Millstone Power Station Unit 3
Safety Analysis Report

Chapter 3: Design of Structures, Components, Equipment, and Systems

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

Table of Contents

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

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

Table of Contents (Continued)

APPENDIX 3A - COMPUTER PROGRAMS FOR DYNAMIC AND STATIC ANALYSIS OF SEISMIC CATEGORY I STRUCTURES, EQUIPMENT, AND COMPONENTS

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

Table of Contents (Continued)

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

List of Tables

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

List of Tables (Continued)

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

List of Tables (Continued)

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

List of Tables (Continued)

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

List of Tables (Continued)

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

List of Tables (Continued)

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

List of Tables (Continued)

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

List of Tables (Continued)

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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

List of Figures

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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

List of Figures (Continued)

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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

List of Figures (Continued)

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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

List of Figures (Continued)

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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

List of Figures (Continued)

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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

List of Figures (Continued)

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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

List of Figures (Continued)

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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

List of Figures (Continued)

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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

List of Figures (Continued)

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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

List of Figures (Continued)

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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

List of Figures (Continued)

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

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

This section evaluates the design bases of Millstone 3 as measured against the NRC General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR 50, as amended through October 27, 1978. The General Design Criteria (GDC) consist of the single failure definition and 55 individual criteria, and are intended to establish the basic requirements for the principal design criteria of nuclear power plants.

The General Design Criteria were not written specifically for the pressurized water reactor; rather, they were intended to guide the design of all water-cooled nuclear power plants. As a result, the criteria are generic in nature and subject to interpretation. 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 criteria is discussed. For the single failure definition and each of the 55 criteria, a specific assessment of the plant design is made and references are included, where necessary, to identify where detailed design information pertinent to each criterion is treated in this FSAR. For the purpose of this report, the terms “important to safety,” “safety related,” and “safety systems” are synonymous.

Based on the contents herein, the Applicants conclude that Millstone 3 satisfies and complies with the GDC, with the exceptions as noted.

3.1.1 CONFORMANCE WITH SINGLE FAILURE CRITERION

3.1.1.1 Single Failure Criterion

Appendix A to 10 CFR 50, as amended through October 27, 1978, defines single failure criterion as follows:

> “A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly) results in a loss of the capability of the system to perform its safety functions.”

3.1.1.2 Definitions of Terms Used in Single Failure Criterion

Active Component Failure

An active component is one in which mechanical movement must occur to complete the component’s intended function. An active component failure is failure of the component to complete its intended function upon demand.

Spurious action of a powered component originating within its actuation system shall be regarded as an active component failure unless specific design features or operating restrictions preclude such spurious action.

Examples of active component failures include the failure of a powered, manual, or check valve (except ECCS check valves within the NSSS vendor original scope of design) to move to its correct position; the failure of an electrical breaker or relay to respond; and the failure of a pump, fan, or emergency diesel generator to start.

Passive Component Failure

A passive component is one in which mechanical movement need not occur for the component to perform its intended function. A passive component failure is the structural failure of the component or the blockage of a process flowpath by the passive component so that the component does not perform its intended function. For fluid pressure boundary components, a passive component failure is a break of, or crack in, the pressure boundary.

Passive components include piping, cables, and valve bodies.

Short Term

The short term is defined as the first 24 hours following the start of an incident.

Long Term

The long term is defined as the period following the short term (i.e., greater than 24 hours), during which time the system safety function is still required as defined in the individual equipment qualification documentation described in Section 3.11.

Related Service Systems

Related service systems are those systems which provide the services necessary to a fluid system to enable that system to complete its intended safety function.

Examples of related service systems for the emergency core cooling system (ECCS) include the safety injection pumps cooling system, charging pumps cooling system, service water system, electric power supply system, protection systems, and the ECCS equipment area ventilation systems.

3.1.1.3 Application of Single Failure Criterion

For incidents of moderate frequency (Chapter 15) that can result in automatic reactor or turbine trip when Millstone 3 is generating power for offsite transmission, Millstone 3 is designed to maintain capability for cold safe shutdown assuming a single failure.

The aggregate of fluid systems provided to mitigate the consequences of an infrequent incident or limiting fault (Chapter 15) are designed to tolerate a single failure in addition to the incident which requires their function, without loss of safety function to Millstone 3. Examples of systems in this category are: the ECCS, the containment depressurization systems, the supplementary leak collection and release systems, the auxiliary feedwater system, and the cooling water systems used in combination with these systems.

The single failure considered is a random failure in addition to:

1. The initiating event for which the system is required,
2. Any failures which are direct consequences of the initiating event, and
3. Loss of offsite electric power if the initiating event results in a trip of either the turbine generator or reactor protection system.

For fluid systems, the single failure is limited to an active component failure during the short term, and assuming no prior failure during the short term, the single failure is either an active or a passive component failure during the long term.

High energy piping ruptures and moderate energy leakage cracks are considered separately as single postulated initial events occurring during normal plant operation. A single active failure is assumed in systems used to mitigate the consequences of the piping failure and to shut down the reactor. The single active failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure.

Where the initial postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential system (i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure), single failures of components in the other train or trains of that system are not assumed, provided that the system is designed to Seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to QA Category I and to the testing and inservice inspection standards appropriate for nuclear safety systems. Examples of systems that qualify as dual-purpose essential systems are the service water system, the reactor plant component cooling system, and the residual heat removal systems.

A single active failure is not assumed in a dual-purpose safe shutdown system only if:

- the initial failure event occurs from a source external to the dual-purpose safe shutdown system;
- the event results in the loss of function of one of the two redundant trains of the shutdown system;
- the event does not result in a reactor or turbine trip; and

- the safe shutdown system's components are not required to be actuated (i.e.; change state upon demand) to achieve cold shutdown.

Under these circumstances, continued operation of the plant is controlled by plant Technical Specifications.

A single active failure is not assumed in a safety-related system if the initial failure event is postulated in one of two redundant trains of that system whose function is not required during normal plant operation or to achieve cold shutdown if the failure does not require an automatic protective action to mitigate the consequences of the failure. Under these circumstances, continue operation of the plant is controlled by plant Technical Specifications.

For electrical and protection systems, the single failure can be either an active or passive failure during the short or long term. This is in accordance with IEEE-308 and Regulatory Guide 1.53 (Section 7.1.2).

For the purpose of single failure criterion, the system subject to the criterion includes the safety system itself and its related service systems, such as reactor plant component cooling, service water, electric power, etc., required for the safety system to perform its safety function. Even though each system indicated above is designed to the single failure criterion, only one single failure and its consequences are assumed to take place in the aggregate of safety systems and related service systems in the plant.

Nonsafety related systems are designed such that their failure does not cause safety related systems to lose their safety function.

If the proper active function of a component is demonstrated despite any reasonable postulated condition, that component is considered especially qualified for service and exempt from active component failure. Examples of such component function exemptions include opening of code safety valves and ECCS swing check valves within the NSSS vendor original scope of design.

This exemption from single active failure consideration applies only to the ECCS system design philosophy employed by the NSSS vendor. As active components (described in FSAR Section 3.9), ECCS check valves are subject to stringent design criteria and their operational readiness is periodically verified in accordance with applicable plant technical specifications and the MP3 Inservice Testing Program.

In the case of ECCS check valves, the differential pressures required to open the check valve in the forward direction, and the reverse flow velocity required to close the check valve, are very small in comparison to the differential pressures and reverse flow velocities which will be generated if the check valve does not change position immediately as required. Therefore, failure of the check valve to open or close is likely to be momentary, and not of sufficient duration to impact the safety function of the system. In the design of the ECCS, therefore, the failure of a safety-related check valve to open when required to pass flow in the forward direction, or the failure of a safety-related check valve to seat to prohibit flow in the reverse direction is not considered to be a credible event.

Passive component failure assumed in failure analysis of a system is defined by review of each component in the system, considering conditions of operation and possible failure or leakage modes. As an example, review of systems involving piping, heat exchangers, valves, flange joints, and system interface barriers results in a definition of a design leak rate for passive component failure evaluation based on maximum flow through a failed packing, a mechanical seal, or a piping crack (having an opening in size equal to half the pipe diameter in length and half the pipe wall thickness in width for non-ECCS systems). Passive component failure is considered for system functional design adequacy and effects from flooding only.

As an exception, for ECCS systems containing recirculated sump fluid during the long term post LOCA mode of operation, only limited passive failures, with leakage rates up to 50 gpm, are postulated to occur in piping components or as pump mechanical seal failures. In some areas of the ESF building and pipe tunnel, these long term failures are precluded by conservative piping system design (refer to Section 3.6 for passive failure exclusion design criteria).

Means are provided to detect and isolate limited passive failures of up to 50 gpm, from ECCS components, within approximately 30 minutes in the ESF building and in approximately 60 minutes in the Auxiliary building (refer to Section 6.3 for ECCS failure modes and effects analysis and a description of leakage detection and isolation design features).

If proper passive functions of a component can be demonstrated despite any reasonable postulated conditions, that component is considered especially qualified for service and exempt from passive component failure. One example of such a case is not assuming a passive component failure in ductwork.

In lieu of a review of all the components in a system, complete severance of a line is assumed as the passive component failure.

Passive component failure or leakage in excess of limits defined in the Technical Specifications (Section 16.1) is not considered in the primary reactor containment vessel boundary and the containment isolation systems.

Conservative piping design criteria have been applied in conjunction with dual isolation valves located outside containment as an acceptable alternative to General Design Criterion 56 for certain containment penetrations described in Section 6.2. Passive failures are not assumed in the line outside containment (between the containment and the outboard isolation valve), provided that the stresses are low, augmented inservice inspection is performed, and protection from postulated missiles is assured.

For design, the mass of fluid discharged through the crack or break prior to effective isolation is considered. Leakage duration is conservatively determined consistent with leakage detection, location, and isolation mechanisms. As a guide, a nominal 30-minute leakage duration is considered conservative for manual operator action outside of the control room.

The plant design is such that all active components of the designated safety systems and related service systems can be proved operational by scheduled periodic operational tests or operational status indications.

3.1.2 CRITERION CONFORMANCE

3.1.2.1 Quality Standards and Records (Criterion 1)

Criterion

"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 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."

Design Conformance

Structures, systems, and components important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

Quality standards applicable to safety related structures, systems, and components are generally contained in codes such as the ASME Boiler and Pressure Vessel Code. The applicability of these codes is specifically identified throughout this report and is summarized in Section 3.2.5. Chapter 17 provides direct reference to the Quality Assurance Program established to provide assurance that safety related structures, systems, and components satisfactorily perform their intended safety functions. The procedures for generating and maintaining appropriate design, fabrication, erection, and testing records are contained within the referenced documents.

3.1.2.2 Design Bases for Protection Against Natural Phenomena (Criterion 2)

Criterion

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

Design Conformance

Those features of plant facilities that are essential to the prevention of accidents that could affect the public health and safety or to the mitigation of accident consequences are designed to:

1. Quality standards that reflect the importance of the function to be performed. Approved design codes are used when appropriate to the nuclear application.

2. Performance standards that enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquake, flooding condition, wind, ice, or other natural phenomena for the site, and credible combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

Features of the facility essential to accident prevention and mitigation of accident consequences, which are designed to withstand the effects of natural phenomena, are:

1. The reactor coolant pressure boundary and containment barriers

2. The controls and emergency cooling systems whose functions are to maintain the integrity of these barriers

3. Systems which depressurize the containment following a loss of coolant accident (LOCA)

4. Power supply and essential services

5. Reactivity systems, monitoring systems, and fuel systems

6. The components used to store and cool spent reactor fuel

All piping, components, and supporting structures of the reactor and safety related systems are designed to withstand a specified seismic disturbance and credible combinations of effects of normal and accident conditions coincident with the effects of natural phenomena. Plant design criteria specify that there is to be no loss of function of such equipment in the event of the safe shutdown earthquake (SSE) ground acceleration acting in the horizontal and vertical directions simultaneously. The dynamic response of Seismic Category I structures to ground acceleration, based on an envelope of characteristics of the site foundation soils and on the critical damping of the foundation and structures, is included in the design analysis.

Design of structures for protection against natural phenomena is described in Section 3.8. Safety related structures have sufficient capacity to accept a combination of normal operating loads, functional loads due to the design basis accident (DBA), and the loadings imposed by the maximum wind velocity, or those due to the SSE, whichever is the larger.

The emergency onsite power sources are not subject to interruption due to earthquake, windstorm, floods, or to disturbances on the external power transmission system.

Power cabling, motors, and other equipment required for operation of the engineered safety features are suitably protected against the effects of the design basis accident (DBA) and from severe external weather conditions, as applicable.

Unit design criteria which ensure protection against natural phenomena are described in Section 3.2 (Classification of Structures, Systems, and Components), Section 3.3 (Wind and Tornado Loadings), Section 3.4 (Water Level (Flood) Design), and Section 3.7 (Seismic DesignS).

3.1.2.3 Fire Protection (Criterion 3)

Criterion

"Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. 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."

Design Conformance

The design of Millstone 3 minimizes the probability and effect of fires and explosions on structures, systems, and components important to safety. Noncombustible and heat-resistant materials are used wherever practical throughout the unit. Fire detection and fire suppression systems of sufficient capacity and capability minimize the adverse effects of fires on structures, systems, and components important to safety. Fire suppression systems are designed to assure that rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Section 9.5.1 and Millstone 3 Fire Protection Evaluation Report describe the fire protection system in detail.

3.1.2.4 Environmental and Missile Design Bases (Criterion 4)

Criterion

"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."

Design Conformance

Structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operating, maintenance, testing, and postulated accidents including LOCA's. These items are either protected from accident conditions or designed to withstand, without failure, exposure to the combination of temperature, pressure, humidity, radiation, and dynamic effects expected during the required operational period.

Physical separation, physical protection, pipe restraints, and redundancy are included in the design of safety related systems to ensure that each such system performs its intended safety function.

In a letter from B. J. Youngblood (NRC) to J. F. Opeka (NNECO) dated June 5, 1985, Millstone 3 was granted an exemption for a period of two cycles of operation from those portions of General Design Criterion 4 which require protection of structures, systems, and components from the dynamic effects associated with postulated breaks in the reactor coolant system primary loop piping.

In Federal Register, Volume 51, No. 70, dated April 11, 1986, the NRC published a final rule modifying General Design Criterion 4 to allow use of leak-before-break technology for excluding from the design basis the dynamic effects of postulated ruptures in primary coolant loop piping in pressurized water reactors. This rule obviates the need for the above exemption.

Structures, systems, and components important to safety are classified as QA Category I and are designed in accordance with the codes and classifications indicated in Section 3.2.5.

Chapter 3 provides the details of the environmental activities and dynamic effects to which the structures, systems, and components important to safety are designed.

3.1.2.5 Sharing of Structures, Systems, and Components (Criterion 5)

Criterion

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

Design Conformance

A system important to safety and shared by Millstone 1, 2, and 3 is the radioactive gaseous waste system discharge stack described in Section 11.3. The sharing of the discharge stack causes no safety impairment of the three units.

The following equipment may be shared and utilized by Millstone Unit 2 to meet its GDC 17 requirements for an alternate offsite source to relieve one of its emergency diesel generators and supply power to minimum post-accident loads:

1. Main Transformer 15G-3X
2. Normal Station Service Transformer 15G-3SA
3. Reserve Station Service Transformer 15G-23SA

The sharing of this equipment does not impair its ability to perform its safety function. The transformers are adequately sized and have sufficient capacity to meet maximum postulated Unit 3 loading requirements while supplying Unit 2 GDC 17 minimum loads.

Other facilities and systems not important to safety within the definitions of GDC 5, but which are shared by the three units are:

1. Environmental monitoring systems
2. Machine shops
3. Offsite transmission lines and switchyard
4. Office buildings
5. Roadway access
6. Railroad access
7. General warehouses

8. Fire protection system

9. Potable (Waterford) water

10. Warehousing facility houses the Millstone 2 condensate polishing system, the Millstone 2 and 3 (removed from service) condensate demineralizer radioactive liquid waste systems. The Unit 2 condensate polishing solid waste system is designed to process radioactive waste from Millstone 2 and 3.

11. Alternate AC (SBO) Diesel Generator (shared by Units 2 and 3)

12. Security System

13. Auxiliary Steam System

3.1.2.6 (Criterion 6)

Criterion 6 has not been promulgated by the NRC.

3.1.2.7 (Criterion 7)

Criterion 7 has not been promulgated by the NRC.

3.1.2.8 (Criterion 8)

Criterion 8 has not been promulgated by the NRC.

3.1.2.9 (Criterion 9)

Criterion 9 has not been promulgated by the NRC.

3.1.2.10 Reactor Design (Criterion 10)

Criterion

"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."

Design Conformance

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to:

1. Assure that fuel damage is not expected during normal core operation and operational transients (Condition I) or any transient conditions arising from

occurrences of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the plant cleanup system and are consistent with plant design bases.

2. Ensure return of the reactor to a safe state following infrequent incident (Condition III) events with only a small fraction of fuel rods damaged, although sufficient fuel damage might occur to preclude immediate resumption of operation.

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

Note that fuel damage as used under Item 1 is defined as penetration of the fission product barrier (i.e., the fuel rod clad). Also note that ANSI N18.2-73 expands the definitions of the four conditions enumerated in Items 1 through 3.

Chapter 4 discusses the design bases and design evaluation of reactor components. Chapter 7 gives the details of the control and protection systems instrumentation design and logic. This information supports the accident analyses of Chapter 15 which show that the acceptable fuel design limits are not exceeded for Condition I and II occurrences.

3.1.2.11 Reactor Inherent Protection (Criterion 11)

Criterion

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

Design Conformance

Prompt compensatory reactivity feedback effects are assured when the reactor is critical by the negative fuel temperature effect (Doppler effect) and by ensuring that the moderator temperature coefficient is maintained within the limits provided in Technical Specification 3/4.1.1.3. Section 4.3.2.3 discusses these reactivity coefficients.

3.1.2.12 Suppression of Reactor Power Oscillations (Criterion 12)

Criterion

"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."

Design Conformance

Total power oscillations of the fundamental mode are inherently stable by the negative power coefficient of reactivity.

Oscillations, due to xenon spatial effects, in the radial, diametral, and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative power coefficient of reactivity.

Oscillations, due to xenon spatial effects, in the axial first overtone mode may occur. Assurance that fuel design limits are not exceeded by xenon axial oscillations is provided by reactor trip functions using the measured axial power imbalance as an input.

Oscillations, due to xenon spatial effects, in axial modes higher than the first overtone are heavily damped due to the inherent design and due to the negative Doppler coefficient of reactivity.

Section 4.3 discusses xenon and samarium stability control.

3.1.2.13 Instrumentation and Control (Criterion 13)

Criterion

"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."

Design Conformance

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

The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems over the full design range of the plant. These systems are described in Chapters 6, 7, 8, 9, 11, and 12.

3.1.2.14 Reactor Coolant Pressure Boundary (Criterion 14)

Criterion

"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."

Design Conformance

The reactor coolant system boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including all anticipated transients, and to maintain the stresses within applicable stress limits (Section 3.9). Reactor coolant pressure boundary materials, selection, and fabrication techniques ensure a low probability of gross rupture or abnormal leakage.

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

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

3.1.2.15 Reactor Coolant System Design (Criterion 15)

Criterion

"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."

Design Conformance

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

Additionally, reactor coolant pressure boundary components achieve a large margin of safety by the use of proven ASME materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and integrated hydrostatic testing of assembled components. Chapter 5 discusses the reactor coolant system design.

3.1.2.16 Containment Design (Criterion 16)

Criterion

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

Design Conformance

A steel-lined reinforced concrete containment structure, maintained at subatmospheric pressure, encloses the entire reactor coolant system with an essentially leaktight barrier, as described in Section 6.2.1. The containment structure and the engineered safety features are designed to withstand internal and external environmental conditions that may reasonably be expected during the life of the unit and to ensure that the short and long term conditions following a LOCA do not exceed the design values. Following a design basis accident (DBA), the containment heat removal systems reduce the containment pressure, as described in Sections 6.1.2 and 6.2.2. Most of the leakage from the containment structure is collected and processed through the supplementary leak collection and release system, described in Section 6.2.3. This process reduces the amount of radioactivity released to the environment.

3.1.2.17 Electric Power Systems (Criterion 17)

Criterion

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

"The onsite electric power 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 accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained."

"Provisions shall be included to minimize the probability of losing electric power from any of the remaining 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 electric power supplies."

Design Conformance

Two connections to the offsite power system are provided. The preferred offsite connection is a backfeed through the main and normal station service transformers with the generator breaker open. The alternate offsite connection is through the reserve station service transformers. Each offsite source has 100 percent capacity for all emergency and normal loads during all phases of operation plus the capacity to supply Millstone Unit 2 GDC 17 requirements through the NSST or RSST as an alternate offsite source for minimum post-accident loads.

Two onsite power systems are provided. Each system has an emergency diesel generator. Each diesel generator has 100 percent capacity for the emergency loads in the event of the postulated accidents or required for reactor cooldown.

The design of the electrical system (Chapter 8) conforms to Criterion 17.

3.1.2.18 Inspection and Testing of Electric Power Systems (Criterion 18)

Criterion

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

Design Conformance

The engineered safety features power supply buses, the power supply buses for equipment required for cooldown, and associated emergency generators, all of which comprise the onsite power system, are arranged for periodic testing of each system independently. During refueling shutdowns, tests are conducted to prove the operability of the automatic starting and load sequencing capability of the emergency generators. Full load testing of each emergency generator is performed periodically. These tests prove the operability of the electric power systems under conditions as close to design as practical to assess the continuity of these systems and condition of the components. The electric power systems and the testing specifications are described in Sections 8.3.1.1 and 16.4.8, respectively.

3.1.2.19 Control Room (Criterion 19)

Criterion

"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."

Design Conformance

The control room provided is equipped to operate the unit safely under normal and accident conditions. Its shielding and ventilation design permits continuous occupancy of the control room for the duration of a DBA without the dose to personnel exceeding 5 rem whole body. Based on 10 CFR 50.67, the applicable dose criterion was modified to 5 rem TEDE.

The auxiliary shutdown panel located in the west switchgear room has equipment, controls, and instrumentation to accomplish, in conjunction with controls and indication located on the adjacent 4160V switchgear, a prompt hot shutdown and the capability for subsequent cold shutdown of the reactor through the use of suitable procedures. The panel is physically located outside the control room. Thus, the uninhabitability of the control room would have no effect on the availability of the auxiliary shutdown panel and adjacent controls (Section 7.4.1.3).

The design of the control building (Section 3.8.4), which houses the control room and the auxiliary shutdown panel area, conforms to Criterion 19. Section 9.4.0 describes the control building ventilation system. Control room habitability is discussed in Section 6.4. Fire protection systems are discussed in Section 9.5.1.

3.1.2.20 Protection System Functions (Criterion 20)

Criterion

"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."

Design Conformance

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

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

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

3.1.2.21 Protection System Reliability and Testability (Criterion 21)

Criterion

"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."

Design Conformance

The protection system is designed for high functional reliability and inservice testability.

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

3.1.2.22 Protection System Independence (Criterion 22)

Criterion

"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."

Design Conformance

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

Section 7.1.2.1.8 provides details of ESF system diversity.

Automatic reactor trips are based upon process parameters and neutron flux measurements. Trips on process parameters include reactor coolant loop temperature measurements, pressurizer pressure and level measurements, and reactor coolant pump underspeed trip. Trips may also be initiated manually or by SIS. Section 7.2 describes all the trips and provides further details.

High quality components, conservative design and applicable quality control, inspection, calibration, and tests are utilized to guard against common-mode failure. Sections 3.10 and 3.11 provide details concerning qualification testing. Qualification testings is performed on the various safety systems to demonstrate functional operation at normal and post-accident conditions of temperature, humidity, pressure, and radiation for specified periods, if required. Typical protection system equipment is subjected to type tests under simulated seismic condition using conservatively large accelerations and applicable frequencies. The test results indicate no loss of the protection function.

3.1.2.23 Protection System Failure Modes (Criterion 23)

Criterion

"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."

Design Conformance

The protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. Sections 7.2 and 7.3 discuss this protection system.

3.1.2.24 Separation of Protection and Control Systems (Criterion 24)

Criterion

"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."

Design Conformance

The protection system is separate and distinct from the control systems. Control systems may be dependent on the protection system in that control signals are derived from protection system measurements, where applicable. These signals are transferred to the control system by isolation devices which are classified as protection components. The adequacy of system isolation is verified by testing under conditions of postulated credible faults. The 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 which satisfies the requirements of the protection system. Distinction between channel and train is made in this discussion. The removal of a train from service is allowed only during testing of the train. Chapter 7 gives further details.

3.1.2.25 Protection System Requirements for Reactivity Control Malfunctions (Criterion 25)

Criterion

"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."

Design Conformance

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

Chapter 15 discusses analyses of the effects of possible malfunctions. These analyses show that for postulated dilution during refueling, startup, or manual or automatic operation at power, the operator has ample time to determine the cause of dilution, terminate the source of dilution, and initiate boration before the shutdown margin is lost. The analyses show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

3.1.2.26 Reactivity Control System Redundancy and Capability (Criterion 26)

Criterion

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

Design Conformance

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

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

The chemical and volume control system maintains the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Chapter 4 presents details of the construction of the RCCA's and Chapter 7 discusses the operation. Chapter 9 describes the means of controlling the boric acid concentration. Chapter 15 includes performance analyses under accident conditions.

3.1.2.27 Combined Reactivity Control System Capability (Criterion 27)

Criterion

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

Design Conformance

The facility is provided with means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. Chapter 4 and 9 discuss these means in detail. Combined use of the rod cluster control system and the chemical shim control system permits the necessary shutdown margin to be maintained during long term xenon decay and plant cooldown. The single highest worth control cluster is assumed to be stuck full-out upon trip for this determination. Chapter 15 describes accident assumptions in detail.

3.1.2.28 Reactivity Limits (Criterion 28)

Criterion

"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."

Design Conformance

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

The maximum positive reactivity insertion rates for the withdrawal of rod cluster control assemblies (RCCA's) and the dilution of the boric acid in the reactor coolant system are limited by the physical design characteristics of the RCCA's and of the chemical and volume control system. Technical Specifications on shutdown margin and on RCCA insertion limits and bank overlaps as functions of power provide additional assurance that the consequences of the postulated accidents are no more severe than those presented in the analyses of Chapter 15. Reactivity insertion rates, dilution, and withdrawal limits are also discussed in Section 4.3. The capability of the chemical and volume control system to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 15.

Assurance of core cooling capability following Condition IV accidents, such as rod ejections, steam line break, etc., is given by keeping the reactor coolant pressure boundary stresses within faulted condition limits as specified by applicable ASME Codes. Structural deformations are checked also and limited to values that do not jeopardize the operation of necessary safety features.

3.1.2.29 Protection against Anticipated Operational Occurrences (Criterion 29)

Criterion

"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."

Design Conformance

The protection and reactivity control systems are designed to assure extremely high probability of performing their required safety functions in any anticipated operational occurrences. Equipment used in these systems is designed, constructed, operated, and maintained with a high level of reliability. Chapter 7 covers details of system design. Also refer to the discussions of GDC-20 through 28.

3.1.2.30 Quality of Reactor Coolant Pressure Boundary (Criterion 30)

Criterion

"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."

Design Conformance

Reactor coolant pressure boundary components are designed, fabricated, inspected, and tested in conformance with ASME Nuclear Power Plant Components Code, Section III. All components are classified according to ANSI N18.2-73 and N18.2a-75 and are accorded the quality measures appropriate to the classification. The design bases and evaluations of reactor coolant pressure boundary components are discussed in Chapter 5.

Leakage is detected by an increase in the amount of makeup water required to maintain a normal level in the pressurizer. The reactor vessel closure joint is provided with a temperature monitored leakoff between double gaskets. Leakage into the reactor containment is drained to the reactor building sump where it is monitored.

