVAC 1106 operating under EXEC 8.

A number of test cases were run to check the program options and validity of solution. One of the practical problems included here is the analysis of conical shell subjected to eccentric line load. The shell is made of two parts, cylindrical and conical, and both are of reinforced concrete with different thicknesses as shown in Figure 3.13-32. The problem has been analyzed by this program and also by the public domain program SABOR III.

Results from the two programs are compared in Figures 3.13-33 through 3.13-36. Figures 3.13-33 and 3.13-34 show a comparison of shell forces along a meridian at 0° (symmetric with respect to the load). Figures 3.13-35 and 3.13-36 show a comparison of shell forces around the circumference at an elevation where the load is applied. As shown in these figures, the results compare favorably.

3.13.1.8 MASS IV - Matrix Analysis of Seismic Stresses (Reactor/Auxiliary Building)

Matrix Analysis of Seismic Stresses (MASS) IV is used for performing seismic analysis of plane and space trusses and frames and plane grids. Either the response spectrum method or the time-history method can be used, depending on the forcing function available. Both methods use the normal mode approach. In the case of time-history analysis, the decoupled differential equations of motion are integrated numerically using Newmark's β-method (Reference 2). Included in the program are input options allowing for member releases, input stiffness between two nodes, and rigid members. The program output includes

a. Stiffness

b. Mass and mass-stiffness triple product matrices

c. Modal periods

d. Eigenvectors and participation factors

e. Modal displacements

f. Member and joint forces

g. Probable and absolute maxima of displacements and forces.

The MASS program was developed by S&L in 1968. Version IV is currently maintained on a UNIVAC 1106 operating under EXEC 8. It has been used successfully since 1971. Two problems for validating the program are presented.

3.13.1.8.1 Problem 1

In the first problem, a three-story shear building is analyzed and compared to a solution obtained by Biggs (Reference 3). The structure is represented by the closed-coupled system shown in Figure 3.13-11. The masses and stiffness values used are also shown. For the analysis, the following response spectrum was used:

Frequency (Hz)	Displacement (in.)
1.00	3.30
2.18	1.40
3.18	0.66

The results obtained by Biggs and from MASS IV are compared in Table 3.13-1.

3.13.1.8.2 Problem 2

In the second problem, results of MASS IV are compared to those obtained by Wilson (Reference 4) using the SAP IV program. At the fixed end of a cantilever beam, an acceleration is applied (Figure 3.13-37). The natural periods calculated by both SAP IV and MASS IV are shown in Table 3.13-2. A comparison of the bending moment at the fixed end of the cantilever beam is shown in Figure 3.13-38. As demonstrated in both examples, MASS IV performs an accurate analysis.

3.13.1.9 PLFEM II - Plate Finite Element Method (Reactor/Auxiliary Building)

Plate Finite Element Method (PLFEM II) is used to analyze plane elastic bodies, plates, and shell structures by the stiffness matrix method. The program uses two finite elements, rectangular and triangular.

Elastic spring supports or an elastic foundation may be considered in the analysis. Orthotropic materials may also be considered in conjunction with the rectangular element. Pressure loads, concentrated forces, nodal displacements, and temperature loads may be considered in the analysis. All loading cases may be factored or combined in any manner.

The program output includes deflections and rotations of all joints and membrane stresses (normal, shearing, and principal) at the center of each element; the resultant moments (x, y, twisting, and principal); and shears and reaction forces. An equilibrium check is made to determine the accuracy of the results.

PLFEM II, developed on a UNIVAC 1108 in 1966, is maintained by S&L. Since May 1972, it has been operating successfully on the S&L UNIVAC 1106 under EXEC 8.

Three sample problems are presented to demonstrate the validity of PLFEM. Plots of the computer results obtained are compared with theoretical results and results obtained by other methods.

3.13.1.9.1 Problem 1

The first problem is an analysis of a rectangular tank filled with water, which was presented by Y. K. Cheung and J. D. Davies in an article in May 1967 (Reference 12). The finite element used was presented by Zienkiewicz and Cheung in the Proceedings of the Institute of Civil Engineers in August 1964 (Reference 13). Experimental results obtained agreed exactly with the finite element results except at a few isolated points where very small differences were noted. The PLFEM grid and loading for the tank problem are shown in Figure 3.13-39. The grid used is the same size as that used by Cheung and Davies. Moments in three regions of the tank are plotted along with the PLFEM results in Figures 3.13-40 through 3.13-42.

3.13.1.9.2 Problem 2

In the second analysis, a rectangular plate with a circular hole in its center is subjected to a uniform plane stress. The grid used in the PLFEM analysis is shown in Figure 3.13-43. Because of double symmetry, only one-quarter of the plate is analyzed. Results obtained from the PLFEM analysis are plotted in Figure 3.13-44 against the exact values as given by Timoshenko and Goodier in Reference 1.

3.13.1.9.3 Problem 3

In the third problem, a square plate having a rectangular hole in its center is analyzed for the effect of a temperature gradient through the plate. The grid used in the PLFEM analysis is shown in Figure 3.13-45. Only one-quarter of the plate is analyzed because of double symmetry. Moment values obtained by PLFEM are plotted for two regions of the plate in

Figure 3.13-46. For comparison, values of the moments obtained by an analysis based on the Hrennekoff framework analogy are also shown.

3.13.1.10 SLSAP - Sargent & Lundy Structural Analysis Program (Reactor/Auxiliary Building and Residual Heat Removal Complex)

The S&L Structural Analysis Program (SLSAP) was developed by E. Wilson of the University of California at Berkeley. It is maintained by S&L. The program uses the stiffness matrix method to analyze two- and three-dimensional frames, trusses, and grids; three-dimensional elastic solids; and axially symmetric solids, plates, and shells, for arbitrary static loads. Dynamic analyses for frequencies and mode shapes, spectral analysis, and numerical integration analyses are also possible.

The program allows materials with arbitrary elastic constants, combined loadings, rigid members, elastic supports, and a combination of different element types.

Included in the program output are displacement and rotations of all joints, nodes, forces, or stresses in members or elements; frequencies and mode shapes; and dynamic response in terms of displacements and forces.

The original version of SLSAP dates back to 1968. S&L currently maintains the SLSAP IV version. The program can successfully operate on either a UNIVAC 1106 or a CDC 6600 computer. It is primarily used for static analysis. Results from the program have been compared with several other static and dynamic computer programs and classical solutions. Two examples of these validation problems are presented.

3.13.1.10.1 Problem 1

The first problem is a cantilever beam under both uniform and concentrated load (the beam was modeled for SLSAP using 10 equal-length beam elements). It has a cross-sectional area of 1 x 2 in., length 10 in., and a Young's modulus of 30 x 10^3 ksi. A uniform load q = 2 kips/in. and a concentrated load of 10 kips at one end of the beam are applied. The results from the program are compared to analytical results obtained by Timoshenko and Gere. Figure 3.13-47 shows excellent agreement for the bending moment obtained in both solutions.

3.13.1.10.2 Problem 2

In the second problem, a simply supported square plate under uniform loading is analyzed. A 10-in.-square by 1-in.-thick square plate with Poisson's ratio = 0.3 and Young's modulus = 30 x 10^3 ksi, was loaded with 1-ksi pressure. The results obtained were compared to those presented by S. Timoshenko and S. Woinowsky-Krieger. The bending moments M_{xx} and M_{yy} for both the x and y symmetry lines obtained in the two solutions are shown in Figure 3.13-48. The maximum bending moment that occurs at the center of the plate differs by only 1.05 percent.

3.13.1.11 SOR III - Shell of Revolution (Reactor/Auxiliary Building)

The Shell of Revolution (SOR III) was developed by Knolls Atomic Power Laboratory for the AEC. It is maintained by S&L. This program analyzes thin shells of revolution subjected to axisymmetric loading by numerically integrating the governing differential equations, using a generalized Adams-Moulton method.

Arbitrary distribution of normal, tangential, and moment surface loadings, as well as edge forces and deflections, may be considered in the axisymmetric loadings. Input of boundary conditions allows the consideration of elastic support conditions. The effect of temperature variations along the meridian or across the thickness also is considered.

The program output includes shell displacements, outer fiber stresses and strains, and stress resultants. Version III was acquired by S&L in 1969 and is currently maintained on S&L's UNIVAC 1106 computer. The S&L version has been modified to punch data for plotting.

Results from this program have been frequently compared with other available solutions and other computer programs to test the validity of the program. One of these comparisons is the analysis of a circular, flat, reinforced-concrete plate. The details of the problem and the boundary conditions are shown in Figure 3.13-49. Results of the SOR III analysis were compared with the finite element program, SABOR III. Figure 3.13-50 shows the bending moment in the meridional and hoop directions. Figure 3.13-51 shows the comparison of radial shear. As shown in these figures, results compare favorably.

3.13.1.12 SSANA - Spring-Slab Analysis (Reactor/Auxiliary Building and Residual Heat Removal Complex)

Spring-Slab Analysis (SSANA) was written to facilitate the reduction of a reinforced-concrete shear building and the equipment in the building to a system of rigid slabs interconnected by weightless linear springs. The program calculates the centroid, total weight, and the weight moment of inertia about the vertical and two horizontal centroidal axes of each slab. The program also calculates the spring stiffness of concrete walls and its distance from the mass centroid.

Spring constants of shear walls are computed based on the following equation:

$$K = \frac{1}{DT}$$

where

DT = DF + DS

DF = Flexural deflection/unit load

DS = Shear deflection/unit load

SSANA was written and is maintained by S&L. It currently operates on a UNIVAC 1106 operating under EXEC 8.

Hand calculations were used to validate the program. As an example of this validation, stiffness and rotary mass were calculated for elements of the structure shown in Figure 3.13-

52. A comparison of results from SSANA and hand calculations shown in Tables 3.13-4 and 3.13-5 demonstrates the accuracy of the program.

3.13.1.13 STRESS-II - Structural Engineering Systems Solver (Reactor/Auxiliary Building)

The Structural Engineering Systems Solver (STRESS-II) was developed by the Massachusetts Institute of Technology. It is maintained by the University Computing Company and is in the public domain. The program uses the stiffness matrix method to analyze plane and space trusses and frames, and plane grids.

The structure can be analyzed for arbitrary joint loads, member loads, temperature changes, and joint displacements. A plotting feature is available with the program. The output includes joint displacements, equilibrium checks and reactions, and member forces.

The version currently used by S&L was adapted to the UNIVAC 1100 Series computer by the Chi Corporation, Cleveland, Ohio, which has maintained it since 1972.

3.13.1.14 STRUDL II - Structural Design Language (Reactor/ Auxiliary Building and Residual Heat Removal Complex)

The Structural Design Language (STRUDL II) was developed by the Massachusetts Institute of Technology. It is maintained by McDonnell Douglas Automation Company. Linear, static, or dynamic analyses may be performed for finite element representations of structures using stiffness matrix methods. Nonlinear static problems and stability problems also may be treated.

The program is capable of analyzing plane trusses and frames, grids and elastic bodies, space trusses and frames, or three- dimensional elastic solids subjected to arbitrary loads, temperature changes, or specified displacements. Either earthquake accelerations or time-history force may be used. In addition to analysis, the program is capable of doing structural steel design according to the American Iron and Steel Institute (AISI) Code, and reinforced or prestressed concrete design according to the ACI Code.

The program output depends on the type of finite element used and the analysis that was performed. Included in the output are displacements and member forces and moments, or element stresses and moments. Eigenvalues, eigenvectors, and time-history response or nodal response may be obtained for dynamic analyses. Member sizes may be obtained if the design portion is used.

This program has been in the public domain since 1968. Two versions are currently being used: one is maintained by the McDonnell Douglas Automation Company on IBM 370 Series hardware, and one is maintained by UNIVAC on the 1100 Series hardware.

3.13.1.15 TEMCO III - Reinforced Concrete Sections Under Eccentric Loads and Thermal Gradients (Reactor/ Auxiliary Building and Residual Heat Removal Complex)

TEMCO analyzes reinforced-concrete sections subject to separate or combined action of eccentric loads and thermal gradients. The effect of temperature is induced in the section by reactions created by the curvature restraint.

The analysis may be done assuming either a cracked or an uncracked section. Material properties can be assumed to be either linear or nonlinear. The program is capable of handling rectangular as well as nonrectangular sections. The program input consists of section dimensions, areas and location of each layer of reinforcing steel, loads, load combinations, and material properties.

The curvature and axial strain corresponding to the given eccentric loads (axial load and bending moment) are determined by an iterative procedure. Thermal gradient is applied on the section by inducing reactions created by the curvature restraint, i.e., there is no curvature change due to a thermal gradient on the section. The axial expansion is assumed to be free after thermal gradient is applied. An iterative procedure is again used for finding the final strain distribution in which equilibrium of internal and external loads is satisfied.

The program output consists of the echo of input, combined loads, final location of neutral axis, final stresses in steel and concrete, and final internal forces. Similar intermediate results (before thermal gradient is applied) can also be output if desired. The program has applications to a wide variety of reinforced-concrete beams and columns, slabs, and containment structures subject to various combinations of external loads and thermal gradients. The program was developed and is maintained by S&L. Since February 1972, the program has been extensively used at S&L on UNIVAC 1106 hardware operating under EXEC 8.

To demonstrate the validity of TEMCO, program results are compared with hand-calculated results. Three example problems are considered. The section and material properties for each problem are given in Table 3.13-6, along with the applied external forces and thermal gradients.

The first problem considered involves a section with two layers of steel under the action of a compressive force applied at the centerline of the section, a bending moment, and a thermal gradient. A cracked analysis of the section is required, assuming nonlinear material properties.

The second problem considered involves a section with two layers of steel under the action of a tensile force applied at the centerline of the section, a bending moment, and a thermal gradient. A cracked analysis of the section is required, assuming nonlinear material properties.

The third problem considered involves a section with two layers of steel under the action of a tensile force applied at the centerline of the section, a bending moment, and a thermal gradient. A cracked analysis of the section is required, assuming linear material properties. The hand-calculated solution was obtained according to the following outlined procedure:

a. Assume the location of neutral axis and the stress distribution to be the same as those given by the program under the given mechanical loading

b. Compute the strain distribution under the given mechanical loading

c. Compute the stress resultants by integration and using the proper stress-strain relationships

d. Check for equilibrium with external mechanical loads

e. If equilibrium is satisfied, compute the curvature imposed on the section by the given thermal gradient

f. Compute the final curvature by subtracting the thermal curvature from the mechanical curvature

g. Compute the new axial strain so that equilibrium is satisfied, keeping the curvature constant

h. Compute the final stress resultants by integration, using the proper stress-strain relationships

i. Compute the thermal moment

j. Check for equilibrium and compare program results with hand-calculated results.

Results obtained using this procedure, together with those computed by TEMCO, are presented in Table 3.13-7. It is concluded that results given by the program agree very well with results obtained by hand calculations, and that equilibrium between internal and external forces is satisfied for all three problems.

3.13.1.16 CAPAN - Cable Pan Analysis (Reactor/Auxiliary Building and Residual Heat Removal Complex)

Cable Pan Analysis (CAPAN) is a computer program used in the analysis of continuous cable pan systems. Section properties of cable pans and allowable stresses for bending about both axes for both seismic and nonseismic conditions are computed. Given a pair of response spectra for any slab, allowable pan support spacing can also be computed.

Peak accelerations (both horizontal and vertical) are used in computing the moments due to seismic loads. The allowable spacing is computed for dead load, dead load plus live load, and dead load plus earthquake. The minimum value is chosen as allowable pan spacing.

CAPAN was developed by S&L in 1972. It is currently maintained on a UNIVAC 1106 operating under EXEC 8.

As an example of validation, the support spacing of a pan 12 in. wide, 4 in. deep, and 14 gage thick was analyzed. CAPAN results and those from hand calculations were compared. As shown in Table 3.13-8, the results are in good agreement.

3.13.1.17 MVI - Matrix Analysis for Seismic Stresses Input Generator (Reactor/Auxiliary Building and Residual Heat Removal Complex)

Mass V Input Generator (MVI) is a computer program that generates data on cable pan hangers. These data are stored on magnetic tape and later used as input to S&L program MASS V (Matrix Analysis for Seismic Stresses) to perform seismic analysis. It is written in Fortran V language and represents the first step of Method l in the design of cable pan hanger systems that support Category I cables.

For a given width and height of hanger, number of levels, and member properties, MVI generates frame geometry and other necessary data in a format acceptable to the MASS V program. Each hanger is loaded with unit mass per level, and subjected to unit horizontal acceleration. Since this is an input-generation program for MASS V, the program is validated by checking generated input for MASS V.

3.13.1.18 ELHAN - Elastic Hanger Analysis (Reactor/Auxiliary Building and Residual Heat Removal Complex)

Elastic Hanger Analysis (ELHAN) is a postprocessor program used in the design of hangers to support Category I cables. It represents Step 3 in the design of Type l cable pan hangers. The hangers described in the program MVI are designed by using ELHAN. The various functions performed by ELHAN are as follows.

a. Reads variations of dead load and horizontal and vertical response spectra for all slabs. The response spectra should be obtained from the project seismic report. For horizontal excitation, the response spectra used is the envelope of maximum response due to north/ south and east/west

b. Reads results of MASS V program from tape as described in Step 2

c. Forces and deflections obtained as above are modified to represent actual load on the hangers and actual floor acceleration (results are stored for unit load and unit acceleration)

d. Computes forces induced due to dead load and vertical excitation

e. Combines and checks stresses.

The program has an option to design vertical members with or without compression criteria. If compression criteria are used, the members with (KL/r) ratios greater than 200 are omitted from the tabulation. The program also has an option to print the data stored on the tape for checking purposes. The program was checked against hand calculations. In all cases the program correctly selected all failing members.

3.13.1.19 RIGHAN - Rigid Hanger Analysis (Reactor/Auxiliary Building and Residual Heat Removal Complex)

Rigid Hanger Analysis (RIGHAN) is a program used for the analysis and design of laterally supported cable pan hangers. Input to the program consists of variations of dead load and hanger widths, member properties, and a set of horizontal and vertical response spectra for each slab. The program is used to compute stresses due to the combined dead load and horizontal and vertical excitations.

Rigid Hanger Analysis was developed by S&L in 1972. It is currently maintained on a UNIVAC 1106 operating under EXEC 8.

A typical hanger used for validation is shown in Figure 3.13-53. Results of the program were compared with hand calculations. As shown in Table 3.13-9, RIGHAN correctly analyzes and designs rigid hangers.

3.13.1.20 MASS V - Matrix Analysis for Seismic Stresses (Reactor/Auxiliary Building and Residual Heat Removal Complex)

Matrix Analysis for Seismic Stresses (MASS V) is used to perform seismic analysis of plane and space trusses and frames and plane grids. Either the response spectrum method or the time-history method can be used, depending on the forcing function available. Both methods use the normal mode approach. In the case of time-history analysis, the decoupled differential equations of motion are integrated numerically using Newmark's β-method (Reference 2).

Included in the program are input options allowing for member releases, input stiffness between two nodes, and rigid members. The program output includes

a. Stiffness

b. Mass and mass-stiffness triple product matrices

c. Modal periods

d. Eigenvectors and participation factors

e. Modal displacements

f. Member and joint forces

g. Probable and absolute maxima of displacements and forces.

The MASS program was developed by S&L in 1968. Version V is currently maintained on a UNIVAC 1106 operating under EXEC 8. It has been used successfully since 1972. Two examples of the problems for validating the program are presented.

3.13.1.20.1 Problem 1

In the first example, a three-story shear building is analyzed and compared to a solution obtained by Biggs (Reference 3). The structure is represented by the closed-coupled system shown in Figure 3.13-11. The masses and stiffness values used are also given in Figure 3.13-11. For the analysis, the following response spectrum was used:

Frequency (Hz)	Displacement (in.)
1.00	3.30
2.18	1.40
3.18	0.66

The results obtained by Biggs and from MASS V are compared in Table 3.13-1.

3.13.1.20.2 Problem 2

In the second example, results of MASS V are compared to those obtained by Wilson et al. (Reference 14) using the SAP IV program.

At the fixed end of a cantilever beam (Figure 3.13-12), an acceleration is applied (Figure 3.13-13). The natural periods calculated by both SAP IV and MASS V are shown in Table 3.13-2. A comparison of the bending moment at the fixed end of the cantilever beam is shown in Figure 3.13-54. As demonstrated in both examples, MASS V performs an accurate analysis.

3.13.1.21 PCAUC - Portland Cement Association, Ultimate Strength Design of Reinforced Concrete Columns (Residual Heat Removal Complex)

Portland Cement Association, Ultimate Strength Design of Reinforced Concrete Columns (PCAUC) is used to design or to investigate reinforced-concrete columns using the ultimate strength theory in accordance with ACI 318-71 Code. The program is capable of designing or investigating tied columns subjected to an axial load combined with uniaxial or biaxial bending moments. The program input consists of the dimensions of sections, material properties, reinforcement requirements, and loading data. The slenderness effect is not included in the present program.

Output from the design part of the program includes the steel reinforcement arrangement, ultimate capacity for all loading cases, and interaction control points data. Output from the investigation part of the program includes either biaxial or uniaxial interaction data. Sargent & Lundy has modified the original PCA program to follow the 1971 ACI building code and to provide more design options and greater capacity.

PCAUC is a modified version of the program "Ultimate Strength Design of Concrete Columns," developed by the Portland Cement Association. The program was obtained by S&L in 1972 and modified. It is currently maintained on the UNIVAC 1106 operating under EXEC 8.

To validate PCAUC, documented results from several problems were compared with PCAUC results. Three of these problems are presented herein.

3.13.1.21.1 Problem 1

The first problem is taken from Wang and Salmon, Reinforced Concrete Design (Reference 15). The reinforcement for a 17 x 17-in. square tied column is designed for compression control loads. The loads include a deadload axial load of 214 kips and bending moment of 47 ft-kips, and a liveload axial load of 132 kips and a bending moment of 23 ft-kips. The reinforcement is designed according to the ACI Code with $f_c' = 3000$ lb/in.^{2} and $f_y = 40{,}000$ lb/in.^{2}

The solution as given in Reference 15 is identical to the solution obtained from PCAUC, shown in Figure 3.13-55. It should be noted that the ultimate capacity provided by PCAUC has been reduced by a factor of 0.7.

3.13.1.21.2 Problem 2

The second problem is also taken from Reference 15. The reinforcement for a tied column 14 in. wide and 20 in. deep is designed for tension control loads with a deadload axial load of 43 kips and bending moment of 96 ft-kips, and a liveload axial load of 32 kips and bending moment of 85 ft-kips. The reinforcement is designed according to ACI Code using symmetrical reinforcement with respect to its width, and with $f_c' = 4500$ lb/in.^{2} and $f_y = 50{,}000$ lb/in.^{2}. The solution given in Reference l5 is identical to the solution obtained from PCAUC, shown in Figure 3.13-56.

3.13.1.21.3 Problem 3

The third problem is taken from Notes on ACI 318-71 Building Code Requirements With Design Applications, by the Portland Cement Association (Reference 16). A square tied column 28 in. x 28 in. is designed for biaxial bending loads for the following service loads.

Service Load	Dead Load	Live Load
Axial	550 kips	300 kips
M_x	320 ft-kips	200 ft-kips
M_y	160 ft-kips	100 ft-kips

The bending is designed according to the ACI Code with $f_c' = 5{,}000$ lb/in^2 and $f_y = 60{,}000$ lb/in.^{2}.

The selected reinforcement obtained from PCAUC, shown in Figure 3.13-57, is identical to that from Reference 16. It should also be noted that the interaction control points obtained by both show good agreement.

3.13.1.22 STAND - Structural Analysis and Design (Reactor/ Auxiliary Building)

Structural Analysis and Design (STAND) is an integrated system programmed to perform analysis and design of structural steel members according to the 1969 AISC Specification. It consists of the following subsystems:

a. Beam edit
b. Rolled beam design
c. Composite beam design
d. Plate girder design
e. Column edit
f. Column design
g. Column baseplate design.

The program input consists of member geometry and basic loadings. The design is performed for specified combinations of basic loadings and overstress factors. For floor framing systems, the program is capable of automatically transferring reactions from tributary beams to supporting members. There are many design control parameters available, such as minimum and maximum depth limitations, shape of the rolled section, location of the lateral support of the compression flange, material grade of yield stress, deflection limitations, flange cutoff criterion, and location of stiffeners.

For columns, the program can be used to account for axial loading as well as uniaxial or biaxial bending. For column baseplate design, only axial load and column combinations are considered.

The program output includes the complete final design and provides the designer with sufficient intermediate information to enable him to evaluate the results. For rolled and composite beam designs, complete details of shop-welded and field-bolted end connections are contained in the output. Supplementary information for economic evaluation of the design is also provided.

STAND was developed and is maintained by S&L. Since May 1972, the program has been extensively used at S&L on UNIVAC 1106 hardware operating under EXEC 8. Some of the principal applications include the design of steel floor framing using various types of horizontal structural elements, and the design of columns or beam columns.

To validate STAND, results from the program were compared with results from example design problems in the Manual of Steel Construction (Reference 17). Four problems are given herein.

3.13.1.22.1 Problem 1

The first problem is a rolled beam design problem (Example 1, pp. 2-4, 5). A beam of 36 ksi steel is designed for a 125 kip-ft angling moment, assuming its compression flange is braced at 6.0-ft intervals. The results, listed in Table 3.13-10, show that STAND selects a more efficient section.

3.13.1.22.2 Problem 2

The second problem is a composite beam design problem (Example 1, pp. 2-143, 144). A non-coverplated composite interior floor beam is designed. Limits of 1.5 in. for deadload deflection and 1.2 in. for liveload deflection are imposed. The results, shown in Table 3.13-11, are nearly identical.

3.13.1.22.3 Problem 3

The third problem is a column design problem with three examples, (Examples 1, 2, and 5, pp. 3-4, 5, 9). The first is the design of a W12 column of 36-ksi steel that will support a concentric load of 670 kips. The effective length with respect to its minor axis is 16 ft, and to its major axis, 31 ft.

The second is the design of an 11-ft-long W12 interior bay column of 36-ksi steel that will support a concentric load of 540 kips. The column, rigidly framed at the top by 30-ft-long, W30 x 116 girders connected to each flange, is braced normal to its web at the top and the base.

The third is the design of a Wl4 column of 36-ksi steel for a tier building, 18-ft story height, that will support a 600-kip gravity load and a 190-kip-ft maximum wind moment, assuming K = l relative to both axes and bending is about the major axis.

The results from all three checks are identical to those in the AISC Manual, and are shown in Table 3.13-12.

3.13.1.22.4 Problem 4

The fourth problem is a plate girder design problem (Example 1, p. 2-108). A welded plate girder is designed to support a uniform load of 3 kips/ft and two concentrated loads of 70 kips as shown in Figure 3.13-58. The compression flange of the girder is laterally supported only at points of concentrated load. The close results are shown in Table 3.13-13.

3.13.1.23 PLGIRD - Plate Girder Design (Reactor/Auxiliary Building)

Plate Girder Design (PLGIRD) is used to design welded plate girders according to specified loadings and geometries. The design criteria are in accordance with the AISC Specification for the Design of Structural Steel for Buildings, 1969. The program can automatically account for variations in steel stress according to the material thickness for seven types of structural steel. The program also takes into account the variation of the weight of the girder along the span due to flange cutoffs. Input to the program consists of the specified minimum or maximum web depth, maximum flange width, maximum plate thickness, and vertical loadings.

PLGIRD was developed by S&L in 1967. It is currently maintained on a UNIVAC 1106 operating under EXEC 8.

To validate PLGIRD, results from the program were compared with results from example design problems in the Manual of Steel Construction (Reference 17). One of these problems is given.

A welded plate girder (Example 1, p. 2-108) is designed to support a uniform load of 3 kips/ft and two concentrated loads of 70 kips, as shown in Figure 3.13-58. The compression flange of the girder is laterally supported only at points of concentrated load. The close results are shown in Table 3.13-13.

3.13.1.24 MESHG (Reactor/Auxiliary Building and Residual Heat Removal Complex)

The program MESHG is written for the UNIVAC 1106/130k machine. It is used as a preprocessor for finite element programs that are currently available. Its main function is to check the geometry of the input by plotting the mesh. Options are available allowing the user to scale the plot, number elements and/or nodes, draw different isometric views of three-dimensional data, rotate axes for two-dimensional data, and plot a vector field. This program is repeatedly verified by inspection of each plotted mesh.

3.13.1.25 COGO (Reactor/Auxiliary Building and Residual Heat Removal Complex)

COGO is a problem-oriented computer language and programming system for solving geometric problems in civil engineering on a digital computer. Each problem is solved by writing a "COGO program," consisting of a series of commands that describe the operations to be performed in order to effect a solution. Data needed to perform each operation are included as part of the command. COGO was developed at the Massachusetts Institute of Technology, and is in the public domain.

3.13.1.26 PIPSYS (Reactor/Auxiliary Building and Residual Heat Removal Complex)

Integrated Piping Analysis System (PIPSYS) is used to analyze piping systems of power plants for static and dynamic loadings, and to compute the combined stresses. The following analyses are performed:

a. Static - analysis of thermal, displacement, distributed, and concentrated weight loadings on piping systems

b. Dynamic - analysis of piping system response to seismic and fluid transient loads

c. Stress combination - computation of the combined stresses in the piping components in accordance with the ASME Boiler and Pressure Vessel (B&PV) Code Section III (Reference 18).

The static, dynamic, and stress combination analyses can be performed independently or in sequence. Results of the static and dynamic analyses can be stored on magnetic tape for use at a later date to perform the stress combination analysis. The piping configuration can be plotted on a Calcomp plotter.

The input consists of the piping system geometry, material properties, and static and dynamic loadings. Various options exist to control the length of the output. The default option generally prints only the summary of input data and final results.

PIPSYS was developed by S&L in 1972. It is currently maintained on a UNIVAC 1106 operating under EXEC 8. To demonstrate the validity of the PIPSYS program, three problems are presented.

3.13.1.26.1 Problem 1

To illustrate the validity of the static portion of PIPSYS, the problem shown in Figure 3.13-59 was analyzed and the results compared to those given in Reference 19. Table 3.13-14 shows the comparison of member end moments. As shown, the results from PIPSYS and Reference 19 are in good agreement.

3.13.1.26.2 Problem 2

To illustrate the validity of the stress combination analysis portion of PIPSYS, the problem outlined in Reference 20 was reanalyzed on the PIPSYS program. The layout of the piping system is shown in Figure 201 of Reference 21. The stress analysis is performed at location 19. The summary of load sets and descriptions is presented in Table 3.13-15. The results of the stress analysis are presented in Tables 3.13-16 and 3.13-17. The notations and equation numbers correspond to the ASME B&PV Code (Reference 18). The PIPSYS results are in very close agreement with those presented in Reference 20.

3.13.1.26.3 Problem 3

To illustrate the validity of the dynamic analysis portion of PIPSYS, a problem was analyzed and the results obtained from PIPSYS were compared with those from two public domain computer programs. These are DYNAL (Reference 22) and NASTRAN (References 23 and 24).

Figure 3.13-60 shows a schematic representation of the piping system analyzed. The system is modeled with simple beam elements with a total of 136 degrees of freedom. Figure 3.13-61 shows the time-dependent blowdown forces at the relief valve locations. Results of PIPSYS are compared with DYNAL and NASTRAN in Table 3.13-18 and in Figure 3.13-62. The results from all three programs are in quite close agreement.

3.13.1.27 NOHEAT

Sargent & Lundy's NOHEAT computer program is used to determine the code quantities $(\Delta T_1)_I$, $(\Delta T_2)_I$, and $(\alpha ATA\text{-}\alpha BTB)_I$, resulting from fast fluid temperature changes. It can be used for piping products and dissimilar metal joints.

Each fitting is modeled using an axisymmetric solid finite element mesh, with a time-dependent forced-convection heat flow boundary condition forced on the inside surface of the pipe. The meshes are internally generated using user-supplied dimensions. The forced-convection boundary conditions are determined using the design thermal transient definitions associated with Envelope Load Set. The following procedure is then used to determine the time-dependent (ΔT_1), (ΔT_2), and $(\alpha ATA\text{-}\alpha BTB)$ quantities:

a. The time-dependent temperature distribution on the fitting is determined by direct time integration of the finite element heat conduction equations

b. Integrations of the temperature distribution at every instant in time are then performed using the code definitions in NB-3653 to determine the quantities (ΔT_1), (ΔT_2), $(\alpha ATA\text{-}\alpha BTB)$.

3.13.1.28 AXTRAN

Sargent & Lundy's AXTRAN computer program is used to determine the code quantity $(\alpha ATA\text{-}\alpha BTB)_I$ resulting from axial temperature distribution along stagnant lines.

The program models the piping as an infinitely long cooling fin. Heat loss from the fin is governed by insulation characteristics that are obtained from the insulation vendor. Input to the program consists of a temperature-versus-time description of the thermal transient to be forced on the model, and a physical description of the model. Output from the program consists of a time history of the temperature distribution along the pipe and a time history of the code nominal stress $Eab(\alpha ATA\text{-}\alpha BTB)_I$.

3.13.1.29 SEISHANG

3.13.1.29.1 SEISHANG (Version 3) (Reactor/Auxiliary Building and Residual Heat Removal Complex)

SEISHANG (Seismic Analysis of Hangers) is used for the analysis and design of electrical cable and heating, ventilation and air conditioning (HVAC) duct support systems. The program computes the allowable spans for cable trays and selects the proper member sections for various types of supports. The input load functions can be in the form of dead load, live load, or dynamic response spectra.

Program input consists of geometric data, material properties, member properties, and external loadings. Program output consists of allowable spans, member sizes, and mechanical response.

SEISHANG was developed at S&L in 1976. It is currently maintained on UNIVAC 1100 Series hardware under EXEC 8.

To demonstrate the validity of the program, two problems are presented.

3.13.1.29.1.1 Problem 1

A typical cable tray, shown in Figure 3.13-63, is analyzed and compared to the solution obtained by hand calculation. The results obtained from SEISHANG and by hand calculation are compared in Table 3.13-19. The results show good agreement.

3.13.1.29.1.2 Problem 2

Two typical HVAC supports, shown in Figures 3.13-64 and 3.13-65, are analyzed and compared to the solution obtained from the DYNAS (09.7.090-9.0) computer program, Reference 25. The results obtained from SEISHANG and from DYNAS are compared in Tables 3.13-20 and 3.13-21. The HVAC support shown in Figure 3.13-64 is also analyzed by the PIPSYS (09.5.065-3.4) computer program, Reference 26. The results obtained from SEISHANG and from PIPSYS are compared in Table 3.13-22. The results show good agreement.

3.13.1.29.2 SEISHANG (Version 4)

SEISHANG (Seismic Analysis of Hangers) is used for the analysis and design of electrical cable and HVAC duct support systems. The program computes the allowable spans for cable trays and selects the proper member sections for various types of supports. The input load functions can be in the form of dead load, live load, or dynamic response spectra.

Program input consists of geometric data, material properties, member properties, and external loadings. Program output consists of allowable spans, member sizes, and mechanical response.

SEISHANG was developed at S&L in 1976. It is currently maintained on UNIVAC 1100 Series hardware under EXEC 8.

To demonstrate the validity of the program, two problems are presented.

3.13.1.29.2.1 Problem 1

A typical cable tray, shown in Figure 3.13-63, is analyzed and compared to the solution obtained by hand calculation. The results obtained from SEISHANG and by hand calculation are compared in Table 3.13-19. The results show good agreement.

3.13.1.29.2.2 Problem 2

A typical cable tray support, shown in Figure 3.13-66, is analyzed and compared to the solution obtained from the PIPSYS (09.5.065-6.1) computer program, Reference 27. The analysis results obtained from SEISHANG and from PIPSYS are compared in Table 3.13-23. For member stress calculation, the results obtained from SEISHANG and from hand calculations are compared in Table 3.13-24. The results show good agreement.

3.13.1.30 SUPS (Reactor/Auxiliary Building and Residual Heat Removal Complex)

The SUPS program includes the PFRAME, CONNECTIONS, CINCH, and APLAN modules discussed below.

3.13.1.30.1 PFRAME

PFRAME (Interactive Plane Frame Analysis) is an interactive program that analyzes two-dimensional frames for static loads, using the stiffness approach. Joint movements are allowed only three degrees of freedom. Members are considered as prismatic beam elements. Loads are defined as global joint forces or member end forces.

Input consists of joint coordinates, fixities, member incidence, material/section properties, and joint/member forces. All input is free format and prompted by the program.

Output consists of joint displacements, rotations, support reactions, member end forces, and moments.

PFRAME is a module of the SUPS package and was developed at S&L in 1982. It is maintained on UNIVAC 1100 Series hardware operating under EXEC 8.

The program's validity is demonstrated by two problems. The first is a continuous beam problem shown in Figure 3.13-67. Table 3.13-25 compares PFRAME's results with the solution shown in Beer and Johnston, Reference 28. The second problem is the frame shown in Figure 3.13-68. Table 3.13-26 compares PFRAME's results with the solution shown in Gere and Weaver, Reference 29. The tables show good comparison.

3.13.1.30.2 CONNECTIONS

CONNECTIONS (Connections Investigation Program) aids in checking the adequacy of connections. The program checks the design of sliding- and friction-type Framed Beam Connections.

Design procedures used are given in Structural Design Standard E7 (Reference 30) and conform to the requirements of the AISC Manual of Steel Construction, 1978 Edition (Reference 31). Criteria for reassessing connections on nuclear projects are discussed in Reference 32. The program is interactive with self-documenting input, and prints out a summary of results. References 33 and 34 are applicable for general information.

The program is a module of the SUPS package and was developed at S&L in 1982. It is currently maintained on UNIVAC 1100 Series hardware operating under EXEC 8.

Two problems have been selected to validate the program for each type of connection except for Connection Detail No. 7.7.5 for which three problems are used. Results obtained from the program were compared with hand-calculation results; the results are identical.

The following examples were selected to validate the program for bolted/bolted sliding-type connections (Detail No. 7.7.1 of Reference 30).

3.13.1.30.2.1 Problem 1

Example 1a - Uncoped Connection (L2 = E3)

General criteria with operating-basis earthquake (OBE) load case. W10x39 beam (A36) and two 7 x 4 x 3/8 x 6 angles (A36) with 7/8-in.-diameter bolts (A325) (two bolts with pitch equal to 3 in. on instanding leg, four bolts in two rows with pitch equal to 3 in. on outstanding leg). The connection is subjected to a vertical load (R_y = 3.83K), a lateral load (R_x = 1.98K), and a torsional moment (M_z = 4.02K-in.).

The dimensions for connection angles are as follows:

L1	=	6.0 in.	E2	=	7.0 in.
L3	=	0.0 in.	E3	=	3.0 in.
OL	=	4.0 in.	E1	=	1.5 in.
			S	=	0.375 in.

On the outstanding angle, the gage is equal to 5.5 in. and the surface condition is Type A.

Results obtained by hand calculation and program analysis are compared in Table 3.13-27.

Example 1b - Coped Connection

All the parameters are the same as in Example 1a, except the coped distances on the top flange are L2 = 5.5 in. and L3 = 1.25 in. Results obtained by hand calculation and program analysis are compared in Table 3.13-28.

3.13.1.30.2.2 Problem 2

The following examples were selected to validate the program for bolted/welded sliding type connection with side plates (Detail No. 7.7.2 of Reference 30).

Example 2a - Uncoped Connection General Criteria

W8x35 member and plate with two bolts, 7/8-in.-diameter and an allowable tension of 44 (ksi). The connection is subjected to a vertical load of 8.0 (K), a lateral load of 2.27 (K), and a torsional moment of 3.05 (K-in.). The material type for the member is A36 steel and for the plate is A588 steel. The dimensions for the connection plate are distance L1 = 5.50 in. and thickness Tp = 0.375 in.

Slot and pitch sizes are 0.25 in. and 3.0 in. respectively. Distances E1 and E3 are 0.5 in. and 2.75 in. respectively. Use a full penetration weld and consider general criteria with OBE load case.

Results obtained by hand calculation and program analysis are compared in Table 3.13-29.

Example 2b - (Coped Connection)

Same as Example 2a, but the member is coped. Distances L2 and L3 are 5.5 in. and 1.25 in. respectively.

Results obtained by hand calculation and program analysis are compared in Table 3.13-30.

3.13.1.30.2.3 Problem 3

The following examples were selected to validate the program for framed shop-welded/field-bolted-type connections (Detail No. 7.2.9 of Reference 30).

Example 3a - Uncoped Connection (L2=E3)

General criteria with OBE load case. W8x24 beam (A36) and two 4 x 3-1/2 x 3/8 x 5.5 angles (A36) with 7/8-in.-diameter bolts (A325) (fillet weld on instanding leg, weld size equal to 5/16 in., four bolts in two rows with pitch equal to 3.0 in. on outstanding leg). The connection is subjected to a vertical load (R_y = 1.98K), a lateral load (R_x = 0.44K), and an axial load (R_z = 1.38K).

The dimensions for the connection angle are as follows:

E3	=	1.25 in.	L1	=	5.5 in.
E	=	3.5 in.	L3	=	0.0 in.
B	=	0.0 in	OL	=	4.0 in.

On the outstanding angle, the gage is equal to 5.25 in., and the surface condition is Type A.

Results obtained by hand calculation and program analysis are compared in Table 3.13-31.

Example 3b - Coped Connection

All the parameters are the same as in Example 3a, except the coped distances on the top flange are L2 = 5.5 in. and L3 = 1.25 in. Results obtained by hand calculation and program analysis are compared in Table 3.13-32.

3.13.1.30.2.4 Problem 4

The following examples were selected to validate field-welded/ shop-welded friction-type connections (Detail No. 7.2.11 of Reference 30).

Example 4a - Uncoped Connection

General criteria with OBE load case. W10x39 beam (A36) and two 4 x 3-1/2 x 3/8 x 5.5 angles (A36) with 5/16-in. fillet weld (E70) on both instanding and outstanding legs. The connection is subjected to a vertical load (R_y = 2.60K), a lateral load (R_x = 1.50K), an axial load (R_z = 1.50K), and a torsional moment (M_z = 5.0 kip-in.).

The dimensions for the connection angle are as follows:

L1	=	5.5 in.	E3	=	1.25 in.
L3	=	0.0 in	E1	=	4.0 in.
E2	=	3.5 in.	B	=	0.0 in.

Results obtained by hand calculation and program analysis are compared in Table 3.13-33.

Example 4b - Coped Connection

All the parameters are the same as in Example 4a, except that the beam is coped on top with distance L2 = 5.5 in. and L3 = 1.25 in. Results obtained by hand calculation and program analysis are compared in Table 3.13-34.

3.13.1.30.3 CINCH

CINCH (Anchor Plate Assembly Analysis) is an interactive prompting program that analyzes individual expansion anchored plates with a single attachment and concrete expansion anchors. Design procedures are given in Structural Design Standard E11 (Reference 35). The program with its assumptions and limitations is consistent with this standard. The program is self-documenting and prints out a summary of results.

The program is a module of the SUPS package and was developed at S&L in 1984. It is currently maintained on UNIVAC 1100 Series hardware operating under EXEC 8.

The program was validated by comparing the program results to detailed hand calculations. Two problems used for this comparison are shown in the following. The CINCH results are compared with hand calculations in Tables 3.13-35 and 3.13-36.

3.13.1.30.3.1 Problem 1 (Initial Design)

Concrete thickness = 18 in.

Attachment size:

X-dimension = 8 in.

Y-dimension = 8 in.

The center of the attachment area is at the center of the plate. The C.G. of the attachment is at the center of the attachment area.

Default material properties are used.

Loading

SSE Case			OBE Case		
M_x	=	10,000 in-lb	M_x	=	8,400 in-lb
M_y	=	12,000 in-lb	M_y	=	10,000 in-lb
M_z	=	5,000 in-lb	M_z	=	4,200 in-lb
F_x	=	800 lb	F_x	=	670 lb
F_y	=	2,000 lb	F_y	=	1,670 lb
F_z	=	11,000 lb	F_z	=	9,200 lb

3.13.1.30.3.2 Problem 2 (As-Built Reassessment)

18 in. x 18 in. x 1 in. plate with 8-3/4-in.-diameter anchors in reinforced concrete.

Attachment size:

X-dimension = 10 in.

Y-dimension = 10 in.

The center of the attachment area is at X = 8.5 in. Y = 9.5 in. The C. G. of the attachment is offset from the center of the attachment area by X offset = -0.5 in., Y offset = 0.5 in.

OBE Load Case

M_x	=	25 kip-in.	F_x	=	5 kips
M_y	=	37 kip-in.	F_y	=	13 kips
M_z	=	20 kip-in.	F_z	=	15 kips

Loads are not reversible.

An edge of concrete is defined parallel to the x-axis at 3 in. above the top of the plate.

Concrete thickness = 18 in.

3.13.1.30.4 APLAN

APLAN (Attachment Plate Analysis) is a finite element program that can analyze rectangular attachment plates mounted on reinforced-concrete or concrete masonry by means of expansion anchors, headed welding studs, or wire embedments.

APLAN communicates with the user through a simple command- oriented language. It uses free-format input that consists of one or more key words interspersed with its arguments. This command language is used to define plate geometry, anchor location, and loading configurations, and to perform finite element analysis. The material properties for standard expansion anchored and embedded plates, as defined by Standard SDS E11 (Reference 35), have default values in the program, but the user can change these.

Efficient finite elements permit the analysis to be performed interactively. These finite elements are used to perform decoupled bending and plane stress analysis of the attachment plate. The bending analysis includes the partial contact of the plate with the concrete wall, which can result in prying and amplification of anchor tension forces. Plane-stress analysis is performed to determine the shear reactions on the anchors.

The output is printed at the terminal. This program has the capability to print out all the finite element solutions for every element and node of the plate. Because the comprehensive information requires enormous printout time, the program is defaulted to echo out all user input data, equilibrium check, all anchor reactions, and maximum element stress location. The user has the option to request the full output.

The program is a module of the SUPS package and was developed at S&L in 1985. It is currently maintained on UNIVAC 1100 Series hardware operating under EXEC 8.

The program was validated by comparing the program results to those generated by the ADINA program (Reference 36). Three problems were used for this comparison.

3.13.1.30.4.1 Problem 1

17 in. x 19 in. x 3/4 in. plate, shown in (a) of Figure 3.13-69.

8-3/4-in.-diameter concrete expansion anchors.

Concentrated load applied at x = 5 in., y = 13 in.

Loads: F_z = 2.50 kips

M_x = 50.20 kip-in.

M_y = 40.0 kip-in.

Comparison of results showed:

Anchor Number	APLAN Reaction	ADINA Reaction
1	0.19	0.19
2	0.04	0.05
3	0.00	0.00
4	1.34	1.35
5	0.00	0.00
6	3.19	3.20
7	1.35	1.36
8	0.48	0.48

3.13.1.30.4.2 Problem 2

15 in. x 15 in. x 1 in. plate.

Four studs 7/8-in.-diameter, 8 in. long at 1 1/2-in. edge distance.

Concentrated load applied at x = 9 in., y = 9 in.

Loads: F_z = 4.00 kips

M_x = 40 kip-in.

M_y = 50 kip-in.

Comparison of results showed:

Stud Number	APLAN Reaction	ADINA Reaction
1	1.17	1.23
2	0.01	-0.0003
3	4.51	4.49
4	2.93	3.0

3.13.1.30.4.3 Problem 3

12 in. x 22 in. x 1/2 in. plate, shown in (b) of Figure 3.13-69.

Twenty-one deformed wire anchors 0.302-in.-diameter, 1 ft-7 in. long.

Loads: F_z = 24 kips

Comparison of results showed the APLAN stress as 17.13 ksi, and the ADINA stress as 17.49 ksi.

3.13.1.31 ADINA (Reactor/Auxiliary Building)

ADINA (Automatic Dynamic Incremental Nonlinear Analysis) is a computer program for the static and dynamic displacement and stress analysis of solids, structures, and fluid-structure systems. The program can be used to perform linear and nonlinear analyses. The structural systems can be composed of combinations of different finite elements. The program presently contains the following element types:

a. Three-dimensional truss element
b. Two-dimensional plane stress or plane strain element
c. Three-dimensional plane stress element
d. Two-dimensional axisymmetric shell or solid element
e. Three-dimensional solid or thick shell elements
f. Three-dimensional two-node beam element
g. Curved beam element
h. Three-node thin plate/shell element
i. Thin shell element
j. Two- and three-dimensional fluid elements.

The nonlinearities may be due to large displacements and non-linear material behavior. The material descriptions presently available are:

For the Truss Elements

a. Linear elastic
b. Nonlinear elastic
c. Thermo-elastic
d. Elastoplastic (isotropic or kinematic hardening)
e. Thermo-elastic-plastic and creep (isotropic or kinematic hardening).

For the Two-Dimensional Elements

a. Isotropic linear elastic
b. Orthotropic linear elastic
c. Isotropic thermo-elastic

d. Curve description model

e. Concrete model

f. Elastic-plastic materials, von Mises (isotropic or kinematic hardening) and Drucker-Prager yield conditions

g. Thermo-elastic-plastic-creep, von Mises condition (isotropic or kinematic hardening)

h. Mooney-Rivlin material.

For the Three-Dimensional Elements

a. Isotropic linear elastic

b. Orthotropic linear elastic

c. Isotropic thermo-elastic

d. Curve description model

e. Concrete model

f. Elastic-plastic materials, von Mises (isotropic or kinematic hardening) and Drucker-Prager yield conditions

g. Thermo-elastic-plastic-creep, von Mises isotropic or kinematic hardening yield condition.

For the Two-Node Beam Element

a. Linear elastic

b. Elastic-plastic, von Mises yield condition.

For Curved Beam Element

a. Linear elastic

b. Elastic-plastic, von Mises yield condition (isotropic or kinematic hardening).

For Three-Node Plate/Shell Element

a. Linear elastic

b. Elastic-plastic, Ilyushin yield condition isotropic hardening.

For the Shell Element

a. Linear elastic

b. Elastic-plastic, von Mises yield condition (isotropic hardening).

The ADINA program is an out-of-core solver, so very large finite element systems can be considered. Also, all structure matrices are stored such that only nonzero elements are processed, resulting in maximum system capacity and solution efficiency.

In dynamic analysis, the frequencies of the system can be calculated, and the system response can be evaluated using mode superposition or implicit direct-time integration (the Newmark method or the Wilson method), or explicit direct-time integration (the central difference method).

In nonlinear analysis, the finite-element system response is evaluated using an incremental solution of the equations of equilibrium. The incremental equilibrium schemes that can be used are an accelerated modified Newton iteration or the BFGS method. Substructuring can be used to increase the solution efficiency.

ADINA was developed by Klaus-Jurgen Bathe (Reference 37) at the Massachusetts Institute of Technology. It is currently maintained by S&L on the UNIVAC 1100 Series hardware operating under EXEC 8. To demonstrate the validity of the major analytical capabilities of ADINA, the test problems are taken from the ADINA User's Manual and are compared with solutions of the S&L version.

3.13.1.31.1 Problem 1

Frequency Analysis of a Tower Cable

The cable stretched between a ground anchor point and tower attach point, shown in Figure 3.13-70, was analyzed for frequencies of vibration. The cable was modeled using 12 truss elements of linear elastic material, as shown in Figure 3.13-70. The cable had an initial tension of 7520 lb. Insulators weighing 510 lb each were located at nodes 2, 4, and 6, and a cluster of six insulators totaling 3060 lb was located at node 8. Nodes 3, 5, 7, and 9 through 12 are intermediate nodes located along the cable without insulators. The total vertical load acting on the cable nodes was 5677.83 lb, which includes the insulator weights and the cable self-weight.

For the frequency analysis, a lumped-mass matrix of the cable has been assumed to which the masses of insulators have been added. The periods of vibration of the cable about the static equilibrium configuration are given in Table 3.13-37.

3.13.1.31.2 Problem 2

Large Displacement Analysis of an Elastic Simply Supported Plate

The simply supported square plate subjected to a uniformly distributed pressure shown in Figure 3.13-71 was analyzed for its large deflection response. One single 16-node shell element was used to model one-quarter of the plate.

Figure 3.13-71 shows the displacement response predicted in the finite element analysis. The computed displacement response compares very closely to the solutions given by Levy (Reference 38). The effect of using different assumptions on the plate edge in-plane displacements was modeled using the constraint equation option in ADINA.

3.13.1.31.3 Problem 3

Thermo-Elastic Static Analysis of a Cantilever Beam

The cantilever beam shown in Figure 3.13-72 was subjected to a linearly varying temperature gradient in the Z-direction. No mechanical loads were applied. The beam was modeled using three 16-node, three-dimensional elements. Since displacements and strains are small, the analysis was carried out for material non-linearities only, and by using appropriate displacement boundary conditions, only the portion of the beam above the neutral surface was included in the finite element model.

Figure 3.13-73 shows the displacement response of the cantilever neutral surface. Excellent agreement with the solution by Boley and Weiner was obtained (Reference 39).

3.13.1.31.4 Problem 4

Static Analysis of a Reinforced-Concrete Beam

The simply supported reinforced-concrete beam subjected to two symmetric concentrated loads, as shown in Figure 3.13-74, was analyzed using ten 6-node, concrete plane-stress elements and 10 steel truss elements. The material properties of the concrete were idealized using the concrete model with the parameters given in Figure 3.13-74. Materially nonlinear-only response was assumed, i.e., large-displacement effects were neglected.

Figure 3.13-75 gives the calculated transverse displacements at the midspan of the beam, for A_{st} = 2.00 in.^{2} for nonlinear static response. The loading scheme used is also shown in this figure. Other results on analysis for A_{st} = 0.62 in.^{2} are compared with the response predicted by Suidan and Schnobrich (Reference 40), who assumed a linear stress-strain relationship for the concrete with the constant Young's modulus equal to E_0 of this analysis, and who modeled the steel reinforcement as a smeared stiffness added to the concrete.

3.13.1.31.5 Problem 5

Analysis of a Beam Subjected to a Traveling Load

The simply supported beam in Figure 3.13-76 was analyzed for its dynamic response. The beam was subjected to a constant-magnitude force traveling across its span at a constant velocity. In the analysis, 20 beam elements were used to model the structure, and small displacements and elastic material conditions were assumed. To model the traveling load, the time function and arrival time option were used in ADINA.

Figure 3.13-76 shows the midspan lateral deflection during the period the load is acting on the beam. The analysis results using ADINA are also compared with one-mode analytical solution given by Biggs (Reference 3).

3.13.1.32 FRAME (Reactor/Auxiliary Building and Residual Heat Removal Complex)

FRAME (Integrated Frame Analysis System) analyzes frames for static and dynamic loadings, performs load combinations, and checks stresses against allowable stresses.

The following analyses are performed:

a. Static: Analysis of distributed and concentrated weight loadings and reaction loadings on frames

b. Dynamic: Analysis of frame response to seismic loads using pseudo-static methods.

The static and dynamic analyses can be performed independently or in sequence.

The input consists of the frame geometry, material properties, and static and dynamic loadings. Various options exist to control the length of the output. The default option generally prints only the summary of input data and final results.

The load combinations are done per user specification. Three methods of combination are used: (1) combination by addition considering signs; (2) combination by addition of absolute values; and (3) combination by square-root-of-the-sum-of-the-squares (SRSS). Loads designated as WT (self-weight of the members and additional lumped weights) are combined considering the signs. Loads designated as RO (reaction loads) or SE (seismic) are combined by either summation of absolute values or by SRSS, as specified by the user. To account for the sign reversal inherent in the RO and SE loads, these loads are combined with the absolute value of the result of the WT-type load combination to give the "worst case" final loads. Two values of the load combination are also obtained for the axial load: the combined WT load with the sign is combined with the combined RO and SE terms with both plus and minus signs. These two values are used in determining the tension and compression stresses.

The stress-checking provisions follow those of the 1978 AISC Specification (Reference 41), and Structural Design Standard E-37 for Mechanical Component Auxiliary Support Steel Framing (Reference 42). AISI Specifications (Reference 43) are used to check the stress levels in Unistrut members (Reference 44).

Allowable stresses calculated using the above-referenced documents are multiplied by an overstress factor (input by the user). However, these stresses are limited to "SLIM*FY", where 'SLIM', defined as stress limiting factor, is calculated as follows:

a. For axial stresses (direct or bending)

$$\text{SLIM} = \frac{1}{\text{Minimum Factor of Safety}} \leq 1.0$$

b. For shear stress

$$\text{SLIM} = \frac{1}{\sqrt{3}\,(\text{Minimum Factor of Safety})} \leq 0.57$$

Minimum factor of safety is input by the user. Overstress factor is 1.0 for normal load combinations.

Presently the program can check member design for Unistrut wide flange, structural tube sections, and single angle members.

FRAME was developed at S&L in 1983. It is currently maintained on UNIVAC 1100 Series hardware operating under EXEC 8.

To demonstrate the validity of the analysis performed by the FRAME program, the following two examples are presented.

To illustrate the validity of the static portion of FRAME, the problem shown in Figure 3.13-77 was analyzed and the results compared to those given in Reference 45. Table 3.13-38 shows comparison of member end forces; the results are in good agreement. To illustrate the validity of the dynamic analysis portion of FRAME, which uses a pseudo-dynamic method of analysis in which the system is analyzed for a static loading equivalent to the mass times the specific acceleration applied at all mass points, the problem shown in Figure 3.13-78 was analyzed. A static selfweight-loading analysis was performed for each of the three global directions. The results for each direction loading were multiplied by the corresponding direction g level to obtain the individual direction-excitation results. The final result is taken as the SRSS of the three direction-excitation results. The results for the program and independent calculations are compared in Table 3.13-39; the results agree.

To demonstrate the validity of the load combinations performed by the FRAME program, the following example is presented. The frame shown in Figure 3.13-77 was analyzed for three loadings: self- weight, WT1; reaction loading as shown in Figure 3.13-77, R01; and seismic OBE, SE1. Two load combinations were generated: WT plus absolute sum of reaction load and seismic; and WT plus SRSS of reaction load and seismic. The individual analysis results at select locations are given in Table 3.13-40. Also shown in Table 3.13-40 is the comparison of the load combination results from FRAME and hand calculations; the results agree.

To demonstrate the validity of the member design portion, the following examples (one for each shape of member) have been selected.

3.13.1.32.1 Problem 1 - Tube Section

A 6 x 4 x 3/8 tube section under the loading shown in Figure 3.13-79 was used to validate the design check for tube sections. Additional data are given in Table 3.13-41. FRAME output is compared with hand calculations in Table 3.13-41; they are in close agreement.

3.13.1.32.2 Problem 2 - Wide Flange Section

A W12x40 section under the loading shown in Figure 3.13-80 was used to validate the design check for wide-flange sections. Additional data are given in Table 3.13-42. FRAME output is compared with hand calculations in Table 3.13-41; they are in close agreement.

3.13.1.32.3 Problem 3 - Unistrut Section

A P1000 Unistrut section under the loading shown in Figure 3.13-81 was used to validate the design check for Unistrut sections. Additional data are given in Table 3.13-42. FRAME output is compared with hand calculations in Table 3.13-41; they are in close agreement.

3.13.1.32.4 Problem 4 - Single Angle Section

A 3 x 2 x 3/8 angle section under loading shown in Figure 3.13-82 was used to validate the design check for single-angle sections. Additional data are given in Table 3.13-42. FRAME output is compared with hand calculations in Table 3.13-41; they are in close agreement.

To demonstrate the validity of the connection module, five problems were selected. The problem data given are in Table 3.13-43. These problems were validated by comparing the FRAME results to the results obtained from hand calculations. A comparison of results is given in Table 3.13-44.

3.13.2 Computer Programs Used by Chicago Bridge & Iron

Subsections 3.13.2.1 through 3.13.2.9 provide a description of the CBI computer programs used for general analysis and design work. Computer program information beyond that included herein, or in the Stress Report and Calculations, is proprietary. However, test problems and verifications are on file and available at CBI.

The description of programs used by CBI includes a discussion on the design of the drywell and torus. There have been extensive modifications to the torus subsequent to its original installation. The details of modifications to the torus are presented in the Fermi 2 Plant Unique Analysis report. See References 46 through 51.

3.13.2.1 Program 405

This is a program used for the analysis of a ring with a constant moment of inertia and modulus of elasticity. The loads are in the ring. The mathematics are based upon the Hardy-Cross column analysis for rings as referenced in Theory of Modern Steel Structures, Vol. II, by Grinter, page 259. The loads can be moments, tangential, or radial to the ring. The printouts are coefficients at incremental distances around the ring. The printout titles for the output are as follows:

X	=	Angle and degrees as measured from a reference axis
V	=	A radial shear with force units acting in a radial direction through the ring
T	=	An axial thrust in the ring with units of force
M/R	=	A coefficient with units of force which when multiplied by the radius to the centroid will equal a moment
EI/RR	=	A coefficient which when multiplied by the radius^2 will equal the rotation of the ring at the point
REI/RRR	=	A coefficient which when multiplied by the radius^3 equals the radial deflection of the point
CEI/RRR	=	A coefficient which when multiplied by the radius^3 will equal the tangential deflection of the point.