Leakage is also detected by measuring the airborne and gaseous activity and activity of the condensate drained from the reactor building air recirculation units. Monitoring the inventory of reactor coolant in the system at the pressurizer, volume control tank and coolant drain collection tanks make available an accurate indication of integrated leakage.

Section 5.2.5 discusses the reactor coolant pressure boundary leakage detection system.

3.1.2.31 Fracture Prevention of Reactor Coolant Pressure Boundary (Criterion 31)

Criterion

"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."

Design Conformance

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

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

1. Ultrasonic Testing - In addition to code requirements, a 100-percent volumetric ultrasonic test of reactor vessel plate for shear wave and a post-hydro test ultrasonic map of all full penetration ferritic pressure boundary welds in the pressure vessel are performed. Cladding bond ultrasonic inspection to more restrictive requirements than those specified in the code are also required to preclude interpretation problems during inservice inspection.

2. Radiation Surveillance Program - In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation and post-irradiation testing of Charpy V-notch and tensile specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM-E-185-82, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," and the requirements of 10 CFR 50, Appendix H.

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

The fabrication and quality control techniques used in the fabrication of the reactor coolant system are consistent with those used for the reactor vessel. The inspections of reactor vessel, pressurizer, piping, pumps, and steam generator are governed by ASME Code requirements. (Refer to Chapter 5.)

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

3.1.2.32 Inspection of Reactor Coolant Pressure Boundary (Criterion 32)

Criterion

“Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.”

Design Conformance

The design of the reactor coolant pressure boundary provides the capability for accessibility during service life to the entire internal surfaces of the reactor vessel, certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads, and external surfaces of the reactor coolant piping except for the area of pipe within the primary shielding concrete. The inspection capability complements the leakage detection systems in assessing the pressure boundary component’s integrity. The reactor coolant pressure boundary is periodically inspected under the provisions of ASME Boiler and Pressure Vessel Code, Section XI. Section 5.2.4 gives details of the Inservice Inspection Program.

Monitoring of changes in the fracture toughness properties of the reactor vessel core region plates forging, weldments, and associated heat treated zones are performed in accordance with 10 CFR 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements.” Samples of reactor vessel plate materials are retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The observed shifts in RT_{NDT} of the core region materials with irradiation are used to confirm the allowable limits calculated for all operational transients. Section 5.3 gives further details.

3.1.2.33 Reactor Coolant Makeup (Criterion 33)

Criterion

"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."

Design Conformance

The chemical and volume control system provides a means of reactor coolant makeup and adjustment of the boric acid concentration. Makeup is added automatically if the level in the volume control tank falls below the normal operating range. The high-pressure centrifugal charging pumps provided are capable of supplying the required makeup and reactor coolant seal injection flow when power is available from either onsite or offsite electric power systems. These pumps also serve as high head safety injection pumps. Functional reliability is assured by provision of standby components assuring a safe response to probable modes of failure. Sections 6.3 and 9.3 include details of system design and Chapter 8, details of the electric power system.

3.1.2.34 Residual Heat Removal (Criterion 34)

Criterion

"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."

Design Conformance

The residual heat removal system, in conjunction with the steam and power conversion system, is designed to transfer the fission product decay heat and other residual heat from the reactor core within acceptable limits. The transfer of the heat removal function from the steam and power conversion system to the residual heat removal system occurs when the reactor coolant system is at approximately 350°F and 375 psig.

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

Suitable redundancy at temperatures above approximately 350°F is provided by the steam generators and associated piping system.

Section 5.4.7 and Chapter 10 give details of the system design.

3.1.2.35 Emergency Core Cooling (Criterion 35)

Criterion

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

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

Design Conformance

An emergency core cooling system is provided to cope with any loss-of-coolant accident in the plant design basis. Abundant cooling water is available in an emergency to transfer heat from the core at a rate that clad metal-water reaction is limited to less than one percent. Adequate design provisions are made to assure performance of the required safety functions even with a single failure.

Section 6.3 includes details of the capability of the systems. Chapter 15 includes an evaluation of the adequacy of the system functions. Performance evaluations are conducted in accordance with 10 CFR 50.46 and Appendix K to 10 CFR 50.

3.1.2.36 Inspection of Emergency Core Cooling System (Criterion 36)

Criterion

"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."

Design Conformance

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

The components outside the containment are accessible for leak-tightness inspection during operation of the reactor.

3.1.2.37 Testing of Emergency Core Cooling System (Criterion 37)

Criterion

"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."

Design Conformance

Active components of the emergency core cooling system can be actuated from the emergency power source at any time during unit operation to demonstrate operability.

Tests are performed during refueling shutdowns to demonstrate proper automatic operation of the emergency core cooling system. An integrated system test is performed. Sections 6.3 and 7.3 describe the above tests.

3.1.2.38 Containment Heat Removal (Criterion 38)

Criterion

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

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

Design Conformance

Heat is removed from the containment structure following a LOCA by the containment depressurization systems, which consist of the quench spray system and the containment recirculation system (Section 6.2.2). The quench spray system, consisting of two 100-percent capacity subsystems, transfers water from the refueling water storage tank to two parallel 360-degree spray headers. The quench spray system transfers heat from the containment atmosphere to water on the containment structure floor. The containment recirculation system, which consists of two 100-percent capacity subsystems (each consisting of two pumps, two coolers, and a 360-degree spray header), transfers heat from the water collected in the containment structure sump to the service water system (Section 9.2.1) via the containment recirculation coolers. The quench spray pumps and the containment recirculation pumps and coolers are located in the engineered safety features building (Section 3.8).

The containment depressurization systems are designed so that no single active failure in the short term or no single active or passive failure in the long term impairs their ability to perform their safety function. Redundant components are isolated, physically and electrically. Each subsystem is connected to a separate electrical bus which can be connected to either offsite or onsite power.

3.1.2.39 Inspection of Containment Heat Removal System (Criterion 39)

Criterion

"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."

Design Conformance

The design of the containment depressurization systems permits appropriate periodic inspection of the important components, as described in Section 6.2.2.4.

3.1.2.40 Testing of Containment Heat Removal System (Criterion 40)

Criterion

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

Design Conformance

The design of the containment depressurization systems permits periodic pressure and functional testing, as described in Section 6.2.2.4.

3.1.2.41 Containment Atmosphere Cleanup (Criterion 41)

Criterion

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

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

Design Conformance

The supplementary leak collection and release system (SLCRS) (Section 6.2.3.2) collects radioactive leakage from the containment to the containment enclosure and contiguous areas following a LOCA.

The quench spray system (Sections 6.2.2 and 6.5.2) sprays borated water into the containment atmosphere to reduce the containment pressure and remove airborne iodine. The pH in the containment sumps is controlled by the dissolution of trisodium phosphate (stored in baskets) in the sump water.

The recirculation spray system (Sections 6.2.2 and 6.5.2) sprays containment sump water into the containment atmosphere to reduce containment pressure and remove airborne iodine.

These systems are sufficiently redundant to perform their safety function assuming a single active failure in the short term or a single active or passive failure in the long term and are operable with either onsite or offsite power.

3.1.2.42 Inspection of Containment Atmosphere Cleanup Systems (Criterion 42)

Criterion

"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."

Design Conformance

The design of the supplementary leak collection and release system permit appropriate periodic inspection of the important components, as described in Section 6.5.1.4.

3.1.2.43 Testing of Containment Atmosphere Cleanup Systems (Criterion 43)

Criterion

"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."

Design Conformance

The design of the supplementary leak collection and release system permits periodic pressure and functional testing of components, as described in Section 6.5.1.4.

3.1.2.44 Cooling Water (Criterion 44)

Criterion

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

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

Design Conformance

The reactor plant component cooling water system, the charging pump cooling system, spent fuel pool cooling and purification system and the safety injection pump cooling system, transfer heat from systems containing reactor coolant to the service water system. Together, these systems transfer heat to the ultimate heat sink from structures, systems, and components important to safety during normal and accident conditions.

These systems are designed with suitable redundancy in components, with leak protection, and with the capability to isolate redundant components. The systems are designed to satisfy the cooling water requirements assuming a single failure and either a loss of onsite or offsite power. Designs of these systems are described in FSAR sections, as follows:

Title	**Section No.**
AC Power Supply System	8.3.1
Service Water System	9.2.1
Reactor Plant Component Cooling Water System	9.2.2.1
Charging Pumps Cooling System	9.2.2.4
Safety Injection Pumps Cooling System	9.2.2.5
Ultimate Heat Sink	9.2.5
Spent Fuel Pool Cooling and Purification	9.1.3

3.1.2.45 Inspection of Cooling Water System (Criterion 45)

Criterion

"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."

Design Conformance

The service water system (Section 9.2.1), the reactor plant component cooling water system (Section 9.2.2.1), the charging pumps cooling system (Section 9.2.2.4), the safety injection pumps cooling system (Section 9.2.2.5), and the spent fuel pool cooling and purification systems (Section 9.1.3) are designed to permit appropriate periodic inspection in order to ensure the integrity of the components and the systems as a whole. In addition, as the service water, reactor plant component cooling water, charging pumps cooling, and spent fuel pool cooling and purification systems function almost continuously during normal unit operation, their capability and integrity are continuously demonstrated. The safety injection pumps cooling system is operated periodically to assure its capability and integrity.

3.1.2.46 Testing of Cooling Water System (Criterion 46)

Criterion

"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."

Design Conformance

The service water system (Section 9.2.1), reactor plant component cooling water system (Section 9.2.2.1), charging pumps cooling system (Section 9.2.2.4), safety injection pumps cooling system (Section 9.2.2.5), and the spent fuel pool cooling and purification system (Section 9.1.3) are designed to permit periodic pressure and functional testing. With the exception of the safety injection pumps cooling system, these systems operate during normal operation and shutdown; thus, the structural and leaktight integrity of the system components, the operability and performance of most of the active components, and the operability of the system as a whole are continuously demonstrated. The active components that cannot be tested during normal system operation are tested during shutdown.

The safety injection pumps cooling system, which is not normally in service, is periodically tested to assure structural and leaktight integrity of its components, the operability and performance of its active components, and the operability of the system as a whole.

The performance of the full operational sequence for the safety related portions of the above systems that brings each system into operation for reactor shutdown, LOCA, or loss of unit power is evaluated periodically in conjunction with the applicable portions of the protection system.

Transfer between normal and emergency power sources is discussed in Section 8.3.

3.1.2.47 (Criterion 47)

Criterion 47 has not been promulgated by the NRC.

3.1.2.48 (Criterion 48)

Criterion 48 has not been promulgated by the NRC.

3.1.2.49 (Criterion 49)

Criterion 49 has not been promulgated by the NRC.

3.1.2.50 Containment Design Basis (Criterion 50)

Criterion

"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 as required by Paragraph 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of 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."

Design Conformance

The containment structure is designed with a leakage rate shown in Table 1.3-3. The containment is designed to withstand, by a sufficient margin, loads above those that are conservatively calculated to result from a DBA as discussed in Section 6.2.1.

3.1.2.51 Fracture Prevention of Containment Pressure Boundary (Criterion 51)

Criterion

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

Design Conformance

Ferritic materials for the containment structure boundary are specified so that, when the liner, equipment latch, personnel lock, penetrations, and fluid system components, including valves required to isolate the system are exposed to postulated accident, test, operating and normal conditions, the corresponding and resultant stress levels are below the maximum stress level permitted by the crack arrest temperature (CAT) curve of NRL Report 6900 for each applicable correspondence temperature condition.

Nil Ductivity Transition Temperature

Figure 23 of NRL Report 6900 shows the fracture analysis diagram (FAD), which plots stress vs temperature in excess of nil ductility transition temperature (NDTT). The containment structure liner is designed so that no stress exceeds the CAT curve shown in this FAD. This approach is very conservative and ensures that flaws of any size are not propagated to a rapid (i.e., brittle) fracture.

Uncertainties in the determination of NDTT are minimized by using the drop weight test, ASTM E208, for material 5/8 inch or thicker. Charpy V-notch tests are required for all material which form part of the containment structure boundary.

3.1.2.52 Capability for Containment Leakage Rate Testing (Criterion 52)

Criterion

"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."

Design Conformance

The design of the containment structure and related equipment, which are subjected to the containment structure test conditions, as described in Section 6.2.6, allows for conducting periodic integrated leakage rate testing of the containment structure.

3.1.2.53 Provisions for Containment Testing and Inspection (Criterion 53)

Criterion

"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."

Design Conformance

The reactor containment is 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, as discussed in Section 6.2.6 (Containment Leakage Testing).

3.1.2.54 Piping Systems Penetrating Containment (Criterion 54)

Criterion

"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."

Design Conformance

The piping systems penetrating the containment structure are designed to minimize leakage. Containment isolation valves provide the capability to seal most penetrations redundantly; Section 6.2.4 describes the few exceptions in detail. Pressure taps provide the capability to perform a Type C (10 CFR 50 Appendix J) test to measure containment isolation valve leakage rates, as outlined in Section 6.2.4.

3.1.2.55 Reactor Coolant Pressure Boundary Penetrating Containment (Criterion 55)

Criterion

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

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

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

“Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for 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.”

Design Conformance

All lines that are part of the reactor coolant pressure boundary and that penetrate the containment structure are provided with containment isolation valves in accordance with the above criterion, as described in Section 6.2.4. Valves outside the containment structure are located as close as practical to the containment structure. All containment penetrations and isolation valves are protected against possible environmental effects including missiles. The isolation valves are subject to periodic Type C tests (10 CFR 50, Appendix J), as outlined in Section 6.2.4, and, automatic valves, upon loss of actuating power, take the position that provides the greatest safety.

There are no additional requirements for the mechanical design of those lines that are part of the reactor coolant pressure boundary and that penetrate the containment structure beyond those required by the applicable standards and codes.

3.1.2.56 Primary Containment Isolation (Criterion 56)

Criterion

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

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

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

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

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

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

Design Conformance

All lines that connect to the containment atmosphere are provided with containment isolation valves in accordance with the above criterion except for those lines identified otherwise in Section 6.2.4. Valves outside the containment structure are located as close as possible to the containment structure. The isolation valves are subject to periodic Type C tests (10 CFR 50, Appendix J) as outlined in Section 6.2.4 and, upon loss of actuating power, take the position that provides the greatest safety.

Instrument line penetrations are in accordance with Regulatory Guide 1.11 (Section 6.2.4) and have a remote manual isolation valve outside of the containment structure.

3.1.2.57 Closed System Isolation Valves (Criterion 57)

Criterion

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

Design Conformance

All lines connected to closed systems are provided with at least one containment isolation valve located as close as possible to the outside of the containment structure as described in Section 6.2.4. The isolation valves are subject to periodic tests (10 CFR 50, Appendix J) as outlined in Section 6.2.4. These automatic isolation valves, upon loss of actuating power, take the position that provides the greatest safety.

3.1.2.58 (Criterion 58)

Criterion 58 has not been promulgated by the NRC.

3.1.2.59 (Criterion 59)

Criterion 59 has not been promulgated by the NRC.

3.1.2.60 Control of Releases of Radioactive Materials to the Environment (Criterion 60)

Criterion

"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."

Design Conformance

In all cases, the design for radioactivity control is based on:

1. The requirements of 10 CFR 20, 10 CFR 50, and Appendix I to 10 CFR 50 for normal operations and for any transient situation that might reasonably be anticipated to occur.

2. 10 CFR 50.67 dose level guidelines for potential accidents of extremely low probability of occurrence

All release paths, including ventilation and process streams, are monitored and controlled as described in Section 11.5.

The activity level of the radioactive gaseous waste effluents subsequent release through the 375-foot Millstone stack are monitored (Section 11.3.2.4). Under conditions of concurrent fuel failure and steam generator tube leakage, some radioactive gas would be present and suitably controlled in the steam jet air ejector discharge in the condenser air removal system (Section 10.4.2) and in the flow from the steam packing exhauster fan in the turbine generator gland seal and exhaust system (Section 10.4.3). The steam jet air ejector discharge is directed to the Millstone stack while the seal steam packing exhauster fan discharges through the condensate polishing enclosure roof.

Control of liquid waste effluents (Sections 11.2 and 11.5) is maintained by batch processing of all liquids, sampling before discharge, and a controlled rate of release. Liquid effluents are monitored for radioactivity and rate of flow. Radioactive liquid waste system capacities are sufficient to handle any expected transient in the processing of liquid waste.

Solid wastes are prepared for offsite disposal by either compaction or solidification (Section 11.4). Solid waste is prepared for shipment by placement in properly labeled containers that meet applicable NRC and Department of Transportation dose rate requirements as detailed in 10 CFR 71, 49 CFR 170-178, and Section 11.4.

3.1.2.61 Fuel Storage and Handling and Radioactive Control (Criterion 61)

Criterion

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

Design Conformance

Safety related components in the fuel storage and handling system (Section 9.1.3.4) are designed to allow periodic inspection and testing to ensure proper operation. Performance of components important to safety in the radioactive liquid and gaseous waste systems is verified by extensive process fluid analysis and continuous radiation monitoring of gaseous effluents, respectively.

The new and spent fuel storage areas are designed to meet the requirements of 10 CFR 20 in providing radiation shielding for operating personnel during new and spent fuel transfer and storage. The fuel transfer canal and spent fuel pool wall thickness are sufficient to shield adjacent work areas to meet the requirements of 10 CFR 20 for personnel access during actual fuel transfer. Waste storage and processing facilities in the auxiliary building and the waste disposal building are shielded to meet the requirements of 10 CFR 20 for operating personnel. Periodic surveys by radiation protection personnel and continuously operated radiation monitors located in areas selected to afford maximum personnel protection (Section 12.1) ensure that radiation design levels are not exceeded during the operating lifetime of the unit.

New and spent fuel handling systems are designed to preclude gross mechanical failures which could lead to significant radioactivity releases. Floor and equipment drains are provided to collect leakage which might occur from valve stem leakoffs, pump seals, and other equipment, and to transfer the leakage to one of the building sumps for eventual processing by the liquid waste system.

Radiation gases and particulates released from components are collected by the reactor plant aerated vents system. Uncontrolled leakage of radioactive gases and particulates which may leak from spent fuel, radioactive waste, or components containing radioactive fluids is collected and treated by the respective building ventilation filtration system (Section 9.4) or supplementary leakage collection and release system (Section 6.5.1). All discharges from these systems are monitored for radioactivity.

Decay heat from spent fuel is dissipated in the water of the spent fuel pool and subsequently removed by the cooling portion of the fuel pool cooling and purification system (Section 9.1.3). Redundancy of fuel pool cooling and purification system components ensures reliability in controlling the spent fuel pool water temperature. Spent fuel pool cooling system operation is continuously monitored in the main control room where spent fuel pool water temperature is both indicated and alarmed. Special tests are not required because at least one pump and heat exchanger are normally in operation when spent fuel is stored in the spent fuel pool.

The piping connected to the spent fuel pool is designed so that an acceptable water level is maintained in the event of a pipe rupture. Instrumentation to annunciate spent fuel pool water level changes above or below preset levels is provided on the fuel pool control panel in the main control room. Redundancy of makeup water sources ensures adequate supply and availability of makeup to the spent fuel pool even under loss of normal electrical power.

3.1.2.62 Prevention of Criticality in Fuel Storage and Handling (Criterion 62)

Criterion

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

Design Conformance

Criticality is prevented in the new fuel storage racks by a combination of geometry and poison material as described in Sections 9.1.1 and 4.3.2.6.

Criticality is prevented in the spent fuel storage area by the physical separation of fuel assemblies, limits on the enrichment, burnup and decay times of the fuel, the use of full length RCCAs in Region 2, and the use of fixed neutron poisons in Region 1 and 2. Soluble boron in the spent fuel pool water is also credited. Sections 9.1.1 and 9.1.2 discuss criticality prevention in more detail.

3.1.2.63 Monitoring Fuel and Waste Storage (Criterion 63)

Criterion

"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."

Design Conformance

The spent fuel pool water temperature is continuously monitored. Safety related temperature indicators are provided in the control room where alarms are also provided should the water temperature increase above a preset level. Spent fuel pool temperature is also indicated and alarmed at the fuel pool cooling and purification panel (FP) in the fuel building.

Safety related spent fuel pool low water level indicating lights are displayed in the control room. Additionally, alarms are provided in the control room and at the FP in the fuel building should the level increase or decrease beyond preset levels. The spent fuel pool cooling system cooler outlet temperature and flow rate are each monitored at the FP. Should either of these parameters or the component cooling system flow rate from the operating cooler exceed preset values, an individual alarm is actuated at the FP and the fuel pool cooling system "trouble" alarm is actuated in the control room.

In the event of high temperature, low flow, or abnormal spent fuel pool level indication, administrative procedures provide for checking the operating status and integrity of the spent fuel pool and support cooling systems, inspecting for spent fuel pool leakage, and ensuring that corrective measures are taken to restore all system parameters to normal.

The fuel pool demineralizer can be aligned to either the SFP or the RWST. The specific conductivities of the SFP and the RWST (the fuel pool demineralizer influent sources) are monitored and recorded weekly. The fuel pool demineralizer effluent specific conductivity is monitored and recorded monthly. In the event of abnormal conductivity levels in the SFP, RWST, or the effluent of the fuel pool demineralizer, administrative procedures provide for taking additional samples to assist in determining the source and/or cause of the abnormal conductivity levels and for ensuring that corrective measures are taken.

Radiation levels in the area of spent fuel storage are continuously monitored by radiation detectors located around the periphery of the storage areas. Other continuously operating radiation detectors are located in the fuel and waste disposal buildings in areas best suited for alerting operating personnel of high local radiation levels. Radiation levels in excess of the preset values for either the fuel building or waste storage areas initiate alarms, both locally and in the control room.

In the event of a high airborne gross activity alarm, the fuel building ventilation system exhaust (Section 9.4.1) is remote manually diverted to its exhaust filtration system. In the waste disposal building, the ventilation system exhaust (Section 9.4.4) can be remote manually diverted to the auxiliary building filtration system upon receipt of a high airborne activity alarm.

3.1.2.64 Monitoring Radioactivity Releases (Criterion 64)

Criterion

"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."

Design Conformance

The containment atmosphere is monitored during normal and transient operations of the reactor plant by the containment structure particulate and gas monitor located in the upper level of the auxiliary building (Section 12.3.4) or by grab sampling. Normal unit effluent discharge paths are monitored during normal plant operation by the ventilation particulate samples and gas monitors in the auxiliary building and engineered safety buildings (Section 11.5). After a postulated accident, the safety related ventilation vent monitors and the safety related Supplementary Leak Collection and Release System monitors are used to monitor the effluents from spaces contiguous to the containment structure including the areas that contain loss-of-coolant accident fluids. In addition, the service water outlet from each pair of containment recirculation coolers is monitored to ensure that any leakage of radioactive fluids into the service water system is detected (Section 11.5). Radioactivity levels in the environs are controlled during normal and accident conditions by the various radiation monitoring systems (Sections 11.5 and 12.3.4) and monitored by the collection of samples as part of the offsite radiological monitoring program.

3.1.3 REFERENCE FOR SECTION 3.1

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.2.1 SEISMIC CLASSIFICATION

Seismic Category I structures, systems, and components are those necessary to ensure:

1. The integrity of the reactor coolant pressure boundary

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

3. The capability to prevent or mitigate the consequences of accidents which could result in potential off site exposures comparable to the guideline exposure of 10 CFR 100.

Seismic Category I structures, systems, and components are designed to remain functional during a safe shutdown earthquake (SSE). Strain limits in excess of yield are allowed provided safety functions are maintained. Seismic Category I structures, systems, and components are also designed to be well within the elastic limit for a vibratory motion at 50 percent of the SSE. This requirement is called the operating basis earthquake (OBE).

The SSE and the OBE are described in Section 2.5. The seismic design of Seismic Category I systems and components is described in Sections 3.7 and 3.10 and of structures in Section 3.8. Seismic Category I structures, systems, and components are listed in Table 3.2–1.

The seismic classification of structures, systems, and components complies with Regulatory Guide 1.29.

Structures, systems, and components designated Seismic Category I in accordance with NRC Regulatory Guide 1.29 are listed in Table 3.2–1.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION

The containment structure and safety related fluid systems, listed in Table 3.2–1, are classified according to the classes listed below. Supports are in the same safety class as the components for which they provide support if failure of the support could cause a loss of a safety function associated with the supported component.

Safety Class 1

Safety Class 1 (SC-1) applies to reactor coolant pressure boundary components whose failure during normal reactor operation would prevent orderly reactor shutdown and cooldown. The reactor coolant pressure boundary is defined in Paragraph 50.2 of 10 CFR 50 as being all those pressure-containing components "... such as pressure vessels, piping, pumps, and valves, which are:

1. Part of the reactor coolant system, or

2. Connected to the reactor coolant system up to and including any and all of the following:

   a. The outermost containment isolation valve in system piping which penetrates the primary reactor containment.

   b. The second of two valves normally closed during normal reactor operation in system piping which does not penetrate the primary reactor containment.

   c. The reactor coolant system safety and relief valves.

According to Section 50.55a of 10 CFR 50, components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary may be downgraded in safety class provided that:

1. "In the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only, or

2. The component is or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only."

"Shut down and cooled down in an orderly manner" is taken as an action where the pressurizer does not empty, assuming normal makeup is available.

<u>Safety Class 2</u>

Safety Class 2 applies to:

1. Pressure containing components of the reactor coolant pressure boundary not covered in Safety Class 1

2. The reactor containment, including those valves and components of systems used to affect isolation of the reactor containment

3. Components necessary to the system safety function of the following:

   a. Residual heat removal system

b. Those portions of the reactor coolant auxiliary systems that form the reactor coolant letdown and makeup loop

c. Reactor containment cooling spray systems

d. Emergency core cooling system including injection and recirculation modes

e. Containment air purification and cleanup systems used to clean up the containment atmosphere and, thereby, to reduce the radioactivity present in leakage from the containment structure, regardless of whether such systems are located inside or outside the containment structure

f. Portions of the main steam and feedwater systems extending from and including the secondary side of the steam generator up to the first restraint beyond the main steam isolation and feedwater isolation valves in the main steam valve building.

g. Portions of the auxiliary feedwater system

h. Hydrogen recombiner system

Safety Class 3

Safety Class 3 applies to:

1. Components necessary to the safety system function of the following:

a. Portions of the reactor auxiliary systems that provide boric acid for the reactor coolant letdown and makeup loop

b. Portions of the auxiliary feedwater system

c. Portions of the reactor plant component cooling and service water systems that transfer heat from components whose heat removal capability serves a safety function or are required for orderly reactor shutdown. This generally includes portions of the reactor plant component cooling and service water systems serving the emergency core cooling, containment heat removal, and residual heat removal systems; also cooling systems for the control room and safety related electrical equipment. If a heat transfer component is classified as Safety Class 2 or 3 just to maintain a flow path or because its failure would cause uncontrollable release of gaseous radioactivity, then the cooling water lines to and from the component do not have to be assigned a safety class.

d. Fuel pool cooling system

e. On site emergency power supply support systems external to the emergency generators (the emergency generators are defined in IEEE-387-1977)

f. Portions of the main steam system that supply steam to the turbine drive of the steam generator auxiliary feedwater pump

g. Air purification and cleanup systems used to clean up the atmosphere after leakage from the containment structure and other air purification and cleanup systems used after accidents

2. Portions of any system whose failure would result in calculated potential exposure comparable to the guideline exposure of 10 CFR 100. The designation of SC 3 applies to portions of the following systems, the failure of which would result in uncontrollable release to the environment of significant gaseous radioactivity normally held up, and meets the intent of 10 CFR 100:

a. Reactor coolant auxiliary systems that form the reactor coolant letdown and makeup loop not covered by Safety Class 2

b. Portions of the radioactive waste processing and handling systems

Nonnuclear Safety Class

This class applies to portions of the unit not covered in Safety Classes 1, 2, or 3. This class includes most of the steam and power conversion systems, radioactive liquid waste system, and portions of the boron recovery system containing degassed liquid.

Quality Group Classification System

The quality group classification system and its relation to industrial codes conform with Regulatory Guide 1.26 (Section 1.8) with the following exceptions:

1. The safety class terminology of ANSI N18.2-1973 and ANSI N18.2a-1975 is used instead of the quality group terminology. Thus, the terms Safety Class 1, Safety Class 2, Safety Class 3, and non-nuclear safety (NNS) are used instead of Quality Groups A, B, C, and D, respectively.

2. Regarding Regulatory Guide 1.26, Positions C.1.e and C.2.c, one safety valve designed, manufactured, and tested in accordance with ASME III, Division 1 (i.e., a code safety valve) is considered acceptable as the boundary between the reactor coolant pressure boundary and lower safety class or NNS line.

Millstone 3 has constructed components in safety-related systems to the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division I, as follows:

1. Quality Group A (Section III, Class 1) components within the reactor coolant pressure boundary comply with Section 50.55a, 10 CFR 50

2. Quality Group B (Section III, Class 2) components comply with the requirements of Subsection NA-2130 of the code

3. Quality Group C (Section III, Class 3) components comply with the requirements of Subsection NA-2130 of the code except for:

   a. Rubber expansion joints 3SWP*EJ6A, B, C, and D, which are procured in accordance with Appendix B requirements or commercially dedicated.

   b. Valves SWP*V673, SWP*V674, SWP*V23, SWP*V22, SWP*V55, and SWP*V56 which are not N stamped but were procured per the requirements of Generic Letter 89-09.

   c. The service water system supply and return piping for the post accident sample cooler, which is designed to ANSI B31.1 requirements and is seismically qualified.

   d. The service water cubicle sump drain lines which are designed to ANSI 31.1 requirements and are seismically qualified.

The boundary of jurisdiction of ASME Code Section III, Classes 2 and 3 process piping extends to and includes the root valve. The appropriate safety class extends from the root valve to the sensing instrument. Seismic Category I supports are employed for Safety Classes 2 and 3 instrument tubing. The tubing used is one-half inch or less in diameter. The requirements for Safety Classes 2 and 3 instrument tubing are listed in Tables 3.2–2, 3.2–3 and 3.2–4.

The safety classes of safety-related fluid systems are given in Table 3.2–1. In addition, the safety class boundaries are shown on the various piping and instrumentation diagrams (P&IDs) located throughout this safety analysis report. The following line designations are used on P&IDs to indicate these boundaries:

Safety Class 1, SC-1 (Quality Group A) = line designator, - 1

Safety Class 2, SC-2 (Quality Group B) = line designator, - 2

Safety Class 3, SC-3 (Quality Group C) = line designator, - 3

Nonnuclear Safety Class, NNS (Quality Group D) = line designator - 4

The QA classification process for the identification of structures, systems, and components as nuclear safety-related and non safety-related is controlled via an Engineering Design Specification. The methodology for system identifications, system interfaces, line designators, safety class boundaries, and identification of nuclear safety related flow paths, equipment and instrumentation is defined and controlled via an Engineering Design Specification(s).

Safety class boundaries on FSAR figures (P&IDs) are extended to the first piping restraint beyond the indicated boundary.

3.2.3 QUALITY ASSURANCE CATEGORIES

Table 3.2–1 lists Millstone 3 structures, systems, and components which are classified QA Category I.

3.2.4 OTHER CLASSIFICATION SYSTEMS

Tornado Design Classification

The tornado design classification conforms to Regulatory Guide 1.117 (Table 1.8-1). Table 3.2–1 lists the tornado protection criteria for Category I structures, systems, and components. Section 3.8.4.1 describes those structures which are not safety-related but whose failure could reduce to an unacceptable safety level the functional capability of any plant feature included in the items listed in the appendix of Regulatory Guide 1.117.

ASME Code Classes

ASME Code Classes 1, 2, 3, and MC are defined in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1971. With regard to pumps, valves, piping, tanks, and pressure vessels, there is a direct one-for-one correlation between Code Classes 1, 2, and 3, and SC 1, 2, and 3 (Section 3.2.2). Exceptions to this are noted on the engineering document. The Codes for the concrete portions of the containment structure, classified as SC 2, are specified in Section 3.8. Metal containment systems, such as the personnel access lock, are classified as Code Class MC. Paragraphs NE 1100 and NE 1140 of ASME Section III delineate the portions of the containment structure classified as Code Class MC.

The code classes of structures, systems, and components are given in Table 3.2–1.

Engineered Safety Features

Engineered safety features (ESF) are those systems used to directly mitigate the consequences of a major loss-of-coolant accident, up to and including the design basis accident, which is the double ended rupture displacement of the largest pipe in the reactor coolant pressure boundary. The following types of systems are classified as ESF:

- containment systems, including containment structure and containment enclosure building;
- ESF actuation systems;

- emergency core cooling system;
- containment heat removal systems;
- containment combustible gas control systems;
- containment isolation systems; and
- supplementary leak collection and release system.

It should be noted that systems supporting the above (e.g., cooling systems and electrical systems) are not classified as ESF, but are safety-related (QA Category I).

IEEE Classification Systems

IEEE Std-308-1971, IEEE Standard Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations, delineates three classes of structures and equipment (Classes I, II, and III) and one class of electric systems (Class 1E). Class 1E electric systems provide the electric power used to shut down the reactor and limit the release of radioactive material following a design basis event (i.e., postulated events used in the design to establish the performance requirements of the structures and systems). Class II and III electric equipment may be supplied from Class 1E electric systems.

3.2.5 TABULATION OF CODES AND CLASSIFICATIONS

This subsection provides a concise compilation of the safety classes, codes, and design classifications of the structures, systems, and components in Table 3.2–1 that are QA Category I.

QA Category I structures, systems, and components are defined in Table 3.2–1. Seismic Category I structures, systems, and components are defined in Section 3.2.1 and are designed in accordance with the seismic design criteria of Sections 3.7, 3.8, and 3.10. Because the definitions of Seismic Category I and QA Category I are different, the various structures, systems, and components falling in one category do not necessarily fall in the other. QA Category I items which are not Seismic Category I are pointed out in the notes column of Table 3.2–1.

The safety class, as defined in Section 3.2.2, is indicated for each QA Category I structure, system, and component to which it applies. The codes applicable to the QA Category I components are also presented; in addition, Table 3.2–1 indicates which structures, systems, and components are designed for tornado resistance and are protected from tornado effects.

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS

(Symbols and references are defined at the end of this Table)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
STRUCTURES						
Containment Structure						
Reinforced Concrete Substructure	2	ACI 318-71	N/A	CS	P	
Reinforced Concrete Superstructure	2	ACI 318-71	N/A	CS	D	
Containment Enclosure Building	N/A	ACI 318-71	N/A	CS	D	
Containment Enclosure Building	N/A	AISC Steel Construction Manual, 7th Edition	N/A	CEB	N/A	
Reinforced Concrete Interior Shields and Walls	2	ACI 318-71	N/A	CS	P	
Containment Structure Liner	2	ASME III	MC	CS	P	(Note 2)
Piping and Duct Penetrations	2	ASME III	2	CS	P	(Note 2)
Electrical Penetrations	2	IEEE-317	N/A	CS	P	
Personnel Access Lock	2	ASME III	MC	CS	P	(Note 2)
Equipment Hatch	2	ASME III	MC	CS	P	(Note 2) (Note 3)
Containment Dome Closure	2	ASME III	MC	CS	D	(Note 2)
Containment Sump Strainer	2	AISC 6th Edition	N/A	CS	P	
Circulating Water Discharge Tunnel	N/A	ACI 318-71	N/A	OY	D	
Demineralizer Water Storage Tank Enclosure	N/A	ACI 318-71	N/A	OY	D	
Cable Tunnel from Auxiliary Building to Control Building	N/A	ACI 318-71	N/A	SB	D	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Main Steam Valve Building	N/A	ACI 318-71	N/A	MSV	D	
Engineered Safety Features Building	N/A	ACI 318-71	N/A	ESB	D	
Auxiliary Building						
Reinforced Concrete Structure	N/A	ACI 318-71	N/A	AB	D	
MCC and Rod Control Area	N/A	ACI 318-71	N/A	AB	P	
Hydrogen Recombiner Building	N/A	ACI 318-71	N/A	HRB	D	
Fuel Building						
Reinforced Concrete Structure	N/A	ACI 318-71	N/A	FB	D (Note)	Capital new fuel pool, spent fuel pool, pipe tunnel below grade, and the fuel pool cooler and pump area only will be Seismic Category I and protected from tornado missiles.
Steel Roof Structure	N/A	ACI 318-71	N/A	FB	D (Note)	Not missile protected. Designed to withstand tornado winds only (Section 3.8.4.3).
Spent Fuel Pool Liner (including Gates)	N/A	ASME VIII	N/A	FB	P	
Spent Fuel Pool Racks	3	ASME III	3	FB	P	
Control Building	N/A	ACI 318-71	N/A	CB	D	
Emergency Generator Enclosures	N/A	ACI 318-71	N/A	EGE	D	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Circulating and Service Water Pumphouse	N/A	ACI 318-71	N/A	CSP	D (Note)	Service water pump cubicles only.
West Retaining Wall	N/A	ACI 318-71	N/A	CSP	D	
Emergency Generator Fuel Oil Transfer Pump Vault	N/A	ACI 318-71	N/A	OY	D	
SYSTEMS						
Reactor Coolant System						
Reactor Vessel	1	ASME III	1	CS	P	
Reactor Internals, Core Supports	1	N/A	N/A	CS	P	Reactor core supports and internals are designed to the intent of Subsection NG of ASME III.
Reactor Head Vent Piping and Valves	1	ASME III	1	CS	P	
Fuel Assemblies	1	N/A	N/A	CS	P	
Full Length Control Rod Drive Mechanism Housing	1	ASME III	1	CS	P	
Part Length Control Rod Drive Mechanism Housing	1	ASME III	1	CS	P	
CRDM Pressure Vessel	1	ASME III	1	CS	P	
CRDM Latch Assembly	NNS	N/A	N/A	CS	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
CRDM Drive Rod Assembly	NNS	N/A	N/A	CS	P	
Control Rods	1	N/A	N/A	CS	P	
Steam Generator						
Tube Side	1	ASME III	1	CS	P	
Shell Side	2	ASME III	2	CS	P	
Reactor Coolant Stop Valves	1	ASME III	1	CS	P	
Pressurizer	1	ASME III	1	CS	P	
Pressurizer Heaters (Groups A&B)	1	No Code	N/A	CS	P	
Pressurizer Spray Valves and Piping	1	ASME III	1	CS	P	
Reactor Coolant Hot and Cold Leg Piping, Fittings, and Fabrication	1	ASME III	1	CS	P	
Surge Pipe, Fittings, and Fabrication	1	ASME III	1	CS	P	
Loop Bypass Line	1	ASME III	1	CS	P	
Reactor Coolant Thermowell	1	ASME III	1	CS	P	
Reactor Coolant Thermowell Boss	1	ASME III	1	CS	P	
Safety Valves	1	ASME III	1	CS	P	
Relief Valves (PORV)	1	ASME III	1	CS	P	Actuators are qualified to IEEE-323-74.
Power Operated Block Valves	1	ASME III	1	CS	P	Actuators are qualified to IEEE-323-74.
Valves to Reactor Coolant System Boundary	1	ASME III	1	CS	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Control Rod Drive Mechanism Head Adapter Plugs	1	ASME III	1	CS	P	
Reactor Coolant Pump						
Reactor Coolant Pump Casing	1	ASME III	1	CS	P	
Main Flange	1	ASME III	1	CS	P	
Thermal Barrier	1	ASME III	1	CS	P	
No. 1 Seal Housing	1	ASME III	1	CS	P	
No. 2 Seal Housing	2	ASME III	1	CS	P	No. 2 Seal Housing is permitted to be Code Class 2; however, it is supplied as Code Class 1 by Westinghouse.
No. 3 Seal Housing	2	ASME III	2	CS	P	
Pressure Retaining Bolting	1	ASME III	1	CS	P	Reactor coolant pump seal bolting is NSS.
Reactor Coolant Pump Seals	N/A	N/A	N/A	CS	P	Special requirements are included in the specifications.
Reactor Coolant Pump Motor						RCP motor is not safety-related.
Shaft Coupling	2	NEMA MG1	N/A	CS	P	
Spool Piece	2	NEMA MG1	N/A	CS	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Armature	2	No Code	N/A	CS	P	
Flywheel	2	No Code	N/A	CS	P	
Motor Bolting	2	No Code	N/A	CS	P	
Upper Oil Cooler						
Tube Side-Component Cooling Water	3	ASME III	3	CS		
Shell Side-oil	2	ASME III	2	CS	P	
Lower Oil Cooler						
Tube Side-Component Cooling Water	3	ASME III	3	CS	P	
Shell Side-oil	2	ASME III	2	CS	P	
Air-Water Coolers	3	ASME III	3	CS	P	
Piping and Valves***	See Note			CS	P	All piping, valves, and other pressure retaining equipment which are inside or part of the reactor coolant pressure boundary (Section 3.2.2) are SC-1 and ASME III. Other equipment outside the boundary is ASME III 2 or NNS.

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Instrumentation and Controls required to perform safety function in QA Category I portions of system***	N/A	IEEE-279-71	N/A	CS / CR / IRR / MRC	P	
Inadequate Core Cooling Instrumentation	N/A	IEEE-323-74	N/A	CS / CR	P	
		IEEE-344-75				
Subcooled/Superheat Margin Monitor						
Core Exit Thermocouple						
Heated Junction Thermocouple						
Core Exit Thermocouple						
Pressure Boundary/Reactor Internal Modifications for Inadequate Core Cooling Instrumentation	1	ASME III	1	CS	P	
Supports for QA Category I Components*	Same as component being supported.					
Chemical and Volume Control System						
Regenerative Heat Exchanger**						
Tube Side	2	ASME III	2	CS	P	
Shell Side	2	ASME III	2	CS	P	
Letdown Heat Exchanger**						
Tube Side	2	ASME III	2	AB	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Shell Side	3	ASME III	3	AB	P	
Mixed Bed Demineralizer***	3	ASME III	3	AB	P	
Cation Bed Demineralizer**	3	ASME III	3	AB	P	
Reactor Coolant Filter**	2	ASME III	2	AB	P	
Volume Control Tank**	2	ASME III	2	AB	P	
Centrifugal Charging Pump**	2	ASME III	2	AB	P	
Seal Water Injection Filter**	2	ASME III	2	AB	P	
Letdown Orifices**	2	ASME III	2	CS	P	
Excess Letdown Heat Exchanger**						
Tube Side	2	ASME III	2	CS	P	
Shell Side	3	ASME III	3	CS	P	
Seal Water Return Filter**	2	ASME III	2	AB	P	
Seal Water Heat Exchanger**						
Tube Side	2	ASME III	2	AB	P	
Shell Side	3	ASME III	3	AB	P	
Letdown Filter	3	ASME III	3	AB	P	
Boric Acid Tanks *	3	ASME III	3	AB	P	
Boric Acid Transfer Pump**	3	ASME III	3	AB	P	
Boric Acid Blender**	3	ASME III	3	AB	P	
Boric Acid Filter**	3	ASME III	3	AB	P	
Boric Acid Tank Orifice*	3	ASME III	3	AB	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Instrumentation and Controls	See Note			AB / CS	P	Pressure retaining portions of inline instruments are the same as connecting piping.
Moderating Heat Exchanger**						
Tube Side	3	ASME III	3	AB	P	
Shell Side	3	ASME III	3	AB	P	
Letdown Chiller Heat Exchanger**						
Tube Side	3	ASME III	3	AB	P	
Shell Side	NNS	ASME VIII	N/A	AB	P	
Letdown Reheat Heat Exchanger**						
Tube Side	2	ASME III	2	AB	P	
Shell Side	3	ASME III	3	AB	P	
Thermal Regeneration Demineralizer**	3	ASME III	3	AB	P	
Piping and Valves inside RCPB* No. 1 Seal Water Injection Lines	1	ASME III	1	CS	P	Figures 9.2–5 (P&ID 105), 9.3–7 (P&ID 103), and 9.3–8 (P&ID 104) delineate SC boundaries.
Piping and Valves outside RCPB*	See Note					Same references as inside RCPB.

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Letdown and charging lines	2	ASME III	2	AB / CS	P	Charging pump alternate minimum flow lines downstream of isolation valves 3CHS*MV8512 A/B to the RWST are ANS Safety Class 4.
Seal water injection and return lines	2	ASME III	2	AB / CS	P	
Mixed-bed and cation-bed lines	3	ASME III	3	AB	P	
Boric acid lines	3	ASME III	3	AB	P	
Thermal regeneration lines	3	ASME III	3	AB	P	
Emergency boration	2	ASME III	2	AB	P	
Supports for QA Category I Components*	Same as component being supported.					
<u>Residual Heat Removal System</u>						
Residual Heat Removal Pump**	2	ASME III	2	ESB	P	
Residual Heat Exchanger**						
Tube Side	2	ASME III	2	ESB	P	
Shell Side	3	ASME III	3	ESB	P	
Piping and Valves inside RCPB*	1	ASME III	1	CS / ESB	P	Figure 5.4–5 delineates SC boundaries.
Piping and Valves outside RCPB*	2	ASME III	2	CS / ESB	P	Figure 5.4–5 (P&ID 112) delineates SC boundaries.

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Valve interlocks for over-pressurization protection	N/A	IEEE-279-71	N/A	CS / ESB / IRR / ESR / CR	P	Section 7.6.2.2 for criteria comments.
Supports for QA Category I	Same as components being supported.					
Emergency Core Cooling System						
Accumulators**	2	ASME III	2	CS	P	
Safety Injection Pumps**	2	ASME III	2	ESB	P	
Piping and Valves inside RCPB*	1	ASME III	1	CS	P	Figures 5.4–5 (P&ID 112), 6.3.2 (P&ID 113), and 9.2.4 (P&ID 114) delineate SC boundaries.
Piping and Valves outside RCPB*						Figures same as inside RCPB.
High and low pressure safety injection lines	2	ASME III	2	CS / ESB	P	
Supports for QA Category I Components *	Same as component being supported.					
Hydrogen Recombiner System*						
Hydrogen Recombiner Blower	2	ASME III	2	HRB	P	
Preheater Coil	2	ASME III	2	HRB	P	
Air Cooler	2	ASME III	2	HRB	P	
Thermal Recombiner	2	ASME III	2	HRB	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Piping and Valves	2	ASME III	2	CS / HRB	P	Figure 6.2–36 (P&ID 115) delineates SC boundaries.
Instrumentation and Controls required to perform safety function	N/A	IEEE-279-71	N/A	HRB	P	
Supports for QA Category I Components *	Same as component being supported.					
Quench Spray System *						
Refueling Water Storage Tank (RWST)	2	ASME III	2	OY	N/A	
Quench Spray Pumps	2	ASME III	2	ESB	N/A	
Piping and Valves, excluding RWST recirculation lines and test piping (NNS)	2	ASME III	2	ESB / CS	N/A	Figure 6.2–36 (P&ID 115) delineates SC boundaries.
Instrumentation and Controls required to perform safety function in QA Category I portions of system	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	OY / ESB / IRR / ESR / CR	P	
Supports for QA Category I Components*	Same as component being supported.					
Containment Recirculation System*						
Containment Recirculation Pumps	2	ASME III	2	ESB	P	
Containment Recirculation Coolers						

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Tube Side	3	ASME III	3	ESB	P	
Shell Side	2	ASME III	2	ESB	P	
Piping and Valves, excluding test piping	2	ASME III	2	ESB / CS	P	Figure 5.4–5 (P&ID 112) delineates SC boundaries.
Instrumentation and Controls required to perform safety function in QA Category I portions of system	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	CS / E / IRR / ESR / ESB / CR	P	
Supports for QA Category I Components	Same as component being supported.					
Emergency Generator Fuel Oil System*						
Emergency Generator Fuel Oil Day Tanks	3	ASME III	3	EGE	P	
Emergency Generator Fuel Oil Transfer Pumps	3	ASME III	3	OY	P	Located underground in emergency generator fuel oil transfer pump vault.
Emergency Generator Fuel Oil Storage Tanks	3	ASME III	3	OY	P	Underground vault.
Piping and Valves	3	ASME III	3	OY / EGE	P	Figure 9.5.2 (P&ID 117) delineates SC boundaries.
Instrumentation and Controls required to perform safety function.	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	EGE / OY / IRR / ESR / CR	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Supports for QA Category I Components*	Same as component being supported.					
Emergency Diesel Engine Air Start System*						
Air Receiver Tanks	3	ASME III	3	EGE	P	
Piping and Valves	3	ASME III	3	EGE	P	
Instrumentation and Controls required to perform safety function	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	EGE / OY / IRR / ESR / CR	P	
Supports for QA Category I Components*	Same as component being supported.					
Emergency Diesel Engine Jacket Water Cooling System*						
Fresh Water Expansion Tank	3	ASME III	3	EGE	P	
Piping and Valves	3	ASME III	3	EGE	P	
Instrumentation and Controls required to perform safety function	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	EGE	P	
Supports for QA Category I Components*	Same as component being supported.					
Emergency Diesel Engine Exhaust and Combustion Air System*						

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Air Piping	3	ASME III	3	EGE	P (Note)	Emergency Diesel Exhaust Piping is not protected for tornado See Section 9.5.8.3
Supports for QA Category I Components*	Same as component being supported.					
Emergency Diesel Engine Lube Oil System*						
Piping	3	Mfgr Std	See Note	EGE	P	Equivalent to ANSI B31.1. See NRC Question Q430.73.
Valves	3	ASME III	3	EGE	P	
Lube Oil Heat Exchanger 3EGO*E3A,B	3	ASME III	3	EGE	P	
Lube Oil Strainers 3EGO*STR1A,B 3EGO*STR2A,B	3	ASME III See Note	3	EGE	P	Additional strainer to be purchased/installed ASME III, or if not available, ASME VIII.
Lube Oil Filter 3EGO*FLT1A	3	ASME III	3	EGE	P	
Heater 3EGO*H1A,B	3	ASME VIII	N/A	EGE	P	
Lube Oil and Pre-Lube Pump 3EGO*P3A,B 3EGO*P4A,B	3	Mfgr Std	N/A	EGE	P	ASME material

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Rocker Arm Lube Oil System*						
Rocker Arm Lube Oil and Prelube Pump	3	Mfgr Std	N/A	EGE	P	
3EGO*P2A,B						
3EGO*P1A,B						
Piping	3	Mfgr Std	See Note	EGE	P	Equivalent to ANSI B31.1. See NRC Question Q430.73.
Valves	3	ANSI B16.5	N/A	EGE	P	
Filter	3	Mfgr Std	N/A	EGE	P	
3EGO*FLT2A,B						
Fuel Pool Cooling and Purification System*						
Fuel Pool Cooling Pumps	3	ASME III	3	FB	P	
Fuel Pool Coolers	3	ASME III	3	FB	P	
Piping and Valves required for cooling	3	ASME III	3	FB	P	Figure 9.1–6 (P&ID 111) delineates SC boundaries.
Service Water Piping for Emergency Makeup to Fuel Pool	3	ASME III	3	FB	P	Figure 9.1–6 (P&ID 111) delineates SC boundaries.
Instrumentation and Controls required to perform safety function in QA Category I portions of system	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	FB / CR / ESR	P	Pressure retaining portions of inline instruments are the same as connecting piping.

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Supports for QA Category I Components	Same as component being supported.					
Containment Isolation System*						
Containment isolation valves and associated piping for all system penetrating containment structure.	2	ASME III	2	CS / FB / MSV / AB / ESB / HRB	P	Individual fluid system figures delineate SC boundaries.
Instrumentation and Controls required to perform safety function	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	CS / FB / MSV / AB / CR / IRR / ESB / HRB	P	
Supports	Same as component being supported.					
Service Water System*						
Service Water Pumps	3	ASME III	3	CSP	P	
Service Water Strainer	3	ASME III	3	CSP	P	
Piping and Valves supplying cooling water to QA Category I equipment	3	ASME III	3	AB / ESB / EGE / OY (Buried) / CR / CSP	P (Note)	A. Figure 9.2–1 (P&ID 133) delineates SC boundaries. B. Service Water vents above the ESF roof are not protected for tornado See Section 9.2.1.3

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Instrumentation and Controls required to perform safety function in QA Category I portions of system.	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	CSP / ESB / EGE / CR / IRR / ESR	P	
Supports for QA Category I Components	Same as component being supported.					
Reactor Plant Component Cooling Water Subsystem*						
Component Cooling Pumps	3	ASME III	3	AB	P	
Component Cooling Surge Tank	3	ASME III	3	AB	P	
Component Cooling Heat Exchangers	3	ASME III	3	AB	P	
Piping and Valves supplying cooling water to QA Category I equipment	3	ASME III	3	AB / CS / ESB / FB	P	Figure 9.2–2 (P&ID 121) delineates SC boundaries.
Instrumentation and Controls required to perform safety function in QA Category I portions of system	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	AB / CS / CR / IRR / ESB / ESR / FB	P	
Supports for QA Category I Components	Same as component being supported.					
Neutron Shield Tank Cooling System*						

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Neutron Shield Tank	1	N/A	N/A	CS	P	Classified as SC-1 because portions support reactor vessel, not because of shielding function.
Charging Pumps Cooling System*						
Charging Pumps Cooling Pumps	3	ASME III	3	AB	P	
Charging Pumps Coolers	3	ASME III	3	AB	P	
Charging Pumps Surge Tank	3	ASME III	3	AB	P	
Piping and Valves	3	ASME III	3	AB	P	
Instrumentation and Controls required to perform safety function	N/A	IEEE-279-71 IEEE-336-71 IEEE-323-74	N/A	AB / CR / IRR	P	
Supports for QA Category I Components*	Same as component being supported.					
Safety Injection Pumps Cooling System*						
Safety Injection Pumps Coolers	3	ASME III	3	ESB	P	
Safety and Injection Pumps Cooling Pumps	3	ASME III	3	ESB	P	
Safety Injection Pumps Surge Tank	3	ASME III	3	ESB	P	
Piping and Valves	3	ASME III	3	ESB	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Instrumentation and Controls required to perform safety function	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	ESB / CR / IRR	P	
Supports for QA Category I Components*	Same as component being supported.					
Fuel Handling System						(Note 3)
Reactor Vessel Head Lifting Device	1 & NNS	N/A	N/A	CS	P	Portions that furnish support to Control Rod Drive Mechanism are SC-1.
Refueling Machine	NNS	CMMA Spec #70	N/A	CS	P	
Spent Fuel Shipping Cask Trolley	NNS	CMMA Spec #70	N/A	FB	P	
Spent Fuel Assembly Handling Tool	NNS	N/A	N/A	FB	P	
Spent Fuel Pit Bridge and Hoist	3	N/A	N/A	FB	P	
Fuel Transfer Tube and Flange	2	ASME III	N/A	FB / CS	P	Portions of Containment Boundary
Fuel Basket	3	N/A	N/A	FB / CS	P	Protects fuel during transportation from damage.
Drive Mechanism and Controls	NNS	N/A	N/A	FB / CS	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Supports for QA Category I Components*	Same as component being supported.					
Main Steam System						
Main Steam Piping and Valves from steam generators up to and including main steam isolation trip valves and isolation valves in main steam supply lines to auxiliary feed pump turbines*	2	ASME III	2	CS / ESB / MSV	P (Note)	A. Figure 10.3–1 (P&ID 123) delineates SC boundaries. B. Main Steam Vents above the MSVB roof are not protected for tornado See Section 10.3.3
Main Steam Piping and Valves from isolation valves to the steam generator auxiliary feed water pump turbine *	3	ASME III	3	ESB	P (Note)	A. Figure 10.3–1 (P&ID 123) delineates SC boundaries. B. TDAFW vent above the ESF roof is not protected for tornado See Section 10.3.3
Main Steam Safety Valves	2	ASME III	2	MSV	P	
Main Steam Pressure Relieving Valves and Bypass Valves	2	ASME III	2	MSV	P	Actuators are qualified to IEEE-323-74.
Main Steam Piping from main steam isolation valves to turbine building	3	ASME III	3	MSV	P	Figure 10.3–1 (P&ID 123) delineates SC boundaries.
Main Steam Flow Restrictors	2	ASME III	2	CS	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Instrumentation and Controls required to perform safety function in QA Category I portions of system***	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	MSV / CR / IRR	P	
Supports for QA Category I Components*	Same as component being supported.					
Auxiliary Feedwater System*						
Demineralized Water Storage Tank (DWST)	3	ASME III	3	OY	P (Note)	Vent above the DWST enclosure is not protected for tornado See Section 10.4.9.3
Steam Generator Auxiliary Feedwater Pumps (Turbine- and Motor-Driven)	3	ASME III	3	ESB	P	
Piping and Valves supplying auxiliary feedwater from DWST to steam generator auxiliary feedwater isolation valves	3	ASME III	3	OY / ESB	P	Figure 10.4–6 (P&ID 130) delineates SC boundaries.
Piping and Valves from steam generator auxiliary feedwater isolation valves to steam generator feedwater lines	2	ASME III	2	ESB / CS	P	Figure 10.4–6 (P&ID 130).