3.13.2.2 Program 601

This program is based on the mathematics of Program 405. In addition, the coefficients have been multiplied by the proper radius. This means that the thrust and moment only have to be divided by the area and section modulus, respectively, to find the stresses at the point.

3.13.2.3 Program 655

This program is based on the theory and equations presented in NASA-TN 1219. In the program the influence of the loads on any ring is not evaluated beyond the adjacent rings. Basically, the only difference between this program and the previous ring programs is that the shear in the ring with loads is transferred into the shell between the rings.

3.13.2.4 Program 7 - 81N - Kalnins' Shell Program

This program was developed by Aerturs Kalnins at Yale University, based on a method of analysis published in Reference 11. The program is used for shell stresses at discontinuities, with the exception of nozzles.

3.13.2.5 Program 772

This is a program for checking nozzle reinforcing. It is designed essentially for containment vessels, and adheres to area replacement criteria specified by ASME Sections III and VIII. The program does no design work, merely checking the adequacy of preselected reinforcing plate dimensions and weld sizes.

3.13.2.6 Program 6 - 20 Cookbook Nozzles

This program computes the local stresses in cylindrical and spherical shells due to a load or a combination of loads acting on a nozzle that penetrates the shell. The solution for local shell stresses is made using the dimensionless parameters (input) from the graphs in the Welding Research Council Bulletin No. 107. When reinforcing is present, these parameters are found using the procedures of Bijlaard, as outlined in Welding Research Council Bulletins 49 and 50.

If a solution for unit loads is desired, the card for loads is left blank. The program assigns unit loads of 1000 lb to the radial load and the shears, and 100 in.-lb to the moments.

Tests are performed in the cylinder and sphere subroutines to see if either an insert or pad plate or no reinforcing is present. Depending on the results of these tests, a particular set of denominators is computed for use in the stress calculations in the stress subroutine. When reinforcing is present, the program checks the stress at the edge of the reinforcing. The thicknesses used in the computation of stresses are described in the following subsection.

3.13.2.7 Program 860 - Rigid Attachment to Spherical Shell

3.13.2.7.1 General

This program computes shell stresses around a rigid attachment to a spherical shell due to any combination of loading, radial, shear, or moment. The program uses the nomenclature, the curves for coefficients, and the mathematics of the Welding Research Council Bulletin No. 107. Given the basic geometry of the attachment, the program will compute the parameters as required from Figures SR-2 and SR-3 and the shell stresses around the attachment.

If the load card is blank, the program assigns unit loads of 1000 lb to P and U1, and 100 in.-lb to M1. There is one printout for each of the unit loads.

If the width of reinforcing is less than 1.65 times the square root of the spherical radius, times either the thickness of the insert or an equivalent thickness for pads, the stresses are also checked at the edge of the reinforcing. All induced moments at the nozzle-to-shell junction and the induced moment M_x at the edge of reinforcing, are increased by 20 percent to satisfy the requirements of Welding Research Council Bulletins 49 and 50 (References 52 and 53) by Bijlaard. If the width of reinforcing is greater than 1.65 times the square root of the spherical radius, times either the insert thickness or equivalent thickness, only the stresses at the nozzle-to-shell junction are computed. None of the induced moments are increased. The thickness used in the solution of stresses about a reinforced nozzle is determined as follows.

3.13.2.7.2 Reinforced Nozzle Parameters For Bijlaard Analysis

At Nozzle-to-Shell Junction

a. T = Thickness of insert or equivalent thickness for pad-type reinforcing. T equivalent = 1/2 $\{(TS + TP)^3 + (TP)^3 + (TS)^3\}^{1/3}$ where TP is the thickness of the pad and TS is the shell thickness without reinforcing. The quantity under the cube root is the average of the moment of inertia of the total thickness of the shell plus pad and the sum of the individual moments of inertia of the shell and the pad

b. All parameters are found using T. The stresses are computed using T. The programs compute the membrane stress using the total thickness of the pad plus shell. However, the parameters contain the equivalent thickness which is divided out to obtain N_x, N_y, or N first

Example: $\frac{N_X T}{P} \quad \frac{P}{T(TS+TP)}$

c. If the width of reinforcing, W, is less than C RT, where R is the mean radius of the shell, T is the thickness defined in item 1, and C is normally assumed to equal 1.65, the reinforcing is assumed to act as a rigid plug and the induced moments M_x and M_y for spheres or M_x and M for cylinders are increased by 20 percent because of effects of reinforcing (References 52 and 53). If W is greater than $C\sqrt{RT}$, the reinforcing is assumed to act as a shell plate with no increase in induced moments.

At Edge of Reinforcing

a. TS = Thickness of shell without reinforcing

T = Thickness of insert or equivalent thickness for pad-type reinforcing. See Item b. above.

b. Parameters for induced moment M_x are found using T. All other parameters are found using TS. The shell stresses are computed using TS. The programs divide out the only equivalent thickness in the parameters containing M_x only

Example: $\frac{M_X RT}{M_1} \quad \frac{6M_1}{(TS)^2 RT}$

c. Increase the parameters containing the induced moment M_x by 20 percent.

3.13.2.8 Program 7-78, Drywell Primary Membrane Stress Analysis

The drywell shell is analyzed for stresses due to the customer- specified loading combinations. Primary membrane stresses are computed for each of the loading combinations, and the resulting stresses are compared to the ASME Code allowables. In addition, the compressive stresses are compared to an allowable buckling stress, and a buckling ratio is computed.

The drywell primary membrane stresses are found using the general equations for an axisymmetrically loaded shell of revolution. The derivation of the general equations can be found in Chapter 14 of Theory of Plates and Shells by Timoshenko (Reference 6). The equations are as follows:

General Equation No. 1: $\frac{N_\phi}{R_\phi} + \frac{N_\theta}{R_\theta} = P$

General Equation No. 2: $2\pi r_o N_\phi \sin\phi + Z = 0$

where

N_ϕ	=	meridional membrane stress resultant
N_θ	=	circumferential membrane stress resultant
R_ϕ	=	radius of curvature in meridional plane
R_θ	=	radius of curvature in circumferential plane
P	=	pressure
ϕ	=	angle between pole of revolution and point
Z	=	resultant of total load on shell
r_o	=	$R_\theta \sin\phi$

It should be noted that the stress resultants at structural discontinuities, such as the cylinder-to-knuckle and knuckle-to- sphere, are the maximum stress resultants. The stress resultants are found using the appropriate equations for the smaller thickness at the point of discontinuity.

Pressure

Top Head - The top head is designed for stresses due to internal and external pressure. The thickness required for internal pressure is found using the formulas in Paragraph UA-4(c) of Section VIII of the ASME Code, while the allowable external pressure is found according to the requirements of Paragraph UG-33 of Section VIII of the ASME Code. This design is in accordance with Code Case 1392.

The top head is also designed for stresses due to a jet load. The stresses resulting from this jet load are computed using Case 20 on page 304 of Reference 54.

Cone - The top cone, if one exists, is designed according to the requirements of Paragraph UA-5 of Section VIII of the ASME Code. This analysis is in compliance with Code Case 1392.

Knuckle - The knuckle pressure stress resultants are analyzed in this section, using the "pressure area method" as outlined in Reference 55.

Cylinder - The cylinder is designed for both internal and external pressure, in accordance with Code Case 1392. The design for internal pressure is made using the equations for thickness of UG-27(c) of Section VIII of the ASME Code.

The external pressure design is made using the method described in Paragraph UG-28 of Section VIII of the ASME Code. The curves in Figure UCS-38.2, which are referred to in Paragraph UG-28, are defined by the following equation:

$$\text{Allowable External Pressure} = \frac{2.6\ E\ [T/D]^{2.5}}{4\left[L/D - 0.45\ [T/D^{0.5}]\right]}$$

where

E = modulus of elasticity of steel

T = thickness of cylinder

D = diameter of cylinder

L = length of cylinder including one-third the vertical height of the knuckle and the lesser of one-third of the length of the cone to its apex or length to the flange

Sphere - The sphere is analyzed for both internal and external pressure. The stress resultants due to internal pressure are found using the general equations reduced to the following form:

$$N_\phi = N_\theta = \frac{PR}{2}$$

where

R = radius of sphere

The sphere is also checked for buckling stresses when subjected to external pressure. A discussion of this buckling analysis is found in the discussion of allowables for compressive stress resultants. The buckling stress resultant due to external pressure is considered in conjunction with the stress resultants due to dead loads and the effects of seismic loading on these dead loads.

Vertical Loads

The vertical loads include, but are not limited to, the weight of the penetrations, compressible material, shell steel, jet deflectors, refueling water, and spray headers. Also included with the vertical loads is the effect of vertical earthquake acting on the above loads. The stress resultants for these loads are found using the general equations reduced to the following forms for the various shapes.

Cylinder

General Equation No. 2 $$N_\phi = \frac{\text{Vertical Load}}{\text{Circumference}} = \frac{\text{Load}}{2\pi R_\theta}$$

General Equation No. 1 $$N_\theta = 0$$

where

P = 0

R_ϕ = ∞

R_θ = circumferential radius

Knuckle

General Equation No. 2 $N_\phi = \frac{\text{Load}}{\text{Circumference} * \sin\phi}$

$= \frac{\text{Load}}{2\pi L_1 (\sin\phi)^2}$

General Equation No. 1 $N_\theta = N_\phi \frac{L_1}{-R_2}$

where (refer to [a] in Figure 3.13-83)

P = 0 for all loads except shell weight and compressible material

R_2 = knuckle radius (negative number)

L_1 = distance from pole to point as measured on the normal

Special consideration is given to the weight of the shell and compressible material. N_ϕ is computed using General Equation No. 2 above. However, the density of the shell is considered to act as a pressure in the radial direction in finding N_θ. Therefore, the General Equation No. 1 for N_θ due to shell weight or compressible material is as follows:

General Equation No. 1 $N_\theta = -\rho + L_1 \cos\phi + N_\phi \frac{L_1}{R_2}$

where

P = $-\rho t \cos\phi$

ρ = density of steel or compressible material

t = thickness of shell

L_1 = distance from pole to point as measured on normal

R_2 = knuckle radius (negative number)

Sphere

General Equation No. 2 $N_\phi = \frac{\text{Weight}}{2\pi R (\sin\phi)^2}$

General Equation No. 1 $N_\theta = -N_\phi$

where

P = 0 for all loads except shell weight and compressible material

R = radius of sphere

The weight of the shell and the weight of the compressible material is again treated as a pressure in the radial direction for finding the stress resultant N_θ.

General Equation No. 1 $N_\theta = -\rho t R \cos\phi - N_\phi$

where

R = radius of sphere

P = $-\rho t \cos\phi$

ρ = density of steel or compressible material

t = thickness of shell

Horizontal Earthquake

The effect of the horizontal earthquake is to produce a shear load acting on the shell at the elevation of the load. This shear is found by multiplying the load by the horizontal earthquake factor for the elevation of the load. This factor is taken from curves for horizontal earthquake given in the customer specifications. From statics the shear load can be considered to produce a moment at a lower elevation. This moment tends to rotate the drywell shell about the plane under consideration.

In the earthquake analysis, the drywell is analyzed as a free- standing, cantilevered column. However, the drywell can be supported by the surrounding building at the stabilizer elevation. This support is separated from the stabilizer of the drywell by a l0 mil-gap. Thus, during the incidence of an earthquake, the vessel may generate a shear in the opposite direction of the shear of the applied loads. This shear is the reaction at the stabilizer elevation, which is treated in the same manner as the other shear loads. The reaction is found using a combination of Castigliano's First Theorem and the unit load method using the following equations:

$$\Delta = \frac{1}{E}\int \frac{M}{I}\frac{\delta M}{\delta P}\,dx + \frac{1}{G}\int \frac{V}{A}\frac{\delta V}{\delta P}\,dx$$

ΔImposed = ΔHorizontal Earthquake Acting on Vessel + ΔUnit Load x Reaction

The stress resultants due to moment are computed using the general equations. These equations have been reduced as follows for the three general shapes in the drywell.

Cylinder

General Equation No. 2 $N_\phi = \frac{\text{Moment}}{\text{Section Modulus}} = \frac{\text{Moment}}{\pi R^2}$

General Equation No. 1 $N_\theta = 0$

where

P = 0

R_ϕ = ∞

R_θ = radius of curvature in circumferential plane

Knuckle (refer to [b] in Figure 3.13-83)

General Equation No. 2 $N_\phi = \frac{\text{Moment}}{\text{Section Modulus} * \sin\phi}$

$$N_\phi = \frac{\text{Moment}}{\pi(L_1 \sin\phi^2 * \sin\phi)}$$

General Equation No. 1 $N_\theta = -N_\phi \frac{L_1}{R_2}$

where

P = 0

L_1 = distance from pole to point as measured on the normal

R_2 = knuckle radius (negative number)

Sphere

General Equation No. 2

$$N_\phi = \frac{\text{Moment}}{\text{Section Modulus} \times \sin\phi^*}$$

$$N_\phi = \frac{\text{Moment}}{(R\sin\phi)^2 \times \sin\phi}$$

General Equation No. 1 $\quad N_\theta = -N_\phi$

where

P = 0

R = radius of sphere

* sin ϕ used to transfer stress resultant into plane of shell.

Drywell Flooded

In the flooding of the drywell, the stress analysis is made for stresses both in the meridional and circumferential directions (Figure 3.13-84). In the meridional direction, floodwater weight adds to the other gravity loads and causes an increase in the compressive stress. These other loads are the weight of the shell steel, the weight of the compressible material (if applicable), the weight of the penetrations, dead loads, and live loads. In the consideration of the meridional stress, the buckling of the shell is the limiting factor. In the circumferential direction, the hydrostatic pressure due to the floodwater increases the total circumferential stresses. The stresses in each direction are analyzed both with and without seismic effect.

Meridional Stress - In the analysis of the meridional stresses, there are two conditions considered critical, and therefore an analysis is made for each.

One condition exists when the floodwater reaches its maximum elevation as specified by the customer. This condition is considered critical because it obviously involves the largest amount of floodwater in the drywell shell, and also because it involves the greatest hydrostatic pressure that the drywell shell will experience under the flooding condition.

The second condition occurs instantaneously as the drywell is filled and the water reaches the critical point P (see Figure 3.13-85). This point P is considered critical for two reasons:

a. With reference to Figure 3.13-85, it can be seen that the maximum water weight that the shell will carry will exist when the water level is at point P or higher. No matter how high the water floods above point P, only the overhanging water (bounded by the shell, point P, and embedment) can be carried by the shell. The remaining water is carried by the internal concrete through the shell into the foundation

b. With reference to Figure 3.13-86, it can be seen that there is an unbalanced hydrostatic pressure acting on the drywell shell between point P and the vertical cylindrical shell. This unbalanced pressure is a buoyant force that is calculated based on Volume (B) with respect to Archimedes' principle. This buoyant force acts upward and thereby reduces the buckling stress.

Considering the above two reasons, it can be seen that the worst loading condition, that is, the maximum water load at embedment, and the minimum bouyant force (equal to 0) will both be attained with the water level at point P.

While the additional weight of the floodwater will increase the buckling stress, the water pressure inside the vessel will permit increasing the critical buckling stress. A calculation for this increase is made, and the result is added to the normal shell allowable buckling stress to give the critical buckling stress.

By combining the compressive meridional stress due to the floodwater with those stresses due to the normal loads, and then dividing this total into the increased critical buckling stress, a factor of safety is calculated.

Circumferential Stress - In the analysis of the circumferential stresses, the general membrane equation from Page 39 of Reference 56 is used:

$$N_\theta = PR - N_\phi$$

where

P = hydrostatic pressure due to floodwater

R = drywell sphere radius

N_ϕ = meridional stress resultants

Allowables

Tensile - The stress that results from the combination of loading for each condition of loading is compared to the allowable general membrane stress intensity. This is in accordance with the requirements of Section III of the ASME Code for Class B vessels.

Compressive - The compressive stress resultants are compared to allowables obtained according to the paragraphs titled "Biaxial Compression-Equal Unit Forces" and "Biaxial Compression-Unequal Unit Forces" of the Welding Research Council Bulletin No. 69. The allowables used are found by assuming that the sphere reacts as a cylinder with a radius equal to the radius of the sphere. There are three cases of loading considered. The allowables for these three cases are

a. Uniaxial compressive stress resultant

$$N_{ALL} = 1.8 * 10^6 \frac{t^2}{R}$$

b. Biaxial equal compressive stress resultants

$$N_{ALL} = 0.9 * 10^6 \frac{t^2}{R}$$

c. Biaxial unequal compressive stress resultants. This case is treated as the summation of a uniaxial condition with the biaxial condition with equal stress resultants (see [c] of Figure 3.13-83)

$$\frac{N_\theta - N_\phi}{1.8*10^6 \frac{t^2}{R}} + \frac{N_\phi}{0.9*10^6 \frac{t^2}{R}} \leq 1$$

3.13.2.9 Program 7-71

As stated in Subsection 3.13.2, the description of programs used by CBI includes a discussion on the design of the drywell and torus. There have been extensive modifications to the torus subsequent to its original installation. The details of modifications to the torus are presented in the Fermi 2 Plant Unique Analysis report. See References 47 through 52.

3.13.2.9.1 Torus Columns and Column "Stubs" Design

The inner and outer columns and the inner and outer column "stubs" that connect the columns to the torus ring are designed by the computer program using the approach illustrated in Figure 3.13-87.

The column "stubs" are welded to the columns by full fusion welds.

Coefficient of friction for lubrite = 0.1

Friction Force = 0.1P (resisted by column knee braces)

Shear due to horizontal seismic force taken by the torus seismic ties.

The inner and outer columns and column "stubs" (which are usually built-up sections) may have different cross-sectional properties but these must qualify as "compact" sections in accordance with AISC specifications.

The inner and outer columns may have different lengths.

The total length of the columns is taken as the distance from the top surface of the column baseplates (PT.C) to the point where the vertical centerline of the corresponding column "stub" intersects the outer surface of the torus ring (PT.D).

To determine the value of the "effective slenderness ratio" under axial compression only, the values of "K" adopted are K = 1.20 for buckling in radial direction and K = 0.65 for buckling in tangential direction (refer to Section 1.8 of AISC Commentary).

The total axial load and bending moment is taken to be the same for the outer column and outer column "stub," and for the inner column and inner column "stub."

The inner column and column "stub" have been designed for an axial load consisting of the dead load of torus and contents, vertical seismic load, and vertical component of inner brace load which is transferred to the column by the torus ring.

The outer column and column "stub" have been designed for an axial load consisting of dead load of torus and contents, vertical seismic load, vertical component of outer knee brace load, and the overturning effect of horizontal seismic force which is conservatively assumed to be applied directly to the outer columns.

The principal factor causing bending moment is the differential temperature expansion 'Δ' between the column "stub" attachment and the attachment of the knee bracing to the torus shell (Figure 3.13-88).

The differential movement of these two points as a result of pressure-induced stresses has been neglected as these are very small compared to the temperature-induced differential movement of these two points.

The column baseplates rest on a lubrite pad, and are free to slide and compensate for the overall expansion of the torus without causing any bending to be induced in the columns and column "stubs."

The bending moment in the columns and column "stubs" is produced by the differential movement of the column stub attachment to the torus ring radially outward with respect to the column base.

This bending moment is given by

$M = \frac{6EI\Delta}{L^2}$ (Refer to AISC, P. 2-127, Case 23)

Here it is assumed that the knee brace transmits the radial movement of the brace attachment to the column baseplates which slide on the lubrite pads; after which the columns are assumed to be fixed at the column bases and the corresponding column "stub" attachments then undergo the differential expansion radially outward with respect to the torus without any rotation at the junction with the torus ring.

The program computes the actual axial and bending stresses for the inner and outer columns and column "stubs" which are shown on the computer printout. Allowable stresses are also calculated per Sections 1.5.1.3 and 1.5.1.4 of the AISC Specification which are also shown on printout. These actual stresses are compared to the corresponding allowable stresses using the ratios per Section 1.6.1 of the AISC Specifications.

In addition, the program also checks the outer and inner column "stub" attachment welds to the torus ring.

<u>Column Baseplate (refer to [a] of Figure 3.13-89.)</u>

The column baseplate is designed for the column axial load per page 3-75 of the AISC Specifications. The program assumes 3000 psi concrete for calculations.

Allowable concrete bearing F_P = 0.25x3000 x F

F = Factor for increasing allowable stress during flooded condition

Required area = $\text{Col Load}/F_P$

Actual Bearing F_{PA} = $\text{Col Load}/\text{Furnished Area}$

Allow bending F_B = (0.75) (yield F)

Required plate thickness = $\sqrt{\frac{3F_{PA}X^2}{F_B}}$

X = Larger of M or N

Column Knee Braces (see [b] of Figure 3.13-89.)

The column knee braces consist of two angles back to back and are designed to take the friction force due to the column base sliding on the lubrite pad with a coefficient of friction equal to (0.1).

The allowable compressive stress is calculated per Section 1.5.1.3 of the AISC Code and is shown on the printout. The maximum distance (L_s) between spacers is calculated per Section 1.18.2.4 and is shown on the printout.

3.13.2.9.2 Torus Support Ring

The stresses in the torus support ring are analyzed by a computer program using the approach as outlined in the following discussion:

The ring is subjected to the following loadings:

a. Column stub reactions

b. Column deflections

c. Column knee brace reactions

d. Header dead load

e. Downcomer jet thrust

f. Internal pressure.

The reaction loads are broken into components so that the three basic loadings (other than internal pressure) are radial, tangential, and bending moment. The magnitude of the loads applied to the ring and their point of application (angular location) are shown on the printout and are specified as radial, tangential, or moment with respective signs. The loads are applied as shown in Figure 3.13-90.

The sign conventions for the ring loads are as follows:

a. Radial = positive when acting inward

b. Tangential = positive when acting clockwise

c. Moment = positive when acting clockwise.

The column stub reactions and moments are broken into two parts and assumed to act 10° apart as shown in Figure 3.13-90. The header dead load and the downcomer jet loads are assumed as radial loads due to their very small angle with the centerline.

The ring loads, as shown on the printout, may be checked by referring to the following:

a.	Column stub axial load	Printout from column stub design section
b.	Column stub bending moment	Printout from column stub attachment weld
c.	Brace axial load	Printout from brace design
d.	Header dead load	See sheet
e.	Downcomer jet load	See customer specifications

For the stress analysis of the ring, the suppression chamber has been assumed as a ring-stiffened cylinder with fixed ends made up of four similar bays. The ring loads as discussed above are placed on the second ring and bending, thrust, and pressure stresses on this ring are calculated by computer program No. 655 based on theory derived in NASA Technical Note No. 1219. Stresses from the internal pressure are calculated by the "pressure area method" from Reference 55.

The computer prints out the component stresses (bending, thrust, and pressure) in addition to the total stresses on the inner and outer flanges of the ring.

3.13.3 Computer Programs Used by Others

Computer programs used by S&L, CBI, and S&W, Michigan, are described in Subsections 3.13.1, 3.13.2, and 3.13.4, respectively. Other significant computer programs used by these and other support organizations (including Edison) are described in this section. A number of computer programs were used by NUTECH Engineers, Inc., in the plant-unique analysis required by the NRC (NUREG-0661, Safety Evaluation Report, Mark I Containment Long-Term Program). The major computer programs used in these analyses are described in the Fermi 2 Plant Unique Analysis reports (References 46 through 51), which were submitted in response to NUREG-0661 requirements.

3.13.3.1 ADLPIPE - Arthur D. Little, Inc.

There are three types of documentation for ADLPIPE. The first is the multitude of hand checks made during the development and change of the program. The second is by the many user groups who have their own method of evaluation and documentation, both analytical and experimental. These groups have contributed immeasurably to the current state of ADLPIPE reliability. The third type is the documentation and internal checks that Arthur D. Little, Inc., has generated.

This third type of documentation and internal checks is in four forms:

a. Fifty-two common errors are checked for and automatically reported
b. All internal program data may be printed during problem solution
c. Sample problems (benchmarks) are compared to other solutions
d. Mathematical techniques used are described.

3.13.3.1.1 Input Check

Automatic message for 52 different types of input error. See Input Preparation Manual.

3.13.3.1.2 Intermediate Data

a. Force vectors are printed prior to inversion of the stiffness matrix
b. Deflection vector is printed after stiffness matrix inversion
c. Member data are printed out after input is read
d. Contracted stiffness matrix is printed prior to inversion

e. Eigenvectors of dynamical matrix are printed after eigenvalue routine

f. Eigenvalues of dynamical matrix are printed after eigenvalue routine

g. Dynamical matrix is printed after formation from stiffness matrix

h. Flexibility matrix is printed after inversion of stiffness matrix

i. Reduced stiffness matrix and mass vector are printed after reduction of stiffness matrix to order of dynamical matrix

j. Contents of logic unit 14, flags, properties, stress coefficients, and moments for each member

k. Modal effective mass for dynamic model/solution evaluation.

3.13.3.1.3 Typical Benchmark Calculations

This section defines and references eight benchmark calculations typical of the verification that has been done with ADLPIPE. The solution to each problem from other sources is compared to the ADLPIPE solution.

Type of Analysis	Checks	Reference
Thermal and dead weight combined	Forces, moments, and deflections through the system	Pressure Vessel and Piping/1972 Computer Program Verification ASME, page 6-1
Dynamic	Natural frequencies of a three-dimensional structure. Mode shapes are checked (not published)	Pressure Vessel and Piping/1972 Computer Program Verification ASME, page 1-1
Stress and usage factor	Checks stress range calculation and fatigue usage factor	Sample Analysis of a Piping System – ASME Class 1, Nuclear
Thermal and dead weight (separate)	Checks for forces, moment, deflections, and stresses, per B31.1	“Stress in Three Dimensional Pipe Bends” by W. Hovgaard, Trans. ASME, Volume 57, 1935, pages 401-465
Thermal	Thermal stress per B31.1. Anchor reactions	Design of Piping Systems, M. W. Kellogg Company, page 47
Dynamic	Natural frequencies, model shapes, and response spectra deflections and moments	Shock and Vibration by Young, ASME

Type of Analysis	Checks	Reference
Stress	All stress coefficients (product of stress indices and geometry) used in Section III Class 1 piping Analysis Checks “either/or” logic specified in footnotes to stress indices	Hand calculations
Stress	Checks all stress components and their sum on selected piping	Hand calculations

3.13.3.1.4 Analytical Description Technique

See Reference 21 and 57 through 60.

3.13.3.2 PASS Teledyne Materials Research

3.13.3.2.1 Introduction

The PASS computer program is a postprocessor to the ADLPIPE computer program which provides an elastic analysis of redundant piping systems subjected to thermal, static, and dynamic loads. The program accepts, as input, the ADLPIPE Math Model describing the piping geometry, and the internal forces, moments, and deflections resulting from the flexibility analysis for various load conditions (dead weight, hydrotest, thermal, seismic inertia, and attachment displacements).

The PASS program also functions as a report generator for the hanger selection summary reports. The summary defines the support system and summarizes in a tabular report style format:

a. Nozzle and anchor loads

b. Hanger and restraint loads

c. A stress summary of selected data points in accordance with the rules of NC-3652 for sustained loads - Equation (8); occasional loads - Equation (9); thermal expansion -Equation (10); and Equation (11) for Class 1 and Class 2 components

d. A stress summary in accordance with the rules of NB-3652 for the primary stress-intensity limit - Equation (9), for Class 1 components only.

3.13.3.2.2 Purpose

The purpose of this program is to determine the adequacy of the piping support system for a given ADLPIPE Math Model by evaluating stresses for sustained loads, occasional loads, and thermal expansion in accordance with the design and analysis philosophy of subsections

NB-3652 and NC-3652 in Section III of the ASME B&PV Code. The program also provides load summaries for anchors and restraints, and reports the maximum loads for each load condition and the required net design load.

3.13.3.2.2.1 Method of Solution

The PASS program is designed to read the ADLPIPE Math Model and determine all network point restraints from the restraint cards. Those points restrained in the six degrees of freedom are considered anchors; other restraint points are defined by a restraint code in the respective X, Y, and Z direction on the network point identification cards of the ADLPIPE Math Model. The program then prints out the ADLPIPE Math Model and a restraint summary table of all network point restraints indicating the direction and type of restraint. The outside diameter and thickness for each member, and bend radii for all elbows, are then determined and a table of member geometries printed. Points to be analyzed are subsequently read by the program, which must include all restraint points in the same sequence as they appear in the ADLPIPE Math Model.

The internal forces, moments, and deflections for each load condition are then read by the program and stored for those data points defined on the network point restraint cards and for those points undergoing stress evaluation. As the data for each load condition are read, the program performs a check on the deflection data such that if a card is missing for a point, the forces will be read as deflections causing a diagnostic to be printed for that load condition. If the data check encounters deflections greater than 20 in., the program finishes reading the input data and then terminates the job. The analyst must then correct his data and resubmit the job.

Once the input data have been read in for all load conditions, the net design loads are determined for each anchor point in the following manner.

a. The deadweight loads and hydrotest loads are retrieved for the point of interest

b. The loads for all thermal conditions are scanned and the maximum positive (+) and maximum negative (-) loads for each direction determined

c. The resultants of the deadweight loads, plus maximum positive thermal and maximum negative thermal, are evaluated and the magnitudes compared to the hydrotest loads, if applicable. The greater of the dead-weight plus-thermal or the hydrotest is used in the computation of the net design load

d. The maximum static load, dead-weight-plus-thermal or hydrotest, is summed up with the maximum seismic load (SSE) plus end effects (SSE, inertial plus building movements). Note: See Revision E changes to above (Subsection 3.13.3.2.2.2).

The same procedure is followed for evaluating the net design load on restraints with the exception of springs and snubbers. Only deadweight and hydrotest loads are considered for a spring and only seismic loads for a snubber.

Thermal and seismic displacements are then determined for all restraint points. The thermal displacements are defined by a maximum and minimum range, while the seismic displacements are to be considered plus (+) and minus (-), since for a normal mode analysis the resultant internal forces and moments are computed from the square root of the sum of the squares of the modal forces and moments.

a. <u>Option 1</u>

$$(F_i)^2 = (F_i)_X^2 + (F_i)_Z^2 + |F_i|_Y$$

$$(M_i)^2 = (M_i)_X^2 + (M_i)_Z^2 + |M_i|_Y$$

$$(\delta_i)^2 = (\delta_i)_X^2 + (\delta_i)_Z^2 + |\delta_i|_Y$$

b. <u>Option 2</u>

$$(F_i)^2 = (F_i)_X^2 + (F_i)_Y^2 + (F_i)_Z^2$$

$$(M_i)^2 = (M_i)_X^2 + (M_i)_Y^2 + (M_i)_Z^2$$

$$(\delta_i)^2 = (\delta_i)_X^2 + (\delta_i)_Y^2 + (\delta_i)_Z^2$$

where

i	=	x, y, z
x, y, z	=	response directions
X, Y, Z	=	shock directions

The program then evaluates stresses for Class 2 specified data points in accordance with the rules of NC-3652.

a. Sustained loads (NC-3652.1)

$$\frac{PD_o}{4t} + 0.75i\left(\frac{M_A}{Z}\right) \leq 1.0\ S_h \quad (8)$$

b. Occasional loads (NC-3652.2)

$$\frac{P_{max}D_o}{4t} + 0.75i\frac{(M_A+M_B)}{Z} \leq 1.2\ S_h \quad (9)$$

c. Thermal expansion (NC-3652.3)

$$i\left(\frac{M_c}{Z}\right) \leq S_A \quad (10)$$

$$\frac{PD_o}{4t} + 0.75i\left(\frac{M_A}{Z}\right) + i\frac{M_c}{Z} \leq (S_h + S_A) \quad (11)$$

The primary stress-intensity limit Equation (9) of NB-3652 is evaluated for design conditions of all Class 1 data points specified in the node list for stress analysis.

$$B_1\left(\frac{PD_o}{2t}\right) + B_2\left(\frac{D_o}{2I}\right)M_i \leq 1.5\ S_m \quad (9)$$

For a complete definition of the preceding equations, refer to subsections NB-3652 and NC-3652 of Section III in the ASME B&PV Code. The program currently evaluates Equations (9), (10), and (11) of NC-3652 with and without moments due to secondary end effects (building or equipment movements). The moments produced by such displacements from seismic inertia effects are included with earthquake moments in the evaluation of Equation (9) in NC-3652.1 and Equation (9) in NB-3652.

The stress intensification factor, i, of NC-3652 is determined by the program in accordance with Figure NC-3672.9(a)-1 for Class 2 components, and stress indices B_1 and B_2 of NB-3652 are determined in accordance with Table NB-3683.2-1 for Class 1 components. If the stress intensification factor, i, or stress indices B_1 and B_2 are provided with the stress input data, these factors will override standard values computed by the program. In addition to printing stress summary tables for all specified stress points, the program determines critical points as those points with the greatest stress to allowable ratio for Eqauations (8) and (9) of NC-3652 (Class 2) and Equation (9) of NB-3652 (Class 1).

3.13.3.2.2.2 Revision E - Design Loads

The PASS program has been updated by Revision E to reflect the following method of computing design loads for nozzle/anchor reactions and hanger/restraint reactions:

Maximum Design Load (+):

DL1 = + SEISMIC (DBE) + E.E. (DBE)

DL2 = + SEISMIC (DBE) + E.E. (DBE) + DYNAMIC (+) + DEAD WEIGHT

DL3 = + SEISMIC (DBE) + E.E. (DBE) + DYNAMIC (+) + THERMAL (+) + DEAD WEIGHT

DL4 = MAX.(+) OF (DEAD WEIGHT OR HYDRO) (+) DESIGN LOAD

= MAX. (+) OF DL1, DL2, DL3, DL4

Minimum Design Load (-):

DL1 = - SEISMIC (DBE) - E.E. (DBE)

DL2 = - SEISMIC (DBE) - E.E. (DBE) + DYNAMIC (-) + DEAD WEIGHT

DL3 = - SEISMIC (DBE) - E.E. (DBE) + DYNAMIC (-) + THERMAL (-) + DEAD WEIGHT

DL4 = MAX. (-) OF (DEAD WEIGHT OR HYDRO) (-) DESIGN LOAD

= MAX. (-) OF DL1, DL2, DL3, DL4

3.13.3.2.3 PASS Verification

This section contains the solution comparisons between PASS and independent hand calculations for a sample problem. The comparisons of (1) anchor and nozzle reactions, (2) hanger/restraint reactions and displacement tolerances, and (3) the Class 2 stress evaluation are presented in Tables 3.13-45 through 3.13-48. The results show very close, if not exact, agreement. The tabulated PASS values, except hanger/restraint reactions in Table 3.13-46, apply to both D and E Revisions of PASS. The hanger/restraint reactions shown in Table 3.13-46 were taken from the Revision E version of the program.

The Revision D version of PASS can overcompute hanger/restraint reactions where there are seismic end effects (seismic anchor movements) load cases. For the seismic end effects case, the Revision D version computes the hanger/restraint reactions by taking the absolute sum of resulting pipe loads on either side of these supports, which is conservative, whereas the Revision E version uses the more correct algebraic sum.

3.13.3.3 Dynamic Analysis of Piping Systems

See Reference 61.

3.13.3.4 SAMIS

See Reference 61.

3.13.3.5 MEL

See Reference 61.

3.13.3.6 SAP

See Reference 61.

3.13.3.7 Time-Dependent Pipe Forces

See Reference 61.

3.13.3.8 SAP IV - Structural Analysis Program

See Reference 61.

3.13.3.9 CVPT Report

Refer to Subsection 3.6.3.1.6.

3.13.3.10 TMRSAP

TMRSAP, a computer program owned by Teledyne Engineering Services (TES), is assigned to perform an elastic analysis of complex piping systems subjected to thermal, static, and dynamic loads.

The piping systems are modeled using either of two element types, namely, boundary element or pipe element (tangent and bend). These elements may be used in a static or dynamic analysis. The pipe element is represented by a straight segment (tangent) or a circularly curved segment (bend); both elements require a uniform section and uniform material properties. Elements can be directed arbitrarily in space. The member stiffness matrices account for bending, torsion, axial, and shear deformations. In addition, the effect of internal pressure on the stiffness of curved pipe elements is considered.

The loads contributed by the pipe elements include gravity in the global directions and loads due to thermal distortions and deformations induced by internal pressure. Forces and moments acting at the member ends and at the center of each bend are calculated in coordinate systems aligned with the member's cross section.

The input consists of the piping system geometry, material properties, and static and dynamic loadings.

Various benchmark problem solutions have been used to verify and qualify the TMRSAP program. The solutions of benchmark problems have been compared with closed-form solutions available in the literature or with solutions obtained using other similar codes.

3.13.3.11 TMRPASS

The TMRPASS computer program determines the adequacy of the piping support system for a given TMRSAP structural model by evaluating stresses for sustained loads, occasional loads, and thermal expansion in accordance with the design and analysis philosophy of Subarticles NB-3652 and NC-3652 in Section III of the ASME B&PV Code. The program also provides design load summaries for anchors and restraints, and reports the maximum loads for each load condition and the required net design load.

The program requires as input the TMRSAP structural model describing the piping geometry, as well as the internal forces, moments, and deflections resulting from the flexibility analyses for various load conditions (dead weight, hydrotest, thermal, seismic inertia, and attachment displacements).

The verification and qualification of TMRPASS were performed by comparing TMRPASS output with the results of hand calculations for a typical piping system.

3.13.3.12 ANSYS

ANSYS, engineering analysis system, is a general-purpose computer program with capabilities for transient heat-transfer analyses; static elastic, plastic, creep, dynamic, and dynamic plastic analyses; large deflection and stability analyses; and one- dimensional fluid-flow analyses. The output from the transient heat-transfer analyses is in the form required to do thermal stress analyses at selected time points in the transient with the same analyses models. The program was formulated and developed by Swanson Analysis Systems, Inc.

3.13.3.13 STAAD-III/STAAD.Pro

STAAD-III is a general-purpose structural analysis program marketed by United Information Systems of Kansas City. It performs a static structural analysis of framed structures using the stiffness method of solution. A natural frequency calculation of a structure can be performed by the program as a user option. Internal structure forces, moments, and stresses and nodal displacements and rotations can be output from the analysis portion of the program.

STAAD-III has a postprocessor that performs an evaluation in accordance with the AISC Specification for Structural Steel. It also performs an evaluation of welded connections.

STAAD.Pro is a comprehensive and integrated finite element analysis and design program capable of analyzing structures exposed to static loading, a dynamic response, wind, earthquake, and moving loads. Its analytical capabilities include linear static, response spectra, time history, cable, imperfection, pushover and non-linear analyses. The program is developed by Bentley Systems, Inc. and is an updated version of the STAAD-III program which they obtained when they acquired Research Engineers International.

3.13.3.14 DYNAFLEX

DYNAFLEX is a computer program used to analyze piping systems for static and dynamic loads and to compute the combined stresses. The following analyses are performed:

a. Static - analysis of distributed and concentrated weight, displacement, and thermal loadings on piping systems

b. Dynamic - analysis of piping system response to seismic loads using the uniform response spectrum method

c. Stress combination - computation of the combined stresses in piping components in accordance with the ASME Code Section III, Subarticle NC-3650, or with the ANSI B31.1 Code for Power Piping.

DYNAFLEX is a proprietary program owned, maintained, and supported by Intercomp, Inc., Houston, Texas, and marketed by United Information Systems of Kansas City.

Test problems verifying the accuracy of the results obtained from DYNAFLEX have been run, comparing results with other piping analysis programs such as ADLPIPE and PIPESD. In addition, program updates are verified using a standard series of problems and also specific problems designed to verify the specific updates made to the program.

3.13.3.15 BASEPLT

The program BASEPLT is a preprocessor to the STARDYNE computer code developed for the specific purpose of analyzing flexible baseplates. The BASEPLT preprocessor generates the input runstream, including control cards, for a STARDYNE/SPRING nonlinear solution of a baseplate analysis. The program is marketed and supported by Control Data Corporation, Minneapolis, Minnesota, and available in the public domain.

3.13.3.16 PISYS/ANSI7

These computer programs are used by GE for piping stress analyses and were written by and meet the Quality Assurance Standards of GE. The programs have been approved for production use by a special committee after independent review and verification. All changes to these programs require verification and approval by this committee. The computer program master files are stored in the GE Energy Division archive tapes.

PISYS performs static and dynamic analyses of piping systems. The analysis modules of PISYS were taken directly from the SAP4G program. The ANSI7 program calculates stresses (and cumulative usage factors) for Class 1, 2, and 3 piping components in accordance with Article NB-3600 and Subarticle NC-3652 of ASME Code Section III. This program also calculates combined loads on piping equipment in accordance with the equipment load combinations given in the Piping Design Specification and compares them with the allowable loads.

3.13.3.17 Holtec Computer Programs

All computer programs utilized by Holtec International to perform the analyses documented in this safety analysis report are benchmarked and verified in accordance with Holtec International's Quality Assurance procedures. The significant programs employed are listed and described below.

3.13.3.17.1 DYNARACK

DYNARACK performs dynamic simulations on systems and structures. It is used to simulate rack structure response to seismic excitation.

3.13.3.17.2 ONEPOOL

ONEPOOL is used to predict SFP bulk temperatures. All discharge scenarios and heat exchanger performances can be modeled.

3.13.3.17.3 FLUENT

FLUENT is a computational fluid dynamics code used to determine fluid motion in the SFP.

3.13.3.17.4 THERPOOL

THERPOOL is utilized to evaluate local pool water and fuel cladding temperatures.

3.13.3.17.5 NITAWL

Part of Oak Ridge National Laboratory's SCALE system of computer codes. It collects cross sections from the 238 group master library for specified materials and compiles them into the proper format for input to KENO-5a. It also calculates the shielded resonance cross sections for U-238.

3.13.3.17.6 KENO-5a

KENO-5a calculates the k-effective of spent fuel storage racks in three dimensions.

3.13.3.17.7 MCNP-4A and MCNP-05P†

MCNP is used to evaluate criticality and shielding problems with a high degree of accuracy.

†Original HOLTEC criticality analyses performed using MCNP-4A were subsequently updated using GNF version of MCNP-05.

3.13.3.17.8 CASMO-4

CASMO-4 is used for spatial and burnup calculations.

3.13.3.17.9 ANSYS

ANSYS is used in conjunction with the dynamics simulation code DYNARACK in spent fuel pool structure evaluations. ANSYS has also been used to evaluate seismic class I Reactor Building 1st and 5th floor stresses in response to seismic excitation with ISFSI loads, as well as for analysis of ISFSI component internal structure and support stresses. Also refer to section 3.13.3.12.

3.13.3.18 AutoPIPE

AutoPIPE is a computer aided engineering (CAE) program for calculation of piping stresses, flange analysis, pipe support design, and equipment nozzle loading analysis under static and dynamic loading conditions. In addition to piping codes, AutoPIPE incorporates ASME, British Standard, API, NEMA, ANSI, ASCE, AISC, UBC, and WRC guidelines and design limits to provide comprehensive analysis of the entire system.

AutoPIPE provides unique capabilities for process, power, oil and gas, nuclear, underground, offshore floating, production, storage, and offloading (FPSO) platform and subsea pipeline areas with international piping codes. Advanced AutoPIPE capabilities include built-in wave loading, buried pipeline analysis, jacketed piping, dynamic loadings, orthotropic fiberglass reinforced plastic (FRP/GRP), and high-density polyethylene (HDPE) plastic piping analysis. It also includes thermal stratification or bowing, thermal transient, pipe/structure interaction, fluid transient with closure time and relief valve utilities, advanced load sequencing, non-linear support gaps and friction and jacketed piping. Local stress calculation to WRC 107, WRC 297, PD 5500, KHK, API 650 is available using AutoPIPE Nozzle.

AutoPIPE quality assurance program has been subjected to numerous nuclear and Nuclear Procurement Issues Committee (NUPIC) audits to 10 CFR 50 App. B, ISO9001, CSA N286.7-99, ASME NQA-1, and ANSI N45.2 standards. AutoPIPE Nuclear provides design of critical safety pipework to ASME Class 1, 2, or 3.

3.13.3.19 GT STRUDL

GT STRUDL is a large-scale general purpose structural analysis computer program. The matrix displacement method of analysis based upon finite element idealization is used throughout the program. GT STRUDL has the ability to perform static and dynamic analysis for framed structures and three-dimensional solid structures.

GT STRUDL is used in the analysis and design of nuclear and nonnuclear linear type pipe supports and seismic Category I duct supports.

3.13.4 Computer Programs Used by Stone & Webster, Michigan, Incorporated

Subsections 3.13.4.1 through 3.13.4.4 describe four computer programs used by S&W. They were applied to piping and support design only.

3.13.4.1 NUPIPE - Elastic Piping Analysis with NB, NC 3600 Solutions

NUPIPE is a program for thermal, deadweight, and seismic analysis done in accordance with Subarticle NB3600, NC3600, or ND3600 of Reference 62. It considers stress intensities as specified in Equations 9 through 14 given in the above-mentioned Subarticle NB3600, and also determines the usage factors for points undergoing analysis of normal, upset, emergency, and faulted conditions. This program accepts the complete geometric and physical description of the piping system, provides a complete error and coordinate check for the inputs, and computes internal forces and moments, support and equipment reactions, and displacements and stress values for a variety of loading cases including weight, thermal expansion, applied forces, applied displacements, and earthquakes.

The NUPIPE program has been verified with ADLPIPE (Reference 63) for thermal, weight, and response spectrum seismic analysis. The results from both the programs are presented in Tables 3.13-49 through 3.13-55. The model used for this comparison is presented in Figure 3.13-91.

The comparison is made also with ASME Benchmark Solution (see Reference 64, Problem 5) for force time-history dynamic response. The model used for this comparison is shown in Figure 3.13-92. The results for comparisons are presented in form of plots in Figure 3.13-92. The natural frequencies are given in Table 3.13-56.

The Class 1 piping stress conforms with the hand calculations. The model used is shown in Figure 3.13-93. The results are tabulated in Tables 3.13-57 and 3.13-58.

3.13.4.2 HTLOAD - Heat Loads

3.13.4.2.1 General Description

HTLOAD is a computer program that performs a finite difference method analysis of piping system response to thermal transients of its contained fluid. The output gives overall thermal growth, linear and nonlinear temperature distribution through the pipe wall, gross discontinuity information (T_A-T_B), and Equations 10 and 11 results of Article NB3600 of ASME Section III.

HTLOAD can analyze piping, with or without a thermal sleeve, that is subject to changes in fluid temperature, velocity, and/or state. The properties of subcooled or saturated water and superheated or saturated steam are taken from the ASME steam tables (Reference 65). The pressure range is from 0.45 psia to 6210 psia.

This computer program also performs thermal analysis for pipes with different insulating conditions, ranging from noninsulated to perfectly insulated. It has stored properties for insulation such as unibestos, asbestos, reflective aluminum, reflective stainless, and calcium silicate. Provision is further made for hand input properties of other insulation types.

Also stored in the program are the piping material properties of carbon steel, austenitic stainless, low-chrome steel, high-chrome steel, and nickel-chrome iron for the temperature range of 32°F to 1600°F.

Program input includes piping material insulation information, time lapse for initial to final fluid temperature, calculation time limit, fluid velocities, initial and final temperature and pressure, and pipe and thermal sleeve dimensions.

HTLOAD requires that each thermal transient be input as a step change, a ramp change, or as a twelve-point arbitrary function.

Output results are used in the calculation of piping stress in accordance with Article NB3600 of ASME Section III. HTLOAD also performs the primary, plus secondary, stress intensity range check (Equation 10) and the peak stress intensity range calculation (Equation 11) from Article NB3600.

3.13.4.2.2 Program Verification

The sample problem selected for solution by HTLOAD consists of a 2-in. Schedule 160, stainless steel pipe with one end connected to a 1/2-in.-thick socket-welded fitting. Saturated water flowing within the piping system changes temperature from 400°F to 500°F in a period of 10 sec. Velocity of fluid is 7560 ft/hr. Input properties are listed in Tables 3.13-59 and 3.13-60.

Reynolds number and heat-transfer coefficients are compared with hand calculations (Reference 66) and are given in Table 3.13-61.

Comparison between HTLOAD and Brock and McNeill's charts (Reference 67) for ΔT_1 and ΔT_2 is given in Table 3.13-62. Table 3.13-63 represents the comparison between TRHEAT (Reference 68) and HTLOAD for ΔT_1, ΔT_2, and T_A-T_B.

3.13.4.3 PITRUST

PITRUST is a program to calculate local stresses in the pipe caused by cylindrical welded attachments under external loadings. This program uses the Bijlaard method, as published in Reference 69, to calculate local stresses in the pipe wall caused by cylindrical welded attachments under external loadings, including pressure, dead load, and combinations of maximum seismic reactions.

Program PITRUST has been verified by comparing its solution of a test problem to the solution of the same problem by an independently written piping local stress program, CYLNOZ, in the public domain. The CYLNOZ piping local stress program was written by Franklin Institute (Philadelphia, Pennsylvania) and is used presently by engineering companies. The test problem is of a 72.375-in. O.D. x 0.375-in.-thick run pipe, reacting under an external loading condition of 1000 lb force (normal and shear) and 1000 in.-lb bending and torsional moments transmitted by a 16-in.-O.D. nozzle. A comparison of results is tabulated in Table 3.13-64. Program PITRUST has been verified also by comparing its solution of the test problem to the experimental results obtained in Reference 70. A comparison of these results is tabulated in Table 3.13-65.

3.13.4.4 PILUG

PILUG is a program to calculate local stresses in the pipe wall caused by rectangular welded attachments under external loadings. This program uses the Bijlaard method, as described in

Reference 69, to calculate local stresses in pipe walls caused by rectangular welded attachments under external loadings, including pressure, dead load, and combinations of maximum seismic reactions.

Program PILUG has been verified by comparing its solution of a test problem to results obtained by hand calculations using the formulations specified in Reference 69. A comparison of results is tabulated in Table 3.13-66.

3.13 COMPUTER PROGRAMS FOR STRUCTURAL ANALYSIS AND DESIGN

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3.13 COMPUTER PROGRAMS FOR STRUCTURAL ANALYSIS AND DESIGN

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TABLE 3.13-1 COMPARISON OF DSASS V, DYNAS, MASS IV, AND MASS V RESULTS WITH BIGGS

Mode Number	Biggs	DSASS V, DYNAS MASS IV, MASS V
	Structural Frequency (Hz)	
1	1.00	1.00
2	2.18	2.18
3	3.18	3.18
	Probable Maximum Story Displacement (in.)	
1	1.50	1.51
2	3.22	3.20
3	4.86	4.68
	Absolute Maximum Story Shear (kip)	
1	3020	3010
2	2080	2068
3	1345	1353
	Probable Maximum Story Shear (kip)	
1	2250	2262
2	1740	1757
3	895	902

TABLE 3.13-2 NATURAL PERIODS FOR THE EIGHT LOWEST FLEXURAL MODES

	Periods in Seconds	
Mode Number	SAP IV	DYNAS, MASS IV, MASS V
1	525.79	525.69
2	85.368	85.369
3	30.965	30.964
4	16.059	16.060
5	9.9006	9.9010
6	6.8276	6.8279
7	5.1865	5.1866
8	4.3777	4.3778

TABLE 3.13-3 INDIA SAMPLE PROGRAM

INTERACTION DIAGRAM AXIAL LOAD VS. BENDING MOMENT REFERRED TO THE PLASTIC CENTROID OF THE SECTION.

LIST OF SYMBOLS

B	= WIDTH OF SECTION, IN.
T	= HEIGHT OF SECTION, IN.
D	= DEPTH OF TENSILE STEEL, IN.
AS	= AREA OF TENSILE STEEL, SQ IN.
DC	= DEPTH OF COMPRESSION STEEL, IN.
ASC	= AREA OF COMPRESSION STEEL, SQ IN.
ES	= MODULUS OF ELASTICITY OF REINFORCING STEEL, KSI
SSY	= YIELD STRESS OF REINFORCING STEEL, KSI
SSU	= ULTIMATE STRESS OF REINFORCING STEEL, KSI
PRESTR	= INITIAL PRESTRAIN OF REINFORCING STEEL
ULTSTR	= ULTIMATE STRAIN OF REINFORCING STEEL
CUS	= 28 DAY STRENGTH OF CONCRETE CYLINDER, KSI
EPSZ	= CONCRETE STRAIN FOR MAXIMUM STRESS
EPSU	= CONCRETE STRAIN AT CRUSHING
NSTESS	= NUMBER OF POINTS IN INTERACTION DIAGRAM
EPEL	= MAXIMUM TOP STRAIN FOR WHICH ID IS COMPUTED

INPUT DATA

B = 12.0000	T = 43.0000	D = 45.0000	AS = 2.7500	DC = 10.0000	ASC = 1.2500
SS = 23000.0000	SSY = 80,0000	SSU = 30,000	ULTSTR = 0.020000	PRESTR = 0.000000	
CUS = 4.5000	EPSZ = 0.002000	EPSU = 0.004000			
NSTESS = 20	EPSL = 0.003000				

RESULTS GIVEN IN THE FOLLOWING ORDER

COUNTER	CURVATURE	TOP STRAIN	AXIAL LOAD (KIP)	BENDING MOMENT (KIP-FT)	AXIAL LOAD DIMENSIONLESS	BENDING MOMENT DIMENSIONLESS	C.R. FACTOR PHI	REDUCED AXIAL LOAD (KIP)	REDUCED BENDING MOMENT (KIP-FT)
INITIAL POINT UNDER UNIFORM COMPRESSION STRAIN + EPSZ									
1	0.00000000	0.00200000	2419.3999	6.0875	1.0984	0.0007	0.8981	2173.3915	5.4674
2	0.00001762	0.00225000	2349.4036	96.0163	1.0654	0.0109	0.8982	2110.2042	86.2406
3	0.00003125	0.00250000	2216.4314	258.0801	1.0080	0.0293	0.8983	1990.9977	231.8123
4	0.00004667	0.00275000	2023.3391	485.1479	0.9184	0.0562	0.8984	1817.8463	444.8601
5	0.00003268	0.00000000	1770.1285	807.2801	0.8034	0.0916	0.8985	1590.8962	725.4495
PRECEDING POINT HAD BOTTOM FIBER STRAIN = ZERO									
6	0.00007316	0.00300000	1377.5877	1191.7730	0.6253	0.1352	0.8389	1238.3646	1071.3288
7	0.00000966	0.00300000	1163.2063	1325.1268	0.5280	0.1504	0.8991	1045.2417	1191.4245
8	0.00010115	0.00300000	385.3843	1400.5915	0.4472	0.1589	0.8992	386.6782	1259.4674
9	0.00011264	0.00300000	833.6598	1447.7928	0.3764	0.1043	0.8904	743.7575	1362.0929
BALANCED POINT, TENSILE STEEL STRAIN = -EPSY									
10	0.00016243	0.00300000	518.9094	1288.7116	0.2355	0.1462	0.3996	466.8167	1159.3245
11	0.00021226	0.00300000	339.3652	1137.9047	0.1540	0.1291	0.8997	305.3398	1023.8163
12	0.00020267	0.00300000	218.2740	1021.0822	0.0991	0.1159	0.8998	195.4099	919.7340

TABLE 3.13-3 INDIA SAMPLE PROGRAM

13	0.00031188	0.00300000	127.3296	930.6383	0.0573	0.1056	0.9999	114.5042	637.4870
14	0.00038159	0.00300000	52.4308	857.1847	0.0230	0.0973	0.9000	47.1876	771.4405
15	0.00041143	0.00300000	-11.2036	797.4842	-0.0051	0.0905	0.9000	-10.0834	717.7247
16	0.00043130	0.00300000	-67.2224	747.8149	-0.0305	0.0849	0.9001	-80.0000	673.0722
PRECEDING POINT HAD TENSILE STEEL STRAIN = -ULSTR									
17	0.00044047	0.00200001	-174.2335	571.9126	-0.0791	0.0649	0.9001	-156.8736	514.7982
18	0.00041954	0.00100000	-269.5452	392.8982	-0.1223	0.0446	0.9002	-242.6467	353.6981
19	0.00038380	0.00000000	-317.0636	209.7994	-0.1439	1.6329	0.9002	-285.4348	269.8928
20	0.00000000	-0.02000000	-360.0000	273.9367	-0.1634	0.0311	0.9003	-380.0080	273.9357

DISTANCE OF PLASTIC CENTROID TO BOTTOM FIBER, IN. = 24.9313

AS AND ASC CHANGED VALUES, READ MOMENT WITH OPPOSITE SIGN ************************************

INPUT DATA

B = 12.0000	T = 48.0000	D = 45.0000	AS = 1.2500	DC = 10.0000	ASC = 2.7500
ES = 29000.0000	SSY = 60.0000	SSU = 90.0000	ULTSTR = 0.020000	PRESTR = 0.000000	
CUS = 4.5000	EPSZ = 0.002000	EPSU = 0.004000			
NSTESS = 20	EPSL = 0.003000				

RESULTS GIVEN IN THE FOLLOWING ORDER

COUNTER	CURVATURE	TOP STRAIN	AXIAL LOAD (KIP)	BENDING MOMENT (KIP-FT)	AXIAL LOAD DIMENSIONLESS	BENDING MOMENT DIMENSIONLESS	C.R. FACTOR PHI	REDUCED AXIAL LOAD (KIP)	REDUCED BENDING MOMENT (KIP-FT)
INITIAL POINT UNDER UNIFORM COMPRESSION STRAIN + EPSZ									
1	0.00000000	0.00200000	2419.8999	-1.8527	1.0984	-0.0002	0.8981	2173.3915	-1.6640
2	0.00001562	0.00225000	2371.9226	64.3211	1.0766	0.0073	0.8982	2130.3883	57.7713
3	0.00003125	0.00225000	2258.0534	206.7475	1.0249	0.0235	0.8983	2038.3176	185.7126
4	0.00004537	0.00275000	2083.4750	431.3483	0.9457	0.0489	0.8984	1871.7781	387.5200
5	0.00008250	0.00300000	1843.1875	738.1239	0.8389	0.0838	0.8986	1660.7331	663.2580
PRECEDING POINT HAD BOTTOM FIBER STRAIN = ZERO									
6	0.00007816	0.00300000	1484.8191	1109.2716	0.6739	0.1259	0.8989	1334.6381	997.0735
7	0.00008866	0.00300000	1292.5999	1224.0333	0.5867	0.1389	0.8990	1162.0587	1100.4092
8	0.00010115	0.00300000	1133.6276	1273.4436	0.5145	0.1445	0.8991	1819.2732	1144.9354
9	0.00011264	0.00300000	999.4452	1290.4057	0.4536	0.1464	0.8992	898.7299	1150.3700
BALANCED POINT, TENSILE STEEL STRAIN = -EPSY									
10	0.00016245	0.00300000	669.1896	1129.0560	0.3037	0.1281	0.8995	601.9251	1015.5674
11	0.00021226	0.00300000	474.8520	963.1077	0.2155	0.1093	0.8996	427.1928	866.4440
12	0.00026207	0.00300000	339.6791	826.0203	0.1542	0.0937	0.8997	305.6222	743.2018
13	0.00031189	0.00300000	234.6630	712.3114	0.1065	0.0808	0.8998	211.1542	640.9513
14	0.00038169	0.00300000	143.7225	611.7068	0.0652	0.0694	0.8999	129.3743	550.4683
15	0.00041149	0.00300000	64.0464	523.6752	0.0291	0.0594	0.9000	57.8385	471.2818
16	0.00046130	0.00300000	-8.0140	444.9537	-0.0036	0.0505	0.9000	-7.2127	400.4611
PRECEDING POINT HAD TENSILE STEEL STRAIN = -ULTSTR									

TABLE 3.13-3 INDIA SAMPLE PROGRAM

17	0.00044047	0.00200001	-135.5442	255.9720	-0.0615	0.0290	0.9001	-122.0039	230.4015
18	0.00041964	0.00100000	-232.6858	84.0563	-0.1056	0.0095	0.9002	-209.4590	75.6658
19	0.00038880	0.00000000	-282.0341	-6.7816	-0.1280	-0.0369	0.9002	-253.8921	-6.1048
20	0.000000	-0.02000000	-360.0000	-83.3721	-0.1634	-0.0035	0.9003	-360.0000	-93.3721

DISTANCE OF PLASTIC CENTROID TO BOTTOM FIBER, IN. = 23.7166

SARGENT & LUNDY
ENGINEERS
CHICAGO

INTERACTION DIAGRAM - P VS. M ABOUT C.G. OF UNCRACKED TRANSFORMED SECTION

YIELD-STRENGTH THEORY

WIDTH OF SECTION (IN.)	=	12.000	AREA OF TENSILE STEEL (IN.)	=	2.750
HEIGHT OF SECTION (IN.)	=	48.000	DEPTH OF TENSILE STEEL (IN.)	=	45.000
ELASTIC MODULUS, STEEL (KSI)	=	29000.	AREA OF COMPRESSIVE STEEL (IN.)	=	1.250
ELASTIC MODULUS, CONCRETE (KSI)	=	3865.	DEPTH OF COMPRESSIVE STEEL (IN.)	=	10.000

28-DAY STRENGTH OF CONCRETE CYLINDER (KSI)	=	4.500
YIELD STRESS FOR REINFORCING STEEL (KSI)	=	54.000
DEPTH OF C.G. OF UNCRACKED TRANSFORMED SECTION (IN.)	=	24.435
MODULAR RATIO	=	8.000
MAXIMUM STRESS OF CONCRETE (KSI)	=	3.925
MAXIMUM STRESS OF REINFORCING STEEL (KSI)	=	48.600
UNIT WEIGHT OF CONCRETE (LB/CU FT)	=	145.000

PHI1 IS THE CAPACITY REDUCTION FACTOR

POSITION NO.	AXIAL LOAD (KIP)	BENDING MOMENT (KIP-FT)	PHI1	REDUCED AXIAL LOAD (KIP)	REDUCED BENDING MOMENT (KIP-FT)
1	2302.69	0.00	0.8992	2068.33	0.00
2	1130.49	796.74	0.8891	1016.45	716.37
3	1056.93	841.20	0.8992	950.38	756.39
4	262.26	846.88	0.8398	235.88	752.02
5	-147.15	212.81	0.0001	-132.45	191.55
6	-194.40	155.97	0.9001	-174.99	140.40
7	-71.30	-54.99	0.9001	-04.18	-49.50
8	-51.38	-90.70	0.9000	-48.24	-81.83
9	316.89	-674.33	0.8998	285.12	-805.73
10	940.50	-908.37	0.8983	845.77	-817.77
11	1172.20	-796.81	0.8091	1053.92	-716.23
12	2302.69	0.00	0.8982	2068.33	0.00

TABLE 3.13-3 INDIA SAMPLE PROGRAM

SARGENT & LUNDY
ENGINEERS
CHICAGO

INTERACTION DIAGRAM - P VS. M ABOUT C.G. OF UNCRACKED TRANSFORMED SECTION

WORKING STRESS DESIGN METHOD

WIDTH OF SECTION (IN.)	=	12.000	AREA OF TENSILE STEEL (IN.)	=	2.750
HEIGHT OF SECTION (IN.)	=	48.000	DEPTH OF TENSILE STEEL (IN.)	=	45.000
ELASTIC MODULUS, STEEL (KSI)	=	29000.	AREA OF COMPRESSIVE STEEL (IN.)	=	1.250
ELASTIC MODULUS, CONCRETE (KSI)	=	3865.	DEPTH OF COMPRESSIVE STEEL (IN.)	=	10.000

ALLOWABLE STRESS OF CONCRETE IN BENDING (KSI)	=	2.700
ALLOWABLE STRESS IN REINFORCING STEEL (KSI)	=	20.000

POSITION NO.	AXIAL LOAD (KIP)	BENDING MOMENT (KIP-FT)
1	1717.20	0.00
2	824.64	618.32
3	768.37	653.56
4	340.10	659.09
5	-60.58	34.98
6	-80.00	60.75
7	-29.34	-23.89
8	-21.14	-38.23
9	389.06	-827.77
10	718.18	-704.17
11	892.56	-618.00
12	1717.20	0.00

TABLE 3.13-4 COMPARISON OF STIFFNESS

Element	Stiffness Program (SSANA) (kip-ft)	Hand Calculations (kip-ft)
1	398821	398880
2	398821	398880
3	398821	398880
4	398821	398880

TABLE 3.13-5 COMPARISON OF WEIGHT MOMENT OF INERTIA ABOUT X-AXIS (Ip)

Element	Weight Inertia Program (SSANA)	Hand Calculations (Ip)
1	2005	2005
2	531	531.25
3	531	531.25
4	32	31.5

TABLE 3.13-6 TEMCO SAMPLE PROBLEM

Section and Material Properties	Problem Number 1	2	3
Thickness, in.	42.0	30.0	42.0
Width, in.	12.0	12.0	12.0
Area of 1st steel layer, in.^{2}	6.25	2.25	3.12
Distance of 1st steel layer, in.	3.0	3.0	3.0
Area of 2nd steel layer, in.^{2}	6.25	4.0	3.12
Distance of 2nd steel layer, in.	37.0	25.0	37.0
Concrete unit weight, lb/ft^3	150.0	150.0	150.0
Concrete compressive strength, lb/in.^{2}	4000.0	4000.0	4000.0
Concrete coef. of thermal expansion, in./°F	5.56×10^{-6}	5.56×10^{-6}	5.56×10^{-6}
Steel yield strength, kip/in.^{2}	45.0	45.0	45.0
Steel modulus of elasticity, kip/in.^{2}	29000.0	29000.0	29000.0
Material properties	Nonlinear	Nonlinear	Linear
Applied axial force, kip	-38.25	76.53	34.65
Applied bending moment, kip-ft	129.75	-9.49	206.25
Inside temperature, °F	82.50	67.50	247.50
Outside temperature, °F	52.50	0.0	115.50

TABLE 3.13-7 TEMCO SAMPLE PROBLEM - RESULTS

Results	Problem Number 1	2	3
Equilibrating axial force given by program, kip	-38.25	76.53	34.65
Equilibrating axial force computed by hand, kip	-38.253	76.53	34.65
Equilibrating bending moment given by program, kip-ft	129.75	-9.49	206.25
Equilibrating bending moment computed by hand, kip-ft	129.752	-9.493	206.25
Thermal moment given by program, kip-ft	-54.58	-21.07	-137.75
Thermal moment computed by hand, kip-ft	-54.585	-21.071	-137.757

TABLE 3.13-8 CABLE PAN ANALYSIS

		Section Modulus	
	Area (in.^{2})	S_x Vert. (in.^{3})	S_y Horiz. (in.^{3})
CAPAN	1.62	1.18	5.96
Hand calculation	1.62	1.12	5.96

TABLE 3.13-9 COMPUTED STRESSES IN MEMBERS

	Results (ksi)	
Member	RIGHAN	Hand Calculation
Vertical	30.053	30.016
Horizontal	29.237	29.210

TABLE 3.13-10 ROLLED BEAM DESIGN PROBLEM

	Maximum Moments(kip-ft)	Section Selected	Section Modulus(in.^{3})
AISC	125	W16x40	64.6
STAND	125.58	W18x40	68.4

TABLE 3.13-11 COMPOSITE BEAM DESIGN PROBLEM

	Bending Moments (kip - ft)		Maximum	Steel	No. of Shear
	Construction Load	Design Load	Shear (kip)	Section	Connectors
AISC	71.3	237.2	26.4	W21x44	42
STAND	71.3	236.5	26.3	W21x44	42

TABLE 3.13-12 COLUMN DESIGN PROBLEM

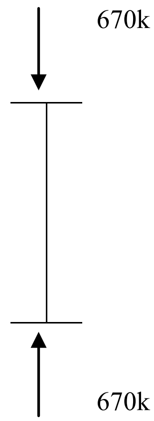
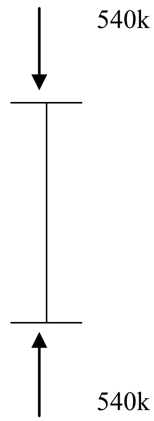
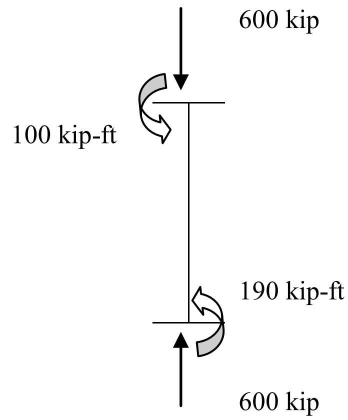
Items	AISC Example 1	AISC Example 2	AISC Example 5
Column design parameters			
AISC solution	W12x161	W12x99	W14x142
STAND solution	W12x161	W12x99	W14x142

TABLE 3.13-13 PLATE GIRDER DESIGN PROBLEM

Results	AISC	STAND
Maximum bending moment (kip-ft)	2054	2045
Maximum vertical shear (kip)	142	141.3
Web Section	1 plate, 70x5/16	1 plate, 70x5/16
Flange section	2 plates, 18x3/4	2 plates, 18x3/4
Stiffener end spacing (ft)	3.5	3.56
Stiffener intermediate spacing (ft)	6.75	6.72
Area of stiffeners furnished (in.^{2})	2.0	1.88

[a] Required area is 1.78 in.^{2}

TABLE 3.13-14 COMPARISON OF MEMBER AND MOMENTS

Stress Lag	Moments from Reference 19 (kip-ft)	Moments from PIPSYS (kip-ft)
M_{AB}	106.0	102.8
M_{BA}	72.0	72.5
M_{BC}	133.0	131.8
M_{CB}	133.0	131.8
M_{CD}	-133.0	-131.8
M_{DC}	-133.0	-131.8
M_{DE}	133.0	131.8
M_{ED}	86.0	84.2
M_{BE}	-158.0	-156.6
M_{EB}	-158.0	-156.6
M_{FE}	106.0	102.8
M_{EF}	72.0	72.5

TABLE 3.13-15 SUMMARY OF LOAD SETS AT GIRTH BUTT WELD WITH CHANGE IN MATERIAL AND WALL THICKNESS, LOCATION 19

Load Set No.	Load Set Description	No. of Transients	F	M_x	M_y	M_z	ΔT_1	T_f (Valve)	T_b (Pipe)	ΔT_2
1	Zero	5	0	0	0	0	0	70	70	0
2	Cold Hydro Test		3590	0	0	0	0	70	70	0
3	Hot Hydro Test, Up	40	2200	251.7	141.6	-7.1	2.4	400	400	0.3
4	Hot Hydro Test, Down		0	0	0	0	-2.4	70	94	-0.3
5	Plant Startup	100	2200	337.2	184.9	-936.0	0	70	70	0
6	Plant Shutdown		0	0	0	0	0	70	70	0
7	Plant Loading	18,300	2200	381.6	204.4	-1169.6	0	70	70	0
8	Plant Unloading		2200	337.2	184.9	-936.0	0	70	70	0
9	Loss of Load, 4.1	80	2515	384.2	204.4	-1183.4	0	70	70	0
10	Loss of Load, 4.2		1500	345.7	186.4	-1011.4	0	70	70	0
11	M.O. + Earthquake	50	2200	408.6	463.3	-1134.1	0	70	70	0
12	M.O. - Earthquake		2200	265.8	-93.5	-737.9	0	70	70	0

TABLE 3.13-16 SIX HIGHEST VALUES OF S_n, GIRTH BUTT WELD WITH CHANGE IN MATERIAL AND WALL THICKNESS, LOCATION 19

Load Set Pair		Values from Reference 20 S_n	Eq. (12)	Eq. (13)	K_a	PIPSYS program S_n	Eq. (12)	Eq. (13)	K_a
3	4	52549	(*)[a]	(*)	1.000	52600	(*)	(*)	1.000
3	9	49883	(*)	(*)	1.000	49900	(*)	(*)	1.000
3	10	49620	(*)	(*)	1.000	49600	(*)	(*)	1.000
3	6	48013	(*)	(*)	1.000	48000	(*)	(*)	1.000
1	3	48013	(*)	(*)	1.000	48000	(*)	(*)	1.000
3	11	47728	(*)	(*)	1.000	47700	(*)	(*)	1.000

[a] Because S_n, calculated by Equation (10), is less than $3S_m$, Equations (12) and (13) are satisfied.

TABLE 3.13-17 SUMMARY OF CALCULATIONS OF CUMULATIVE USAGE FACTOR, GIRTH BUTT WELD WITH CHANGE IN MATERIAL AND WALL THICKNESS, LOCATION 19

Load Set Pair		Values Based On Reference 20		Values from PIPSYS Program	
i	j	$\frac{S_p K_e}{2}$	Usage Factor	$\frac{S_p K_e}{2}$	Usage Factor
3	9	40338	0.0050	40300	0.005
4	9	34400	0.0029	34400	0.003
1	11	29806	0.0002	29800	0.000
6	11	29806	0.0020	29800	0.002
6	7	29163	0.0023	29200	0.002
2	10	26254	0.0002	26300	0.000
10	12	93170	0.0000	93200	0.000
Cumulative Usage Factor			0.0126		0.0124

TABLE 3.13-18 MODAL FREQUENCIES (Hz)

Mode No.	PIPSYS	NASTRAN	DYNAL
1	6.07	6.085764	6.0821088
2	10.69	10.94144	10.936468
3	11.48	11.66862	11.666215
4	14.76	15.20947	15.204282
5	20.12	22.25613	22.135260
6	23.87	28.53255	28.505264
7	25.32	30.58105	30.530972
8	28.80	31.22073	31.190062
9	30.00	32.27319	32.199679
10	42.39	43.14653	43.135100
11	42.95	43.50436	43.497053
12	58.02	58.19336	57.991710
13	77.78	76.62025	71.996751
14	90.74	93.69710	92.12974
15	91.8	96.04482	95.167976
16	93.39	97.81956	97.410131
17	96.96	99.40727	98.209594
18	101.42	104.6169	101.64513
19	102.14	105.4910	103.80206
20	103.03	107.7136	107.52304

TABLE 3.13-19 ALLOWABLE SHEAR, MOMENT, AND SPAN OF CABLE TRAY - COMPARISON OF RESULTS FROM SEISHANG AND HAND CALCULATION

	SEISHANG	Hand Calculation
Vertical shear, static, kip	16.05	16.05
Positive bending moment, static, kip-in.	50.64	50.83
Negative bending moment, static kip-in.	57.62	57.64
Vertical shear, seismic, kip	20.84	20.81
Horizontal shear, seismic, kip	12.84	12.83
Positive bending moment, seismic kip-in.	67.51	67.61
Negative bending moment, seismic kip-in.	76.83	76.82
Horizontal bending moment, seismic kip-in.	153.61	153.59
Span, ft	20.78	20.75

TABLE 3.13-20 CEILING-MOUNTED SUPPORT - COMPARISON OF RESPONSES FROM SEISHANG AND DYNAS

		SEISHANG	DYNAS
Horizontal period, sec		0.1742	0.1765
Vertical period, sec		0.0092	0.0093
Forces and moments due to horizontal seismic			
Vertical element (No. 1)	axial, lb	1600	1607
	shear, lb	770	772
	bending, lb-in.	17100	17208
Horizontal element (No. 9)	axial, lb	25	26
	shear, lb	302	304
	bending, lb-in.	10900	10944
Forces and moments due to vertical seismic			
Vertical element (No. 1)	axial, lb	383	340
	shear, lb	0	2
	bending, lb-in.	30	24
Forces and moments due to dead load			
Vertical element (No. 1)	axial, lb	776	774
	shear, lb	0	0
	bending, lb-in.	30	0

TABLE 3.13-21 WALL-MOUNTED SUPPORT - COMPARISON OF RESPONSES FROM SEISHANG AND DYNAS

		SEISHANG	DYNAS
Horizontal period, sec		0.0067	0.0067
Vertical period, sec		0.1065	0.1080
Forces and moments due to horizontal seismic			
Vertical element (No. 6)	axial, lb	0	1
	shear, lb	2	2
	bending, lb-in.	35	48
Horizontal element (No. 11)	axial, lb	101	105
	shear, lb	2	2
	bending, lb-in.	23	24
Forces and moments due to vertical seismic			
Vertical element (No. 6)	axial, lb	39	0
	shear, lb	131	128
	bending, lb-in.	2700	2676
Forces and moments due to dead load			
Vertical element (No. 1)	axial, lb	717	702
	shear, lb	303	329
	bending, lb-in.	4910	5208

TABLE 3.13-22 INTERACTION COEFFICIENTS OF THE CEILING - MOUNTED SUPPORT - COMPARISON OF RESULTS FROM SEISHANG AND PIPSYS

INTERACTION COEFFICIENT	SEISHANG	PIPSYS
Vertical element (No. 2)	0.617	0.620
(No. 5)	0.520	0.516
Horizontal element (No. 6)	0.683	0.678
Brace element (No. 3)	0.569	0.553

TABLE 3.13-23 COMPARISON OF SEISHANG AND PIPSYS

ANALYSIS RESULTS FOR HANGER SHOWN IN FIGURE 3.13-66

				SEISHANG	PIPSYS
Highest period, sec				0.2349	0.2349
Lowest period calculated, sec				0.0281	0.0281
Forces/moments/disp.					
Load	Element/Node	End	Force/Moment/Disp.		
Dead load	1	i	Axial, lb	-1047	-1047
(a vertical element)		i	Shear, b, lb	1	1
		i	Bending, c ft-lb	0	0
	9	j	Axial, lb	361	361
(a horizontal element)		j	Shear, c, lb	0	0
		j	Bending, c, ft-lb	5	5
	Node 36		y-disp	-0.013	-0.013
Seismic	1	i	Axial, lb	933	933
(a vertical)		i	Shear, b, lb	460	460
		i	Bending, c ft-lb	2596	2596
	9	j	Axial, lb	517	517
(a horizontal)		j	Shear, c, lb	463	463
		j	Bending, b, ft-lb	515	515
	Node 36		z-disp	0.123	0.123

TABLE 3.13-24 INTERACTION COEFFICIENT CALCULATED FOR HANGER SHOWN IN FIGURE 3.13-66: COMPARISON OF SEISHANG RESULTS AND HAND CALCULATIONS

Loading	Type Interaction Equation	Member	Interaction	
			SEISHANG	Hand Calculation
Dead weight and dynamic	Tension and bending	1- end i	0.549	0.549
	Compression and bending	1- end i	0.514	0.514
	Compression and bending	9- end j	0.244	0.243
Dead weight	Tension and bending	1- end i	0.030	0.030

TABLE 3.13-25 COMPARISON OF PFRAME VERSUS BEER AND JOHNSTON FOR CONTINUOUS BEAM PROBLEM

Supports	Support Reactions (k) PFRAME	Support Reactions (k) Beer & Johnston
B	23	23
E	7	7

Forces at Joints	Shear (k)		Bending Moment (k-ft)	
	PFRAME	Beer & Johnston	PFRAME	Beer & Johnston
A	-8	-8	-0.0000095	0
B	+15	+15	-40	-40
C	+5	+5	+50	+50
D	-7	-7	+70	+70
E	+7	+7	+0.0000019	0

TABLE 3.13-26 COMPARISON OF PFRAME VERSUS GERE AND WEAVER FOR PLANE FRAME PROBLEM

Joint Displacements/Rotations (in.)

	PFRAME			Gere & Weaver		
Joints	D_x	D_y	R_3	D_x	D_y	R_3
1(A)	0	0	0	0	0	0
2(B)	-0.020261	0.099359	-0.0017976	-0.02026	-0.09936	-0.001797
3(C)	0	0	0	0	0	0

Support Reactions (k, in.)

Support	PFRAME			Gere & Weaver		
Joints	F_x	F_y	M_z	F_x	F_y	M_z
1(A)	20.261	13.138	436.64	20.26	13.14	436.6
2(B)	-20.261	40.862	-889.52	-20.26	40.86	-889.5

Member Force (k, in.)

Member		PFRAME			Gere & Weaver		
	Joint	F_x	F_y	M_z	F_x	F_y	M_z
M1	2(B)	20.261	13.138	436.64	20.26	13.14	436.6
	1(A)	-20.261	10.862	-322.86	-20.26	10.86	-322.9
M2	1(A)	28.726	-4.5333	-677.14	28.72	-4.52	-677.1
	3(C)	-40.726	20.533	-899.52	-40.73	20.53	-899.5

TABLE 3.13-27 UNCOPED SLIDING CONNECTION FIELD BOLTED ANGLE DETAIL NO. 7.7.1 (GENERAL CRITERIA - OBE)

			Connections			Hand Calculation		
Item	Member	Units	Actual	Allowable	Ratio	Actual	Allowable	Ratio
Shear (forces/stresses	Bolt in-leg	kips	1.92	10.08	0.19	1.92	10.08	-
	Bolt out-leg	kips	2.52	10.52	0.24	2.52	10.50	-
	Angle in-leg (FY)	ksi	2.80	14.40	0.19	2.80	14.40	-
	Angle in-leg (FU)	ksi	4.07	17.40	0.23	4.07	17.40	-
	Angle out-leg(FY)	ksi	2.79	14.40	0.19	2.79	14.40	-
	Angle out-leg(FU)	ksi	4.06	17.40	0.23	4.06	17.40	-
	Beam web (FY)	ksi	3.37	14.40	0.23	3.37	14.40	-
	Beam web (FU)	ksi	5.43	17.40	0.31	5.43	17.40	-
Bending and axial	Angle out-leg	-		-	2.64		-	2.640
	Beam Web	-		-	0.89		-	0.894
Prying action	Angle out-leg	kips	4.17	26.46	-	4.17	26.45	-

	Member	Actual Minimum	Required Minimum
Edge distance	Beam (parallel to slot)	1.50 in.	1.50 in.
	Beam (normal to slot)	3.47 in.	1.13 in.
	Angle in-leg (rolled edge)	2.50 in.	1.34 in.
	Angle in-leg (sheared edge)	1.50 in.	1.25 in.
	Angle out-leg (rolled edge)	1.41 in.	1.13 in.
	Angle out-leg (sheared edge)	1.50 in.	1.28 in.

Connection adequacy = not adequate

TABLE 3.13-28 COPED SLIDING CONNECTION FIELD BOLTED ANGLE DETAIL NO. 7.7.1 (GENERAL CRITERIA - OBE)

			Connections			Hand Calculation		
Item	Member	Units	Actual	Allowable	Ratio	Actual	Allowable	Ratio
	Bolt in-leg	kips	1.92	10.08	0.19	1.92	10.08	-
	Bolt out-leg	kips	2.52	10.52	0.24	2.52	10.50	-
	Angle in-leg (FY)	ksi	2.80	14.40	0.19	2.80	14.40	-
	Angle in-leg (FU)	ksi	4.07	17.40	0.23	4.07	17.40	-
Shear (forces/stresses	Angle out-leg (FY)	ksi	2.79	14.40	0.19	2.79	14.40	-
	Angle out-leg (FU)	ksi	4.06	17.40	0.23	4.06	17.40	-
	Beam web (FY)	ksi	3.44	14.40	0.24	3.44	14.40	-
	Beam web (FU)	ksi	5.43	17.40	0.31	5.43	17.40	-
Bending and axial	Angle out-leg	-	-	-	4.44		-	4.44
	Beam Web	-	-	-	2.75		-	2.75
Prying action	Angle out-leg	kips	4.17	26.46	-	4.17	26.46	-

	Member	Actual Minimum	Required Minimum
Edge distance	Beam (parallel to slot)	1.50 in.	1.50 in.
	Beam (normal to slot)	2.22 in.	1.50 in.
	Angle in-leg (rolled edge)	2.50 in.	1.34 in.
	Angle in-leg (sheared edge)	1.50 in.	1.25 in.
	Angle out-leg (rolled edge)	1.41 in.	1.13 in.
	Angle out-leg (sheared edge)	1.50 in.	1.25 in.

Block Shear	Beam web	-	-	-	0.12	-	-	0.124

Connection adequacy = not adequate

TABLE 3.13-29 UNCOPED SLIDING CONNECTION FIELD WELDED PLATE DETAIL NO. 7.7.2 (GENERAL CRITERIA - OBE)

Edge Distance Results
Connections versus Hand Calculations

Member	Actual		Allowable		Ratio	
	Hand Calc	Connections	Hand Calc	Connections	Hand Calc	Connections
Bolt in-leg	4.00 k	4.00 k	12.17 k	12.17 k	0.33	0.33
Plate (FY)	3.34 ksi	3.34 ksi	20.00 ksi	20.00 ksi	0.17	0.17
Plate (FU)	5.06 ksi	5.06 ksi	21.00 ksi	21.00 ksi	0.24	0.24
Beam web (FY)	4.50 ksi	4.50 ksi	14.40 ksi	14.40 ksi	0.31	0.31
Beam web (FU)	12.92 ksi	12.92 ksi	17.40 ksi	17.40 ksi	0.74	0.74

Shear (Force/Stress) Results
Connections versus Hand Calculations

Member	Interaction Ratio	
	Hand Calc	Connections
Full penetration weld	1.294	1.29
Beam web	1.074	1.07

Bending & Axial Load Results
Connections versus Hand Calculations

Member		Actual Minimum	Required Minimum
Edge distance	Beam (Parallel to slot)	0.50 in	1.50 in
	Beam (Normal to slot)	2.56 in.	1.13 in.
	Plate (Parallel to slot)	1.50 in.	1.63 in.
	Plate (Normal to slot)	1.25 in.	1.50 in.

TABLE 3.13-30 COPED SLIDING CONNECTION FIELD WELDED PLATE DETAIL NO. 7.7.2 (GENERAL CRITERIA - OBE)

Member	Actual		Allowable		Ratio	
	Hand Calc	Connections	Hand Calc	Connections	Hand Calc	Connections
Bolt in-leg	4.00 k	4.00 k	12.17 k	12.17 k	0.33	0.33
Plate (FY)	3.34 ksi	3.34 ksi	20.00 ksi	20.00 ksi	0.14	0.14
Plate (FU)	5.06 ksi	5.06 ksi	21.00 ksi	21.00 ksi	0.24	0.24
Beam web (FY)	4.91 ksi	4.91 ksi	14.40 ksi	14.40 ksi	0.34	0.34
Beam web (FU)	12.92 ksi	12.92 ksi	17.40 ksi	17.40 ksi	0.74	0.74

Shear (Force/Stress) Results
Connections versus Hand Calculations

Member	Interaction Ratio	
	Hand Calc	Connections
Full penetration weld	3.600	3.60
Beam web	3.563	3.56

Bending & Axial Load Results
Connections versus Hand Calculations

	Member	Actual Minimum	Required Minimum
Edge distance	Beam (Parallel to slot)	0.50 in	1.50 in
	Beam (Normal to slot)	1.31 in.	1.50 in.
	Plate (Parallel to slot)	1.50 in.	1.63 in.
	Plate (Normal to slot)	1.25 in.	1.50 in.

	Ratio	
	Hand Calc	Connections
Block Shear in Beam Web	0.430	0.43

TABLE 3.13-31 UNCOPED FRICTION CONNECTION SHOP WELDED/FIELD BOLTED DETAIL NO. 7.2.9 (GENERAL CRITERIA - OBE)

			Connections			Hand Calculation		
Item	Member	Units	Actual	Allowable	Ratio	Actual	Allowable	Ratio
	Weld A in-leg	K/in.	0.75	4.50	0.17	0.75	4.50	-
	Bolt out-leg	Kips	0.85	10.49	0.08	0.85	10.49	-
	Angle in-leg (FY)	Ksi	0.82	14.40	0.06	0.82	14.40	-
Shear (forces/stresses	Angle out-leg (FY)	Ksi	0.49	14.40	0.03	0.49	14.40	-
	Angle out-leg (FU)	Ksi	0.75	17.40	0.04	0.75	17.40	-
	Beam web (FY)	Ksi	1.79	14.40	0.12	1.79	14.40	-
Bending and axial	Angle out-leg	-	-	-	0.44		-	0.444
	Beam Web	-	-	-	0.59		-	0.590
Prying action	Angle out-leg	Kips	0.11	26.46	-	0.110	26.458	-

	Member	Actual Minimum	Required Minimum
Edge distance	Angle out-leg (shear edge)	1.25 in.	1.25 in.
	Angle out-leg (rolled edge)	1.50 in.	1.13 in.

Connection adequacy = O.K.

TABLE 3.13-32 COPED FRICTION CONNECTION SHOP WELDED/FIELD BOLTED DETAIL NO. 7.2.9

Item	Member	Units	Connections Actual	Connections Allowable	Connections Ratio	Hand Calculation Actual	Hand Calculation Allowable	Hand Calculation Ratio
Shear (forces/stresses	Weld A in-leg	K/in.	0.75	4.50	0.17	0.75	4.50	-
	Bolt out-leg	Kips	0.85	10.49	0.08	0.85	10.49	-
	Angle in-leg (FY)	Ksi	0.82	14.40	0.06	0.82	14.40	-
	Angle out-leg (FY)	Ksi	0.49	14.40	0.03	0.49	14.40	-
	Angle out-leg (FU)	Ksi	0.75	17.40	0.04	0.75	17.40	-
	Beam web (FY)	Ksi	1.79	14.40	0.12	1.79	14.40	-
Bending and axial	Angle out-leg	-	-	-	0.86	-	-	0.864
	Beam Web	-	-	-	0.88	-	-	0.880
Prying action	Angle out-leg	Kips	0.11	26.46	-	0.110	26.458	-

	Member	Actual Minimum	Required Minimum
Edge distance	Angle out-leg (rolled edge)	1.25 in.	1.25 in.
	Angle out-leg (sheared edge)	1.50 in.	1.13 in.

Connection adequacy = O.K.

TABLE 3.13-33 UNCOPED FRICTION CONNECTION SHOP WELDED/FIELD WELDED DETAIL NO. 7.2.11 (CONNECTIONS VERSUS HAND CALCULATIONS)

		Connections				Hand Calculation		
Item	Member	Units	Actual	Allowable	Ratio	Actual	Allowable	Ratio
	Weld A in-leg	K/in.	1.19	4.50	0.26	1.19	4.50	-
	Bolt B out-leg	K/in.	1.83	4.50	0.41	1.83	4.50	-
Shear (forces/stresses	Angle in-leg	Ksi	3.43	14.40	0.24	3.43	14.40	-
	Angle out-leg	Ksi	0.73	14.40	0.05	0.73	14.40	-
	Beam web	Ksi	4.29	14.40	0.30	4.29	14.40	-
Bending and axial	Angle out-leg	-	-	-	1.52	-	-	1.519
	Beam Web	-	-	-	1.20	-	-	1.201

Connection adequacy = not O.K.

TABLE 3.13-34 COPED FRICTION CONNECTION SHOP WELDED/FIELD WELDED DETAIL NO. 7.2.11 (CONNECTIONS VERSUS HAND CALCULATIONS)

		Connections				Hand Calculation		
Item	Member	Units	Actual	Allowable	Ratio	Actual	Allowable	Ratio
	Weld A in-leg	K/in.	1.19	4.50	0.26	1.19	4.50	-
	Bolt B out-leg	K/in.	1.83	4.50	0.41	1.83	4.50	-
Shear (forces/stresses	Angle in-leg	Ksi	3.43	14.40	0.24	3.43	14.40	-
	Angle out-leg	Ksi	0.73	14.40	0.05	0.73	14.40	-
	Beam web	Ksi	4.29	14.40	0.30	4.29	14.40	-
Bending and axial	Angle out-leg	-	-	-	3.41	-	-	3.408
	Beam Web	-	-	-	2.86	-	-	2.862

Connection adequacy = not O.K.

TABLE 3.13-35 CINCH VALIDATION PROBLEM 1 OBE CASE

Item	CINCH	Hand Calc
Total M_x including increases	27,720 in-lb	27,720 in-lb
Total M_y including increases	29,320 in-lb	29,320 in-lb
Total M_z including increases	--	9,114 in-lb
Amplified anchor forces		
T_1	1.322 kips	1.33 kips
T_2	1.895 kips	1.89 kips
T_3	2.468 kips	2.47 kips
T_4	0.716 kips	0.71 kips
T_5	1.862 kips	1.86 kips
T_6	0.110 kips	0.11 kips
T_7	0.683 kips	0.68 kips
T_8	1.256 kips	1.26 kips
Maximum shear per anchor	0.338 kips	0.338 kips
Maximum total anchor force	2.95 kips	2.95 kips
Plate bending stresses		
M_y left	7.253 ksi	7.2 ksi
M_y right	2.614 ksi	2.6 ksi
M_x top	7.126 ksi	7.1 ksi
M_x bottom	2.741 ksi	2.7 ksi
Maximum concrete stress	0.0 ksi	0.0 ksi
Allowable anchor force	3.40 ksi	3.4 kips
Allowable bending stress	27 ksi	27 ksi

TABLE 3.13-36 CINCH REASSESEMENT PROBLEM RES02

Item		CINCH	Hand Calc
Pullout area anchor	#1	83.74 in^2	83.74 in^2
	2	78.46 in^2	78.46 in^2
	3	54.12 in^2	54.12 in^2
	4	78.46 in^2	78.46 in^2
	5	51.58 in^2	51.59 in^2
	6	83.74 in^2	83.74 in^2
	7	78.46 in^2	78.46 in^2
	8	54.12 in^2	54.12 in^2
Ultimate tension force anchor	#1	20.684 kips	20.68 kips
	2	15.129 kips	15.13 kips
	3	10.435 kips	10.44 kips
	4	15.129 kips	15.13 kips
	5	9.946 kips	9.95 kips
	6	20.684 kips	20.68 kips
	7	15.129 kips	15.13 kips
	8	10.435 kips	10.44 kips
Ultimate shear force anchor	#1	21.561 kips	21.56 kips
	2	15.771 kips	15.77 kips
	3	10.877 kips	10.88 kips
	4	15.771 kips	15.77 kips
	5	10.368 kips	10.37 kips
	6	21.561 kips	21.56 kips
	7	15.771 kips	15.77 kips
	8	10.877 kips	10.88 kips
Allowable plate bending stress		27.0 ksi	27.0 ksi
Allow. conc. compres. stress		4.091 ksi	4.090 ksi

TABLE 3.13-36 CINCH REASSESMENT PROBLEM RES02

Item		CINCH	Hand Calc
Amplification factor			
Moment		1.04	1.04
Tension		1.13	1.13
Resultant anchor tension anchor	#1	2.421 kips	2.42 kips
	2	3.338 kips	3.33 kips
	3	4.255 kips	4.26 kips
	4	1.227 kips	1.23 kips
	5	3.061 kips	3.05 kips
	6	0.034 kips	0.04 kips
	7	0.951 kips	0.95 kips
	8	1.868 kips	1.86 kips
Resultant anchor shear anchor	#1	1.857 kips	1.858 kips
	2	1.908 kips	1.908 kips
	3	1.973 kips	1.974 kips
	4	1.685 kips	1.685 kips
	5	1.812 kips	1.813 kips
	6	1.515 kips	1.514 kips
	7	1.576 kips	1.576 kips
	8	1.655 kips	1.655 kips
Shear-tension interaction anchor	#1	0.159	0.159
	2	0.273	0.273
	3	0.514	0.514
	4	0.147	0.147
	5	0.385	0.384
	6	0.07	0.07
	7	0.135	0.135
	8	0.258	0.258

TABLE 3.13-36 CINCH REASSESMENT PROBLEM RES02

Item		CINCH	Hand Calc
Edge check interaction anchor	#3	3.339	3.345
	5	2.723	2.719
	8	1.968	1.966
Plate bending moments			
Right face		-6.45 kip-in.	-5.66 kip-in.
Left face		-18.082 kip-in.	-17.98 kip-in.
Top face		-16.491 kip-in.	-16.5 kip-in.
Bottom face		-8.836 kip-in.	-8.81 kip-in.
Plate Bending Stresses			
Right face		2.150 ksi	1.89 ksi
Left face		6.027 ksi	5.99 ksi
Top face		5.497 ksi	5.50 ksi
Bottom face		2.945 ksi	2.94 ksi

NOTE: Difference in values for the plate bending moments and stresses are due to the approximation used in the hand calculations. Considering this, the program results are concluded to be correct.

TABLE 3.13-37 VIBRATION PERIODS OF CABLE IN STATIC EQUILIBRIUM CONFIGURATION (ADINA - Validation Problem 1)

Mode Number	Period (sec) Manual	Period (sec) S&L'S ADINA
1	4.42	4.42
2	2.31	2.309
3	1.21	1.211
4	1.16	1.164
5	0.929	0.9294

TABLE 3.13-38 COMPARISON OF RESULTS AT SELECTED LOCATIONS FOR STATIC ANALYSIS FROM FRAME AND REFERENCE 45

	Results from FRAME	Results from Reference 45
Shear force at node 2, member 2 (LOCAL)	31 lb	30 lb
Moment at node 3 member Z	948 ft-lb	950 ft-lb
Shear at node 4 member 3	104 lb	100 lb

TABLE 3.13-39 COMPARISON OF RESULTS AT SELECTED LOCATIONS FOR DYNAMIC ANALYSIS FROM FRAME AND HAND CALCULATIONS

	X-Direction Loading		Y-Direction Loading		Z-Direction Loading		1.5x SRSS	
	Static Analysis	Acceleration (g)	Static Analysis	Acceleration (g)	Static Analysis	Acceleration (g)	FRAME	Calculator
X-displacement at node 2, in.	0.00654	1.3	-0.00148	2.1	0	1.3	0.0136	0.0136
Axial force, (local) member 3, node 4	-1574	1.3	-1878	2.1	0	1.3	6666	6665
Moment about Z axis, member 1 node 1, ft-lb	10570	1.3	2865	2.1	0	1.3	22501	22501
Moment about Y member 3, node 4, ft-lb	0	1.3	0	2.1	7263	1.3	14164	14163

TABLE 3.13-40 COMPARISON OF RESULTS AT SELECTED LOCATIONS FOR LOAD COMBINATIONS FROM FRAME AND HAND CALCULATIONS

Member 2, node 3 local coordinates	Axial		Shear		Moments		
Analysis results	F_a lb		F_b lb	F_c lb	M_a ft-lb	M_b ft-lb	M_c ft-lb
WT1	-895		687	0	0	0	-6593
R01	4868		31	0	0	0	948
SE1	2250		2374	1253	458	3152	12860
Load combination	Tension compression for stress calculations						
WT + R01 + SE							
FRAME	6223	8013	3092	1253	458	3152	20402
Hand calculations	6623	8013	3092	1253	458	3152	20401
$WT + RO^2 + SE^2$							
FRAME	4467	6258	3061	1253	458	3152	19488
Hand calculations	4468	6258	3061	1253	458	3152	19488

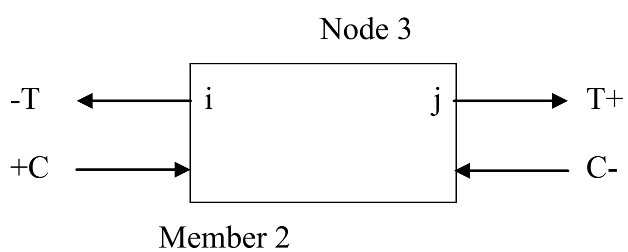


TABLE 3.13-40 COMPARISON OF RESULTS AT SELECTED LOCATIONS FOR LOAD COMBINATIONS FROM FRAME AND HAND CALCULATIONS

Node 3 Displacements, in. analysis results	X	Y	Z	θ_x	θ_y	θ_z
LOT	0.652-02	0.295-02	0	0	0	-.283.03
RO	0.765-03	0.230-02	0	0	0	0.195.03
SE	0.137-01	0.672-02	0.562-01	0.228-01	0.485-02	0.185-02
Load combination						
WT + R01 + SE						
FRAME	0.209-01	0.120-01	0.562-01	0.228-01	0.485-02	0.233.02
Hand calculations	0.209-01	0.120-01	0.562-01	0.228-01	0.485-02	0.233.02
$WT + R0^2 + SE^2$						
FRAME	0.202-01	0.101-01	0.562-01	0.228-01	0.485-02	-0.214.02
Hand calculations	0.202-01	0.101-01	0.562-01	0.228-01	0.485-02	-0.214.02

TABLE 3.13-41 COMPARISON OF FRAME STRESS CHECK OUTPUT AND HAND CALCULATION

	Problem 1				Problem 2				Problem 3				Problem 4			
	FRAME		Hand Calc		FRAME		Hand Calc		FRAME		Hand Calc		FRAME		Hand Calc	
	ACT	ALL	ACT	ALL	ACT	ALL	ACT	ALL	ACT	ALL	ACT	ALL	ACT	ALL	ACT	ALL
Minor axis bending stress (ksi)	0.45	32.73	0.45	32.73	1.82	27.5	1.82	37.5	20.41	21.6	20.41	21.60	58.39	21.06	58.38	21.06
Major axis bending stress (ksi)	0.61	32.73	0.61	32.73	0.87	30.0	0.87	30.0	22.16	21.6	22.7	21.60	39.03	17.69	39.37	17.68
Axial stress(ksi)	1.22	32.73	1.22	32.73	4.24	22.34	4.24	22.34	18.02	15.43	18.02	15.43	1.45	1.92	1.45	1.92
Shear stress(ksi)	2.46	18.9	2.46	18.09	6.01	20.0	6.03	20.0	23.44	14.40	23.45	14.40	54.33	14.04	53.04	14.04
Maximum interaction ratio	0.130	-	0.13	-	0.301	-	0.30	-	4.651	-	4.65	-	14.30	-	14.30	0

TABLE 3.13-42 DESIGN PARAMETERS FOR FRAME STRESS CHECK VALIDATION PROBLEMS

	Problem 1	Problem 2	Problem 3
Unbraced length in 2-direction	25 in.	120 in.	20 in.
Effective length factor in 2-direction	1.0	1.0	1.0
Unbraced length in 3-direction	40 in.	120 in.	40 in.
Effective length factor in 3-direction	1.0	1.0	1.0
Effective length in bending	40 in.	120 in.	40 in.
Overstress factor	1.6	1.0	1.0
Minimum factor of safety	1.1	1.0	1.0
Yield stress	36 ksi	50 ksi	36 ksi

	Problem 4
Unbraced length in 1-direction	120 in.
Effective length factor in 2-directions	1.0
Unbraced length in 3-direction	120 in.
Effective length factor in 3-direction	1.0
Effective length in bending	120 in.
Overstress factor	1.0
Minimum factor of safety	1.0
Yield stress	36 ksi

TABLE 3.13-43 INPUT DATA FOR VALIDATION PROBLEMS FOR THE CONNECTION CHECK MODULE OF FRAME

				Forces (kip and kip - in.)					
Problem No.	Member Size	Weld Type	Weld Size	F_a	F_b	F_c	M_a	M_b	M_c
5	W 5x16	4	1/4"	0.31	0.72	2.47	12.33	44.03	12.22
6	L 3x3x1/4	2	3/16"	0.15	0.17	0.08	0.21	2.04	4.22
7	Z 3x9.8	3	1/4"	0.9	0.34	0.09	0.22	2.15	10.77
8	T 6x4x.25	1	1/4"	1.0	2.0	3.0	4.0	5.0	6.0
9	C 4x5.4	1	1/4"	0.045	0.355	0.045	0.030	1.03	11.26

TABLE 3.13-44 COMPARISON OF FRAME AND HAND-CALCULATED RESULTS FOR CONNECTION MODULE

	FRAME OUTPUT			HAND CALCULATION			
				Connection			
Problem No.	Actual Weld Stress (ksi)	Allowable Weld Stress (ksi)	Ratio	Actual Weld Stress (ksi)	Allowable Weld Stress (ksi)	Ratio	Adequacy
5	5.85	4.72	1.239	5.854	4.724	1.239	Fail
6	1.99	3.54	0.561	1.986	3.543	0.5605	Pass
7	1.17	4.72	0.247	1.166	4.724	0.2468	Pass
8	0.47	4.72	0.1	0.47	4.724	0.0996	Pass
9	1.49	4.72	0.315	1.488	4.724	0.315	Pass

TABLE 3.13-45 PASS - COMPARISON OF NOZZLE AND ANCHOR REACTIONS

Node 901	F_x (lb)		$F_{y\,b}$ (lb)		F_z (lb)	
	PASS	HAND	PASS	HAND	PASS	HAND
Design load (+)	1616	1617	1386	1385	719	719
Design load (-)	-499	-501	-348	-347	-2676	-2676

	M_x (in. – lb)		M_y (in. – lb)		M_z (in. – lb)	
	PASS	HAND	PASS	HAND	PASS	HAND
Design load (+)	87192	87192	24819	24820	17033	17031
Design load (-)	-31632	-31632	-87739	-87740	-35747	-35745

Node 910	F_x (lb)		F_y (lb)		F_z (lb)	
	PASS	HAND	PASS	HAND	PASS	HAND
Design load (+)	30	29	329	331	57	57
Design load (-)	-51	-50	-168	-170	-35	-35

	M_x (in. – lb)		M_y (in. – lb)		M_z (in. – lb)	
	PASS	HAND	PASS	HAND	PASS	HAND
Design load (+)	3632	3633	971	971	3510	3508
Design load (-)	-6267	-6268	-964	-963	-6151	-6149

TABLE 3.13-46 PASS - COMPARISON OF HANGER/RESTRAINT LOADS

Node	Restraint Type and Direction	Design Load (+)		Design Load (-)	
		PASS	HAND	PASS	HAND
65	Y RIGID	2538	2538	-835	-835
65	Z RIGID	2279	2280	-745	-745
395	Y RIGID	222	222	-104	-104
430	Y RIGID	388	388	-158	-158
430	Z RIGID	341	339	-389	-387

TABLE 3.13-47 PASS - COMPARISON OF HANGER/RESTRAINT DISPLACEMENTS

Thermal Displacements

	DX (in.)		DY (in.)		DZ (in.)	
Node	PASS	HAND	PASS	HAND	PASS	HAND
65 (+)	0.000	0.000	0.000	0.000	0.000	0.000
65 (-)	-0.034	-0.034	-0.000	-0.000	-0.000	-0.000
395 (+)	0.097	0.097	0.000	0.000	0.000	0.000
`395 (-)	-0.000	-0.000	-0.000	-0.000	-0.158	-0.158
430 (+)	0.000	0.000	0.000	0.000	0.000	0.000
430 (-)	-0.004	-0.004	-0.000	-0.000	-0.000	-0.000

Seismic Displacements (±)

	DX (in.)		DY (in.)		DZ (in.)	
Node	PASS	HAND	PASS	HAND	PASS	HAND
65	0.120	0.120	0.000	0.000	0.000	0.000
395	0.213	0.212	0.000	0.000	0.388	0.386
430	0.148	0.148	0.000	0.000	0.000	0.000

TABLE 3.13-48 PASS - CLASS 2 STRESS EVALUATION COMPARISON[a]

Node	Component Type	Equation 8		Equation 9 With End Effects		Equation 10 Without End Effects	
		PASS	HAND	PASS	HAND	PASS	HAND
15	Tee-run	2390	2390	5352	5351	15698	15697
15	Tee-branch	2465	2464	4948	4947	6894	6893
15	Tee-run	1645	1645	5969	5968	9485	9484
90	Run	1653	1653	2684	2685	1323	1323
100	Elbow	1295	1295	3695	3696	5807	5807
110	Run	1117	1118	3151	3151	1128	1128
125	Tee-run	1897	1897	5509	5509	3781	3781
125	Tee-branch	1281	1280	6969	6967	1991	1991
125	Tee-run	1863	1863	6041	6041	3448	3448
155	Run at restraint	745	745	1081	1106	405	406
380	Reducer	1641	1641	3570	3575	1883	1886
395	Run at restraint	1045	1045	3885	3890	1922	1925
410	Elbow	1112	1112	4584	4590	6187	6196
910	Anchor	2889	2889	4520	4522	24	24

[a] Refer to NC - 3652 of Section III of ASME B&PV Code, winter 1972 addenda.

TABLE 3.13-49 COMPARISON OF SUPPORT REACTION DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

Node	Program	Forces (lb)			Moments (in. - lb)		
		FX	FY	FZ	MX	MY	MZ
170	NUPIPE	-9154	7541	4492	-5952	-823420	1241512
	ADLPIPE	-9178	7540	4492	-5529	-823420	1241512
218	NUPIPE		16650				
	ADLPIPE		16622				
330	NUPIPE	34532	-33620	-31750	-486338	-1516811	573673
	ADLPIPE	34511	-33608	-31736	-486386	-1519359	573438
390	NUPIPE		8631				
	ADLPIPE		8678				
430	NUPIPE	1702	798	12553	-28147	164346	248852
	ADLPIPE	1746	768	12541	-26917	166180	250956

TABLE 3.13-50 COMPARISON OF DEFLECTIONS AND ROTATIONS DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

Node	Program	Deflection (in.)			Rotation (rad)		
		DX	DY	DZ	RX	RY	RZ
197	NUPIPE	0.348	-0.141	0.230	-0.0026	0.0025	-0.0084
	ADLPIPE	0.348	-0.141	0.229	-0.0026	0.0025	-0.0084
212	NUPIPE	1.120	0.052	-0.023	-0.0092	-0.0051	-0.0115
	ADLPIPE	1.120	0.052	-0.023	-0.0092	-0.0051	-0.0115
230	NUPIPE	1.276	-0.028	-0.548	-0.0066	-0.0044	0.0024
	ADLPIPE	1.276	-0.027	-0.548	-0.0066	-0.0044	0.0024
260	NUPIPE	0.512	-0.001	-0.520	-0.0034	-0.0005	0.0035
	ADLPIPE	0.512	-0.000	-0.520	-0.0035	-0.0005	0.0035
390	NUPIPE	0.066	-0.000	0.249	-0.0010	0.0026	-0.0020
	ADLPIPE	0.067	-0.000	0.248	-0.0010	0.0026	-0.0020
420	NUPIPE	-0.029	-0.079	0.011	-0.0002	-0.0002	-0.0007
	ADLPIPE	-0.029	-0.079	0.011	-0.0002	-0.0002	-0.0007

TABLE 3.13-51 COMPARISON OF STRESS DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

Node	NUPIPE	ADLPIPE
180	18989	19013
199	17703	17731
214	23958	23955
236	14427	14416
265	6254	6251
305	12539	12532
344	11845	11838
370	6295	6296
395	3476	3473
430	3282	3308

TABLE 3.13-52 COMPARISON OF INTERNAL FORCES DUE TO DEADWEIGHT ANALYSIS

		Forces (lb)			Moments (in.-lb)		
Node	Program	FX	FY	FZ	MX	MY	MZ
197	NUPIPE	295	2337	14	-35864	5218	51979
	ADLPIPE	290	2341	15	-35108	5231	52081
212	NUPIPE	295	3306	14	59390	-5394	14010
	ADLPIPE	299	3310	15	59735	-5500	14542
360	NUPIPE	330	2781	-29	30930	-22748	-84971
	ADLPIPE	326	2783	-32	31920	-23105	-82784
390	NUPIPE	330	4933	-29	-255351	701	126476
	ADLPIPE	336	4707	-32	-256444	916	126716
420	NUPIPE	330	-492	-29	-8972	27075	82202
	ADLPIPE	336	-497	-32	-9181	27724	80676

TABLE 3.13-53 COMPARISON OF DEFLECTIONS AND ROTATION DUE TO DEAD WEIGHT

Node	Program	Deflection (in.)			Rotation (rad)		
		DX	DY	DZ	RX	RY	RZ
197	NUPIPE	0.007	-0.014	-0.004	0.0001	0.0001	0.0002
	ADLPIPE	0.007	-0.014	-0.004	0.0001	0.0001	0.0002
212	NUPIPE	-0.005	-0.013	0.013	0.0006	0.0001	0.0004
	ADLPIPE	-0.005	-0.013	0.013	0.0006	0.0001	0.0004
360	NUPIPE	-0.008	-0.068	0.024	0.0004	-0.0000	-0.0004
	ADLPIPE	-0.009	-0.069	0.024	0.0004	0.0000	-0.0004
390	NUPIPE	-0.014	-0.000	-0.003	0.0002	-0.0002	-0.0005
	ADLPIPE	-0.015	-0.000	-0.003	0.0002	-0.0002	-0.0005
420	NUPIPE	-0.001	0.002	-0.001	-0.0000	-0.0001	-0.0002
	ADLPIPE	-0.001	0.002	-0.001	-0.0000	-0.0001	-0.0002

TABLE 3.13-54 COMPARISON OF STRESSES DUE TO DEAD WEIGHT

NODE	NUPIPE (psi)	ADLPIPE (psi)
180	685	694
199	448	458
214	667	679
236	2472	2449
265	530	524
305	515	522
344	635	631
370	679	677
395	575	580
430	1101	1091

TABLE 3.13-55 COMPARISON OF NATURAL FREQUENCIES (NUPIPE VERSUS ADLPIPE)

Node	1st	2nd	3rd	4th	5th
NUPIPE	7.109	9.328	12.297	14.681	18.043
ADLPIPE	7.118	9.329	12.492	14.427	17.714

TABLE 3.13-56 COMPARISON OF NATURAL FREQUENCIES (NUPIPE Versus BENCHMARK Pr.)