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Instrumentation and Controls required to perform safety function in QA Category I portions of system	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	ESB / CR / IRR / OY / CS	P	
Supports for QA Category I Components	Same as component being supported.					
Feedwater System						
Steam Generator Feedwater Piping and Valves inside containment structure up to and including the first restraint beyond isolation valve outside containment structure *	2	ASME III	2	MSV / CS	P	Figure 10.4–6 (P&ID 130) delineates SC boundaries.
Feedwater Piping and Valves from the first restraint beyond isolation valve outside containment to turbine building wall	3	ASME III	3	MSV	P	Figure 10.4–6 (P&ID 130) delineates SC boundaries.
Instrumentation and Controls required to perform safety function in QA Category I portions of system***	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	MSV / CR / CS	P	
Supports for QA Category I Components*	Same as component being supported.					
Steam Generator Blowdown System						

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Piping and Valves from steam generator to containment isolation valves	2	ASME III	2	CS / MSV	P	Figure 10.3–1 (P&ID 123) delineates SC boundaries.
Reactor Plant Sampling Systems*						
Piping and Valves for reactor coolant loop, pressurizer, and safety injection accumulator sampling up to and including first isolation valve outside containment structure	2	ASME III	2	CS / AB	P	Figure 9.3–2 (P&ID 144) delineates SC boundaries.
Instrumentation and Controls in QA Category I portions of system	N/A	IEEE-279-71 IEEE-323-74 IEEE-336-71	N/A	CS / AB/ CR / IRR	P	
Sample lines originating from safety-related components, up to and including remotely operated sample selection valve or second manual isolation valves	2 or 3	ASME III	2 or 3	AB / ESB	P	Figure 9.3–2 (P&ID 144) delineates SC boundaries.
Supports for QA Category I Components*	Same as component being supported.					
Reactor Plant Aerated Drains						

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Instrumentation and Control (sump level indication) required to provide leak detection	N/A	IEEE-279-71 IEEE-336-71 IEEE-323-74	N/A	AB / ESB	P	
Porous Concrete Groundwater Sump, Piping & Containment Recirculation Cubicle Sumps	3	ANSI B31.1	N/A	ESB	P (Note)	A. Figure 9.3–6 (P&ID 106D) delineates SC boundaries B. SRW standpipe above the ESF roof is not protected for tornado
Containment Monitoring System						
Instrumentation and Controls required to monitor hydrogen concentration	N/A	IEEE-279-71 IEEE-336-71 IEEE-323-74	N/A	CS	P	
Heating, Ventilation, and Air Conditioning System*						
ESF Building						
All air-conditioning systems	3	SMACNA ARI, AMCA	ASME III	ESB	P	ASME III, Class 3 for service water side of refrigeration condenser.

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Emergency Ventilation System for mechanical equipment room and auxiliary feedwater pump room	3	SMACNA, AMCA	N/A	ESB	P	
Emergency Generator Enclosure						
Ventilation (except normal exhaust fan)	3	SMACNA, AMCA	N/A	EGE	P	
Control Building						
All air-conditioning systems, including control building chilled water system	3	SMACNA, AMCA	ASME III	CB	P	ASME III, Class 3 for chilled water and service water pressure-containing components.
Battery rooms ventilation and chiller room ventilation (excluding the control room kitchenette and toilet exhausts)	3	SMACNA, AMCA	N/A	CB	P	
Control Room Pressurization						Control room post-accident dose analyses do not credit this system or its components.
Piping	3	ANSI B31.1	N/A	CB	P	
Tanks	3	ASME VIII	N/A	CB	P	
Valves	3	ANSI B31.1	N/A	CB	P	
Control Room Emergency Ventilation	3	Note A	N/A	CB	P	See ESF Filter Systems, below.

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Auxiliary Building						
Charging pumps and service water pumps cubicles ventilation (excluding auxiliary building exhaust filtration system)	3	SMACNA, AMCA	N/A	AB	P	
Auxiliary building exhaust filtration	3	Note A		AB	P	See ESF Filter Systems, below.
MCC and rod control area air-conditioning	3	SMACNA, AMCA	ASME III	AB	P	ASME III, Class 3 for service water coils.
Hydrogen Recombiner Building						
Hydrogen recombiner cubicles ventilation	3	SMACNA, AMCA	N/A	HRB	P	
Main Steam Valve Building						
Main Stream Valve building ventilation (except normal exhaust fans)	3	SMACNA, AMCA	N/A	MSV	P	
Yard Structures						
Service Water Pumphouse ventilation	3	SMACNA, AMCA	N/A	CSP	P	
Fuel Building						

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Fuel Building Filtration System	3	Note A		AB	P	A. See ESF Filter Systems, below. This system is not credited for post-accident analyses.
Supplementary Leak Collection and Release System (SLCRS)	3	Note A		AB	(Note B)	B. Only portions inside tornado protected buildings.
Instrumentation and Controls required to perform safety function for QA Category I ventilation systems	N/A	IEEE-279-71 IEEE-336-71	N/A	ESB / EGE / CB / CSP / AB	P	Only portions required to mitigate effects of LOCA will be Seismic Category I.
Supports for QA Category I Components*	Same as component being supported.					
Fuel Building						
Fuel Building Exhaust Filtration	3	See Note				See ESF Filter Systems, below. This system is not credited for post-accident analyses.
ESF Filter Systems and SLCRS						
Fans	3	AMCA	N/A	Note C	Note C	C. For location and tornado criteria, see individual system listing.
Motors	3	NEMA, IEEE-323, 344, 334	N/A	Note C	Note C	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Filter Trains	3	ANSI N509	N/A	Note C	Note C	
Duct	3	SMACNA	N/A	Note C	Note C	
Dampers	3	AMCA	N/A	Note C	Note C	
Filter segment drainage up to and including the isolation valves	3	ANSI B31.1	N/A	AB	Note C	Applicable only to Auxiliary Building Ventilation Filters, Fuel Building Ventilation Filters and SLCRs.
Electrical Systems*						
Emergency Generators, including Auxiliaries	N/A	IEEE-308-74 IEEE-323-74	N/A	EGE	P	
Unit Batteries and Chargers	N/A	IEEE-308-74 IEEE-323-74	N/A	BR	P	
Vital Bus and Inverters	N/A	IEEE-308-74 IEEE-323-74	N/A	ESR	P	
Emergency Unit Substations	N/A	IEEE-308-74 IEEE-323-74	N/A	ESR / MAC	P	
Emergency Station Service Switchgear	N/A	IEEE-308-74 IEEE-323-74	N/A	ESR	P	
Emergency Motor Control Centers	N/A	IEEE-308-74 IEEE-323-74	N/A	ESR / MRC / EGE / CSP / ESB	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Electric Motors-Pumps	N/A	IEEE-334-74	N/A	Misc	P	(Note 1)
Control Panelboards	N/A	IEEE-308-74	N/A			
Main control board and panels with safety related functions				CR	P	
Diesel engine driven emergency generators panel				RCR	P	
Radiation monitor panel				CR	P	
Auxiliary shutdown panel				ESR	P	
Air-conditioning control panel				CR	P	
Hydrogen recombiner control panel				HRB	P	
Trays, Conduits, and Ducts Carry Safety Related Wiring	N/A	N/A	N/A	CS / CSP / AB / CB / FB / EGE / ESB / HRB OY / MSV / MRC	P	
Class 1E AC Instrumentation, Control and Power Cables - Essential Buses (orange/purple trains)	N/A	IEEE-323-74 IEEE-383-74	1E	CS /CSP AB / ESF / CB / FB / HRB / OY / MSV/ MRC / EGE	P	
Class 1E DC Power Cables	N/A	IEEE-323-74 IEEE-383-74	1E	CS/ CSP / AB / ESF / CB / FB / MSV / EGE / HRB	P	

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
Class 1E dc Switchgear, Distribution Panels, and Protective Relays	N/A	IEEE-323-74 IEEE-344-75	1E	CB / CSP	P	
Emergency Lighting Battery Pack Supports	N/A	Mfgr Std	See Note	CS / AB CSP / ESF / FB / HRB / MSV / CB / EGE	P	Seismic Category I only.
<u>Reactor Trip System</u> ***						
All portions of reactor trip system which must operate to safely shut down reactor to hot subcritical condition	N/A	IEEE-279-71	N/A	CS / CR / IRR	P	Includes instrumentation and control components from and to turbine impulse pressure transmitters.
<u>Incore Instrumentation System</u>***						
Instrumentation and Conduit Tubes	1	ASME III	1 (Note)	CS	P	Design Basis allowed use of Subsection NC 3600
Bottom Mounted Instrumentation Thimble Tubes	2	Mfgr. Std.	N/A	CS	P	
<u>Engineered Safety Features Actuation System</u> ***	N/A	IEEE-279-71	N/A	CS / CR	P	
<u>Miscellaneous</u> *						
Containment Structure Polar Cran	N/A	N/A	N/A	CS	P	Designed for earthquake in unloaded condition.

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes

LEGEND

General Symbols

N/A - Not applicable

SC - Safety class

NNS - Nonnuclear safety

QA Category - Quality assurance category

RCPB - Reactor coolant pressure boundary

* - SWEC Scope of Supply

** - WNES Scope of Supply

*** - Scope of Supply shared between WNES and SWEC

Tornado Criteria Symbols

P - Protected from tornado effects by a structure or because below grade

D - Designed to withstand tornado effects

Location Symbols

AB - Auxiliary building

BR - Battery room (in control building)

CB - Control building

CR - Control room (in control building)

CS - Containment structure

CEB - Containment enclosure building

CSP - Circulating and service water pumphouse

EGE - Emergency generator enclosure

ESB - Engineered safety features building

ESR - Emergency switchgear room (in control building)

FB - Fuel building

HRB -Hydrogen recombiner building

IRR -Instrument rack room (in control building)

MRC -MCC and rod control area (in auxiliary building)

MSV -Main steam valve building

OY -Outside, yard

SB -Service building

TB -Turbine building

TABLE 3.2–1 LIST OF QA CATEGORY I AND SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

	ANS Safety Class	Code (Note 1)	Code Class	Location	Tornado Criterion	Notes
			WDB -Waste disposal building			

NOTES:

Note 1. The mechanical system components satisfy the codes and addenda (ASME Section III, Division 1) in effect at the time of component order.

Note 2. There was no applicable code for the design of concrete containment structure liners at the construction of the Millstone 3 liner. However, ASME Sections III and VIII, 1971 Edition, were used as guides. See Section 3.8.1.2.3.

Note 3. Protection “p” is provided by the tornado missile shield blocks. Based on a probabilistic analysis, installation of the tornado missile shield blocks is not required during Modes 5 and 6.

Note 4. This FSAR table identifies safety-related pumps for a given system. Unless otherwise indicated, motors for these safety-related pumps are also safety-related and included under the same safety class.

TABLE 3.2–2 SAFETY CLASSES 2 AND 3 INSTRUMENT TUBING REQUIREMENTS

	ASME III (All Ref. Summer 1973)	QA Category I Program
Organization	Required	Same
Training	Required	Same
Design Specification	Pressure boundary integrity for SSE and dead load, thermal	Same, except for code references to certification in specification [1]
Engineering, Design, and Document Control	Category I	Same
Procurement Control	ASME-approved suppliers	ASME III design and materials, Category I supplier, no N-stamp required
Receiving, Inspection, Identification, Storage, and Handling Control	Physical inspection and review of documentation; ANSI storage and material identification	Same
Fabrication and Installation Control	Control Drawing Package, FQC, and ANI review and established holdpoints; material traceability	Same, except no mandatory holdpoints; no third party documentation review; no individual packages per drawing; normal Category I IR System (Tables 3.2–3 and 3.2–4)
Field Welding and Brazing Control	ASME III Procedures - Weld data package each weld; ASME IX welders [2]	Same (Table 3.2–4)
Bolted and Other Mechanical Joints	Data sheet for special bolted joints	No special bolted joints; mechanical fittings installed to MFG requirements; documented on inspection report
Tubing Supports	Design to AISC, 7th Edition (Refer to NRC Question 210.36 for Details)	Same

TABLE 3.2–2 SAFETY CLASSES 2 AND 3 INSTRUMENT TUBING REQUIREMENTS (CONTINUED)

	ASME III (All Ref. Summer 1973)	QA Category I Program
Heat Treatment and Special Operations and Repairs	Not Applicable	Same
Fabrication and Installation Inspection	FQC, ANI, ASME acceptance; material traceability required of selected components to specific point of installation	Same, except limited third party surveillance; Category I material marking or exclusive purchase of Category I material (Tables 3.2–3 and 3.2–4)
Nondestructive Testing	Dye penetrant for Class 2, visual for Class 3; traceability (2)	Same
Nonconformances	N&D	Same
Control of Measuring and Test Equipment	Required	Same
Authorized Nuclear Inspector and Code Certification	N-stamp	Not Applicable
Quality Assurance Audit Program	SWEC, ASME, ANI, NUSCO	SWEC, NUSCO
Company Quality Assurance and Control Manual	SWEC QA Program Manual SWEC ASME III Control Manual	SWEC QA Program Manual
Final Documentation	As-built data package FQC-ANI Certification	Documented on inspection reports; FQC acceptance
Certificate Holder (Installation Subcontractor)	Not Applicable	Same
Pressure Testing	1.25 times design pressure FQC/ANI to witness	Same
SWEC's Responsibilities when Owner's Designee	Prepare code data forms, N5, N3, ANI witness N-stamp	Not Applicable

TABLE 3.2–2 SAFETY CLASSES 2 AND 3 INSTRUMENT TUBING REQUIREMENTS (CONTINUED)

	ASME III (All Ref. Summer 1973)	QA Category I Program
SWEC Operations under ASMI Section XI	Governs repair of components	Not Applicable

NOTES:

1. For tubing which serves a nonsafety-related function attached to a Category I process pipe, functional capability is not required; therefore, the allowable stress values for ASME equation 9 utilize the faulted allowable of 2.4 Sh for all loading conditions. The faulted allowable for this tubing will ensure that the pressure boundary is maintained, thereby protecting the safety-related function of the process piping.
2. Compression fittings are used exclusively for tubing installation except where transition from pipe to tubing is required. The pipe-to-tubing transition is controlled via piping specification requirements as part of the ASME III piping system.

TABLE 3.2–3 INSTRUMENT TUBING EXAMINATION AND TESTING PROGRAM

A. Safety Classes 2 and 3 Socket and Butt Welds

1. One hundred percent visual inspection by the construction department prior to release to Field Quality Control (FQC) - (document via construction checklist).
2. One hundred percent FQC inspection using ASME III NDE procedures [(1)] (document via IR).
3. Third party inspector (ANI) witness NDE, percentage as determined by ANI.
4. In-process surveillance inspections performed by FQC - (document via IR).
5. One hundred percent pressure tested per ASME III pressure test requirements with 100 percent visual inspection of welds - (document via Pressure Test Report).

B. Safety Classes 2 and 3 Compression Fittings

1. One hundred percent of fitting make-up by construction prior to release to FQC using vendor's recommended practices and inspection tools (document via construction checklist).
2. One hundred percent of fitting make-up by FQC using vendor's recommended practices and inspection tools (document via IR).
3. In-process surveillance inspection performed by FQC (document via IR).
4. One hundred percent pressure tested per QA Category I pressure test requirements with 100 percent visual inspection of fittings (document via Pressure Test Report).

NOTE:

1. For Class 2, LP is required, for Class 3, visual inspection is required.

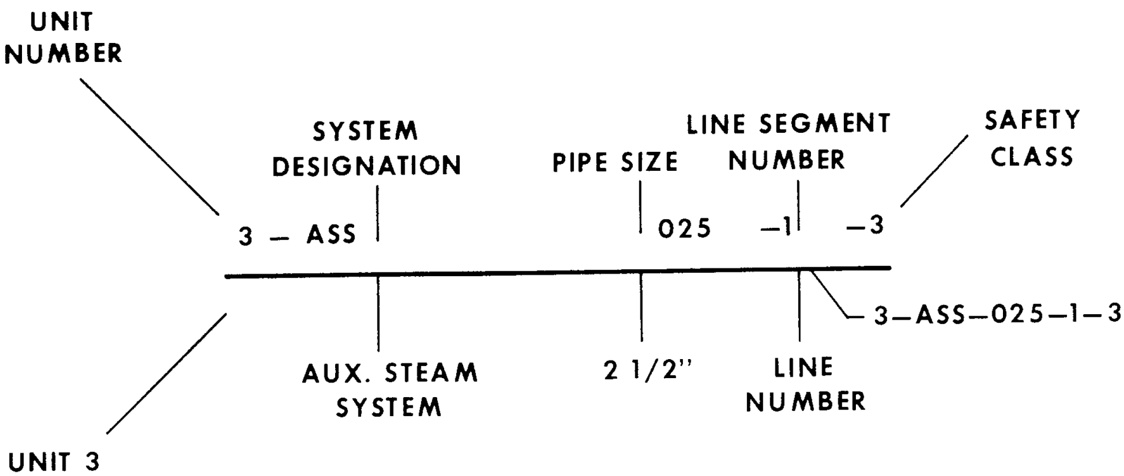
TABLE 3.2–4 COMPARISON OF PROPOSED TUBING EXAMINATION AND TESTING WITH ASME III REQUIREMENTS

ASME Program Classes 2 and 3	QA Category I Exam/Testing Program
1. One hundred percent visual inspection.	1. Same
2. One hundred percent LP inspection (1)	2. Same (1)
3. Surveillance by ASME and ANI; approximately 10 percent in process activities	3. Surveillance by a third party inspector; approximately 10 percent in-process activities - includes welding and weld hydros
4. Hydro - 100 percent inspection by FQC and ANI at 1.25 times design pressure	4. Pressure test (Hydro or Pneumatic) -100 percent inspection by FQC at 1.25 times design pressure; ANI to witness.
5. Surveillance inspection performed by FQC; i.e., In process Welding, Weld material Control, Material Control	5. Same
6. All inspection performed, with the exception of Item 5, are documented in the weld data packages; i.e., Weld Data Sheets	6. Same
7. Welders and procedures to be qualified to ASME IX	7. Same

NOTE:

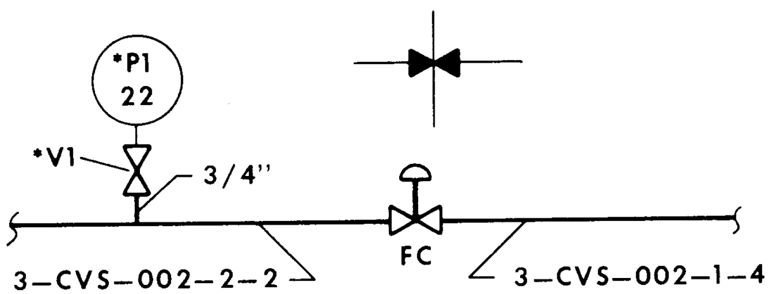
1. For Class 3, LP is not required by ASME III.

FIGURE 3.2–1 LINE DESIGNATION, SAFETY CLASS BOUNDARIES



LINE NUMBER –3ASS025001–3
REFER TO SECTION 6.3.1

LINE DESIGNATION



SAFETY CLASS CHANGE

3.3 WIND AND TORNADO LOADINGS

3.3.1 WIND LOADINGS

This section discusses the design wind load on Seismic Category I structures.

3.3.1.1 Design Wind Velocity

The maximum wind experienced in the vicinity of the Millstone site was associated with a 1960 hurricane. According to the preliminary site evaluation for Millstone 1 (TRC 1965), the Montauk Point Coast Guard Station on the tip of Long Island, across Long Island Sound from Millstone Point, recorded a wind speed of 115 mph with gusts to 140 mph. This wind speed is greater than the 1,000 year mean recurrence interval wind speed for the New Haven area (Hollister 1970). Velocity profiles are as given in ANSI A58.1 - 1972 (ANSI 1972).

All Seismic Category I structures are designed to remain elastic under the 115 mph basic wind speed. The gust factors are included in the determination of effective velocity pressures as described in ANSI A58.1. Because the site is considered to be flat, open country, the 1/7 power law for vertical velocity distribution is used.

3.3.1.2 Determination of Applied Forces

ANSI A58.1-1972 (ANSI 1972) is used to develop wind pressures and distributions on structures. The effective external velocity pressures for structures, q_f, and for portions thereof, q_p, at various heights above ground are in accordance with Tables 5 and 6 of ANSI A58.1 for exposure C. Effective velocity pressures for calculating internal pressure (q_m) are in accordance with ANSI A58.1 Table 12. External pressure coefficients (C_p) are in accordance with ANSI A58.1 Tables 7 and 10 and internal pressure coefficients (C_{pi}) are in accordance with ANSI A58.1 Table 11.

A step function of pressure with height is used. The specified resultant design wind pressure at a given height is applied over a height zone defined by one-half the difference in adjacent heights for which the design wind pressures are specified. The resultant design wind pressure acts normal to the surface of the structure being considered.

3.3.2 TORNADO LOADINGS

Seismic Category I structures requiring tornado design are listed in Table 3.2–1.

3.3.2.1 Applicable Design Parameters

Tornado design parameters are in conformance with Regulatory Guide 1.76 (Section 1.8) as described below:

Maximum wind speed	360 mph
Rotational speed	290 mph

Translational speed	70 mph (max)
Radius of maximum rotational speed	150 ft
Pressure drop	3.0 psi
Rate of pressure drop	2.0 psi/sec

The maximum wind speed is the sum of the maximum rotational speed component and the maximum translational speed component.

Table 3.5–13 lists the tornado generated missiles and their characteristics.

3.3.2.2 Determination of Forces on Structures

Total tornado load is determined as a result of tornado wind (W_w), differential pressure (W_p), and missile (W_m) loadings.

Tornado Wind Load (W_w)

Tornado wind load is determined as follows:

$$W_w = q\ C_p$$

where:

q = External effective velocity pressure

C_p = External wind pressure coefficient for structure being considered

The external effective velocity pressure (q) is determined in accordance with ASCE Paper No. 3269 (ASCE 1961) as follows:

$$q = 0.00256\ v^2 \text{ psf}$$

where:

v = Applicable tornado wind speed in mph

The external wind pressure coefficient (C_p) defines the pressure acting on the surface of the structure. The external pressure coefficients for rectangular shape structures are in accordance with ANSI A58.1. The coefficients for cylindrical shape structures are based on Table 4(f) of ASCE Paper No. 3269 (ASCE 1961). The coefficients for the containment structure dome are based on Wind Stresses in Domes (Gondikas and Salvadori 1960). Coefficients for structural steel shapes are based on ASCE Paper No. 3269 (ASCE 1961).

Tornado Differential Pressure Load (W_p)

For all reinforced concrete structures, with the exception of the fuel building and the emergency generator enclosure, the exterior walls and roofs were designed for nonvented conditions (full 3 psi pressure drop).

For the fuel building, the section of metal roof at elevation 55 feet-3 inches and between column lines G.5 and H is capable of venting the building. For this reason, the design is based on two conditions. The first condition considers the structure to be vented through the area covered by this metal roof. Those sections of walls and floors of interior cubicles which are affected by the venting of the building through the roof area are designed for the full 3 psi pressure drop. For the second condition, it is assumed that this roof does not vent the building. Therefore, the exterior walls and roofs of the building are designed for the full pressure drop (3 psi).

For the emergency generator enclosure, the diesel generator muffler cubicle above elevation 51 feet-0 inch is capable of venting. Therefore, the design of this building is also based on two conditions. The first condition considers the diesel generator muffler cubicle above elevation to be vented by the large openings in the exterior walls of the cubicle. The portions of walls and floors of interior cubicles which are affected by venting of the muffler cubicle are designed for the full 3 psi pressure drop. For the second condition, it is assumed that this cubicle does not vent and the exterior walls and the roof of the structure affected are designed for the full 3 psi pressure drop.

A dynamic load factor of 1.0 is applied to the pressure drop since all wall and floor panels subject to pressure drop have frequencies greater than 4 Hz as shown in Table 3.3-1. Figure 3.3–1 presents the pressure drop time history and the calculated dynamic load factor curve for the pressure drop. Comparing the two, it is evident that the Millstone 3 structural elements experience no dynamic amplification during the pressure drop condition.

Tornado Missile Load (W_m)

Section 3.5.3 describes the methods of analysis to determine the impact effects of tornado missiles.

Total Tornado Load (W_t)

The total tornado load to be applied to Seismic Category I structures is determined from the most adverse combination of individual tornado loadings. The loading combinations considered are:

$W = W_w$

$W = W_p$

$W = W_m$

$W = W_w + 0.5\ W_p$

$W = W_w + W_m$

$$W = W_w + 0.5\ W_p + W_m$$

The total tornado load is then combined with other loads as specified in Sections 3.8.1.3, 3.8.3.3, 3.8.4.3, and 3.8.5.3.

3.3.2.3 Effect of Failure of Structures or Components not Designed for Tornado Loads

Structures which do not require Category I design are either located so that structural failure does not affect the ability of safety related structures or systems to perform their intended design function, or designed so that they do not collapse under tornado wind load. The metal siding and roofing of the service, turbine, waste disposal, containment enclosure buildings, and portions of the fuel building are assumed to blow off under tornado wind load. The resulting siding and roofing missiles are less severe than the missiles described in Section 3.5. The structural steel framing of these structures is designed to withstand tornado wind loads so as not to compromise the integrity of any safety related structure, system, or component.

3.3.3 REFERENCES FOR SECTION 3.3

3.3-1 American National Standards Institute (ANSI) 1972. A58.1 1972, Building Code Requirements for Minimum Design Loads in Buildings and Other Structures. New York, N.Y.

3.3-2 American Society of Civil Engineers (ASCE) 1961. Wind Forces on Structures. In: Transactions of the American Society of Civil Engineers, Paper No. 3269, Vol 126, Part II.

3.3-3 Gondikas, P. and Salvadori, M.C. 1960. Wind Stresses in Domes. Engineering Mechanics Division, American Society of Civil Engineers (ASCE).