Node	1	2
NUPIPE	2.407	13.537
Benchmark Pr.	2.3288	13.0808

TABLE 3.13-57 NUPIPE VERSUS HAND CALCULATION

Point No.: 20	Hand Calculation	NUPIPE
Min. wall thickness	0.032 in.	0.032 in.
Primary stress (Eq. 9)	3,713 psi	3,712 psi
Primary and secondary stress (Eq. 10)	16,041 psi	16,038 psi
Alternating stress (Eqs. 11 & 14)	13,468 psi	13,465 psi
Usage factor	0.0654	0.0631
Point No.: 30		
Min. wall thickness	0.047 in.	0.047 in.
Primary stress (Eq. 9)	8,748 psi	8,741 psi
Primary and secondary stress (Eq. 10)	117,655 psi	117,546 psi
Expansion stress (Eq. 12 and Eq. 13)	99,884 psi 18,252 psi	99,781 psi 18,246 psi
Alternating stress (Eq. 14)	218,258 psi	217,811 psi
Usage factor	Out of Range	

TABLE 3.13-58 INDIVIDUAL PAIR USAGE FACTOR FOR POINT NO. 30

	Hand Calculation	NUPIPE
Pair (1,5)	0.183	0.1803
Pair (1,8)	1.660	1.7361
Pair (1,9)	0.0001	0.0001
Pair (1,10)	Out of Range	
Pair (5,8)	Out of Range	
Pair (5,9)	0.221	0.2646
Pair (5,10)	0.747	0.8051
Pair (8,9)	0.857	0.8832
Pair (8,10)	5.5518	5.8608
Pair (9,10)	0.0001	0.0001

TABLE 3.13-59 PIPE MATERIAL PROPERTIES

Property	Temperature (°F)	Value
Thermal conductivity	450	10.01 Btu/°F/hr/ft
Thermal diffusivity	450	0.164 ft^2/hr
Young’s modulus	70	28.3 x 10^6 psi
Coefficient of thermal expansion	70	9.11 x 10^6 in./in.°F

TABLE 3.13-60 FLUID MATERIAL/THERMAL PROPERTIES

Property	Temperature (°F)	Value
Density	450	51.300 lb/ft^3
Viscosity	450	0.2920 lb/hr/ft
Specific heat	450	1.135 Btu/lb/°F
Conductivity	450	0.3650 Btu/°F/hr/ft
Volume expansion coefficient	450	0.0009/°F

TABLE 3.13-61 COMPARISON OF HTLOAD WITH HAND CALCULATION

	HTLOAD	Hand Calculation
Reynolds number	186,700	186,700
Heat transfer coefficient	946.8 Btu/°F/hr/ft^2	946.8 Btu/°F/hr/ft^2

TABLE 3.13-62 COMPARISON OF HTLOAD WITH CHARTS OF BROCK AND MCNEILL

Parameter	Charts	HTLOAD
Maximum ΔT_1	43.31 °F	45.14 °F
Maximum ΔT_2	8.50 °F	8.36 °F

TABLE 3.13-63 COMPARISON OF HTLOAD WITH TRHEAT

Parameter	TRHEAT	HTLOAD
Maximum ΔT_1	44.70 °F	45.14 °F
Maximum ΔT_1	8.69 °F	8.36 °F
Maximum $T_A - T_B$	19.03 °F	19.08 °F

TABLE 3.13-64 COMPARISON OF PITRUST WITH FRANKLIN INSTITUTE PROGRAM CYLNOZ AND HAND CALCULATION

Source of Stress	Franklin Institute Corrected Values	Output from PITRUST	Hand Calculation
Circumferential			
p (Normal)	395.	399.	399.99
p (Bending)	1,875	1,883	1,877.3
Mc (Normal)	35.85	35.57	36.06
Mc (Bending)	364.7	366.6	354.3
Ml (Normal)	79.05	79.66	79.54
Ml (Bending)	90.52	80.57	79.42
Axial			
p (Normal)	813.	812.	814.8
p (Bending)	812.3	827.	810.6
Mc (Normal)	91.79	105	95.45
Mc (Bending)	158.8	160	158.8
Ml (Normal)	37.06	37.0	37.12
Ml (Bending)	117.9	105.	103.85
Shear stress by Mr	6.63	6.63	6.63
Shear stress by Vc	106.1	106.1	106.1
Shear stress by Vl	106.1	106.1	106.1

TABLE 3.13-65 COMPARISON OF PITRUST WITH REF. 70 RESULTS

Location and Cause	PITRUST Results	Exp. Results (Ref. 70)
Element "A"		
Longt. Moment		
Circumf. stress	20,438.9 psi	20,000 psi (Fig. 16, Ref. 70)
Axial stress	26,292.6 psi	25,000 psi
Element "B"		
Circumf. Moment		
Circumf. stress	22,016.2 psi	24,000 psi (Fig. 15, Ref. 70)
Axial stress	13,105.8 psi	13,000 psi

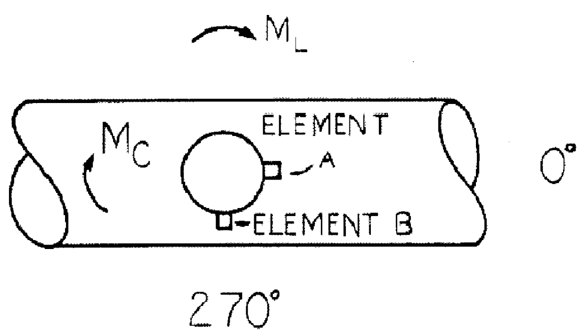


TABLE 3.13-66 COMPARISON OF PILUG COMPUTER PROGRAM OUTPUT WITH HAND CALCULATIONS

Test Problem: Run pipe OD = 17 in.; Run pipe thickness = 0.812 in.;
Axial Length of Lug = 12 in.;
Width of lug along circumf = 3 in.;

Loads: P=3399 lb; Vc = -1788 lb; Vl = 2478 lb;
Mc = 81834 in.-lb; Ml = 103320 in.-lb;
Mt=76284 in.-lb

Stress in Circumferential Direction

Figure	β	Stress from Hand Cal.	Computer Output	Remarks
3C	0.5485	387	330	Membrane stress due to P
1C	0.326	2165	2160	Bending stress due to P
3A	0.294	671	629	Membrane stress due to Mc
1A	0.388	18976	19904	Bending stress due to Mc
3B	0.467	3014	2961	Membrane stress due to Ml
1B	0.416	6143	5969	Bending stress due to Ml

Stress in Axial Direction

Figure	β	Stress from Hand Cal.	Computer Output	Remarks
4C	0.4447	683	690	Membrane stress due to P
2C	0.4632	773	792	Bending stress due to P
4A	0.294	1897	1864	Membrane stress due to Mc
2A	0.550	6357	5942	Bending stress due to Mc
4B	0.467	2365	2328	Membrane stress due to M1
2B	0.582	4989.7	4842	Bending stress due to M1

Shear Stress

Figure	β	Stress from Hand Cal.	Computer Output	Remarks
		1304.8	1304.8	Shear stress due to Mt
		-366.99	-366.99	Shear stress due to V1
		127.15	127.16	Shear stress due to V
		127.15	127.16	Shear stress due to V

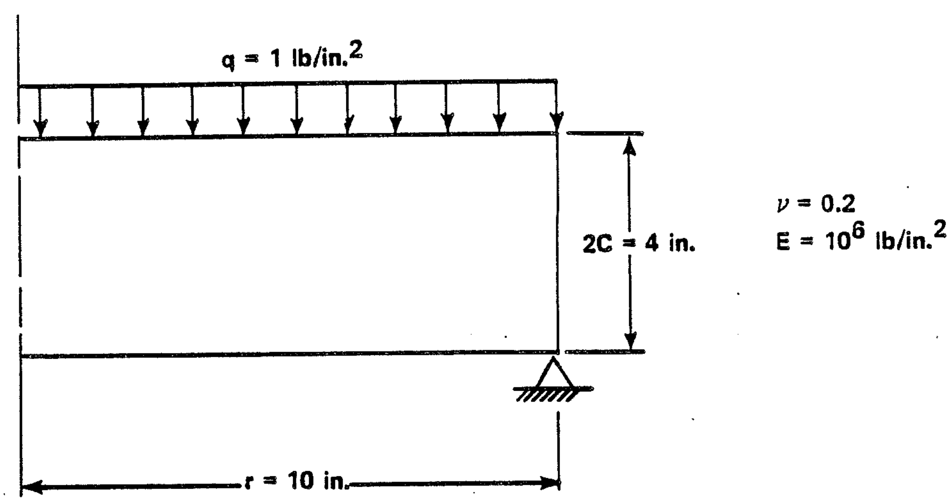


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FIGURE 3.13-1

UNIFORMLY LOADED THICK CIRCULAR PLATE

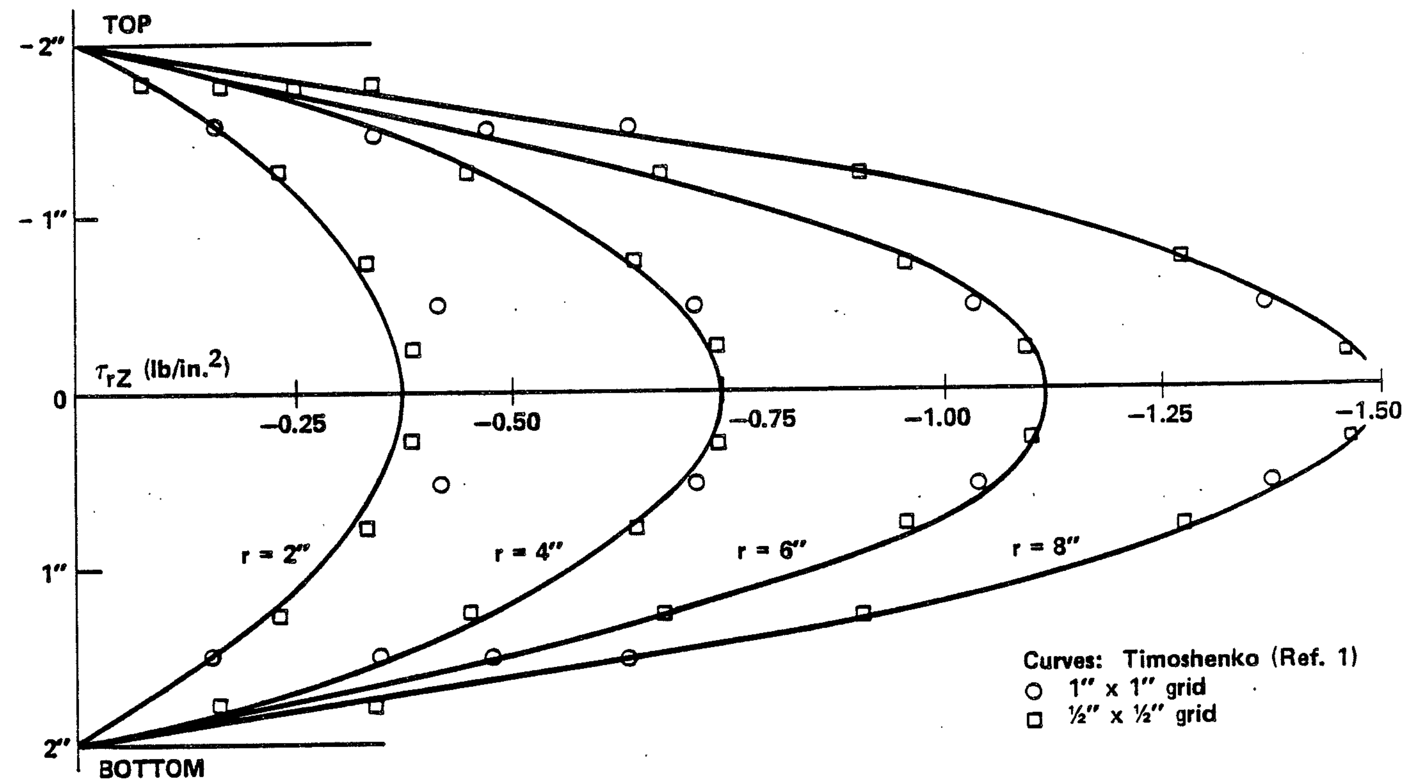

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FIGURE 3.13-2

UNIFORMLY LOADED THICK CIRCULAR PLATE
τ_{rZ} STRESSES

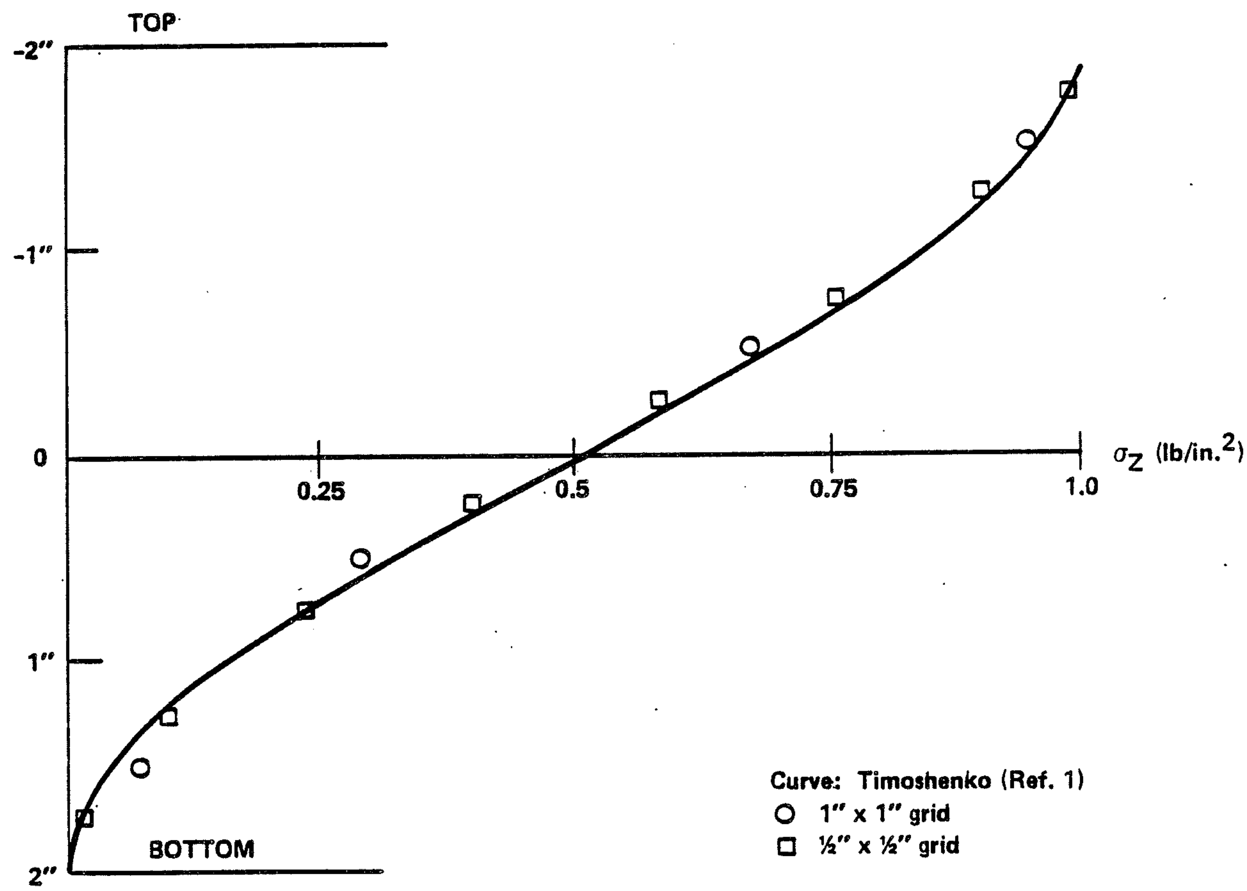

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FIGURE 3.13-3

UNIFORMLY LOADED THICK CIRCULAR PLATE
σ_Z STRESSES – PSI

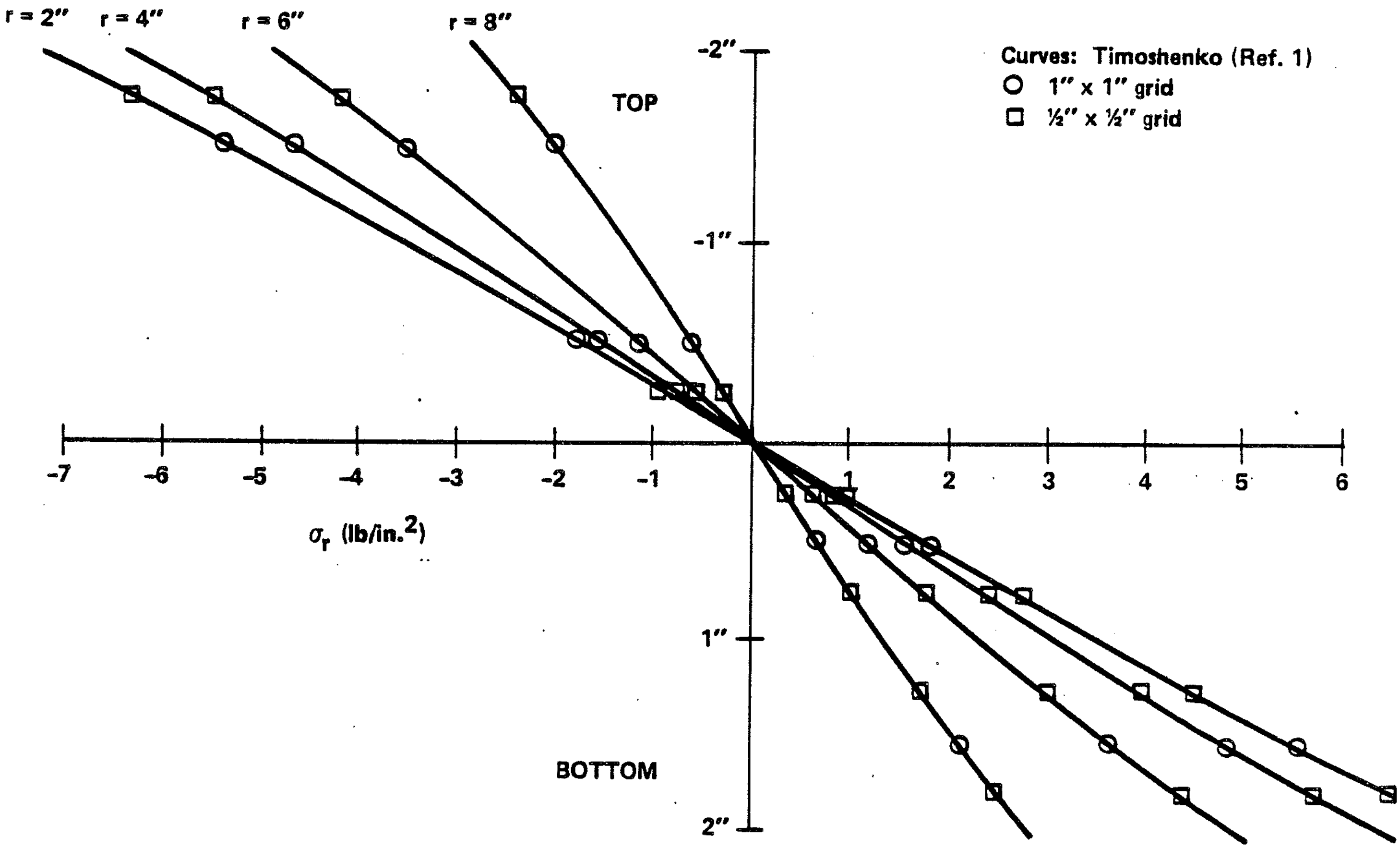

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FIGURE 3.13-4

UNIFORMLY LOADED THICK CIRCULAR PLATE
σ_r STRESSES

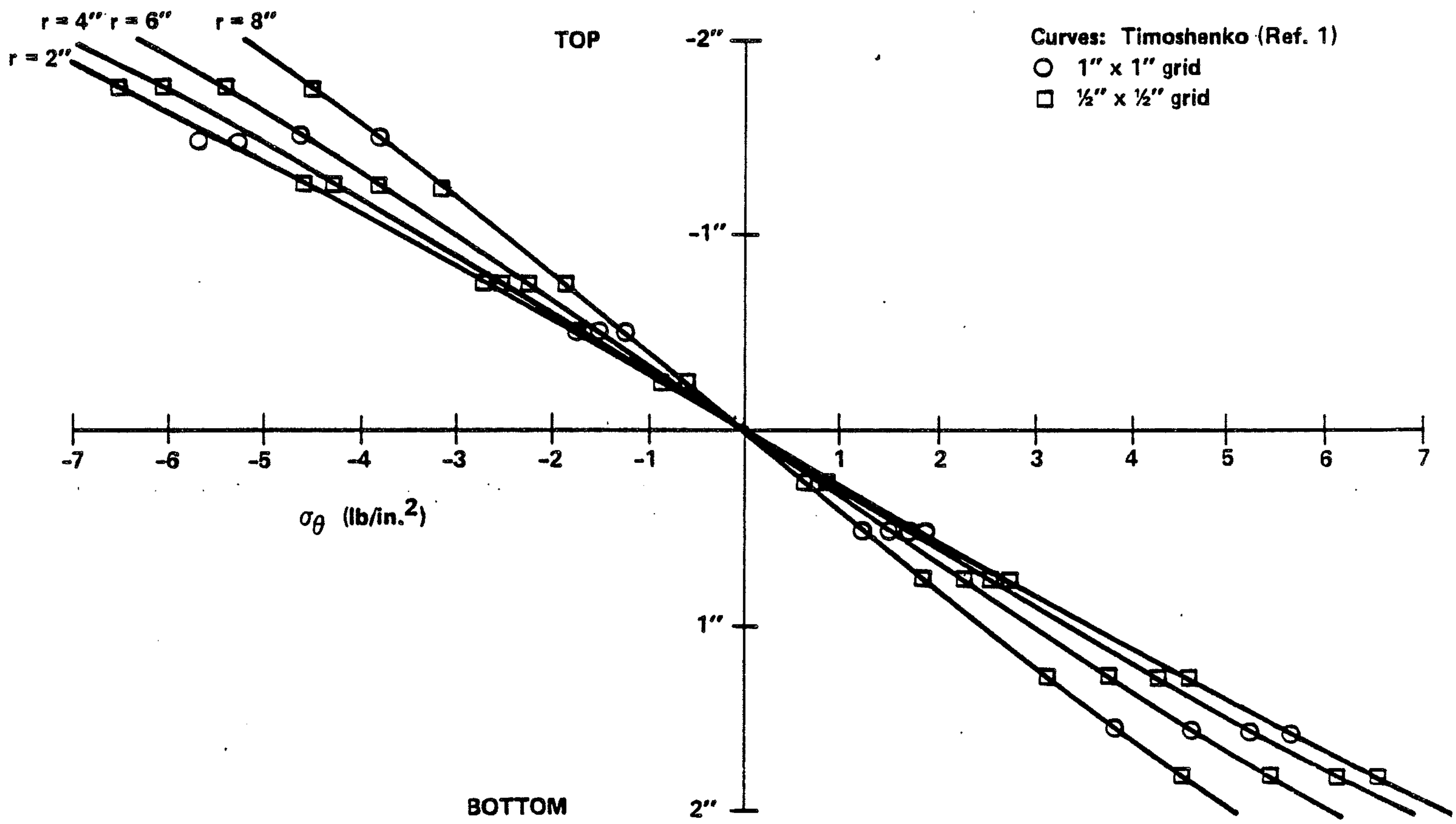

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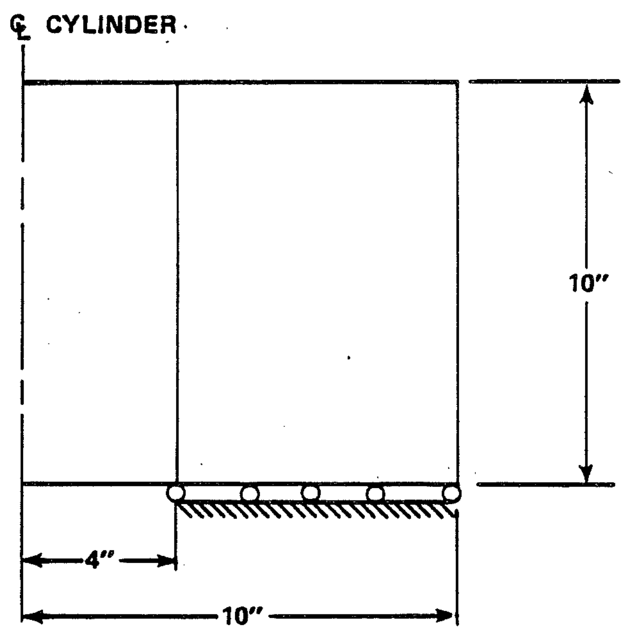
FIGURE 3.13-5

UNIFORMLY LOADED THICK CIRCULAR PLATE σ_θ STRESSES



γ = 200 lb/in.^{3}
ν = 0.2
E = 10^6 lb/in.^{2}

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FIGURE 3.13-6

THICK CYLINDER GEOMETRY AND MATERIAL PROPERTIES

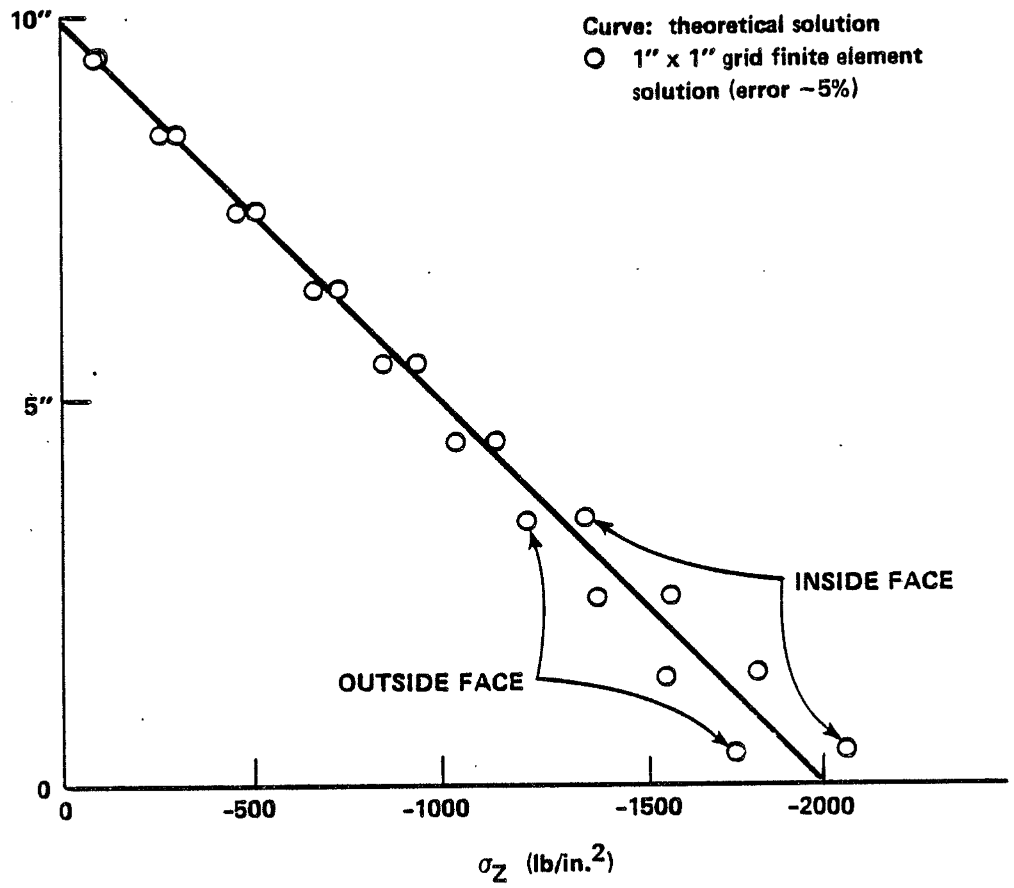

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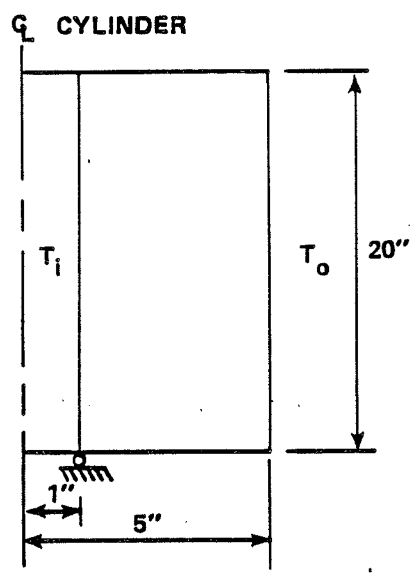
FIGURE 3.13-7

THICK-CYLINDER σ_Z DUE TO DEAD WEIGHT



$\nu = 0.2$

$E = 10^6$ lb/in.^{2}

$a = 1/3000$ in./in.-°F

$T_i = 10°F$

$T_o = 0°F$

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FIGURE 3.13-8

THICK-CYLINDER GEOMETRY FOR THE TEMPERATURE PROBLEM

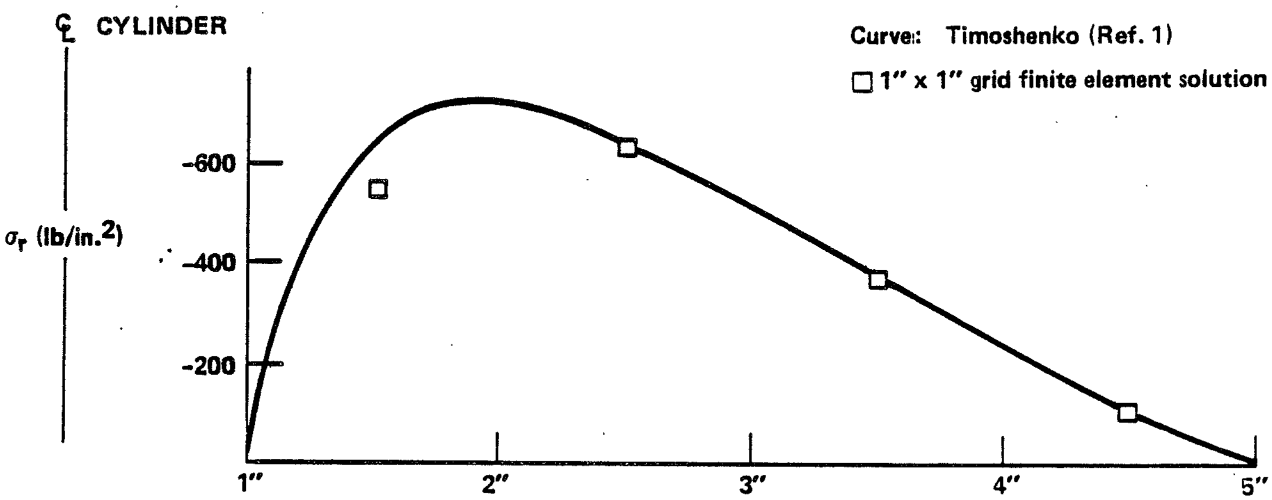

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FIGURE 3.13-9

THICK-CYLINDER σ_r STRESSES DUE TO TEMPERATURE GRADIENT

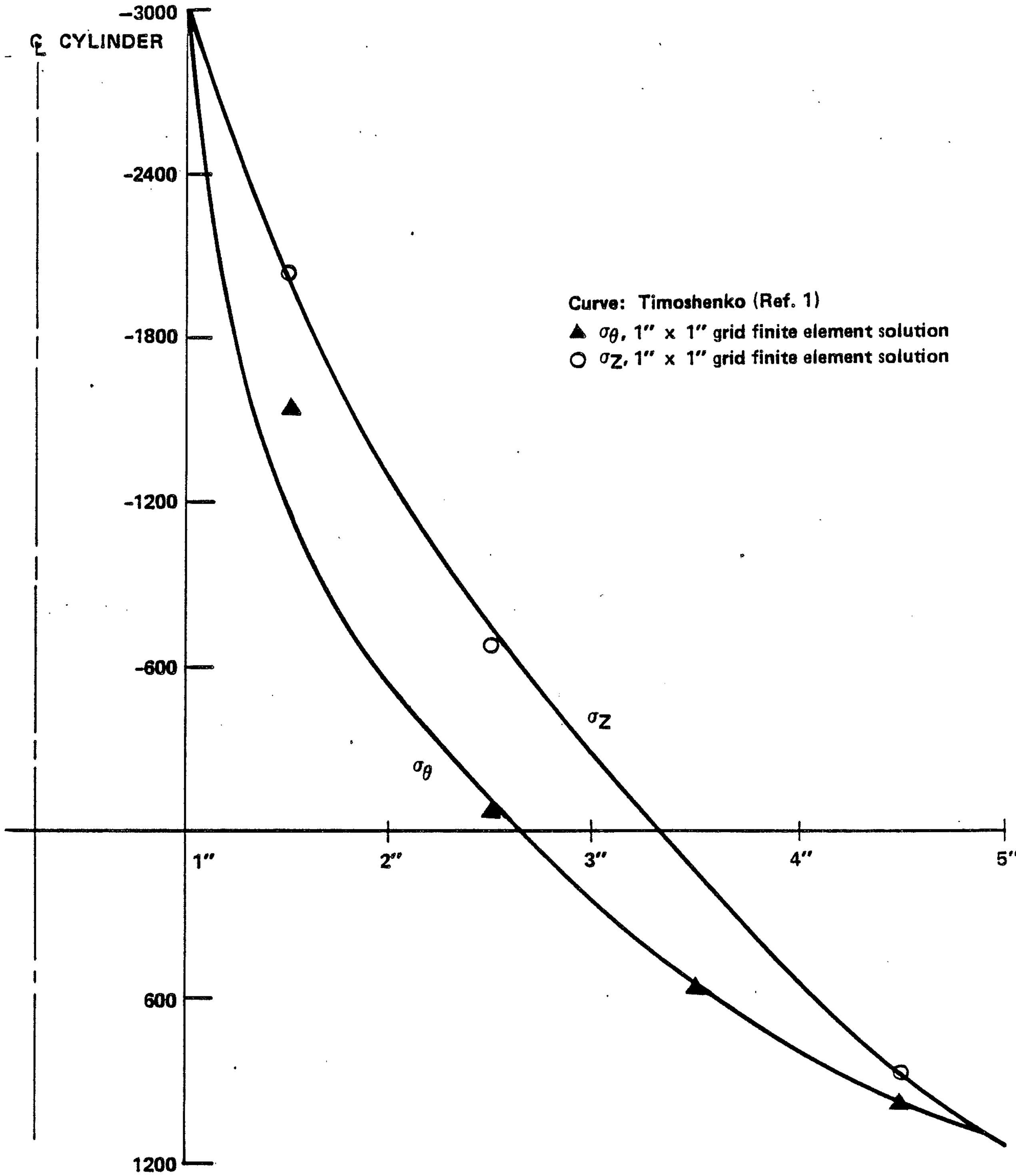

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FIGURE 3.13-10

THICK-CYLINDER σ_Z AND σ_θ STRESSES DUE TO TEMPERATURE GRADIENT

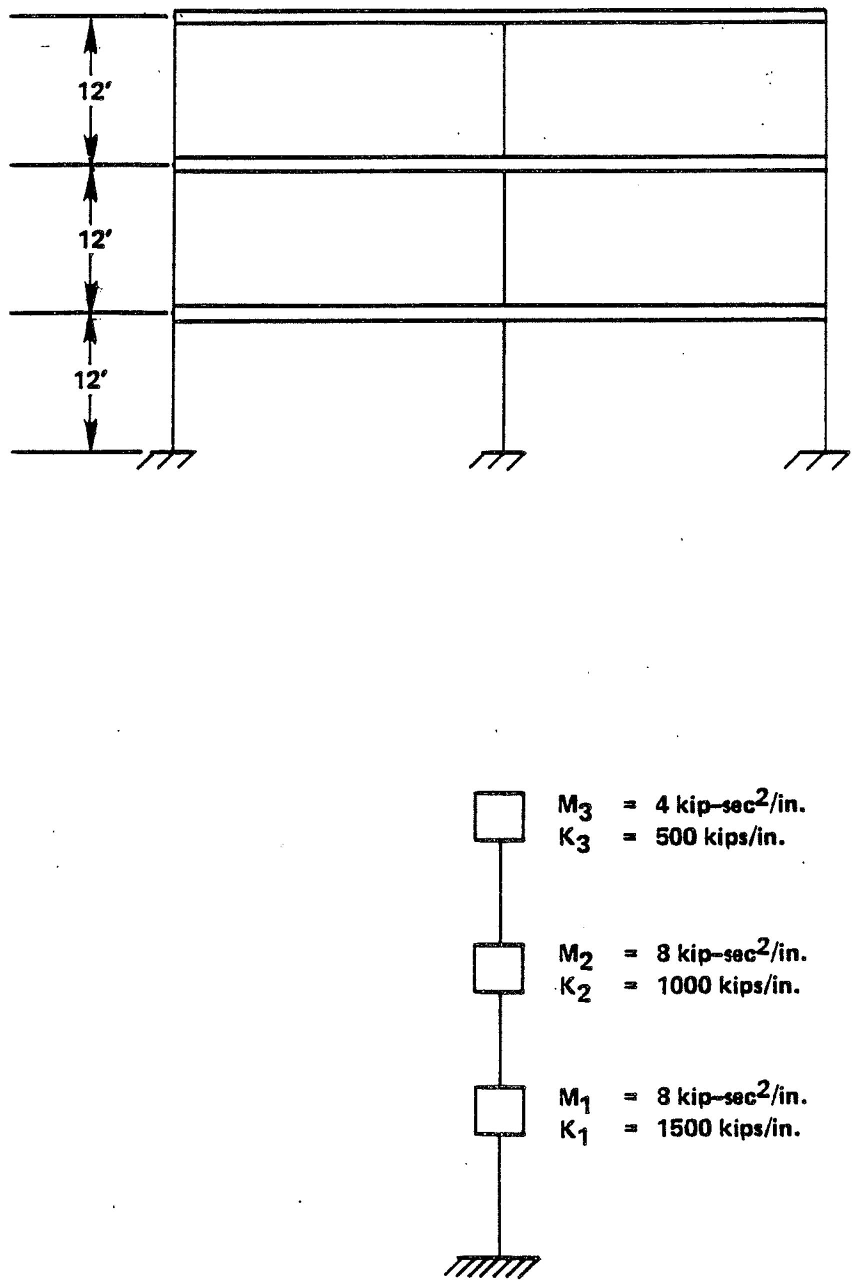

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FIGURE 3.13-11

THREE-STORY SHEAR BUILDING

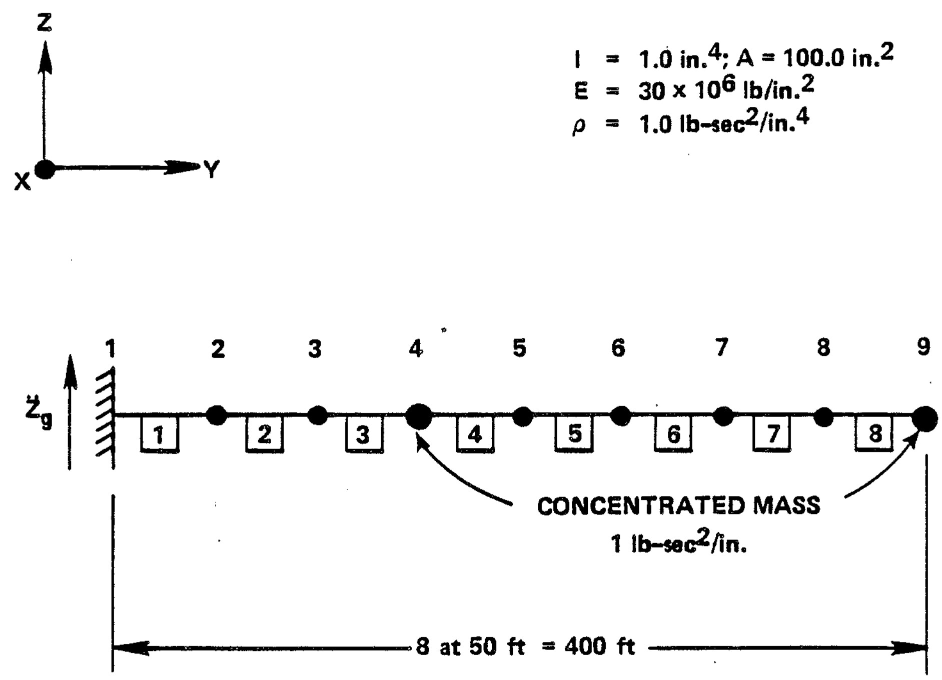

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FIGURE 3.13-12

NODE AND BEAM NUMBER ASSIGNMENTS FOR THE CANTILEVER MODEL

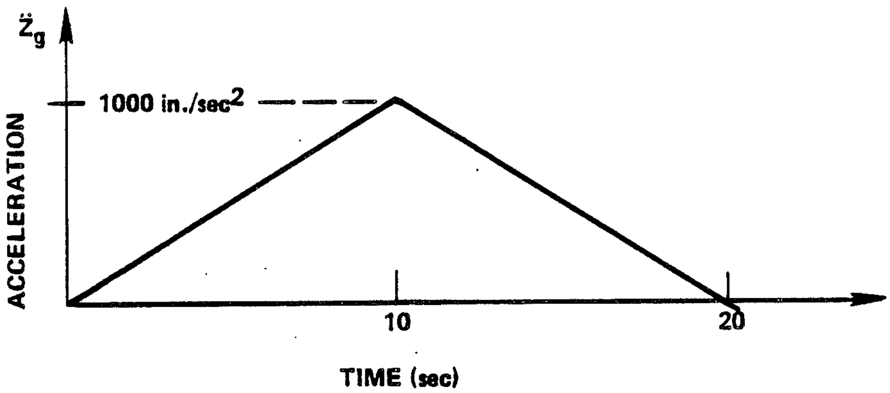

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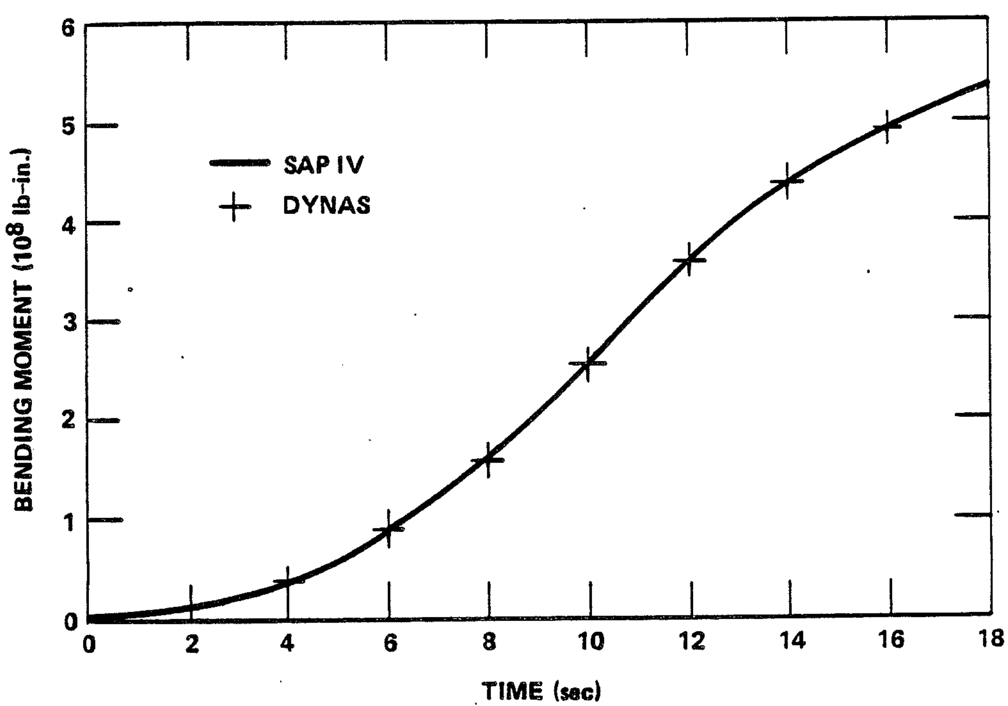
FIGURE 3.13-13

GROUND ACCELERATION APPLIED AT NODE 1



MOMENT AT NODE 1 (FIXED END OF CANTILEVER)

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FIGURE 3.13-14

CANTILEVER RESPONSE

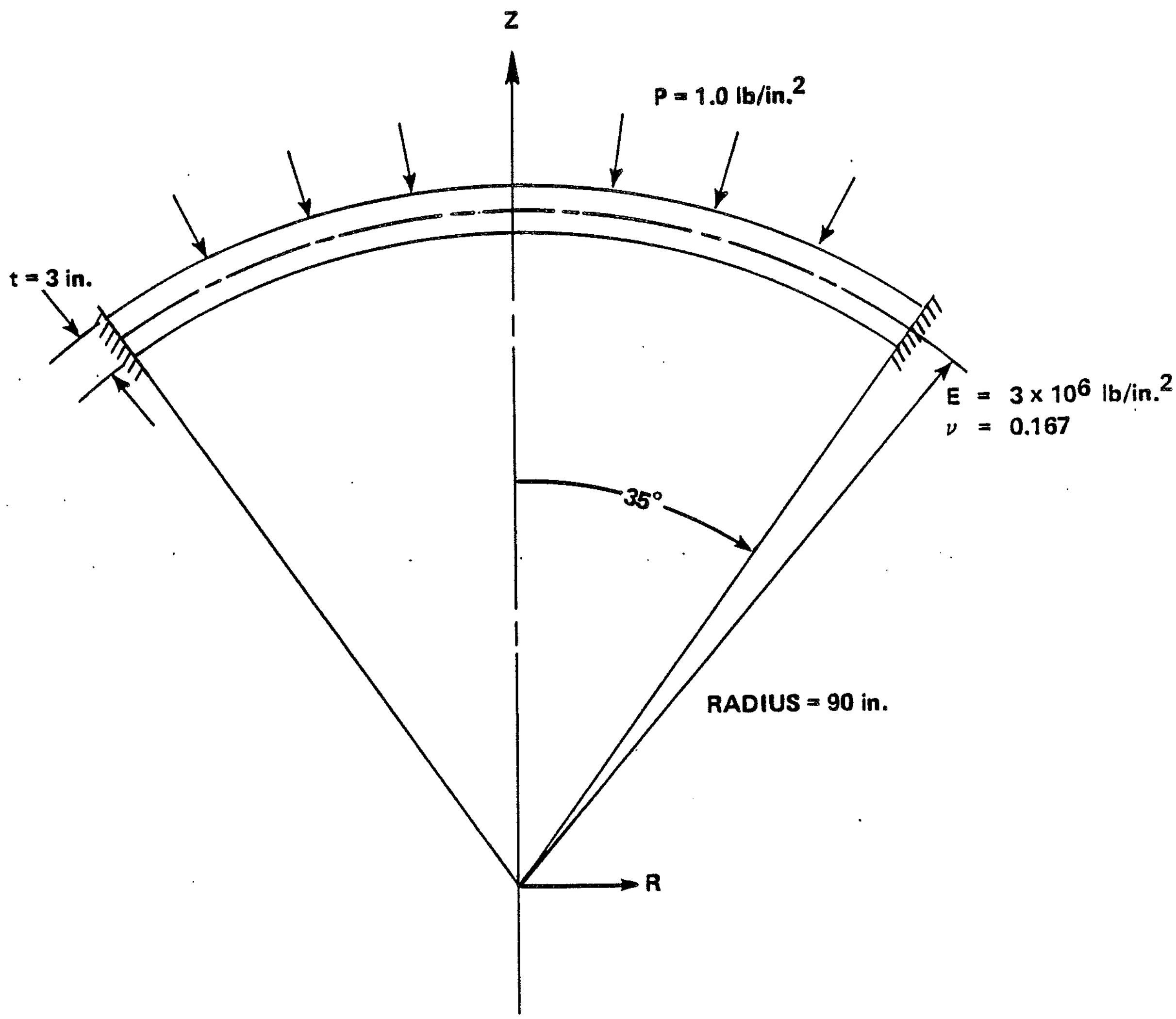

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FIGURE 3.13-15

SHALLOW SPHERICAL SHELL

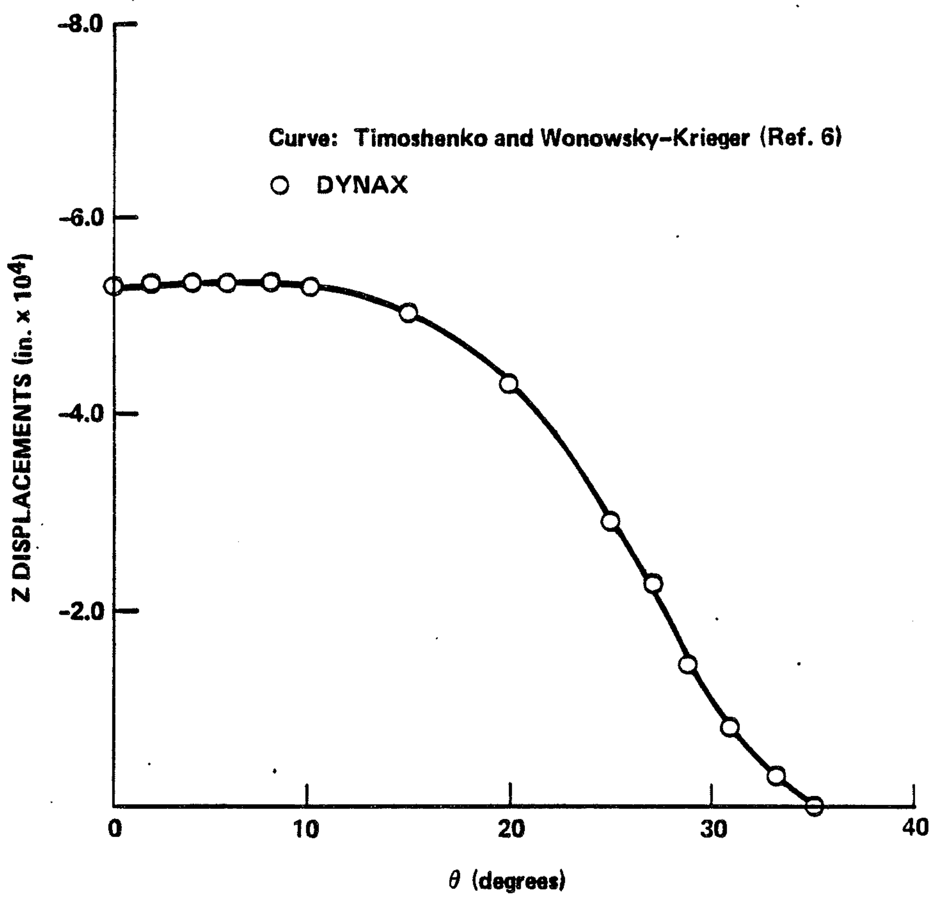

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FIGURE 3.13-16

AXIAL DISPLACEMENT – SHALLOW SPHERICAL SHELL

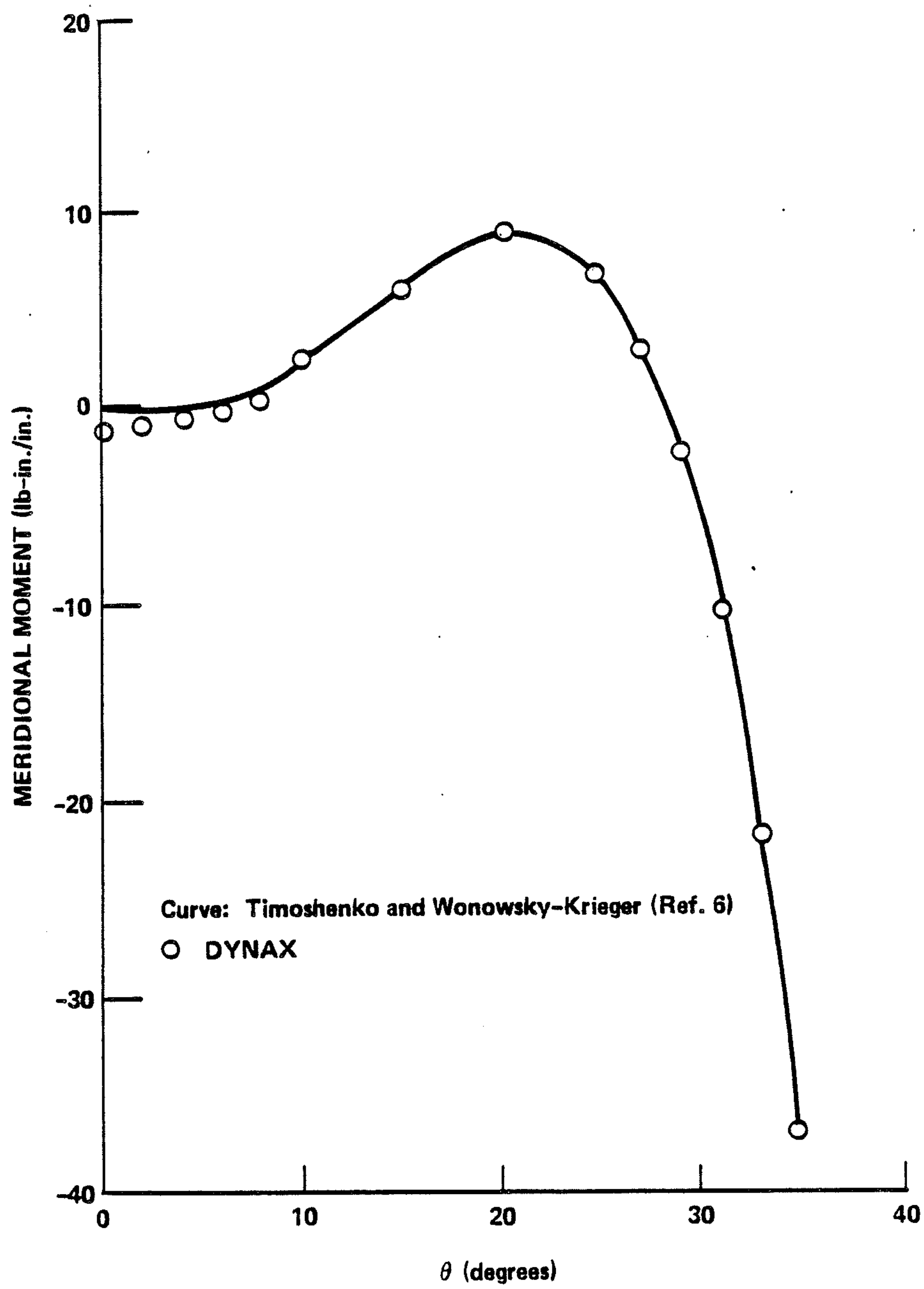

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FIGURE 3.13-17

MERIDIONAL MOMENT – SHALLOW SPHERICAL SHELL

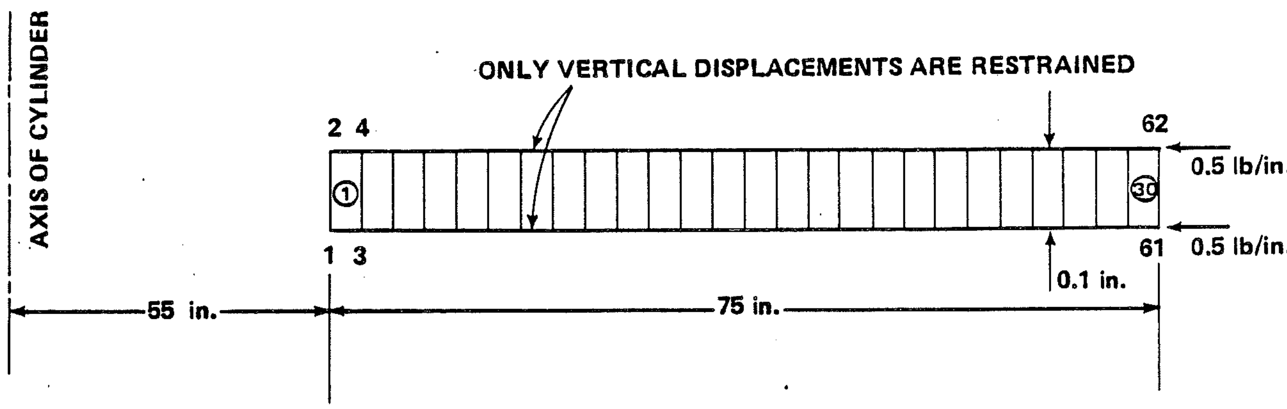

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FIGURE 3.13-18

FINITE ELEMENT IDEALIZATION OF THICK-WALLED CYLINDER

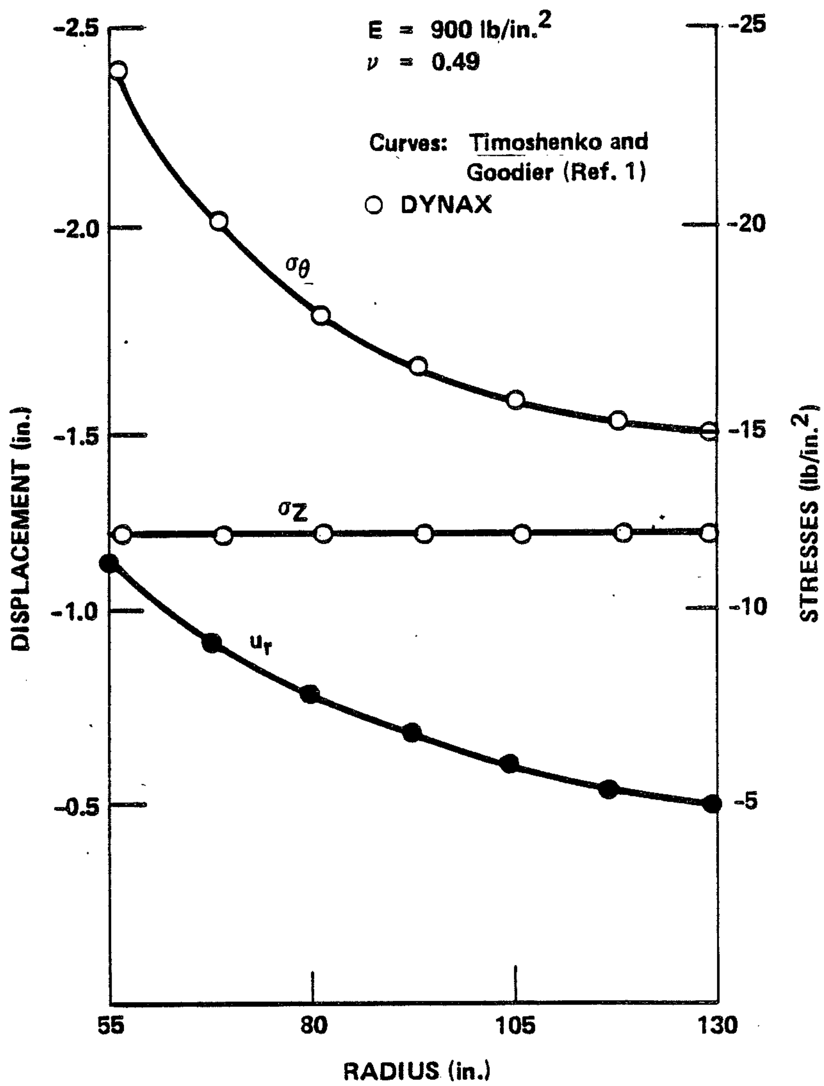

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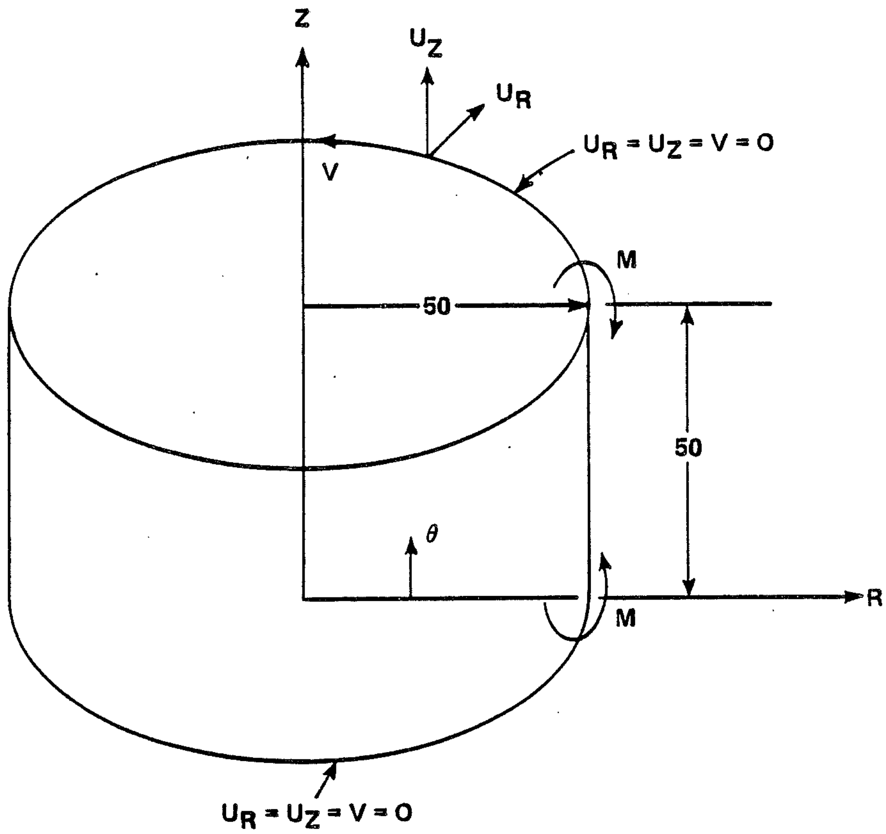
FIGURE 3.13-19

STRESSES AND DISPLACEMENTS
THICK-WALLED CYLINDERS



T = Shell thickness = 1 in.
M = 1 lb-in./in.
E = 91 lb/in.^{2}
ν = 0.3
N = Fourier harmonic number

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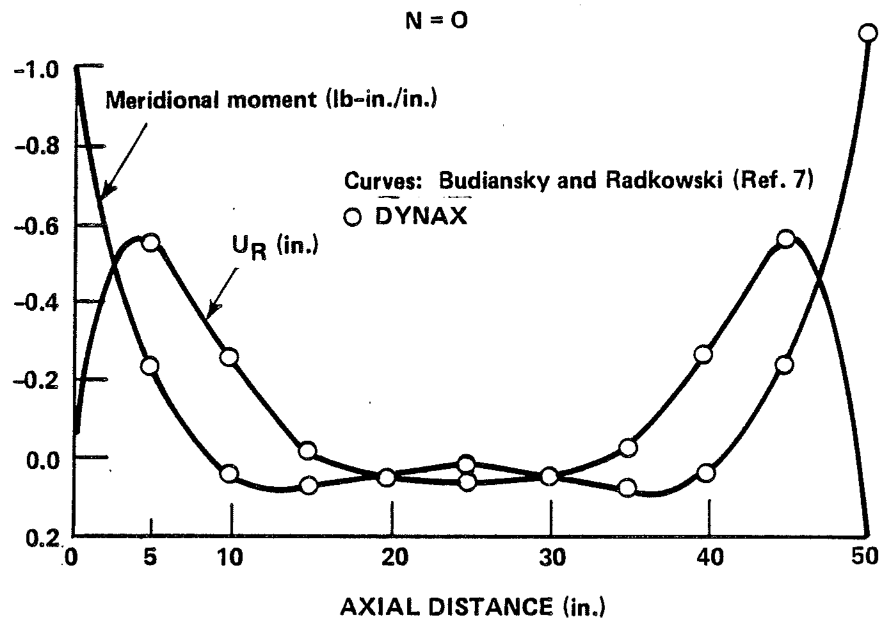
FIGURE 3.13-20

CYLINDER UNDER HARMONIC LOADS



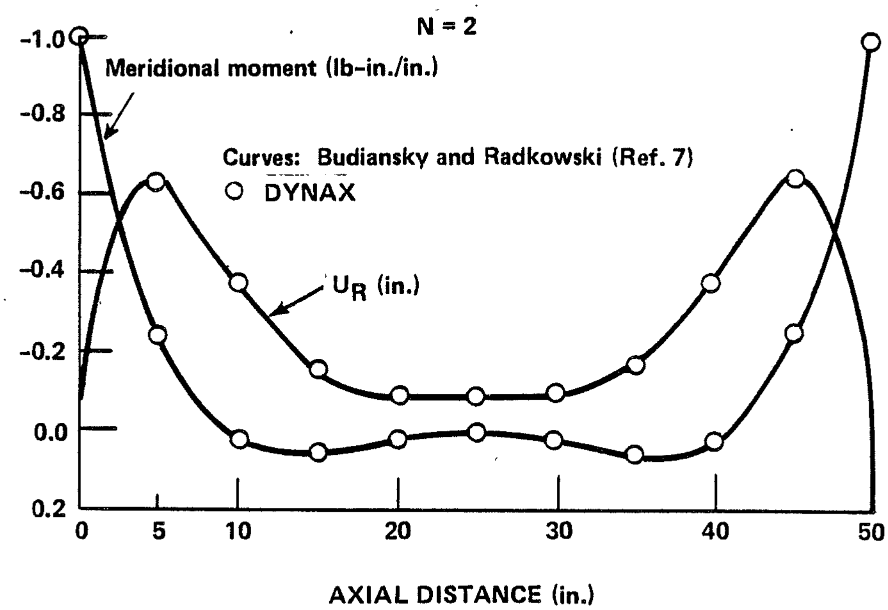


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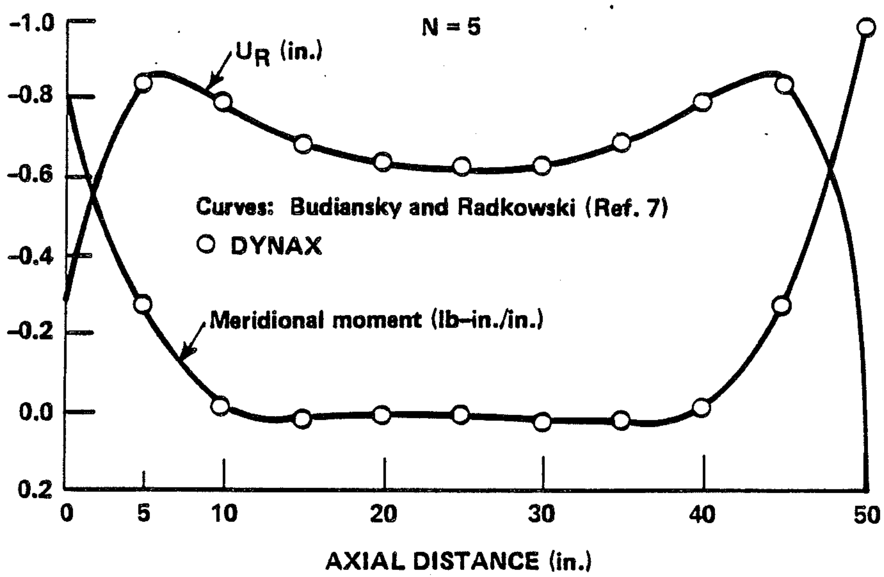
FIGURE 3.13-21, SHEET 1

MERIDIONAL MOMENTS AND DEFLECTIONS OF CYLINDER



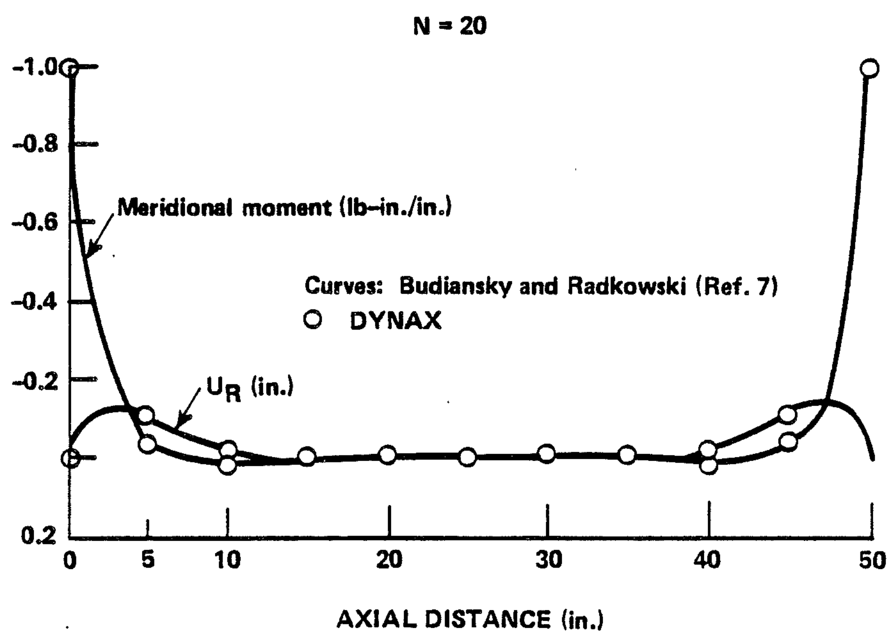


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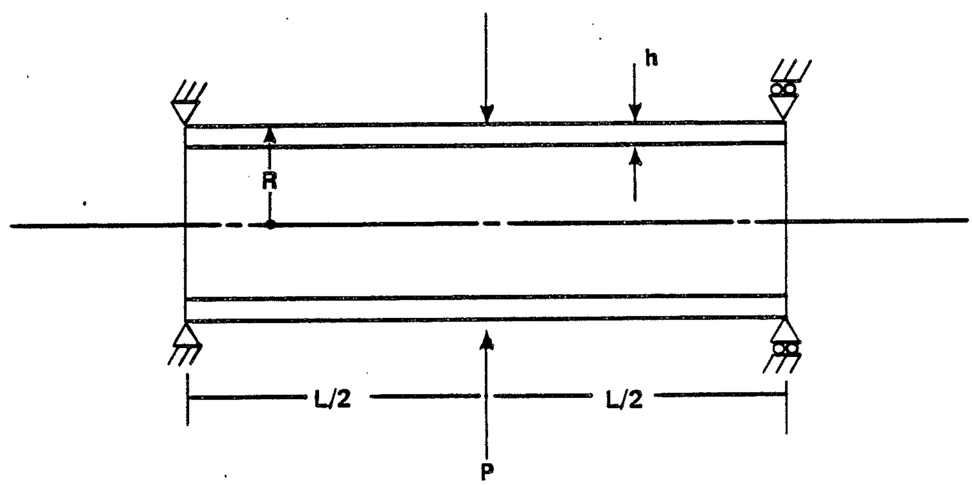
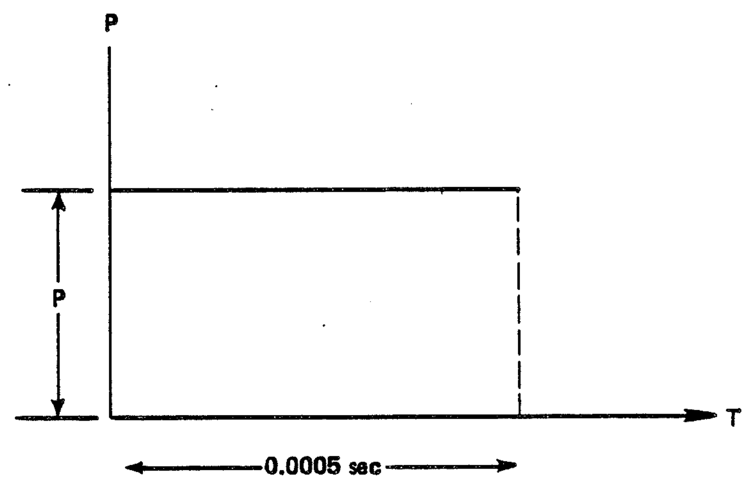
FIGURE 3.13-21, SHEET 2

MERIDIONAL MOMENTS AND DEFLECTIONS OF CYLINDER





TIME HISTORY LOADING

L = 18 in.
P = 500 lb
R = 3 in.
h = 0.3 in.
E = 30×10^6 lb/in.^{2}

Mass density (ρ) = 0.0187 lb-sec^2/in.^{4}

$\nu = 0.3$

Time step = 0.000005 sec

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FIGURE 3.13-22

SUDDENLY APPLIED RING (LINE) LOAD

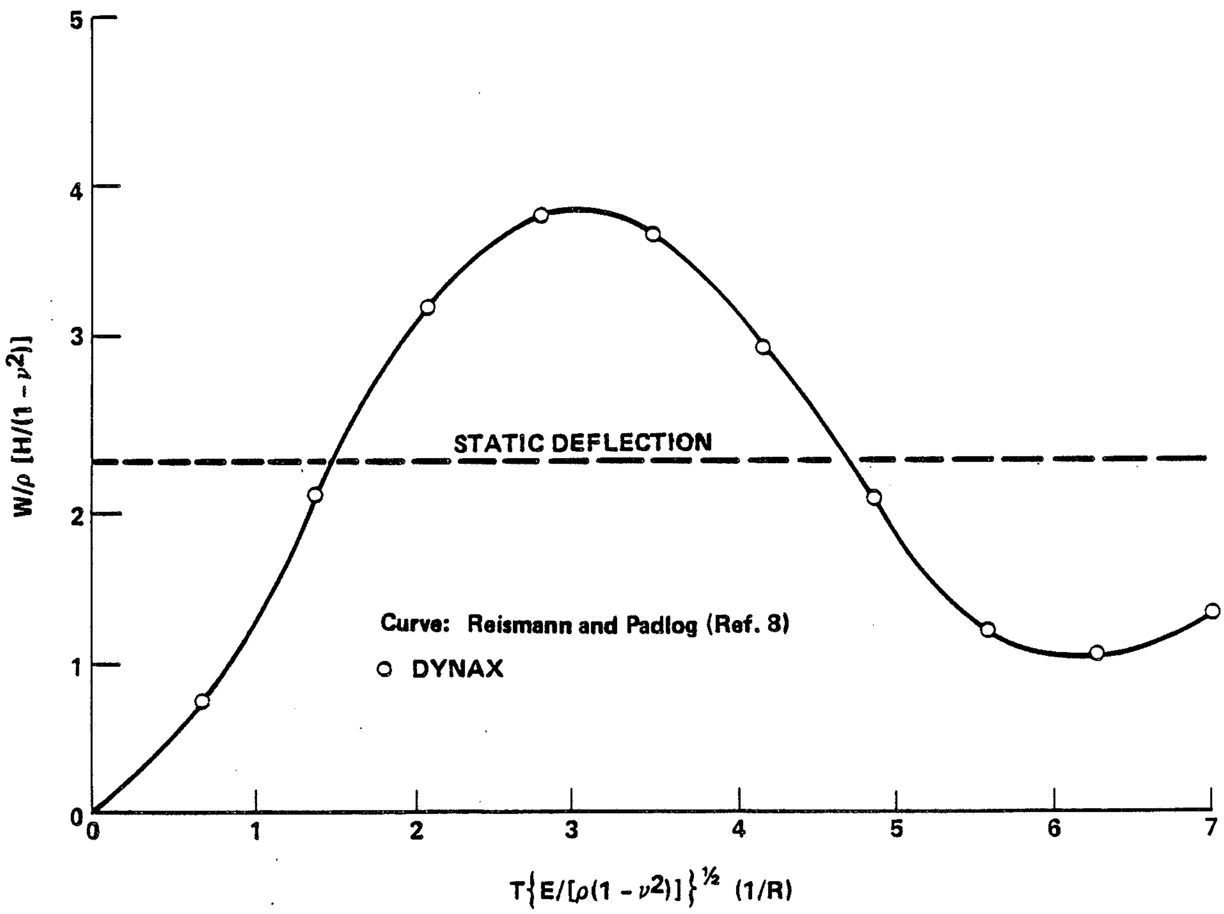

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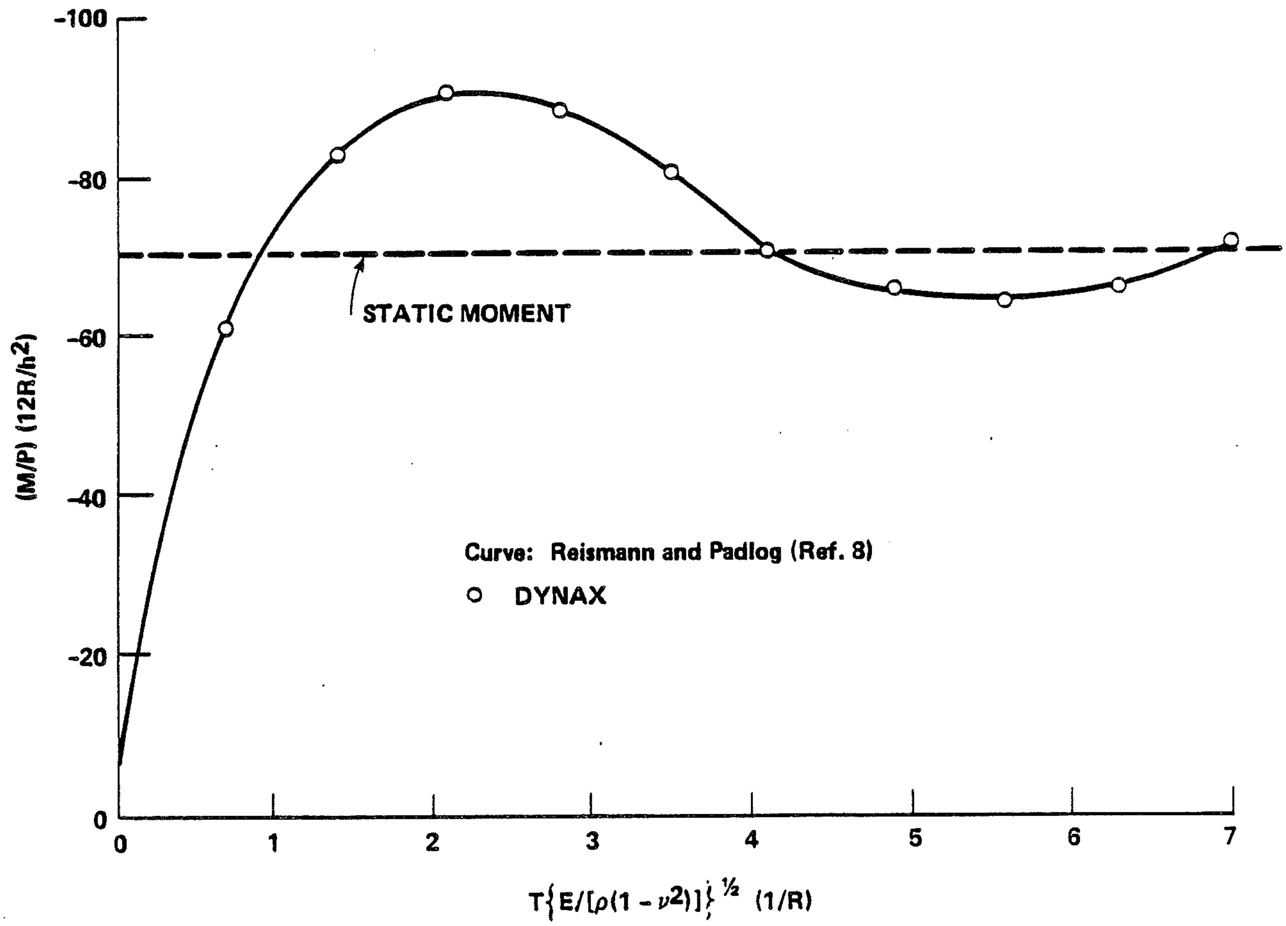
FIGURE 3.13-23

RADIAL DISPLACEMENT (W) VERSUS TIME



M = Meridional moment

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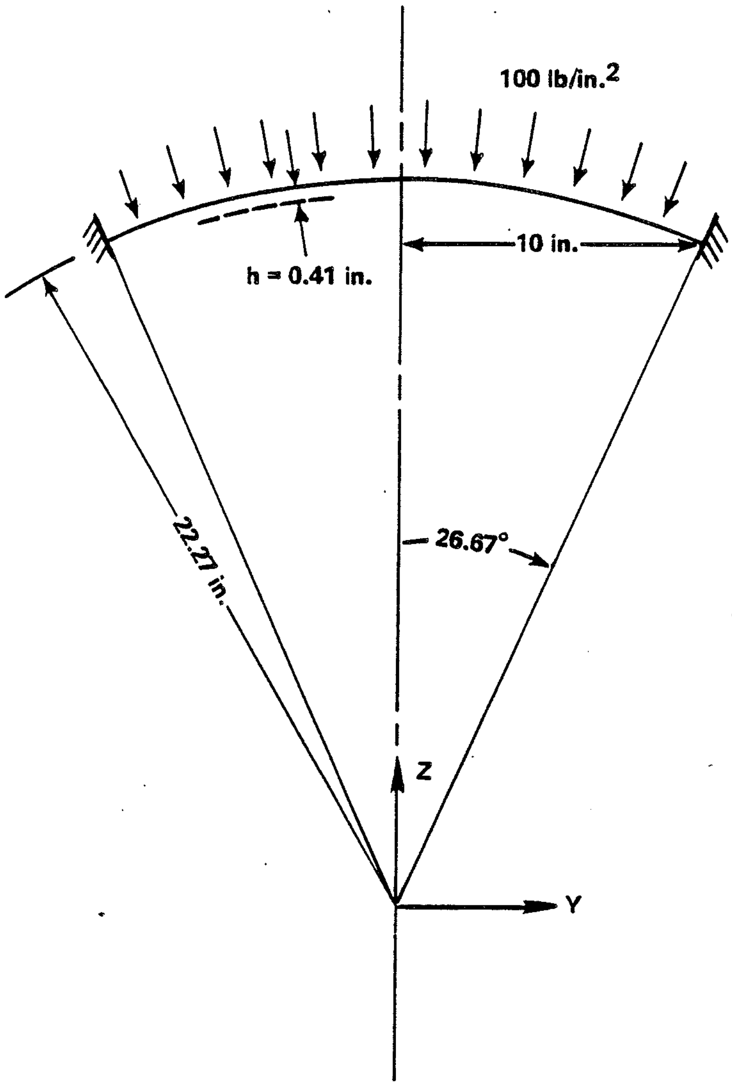
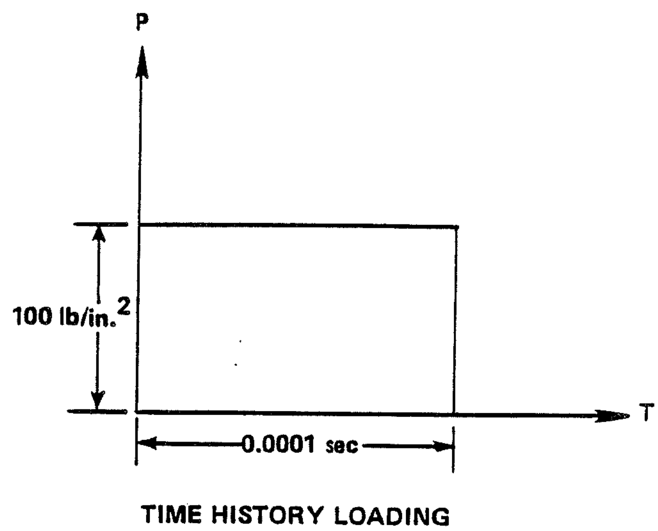
FIGURE 3.13-24

BENDING MOMENT VERSUS TIME
SUDDENLY APPLIED RING (LINE) LOAD





$E = 10.5 \times 10^6$ lb/in.^{2}
$\nu = 0.3$
$\rho = 2.46 \times 10^{-4}$ lb-sec^2/in.^{4}

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FIGURE 3.13-25

DIMENSIONS AND TIME HISTORY OF LOADING FOR SHALLOW SPHERICAL CAP WITH CLAMPED SUPPORT UNDER SUDDEN UNIFORM LOAD AS ANALYZED IN PROBLEM 5 (DYNAX)

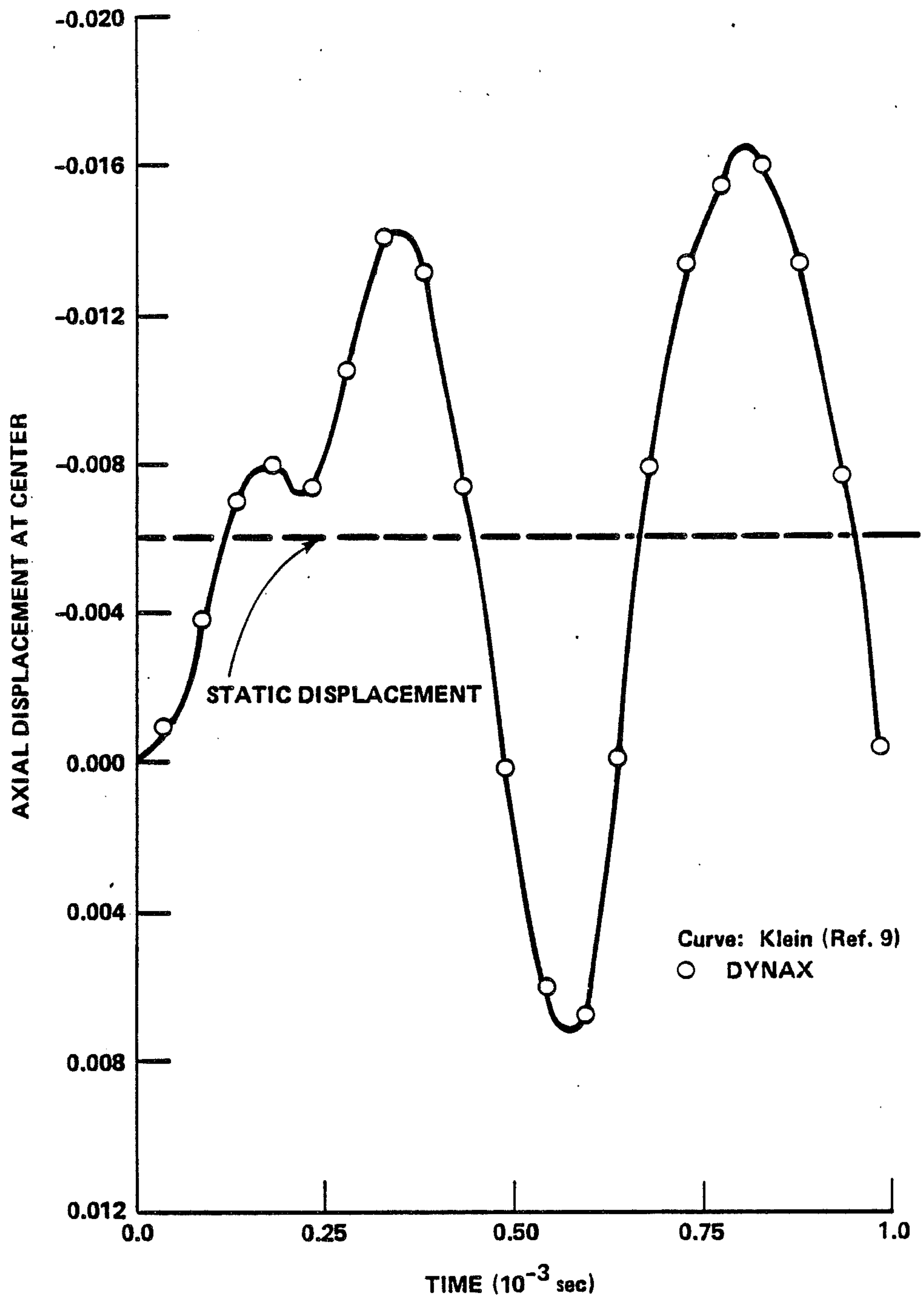

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FIGURE 3.13-26

AXIAL DISPLACEMENT OF SPHERICAL CAP
UNDER DYNAMIC LOAD

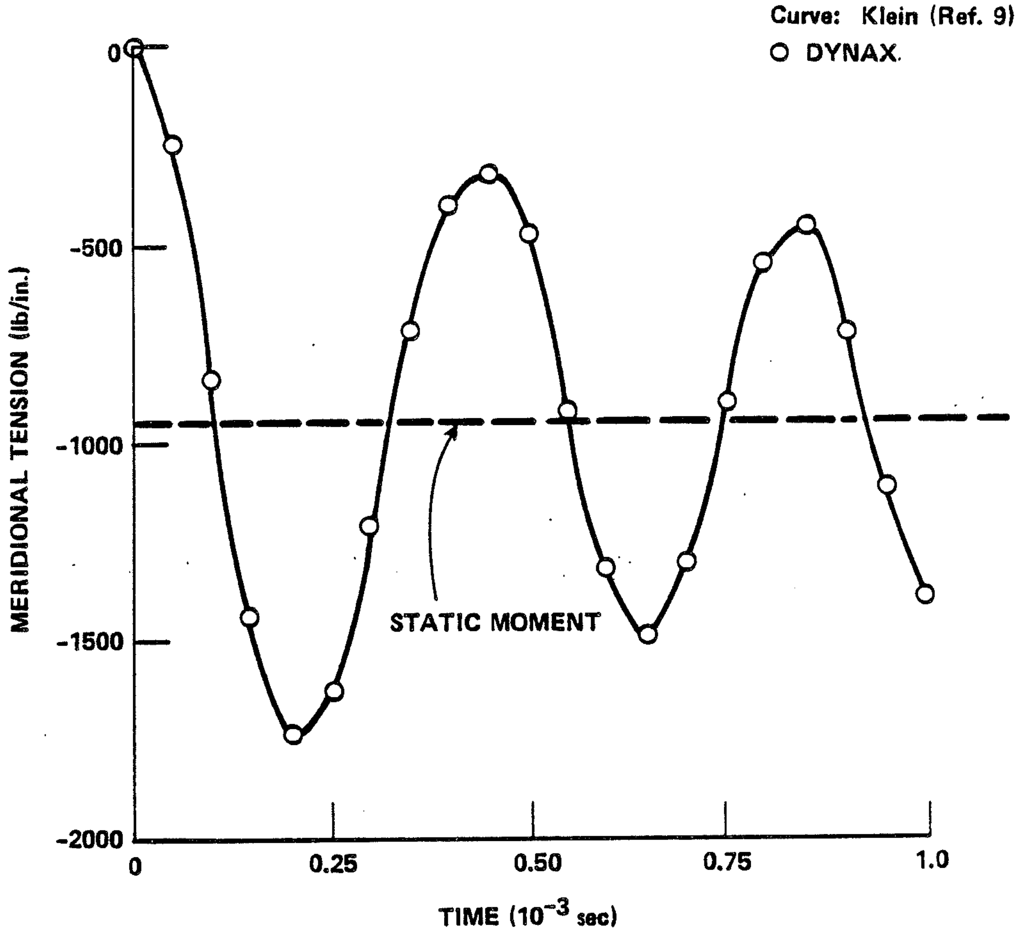

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FIGURE 3.13-27

MERIDIONAL TENSION OF SPHERICAL CAP UNDER DYNAMIC LOAD

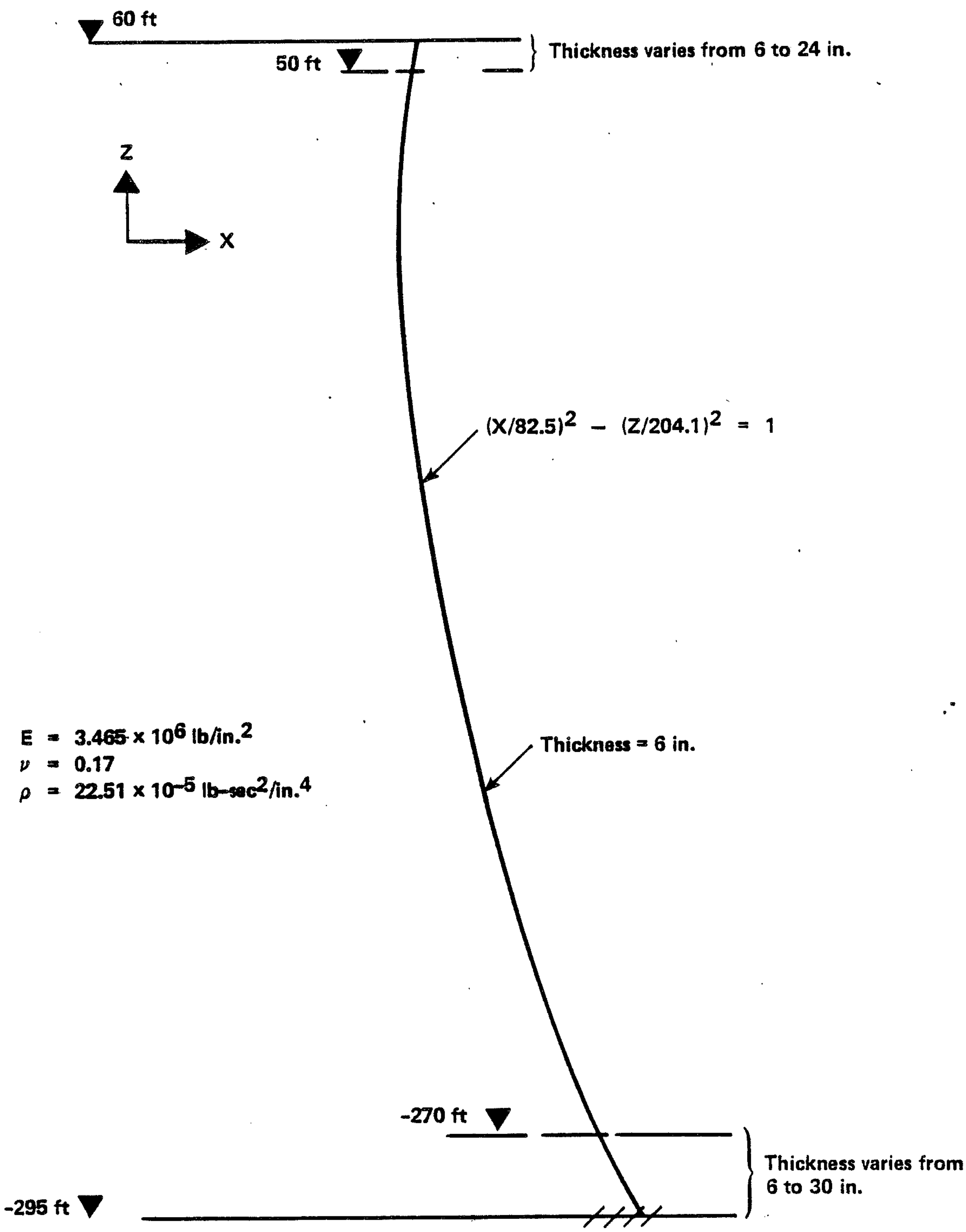

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FIGURE 3.13-28

HYPERBOLIC COOLING TOWER

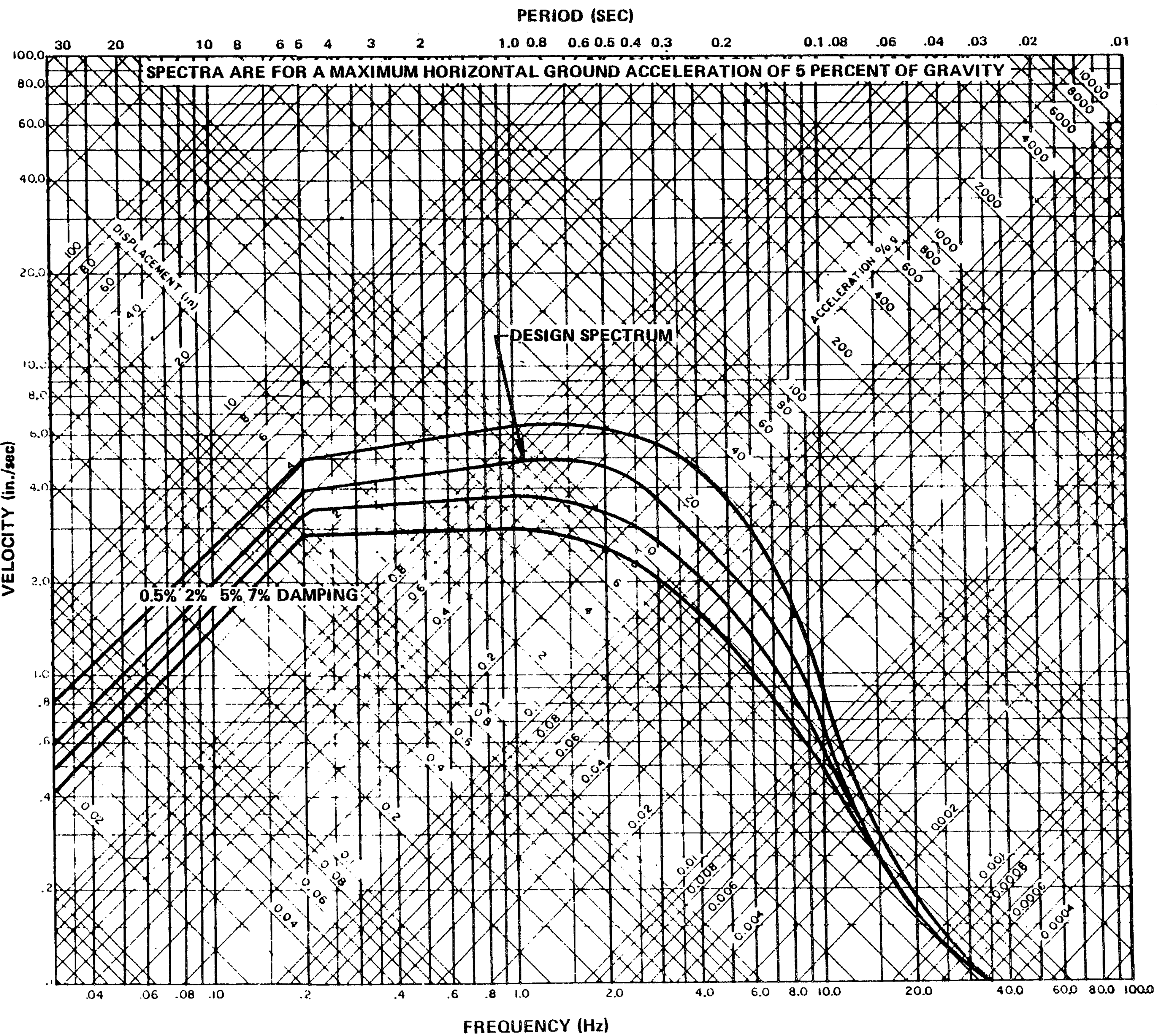

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FIGURE 3.13-29
SPECTRUM OF DESIGN EARTHQUAKE

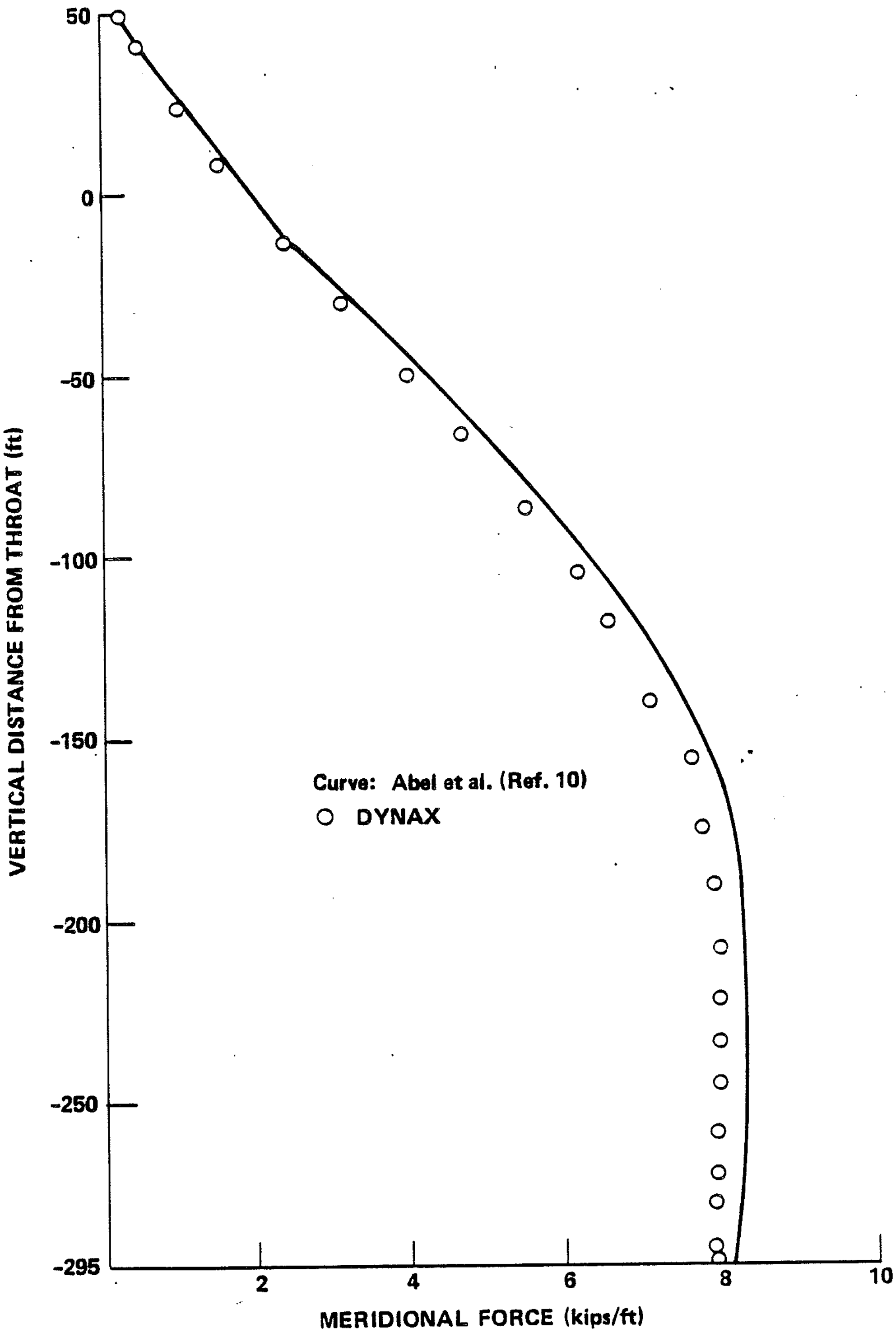

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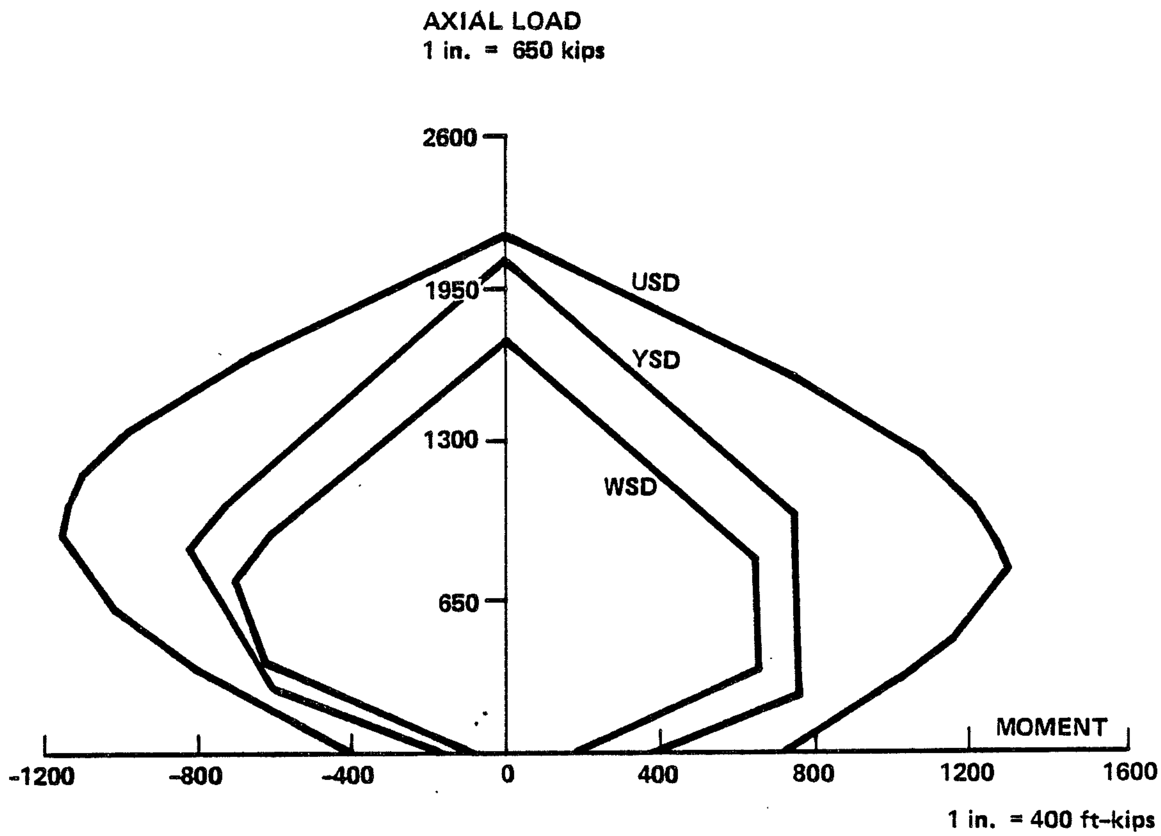
FIGURE 3.13-30

COOLING TOWER MERIDIONAL FORCE



LEGEND (see Table 3.13–3)

TEST 4 48 x 12 STEEL
T = 2.75/40
C = 1.25/10
FCP = 4.5
FP = 60

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FIGURE 3.13-31

INTERACTION DIAGRAM – AXIAL LOAD VERSUS BENDING MOMENT

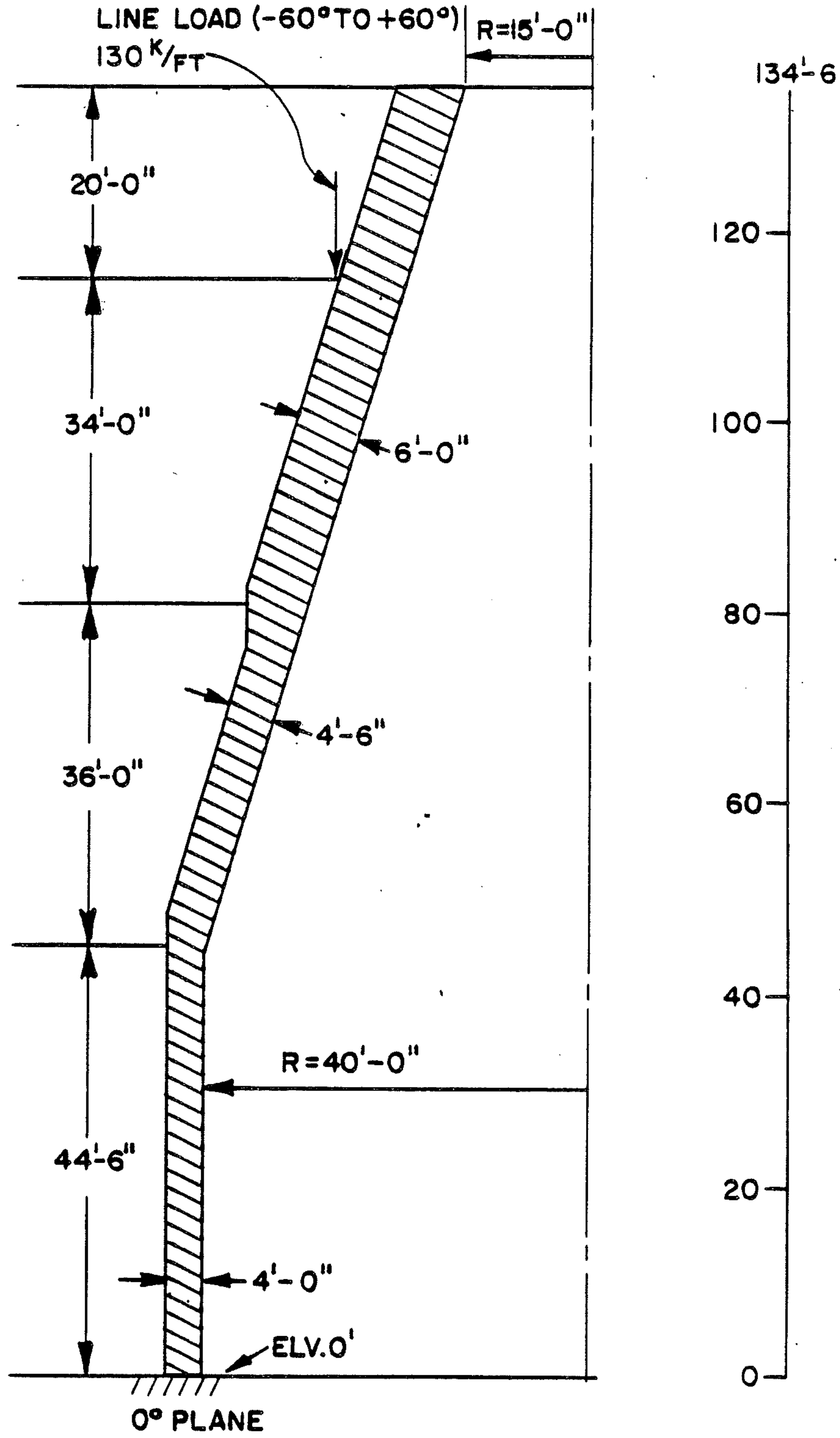

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FIGURE 3.13-32

KALSHEL VALIDATION EXAMPLE
ECCENTRIC LINE LOAD

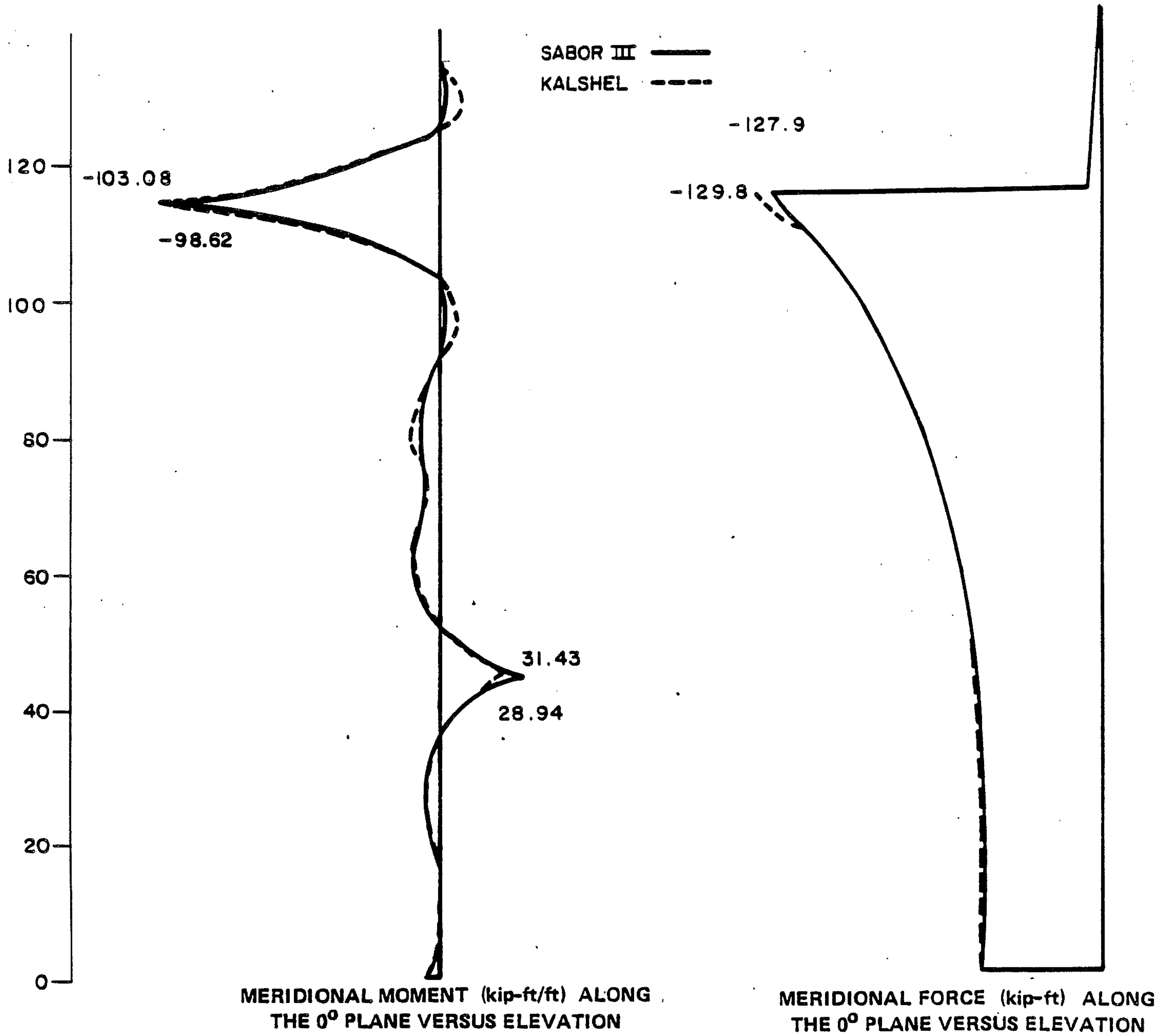

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FIGURE 3.13-33

SABOR III AND KALSHEL
VALIDATION EXAMPLE
MERIDIONAL MOMENT AND FORCE

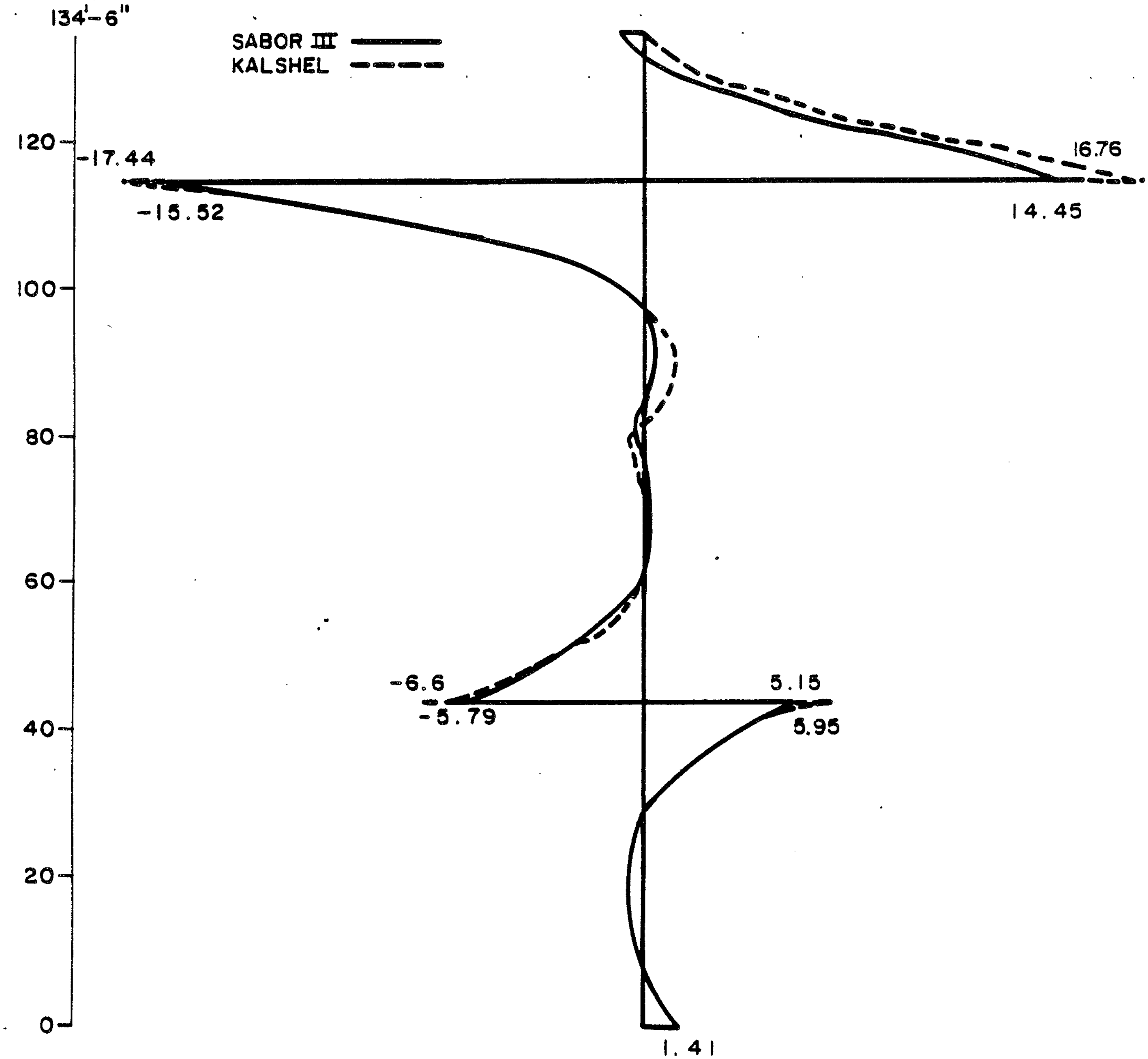

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FIGURE 3.13-34

RADIAL SHEAR (KIP/FT) ALONG THE 0° PLANE VERSUS ELEVATION (FT)

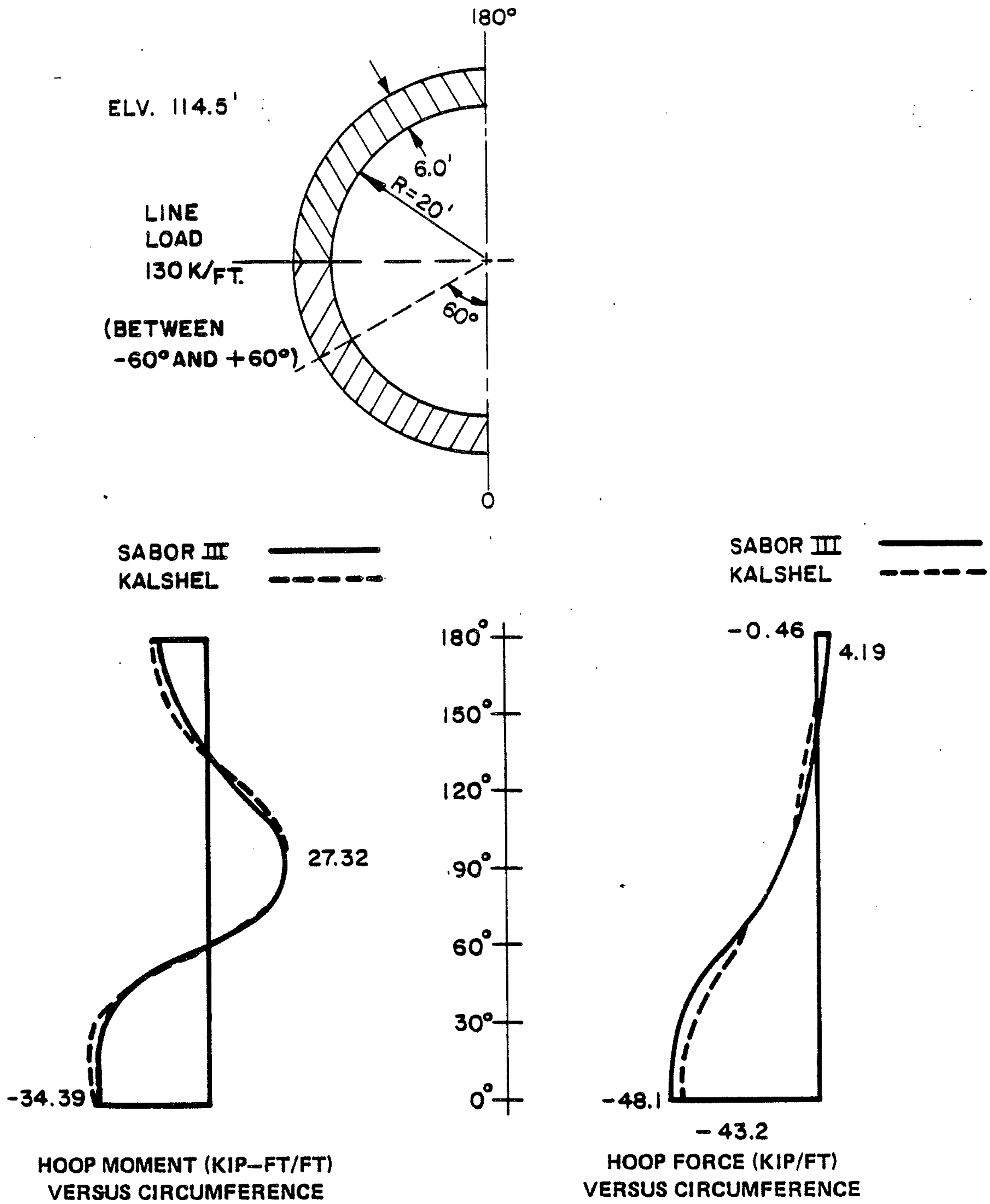

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FIGURE 3.13-35

SABOR III AND KALSHEL
VALIDATION EXAMPLE
HOOP MOMENT AND FORCE

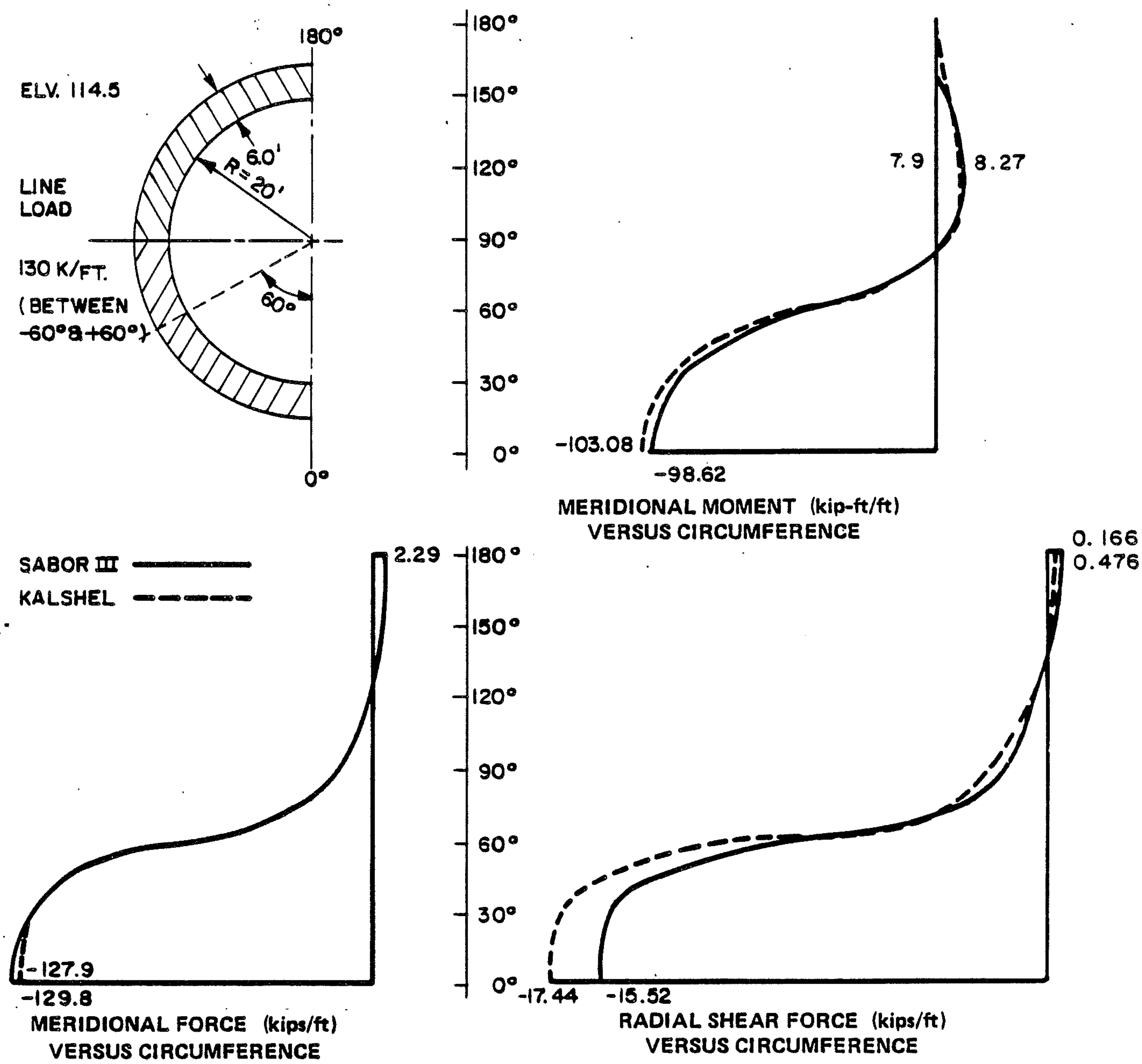

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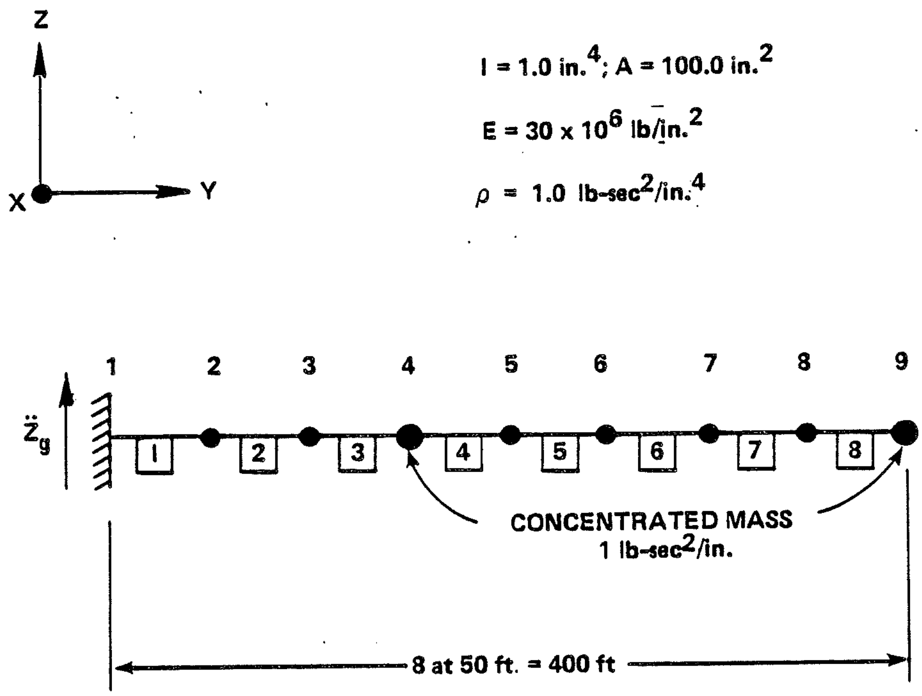
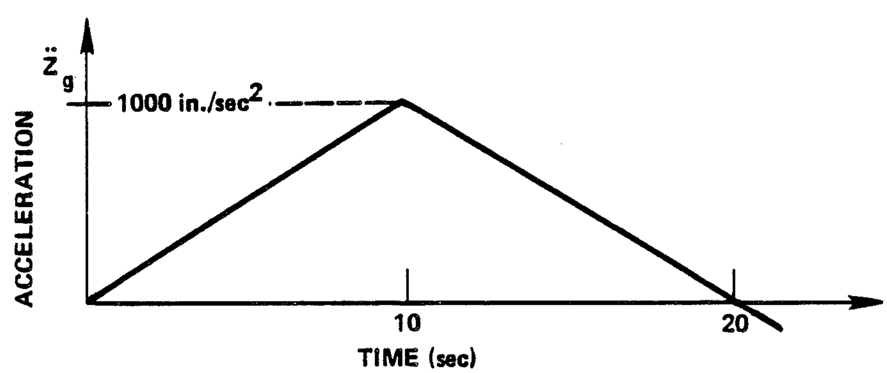
FIGURE 3.13-36