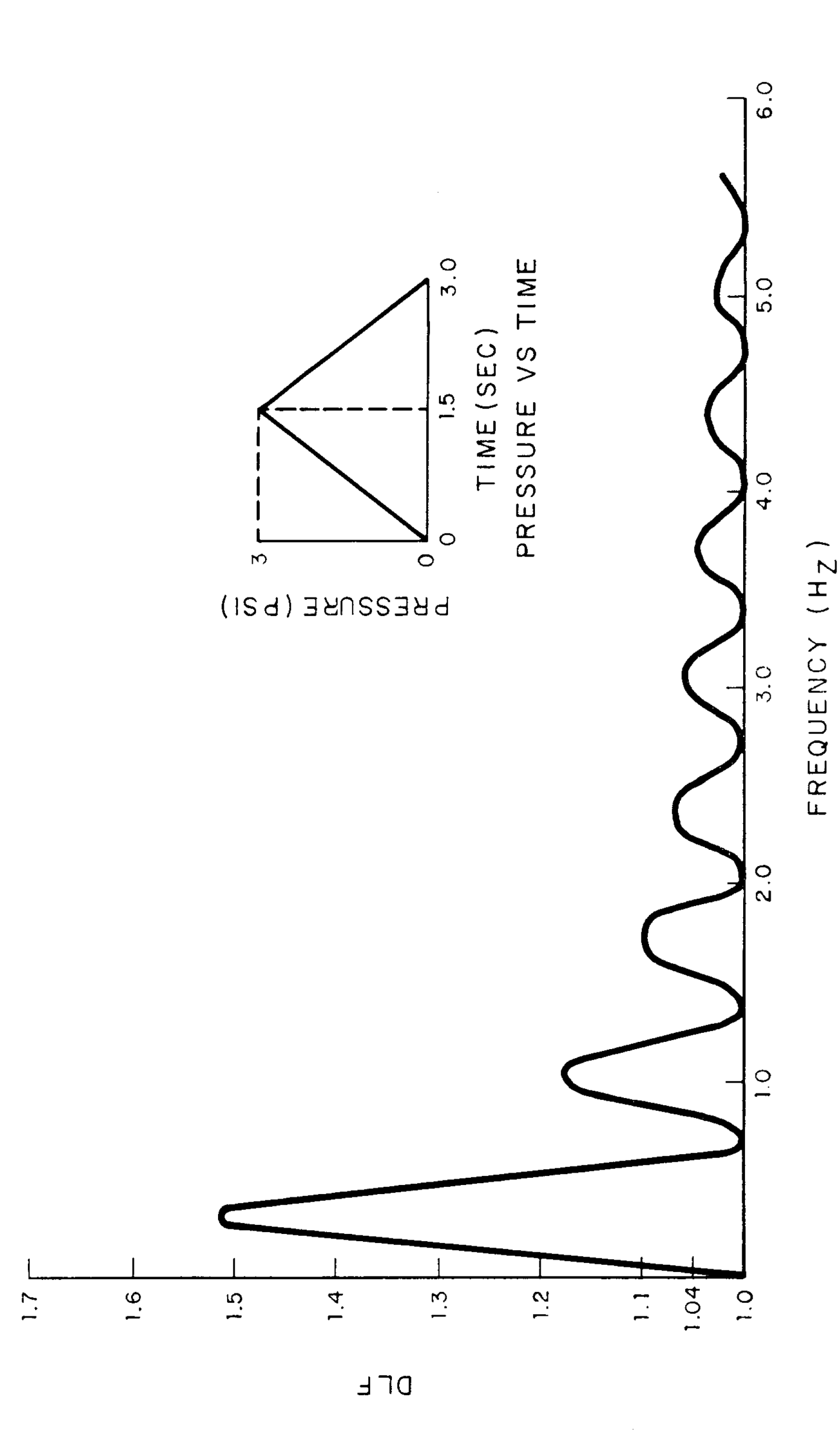
3.3-4 Hollister, S. C. 1970. The Engineering Interpretation of Weather Bureau Records for Wind Loading on Structures. National Bureau of Standards (U.S.), Building Sciences Service, 30.

3.3-5 TRC Service Corporation 1965. Millstone Unit No. 1, Preliminary Site Evaluation.

TABLE 3.3-1 STRUCTURAL PANELS SUBJECT TO TORNADO PRESSURE DROP

PANEL	SIZE (FT)	THICKNESS (FT)	FUNDAMENTAL FREQUENCY (HZ)	REMARKS
MSVB North Wall	36x73	2	6.5	Simple Supports
Aux. Bldg. Roof	49.5x63	2	4.6	Simple Supports
Aux. Bldg. Roof	126.5x26.5	2	9.0	Simple Supports Beam
Fuel Bldg. Roof	59.5x66.5	2	4.0	Simple Supports
Control Bldg. Roof	57.5x102	2	4.2	Partial Fixity

FIGURE 3.3–1 DYNAMIC LOAD FACTOR CURVE FOR TORNADO PRESSURE DROP



3.4 WATER LEVEL (FLOOD) DESIGN

This section discusses the flood and the highest groundwater level design for Seismic Category I structures and components. Internal flooding has been addressed as part of equipment qualification requirements and is described in Section 3.11.

3.4.1 FLOOD PROTECTION

3.4.1.1 Flood Protection Measures for Seismic Category I Structures

The design basis flood (maximum combination of storm surge and wave runup) established for Millstone 3 is elevation +23.8 feet msl and the maximum still water level is elevation +19.7 feet (Section 2.4.5). All unit safety-related structures and equipment, except the circulating and service water pumphouse, are protected from flooding by the site grade of elevation +24 feet msl.

Each pair of service water pumps and pump motors is located at elevation +14.5 feet msl inside individual watertight cubicles in the seismically designed pumphouse (Figure 3.4–1). The walls of these cubicles are watertight up to elevation +25.5 feet msl protecting the pump motors and associated electrical equipment from wave action and probable maximum hurricane (PMH) surge.

All accesses to safety-related structures and facilities are at an elevation of 24 feet-6 inches above the nominal site grade elevation of 24 feet-0 inch and are consequently protected from flooding due to groundwater, storm surge, and direct rainfall, except for the doors that are discussed in Section 2.4.2.3. The two access openings to the service water cubicles inside the pumphouse which are below the flood protection level of +23.8 feet msl are fitted with watertight steel doors capable of withstanding the maximum hydrostatic load occurring at their respective location. Pumphouse roof ventilators are weatherproof and located above the maximum accumulation of snow resulting from the occurrence of the 100-year accumulated snow depth of 52 inches. Equipment access openings on the pumphouse roof over the service water cubicles are fitted with watertight covers. During normal plant operation, the service water cubicles have open drain lines installed in the cubicle sump to enable the service water pump seal water leak off to drain directly into the intake structure pump bay. During severe weather or flooding conditions, the drain lines are isolated and the service water cubicle sumps are drained using sump pumps 3PBS-P1A and 3PBS-P1B.

Foundations of safety-related structures are constructed of reinforced concrete. All subgrade joints between walls and slabs are sealed with waterstops cast in concrete.

The storm drain system uses catch basins and underground conduits and/or drainage ditches to convey runoff to Niantic Bay. Roof and site storm drain systems are designed for a maximum precipitation of 6.5 inch per hour. The probable maximum precipitation of 70.4 inch per hour for a 5-minute period (Hansen, et al., 1982) would result in temporary flooding of the site area in the vicinity of the safety-related buildings as described in Section 2.4.2.3.

In addition to the storm drains, surface runoff from the higher ground to the north and east of plant perimeter roads is intercepted by open ditches and drained into Niantic Bay. Further discussion on the effects of local intense precipitation can be found in Section 2.4.2.3.

The seaward wall of the intake structure is constructed of reinforced concrete, designed to withstand the forces of a standing wave, or clapotis, with a maximum crest elevation of +41.2 feet msl. Further discussion on the effects of PMH storm surge and wave action, including the resultant pressure distribution on the intake wall, can be found in Section 2.4.5.

Combinations of the maximum surge level with a coincident wave and the maximum wave height with a coincident surge for three different speed PMH were examined to determine the maximum uplift pressure on the pumphouse floor. The most critical combination is the maximum wave height of 16.2 feet and a surge level of elevation 19.7 feet msl associated with the slow speed PMH.

The calculated maximum uplift pressure on the pumphouse floor due to the most critical combination of wave action and storm surge during PMH conditions is 863 lb/sq ft. The floor is designed to withstand more than this pressure, precluding the possibility of failure.

The water level fluctuations within the pumphouse, resulting from storm surge and wave action, are dampened by the energy lost in passage through the restricted openings in the trash racks, traveling screens, and operating deck. Internal water level fluctuations are further attenuated because water must enter the structure through a submerged opening (elevation -7 to -30 feet msl) through which the pressure response factor is less than unity.

The discharge outfall structure is also designed to withstand maximum wave forces induced by the most critical combination of wave action and storm surge during PMH conditions. The maximum horizontal pressure is determined for the combination of a maximum wave height of 20.1 feet and a maximum surge level of 19.7 feet msl. The maximum vertical pressure on the upper outfall structure is obtained for a minimum water submergence at the surge level of elevation 9.46 feet msl and a coincident breaking wave of 14.7 feet. The calculated maximum horizontal pressure and the maximum vertical pressure on the discharge outfall structure for the above-mentioned combination events are 2,325 lb/sq ft and 585 lb/sq ft, respectively.

Section 2.5.5.1 discusses shoreline protection in the vicinity of the pumphouse.

Flood protection complies with Regulatory Guide 1.102, Flood Protection for Nuclear Power Plants, as follows:

1. C1 Flood protection is accomplished by the unit's location on a “Dry Site,” with the exception of the circulating and service water pumphouse which utilizes “Incorporated Barriers.”

2. C2 Not applicable

   Refer to Section 1.8 for clarification to Position C1.

3.4.1.2 Permanent Dewatering System

There is no safety-related dewatering system for lowering groundwater levels for Millstone 3. This system is not applicable. Removal of water which bypasses the membrane installed below the Containment Structure is collected in the Engineered Safety Features Building porous concrete groundwater sump and is removed by the Underdrain System (see Section 9.3.3).

3.4.2 ANALYTICAL AND TEST PROCEDURES

Ground levels of all Category I structures, except for the circulating and service water pumphouse and the discharge structure, are located above the design basis flood (DBF) level. This level is based on the maximum combination of storm surge due to the PMH and associated wave run-up (Section 3.4.1). Structures located above this level are designed for the hydrostatic effects of uplift and lateral water pressure resulting from the DBF or normal groundwater, whichever is more severe. Groundwater levels are based on piezometric readings taken at the site (Figure 2.5.4–37).

The circulating water discharge structure and discharge tunnel and the circulating and service water pumphouse are located below the DBF level.

The circulating water discharge structure and discharge tunnel are designed for the hydrostatic and dynamic effects of the DBF as described in Section 3.4.1.

The circulating and service water pumphouse is designed laterally for a standing wave and for uplift on the operating floor due to confined wave action within the pumphouse (Section 3.4.1).

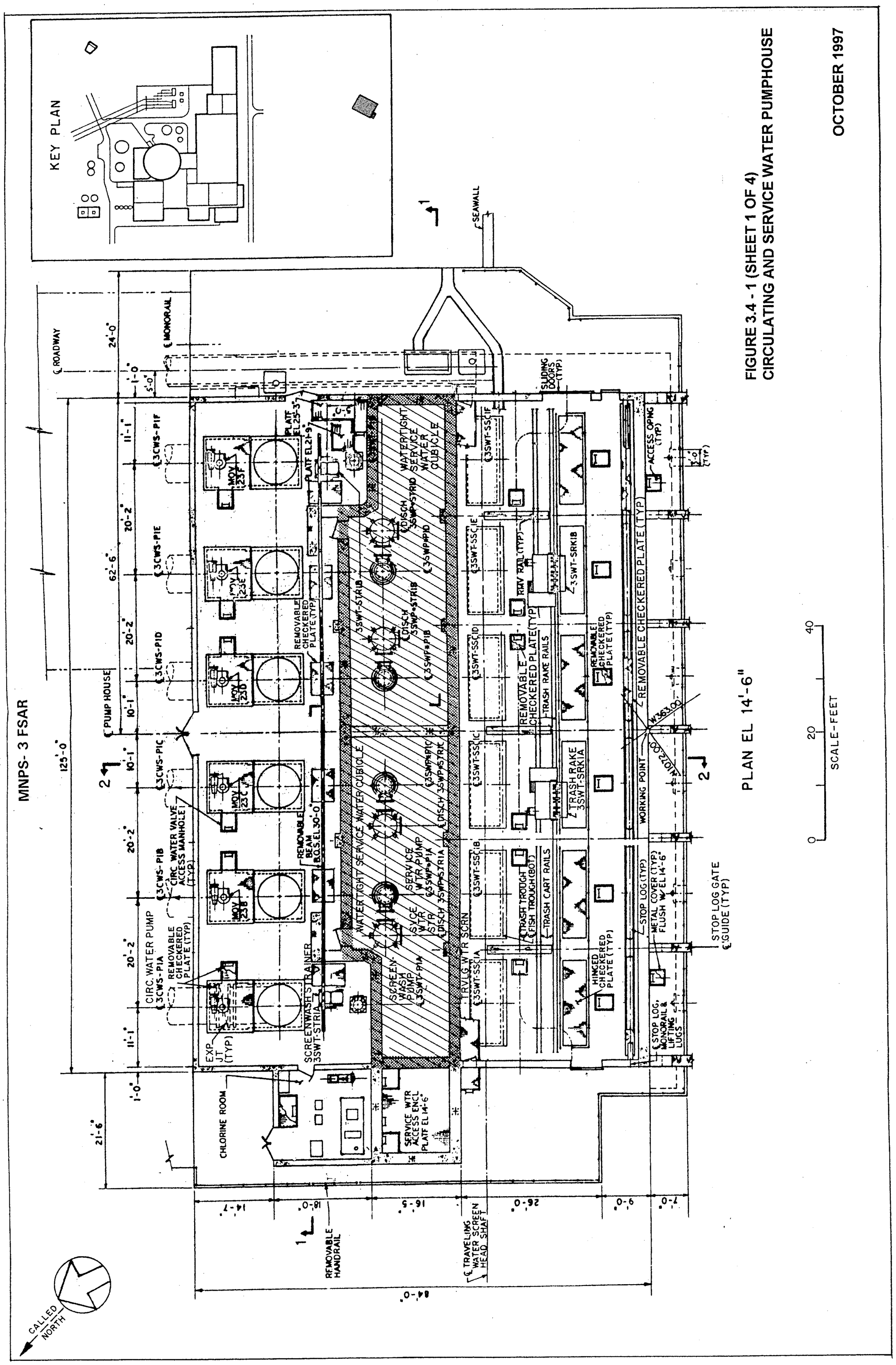
Foundation loadings used in the design reflect saturated soil conditions, where applicable.

The design wind loading described in Section 3.3.1 is applied concurrently with the hydrostatic and dynamic effects of the DBF (Sections 3.8.1.3, 3.8.3.3, and 3.8.4.3). Tornado loading is not applied concurrently with the DBF.

3.4.3 REFERENCE FOR SECTION 3.4

3.4-1 Hansen, E.M., Schreiner, L.C., and Miller, J.F, 1982. Application of Probable Maximum Precipitation Estimate - U.S. East of the 105th Meridian. Hydrometeorological Report No. 52, National Weather Service, NOAA, U.S. Department of Commerce, Washington, D.C.

FIGURE 3.4–1 CIRCULATING AND SERVICE WATER PUMPHOUSE (SHEET 1 OF 4)



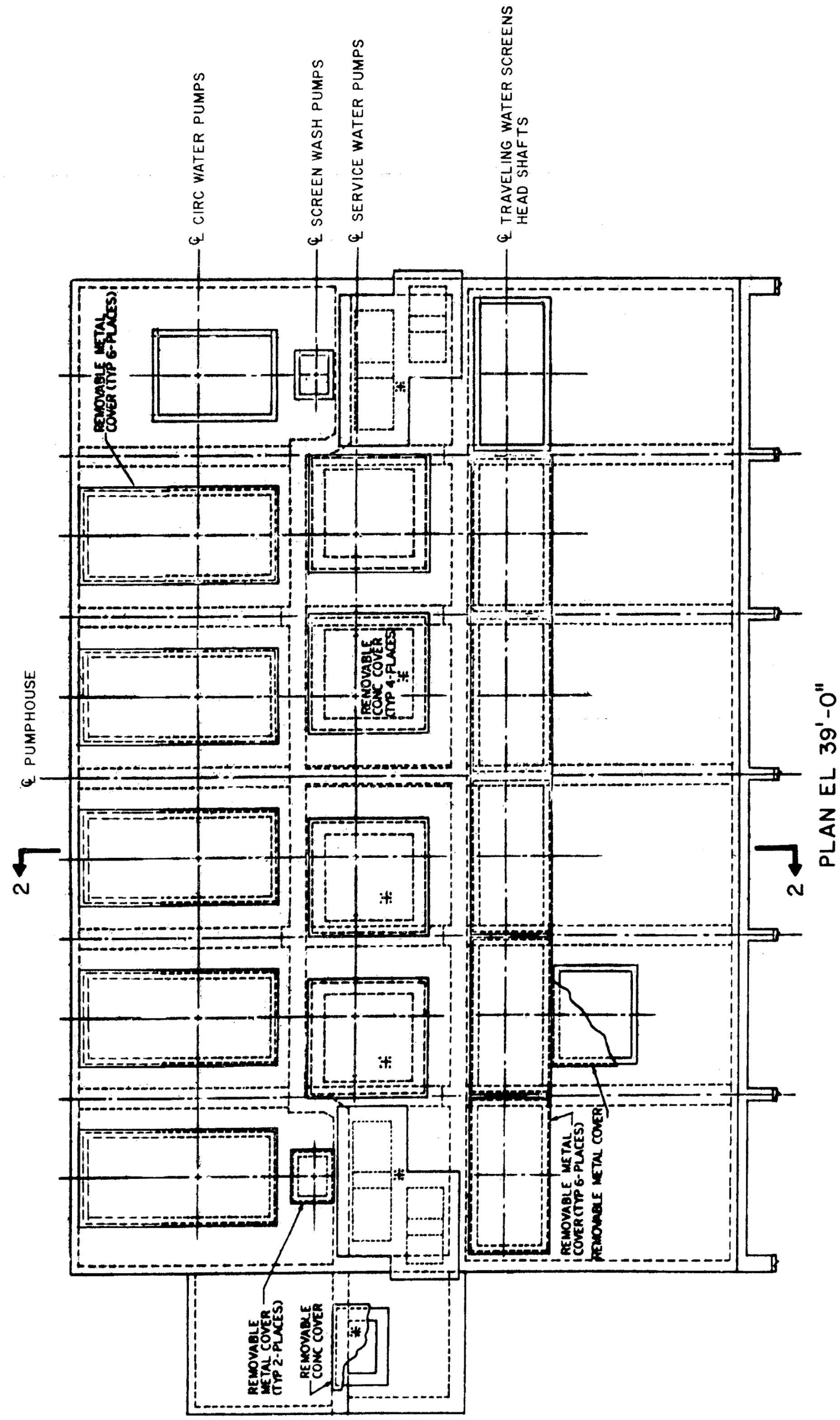


FIGURE 3.4–1 CIRCULATING AND SERVICE WATER PUMPHOUSE (SHEET 2 OF 4)

FIGURE 3.4–1 CIRCULATING AND SERVICE WATER PUMPHOUSE (SHEET 3 OF 4)

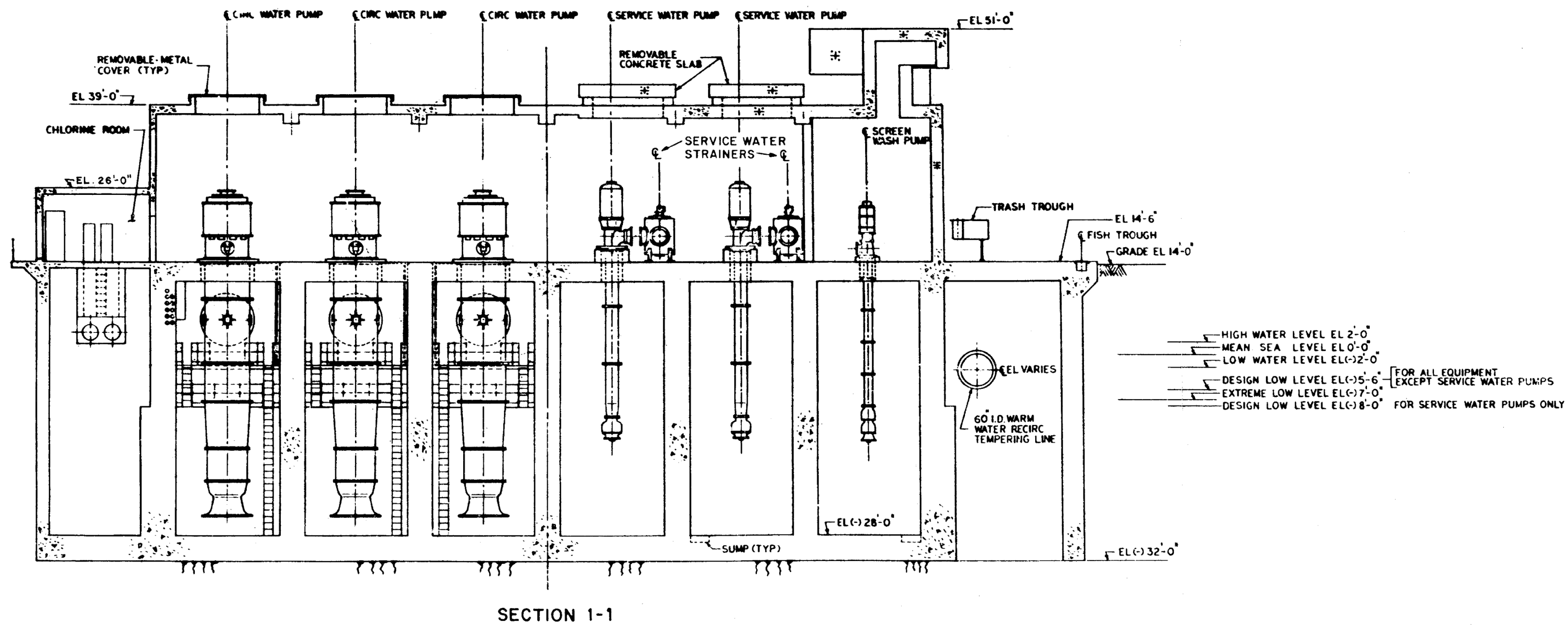
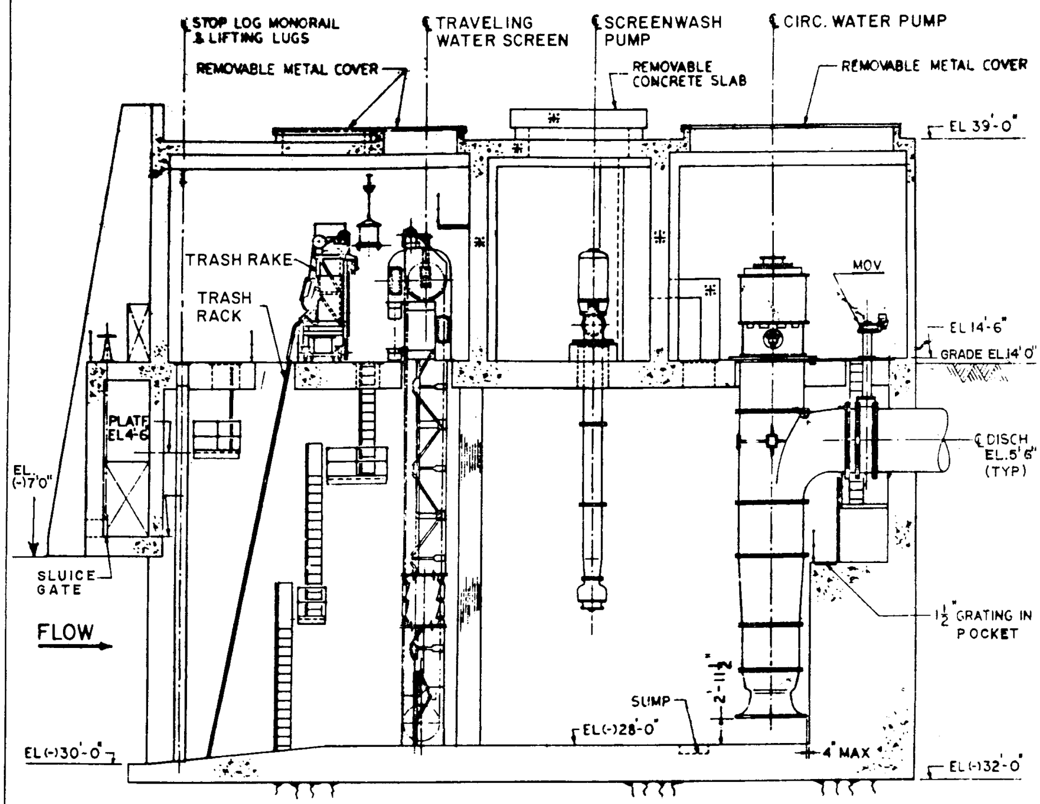


FIGURE 3.4–1 CIRCULATING AND SERVICE WATER PUMPHOUSE (SHEET 4 OF 4)



SECTION 2-2

FIGURE 3.4-1 (SHEET 4 OF 4)
CIRCULATING AND
SERVICE WATER PUMPHOUSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

3.5 MISSILE PROTECTION

3.5.1 MISSILE SELECTION AND DESCRIPTION

Systems and components located both inside and outside the containment have been examined to identify and classify potential missiles. Two broad categories of systems and components are reviewed to determine the potential for generating missiles; pressurized components and high speed rotating machinery. Only designs where a single failure could lead to missile ejection are considered. The basic approach to ensure missile protection of systems and components both inside and outside of containment involves the following considerations:

1. examination of systems in order to identify and classify potential missile sources;

2. evaluation of the design adequacy of equipment to preclude generation of missiles; and

3. evaluation of the effects of the generation of missiles where the potential exists and provisions for protection against them.

The objective is to ensure design adequacy against generation of missiles and means of protecting essential structures, systems, and components should a missile be generated.

3.5.1.1 Internally Generated Missiles (Outside Containment)

The design bases consider missiles generated outside the containment but internal to the plant site. These shall not cause damage that may affect the safe shutdown or cause radiation release during operating conditions, and postulated accident conditions associated with the effects of missile formation. Table 3.5–1 identifies the safety-related structures, systems, and components outside the containment required for safe shutdown of the reactor under all conditions of plant operation.

Valves in high energy fluid systems are evaluated as potential missile sources. Valves are typically designed with parts which are removable for maintenance. It is these removable parts which present the most significant potential for missile producing failures. Valves provided with back-seated stems are not considered credible sources of missiles. This design feature effectively eliminates the possibility of ejecting valve stems even if the stem threads fail. Valve bonnets are considered credible sources for missiles in cases where the bonnet is bolted to the body. The bonnet and the connection bolts are postulated to be ejected. Valve bonnet missiles are not considered credible where the bonnet is welded to the body, the bonnet is integral with the body, or the bonnet bolts are torqued in a controlled manner. Credit may be taken for air and motor operators which interfere with the ejection of valve stems and bonnets. The Applicant’s review of valves located outside containment has indicated that all the valves in the high energy systems can be categorized as follows.

1. Valves that are isolated or enclosed from safety related equipment. These valves do not impose any danger.

2. Valves that are not enclosed or isolated. These valves are reviewed for a credible failure mechanism which could eject a missile. If a credible failure mechanism is determined, then failure is admitted and the consequences of such a failure are assessed. If an unacceptable interaction occurs, then the missile source is subject to reorientation if feasible, the target is subject to relocation if feasible, or a barrier is provided to prevent interaction. An unacceptable interaction is an interaction with an essential structure, system, or component.

Centrifugal pumps and fans located outside the containment in areas containing safety related components have been evaluated for missiles caused by overspeed or failure. The maximum no-load speed of these centrifugal pumps and fans is equivalent to the maximum operating speed of their motors. Consequently, no overspeed is expected and missiles associated with overspeed conditions in centrifugal pumps and fans outside the containment are not postulated.

Fans are further evaluated for missile generation under normal operating speeds due to fatigue failure or manufacturing defects. Fan fragments are postulated only where a credible single failure mechanism results in fragmentation. Such fragments have been shown either to lack sufficient energy to penetrate the fan housing or to result in acceptable interactions with essential targets. In this assessment, the fragments are assumed to be unimpeded by any flexible connections between the fan housing and attached ducting.

The auxiliary feedwater pump turbine is equipped with redundant overspeed detection devices and a regularly tested turbine trip valve, as such overspeed in the turbine is considered credible only up to the trip setting at 10 percent over rated speed. At this speed there exist substantial margins between the energy available in the fragments generated from the turbine-driven pump and the energy required to escape the pump casing; therefore, missiles are not postulated from the pump component. Similarly, fragments generated from the turbine component lack the penetrating geometry and the energy to perforate its casing. As such, neither the pump nor the turbine driver is considered a source of internally generated missiles and no essential systems or components can be adversely affected. Nonetheless, the auxiliary feedwater pump turbine is located and oriented within a concrete cubicle to prevent any generated missiles from affecting other safety systems, such as the motor-driven auxiliary feedwater pumps, in adjacent cubicles.

The motor-generator that provides power to the control rod drive mechanisms (CRDM) is located outside the containment. The flywheel on this component has been evaluated as a potential missile. The fabrication specifications of the motor-generator-set-flywheel control the material to meet ASTM-A533-70, Grade B, Class I with inspections per MIL-I-45208A and flame cutting and machining operations governed to prevent flaws in the material. Nondestructive testing consisting of nilductility (ASTM-E-208), Charpy V-notch (ASTM-A593), ultrasonic (ASTM-A577 and A578), and magnetic particle (ASME Section III, NB2545) is performed on each flywheel material lot. In addition to these requirements, stress calculations are performed consistent with guidelines of ASME Section III, Appendix A, to show the combined stresses due to centrifugal forces and the shaft interference fit shall not exceed one-third of the yield strength at normal operating speed (1,800 rpm) and likewise, shall not exceed two-thirds of the yield strength at 25 percent overspeed. However, no overspeed is expected for the following reason: the flywheel weighs approximately 1,300 pounds and has dimensions of 35.36 inches in diameter and

4.76 inches in width. The flywheel, mounted on the generator shaft and directly coupled to the motor shaft, is driven by a 200 hp, 1,800 rpm synchronous motor. The torque developed by the motor is insufficient for overspeed. Therefore, there are no credible missiles from the CRDM motor-generator flywheel.

Evaluation of missiles being generated from the emergency generator enclosure concluded that there is no need to evaluate missile generation. Safety related emergency generators are located in a structure designed for tornado missile protection; consequently, missiles from the diesel engines are considered unable to penetrate this structure. The essential diesel generator systems are redundant and separated so that a missile generated by one diesel engine will not affect the other. Doors are offset from the generators' axes precluding the possibility of missiles exiting from the doorway of the structure.

3.5.1.2 Internally Generated Missiles (Inside Containment)

The design bases are such that missiles generated within the reactor containment will not cause loss of function in any redundant engineered safety feature nor radiation release or damage the containment boundary.

In addition, a missile accident which is not caused by a LOCA shall not initiate a LOCA.
Table 3.5–2 identifies the structures, systems, and components inside the containment whose failure could lead to offsite radiological consequences or which are required for safe plant shutdown to a cold condition assuming an additional single failure.

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

3.5.1.2.1 Missile Selection and Description

Failure of the reactor vessel, steam generators, pressurizer, and reactor coolant pump casings leading to missile generation are not considered credible because of the combination of material characteristics, inspections, quality control during fabrication, erection, and operation, conservative design, and prudent operation as applied to the particular component.

The reactor coolant pump flywheel is not considered a source of missiles for the reasons discussed in Section 5.4.1. Nuts and bolts are of negligible concern because of the small amount of stored elastic energy.

Centrifugal pumps, fans, and air compressors (centrifugal and axial) located inside the containment have been evaluated for missiles associated with overspeed failure. The maximum no-load speed of these centrifugal pumps, fans, and air compressors is equivalent to the operating speed of their motors. Therefore, no overspeed is expected and missiles associated with overspeed conditions in centrifugal pumps, fans, or air compressors within the containment are not postulated.

Fans are further evaluated for missile generation under normal operating speeds due to fatigue failure or manufacturing defects. Fan fragments are postulated only where a credible single failure mechanism results in fragmentation. Such fragments have been shown either to lack sufficient energy to penetrate the fan housing or to result in acceptable interactions with essential targets. In this assessment, the fragments are assumed to be unimpeded by any flexible connections between the fan housing and attached ducting.

The following nuclear steam supply system components are considered to have a potential for missile generation inside the reactor containment:

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

2. valves;

3. temperature and pressure sensor assemblies; and

4. pressurizer heaters.

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

1. Control rod drive mechanisms are shop tested at 4,100 ± 75 psi.

2. Control rod drive mechanism housings are individually hydrotested to 3,107 psi after they are installed on the reactor vessel to the head adapters and checked again during the hydrotest of the completed reactor coolant system.

3. Control rod drive mechanism housings are made of Type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures that are encountered.

However, it is postulated that the top plug on the control rod drive mechanism could become loose and would be forced upward by the water jet. The following sequence of events is assumed.

1. The drive shaft and control rod cluster are forced out of the core by the differential pressure of 2,500 psi across the drive shaft.

2. The drive shaft and control rod cluster, latched together, are assumed fully inserted when the accident starts.

3. After travelling approximately 12 feet, the rod cluster control spider hits the underside of the upper support plate.

4. Upon impact, the flexure arms in the coupling join the drive shaft and the control cluster fracture freeing the drive shaft from the control rod cluster.

5. The control cluster is completely stopped by the upper support plate. However, the drive shaft continues to accelerate upward and strikes the control rod drive mechanism missile shield provided.

The control rod drive mechanism (CRDM) missiles are summarized in Table 3.5–3. The missile velocities have been calculated by balancing the forces due to the water jet. No spreading of the water jet has been assumed. CRDM missile impact velocity, kinetic energy, and penetration are considered in the design and layout of the missile shield.

Valve stems in motor-operated (MOV) or air-operated (AOV) valves are not considered credible sources of missiles. All the isolation valves installed in the reactor coolant system have stems with a back seat. This effectively eliminates the possibility of ejecting valve stems even if the stem threads fail.

Valves within the reactor coolant pressure boundary have been reviewed to identify potential missiles. This review identified no credible failures that could result in missile generation, except for valve missiles in the region where the pressurizer extends above the operating floor. Valves in this region are the pressurizer safety valves, the motor-operated isolation valves in the relief line, the air-operated relief valves, and the air-operated spray valves. Although failure of these valves is not considered credible, failure of the valve bonnet body bolts is, nevertheless, postulated and the integrity of the containment liner and safety related equipment from the resultant bonnet missile is assured by the pressurizer cubicle walls and roof which act as a missile barrier.

The missile characteristics of the valves in the region where the pressurizer extends above the operating deck are given in Table 3.5–4.

The only credible sources of jet-propelled missiles from the reactor coolant piping and piping systems connected to the reactor coolant system are the temperature and pressure sensor assemblies. The resistance temperature sensor assemblies are of two types: “with well” and “without well.” Two rupture locations have been postulated: around the weld (or thread) between the temperature element assembly and the boss of the “without well” element, and the weld (or thread) between the well and the boss for the “with well” element.

The missile characteristics of the piping temperature sensor assemblies are given in Table 3.5–5. A 10-degree half-angle expansion water or steam jet has been assumed. The missile characteristics of the piping pressure element assemblies are less severe than those of Table 3.5–5.

Temperature and pressure sensors are installed on the reactor coolant pumps close to the radial bearing assembly. A hole is drilled in the gasket and sealed on the internal end of a steel plate. In evaluating missile potential, it is assumed that this plate breaks and the pipe plug on the external end of the hole becomes a missile.

The missile characteristics of the reactor coolant pump temperature sensor, the instrumentation well of the pressurizer, and the pressurizer heaters are given in Table 3.5–6. A 10-degree expansion jet has been assumed.

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

The initial flight direction of these missiles has been determined and only low kinetic energies are possible. Only fragile components would be damaged by this missile category. Protection was implemented by keeping such components as electrical equipment and cables out of the primary flight path.

Pressurizer heaters could loosen and become jet-propelled missiles. The integrity of safety related equipment from postulated heater vessels is assured by the pressurizer cubicle floor and walls.

All valves have been designed against bonnet-to-body connection failure and subsequent bonnet ejection by means of:

1. Compliance with the ASME Code, Section III

2. Control of load during the bonnet-to-body connection stud tightening process

The proper stud torquing procedures limit the stress of the studs to the allowable limits established in the ASME Code. This stress level is far below the material yield. The valves are hydrotested per the ASME Code, Section III. The bodies and bonnets are volumetrically and surface tested to verify soundness. Critical valves are also designated for inservice inspection to ASME Code, Section XI, requirements (Section 6.6).

Non-ASME III valves were examined for potential missiles. Provision for protection against them was undertaken based on the following:

1. Valve stems are provided with back seats. This prevents stem ejection should the threads fail.

2. Valves are oriented to prevent postulated missiles from impacting critical targets.

3. Probability of failure is low ($P \leq 10^{-4}$ per year), so a low impact probability may be argued against generation of missiles.

4. Barriers erected specifically for protection against internal missiles.

Missiles generated due to a seismic event are addressed in Section 3.7B.3.13.

3.5.1.2.2 Missile Protection Provided

Safety related structures, systems, and components whose safety function might be impaired, are protected from postulated missiles by:

1. Locating the systems or components in individual missile proof structures

2. Physically separating redundant systems or components of the system

3. Providing special localized protective shields or barriers.

The ability of structures or barriers to withstand the effects of potential internally generated missiles is discussed in Section 3.5.3.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Turbine Placement and Orientation

Figure 3.5–1 shows the turbine placement and orientation for the three unit site. This figure also indicates the ±25-degree missile ejection zone with respect to the low pressure turbine wheels for each turbine unit "within reach" of plant structures. Figure 3.5–2 shows an elevation view of Millstone 3 structures within the ±25-degree missile ejection zone.

3.5.1.3.2 Missiles Identification and Characteristics

A bounding value of 1.0E-2 per year for the probability of unacceptable damage in the event that a turbine missile is generated by wheel failure, is used.

3.5.1.3.3 Target Description

Heavy zones on Figure 3.5–1 identify target areas for low trajectory missiles. Table 3.5–7 provides applicable elevations over which those wall areas are considered as targets for each heavy line. Areas other than those shown by the heavy lines have been excluded as targets because they are either outside the ejection zone for low trajectory missiles or they meet the criteria that the probability of the missile strike damaging its target is zero, $P_3 = 0$.

Target areas for high trajectory missiles are the roof areas of all Category I structures. These targets are shown as shaded areas on Figure 3.5–1.

3.5.1.3.4 Probability Analysis

Regulatory Guide 1.115 Revision 1, "Protection Against Low-Trajectory Turbine Missiles" states that the NRC staff considers a hazard due to low trajectory turbine generated missiles of less than 1.0E-7 per year an acceptable risk rate for the loss of an essential system for a single event. The methodology is based on maintenance and inspection activities which minimize the potential for missile generation. Using the missile damage probability of 1.0E-2 (for "unfavorable orientation"), the probability of turbine failure, P_1, must be shown to be less than 1.0E-5 in order to meet overall R.G. 1.115 acceptable risk rate.

The turbine system maintenance program is established based on manufacturer's calculations of missile probabilities. The program ensures that the 1.0E-5 probability is maintained. Higher probabilities of turbine failure are acceptable for short duration's (for example, to support on-line maintenance and surveillance activities) based on criteria reviewed and accepted by the NRC (NUREG-1048, Supplement No. 6, Appendix U).

3.5.1.3.5 Turbine Overspeed Protection

The turbine control system is an electrohydraulic control (EHC) system that includes both digital and analog circuitry, electronic servo hardware, and hydraulic valve actuators.

The EHC provides a normal overspeed protection system and an emergency overspeed protection system to limit turbine overspeed. These two systems are essentially separate and independent. The normal overspeed protection system is part of the turbine load and speed control system and is designed to limit turbine overspeed without a turbine trip under all load conditions. The emergency overspeed protection system is part of the emergency trip system and is designed to trip the turbine if the turbine speed exceeds 110 percent of rated speed (Section 10.2).

3.5.1.3.6 Turbine Valve Testing

The main turbine generator control, main stop, intercept, and reheat stop valves are routinely tested (Section 10.2.3.6).

Control and main stop valves are tested one at a time, and as each test is completed, the valve is returned to its original position before the next valve is tested. Intercept and reheat stop valves are interlocked so that a pair of these valves in one crossover pipe is tested together. For this test, one pair of pipe is tested and the valves returned to the open position before the next pair is tested.

3.5.1.4 Missiles Generated by Natural Phenomena

The only credible missiles generated by natural phenomena are those generated by a design basis tornado. Those Category I structures designed to withstand the effects of tornado missiles and the systems and components thus protected are identified in Tables 3.5–1 and 3.5–2.

A minimum of 2 feet thickness of reinforced concrete having a minimum strength of 3,000 psi (28 day compressive strength) was used for walls, roofs, and floors designated as missile protection. The minimum reinforcing steel each way in each face of any square foot of wall or slab providing missile protection is 1.85 square inches.

Postulated missiles generated by the design basis tornado (Section 3.3.2) are listed in Table 3.5–13, which includes all parameters necessary to determine missile penetration. These missiles are considered capable of striking in any orientation.

Ventilation openings in the various facility buildings housing essential shutdown equipment are protected by reinforced concrete labyrinths.

There are no other design basis missiles resulting from flood or any other natural phenomena described in Section 2.2.3.

3.5.1.5 Missiles Generated by Events Near the Site

No missiles of any significance are expected to be generated by events near the site due to distances from nearby transportation routes. The possibility of missiles from a Providence & Worcester (P&W) railroad tank car explosion was considered; however, it was determined that such a tank car explosion would not generate significant missiles at the plant site. For a detailed description, see Section 2.2.3.

3.5.1.6 Aircraft Hazards

A study of the probability of aircraft which use the nearby airport and airways colliding with the safety related structures of the Millstone site has been conducted. Due to the conservatism of the analysis, the values derived by the analytical model are believed to be substantially higher than the true probability. The study concludes that the aircraft accident probability would be less than 1.3×10^{-7} per year for a number of years since no increase in air traffic is projected in the vicinity of the site.

3.5.2 STRUCTURES, SYSTEMS, AND COMPONENTS TO BE PROTECTED FROM EXTERNALLY GENERATED MISSILES

Missiles to be considered in this section are identified in Section 3.5.1.

As discussed in Section 3.5.1, all plant systems and components must be protected whose failure can lead to unacceptable offsite radiological consequences or which are required to shut down the reactor and maintain it in a safe condition, assuming an additional single failure. Safety related structures, systems, and components that are required for a safe shutdown of the reactor are identified in Tables 3.5–1 and 3.5–2. All components containing radioactive fluids are protected or fall within the results of the design basis offsite radiological analyses which are presented in Sections 15.7.1, 15.7.2, and 15.7.3.

3.5.3 BARRIER DESIGN PROCEDURES

Missile barriers are designed to withstand the effects of missiles described in Section 3.5.1 without compromising plant safety.

Most internally generated missiles have low to moderate energy. Consequently most barriers erected specifically for protection against internal missiles are rather light and usually protect only a limited area. Small steel barriers, rather than concrete structures are generally most suitable. There are a few exceptions, such as the control rod drive missiles, where sufficient energy exists to make a concrete barrier more practical.

3.5.3.1 Concrete Barriers

Missile barrier design requirements include:

1. Prevent perforation of the missile into safety related areas with one or more barriers

2. Scabbing particles are not generated or are limited to energy levels which still permit safe shutdown of the plant

3. Structural response to missile impact permits the barrier and its supports to safely carry other loads during and after impact. In addition, the deflection of the barrier does not impair the safety or safety function of a Category I system.

For concrete barriers, the requirement to stop the missile is fulfilled by showing that perforation of the barrier, or the final barrier in a multiple barrier design, does not occur. This is established based either on test data (Nusbaum et al., 1976; Stephenson 1976; and Rotz 1975), or an empirical evaluation of these test data.

Determination of whether scabbing occurs is also based on procedures set forth in Stone & Webster Engineering Corporation Topical Report, SWEC-7703, Appendix B (SWEC 1977).

The overall structural response to missile impact is based on dynamic time history calculations for elastic-plastic behavior of the missile and the barrier. The general method for calculating overall structural response is described in “Introduction to Structural Dynamics” (Biggs 1964). The ultimate load capacity of concrete barriers is determined by yield line theory. The barriers are designed so that the calculated ductility ratio of the barriers for any load combination is less than the maximum allowable ductility ratio.

Ductility ratio is defined as the ratio of maximum acceptable displacement (Xm) to the displacement at the effective yield point (Xy) of the structure.

For analytical procedures for analyzing concrete barriers for overall structural response to tornado missiles refer to Stone & Webster Engineering Corporation Topical Report SWEC 7703, Appendix C (SWEC 1977).

If the concrete barrier is required to carry loads during and after missile impact, the maximum allowable ductility ratio is limited to a factor of 10. In particular, for beam-column members, where the compressive load is equal to or less than one-third of that which produces balanced strain conditions for the cross section, the allowable ductility ratio is 10.

Refer to Table 3.5–14 for examples of maximum barrier elastic deflection, maximum barrier deflection, and ductility ratio as a function of missile and barrier span for 24 inch concrete barriers.

For an example of maximum barrier elastic deflection, maximum barrier deflection, and ductility ratio for a beam-column, refer to Table 3.5–15.

If a concrete barrier is not required to carry other loads during and after impact, the maximum allowable rebar elongation is limited to 5 percent.

3.5.3.2 Steel Barriers

Steel barriers are designed with consideration given to local perforation, overall response, and support loads.

The thickness of the barrier plate required to prevent perforation is determined from the Stanford equation (Gwaltney 1968). The BRL formula for steel perforation is used if the Stanford equation is not applicable. However, impacts near a support are most limiting with respect to perforation and where these are postulated, the Stanford equation is used with the artificial, but conservative, assumption that the support spacing equals the missile diameter.

The preferred design is a single plate supported at the corners and, if necessary, reinforced by bolting around the periphery to channels or other rolled shapes. The overall response is evaluated considering a central impact and equating the missile energy to the energy absorbing capacity of the barrier plate. Plastic limit theory and an allowable of 50% ultimate uniform strain are used to determine this capacity.

The load delivered to the supports is dynamic and equal to the plastic limit load of the plate. Since the impact may occur near one support, each support is sized accordingly. When the resulting load is too high for founding structure, soft supports are used. In this case the supports, whether elastic or plastic, are designed to accept the available energy:

$$E = E_o \frac{m}{M + m}$$

where

E_o = Initial missile energy,

m = Missile mass, and

M = Barrier mass participating in missile momentum transfer

3.5.3.3 Design Evaluation

Only one missile is postulated at a time and is assumed to strike the barrier end-on. The missiles have two effects on structures, walls, or any other barrier: local effects and overall response. The local effects include penetration, perforation, and spalling or scabbing. The overall response includes the flexural and shear effects.

Reinforced concrete external roofs and walls of Seismic Category I structures form barriers against tornado-generated missiles. Buried underground, safety related duct runs and piping are also protected.

Local Effect: The estimate of missile penetration in concrete barrier is based on the modified Petry formula (Amirikian 1950). Sufficient thickness of concrete is provided to prethickness determined for an infinitely thick slab is provided. Determination of penetration by missiles into steel plates is in accordance with the Stanford Research Institute formula (Gwaltney 1968).

Overall Barrier Response: The overall response of structural barriers to missile impact is determined by time history analysis of the slab missile system. The ductility ratio so determined must be less than the allowable ductility ratio. This ductility ratio is a function of the controlling nature of the structural behavior. For beams, walls, and slabs where flexural controls design, the permissible ductility ratios are given in Section 3.5.3.1 and are based on Stone & Webster Engineering Corporation Topical Report, SWEC 7703 (SWEC 1977). Flexural strength is determined from an ultimate strength theory with the limitations on ductility.

An example of a missile shield that has been designed using these procedures is the control rod drive mechanism missile shield.

The shield is provided over the control rod drive mechanisms to block any missiles which might be associated with a fracture of the pressure housing of any mechanism. This shield is constructed of reinforced concrete with a steel facing plate, and is located above, and as near as possible, to the housing. This limits the velocity of the ejected missile and prevents missiles from bypassing it. The shield is designed using procedures given in ORNL-NSIC-5 (Greenstreet et al., 1965).

3.5.3.4 Secondary Missiles

Concrete impacts produce secondary missiles from the front face of the barrier. These missiles have considerably less energy than the primary missile. Concrete secondary missiles were only considered for a postulated primary missile which has more than 4 ft-kips of kinetic energy. Our review showed that scabbing does not damage safety related equipments.

Secondary missiles from impacts are limited to components that may be torn loose by the primary missile. The energy dissipated in creating secondary missiles cause the remaining energy in the primary missile to be negligible. Only concrete walls were found to exhibit this phenomena of loosing parts or scabbings. Nevertheless, all secondary missiles were found passive.

3.5.4 REFERENCES FOR SECTION 3.5

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TABLE 3.5–1 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS OUTSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN*

Structures	Location**	FSAR Section	FSAR Figure
Containment penetrations	AB, EF, MS, HR	3.8.1	3.8–18, 3.8–19
Containment hatches (personnel and equipment) outside containment	AB, HR	3.8.1	3.8–21, 3.8–22
Containment reinforced concrete external superstructure	CS	3.8.1	3.8–1 thru 3.8–14 and 3.8–19
Cable tunnel from containment structure to control building	CB, AB, SB	3.8.4	3.8–62 thru 3.8–67 and 8.3–1
Building reinforced concrete walls surrounding safety related systems and components	AB, EF, FB, SB	3.8.4	3.8–62 thru 3.8–67
Main steam and feedwater valve Areas	MS	3.8.4	3.8–62 thru 3.8–67
Engineered safety features areas	EF	3.8.4	3.8–62 thru 3.8–67
Control building	CB	3.8.4	8.3–1
Service water pumphouse	SP	3.8.4	3.8–69 and 3.4-1
Diesel generator building	DG	3.8.4	3.8–68, 3.8–69
Diesel generator fuel oil pump House	OY	3.8.4, 3.8.5	3.8–68, 3.8–69
Systems			
Chemical and Volume Control System		9.3.4	9.3–7 and 9.3–8
Charging pumps	AB		
Seal water injection filter	AB		
Boric acid tanks	AB		
Boric acid transfer pumps	AB		
Piping, valves, and portions of the system required for charging, seal water injection and boration	AB		

TABLE 3.5–1 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS OUTSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN* (CONTINUED)

Structures	Location**	FSAR Section	FSAR Figure
Instrumentation, cable, and controls required to perform a safety function in the above portions of system	AB, CR, SB		
Residual Heat Removal System		5.4	5.4–4
Residual heat removal pumps	EF		
Residual heat removal exchanger	EF		
Tube side			
Shell side			
Piping and valves required to cool and maintain the RCS in a cold shutdown condition	EF, AB		
Instrumentation, cable, and controls required to perform a safety function in the above portions of system	EF, AB, SB, CR		
Emergency Core Cooling System		6.3	6.3–1 thru 6.3–4
High-head safety injection pumps (charging pumps)	AB		
Low-head safety injection pumps (residual heat removal pumps)	EF		
Piping and valves required for injection to the RCS	EF, AB		
Instrumentation, cable and controls required to perform a safety function in the above portions of system	EF, AB, SB, CR		
Quench Spray System		6.2.2	6.2–36
Refueling water storage tank, quench spray pumps, piping and valves required to perform intended safety functions	EF, OY		
Instrumentation, cables and controls required to perform a safety function in the above portions of system	EF, OY, CR		
Emergency Generator Fuel Oil Storage and Transfer Systems		9.5.4	9.5–2

TABLE 3.5–1 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS OUTSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN* (CONTINUED)

Structures	Location**	FSAR Section	FSAR Figure
Emergency generator fuel oil day Tanks	DG		
Emergency generator fuel oil transfer pumps	DG fuel oil transfer Pump house		
Emergency generator fuel oil storage tanks	OY (Underground)		
Piping and valves required to perform a safety function	OY/DG		
Instrumentation, cable, and controls required to perform a safety function in the above portions of system	DB / OY IR / ER		
Emergency Generator Cooling Water System		9.5.5	9.5–3
Emergency generator cooling water exchanger	DG		
Cooling water electric immersion heater pressure housing	DG		
Emergency generator cooling water expansion tank	DG		
Motor driven circulating water pump	DG		
Piping and valves required to perform a safety function	DG		
Instrumentation and controls required to perform a safety function in the above portions of system	DG		
Emergency Generator Starting System		9.5.6	9.5–3
Emergency generator air starting storage tanks	DG		
Piping and valves required to perform a safety function	DG		
Instrumentation, cable, and controls required to perform a safety function in the above portions of system	DG		

TABLE 3.5–1 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS OUTSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN* (CONTINUED)

Structures	Location**	FSAR Section	FSAR Figure
Emergency Generator Lubrication System		9.5.7	9.5–3
Emergency generator lubrication oil cooler	DG		
Emergency generator lubrication oil filter	DG		
Piping and valves required to perform a safety function	DG		
Instrumentation and controls required to perform a safety function in the above portions of system	DG		
Emergency Generator Air Intake and Exhaust System		9.5.8	9.5–4
Air filter	DG		
Air intake silencer	DG		
Exhaust silencer	DG		
Piping required to perform a safety function	DG		
Service Water System		9.2.1	9.2–1
Service water pumps	SP		
Piping and valves supplying cooling water to safety related equipment	AB, EF, OY (buried)		
Instrumentation, cable, and controls required to perform a safety function in the above portions of system	EF, DG, CB		
Reactor Plant Component Cooling Water System		9.2.2.1	9.2–2
Reactor plant component cooling pumps	AB		
Reactor plant component cooling surge tank	AB		
Reactor plant components cooling heat exchangers	AB		
Piping and valves supplying cooling water to safety related equipment required for shutdown	AB, SB		

TABLE 3.5–1 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS OUTSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN* (CONTINUED)

Structures	Location**	FSAR Section	FSAR Figure
Instrumentation, cable and controls required to perform a safety function in the above portions of system	AB, CB		
Steam Generator Blowdown System		10.4.8	10.3–1
Piping and valves from containment up to and including the blowdown isolation valves	MS		
Instrumentation, cabling, and controls required to trip close the isolation valves	MS		
Main Steam System		10.3	10.3–1, 10.3–2, 10.3–3, 10.3–4
Main steam piping and valves from containment up to and including main steam isolation stop valves and all piping in the valve house	MS		
Main steam piping and valves from main steam lines to steam generator auxiliary feedwater pump turbine	MS		
Instrumentation, cable, and controls required to trip close the MS stop valves and bypass valves	MS / CR / IR		
Steam generator atmospheric relief valves (can be operated manually)	MS		
Auxiliary Feedwater System		10.4.9	10.4–9
Demineralized water storage tank	OY		
Auxiliary feedwater pumps (turbine- and motor-driven)	EF		
Piping and valves supplying auxiliary feedwater from ST to containment penetration	OY, EF		
Instrumentation, cable and controls required to perform a safety function in the above portions of system			
Feedwater System		10.4.7	10.4–6

TABLE 3.5–1 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS OUTSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN* (CONTINUED)

Structures	Location**	FSAR Section	FSAR Figure
Feedwater piping and valves from containment up to and including the feedwater isolation valves	MS		
Instrumentation, cable, and controls required to trip close the feedwater isolation valves and to trip the main feedwater pumps			
Air Conditioning, Heating, Cooling and Ventilation Systems		9.4	
Emergency generator enclosure ventilation	EGE	9.4.5	9.4–3
Engineered safety features area unit cooler and ductwork	EF / AB	9.4.2	9.4–2
Control building heating and ventilation, including control building chilled water system	CB	9.4.0	9.4–1
Service water pumphouse ventilation	SP	9.4.7	9.4–3
Instrumentation and control required to perform a safety related function for the above portions of system	EF / EGE / CB / AB		
Lighting System		9.5.3	None
Emergency lighting	AB/CR		
Electrical Systems		8.3	8.3–1 thru 8.3–3
Emergency generators	EGE		
Batteries, chargers, and battery switchboards	ER		
Vital bus panels and inverters	ER		
Essential unit substations	ER		
Essential metalclad switchgear	ER		
Essential motor control centers	ER		
Control panelboards:	CR		
Main control board and panels			
Emergency generator panel	ER		

TABLE 3.5–1 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS OUTSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN* (CONTINUED)

Structures	Location**	FSAR Section	FSAR Figure
Auxiliary shutdown panel	ER		
Control room air-conditioning control panel	CR		
Cabling and raceway supports for safety related equipment required for safe shutdown	All locations outside containment		
Motors for safety related equipment required for safe shutdown	Same as components		
Instrumentation and cables required for the safety related portion of the above electrical systems	All locations outside containment		
Reactor Trip Systems			
All portions of the reactor trip and safeguards actuation channels including sensors, circuitry, and processing equipment required to control and safety shut down the reactor to the cold shutdown condition (the protection circuits used to trip the reactor on undervoltage, underfrequency, and turbine trip are excluded)	CR / IR	7.2, 7.4	7.2–1 thru 7.2-4
Controls for defeating automatic safety injection actuation during a cooldown and depressurization	CR / IR	7.3, 7.4	None
Indication of the following plant parameters should be available to the operator:		7.4, 7.5, 7.7	7.7–1, 7.7–4, 7.7–5, 7.7–6
Indication of plant parameters required for safe shutdown (see Table 7.5–1)	CR/IR		
Containment Isolation System			
Containment isolation valves and associated piping for all systems penetrating containment structure	AB, EF, MS	6.2.4	6.2–47

TABLE 3.5–1 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS OUTSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN* (CONTINUED)

Structures	Location**	FSAR Section	FSAR Figure
Instrumentation, cable and controls required to perform a safety function in the above portions of system	AB		

NOTES:

* The applicable seismic category and quality group classification for the equipment listed in this table is delineated in Table 3.2–1.