SABOR III AND KALSHEL
VALIDATION EXAMPLE
MERIDIONAL MOMENT AND FORCE AND
RADIAL SHEAR FORCE



(a) NODE AND BEAM NUMBER ASSIGNMENTS FOR THE CANTILEVER MODEL



(b) GROUND ACCELERATION APPLIED AT NODE 1

FIGURE 3.13-37

RESPONSE HISTORY ANALYSIS OF CANTILEVER BEAM

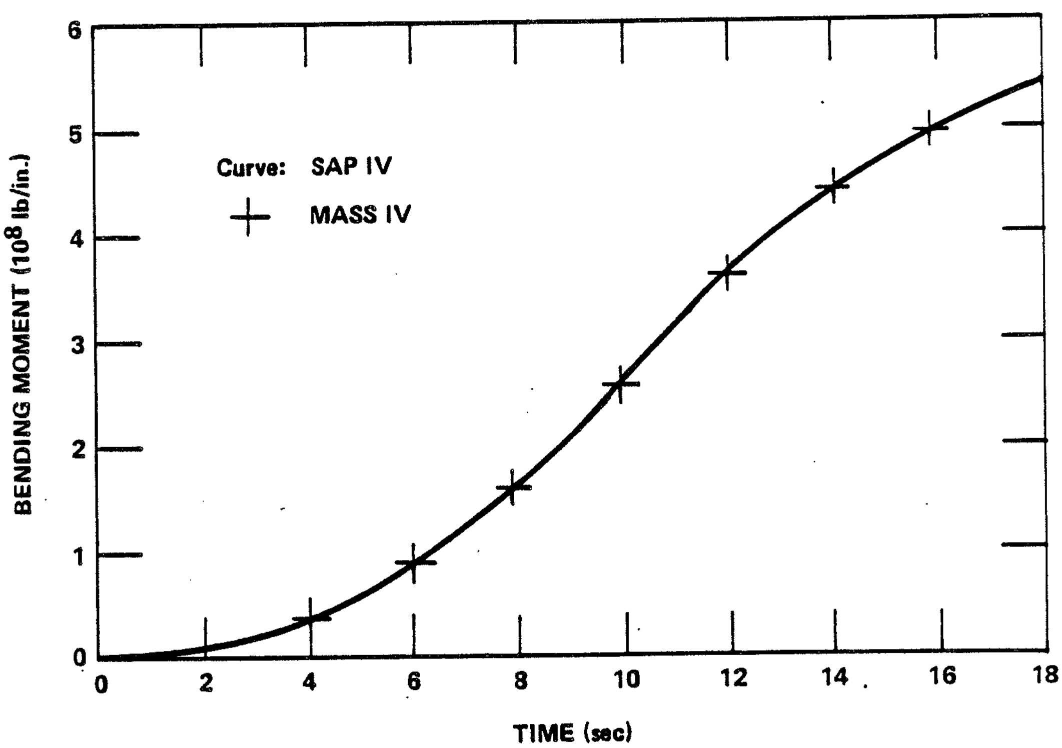


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FIGURE 3.13-38

CANTILEVER RESPONSE
MOMENT AT NODE 1
FIXED END OF CANTILEVER

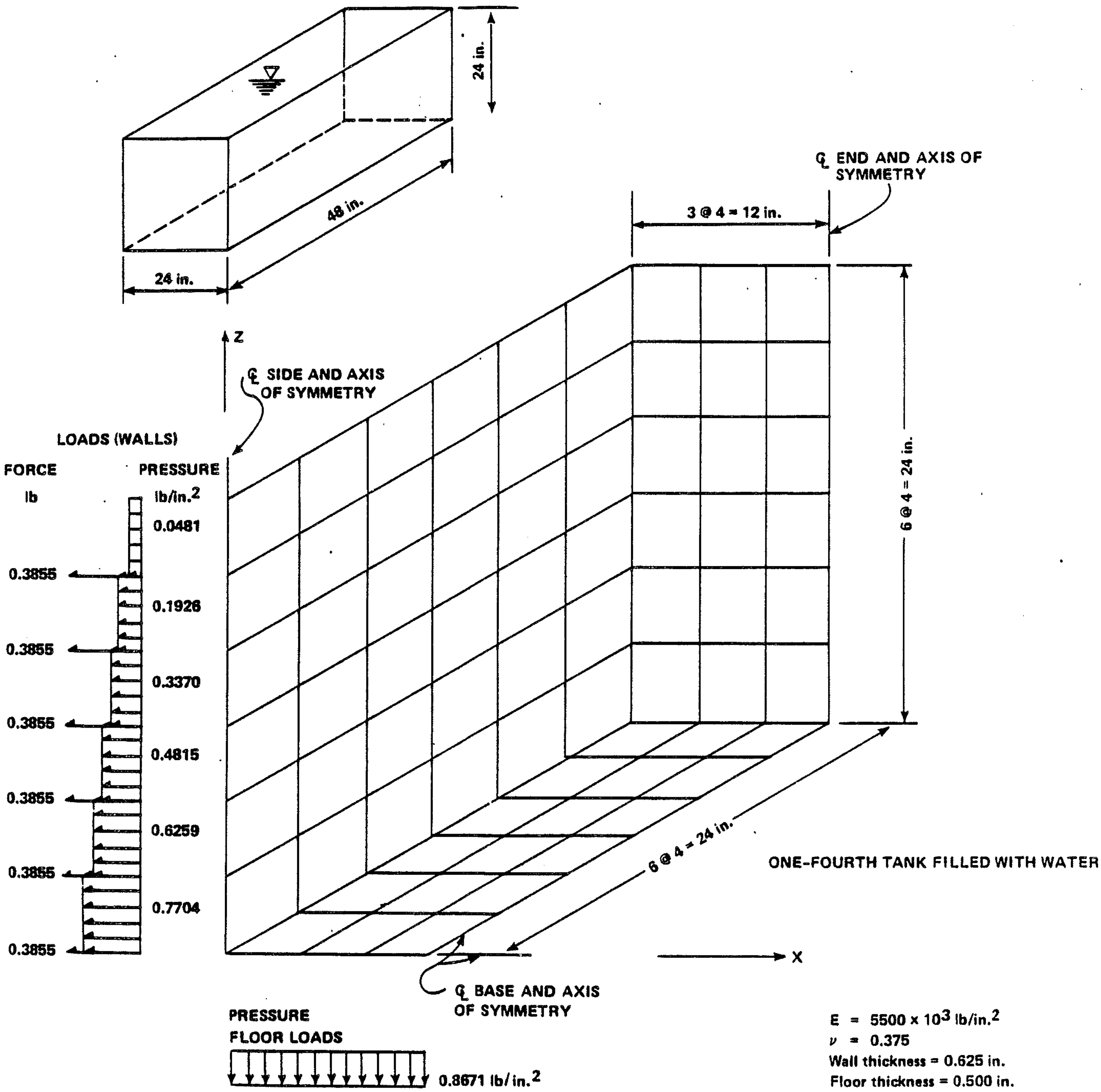

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FIGURE 3.13-39

RECTANGULAR TANK FILLED WITH WATER

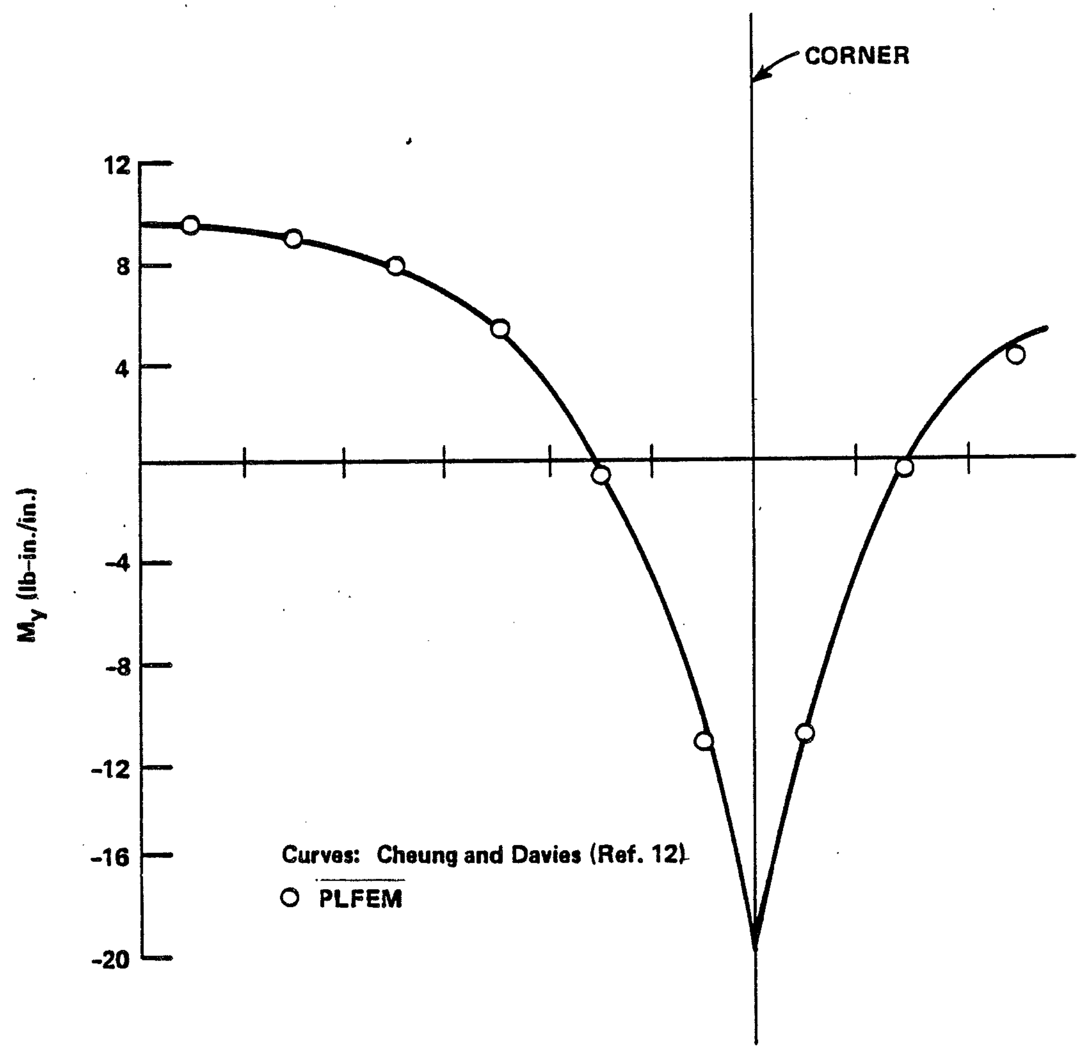

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FIGURE 3.13-40

MOMENT M_y AT HORIZONTAL CENTERLINE OF WALLS

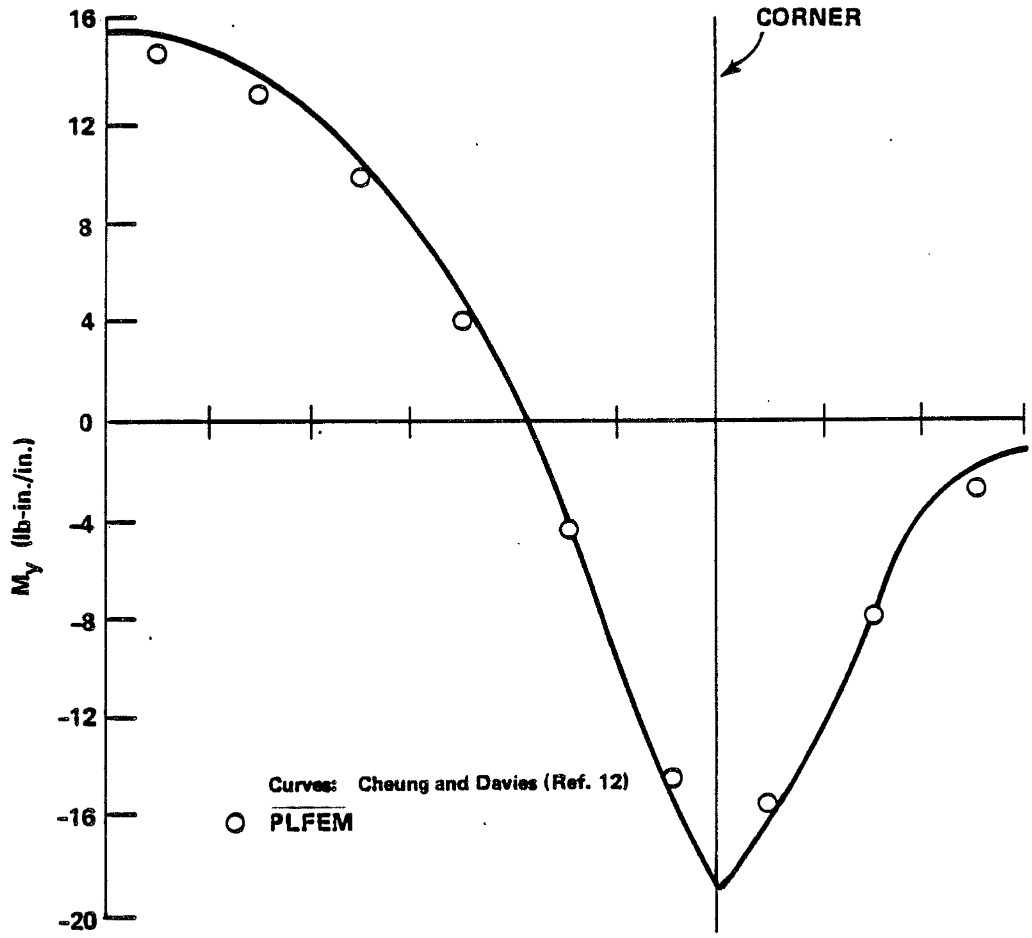

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FIGURE 3.13-41

MOMENT M_y AT TOP OF WALL

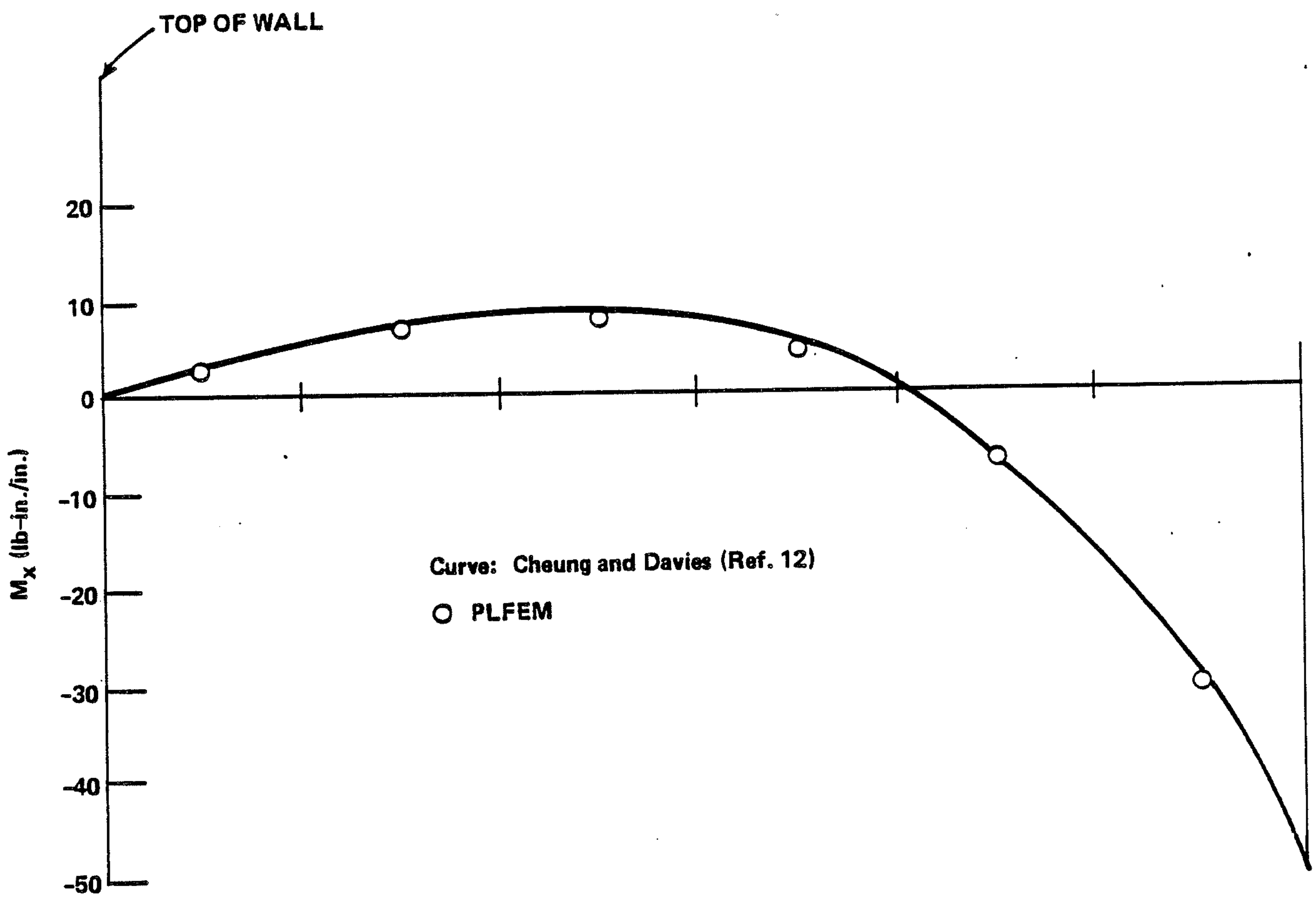

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FIGURE 3.13-42

MOMENT M_X ALONG THE CENTERLINE OF LONG WALL

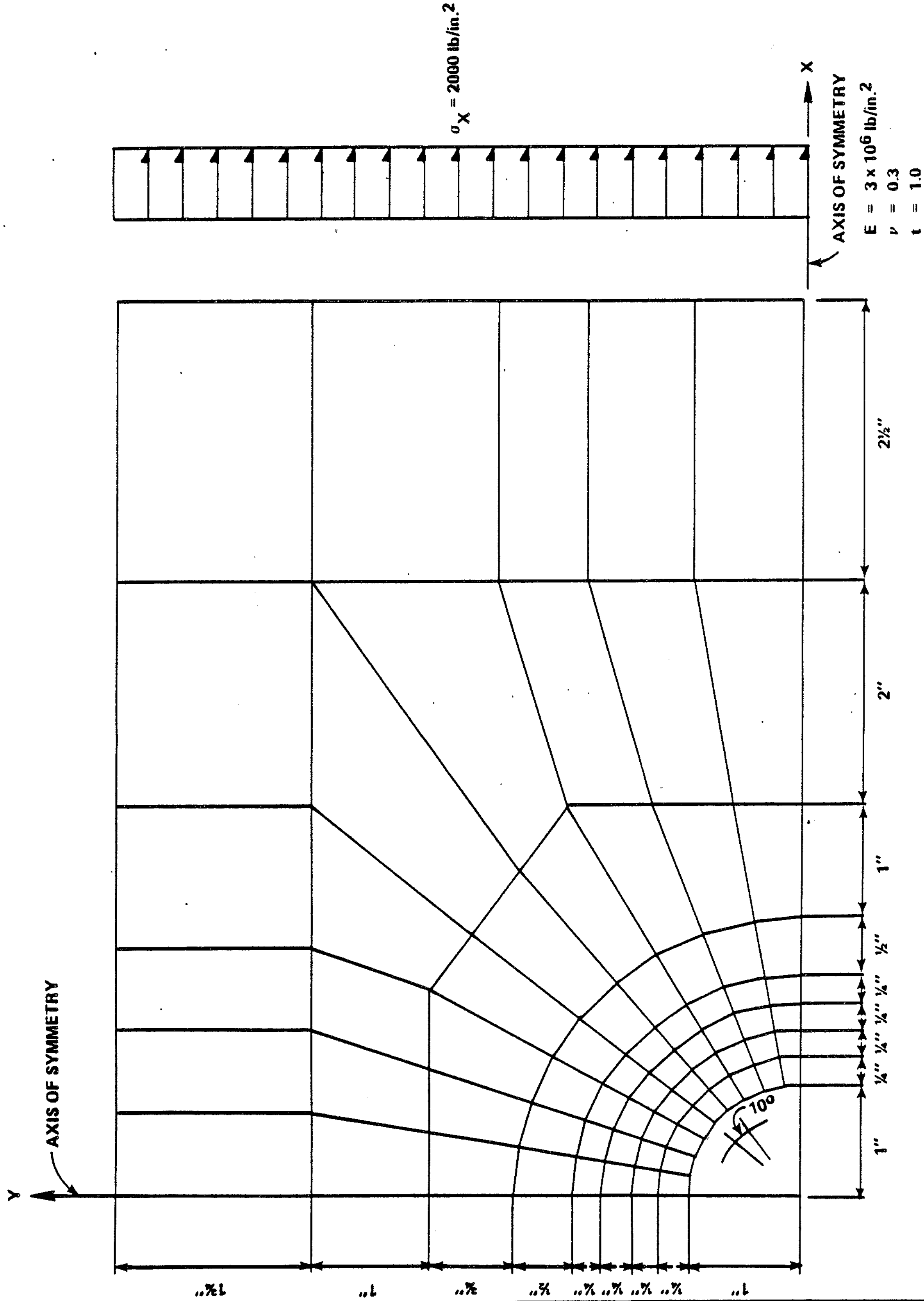

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FIGURE 3.13-43

PLATE WITH CIRCULAR HOLE UNDER UNIFORM TENSION

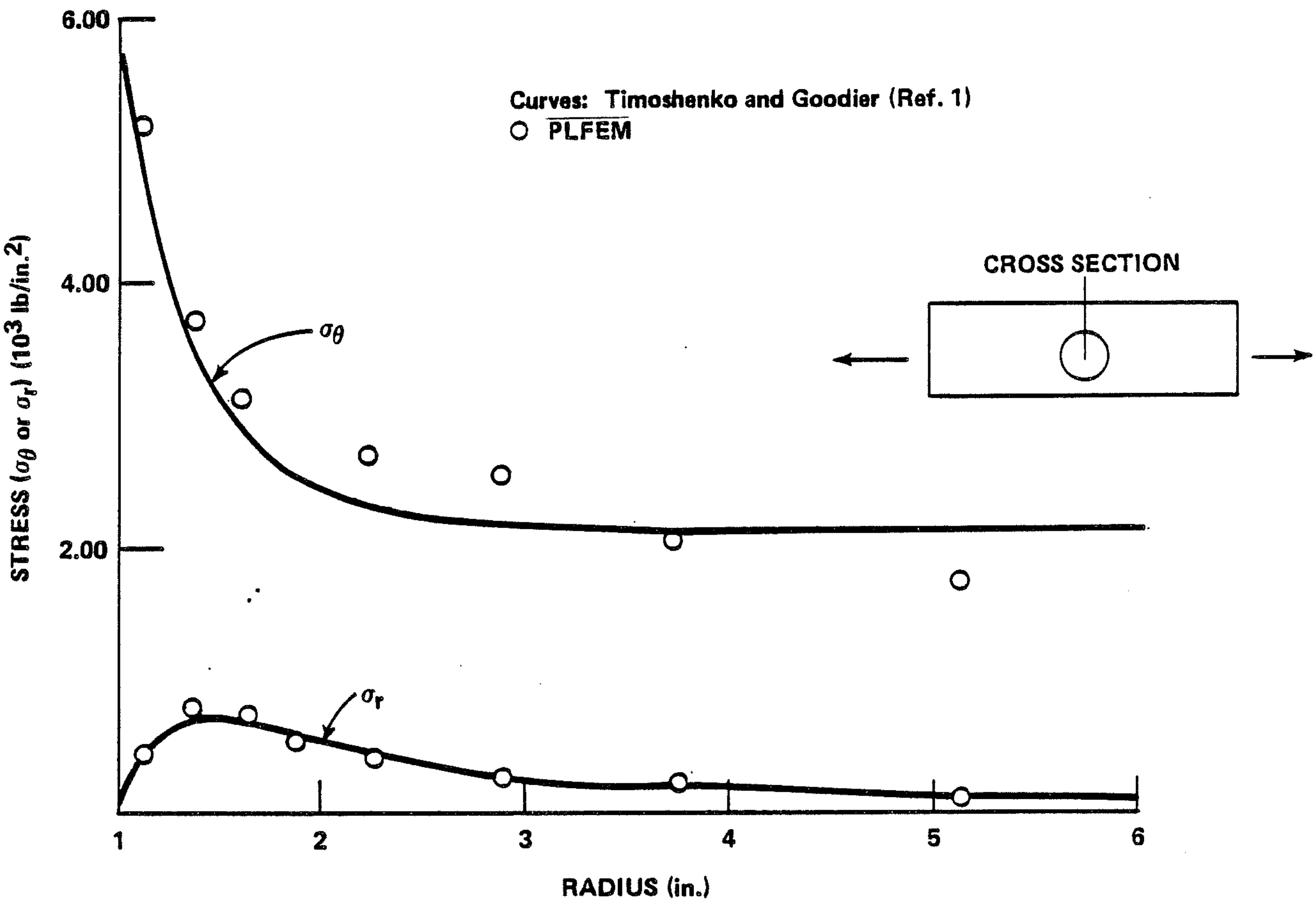

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FIGURE 3.13-44

STRESSES IN PLATE WITH CIRCULAR HOLE UNDER UNIFORM TENSION

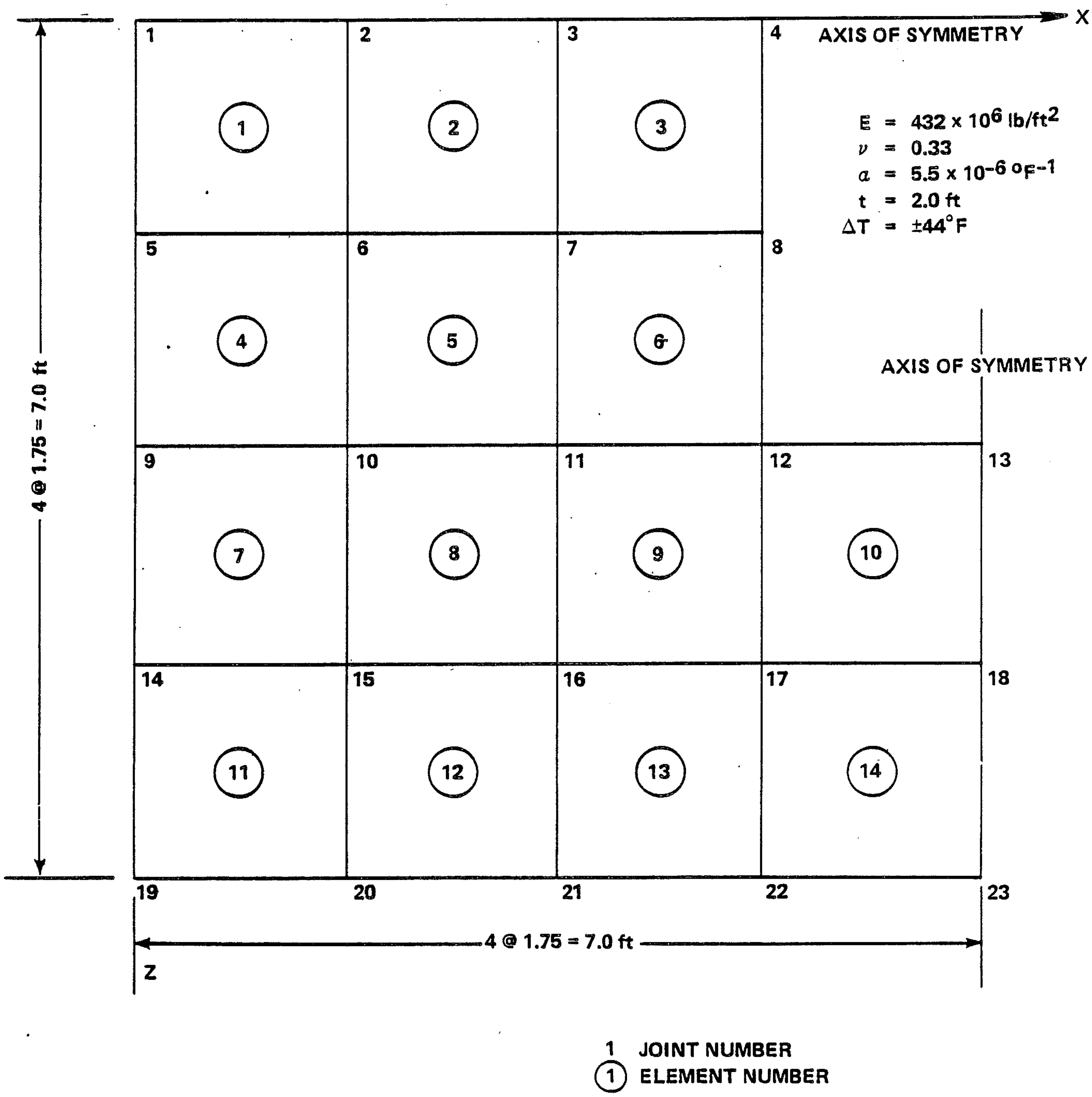

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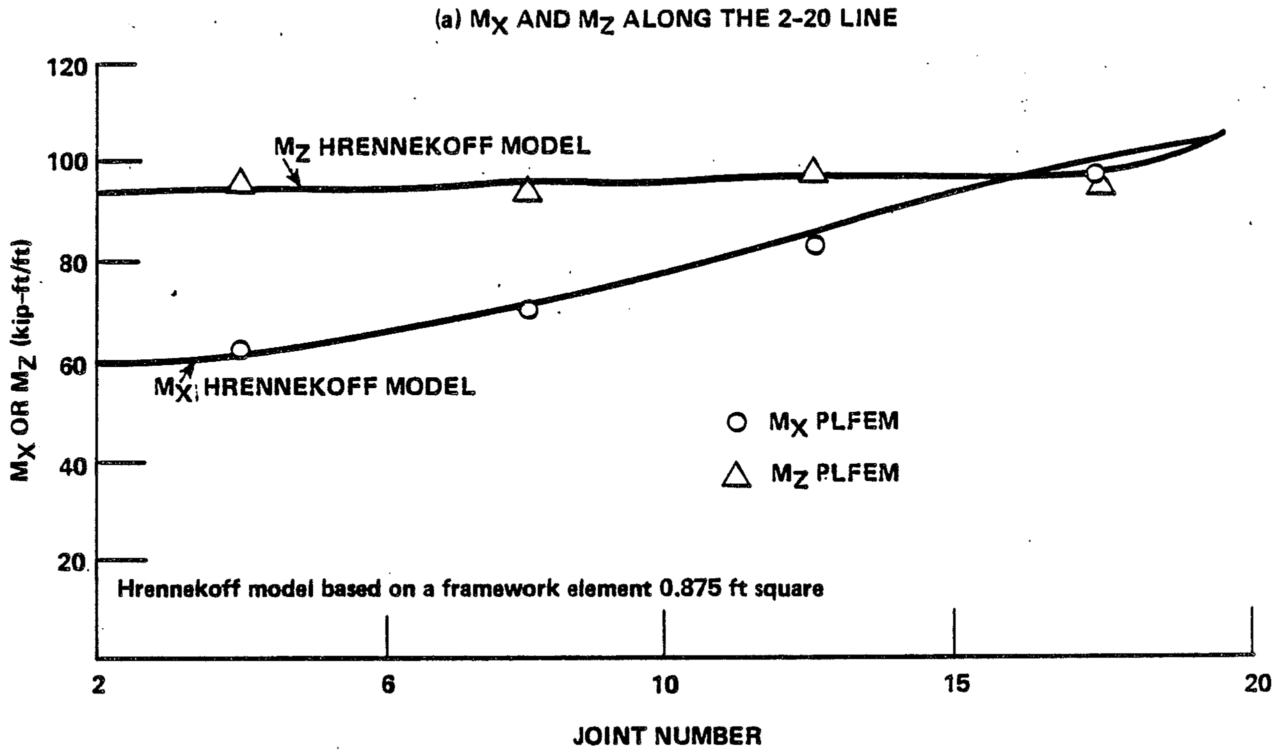
FIGURE 3.13-45

SQUARE PLATE WITH RECTANGULAR HOLE SUBJECTED TO TEMPERATURE VARIATION



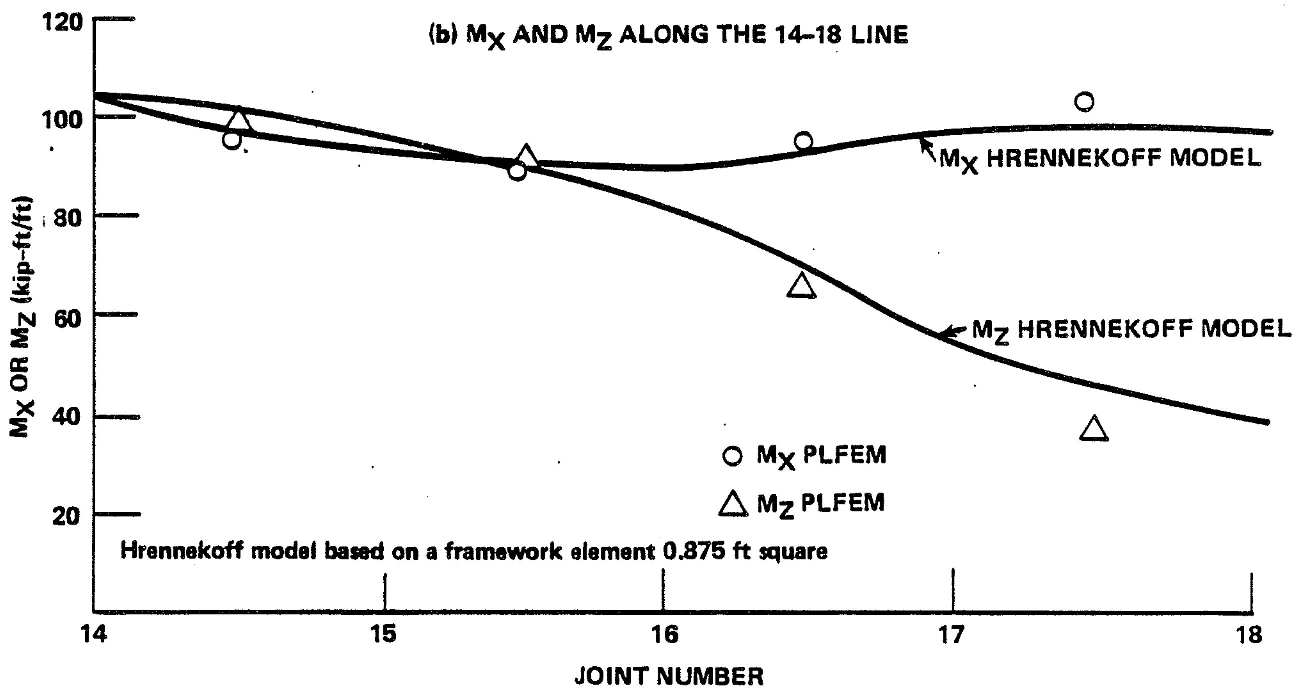


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FIGURE 3.13-46

MOMENTS IN PLATE DUE TO TEMPERATURE VARIATION

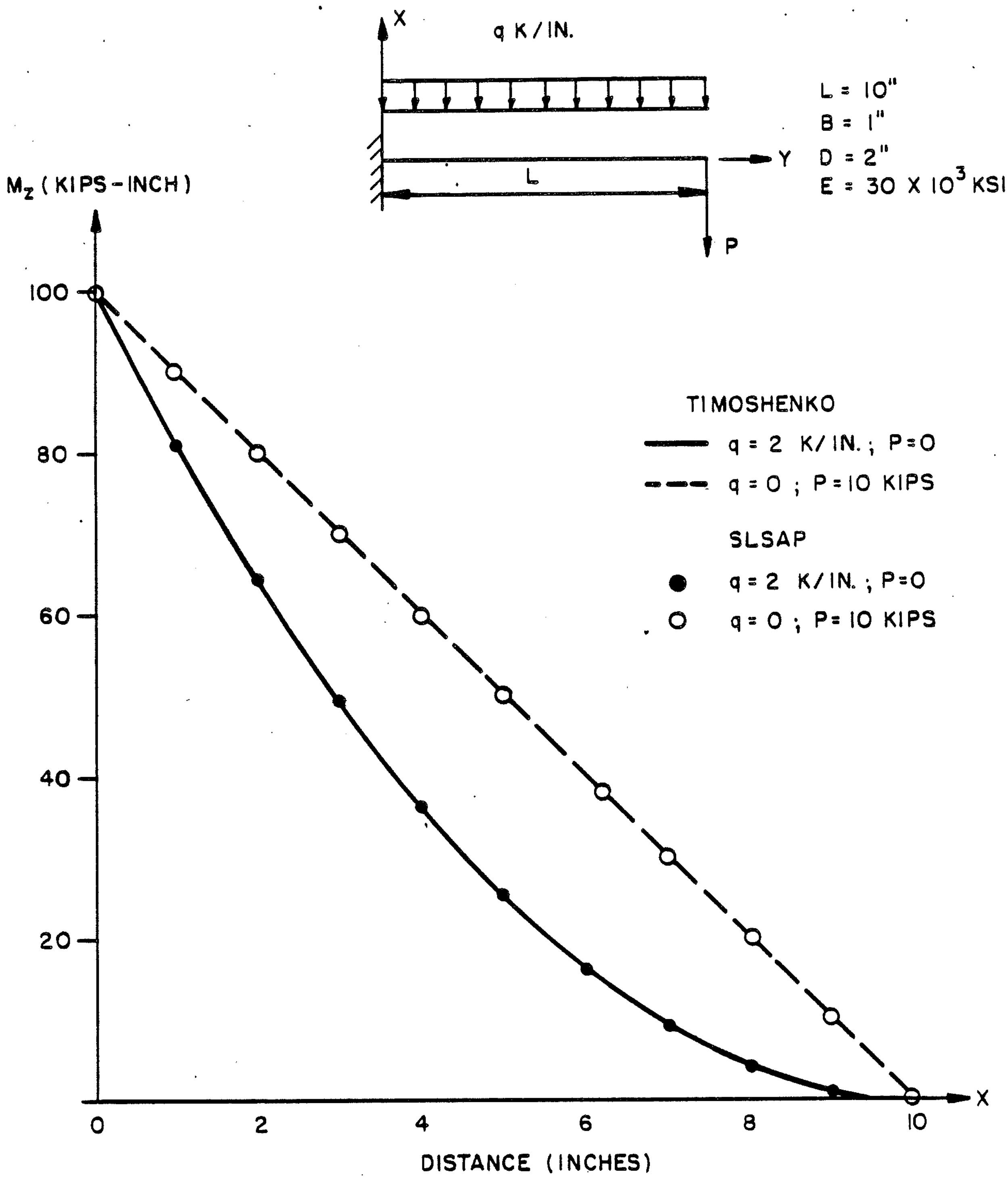

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FIGURE 3.13-47

BENDING MOMENTS IN A CANTILEVER BEAM

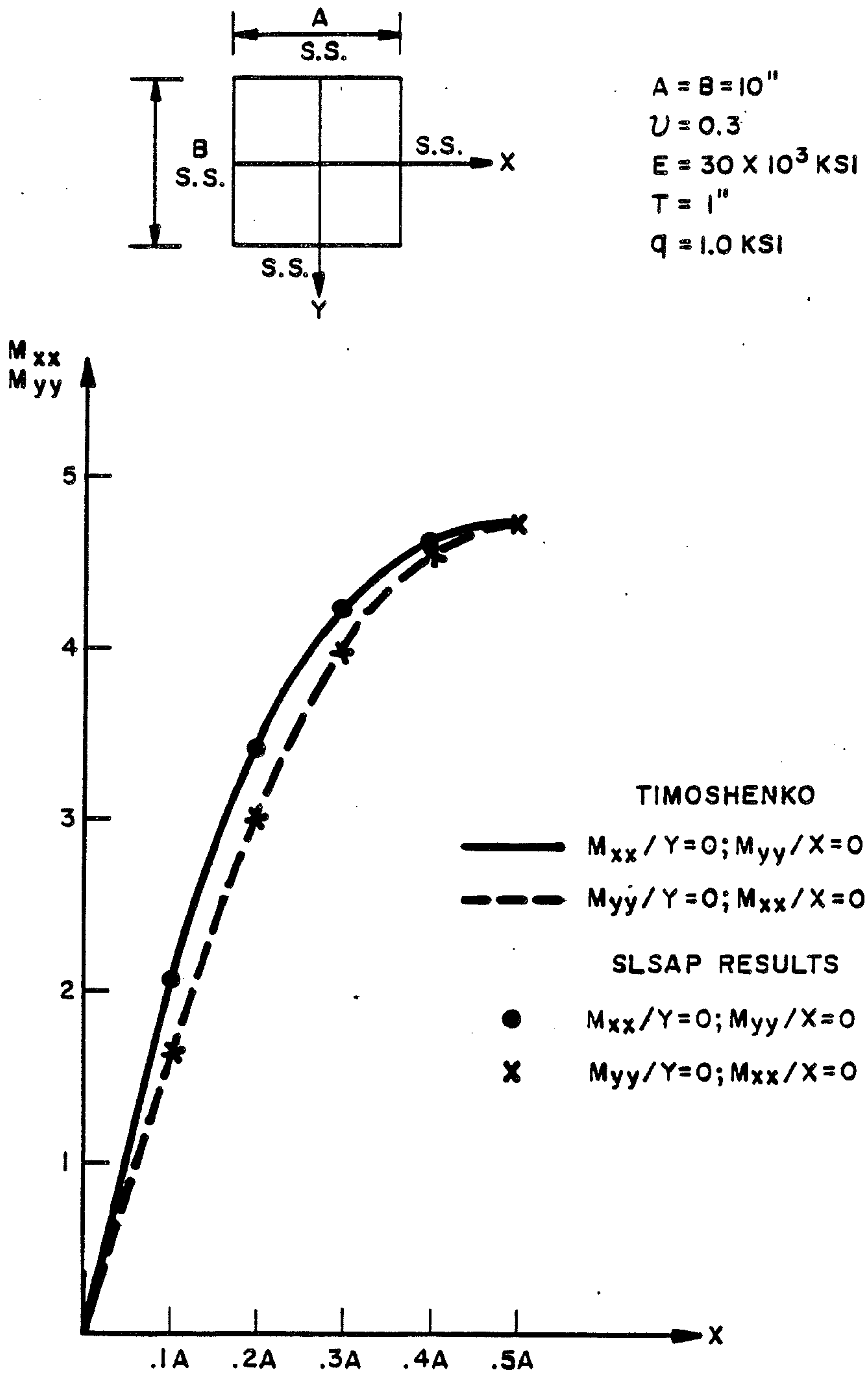

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FIGURE 3.13-48

BENDING MOMENTS IN A SIMPLY SUPPORTED PLATE

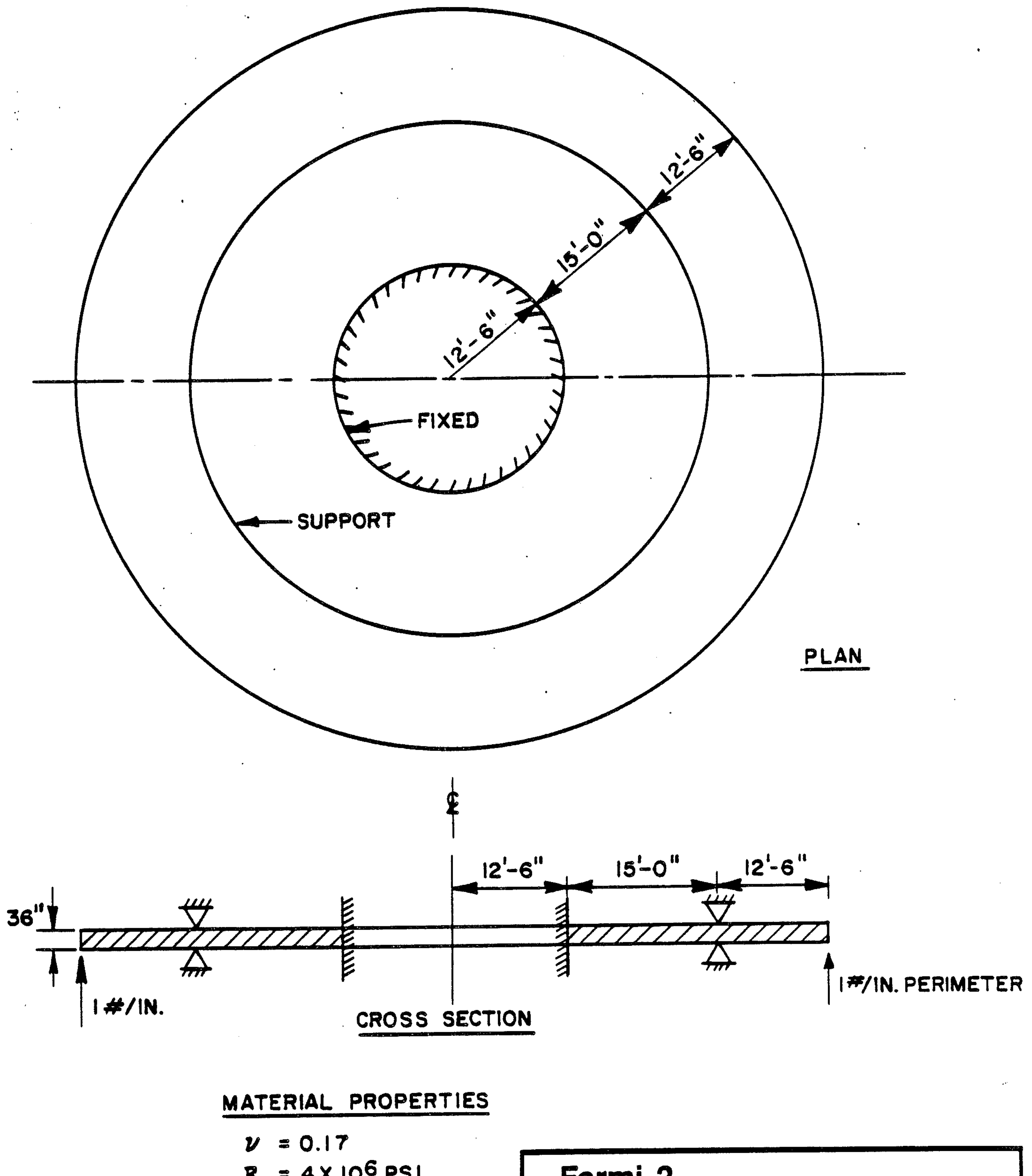

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FIGURE 3.13-49

CIRCULAR PLATE FOR SOR III EXAMPLE

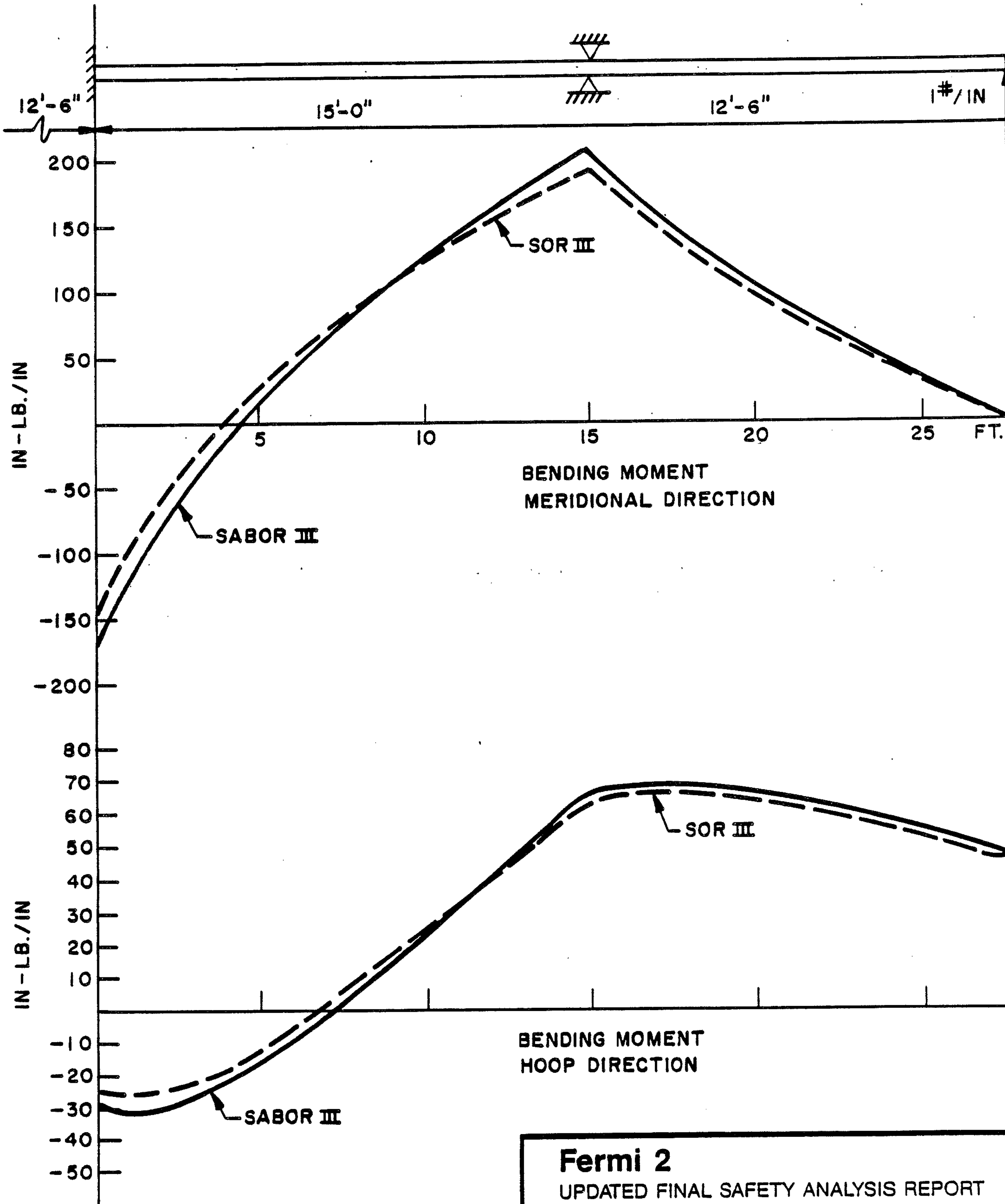

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FIGURE 3.13-50

MOMENT COMPARISON, SABOR III AND SOR III

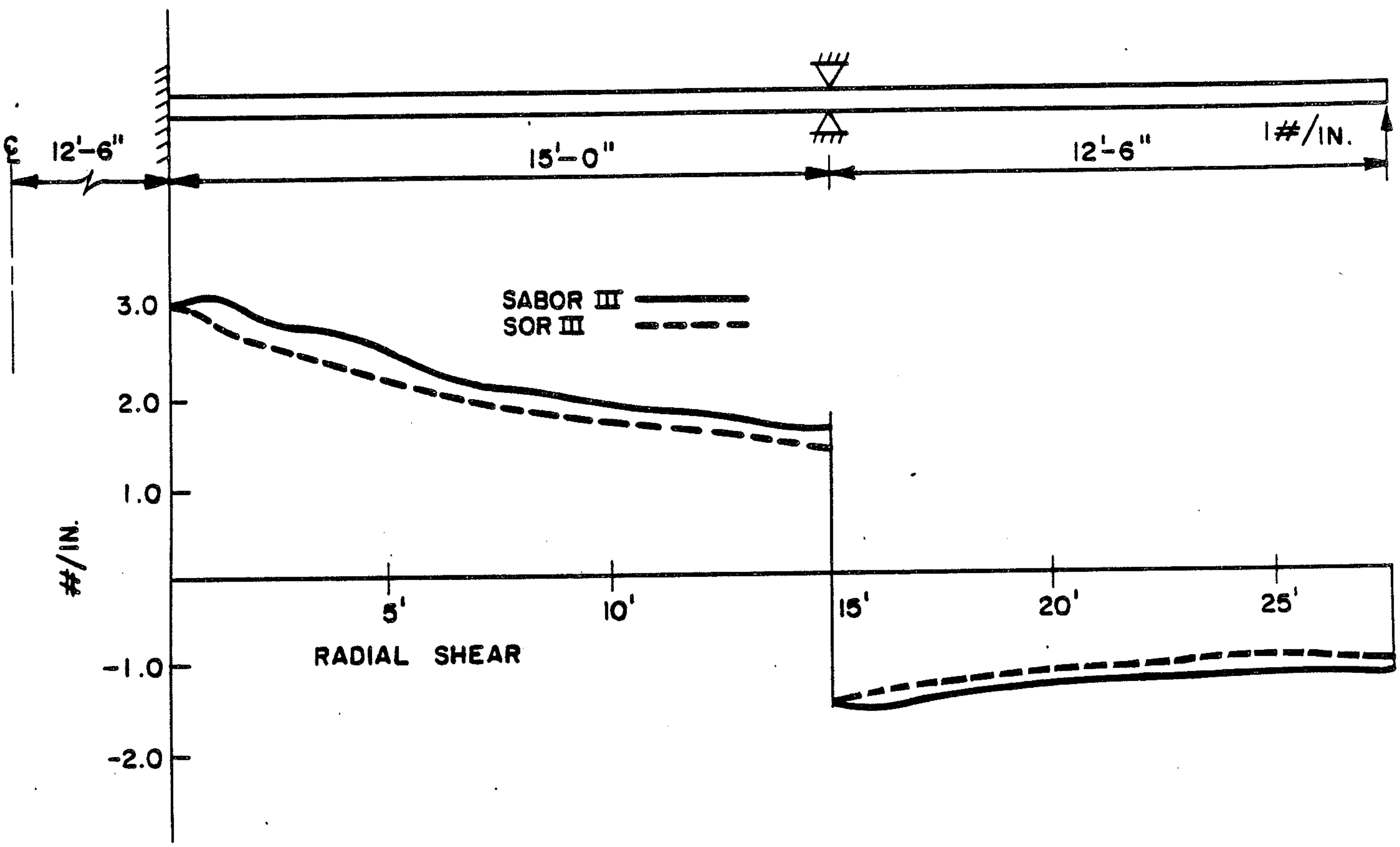

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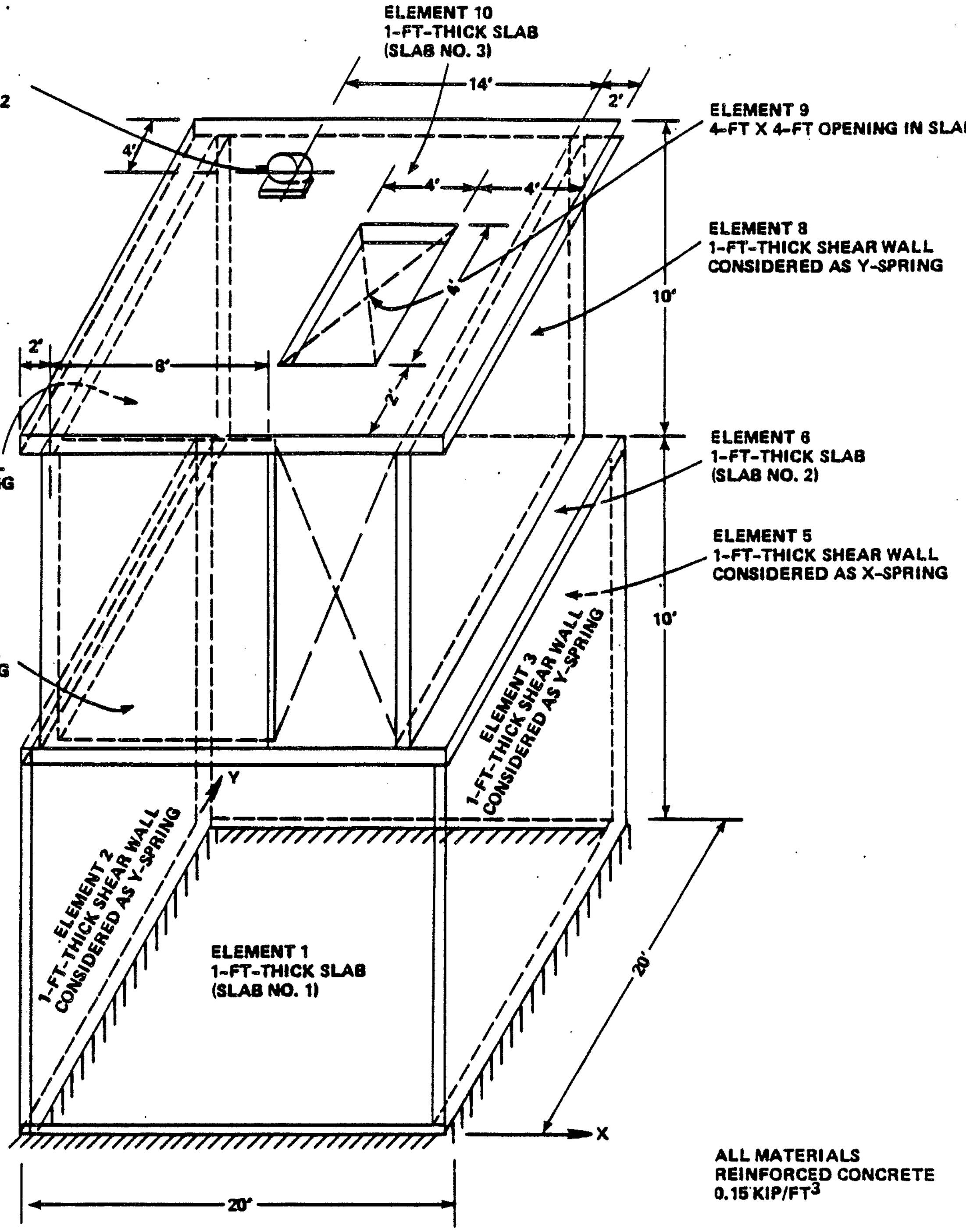
FIGURE 3.13-51

RADIAL SHEAR COMPARISON FOR SABOR III AND SOR III



ALL MATERIALS
REINFORCED CONCRETE
0.15 KIP/FT^3

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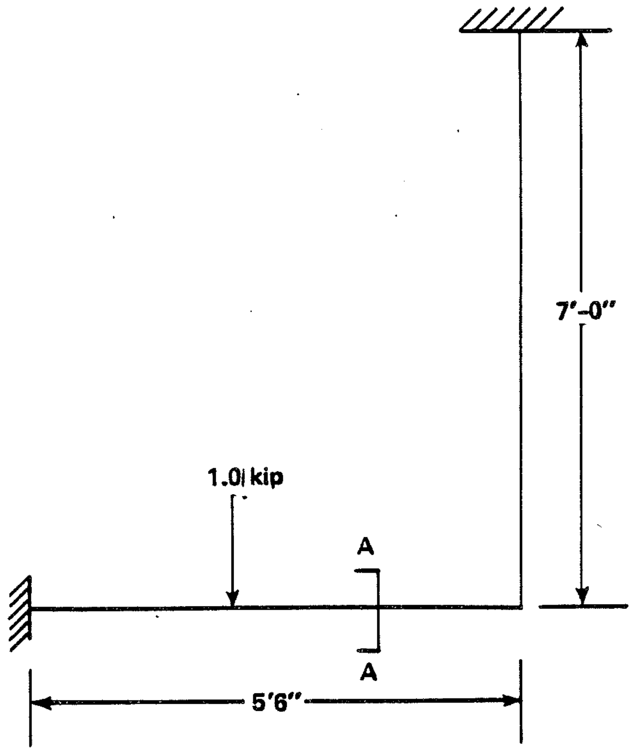
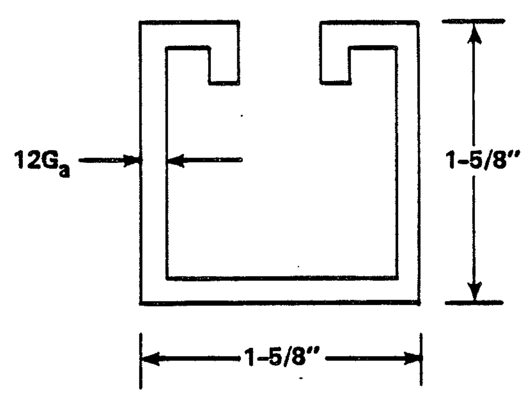
FIGURE 3.13-52

BUILDING COMPLEX





SECTION AA

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FIGURE 3.13-53

TYPICAL CABLE PAN HANGER

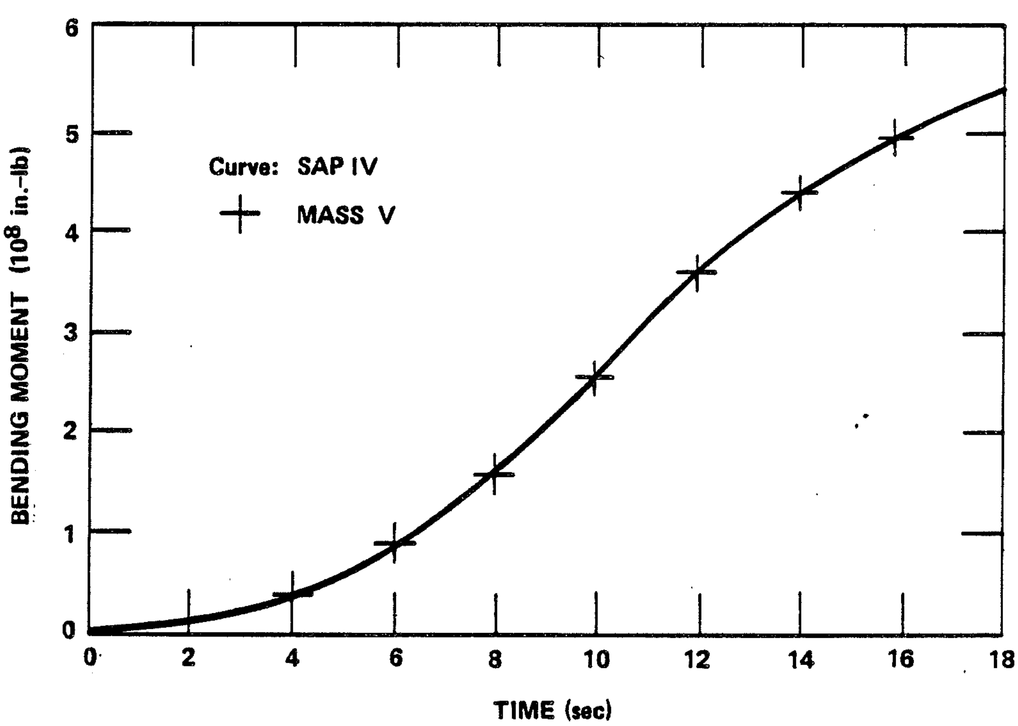

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FIGURE 3.13-54

CANTILEVER RESPONSE
MOMENT AT NODE 1
FIXED END OF CANTILEVER

DESIGN OF TIED COLUMN

B= 17.00 T= 17.00 FC= 3.000 FY=40.000 PHIC= 0.700 PHIB= 0.0900

USE- 10 NO. 9 BARS. AST = 10.00 SQ. IN. = 3.47 PCT. COVER = 1.500 IN.

	ROW 1	ROW 2	ROW 3	ROW 4
NO. OF BARS	2	2	3	3
COVER	1.500	1.500	1.500	1.500

LOAD CASE	APPLIED FORCES AP	AMX	AMY	ULTIMATE CAPACITY UP	UMX	UMY	UP/AP
1	525.	0.	105.	563.	0.	113.	1.072
2	525.	75.	0.	603.	86.	0.	1.148

INTERACTION CONTROL POINTS REQUESTED

	PZ	PB	MB	MZ
X -AXIS	778.0	304.7	166.2	176.2
Y -AXIS	778.0	245.8	234.6	199.7
Z -AXIS	778.0	314.6	167.2	193.7

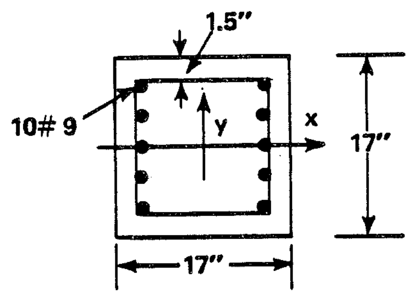

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FIGURE 3.13-55

VALIDATION PROBLEM 1
DESIGN OF A TIED COLUMN
COMPRESSION CONTROL

DESIGN OF
TIED COLUMN

B= 14.00 T= 20.00 FC= 4.500 FY=50.000 PHIC= .700 PHIB= .900

USE- 6 NO.11 BARS. AST = 9.36 SQ.IN. = 3.35 PCT. COVER= 1.500 IN.

	ROW 1	ROW 2	ROW 3	ROW 4
NO. OF BARS	3	3	0	0
COVER	1.500	1.500	1.500	1.500

LOAD CASE	APPLIED FORCES AP	AMX	AMY	ULTIMATE CAPACITY UP	UMX	UMY	UP/AP
1	115.	279.	0.	122.	295.	0.	1.057
2	115.	0.	14.	801.	0.	94.	6.966

INTERACTION CONTROL POINTS REQUESTED

	PZ	PB	MB	MZ
X -AXIS	1052.2	317.9	353.8	282.8
Y -AXIS	1052.2	315.4	187.2	180.3
Z -AXIS	1052.2	310.9	231.3	254.0

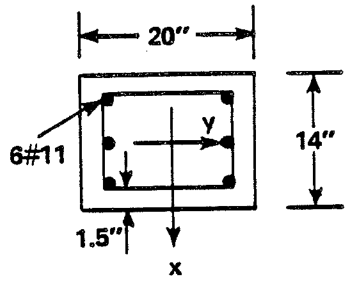

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FIGURE 3.13-56

VALIDATION PROBLEM 2
DESIGN OF A TIED COLUMN
TENSION CONTROLS

DESIGN OF
TIED COLUMN

B= 28.00 T= 28.00 FC= 5.000 FY=60.000 PHIC= .700 PHIB= .900

USE- 12 NO.11 BARS. AST = 18.72 SQ.IN. = 2.39 PCT. COVER = 1.500 IN.

	ROW 1	ROW 2	ROW 3	ROW 4
NO. OF BARS	4	4	2	2
COVER	1.500	1.500	1.500	1.500

LOAD CASE	APPLIED FORCES AP	AMX	AMY	ULTIMATE CAPACITY UP	UMX	UMY	UP/AP
1	1330.	790.	0.	1626.	966.	0.	1.223
2	1330.	0.	394.	2216.	0.	655.	1.666
3	1330.	790.	394.	1388.	824.	411.	1.044

INTERACTION CONTROL POINTS REQUESTED

	PZ	PB	MB	MZ
X -AXIS	3062.9	983.0	1167.4	999.1
Y -AXIS	3062.9	983.0	1167.4	999.1
Z -AXIS	3062.9	910.2	949.7	947.4

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FIGURE 3.13-57

VALIDATION PROBLEM 3
DESIGN OF A TIED COLUMN
BIAXIAL BENDING

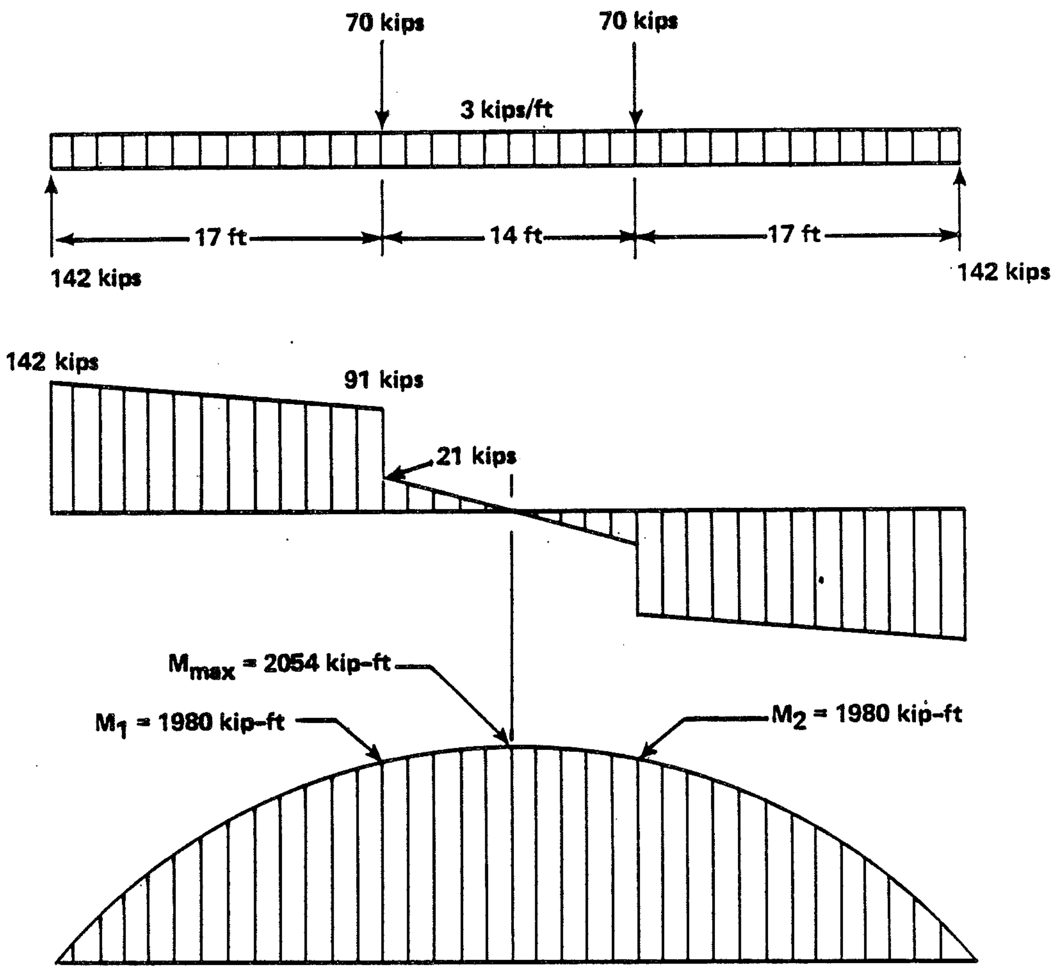

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FIGURE 3.13-58

SHEAR AND MOMENT DIAGRAMS

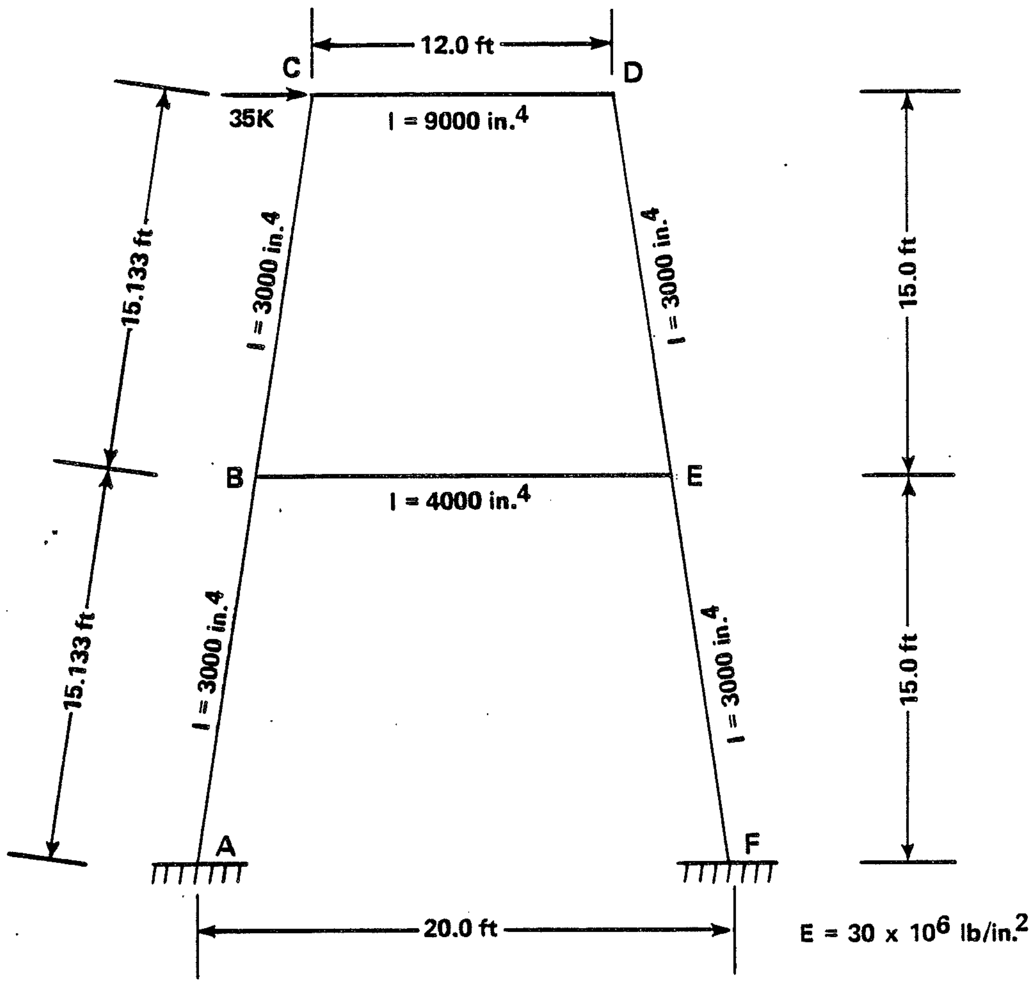

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FIGURE 3.13-59

EXAMPLE FOR STATIC ANALYSIS

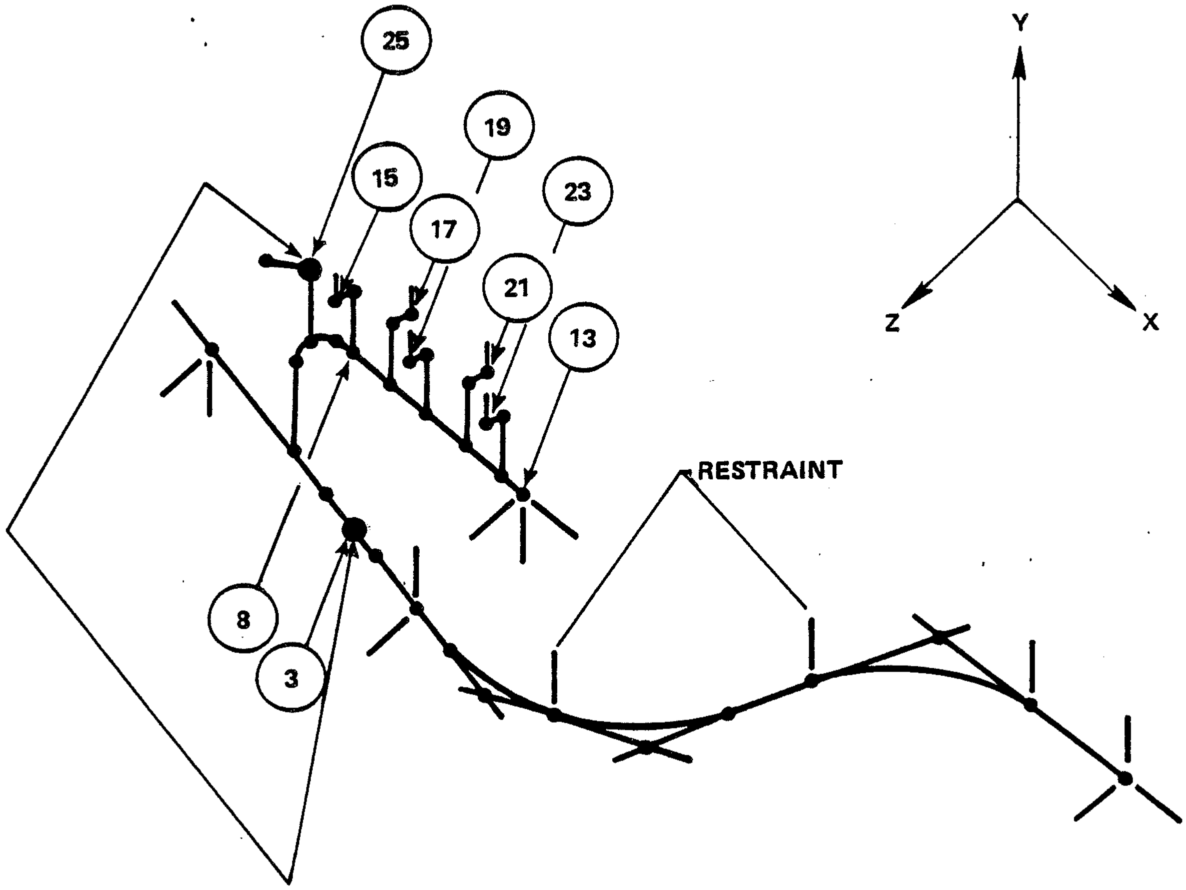

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FIGURE 3.13-60

STRUCTURAL MODEL OF PIPING SYSTEM

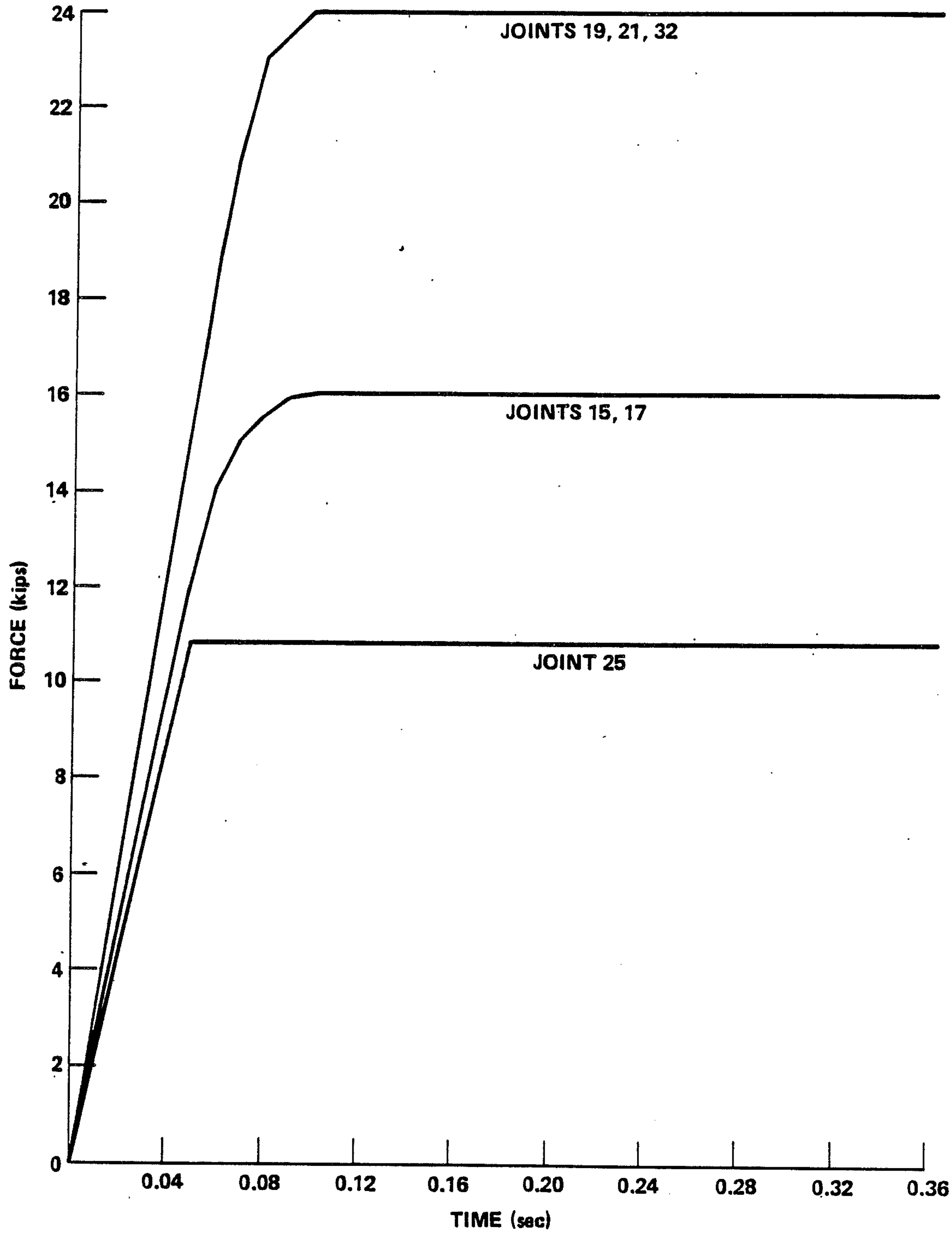

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FIGURE 3.13-61

LOAD TIME HISTORY

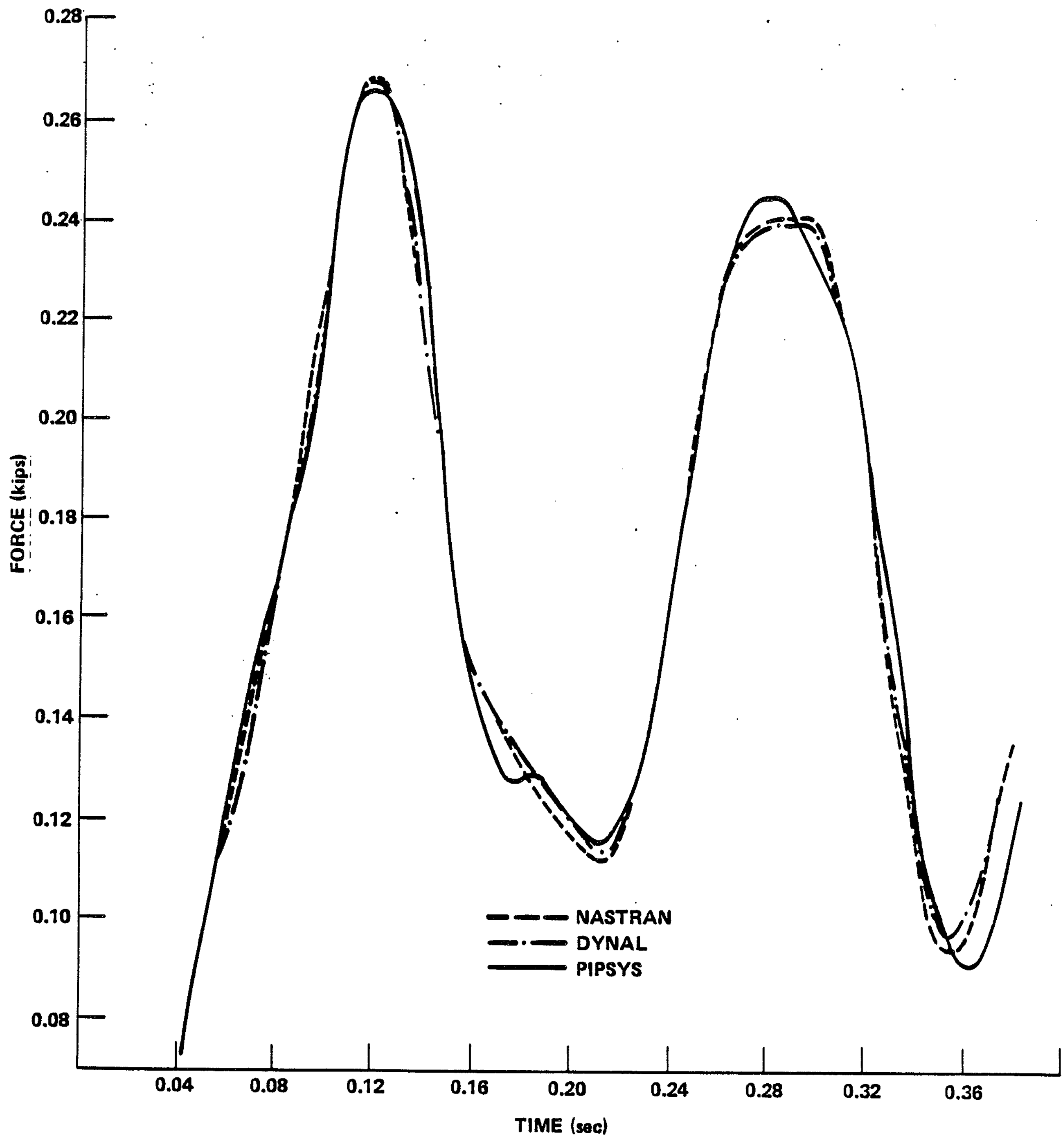

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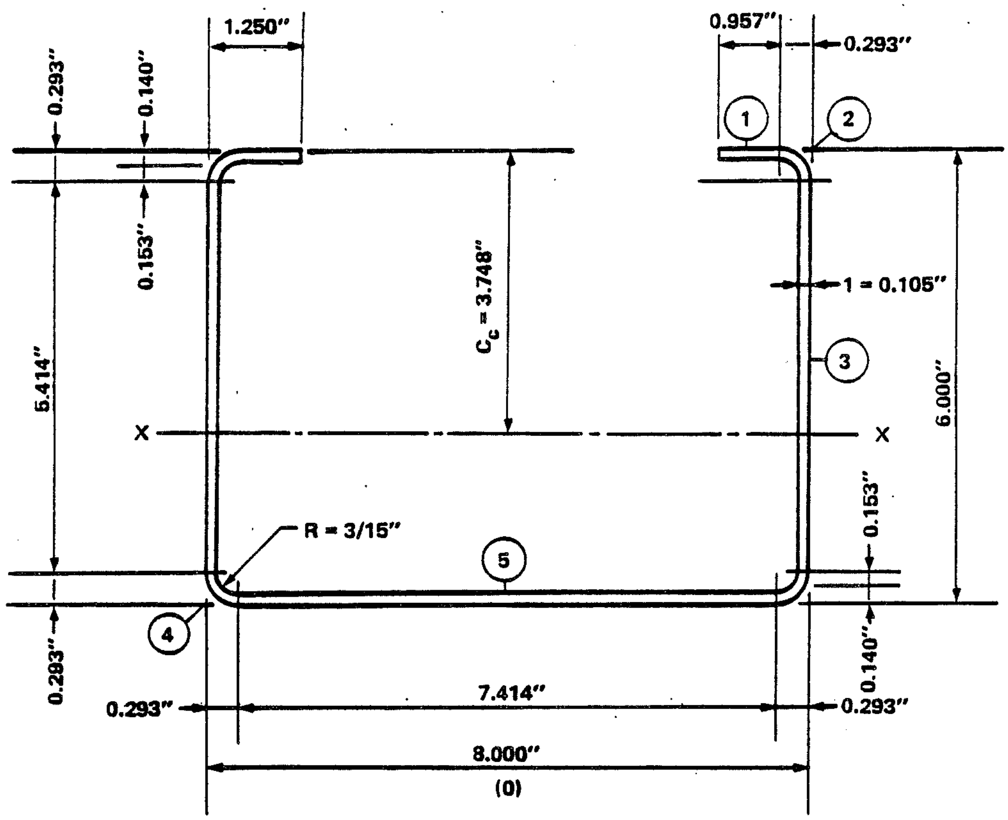
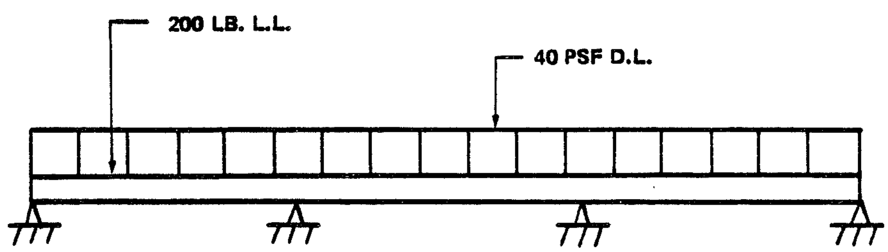
FIGURE 3.13-62

DISPLACEMENT VERSUS TIME
JOINT 8, Z DIRECTION





VERTICAL SEISMIC DESIGN LOAD = 1.5g
HORIZONTAL SEISMIC DESIGN LOAD = 4.5g

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FIGURE 3.13-63

CABLE TRAY MODEL FOR "SEISHANG" PROGRAM

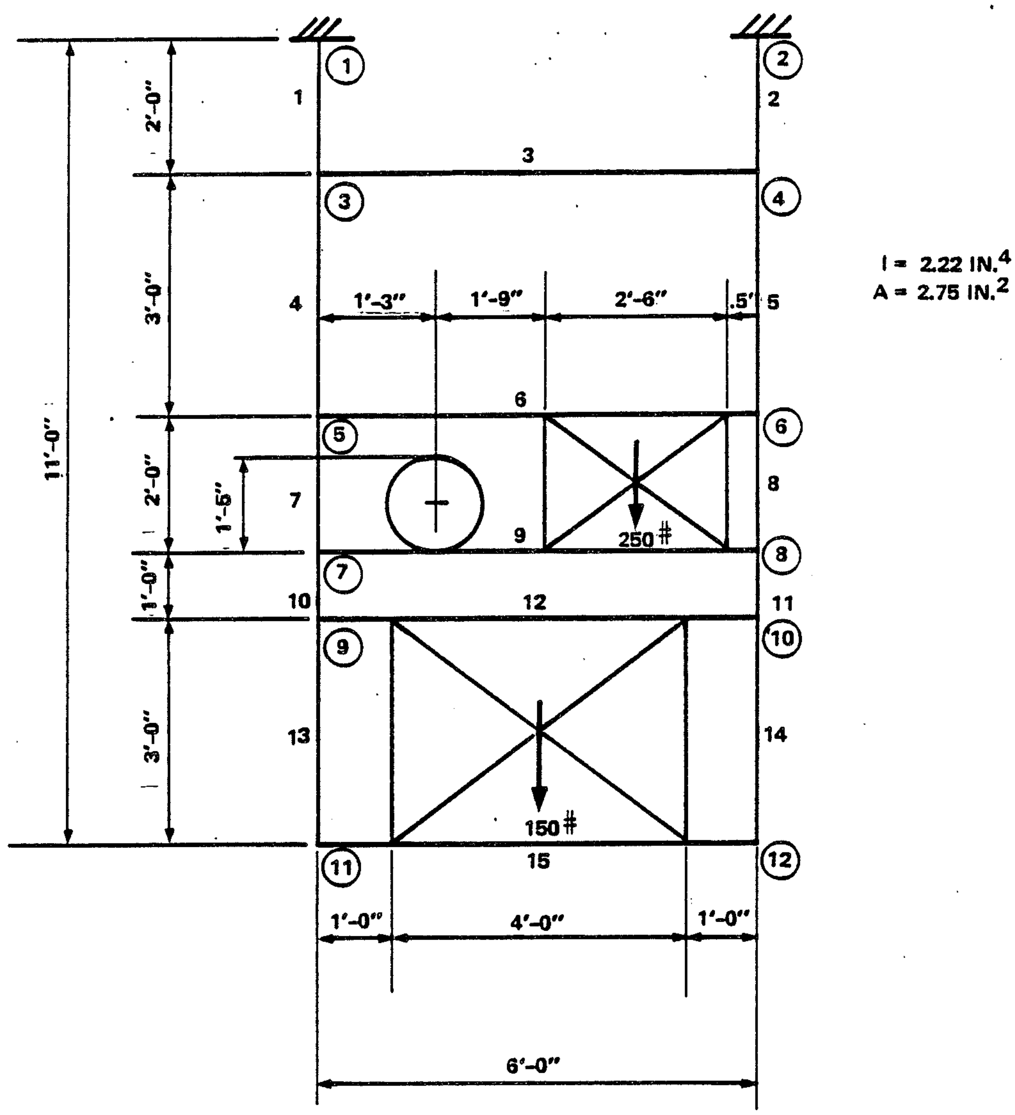

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FIGURE 3.13-64

CEILING MOUNTED SUPPORT MODEL FOR "SEISHANG" PROGRAM

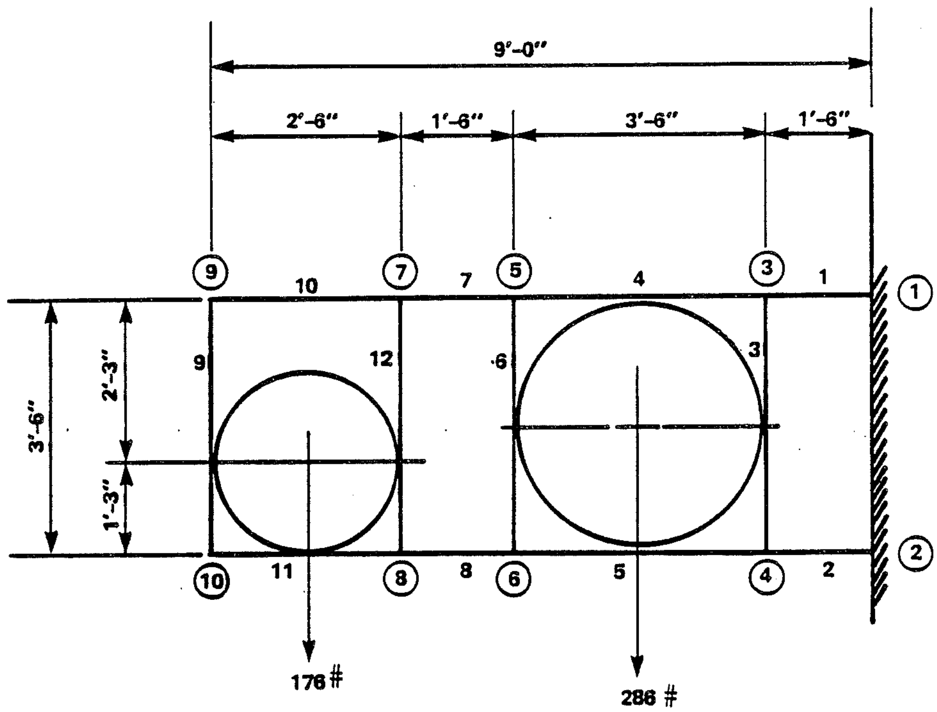

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FIGURE 3.13-65

WALL MOUNTED SUPPORT MODEL FOR "SEISHANG" PROGRAM

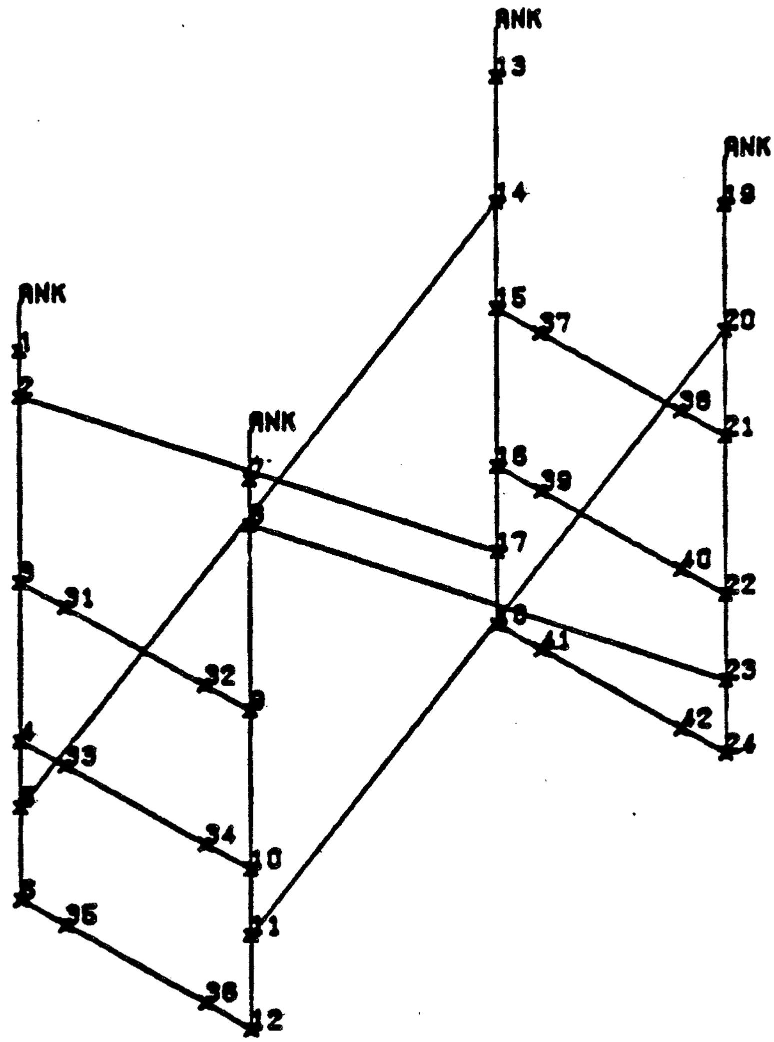

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FIGURE 3.13-66

CABLE TRAY SUPPORT

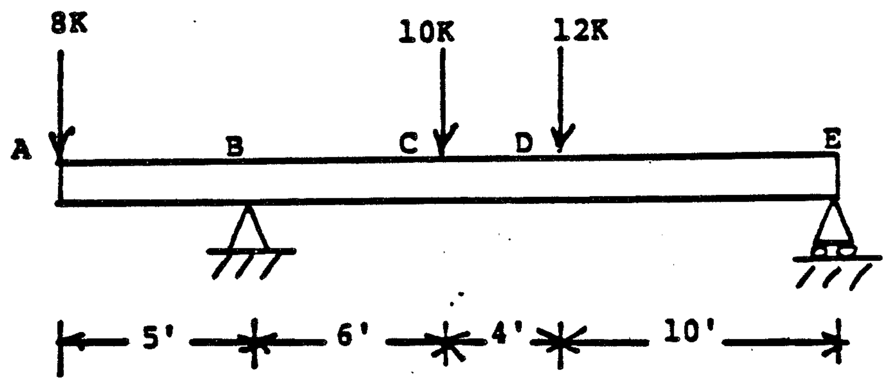

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FIGURE 3.13-67

CONTINUOUS BEAM PROBLEM
"PFRAME" VALIDATION

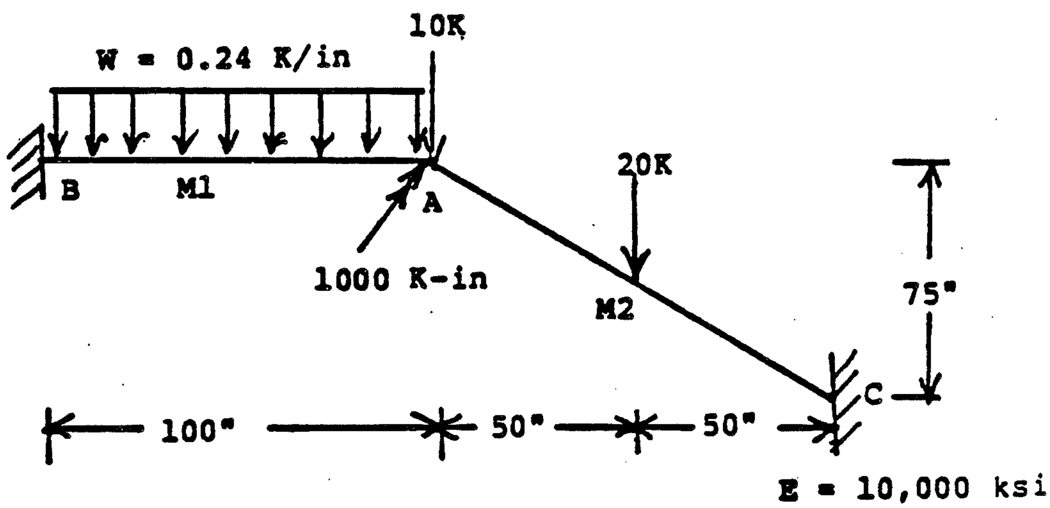

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FIGURE 3.13-68

PLANE FRAME PROBLEM
"PFRAME" VALIDATION

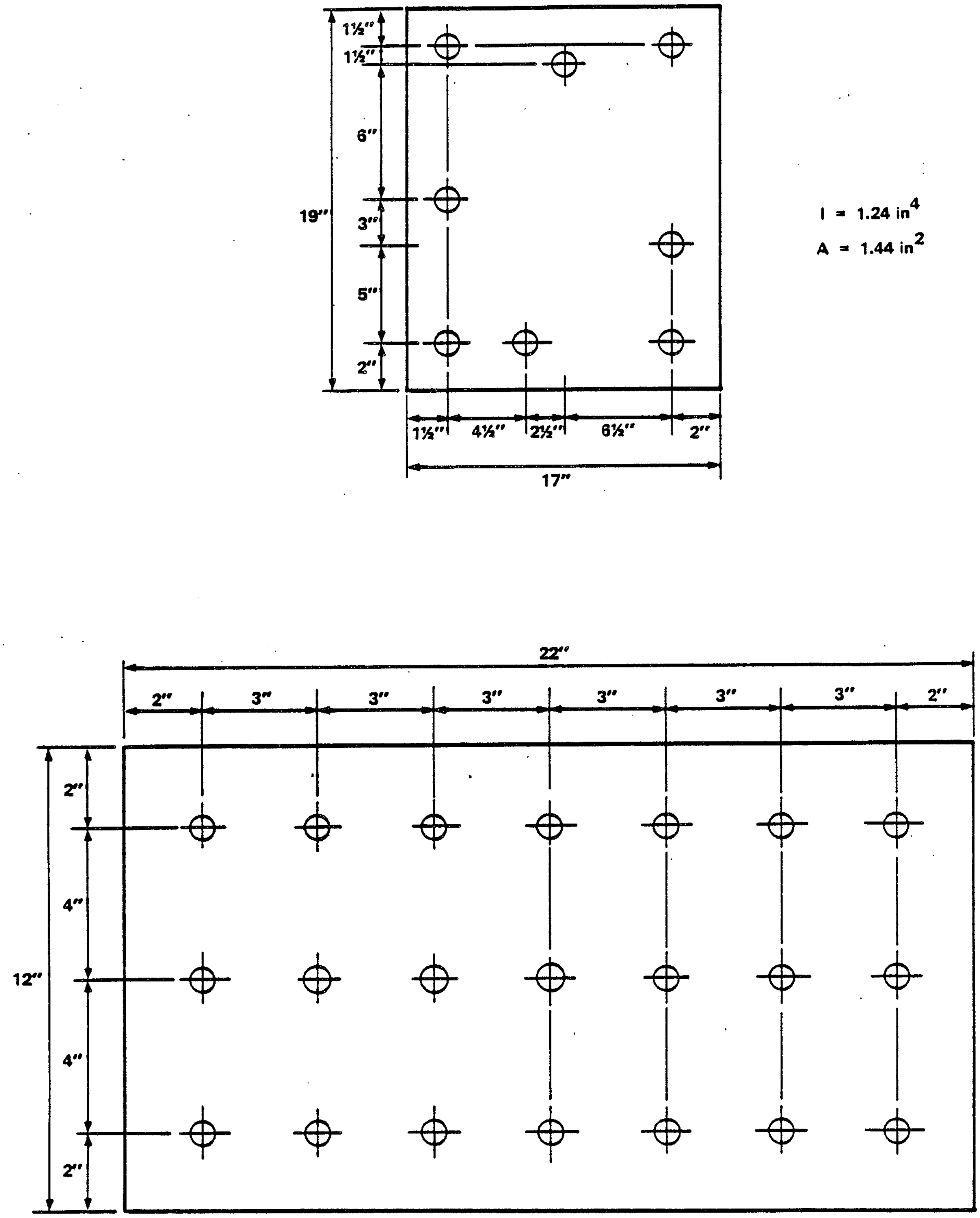

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FIGURE 3.13-69

BASE PLATE ANALYSIS

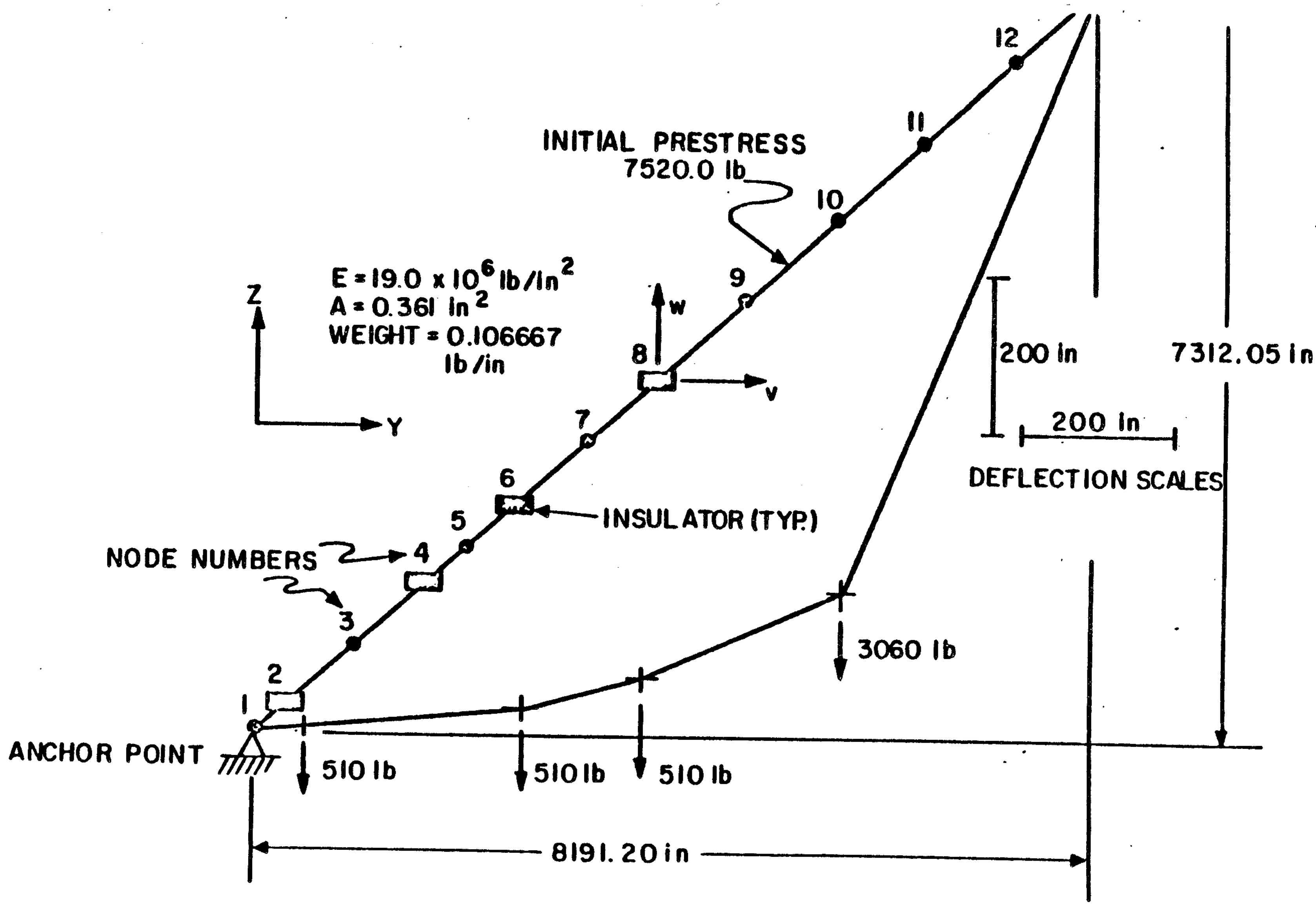

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FIGURE 3.13-70

STATIC CONFIGURATION OF TOWER CABLE "ADINA" – VALIDATION PROBLEM 1

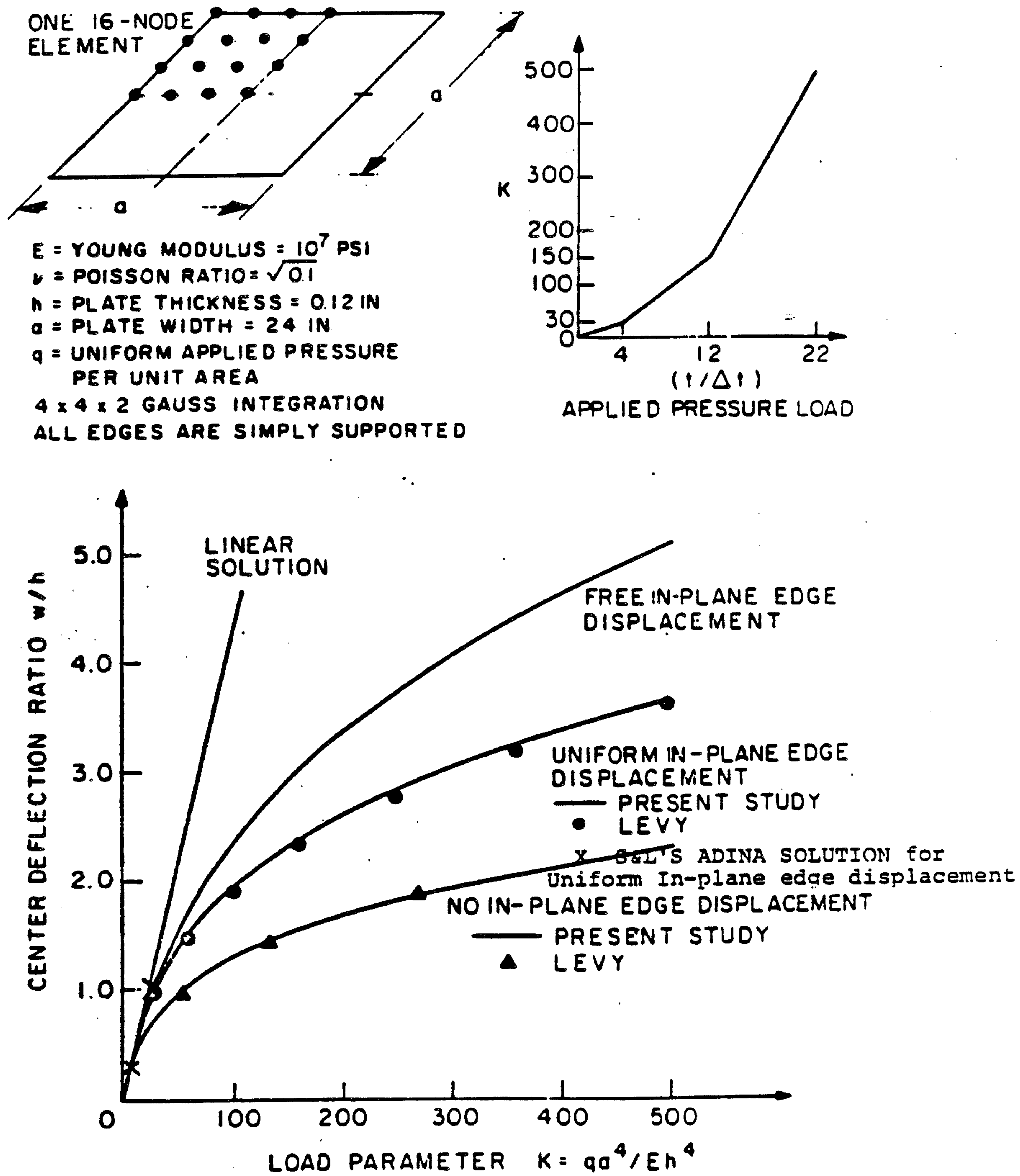

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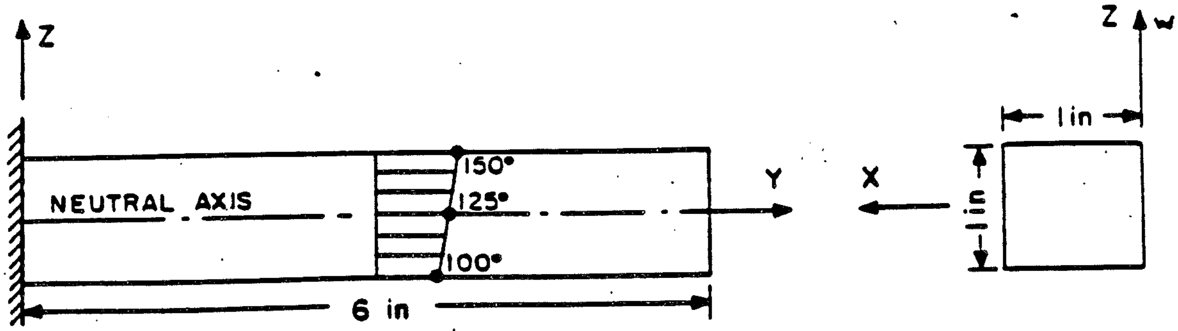
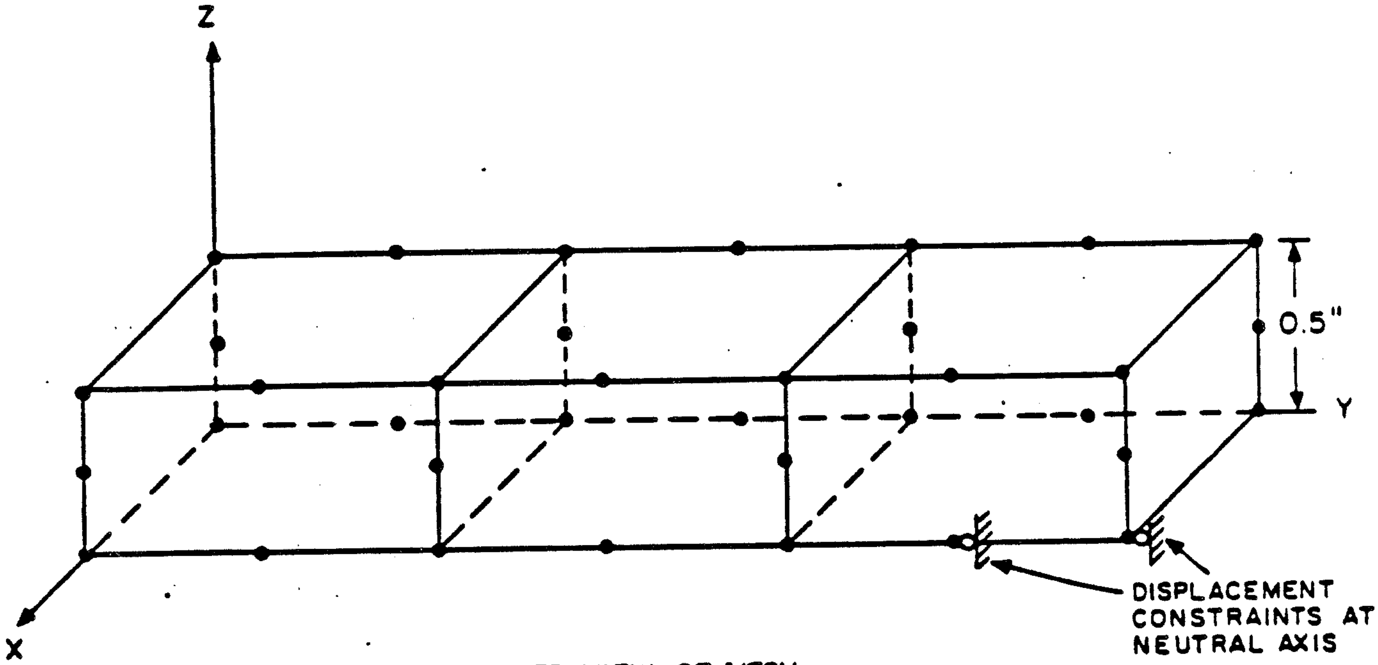
FIGURE 3.13-71

LARGE DEFLECTION ANALYSIS OF A SIMPLY SUPPORTED SQUARE PLATE SUBJECTED TO PRESSURE LOADING
"ADINA" – VALIDATION PROBLEM 2



SIDE AND END VIEWS



ENLARGED VIEW OF MESH

	70°F	200°F	
E	$30. \times 10^6$	$28. \times 10^6$	lb/in^2
ν	0.0	0.0	
α	5.5×10^{-6}	6.0×10^{-6}	in/in/°F

REFERENCE TEMPERATURE = 125° F

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FIGURE 3.13-72

FINITE ELEMENT MESH OF CANTILEVER BEAM "ADINA" – VALIDATION PROBLEM 3

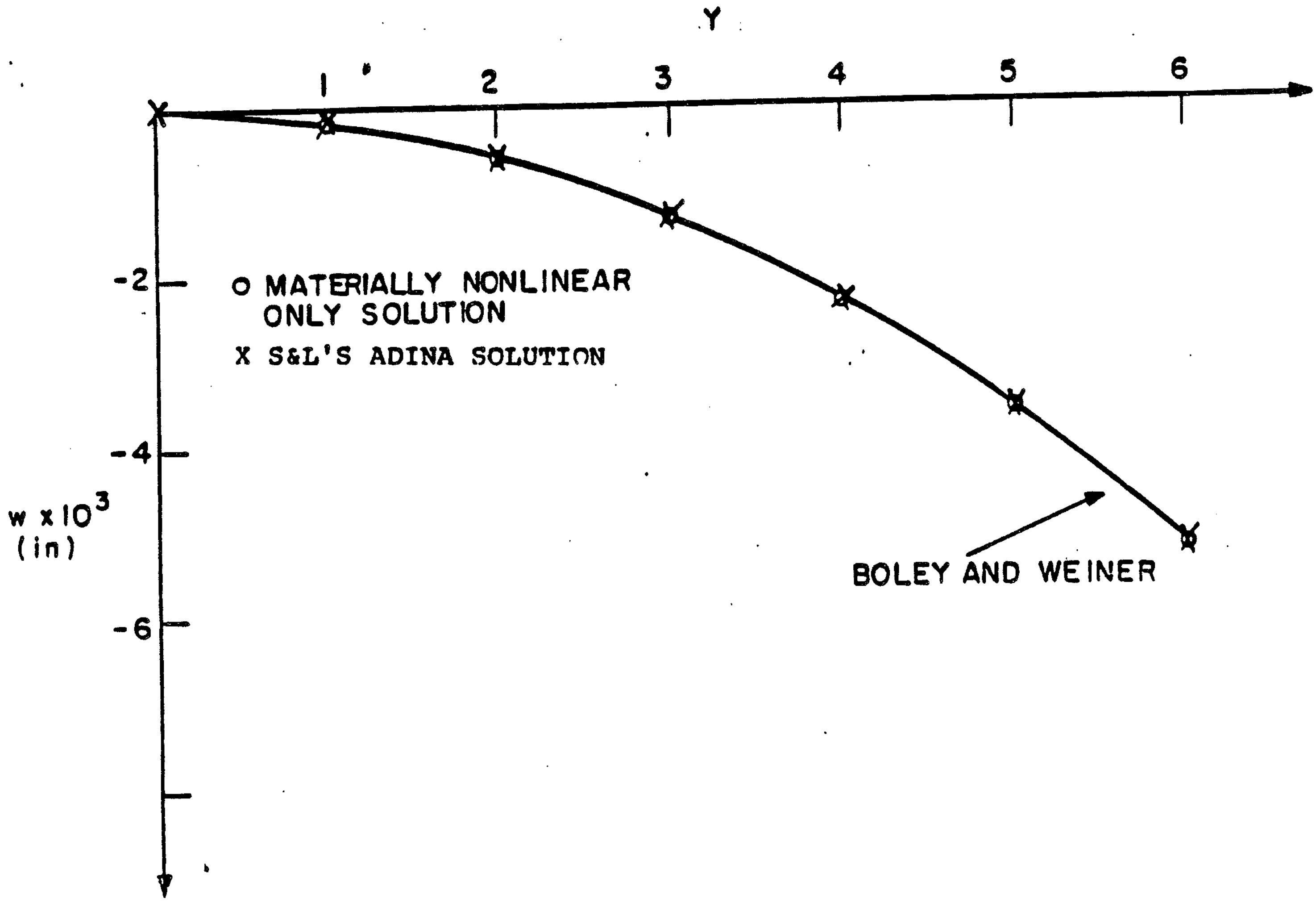

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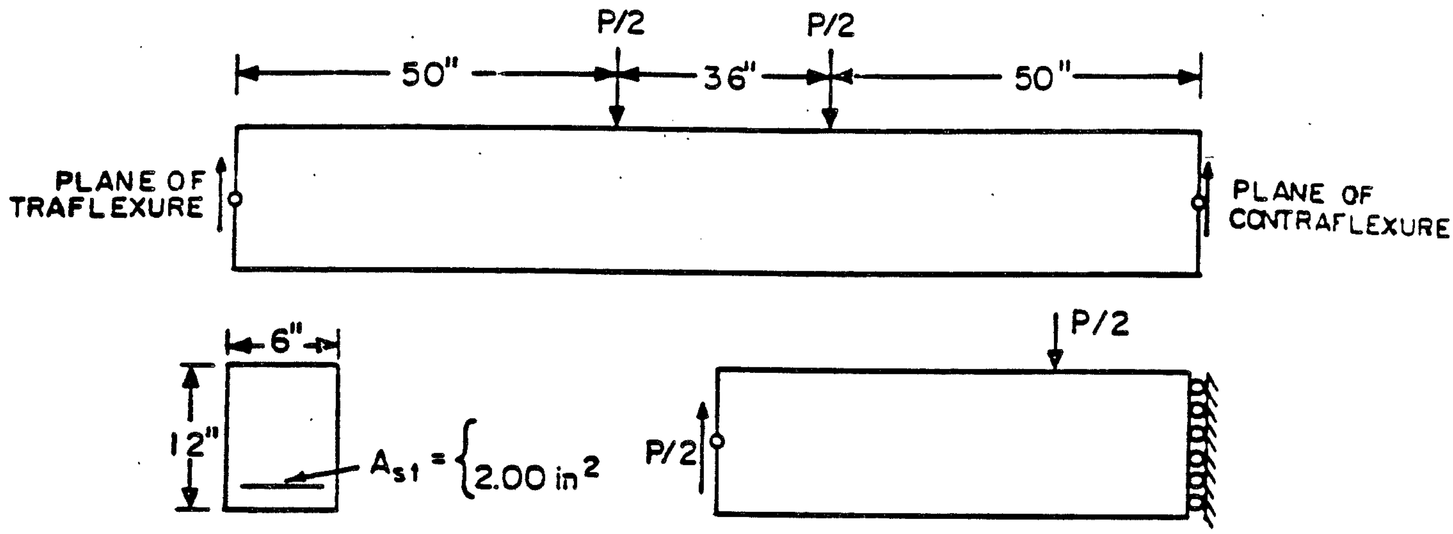
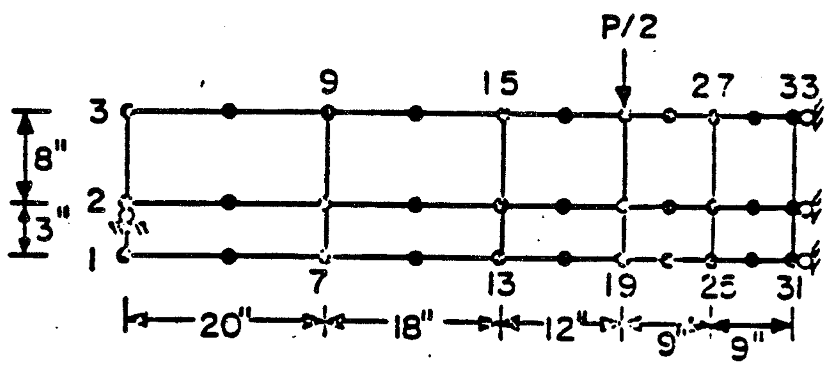
FIGURE 3.13-73

ELASTIC DISPLACEMENT RESPONSE OF CANTILEVER "ADINA" – VALIDATION PROBLEM 3



BEAM DIMENSIONS



FINITE ELEMENT IDEALIZATION

MATERIAL PROPERTIES:

$\tilde{\sigma}_c$ = -3740 psi

$\tilde{\sigma}_t$ = 458 psi

$\sigma_{y\,steel}$ = 44000 psi

$\tilde{E}_{o_{concrete}}$ = 6100 ksi

ν = 0.2

E_{steel} = 30000 ksi

$E_{t_{steel}}$ = 300 ksi

$\tilde{\sigma}_u$ = -3225 psi

$\tilde{e}_u$ = -.003 in/in

$\tilde{e}_c$ = -.002 in/in

$\rho_{concrete}$ = 0.2172×10^{-3} slugs/in^3

ρ_{steel} = 0.7339×10^{-3} slugs/in^3

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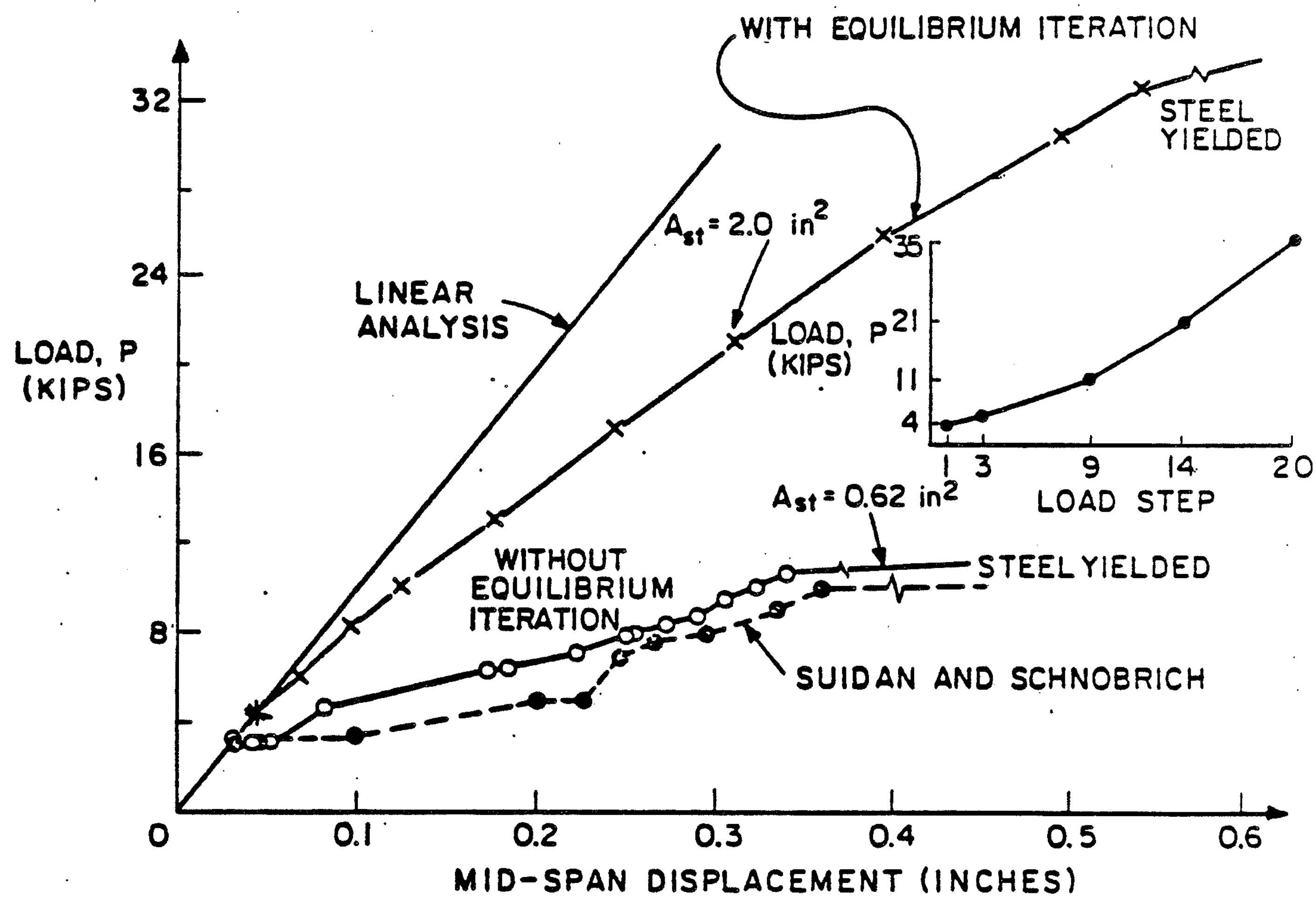
FIGURE 3.13-74

REINFORCED CONCRETE BEAM
"ADINA" – VALIDATION PROBLEM 4



*S&L'S Version for Nonlinear Static Response
Ast = 2.0

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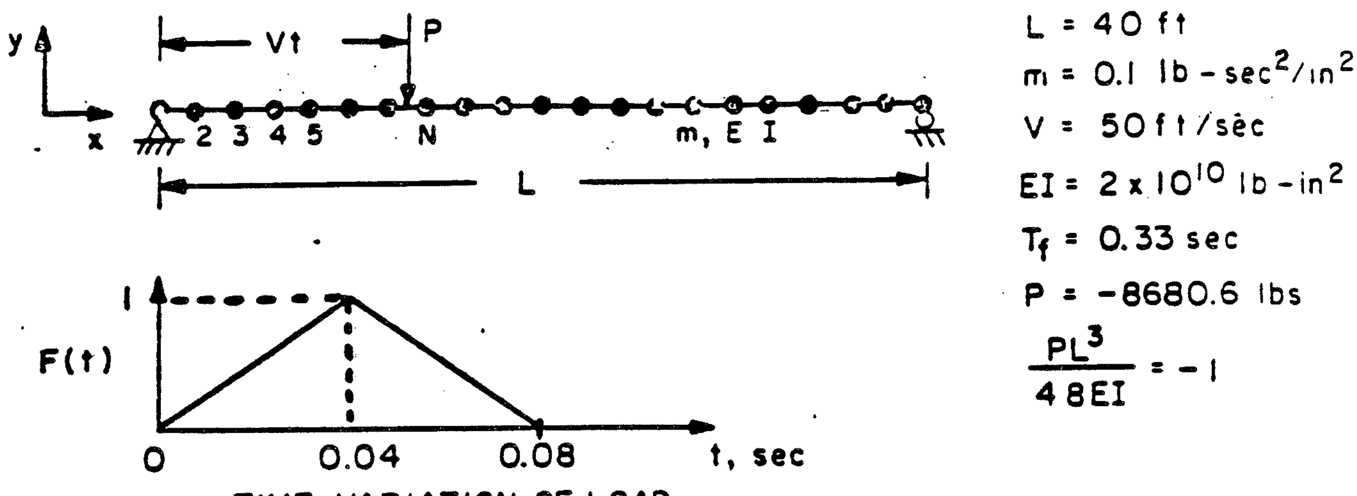
FIGURE 3.13-75

LOAD DISPLACEMENT CURVE FOR THE CONCRETE BEAM – "ADINA" – VALIDATION PROBLEM 4



TIME VARIATION OF LOAD

NOTE: ARRIVAL TIME OF LOAD AT N^{th} NODE = (N-2) x.04 SEC.
THE MAGNITUDE AND DURATION OF ACTION OF THE LOAD AT A NODE IS OBTAINED FROM THE ABOVE TIME FUNCTION, SHIFTED BY THE CORRESPONDING ARRIVAL TIME.