** Location Symbols

AB - Auxiliary building
CB - Control building
CR - Control room (in control building)
CS - Containment structure
EGE - Emergency generator enclosure
EF - Engineered safety features building
ER - Emergency switchgear room (in control building)
IR - Instrument rack room (in control building)
MS - Main steam valve enclosures
SP - Service water pumphouse
OY - Outside, yard
AT - Auxiliary feedwater storage tank
HR - Hydrogen recombiner building
SB - Service building
FB - Fuel building

TABLE 3.5–2 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS INSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN*

Structures	Location **	FSAR Section
Containment Structure		3.8.1, 3.8.3
Reinforced concrete interior substructure	CS	
Reinforced concrete interior shields and walls	CS	
Containment structure liner	CS	
Piping and duct penetrations	CS	
Electrical penetration/assemblies	CS	
Personnel access lock	CS	
Equipment hatch	CS	
Systems		
Reactor Coolant System		5.1, 5.2, 5.3, 5.4
Reactor vessel	CS	
Control rod drive mechanisms	CS	
Steam generators	CS	
Pressurizer	CS	
Piping, fitting and valves within RCPB	CS	
Reactor coolant thermowell	CS	
Reactor coolant thermowell boss	CS	
Safety valves	CS	
Relief valves	CS	
Control rod drive mechanism head adapter plugs	CS	
Isolation valves for reactor coolant system branch line	CS	
Reactor coolant pump	CS	
Reactor coolant pump casing	CS	
Main flange		
Thermal barrier		
Systems	Location **	FSAR Section
No. 1 seal housing		

TABLE 3.5–2 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS INSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN* (CONTINUED)

No. 2 seal housing		
Bolting		
Impeller		
Diffuser		
Reactor coolant pump motor	CS	
Rotor		
Flywheel		
Shaft		
Shaft coupling		
Bearings		
Upper oil cooler	CS	
- Tube side - ccw		
- Shell side - oil		
Lower oil cooler	CS	
- Tube side - ccw		
- Shell side - oil		
Instrumentation cable and controls required to perform a safety function in the above portion of system	CS	
Chemical and Volume Control System		9.3.4
Piping, valves, and portions of the system required for charging, seal water injection, and boration	CS	
Instrumentation, cable, and controls required to perform a safety function in the above portion of system	CS	
Residual Heat Removal System		5.4
Piping and valves required to cool and maintain the RCS in a cold shutdown condition	CS	
Instrumentation, cable, and controls required to perform a safety related function in the above portions of system	CS	

TABLE 3.5–2 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS INSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN* (CONTINUED)

Εμεργενχψ Χορε Χοολινγ Σψστεμ		6.3
ESF pumps	CS	
Piping and valves required for injection to the RCS	CS	
Systems	Location**	FSAR Section
Instrumentation, cable and controls required to perform a safety related function in the above portions of system	CS	
Containment Isolation System		6.2.4
Containment isolation valves and associated piping for all systems penetrating containment structure	CS	
Instrumentation, cable and controls required to perform a safety related function in the above portions of system	CS	
Reactor Plant Component Cooling Water System		9.2.2.1
Piping and valves supplying cooling water to safety related equipment required for shutdown	CS	
Instrumentation, cable and controls required to perform a safety related function in the above portions of system	CS	
Auxiliary Feedwater System		10.4.9
Piping and valves supplying auxiliary feedwater to the steam generators	CS	
Feedwater System		10.4.7
Piping and valves supplying auxiliary feedwater to the steam generator	CS	
Air-Conditioning, Heating, Cooling, and Ventilation Systems		9.4.6
Containment recirculation unit coolers and associated equipment excluding chillers	CS	
Instrumentation, cable, and controls required to perform a safety related function for the above air-conditioning, heating, cooling and ventilation systems	CS	

TABLE 3.5–2 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS INSIDE CONTAINMENT REQUIRED FOR SAFE REACTOR SHUTDOWN* (CONTINUED)

Electrical Systems		8.3
Cabling and raceways for safety related equipment required for safe shutdown	CS	
Motors for safety related equipment required for safe shutdown	CS	
Systems	Location **	FSAR Section
Reactor Trip System		See Table 3.5–1
Those portions of the reactor trip system listed in Table 3.5–1 that are located inside containment	CS	

NOTES:

* The applicable seismic category and quality group classification for the equipment is listed in Table 3.2–1.

** Location Symbols

CS - Containment Structure

TABLE 3.5–3 SUMMARY OF CONTROL ROD DRIVE MECHANISM MISSILE ANALYSIS

	Typical Examples of Postulated Missiles	Missile Weight (lb)	Impact Velocity (ft/sec)	Kinetic Energy (ft-lb)	Penetration (in)	Assumptions
1.	Mechanism housing cap	11	90	1,380	0.05	Plug becomes loose and is accelerated by the water jet
2.	Mechanism top cap and drive rod assembly impacting on the same missile shield spot	133	150	46,757	0.08	Drive shaft further pushes the plug into the shield
3.	Drive shaft latched to mechanism	1,500	12.1	1,490	0.057	-

TABLE 3.5–4 VALVE-MISSILE CHARACTERISTICS*

Missile Description	Weight (lb)	Flow Discharge Area (in^2)	Thrust Area (in^2)	Weight to Impact Area (in^2)	Impact Area Ratio (psi)	Velocity (fps)
Safety valve bonnet (3 inches x 6 inches) or 6 inches x 6 inches	350	2.86	80	24	14.6	110
3-inch motor-operated isolation valve bonnet (plus motor and stem)	400	5.5	113	28	14.1	135
3-inch solenoid-operated relief valve bonnet (plus stem)	75	1.8	20	20	3.75	115
4-inch air-operated spray valve	200	9.3	50	50	4	190

NOTE:

* Typical values based upon manufacturer design data

TABLE 3.5–5 PIPING TEMPERATURE ELEMENT ASSEMBLY - MISSILE CHARACTERISTICS

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

Characteristics	Without Well	With Well
Flow Discharge Area (in^2)	0.11	0.60
Thrust Area (in^2)	7.1	9.6
Missile Weight (lb)	11.0	15.2
Area of Impact (in^2)	3.14	3.14
Missile Weight Impact Area (psi)	3.5	4.84
Velocity (ft/sec)	20.0	120.0

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

Characteristics	Without Well	With Well
Discharge Area (in^2)	0.11	0.60
Thrust Area (in^2)	3.14	3.14
Missile Weight (lb)	4.0	6.1
Area of Impact (in^2)	3.14	3.14
Missile Weight Impact Area (psi)	1.27	1.94
Velocity (ft/sec)	75.0	120.0

TABLE 3.5–6 CHARACTERISTICS OF OTHER MISSILES POSTULATED WITHIN REACTOR CONTAINMENT

Characteristics	Reactor Coolant Pump Temperature Sensor	Instrument Well of Pressurizer	Pressurizer Heaters
Weight (lb)	0.25	5.5	15
Discharge Area (in^2)	0.50	0.442	0.80
Thrust Area (in^2)	0.50	1.35	2.4
Impact Area (in^2)	0.50	1.35	2.4
Missile Weight			
Impact Area (psi)	0.5	4.1	6.25
Velocity (ft/sec)	260	100	55

TABLE 3.5–7 TURBINE-TARGET DISTANCES AND IMPACT AREAS

	High Trajectory Targets	Target Area (sq ft)	R * (ft)	y ** (ft)
1.	Control Building	11,730	216	15.5
2.	Emergency Generator Enclosure	3,999	303	-13.0
3.	Fuel Oil Storage Tanks	1,736	301	-44.5
4.	Auxiliary Building	15,975	265	24.5
5.	Main Steam Valve Building	1,856	171	14.9
6.	Fuel Building	9,701	346	37.0
7.	Engineered Safeguards Building	8,241	333	24.0
8.	Demineralizer Water Storage Tank	1,133	423	11.0
9.	Refueling Water Storage Tank	2,641	406	87.0
10.	Hydrogen Recombiner Buildings	720	307	-17.7
11.	Containment Structure	8,721	256	108.3
12.	Service Water Pumps	2,750	570	-45.0

	Low Trajectory Targets	Target Area (sq ft)	R * (ft)	y ** min (ft)	y ** max (ft)
1.	Auxiliary Building Wall ***	0	234	14.9	24.5
2.	Main Steam Valve Building Wall	1,560	157	-12.0	14.9
3.	Refueling Water Storage Tank Wall	816	383	-17.7	18.0
4.	Hydrogen Recombiner Building Wall	819	295	-45.0	-17.7
5.	Containment Wall - Section 1	2,657	188	14.9	86.9
6.	Containment Wall - Section 2	2,372	188	-45.0	86.9

NOTES:

* R = average distance from turbine
** y = height above spin axis of turbine
*** Auxiliary building wall can be hit by missiles from only one wheel

TABLE 3.5–8 DELETED BY FSARCR 02-MP3-13

TABLE 3.5–9 DELETED BY FSARCR 02-MP3-13

TABLE 3.5–10 DELETED BY FSARCR 02-MP3-13

TABLE 3.5–11 DELETED BY FSARCR 02-MP3-13

TABLE 3.5–12 DELETED BY FSARCR 02-MP3-13

TABLE 3.5–13 POSTULATED TORNADO-GENERATED EXTERNAL MISSILES

Missile Description *	Weight (lbs)	Horizontal Velocity ** (mph)
1. 4 x 12 in plank, 12 ft long at 50 lb/ft^3	200	210
2. Utility Pole 13.5 inch diameter, 35 feet long at 43 lb/ft^3	1,496	120
3. 1 inch solid steel rod, 3 feet long	8	130
4. 3 inch schedule 40 pipe, 10 feet long	78	140
5. 6 inch schedule 40 pipe 15 feet long	285	120
6. 12 inch schedule 40 pipe, 15 feet long	743	110
7. Automobile frontal area 20 ft^2	4,000	50

NOTES:

* Missiles 1 through 6 are considered at all elevations. Maximum trajectory height of Missile 7 is 25 feet above grade.

** Vertical velocities of 70 percent of horizontal velocities are used, except for Missile 3 which has the same speed in all directions.

TABLE 3.5–14 BARRIER DEFLECTION AND DUCTILITY RATIOS FOR TORNADO-BORNE MISSILES PLUS 360 MPH TORNADO WIND

Missile Description	Mass (slugs)	Velocity (fps)	Momentum (lb sec)	Barrier Span (ft)	Maximum Elastic Barrier Deflection (Xy) (ft)	Barrier Deflection (Xm) (ft)	Barrier Ductility
12 inch schedule 40 pipe 15 feet long 743 lb	23.07	161.33	3721.96	5.	0.0038	0.0069	1.79
				10.	0.0154	0.0266	1.73
				20.	0.0615	0.0398	0.65
				30.	0.1383	0.0489	0.35
4,000 lb auto	124.2	73.33	9108.	5.	0.0038	0.0295	7.68
				10.	0.0154	0.0202	1.32
				20.	0.0615	0.0414	0.67
				30.	0.1383	0.0843	0.61
13.5 inch diameter utility pole 35 feet long 1,496 lb	46.46	176	8177	5.	0.0038	0.0031	0.80
				10.	0.0154	0.0122	0.79
				20.	0.0615	0.0499	0.81
				30.	0.1383	0.0871	0.63

Barrier Information:

Reinforcement: #11 at 10 inches GR 40 each way, each face

Concrete: 3,000 psi

Barrier Depth: 24 inches

TABLE 3.5–15 BARRIER DEFLECTION AND DUCTILITY RATIO FOR A BEAM-COLUMN PLUS 360 MPH TORNADO WIND

Barrier Description	Missile Description	Mass (slug)	Velocity (fps)	Momentum (lb sec)	Maxim Barrier Elastic Deflection (ft)	Maximum Barrier Deflection (ft)	Barrier Ductility Ratio	Barrier Compression Load (#)	Maximum Allowable Ductility Ratio
Refer to Figure 3.5–6	13.5 inch diameter utility pole 35 ft long	46.46	176	8177	0.03	0.03	10.	Barrier dead load only	10.

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.5–1 TURBINE PLACEMENT AND ORIENTATION FOR THREE-UNIT SITE

[redacted]

FIGURE 3.5–2 TURBINE PLACEMENT AND ORIENTATION PROFILE

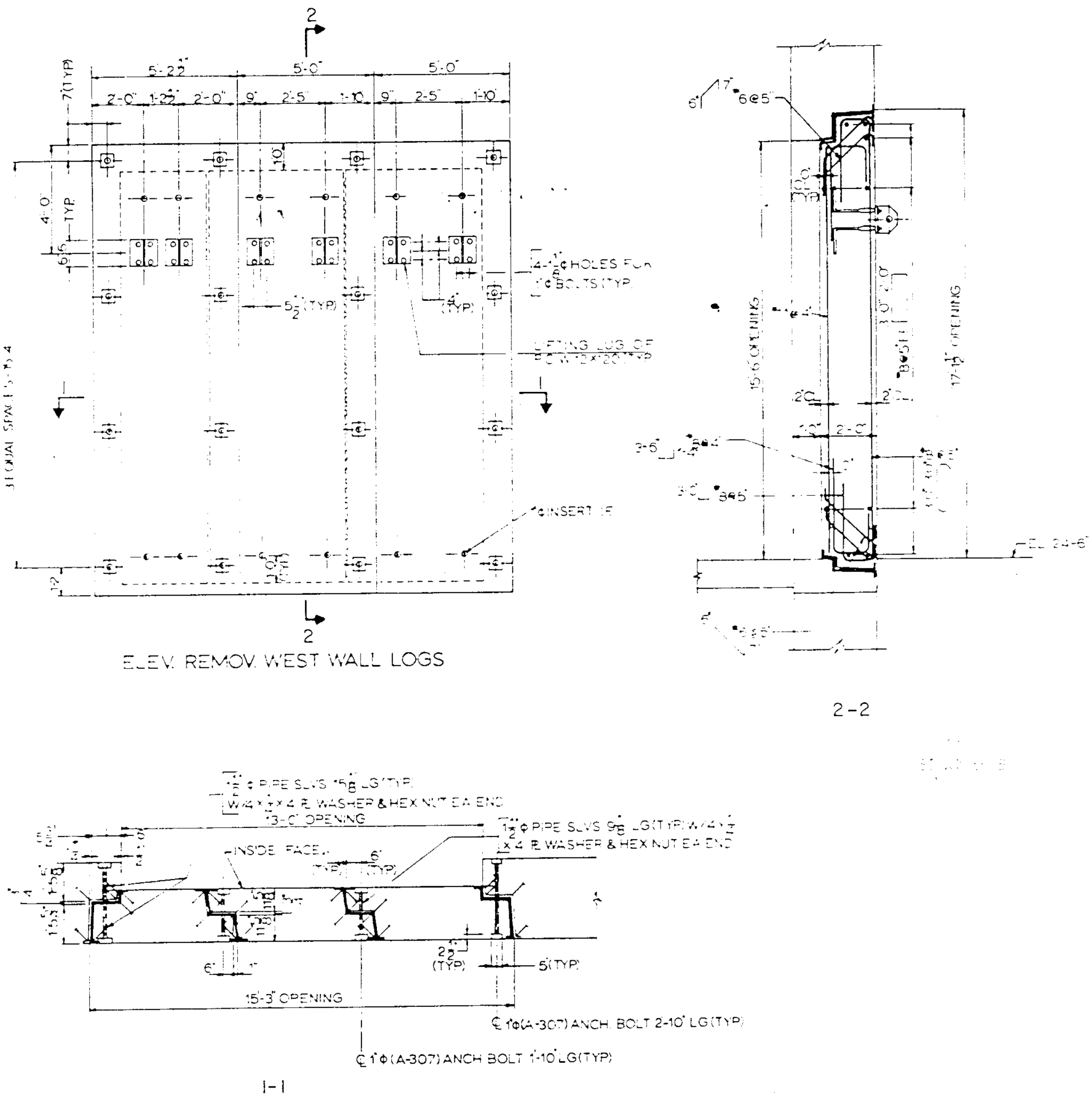
Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.5–3 DELETED

FIGURE 3.5–4 DELETED

FIGURE 3.5–5 DELETED

FIGURE 3.5–6 MISSILE BARRIER BEAM COLUMN



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

This section addresses the subject of a postulated pipe break/crack inside and outside the containment. It also presents the results of analyses initiated in response to NRC Regulatory Guide 1.46 for inside containment and the Giambusso letter of December 18, 1972, for outside containment. The methods of evaluation, however, reflect the approach and methodology contained in the Branch Technical Positions ASB 3-1 and MEB 3-1 as qualified in Sections 3.6.1 and 3.6.2.

3.6.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS INSIDE AND OUTSIDE OF CONTAINMENT

This section describes the design criteria and bases for protecting essential3 equipment from the effects of piping failures inside and outside of containment. These criteria and bases used in the design ensure that:

1. Functions of essential equipment necessary to maintain the capability of a safe plant shutdown during any piping failures are preserved.

2. The plant can be safely shut down to a cold shut down condition without offsite power.

3. Offsite doses in excess of applicable guidelines do not occur.

As used in this section, essential equipment is defined as the structures, systems, portions of systems, and components required to mitigate the effects of the postulated pipe rupture, to shut down the plant safely to a cold condition without offsite power, and to maintain the cold shutdown condition assuming a concurrent single active failure.

3.6.1.1 Design Bases

3.6.1.1.1 Design Basis Protection Criteria

Within the plant there are high and moderate energy piping systems which are postulated to fail. Failures of these systems are evaluated to determine their deleterious effects on essential equipment and those essential systems, components, and structures that are susceptible to the effects of these failures are provided protection. The criteria for protection against pipe rupture inside the containment are contained in NRC Regulatory Guide 1.46, as described in Section 1.8, the criteria for protection against pipe rupture outside containment conform to the NRC Technical Branch Position ASB 3-1. Section 3.6.1.1.4 discusses design features recommended for protection.

3.6.1.1.2 Design Basis Pipe Break/Crack Criteria

The design basis pipe breaks are postulated to occur in high energy systems (or portions of systems) in accordance with the break criteria (Section 3.6.2.1). A high energy system is defined as a fluid system which operates during normal plant operating conditions and has a maximum operating pressure exceeding 275 psig or a maximum operating temperature exceeding 200°F. Figure 3.6–1 presents these boundaries graphically, including the differences between inside and outside the containment.

Maximum operating temperature and pressure are the maximum temperature and pressure in the fluid systems during occurrences which are expected during normal plant conditions. Normal plant conditions are defined as startup, operation at power, hot standby, or reactor cooldown to cold shutdown conditions.

Through-wall leakage cracks are postulated to occur in moderate energy systems (located outside containment only) in accordance with Sections 3.6.2.1.2 and 3.6.2.1.3.

A moderate energy system is defined as a fluid system or portion of a fluid system which is pressurized during normal plant operating conditions but which has both a maximum operating pressure equal to 275 psig or less and a maximum operating temperature equal to 200°F or less.

3.6.1.1.3 Essential Systems, Components, and Structures

Table 3.6–5 lists all essential equipment required to shut down the plant safely and to mitigate the consequences of postulated piping failures. Figures 3.6–2 through 3.6–7 show the relative locations of essential equipment within the plant. Thus, a determination can be made of which high and moderate energy systems are remote from or proximate to essential systems, components, and structures.

3.6.1.1.4 Design Approach

The following subsections describe the design features provided to protect essential systems, components, and structures and to mitigate the consequence of piping failures.

3.6.1.1.4.1 Separation

A primary objective in the piping layout and plant arrangement is to satisfy the separation criteria, so that the effects of postulated pipe breaks at any location are isolated or physically remote from essential structures, systems, and components.

Redundant essential equipment is located in separate cubicles so that a pipe failure in one cubicle cannot affect the backup equipment. Inside each redundant cubicle, high energy piping is located with as much separation as possible from the essential equipment so that the effects of pipe whip or jet impingement are minimized.

Whenever possible, safety related electrical and control system equipment is located in areas which do not have piping (e.g., the emergency switchgear area). Cable tray runs and instrumentation process lines which cannot be separated from piping runs are located as far as practical from postulated piping failure locations. Every attempt is made to physically separate redundant cable runs and instrumentation lines. For all safety components where physical separation is impossible, one or more of the design features described below are provided to ensure that essential equipment remains operable.

3.6.1.1.4.2 Enclosures

Where remote physical separation is not practical, protective enclosures are used. For example, each charging pump is located in a separate cubicle with concrete walls between redundant pumps. Redundant electrical and control systems in the area of high energy piping are located as far as possible from postulated pipe break locations so that potential damage is minimized. Section 3.8.4 discusses the concrete design of these enclosures.

3.6.1.1.4.3 Restraints

Where neither of the above design approaches is feasible, protection is provided by pipe restraints to limit pipe movement following a rupture to restrict the area affected by the failure. Placement of restraints is based on postulated pipe break locations. The design of the restraints for these locations limits pipe whip to acceptable levels and reduces jet impingement thereby preventing damage to essential equipment. Section 3.6.1.3.3 describes the design of pipe rupture restraints.

3.6.1.2 Description

Every attempt, whenever practical, is made to satisfy the separation criteria by locating high and moderate energy systems physically remote from essential equipment. Where plant arrangement and/or piping layout does not permit complete physical separation, protective enclosures are provided. When these methods are not practical, restraints are installed to protect essential equipment.

Table 3.6–1 lists all high energy systems remote from essential systems, components, and structures. A major portion of these systems is in the turbine building where no essential equipment is located, thus satisfying the physical separation criterion. Pipe breaks are only postulated for the main steam line in this area due to its proximity to the control building.

Table 3.6–2 summarizes all high energy systems that are located in proximity to essential systems, components, and structures. The majority of essential systems, and components are found in the Engineered Safety Features, Auxiliary Building, and the Containment Structure. Pipe breaks and through wall leakage cracks are postulated in systems that are proximate to these essential systems, and components.

Table 3.6–3 contains all moderate energy systems outside the containment separated from essential systems, components, and structures by plant arrangement and/or piping layout. Therefore, these systems are not evaluated for pipe rupture.

Table 3.6–4 includes all moderate energy systems located outside containment in proximity to essential systems, components, and structures. Through-wall cracks are postulated in these systems or portions of systems to evaluate the effects of fluid spray and flooding. Jet impingement effects from cracks are not evaluated.

Each system listed in Table 3.6–2 is referenced by pipe break location drawing (figure number), as applicable, showing all postulated pipe breaks. Section 3.6.1.3.3 provides a summary of all postulated pipe breaks. It encompasses their detrimental effects on essential equipment along with single active failure criteria, loss of offsite power, seismic events, and all other design considerations.

3.6.1.3 Safety Evaluation

3.6.1.3.1 Operability of Essential Systems and Components

The operability of an essential system or a component following a postulated piping failure may be dependent on one or more of the following assumptions in addition to consideration of the effects of the failure.

Single Failure Criterion

A single active component failure is assumed to occur in essential systems used to mitigate consequences of the postulated piping failure and to shut down the reactor. The single active component failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure, such as unit trip and loss of offsite power. Section 3.1.1 defines this failure criterion and its applications.

Loss of Offsite Power

Offsite power is assumed unavailable if a trip of the turbine generator system or reactor protection system is a direct consequence of the postulated piping failure. However, a single failure of one emergency generator or one Class 1E bus can be assumed as the single failure if this assumption is the most limiting.

Seismic Event

Credit for mitigating the consequences of a postulated event may be taken only for those systems and components designed to Seismic Category I requirements, except for the High Energy Line Break (HELB) isolation systems for the auxiliary steam and the hot water heating systems. Although the electrical sensing and isolation devices are Category I and located in a Category I building, the valves are not Category I as they are installed in a non-Category I piping system located in a non-Category I structure. The location of the isolation valves provides for maximum isolation capability outside the area for which isolation is intended.

All available systems, including those actuated by operator actions, are used to mitigate the consequences of a postulated event. Judging the availability of systems includes consideration of the postulated failure and its direct consequences (e.g., unit trip and loss of offsite power) and the assumed single active component failure plus its direct consequences. The feasibility of the operator to take action is judged on the availability of ample time and adequate access to equipment for performing the proposed actions. Regulatory Guide 1.62 provides guidance in evaluating the feasibility of operator action.

3.6.1.3.2 Failure Mode and Effects

An analysis of breaks in high energy systems, cracks in moderate energy systems, and the consequent failure modes and effects (e.g., environmental, pipe whip, and jet impingement) must include consideration of their sources and targets. The source comprises the pipe which is postulated to fail and the resulting effects of the failure. The target comprises structures, systems, and components considered essential for shutting down the plant safely, maintaining the safe shutdown, and mitigating the effects of the postulated pipe failure.

Interactions between sources and targets are analyzed individually to determine how each affects essential equipment in the area of the source. The interactions analyzed are pipe whip, jet impingement, and environmental effects.

Pipe Whip

Section 3.6.2.1 describes the criteria for determining break locations in piping systems. Section 3.6.2.2 describes the pipe whip analysis for defining forcing functions and pipe responses. The effects of a particular piping failure are evaluated for each target required to mitigate the consequences of the failure. This evaluation is based on the location of the break and the forces generated by the whipping pipe. The targets impacted by the pipe are either designed to withstand these forces or are protected by rupture restraints (Section 3.6.1.3.3).

Jet Impingement

The blowdown forces are calculated (Section 3.6.2.2), and the location and direction of the resulting jet of fluid determined, for each piping failure which could affect an essential target. Essential targets impacted by the fluid must either be evaluated for effects of these jet forces or shielded as discussed in Section 3.6.2.2.

Environmental Effects

The environmental effects of postulated pipe breaks are established as design basis environmental conditions for essential equipment affected by these pipe breaks and are included as environmental qualification criteria for the essential equipment. Section 3.11 describes the environmental qualification of essential equipment. The following environmental effects are considered:

1. Fluid spray - Essential equipment located proximate to high and moderate energy fluid systems are either designed to withstand the most severe environmental conditions resulting from fluid spray without loss of safety function or the most

severe environmental conditions resulting from fluid spray which are humidity of 100 percent, maximum expected temperature due to the ruptured fluid system, and water spraying essential equipment.

2. Flooding - Compartments and areas containing essential equipment are examined for flooding potential. Flooding height is based on the leak rate from the piping failure and the time required to detect and isolate the leak. Compartment floor and wall structural integrity is maintained when subjected to the hydrostatic head resulting from the calculated flood level.