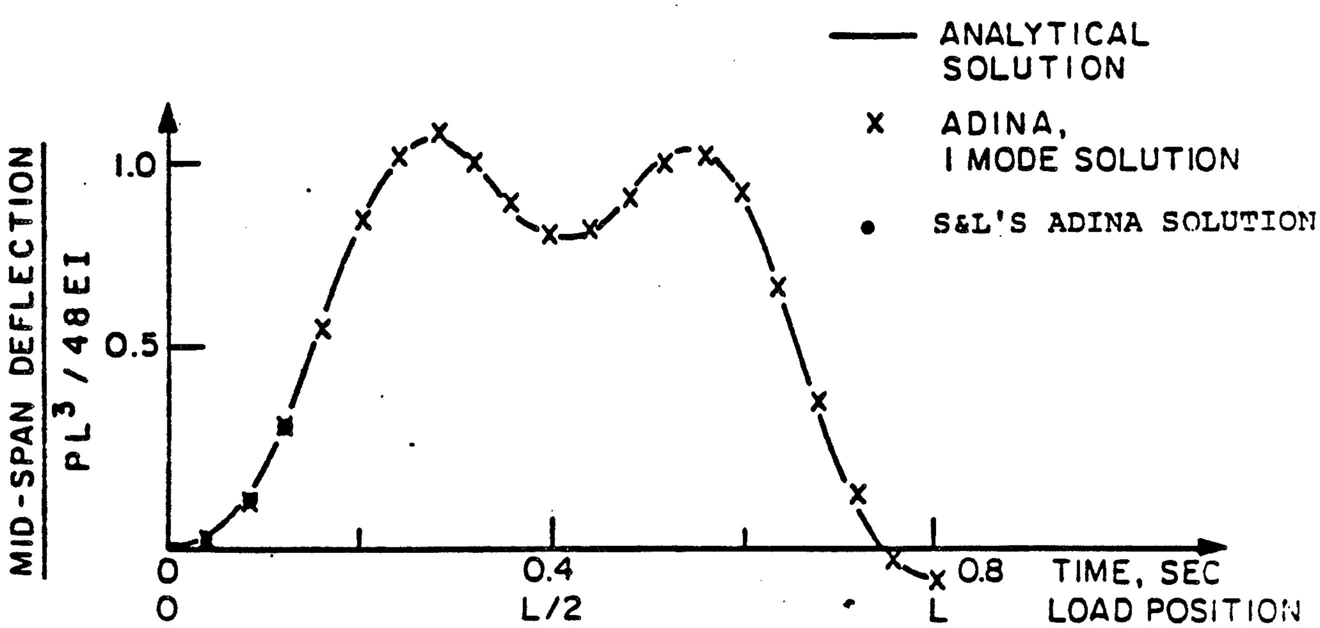


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FIGURE 3.13-76

ANALYSIS OF A BEAM SUBJECTED TO TRAVELLING LOAD "ADINA" – VALIDATION PROBLEM 5

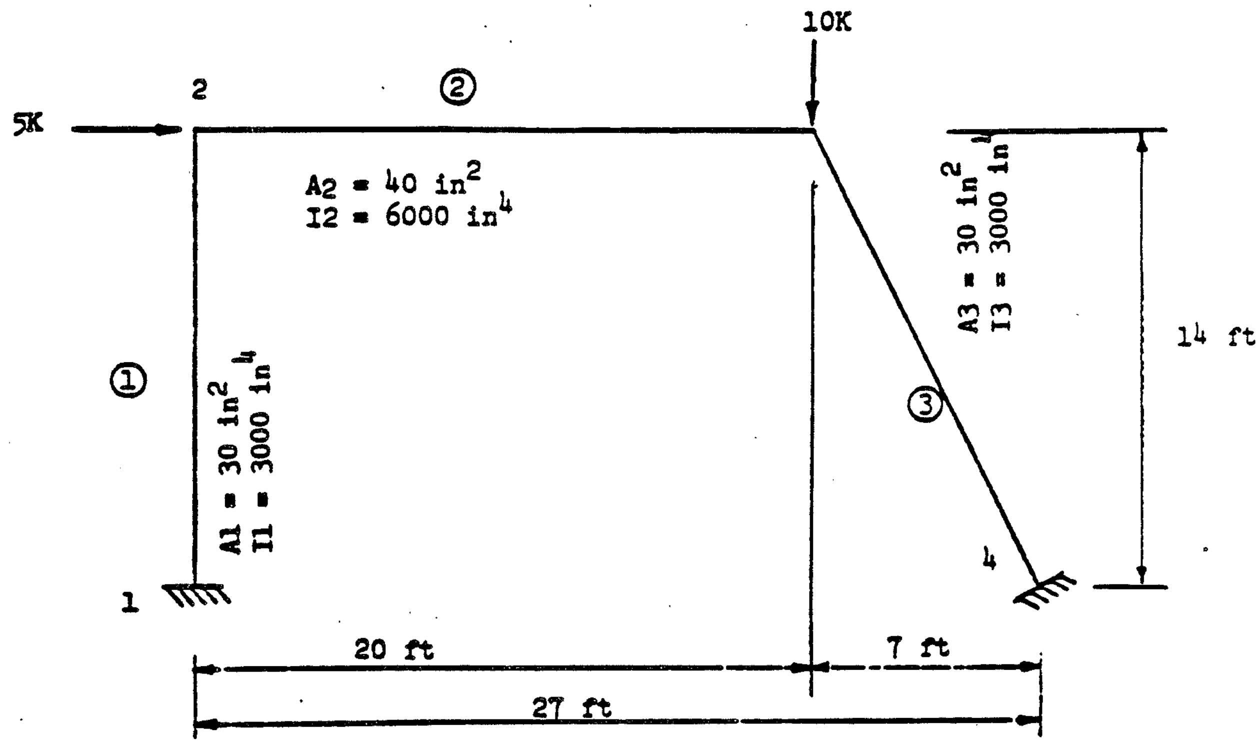

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FIGURE 3.13-77

FRAME USED TO VALIDATE STATIC ANALYSIS OF "FRAME"

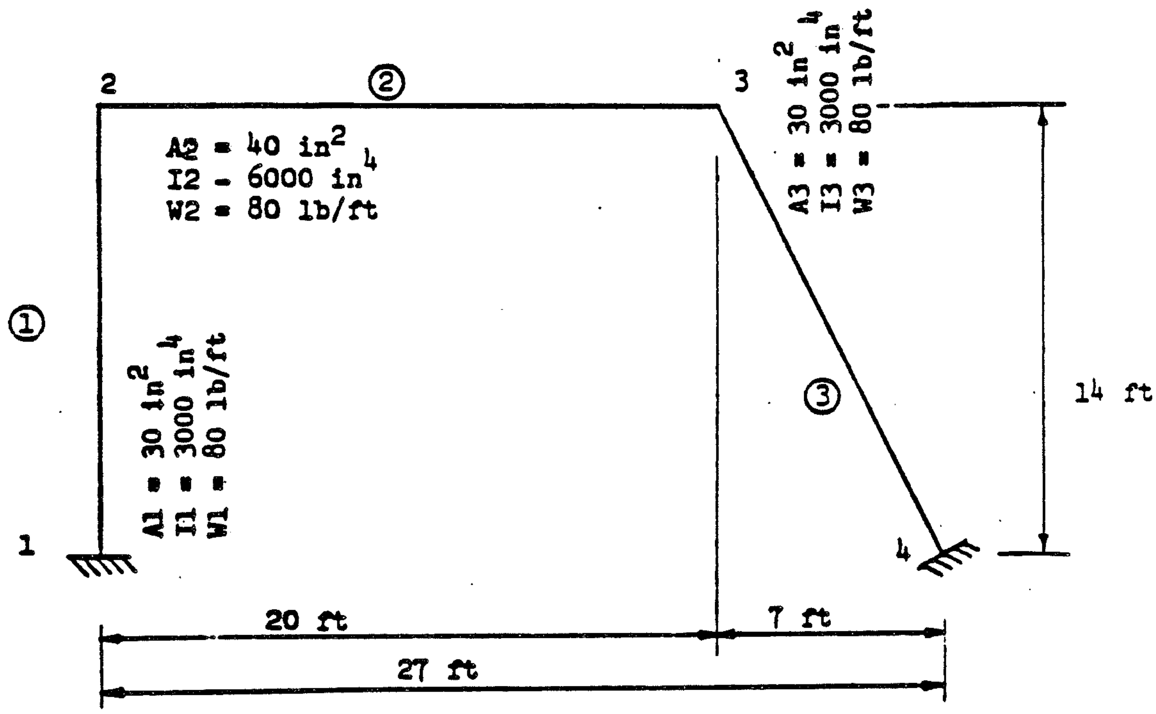

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FIGURE 3.13-78

FRAME USED TO VALIDATE DYNAMIC ANALYSIS OF "FRAME"

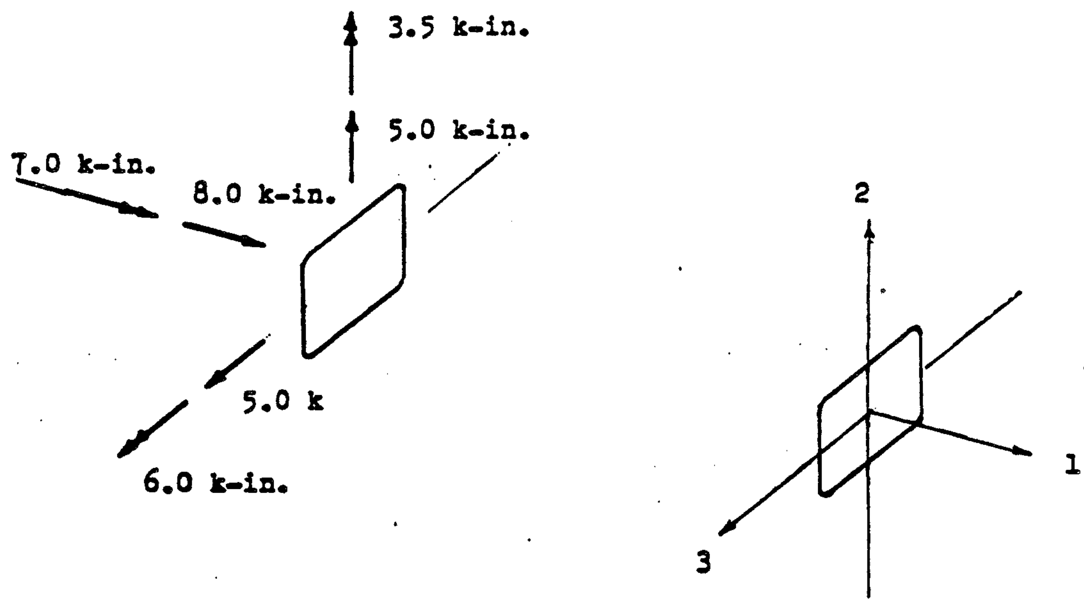

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FIGURE 3.13-79

LOADING FOR "FRAME" STRESS CHECK VALIDATION PROBLEM 1

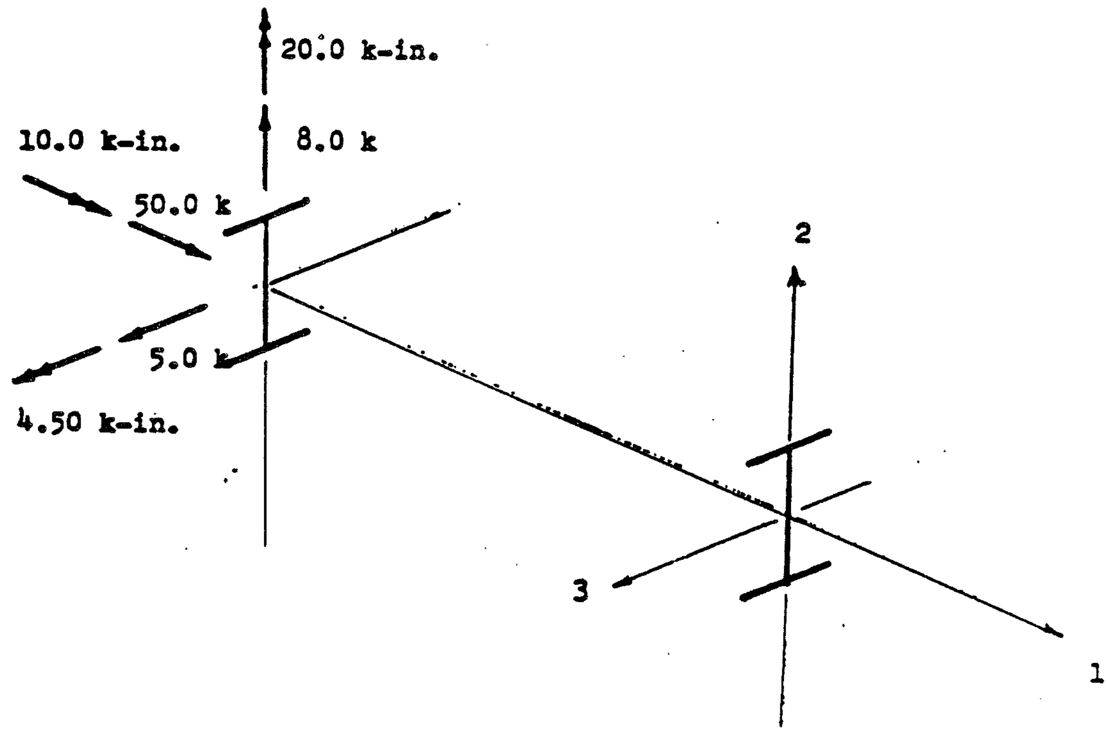

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FIGURE 3.13-80

LOADING FOR "FRAME" STRESS CHECK VALIDATION PROBLEM 2

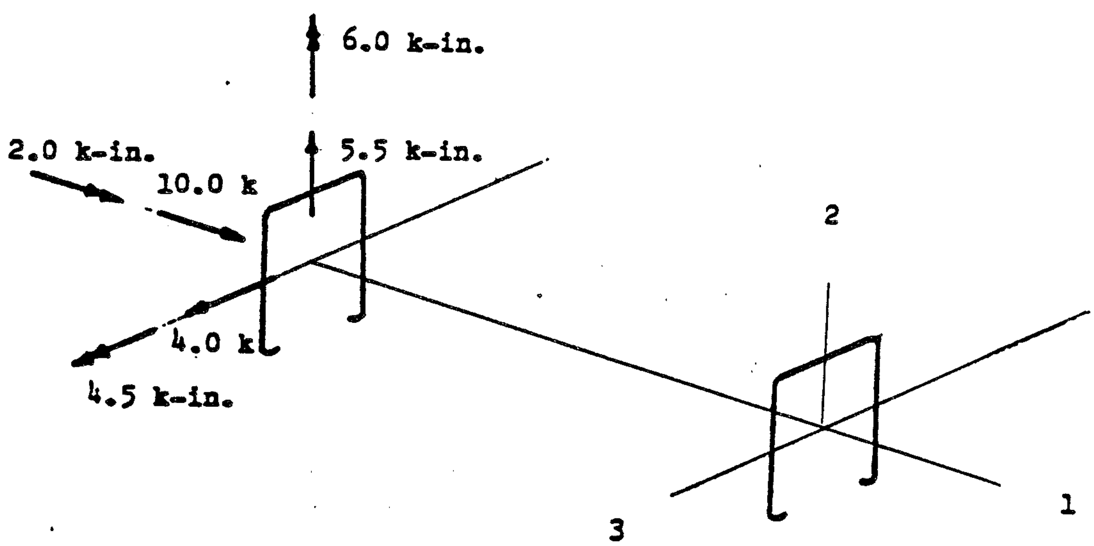

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FIGURE 3.13-81

LOADING FOR "FRAME" STRESS CHECK VALIDATION PROBLEM 3

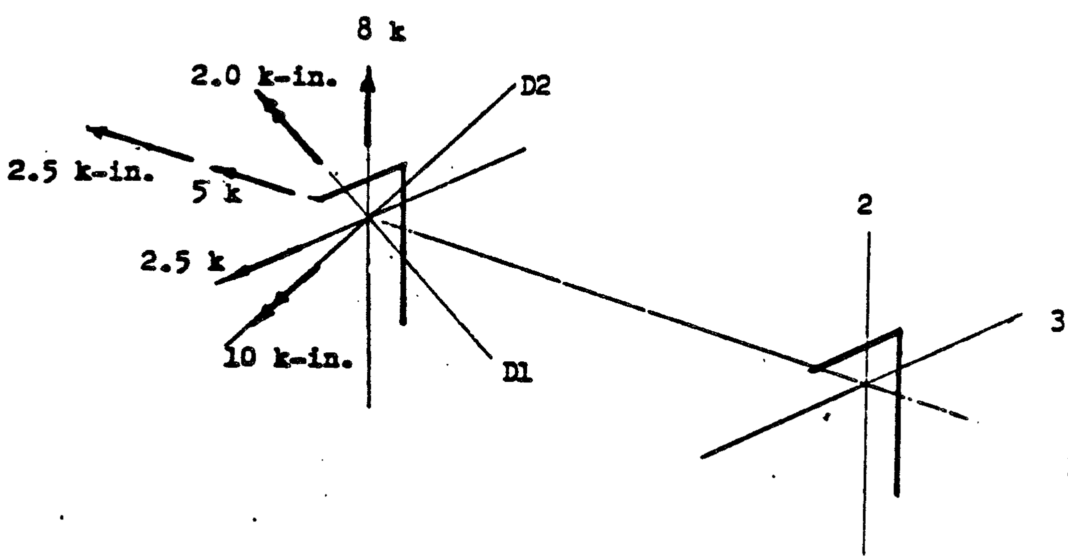

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FIGURE 3.13-82

LOADING FOR "FRAME" STRESS CHECK VALIDATION PROBLEM 4

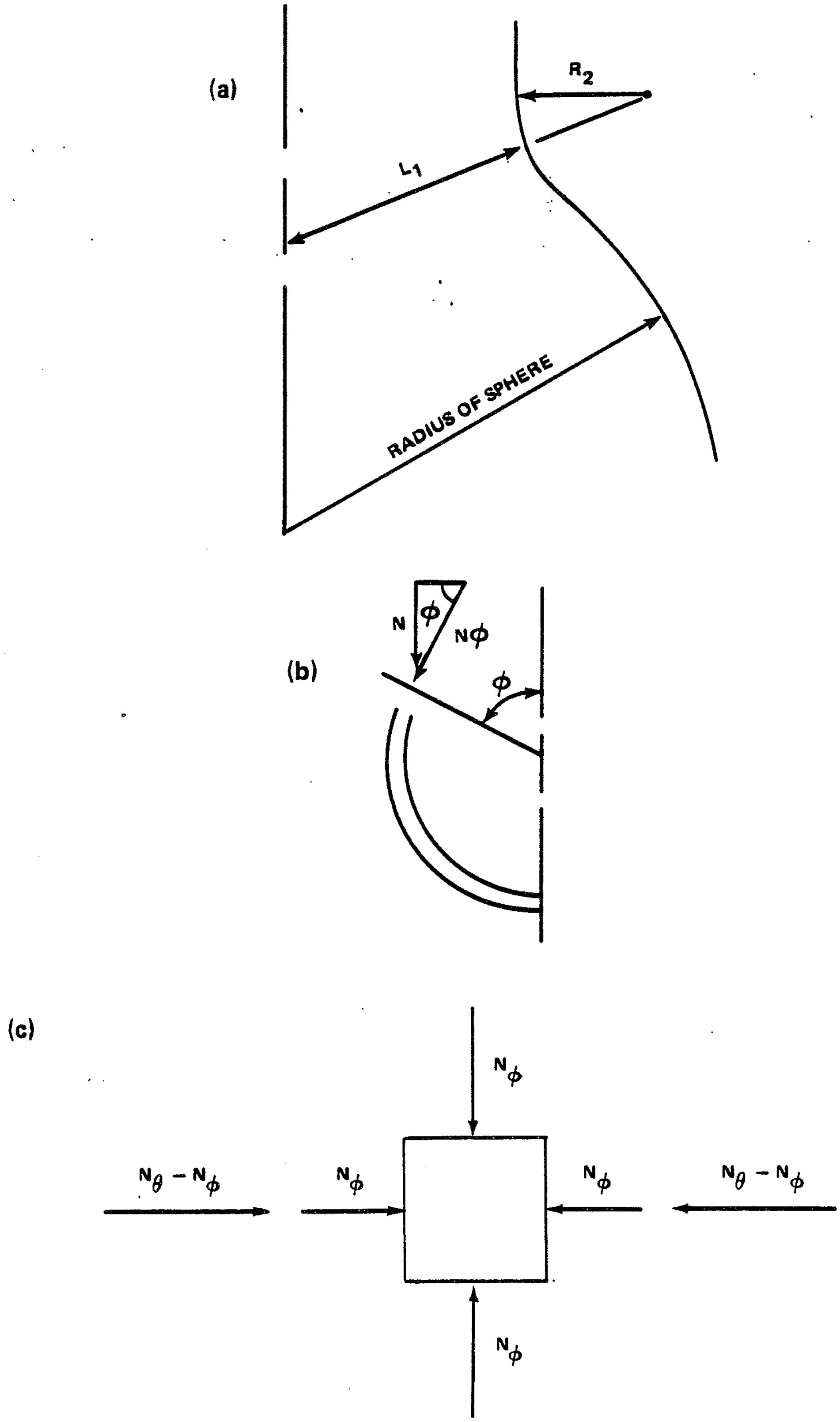

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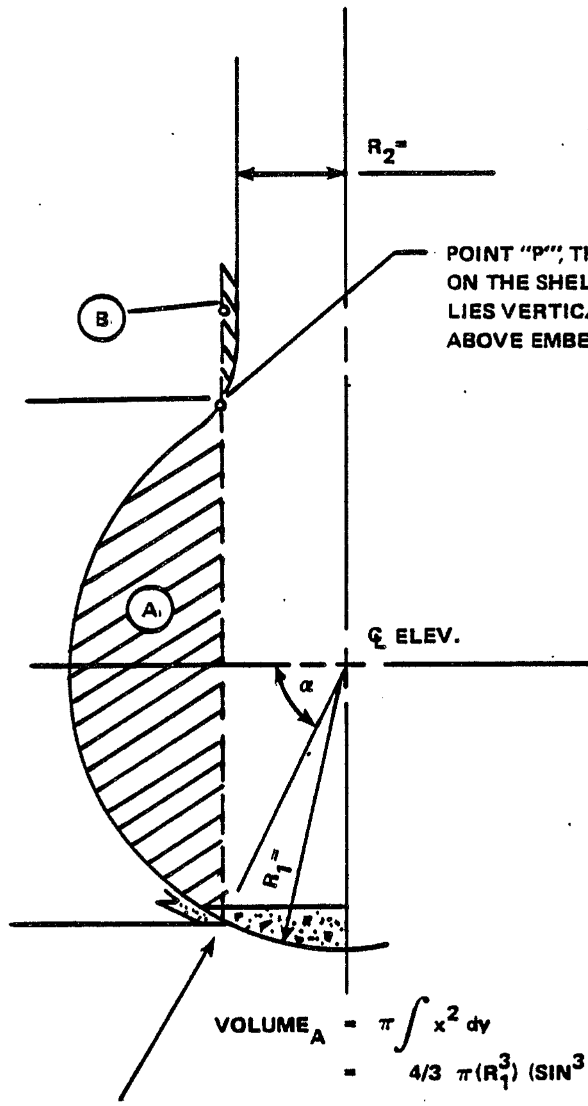
FIGURE 3.13-83

DRYWELL ANALYSIS



$$\text{VOLUME}_A = \pi \int x^2\,dy$$
$$= 4/3\ \pi (R_1^3)(\sin^3 \alpha)$$

PROGRAM 7–78 (SUBSECTION 3.13.2.8)
DRYWELL PRIMARY MEMBRANE
STRESS ANALYSIS

CHICAGO BRIDGE & IRON COMPANY

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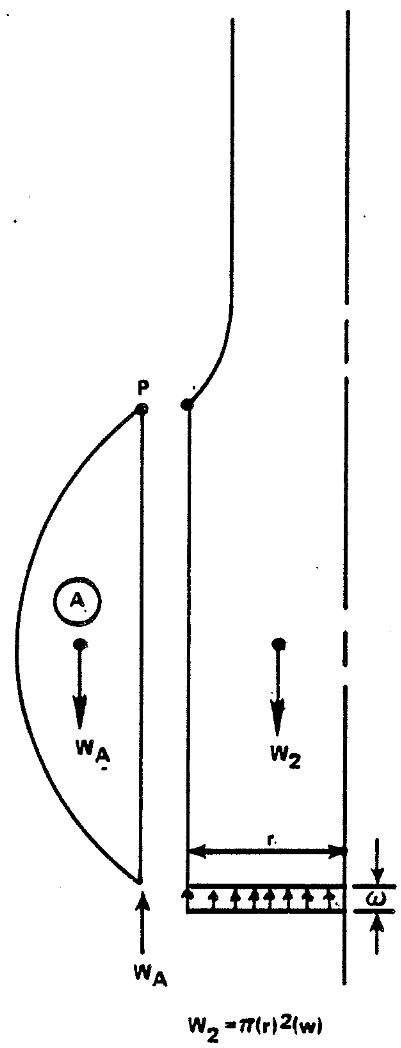
FIGURE 3.13-84

EMBEDMENT LOCATION AT WHICH THE STRESSES ARE ANALYZED



PROGRAM 7–78 (SUBSECTION 3.13.2.8)
DRYWELL PRIMARY MEMBRANE
STRESS ANALYSIS

CHICAGO BRIDGE & IRON COMPANY

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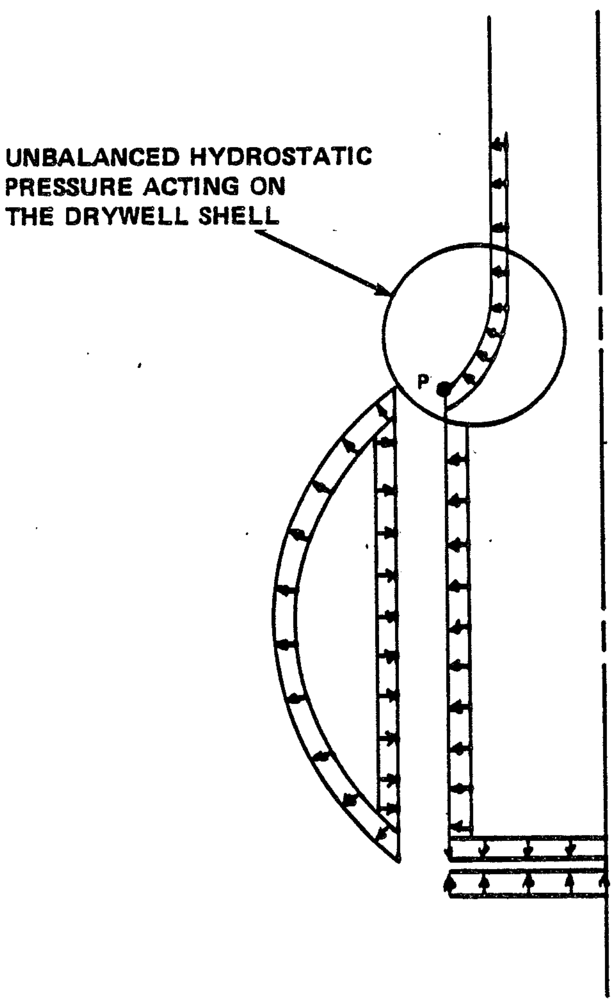
FIGURE 3.13-85

FREE-BODY DIAGRAM, WATER WEIGHT



PROGRAM 7–78 (SUBSECTION 3.13.2.8)
DRYWELL PRIMARY MEMBRANE
STRESS ANALYSIS

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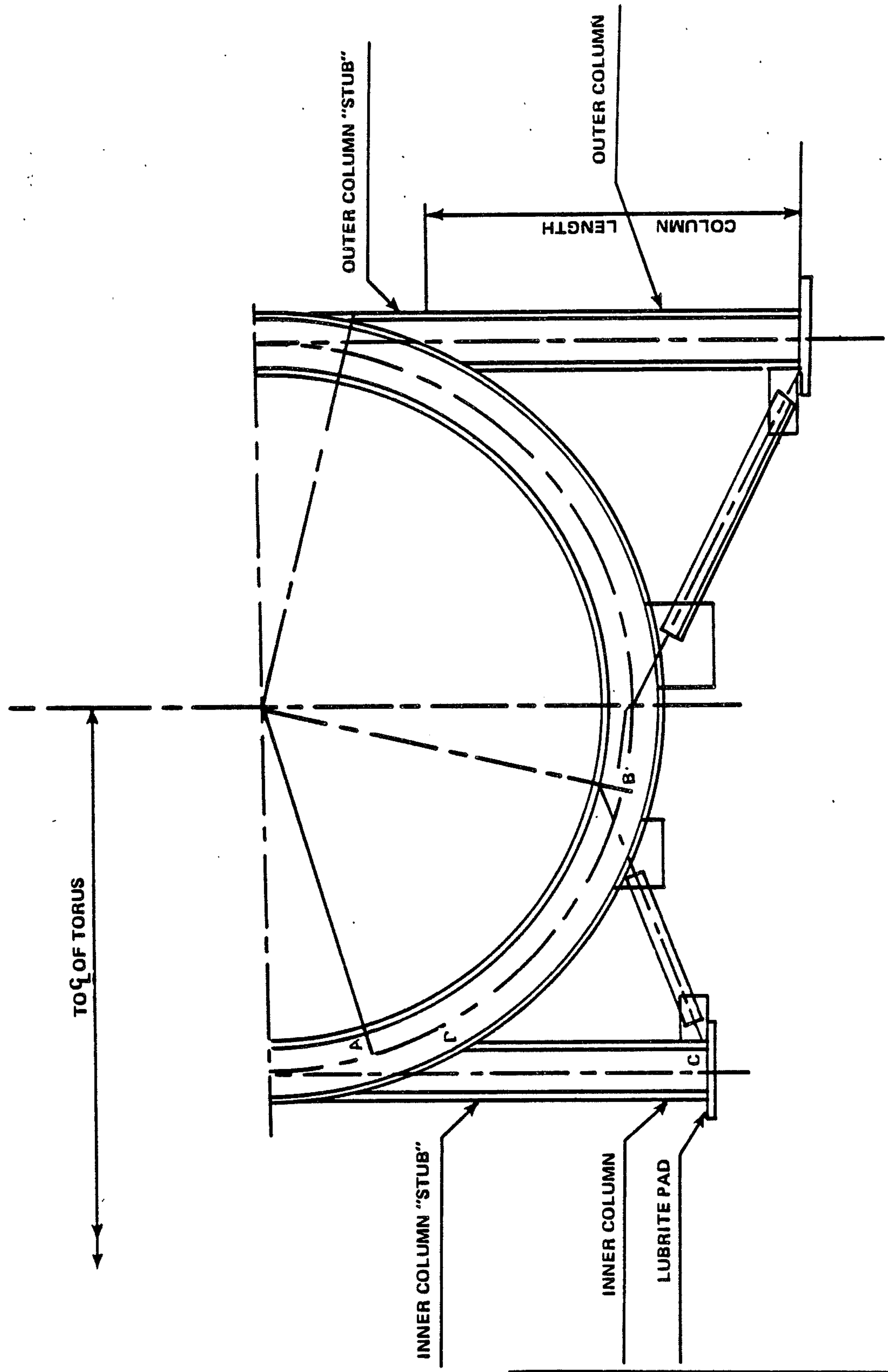
FIGURE 3.13-86

FREE-BODY DIAGRAM, WATER PRESSURE



PROGRAM 7–71 (SUBSECTION 3.13.2.9)
TORUS COLUMNS AND COLUMN
"STUBS" DESIGN

CHICAGO BRIDGE & IRON COMPANY

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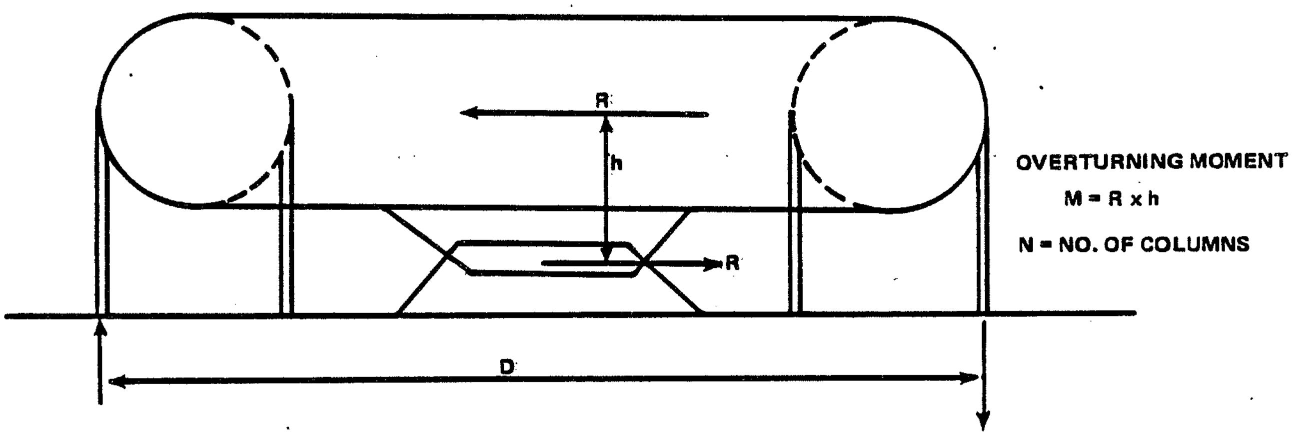
FIGURE 3.13-87

TORUS COLUMNS AND COLUMN "STUBS" DESIGN



COLUMN AXIAL LOAD DUE TO OVERTURNING

MOMENT = $\frac{4M}{ND}$

PROGRAM 7–71 (SUBSECTION 3.13.2.9)
TORUS COLUMNS AND COLUMN
"STUBS" DESIGN

CHICAGO BRIDGE & IRON COMPANY

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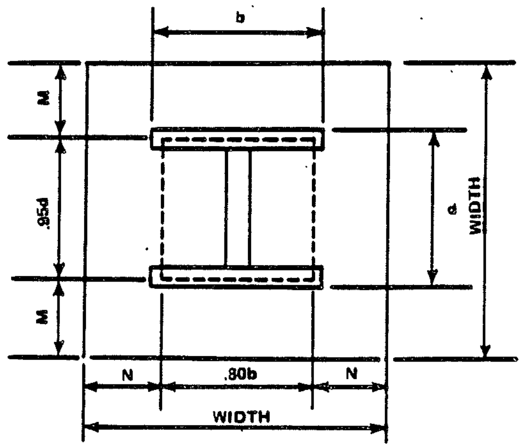
FIGURE 3.13–88

COLUMN AXIAL LOAD DUE TO OVERTURNING

(a)



(b)

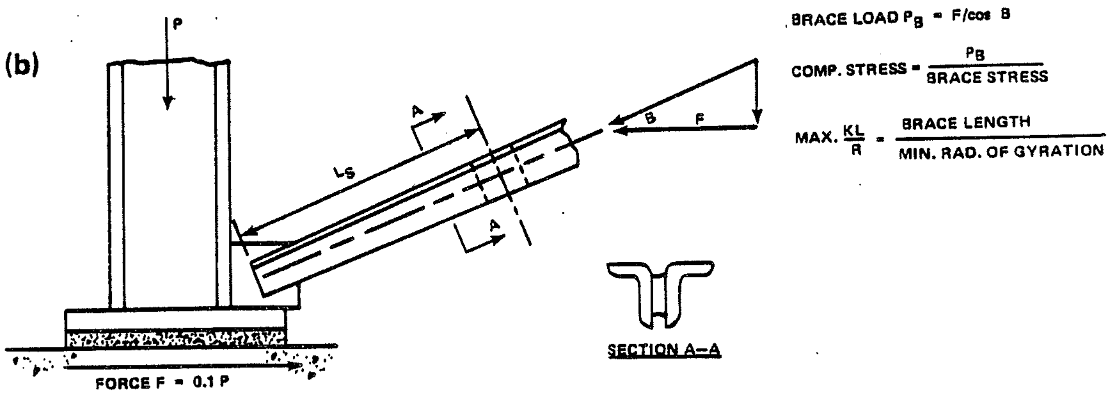


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FIGURE 3.13-89

SUPPRESSION POOL COLUMNS

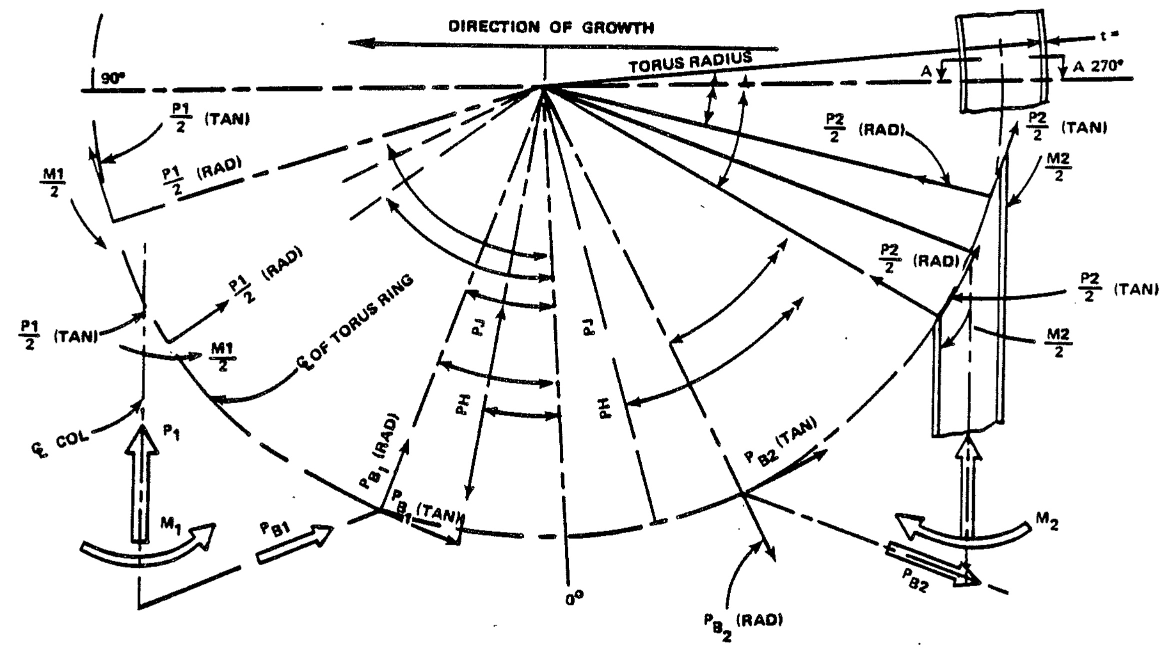

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FIGURE 3.13-90

SUPPRESSION POOL LOADS

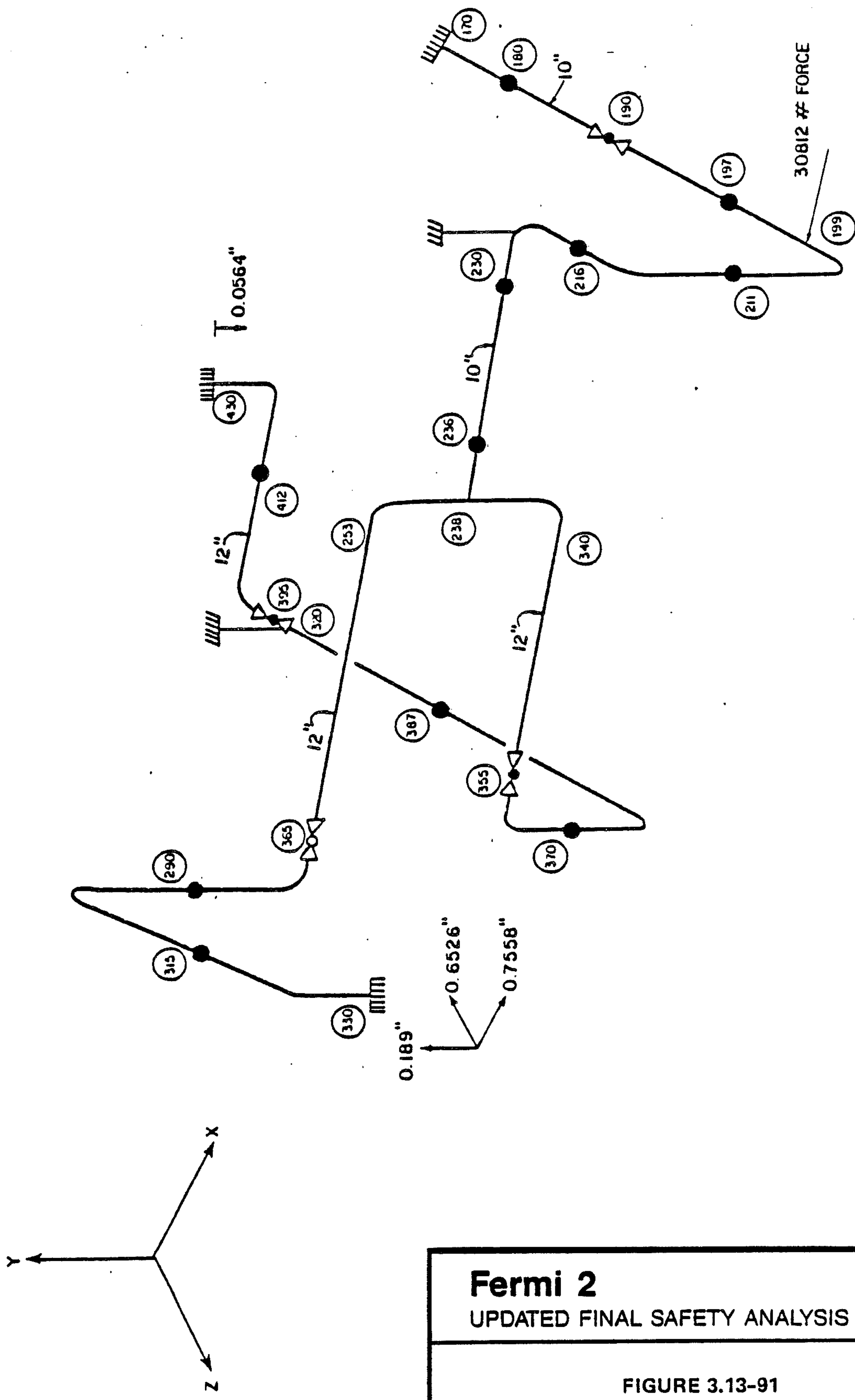

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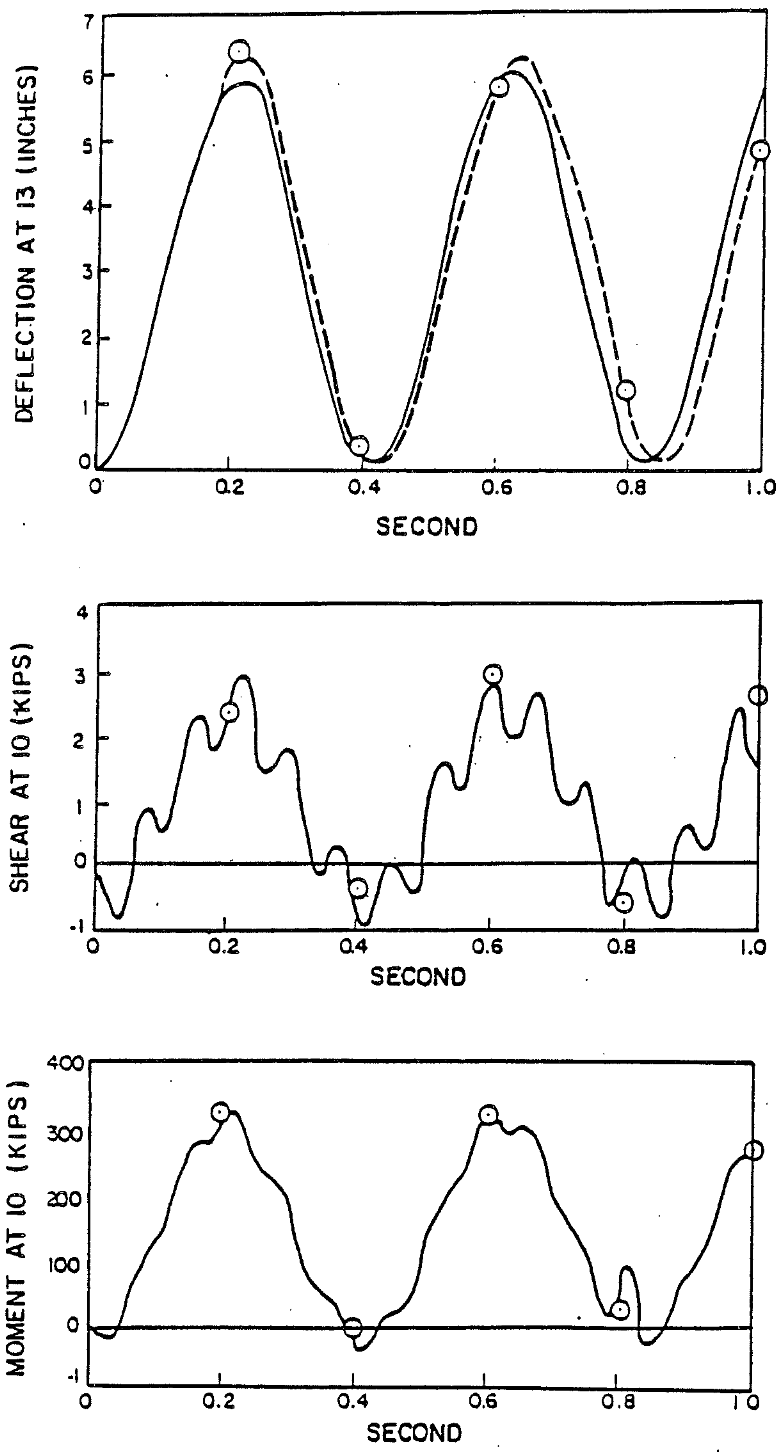
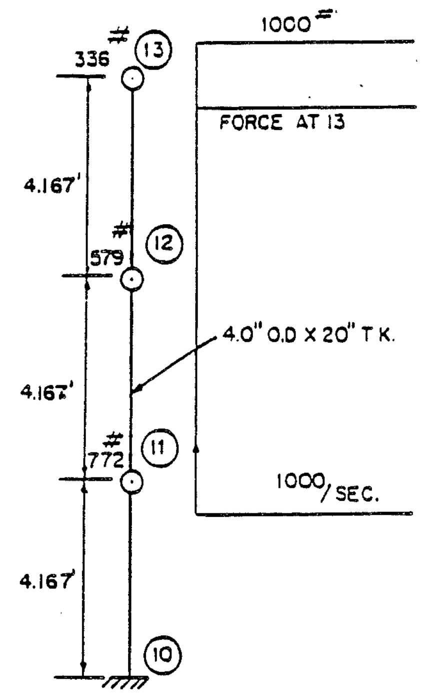
FIGURE 3.13-91

MATHEMATICAL MODEL FOR FLEXIBILITY ANALYSIS VERIFICATION





ASME BENCHMARK PROBLEM 5
MODEL

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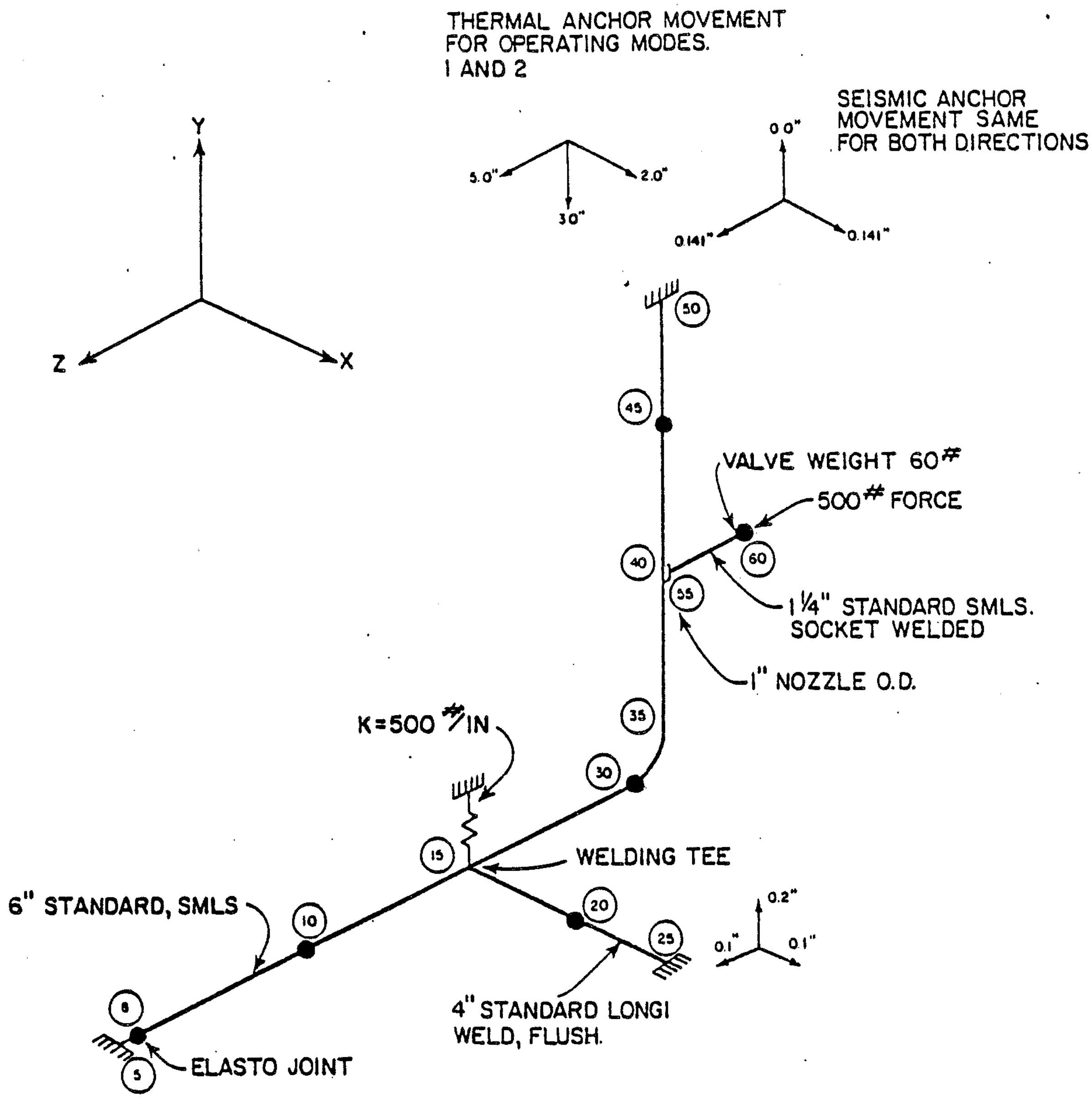
FIGURE 3.13-92

NUPIPE PROGRAM FORCE TIME-HISTORY VERIFICATION



OPERATING CONDITIONS			
OPER MODE	PIPE	PRESSURE (PSI)	TEMPERATURE °F
1	6"	200	400
	4"	"	"
	1¼"	"	"
E/α₁Ta-αbTb/AT 15 = 440 PSI			
2	6"	200	700
	4"	0	70
	1¼"	200	700
$\alpha \Delta T_1 = 0.0002$ $\alpha \Delta T_2 = 0.0004$ IN/FT			
3	6"	700	70
	4"	700	70
	1¼"	700	70

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FIGURE 3.13-93

MATHEMATICAL MODEL FOR CLASS 1 STRESS VERIFICATION