3. Pressure buildup in compartments - The calculation of pressure and temperature buildup in compartments is based on the maximum operating fluid conditions in the pipe which fails, the available vent area in the cubicle, and the volume within the cubicle. Pipe failure locations are determined as described in Section 3.6.2. The results of the pressure calculations indicate the required pressure differential design or the vent area required for building compartments. Maximum calculated temperatures are used to determine essential equipment qualification requirements.

3.6.1.3.3 Pipe Break/Crack Analysis

Each pipe break and crack postulated in the high and moderate energy systems, listed in Tables 3.6–2 and 3.6–4, have been systematically analyzed for their detrimental effects on safe shutdown targets. The following paragraphs describe in detail the interactions between the targets and the postulated breaks and cracks in each of the piping systems identified herein:

1. Main Steam System

2. Feedwater System

3. Reactor Coolant System

4. Chemical and Volume Control System

5. Steam Generator Blowdown System

6. Auxiliary Feedwater System

Main Steam System

Break Locations

Pipe breaks in the main steam piping inside the containment are postulated at terminal ends and intermediate locations defined in Section 3.6.2.1.1. The main steam lines from steam generators 3RCS*SG1A and 3RCS*SG1D up to the containment penetrations are mirror-images of each other; therefore, pipe breaks are postulated at identical locations for both lines. Similarly, the other pair of main steam lines originating from steam generators 3RCS*SG1B and 3RCS*SG1C are symmetrical and the pipe break locations in the individual line are identical. Only one line of each pair is evaluated for the effects of the postulated piping failure; however, the results are also applicable to the other identical lines.

The portion of main steam piping between the containment penetration (inboard side) and the first restraint outboard of the isolation valve is considered a break exclusion zone. The stress distribution along this section of the main steam piping satisfies the break exclusion requirement (ASME Code, Section III, NC-3600; also Section 3.6.2):

$$\text{Eq. (9)} + \text{Eq. (10)} \leq 0.8(1.2S_h + S_A) \qquad (3.6.1)$$

The remainder of the main steam piping is physically located remote from essential systems, components, and structures except for a relatively short section near the control building. However, breaks are postulated in accordance with Section 3.6.2.1.2 to ensure that protection is provided to preserve the integrity of the containment penetrations and the operability of the isolation valves in the event that pipe rupture should occur at any of the postulated locations. Consideration is given to potential damage to the control building wall which may adversely affect the safety related equipment in the control room. Figures 3.6–8 and 3.6–9 show all postulated pipe breaks inside and outside of the containment. Table 3.6–6 lists the postulated break locations.

No single active failure has been assumed concurrent with breaks which are postulated in the break exclusion zone since the break is postulated to evaluate environmental effects only.

Separation

As mentioned above, the main steam piping outside the containment, extending from the main steam valve building wall (Col. F, Figure 3.6–8) to the turbine building satisfy the physical separation criterion. Pipe breaks, however, are postulated in this region to assure that occurrence of such breaks do not jeopardize the integrity of the containment penetrations and operability of the isolation valves and essential equipment located in the control building.

Pipe Whip Effects

Table 3.6–7 summarizes the results of pipe whip analysis inside and outside of containment. To prevent whipping of the main steam piping inside the containment, whip restraints are provided to protect the crane wall and the adjoining containment wall. Similarly, no whipping is permitted for the main steam piping outside containment to prevent damage to containment penetrations, isolation valves, and control building wall by placing restraints at specified locations shown on Figure 3.6–8. Details of these restraints are discussed in the next section.

Pipe Restraints

The pipe rupture restraints on the Main Steam System are shown on Figure 3.6–8 along with postulated pipe rupture locations. Notes to Figure 3.6–8 provide location, type, and function for each pipe rupture restraint.

Jet Impingement Effects

Jet impingement effects are addressed in Table 3.6–8.

Environmental Effects

Refer to Section 3.11B.1 for information.

Conclusion

Adequate protection and/or separation is provided for essential systems and components required to mitigate the consequences of main steam piping ruptures.

Essential systems, structures, and components remain functional subsequent to a postulated main steam line break.

Feedwater System

Break Locations

Pipe breaks in the feedwater piping inside the containment are postulated at terminal ends and intermediate locations in accordance with Section 3.6.2.1.1.

Pipe breaks are excluded in the main steam valve building between the containment penetration (inboard side) and the first restraint on the 20-inch feedwater lines and between the tap into the 20-inch line and the first restraint on the 8-inch bypass line, inasmuch as the stress distribution along this portion of the feedwater lines fully met the break exclusion requirement (Equation 3.6-1).

The portion of feedwater lines upstream of the break exclusion zone are postulated at terminal ends and intermediate locations in accordance with Section 3.6.2.1.2. Figures 3.6–10 and 3.6–11 show all the terminal and intermediate breaks inside and outside of containment. Table 3.6–9 lists the postulated break locations.

Separation

The feedwater piping upstream of the break exclusion zone, extending from the main steam valve building wall (Col. F) to the turbine building fully meet the physical separation criterion, thereby precluding potential damage to essential equipment. Pipe breaks, however, are postulated to evaluate the effects of these breaks on the break exclusion zone to ensure that the integrity of the containment penetrations and the operability of the isolation valves are not lost in the event that this piping should fail.

Pipe Whip Effects

Table 3.6–10 summarizes the results of pipe whip analysis inside and outside containment. Whipping of feedwater piping inside the containment is not allowed except in the area where the whipping pipe cannot potentially damage an essential system, component, or structure. Such a case is confined only to the terminal breaks at the steam generator feedwater nozzles. The relatively short section of the feedwater piping in the turbine building is separated from the control building wall by approximately 40 feet at their nearest approach. A whipping feedwater pipe can not impact the wall.

Pipe Restraints

The pipe rupture restraints on the Main Feedwater System are shown on Figure 3.6–11. The postulated pipe rupture locations are shown on Figure 3.6–10. Notes to Figure 3.6–11 provide location, type, and function for each pipe rupture restraint.

Jet Impingement Effects

Jet impingement effects are addressed in Table 3.6–11.

Environmental Effects

Refer to Section 3.11 for information.

Conclusion

Adequate protection and/or separation is provided for essential systems and components required to mitigate the consequences of piping ruptures.

<u>Reactor Coolant System</u>

Determination of the design basis break locations in the reactor coolant system are based on the criteria defined in Regulatory Guide 1.46. It takes into account certain modifications or exceptions specified in Section 1.8 and qualified in Section 3.6.2.1. For clarity of analysis, discussion of breaks and their consequential effects fall into three separate areas: 1. primary coolant piping (including Loop Stop Valve Bypass Piping, Excess Letdown Piping, Loop Drain Piping, Letdown Line, and Normal Charging Line) 2. pressurizer cubicle piping, and 3. safety injection system.

1. Primary Coolant Piping

Break Locations

Figures 3.6–12 and 3.6–33 show all pipe breaks in the primary coolant piping at postulated locations defined in Section 3.6.2.1. Each break location and orientation is determined on the basis of stress and fatigue analysis. Table 3.6–12 summarizes the postulated break locations. Only one loop (loop A) of the four identical loops is analyzed; however, results of analysis are applicable to all loops by virtue of their symmetry.

The mechanistic effects of pipe breaks in the RCS hot leg, cold leg, and crossover leg need not be addressed due to the GDC4 exemption for primary loop pipe breaks.

Separation

Protection criteria dictate physical separation of each of the reactor coolant loops in order to assure the continued integrity and operability of essential systems, components, and structures required to safely shut down the plant in the event of a loss of coolant accident, main steam, or feedwater break. This protection requirement is achieved by isolating each reactor coolant loop within a cubicle. Section 3.8.1 discusses the design of the cubicle.

Pipe Whip Effects

Table 3.6–13 summarizes the results of pipe whip analysis of the reactor coolant loop bypass piping. Whipping of the ruptured coolant bypass piping originating at any postulated break locations jeopardizes the integrity and operability of the safe-shutdown targets identified in the table. To meet the relevant protection criterion whip restraints are installed at locations where whipping is prevented. Details of these restraints are as follows.

Pipe Restraints

The pipe rupture restraints on the Reactor Coolant System Primary Coolant Piping are shown on Figure 3.6–12. The postulated pipe rupture locations are shown on Figure 3.6–12 for the Loop Stop Valve Bypass lines. Notes to Figure 3.6–12 provide location, type, and function for each pipe rupture restraint.

The pipe rupture restraints originally installed to prevent primary loop pipe whip have been deactivated due to the GDC4 exemption.

Jet Impingement Effects

Jet impingement effects are addressed in Table 3.6–14.

Environmental Effects

Refer to Section 3.11 for information.

Conclusion

Adequate protection and/or separation is provided for essential systems and components required to mitigate the consequences of RCS loop (including Loop Stop Valve Bypass Piping, Excess Letdown Piping, Loop Drain Piping, Loop Fill Piping, Letdown Line, and Normal Charging Line) piping ruptures.

2. Pressurizer Cubicle Piping

Break Locations

Figure 3.6–14 shows the design basis breaks postulated in accordance with the rules specified in Section 3.6.2.1. Table 3.6–15 summarizes the postulated break locations.

Separation

The pressurizer tank located in proximity to the reactor coolant loop is enclosed in a cubicle to satisfy the protection criterion. The pressurizer surge line, however, penetrates the cubicle wall as it connects to the hot leg of the reactor coolant loop (Loop B). Terminal and intermediate breaks are postulated in the surge line to assess their effects (pipe whip, jet impingement and environment) on the integrity of the pressurizer tank, pressurizer cubicle, and the reactor coolant system components located within the immediate vicinity of the postulated breaks.

Pipe Whip Effects

Results of pipe whip analysis are summarized in Table 3.6–16.

Pipe Restraints

The pipe rupture restraints on the Reactor Coolant System Pressurizer Cubicle Piping are shown on Figure 3.6–14. The postulated pipe rupture locations are shown on Figure 3.6–14. Notes to Figure 3.6–14 provide location, type, and function for each pipe rupture restraint.

Jet Impingement Effects

Jet impingement effects are addressed in Table 3.6–17.

Environmental Effects

Refer to Section 3.11 for information.

Conclusion

Adequate protection and/or separation is provided for essential systems and components required to mitigate the consequences of the Reactor Coolant System - Pressurizer Cubicle Piping ruptures.

3. Safety Injection System

The safety injection system consists of the following major components:

1. Four accumulators which discharge borated water into each cold leg of the reactor coolant system.

2. Two charging pumps which supply borated water for cooling to each cold leg of the reactor coolant system (SI mode of CVCS).

3. Two safety injection pumps which supply borated water to each cold and hot leg of the reactor coolant system.

4. Two residual heat removal pumps which supply borated water for core cooling to each cold leg and two hot legs when the reactor coolant system pressure is low.

All of these components interconnect with the reactor coolant system which operate following a loss-of-coolant accident. Where the safety injection lines connect to the reactor coolant piping, the pressure boundary consists of the piping and valves leading to these connecting lines up to and including the second valve in each line. Pipe break postulation is confined only within the defined pressure boundary of the reactor coolant system with the remainder of the connecting systems excluded from the effects of pipe rupture.

Break Locations

Figure 3.6–13 shows all postulated break locations within the reactor coolant pressure boundary except for the accumulator piping where break postulation is extended beyond the pressure boundary up to the accumulator tank discharge nozzle. Note that each pair of the accumulator tanks is located 180 degrees apart and that the main piping runs from each pair are symmetrical to each other. Only one piping run is analyzed for pipe breaks and their consequential effects; however, the results are applicable also to the remaining identical runs. Table 3.6–18 lists the postulated break locations.

Separation

The safety injection accumulator tanks are conveniently located at the basement level of the containment structure, physically remote from high energy systems. Protection of each accumulator tank and its associated piping and valves from the effects of failure in the safety injection piping is further enhanced by judicious pipe routing, resulting in adequate physical separation.

Pipe Whip Effects

Table 3.6–19 summarizes the pipe whip analysis.

Pipe Restraints

The pipe rupture restraints on the Safety Injection System are shown on Figure 3.6–13. The postulated pipe rupture locations are shown on Figure 3.6–13. Notes to Figure 3.6–13 provide location, type, and function for each pipe rupture restraint.

Jet Impingement Effects

Jet impingement effects are addressed in Table 3.6–20.

Environmental Effects

Refer to Section 3.11 for information.

Conclusion

Adequate protection and/or separation is provided for essential systems and components required to mitigate the consequences of safety injection system piping ruptures.

Chemical and Volume Control System

The chemical and volume control system consists of charging, letdown, and seal water system; chemical control, purification, and makeup system; and boron thermal regeneration system.

Only the charging, letdown, and seal water system is systematically analyzed for pipe rupture, since it qualifies as a high energy system. Although the remaining systems function in conjunction with the charging and letdown lines, they are moderate-energy systems by definition.

Summaries for postulated pipe breaks and dynamic effects for the charging, letdown, and seal water subsystems of the CHS are given in Tables 3.6-21 through 3.6-29.

a. CHS Charging Lines:

Break Locations

Figure 3.6–15 illustrates the design basis break locations.

Separation

The charging lines inside and outside the containment fully meet the separation criteria, except the relatively short section that connects to the reactor coolant system. Separation of the charging lines from the essential equipment is achieved as follows:

Each charging pump is isolated in a cubicle so that a piping failure in one cubicle does not affect the other pumps and their associated piping and valves.

A portion of the charging line inside the containment is enclosed within the regenerative heat exchanger cubicle and the continuing piping run in the annulus area is physically routed away from safety related essential equipment. Similarly, the charging lines in the auxiliary building are routed in an area which precludes potential damage to essential equipment.

Pipe Whip Effects

Since the separation criterion is fully met in the regenerative heat exchanger cubicle, the annulus area, and in the auxiliary building, prevention of pipe whip to mitigate its consequential effects in these areas is unnecessary. Table 3.6–22 provides the results of the pipe whip effects for the postulated pipe breaks on the CHS - Charging Line.

Pipe Restraints

The pipe rupture restraints on the CVCS charging line are shown on Figure 3.6–15 along with the postulated pipe rupture locations. Notes to Figure 3.6–15 provide location, type, and function for each pipe rupture restraint.

Jet Impingement Effects

Jet impingement effects are addressed in Table 3.6–23.

Environmental Effects

Refer to Section 3.11 for information.

b. CHS Letdown Line:

Break Locations

Pipe breaks in the CHS letdown line are postulated at terminal ends and intermediate locations (Figure 3.6–16). However, breaks are excluded in the portion of letdown piping from the containment penetration to and including the containment isolation valves to ensure that Letdown Line breaks may be isolated assuming the most limiting single active failure as discussed in Chapter 15, Section 15.6.2. The stresses within this portion of the letdown line satisfy the break exclusion requirement (Equation 3.6-1).

Separation

The letdown line inside and outside the containment is located with as much separation as possible from essential equipment by routing it in the regenerative heat exchanger cubicle, continuing in the annulus area as it penetrates the containment into the auxiliary building.

Pipe Whip Effects

The effects of pipe whip have been mitigated by the installation of rupture restraints. Table 3.6–25 provides the results of pipe whip effects for the CHS-Letdown Line.

Pipe Restraints

The pipe rupture restraints on the CHS letdown line are shown on Figure 3.6–16 along with the postulated pipe rupture locations. Notes to Figure 3.6–16 provide location, type, and function for each pipe rupture restraint.

Jet Impingement Effects

Jet impingement effects are addressed in Table 3.6–26.

Environmental Effects

Refer to Section 3.11 for information.

Conclusion

Adequate protection and/or separation is provided for essential systems and components required to mitigate the consequences of CHS charging and letdown piping ruptures.

c. CHS Seal Water Injection Line

Break Locations

Pipe breaks in the CHS seal water injection lines are postulated at terminal ends and intermediate locations and these postulated pipe break locations are shown on Figure 3.6–17 and listed in Table 3.6–27.

Separation

The seal water injection line, inside and outside containment, is located with as much separation as possible from essential equipment by routing it in the annulus area as it penetrates the containment to the auxiliary building.

Pipe Whip Effects

Table 3.6–28 indicates pipe whip effects. Pipe whip of the seal water injection lines, subsequent to a postulated pipe break, is not sustained since these lines connect the charging pumps to the reactor coolant pumps. The flow in the seal water injection line is limited.

Pipe Restraints

Based upon the pipe whip effects noted above, pipe whip restraints are not required.

Jet Impingement Effects

Jet impingement effects are addressed in Table 3.6–29.

Environmental Effects

Refer to Section 3.11 for information.

Conclusion

Essential systems, structures, and components remain functional subsequent to a postulated pipe rupture of the CHS seal water injection system.

Steam Generator Blowdown System

Break Locations

Pipe breaks in the steam generator blowdown piping are postulated at terminal ends and intermediate locations shown on Figure 3.6–18.

The break exclusion criteria is invoked on portions of the steam generator blowdown piping from the containment penetration (outboard side) up to the isolation valves to ensure the integrity of a minimum of two intact and functioning Steam Generators. The stresses along this section of the steam generator blowdown piping fully meet the break exclusion requirements (Equation 3.6-1).

Pipe breaks are postulated at terminal and intermediate locations in the steam generator piping extending from the break exclusion zone through the main steam valve building to the turbine building. Table 3.6–30 summarizes these breaks.

Separation

Every attempt has been made to locate the steam generator blowdown piping in areas physically remote from essential equipment to minimize the consequential effects of pipe rupture.

Pipe Whip Effects

Table 3.6–31 summarizes the pipe whip effects resulting from the postulated piping failure.

Pipe Restraints

The pipe rupture restraints on the steam generator blowdown system are shown on Figure 3.6–18 along with the postulated pipe rupture locations and data for these rupture restraints.

Jet Impingement Effects

Jet impingement effects are addressed in Table 3.6–32.

Environmental Effects

Refer to Section 3.11 for information.

Conclusion

Adequate protection and/or separation is provided for essential systems and components required to mitigate the consequences of steam generator blowdown piping ruptures.

Auxiliary Feedwater System

Break Locations

Based upon application of the criteria specified in Section 3.6.2.1.1, pipe breaks on the auxiliary feedwater system inside the containment are postulated at terminal ends and intermediate locations.

Based upon application of the criteria specified in Section 3.6.2.1.2, pipe breaks in the auxiliary feedwater system outside the containment are postulated at terminal ends and at intermediate locations.

Auxiliary feedwater system pipe break locations and piping geometry are shown on Figure 3.6–32. Postulated pipe break locations for the Auxiliary Feedwater System are listed in Table 3.6–34.

Separation

The Auxiliary Feedwater System satisfies the physical separation criterion Inside the Containment. Two Auxiliary Feedwater piping runs are routed along the exterior Crane Wall and the other two piping runs are routed along the interior Crane Wall at a higher elevation.

Inside the Engineered Safety Features (ESF) Building, the Auxiliary Feedwater System piping from each Motor Driven Pump is separated from the piping of the other Motor Driven Pump by the ESF Building walls. This prevents damage to a pair of intact Auxiliary Feedwater System pipes subsequent to a pipe break on any other Auxiliary Feedwater pipe.

Portions of the Turbine Driven Auxiliary Feedwater Pump discharge piping are high energy since this piping is pressurized due to operation of the Motor Driven Pumps. The Turbine Driven Auxiliary Feedwater Pump is separated from the Motor Driven Pumps by the ESF Building walls. Turbine Driven Auxiliary Feedwater Pump discharge piping does not meet the separation criterion in one area of the ESF Building. Four piping runs associated with the Turbine Driven Auxiliary Feedwater Pump discharge piping by necessity are routed in a common area. These four piping runs are of the same nominal pipe size and wall thickness where the routing is common. Based upon the criteria in Section 3.6.2.1, a break in one line does not affect the structural integrity of the adjacent lines.

Pipe Whip Effects

Pipe whip of the auxiliary feedwater system is not sustained since backflow from each main feedwater line is prevented by dual check valves installed in the system and flow from each auxiliary feedwater pump is limited. Table 3.6–35 provides the results of pipe whip effects for the Auxiliary Feedwater System.

Pipe Restraints

Pipe whip restraints are not required for postulated pipe breaks on the Auxiliary Feedwater System.

Jet Impingement Effects

Jet impingement effects are addressed in Table 3.6–36.

In general, jet impingement forces, subsequent to postulated pipe breaks on the auxiliary feedwater system, are small due to the lack of upstream energy reservoirs to sustain pressure in the steady state.

Environmental Effects

Refer to Section 3.11B.1 for information.

Conclusion

Essential systems, structures, and components remain functional subsequent to a postulated pipe rupture of the auxiliary feedwater system.

3.6.2 DETERMINATION OF BREAK LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

This section describes the design bases used for defining postulated pipe break and crack locations in high- and moderate-energy piping systems inside and outside of the containment, the methods of analysis used to evaluate the jet reaction forces at the break locations, and the jet impingement effects and loading effects on adjacent essential systems, components, and structures.

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

Pipe breaks and cracks are postulated in those high- and moderate-energy piping systems located in proximity to essential systems, components, and structures required for the safe shutdown of the plant. All postulated breaks and cracks are evaluated for potential damage to essential systems, components, and structures due to pipe whip, jet impingement, and environmental effects. If the damage is unacceptable, protective measures are provided either by rerouting of piping, relocation of essential equipment, or providing enclosures. Where this is not feasible, pipe whip restraints and/or jet impingement shields are installed to protect the essential systems, components, and structures.

An unrestrained whipping pipe is considered capable of causing circumferential and longitudinal breaks, individually, in impacted pipes of smaller nominal pipe size and developing through-wall cracks in equal or larger nominal pipe sizes with thinner wall thicknesses. The impact into pipes of equal or larger nominal pipe sizes and wall thicknesses is considered inconsequential and is not evaluated. In all cases, the effects of jet impingement are less severe than the corresponding pipe whip impact. (In the limiting case of no clearance between the broken pipe and target pipe, the impact force equals the total thrust force and is localized whereas the jet force is less, being moderated by a shape factor, and is distributed. As the gap increases, the whip impact is enhanced but the jet intensity diminishes and becomes more diffuse.) Therefore, jet impingement on pipes of equal and larger nominal pipe size and wall thickness also is considered inconsequential and is not evaluated. In all other cases, the jet interaction is either avoided or the target pipe and its supports are evaluated for the jet load using Service Level D allowables.

Design basis break and crack locations, type, and orientation are postulated in accordance with the following sections.

3.6.2.1.1 Criteria for Inside Containment

Break Locations - ASME Section III Code Class 1 Piping

Breaks in ASME Section III Code Class 1 high-energy piping are postulated to occur at the following locations in each piping run or branch run:

1. At terminal ends of the pressurized portions of the runs (terminal ends are extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves) or pipe anchors. A branch connection to a main piping run is a terminal end of the branch run, except where the branch run is classified as part of a main run in the stress analysis and is shown to have a significant effect on the main run behavior.

2. At intermediate locations between terminal ends where either of the following criteria are exceeded:

   a. At any intermediate locations between terminal ends where the maximum stress intensity ranges, for normal and upset plant conditions, and for a 1/2 safe shutdown earthquake (OBE) event transient exceed 2.4Sm (the design stress intensity as specified in Section III of the ASME Boiler and Pressure Vessel Code), calculated by Equation 10 and either Equation 12 or Equation 13 in Paragraph NB-3653 of the ASME Code, Section III, or

   b. At any intermediate locations between terminal ends where the cumulative usage factor U (the cumulative usage factor as specified in Section III of the ASME Boiler and Pressure Vessel Code), derived from the piping fatigue analysis under the loadings associated with OBE and operational plant conditions, exceeds 0.1.

Break Locations - ASME Section III Code Classes 2 and 3 Piping

Breaks in ASME Section III, Code Classes 2 and 3, high energy piping systems are not postulated at locations delineated in Section 3.6.2.1.2, Item 2. The portions of piping within the break exclusion zone are designed to meet the requirements of ASME Section III, Subarticle NE-1120 and the additional criteria specified in Section 3.6.2.1.2, Items 2a through 2f.

Breaks in ASME Section III Code Classes 2 and 3 high-energy piping are postulated to occur at the following locations in each piping run or branch run:

1. At terminal ends of the pressurized portions of the runs, and

2. At intermediate locations selected by either of the following criteria:

    a. At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve.

    b. At each location where the stress ranges associated with normal and upset plant conditions and a OBE event, calculated by Equations 9 and 10, Paragraph NC-3652 of the ASME Code, Section III, exceed $0.8\ (1.2S_h + S_A)$.

    S_A is the stress calculated by the rules of NC 3600 and ND 3600 for Classes 2 and 3 components, respectively, of ASME Code Section III, 1971 edition up to and including the summer 1973 addenda. S_A is the allowable stress range for expansion stress calculated by the rules of NC 3600 of ASME Code Section III, 1971 edition up to and including the summer 1973 addenda.

3.6.2.1.2 Criteria for Outside Containment

High-Energy Piping Systems

1. Piping Systems Separated from Essential Structures, Systems and Components - A primary objective in the piping layout and plant arrangement is to have adequate separation, so that the effects of postulated pipe breaks at any location are isolated or physically remote from essential structures, systems, and components. Pipe breaks are not postulated in these separated high-energy piping systems.

2. Piping Systems in Containment Penetration Areas Break Exclusion Zone - Breaks are not postulated in certain high energy piping systems in the containment penetration areas. Breaks are not postulated in portions of high energy piping between the first restraint outboard of the containment isolation valve and the pipe to flued head weld on the inboard side of the containment penetration for the 30 Inch Main Steam System and the Main Feedwater System. For the Letdown Line, the 3 Inch Main Steam System in the Engineered Safety Features (ESF) Building, and the Steam Generator Blowdown System, breaks are not postulated between the outboard containment isolation valve and the pipe to flued head weld on the outboard side of the containment penetration. Portions of this piping outboard of the isolation valve, for which failure could affect the leaktight integrity of the containment structure, are provided with pipe whip restraints capable of resisting bending and torsional moments produced by the postulated piping failure outboard of the first restraint beyond the containment isolation valves. These restraints are located as close as practical to the containment isolation valves.

The restraints are designed to withstand the loadings imposed by a postulated pipe rupture so that isolation valve structural integrity, operability, and leak tight integrity of the associated containment penetration is ensured.

The portions of piping within the break exclusion zone are designed to meet the requirements of ASME Code Section III, Subarticle NE-1120 and the following additional design requirements:

a. The following design stress and fatigue limits should not be exceeded for Class 2 piping:

   - The maximum stress ranges as calculated by Equations 9 and 10 in Paragraph NC-3652, ASME Code, Section III, considering normal and upset plant conditions (i.e., sustained loads, thermal expansion and a 1⁄2 SSE event) do not exceed 0.8 ($1.2S_h + S_A$).

   - The maximum stresses as calculated by Equation 9 in Paragraph NC-3652 under the loadings resulting from a postulated piping failure beyond these portions of piping do not exceed $1.8S_h$. This stress limit is applied between the containment isolation valve and the containment penetration. The portion of piping within the pipe break exclusion zone between the containment isolation valve and the first pipe rupture restraint must not develop a plastic hinge subsequent to postulated piping failures beyond the pipe break exclusion zone.

b. Welded attachments, for pipe supports or other purposes, to these portions of piping are permitted when a detailed stress analysis demonstrates compliance with the limits described in (a).

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

d. The length of these portions of piping is reduced to the minimum length practicable.

e. The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) does not require welding directly to the outer surface of the piping (e.g., flued integrally forged pipe fittings are used) except where such welds are capable of 100-percent volumetric inservice inspection. This criterion is also applicable to the portion of piping between the containment and the inside containment isolation valves.

f. THIS ONE-For these portions of high-energy piping, inservice examination is performed in accordance with the requirements specified in ASME Code, Section XI, with the exception that examination of circumferential and longitudinal pipe welds between and including the boundaries of the pipe break exclusion zones is performed in accordance with the risk-informed methodology established in ASME Code Case N-716-1, Alternative Classification and Examination Requirements, Section XI, Division 1. The examination includes the valve and pipe pressure boundaries. Details of containment penetration, identification of pipe welds, access for inservice inspection, and points of fixity and discontinuity are provided in Section 6.6.

3. Piping Systems not designated as Break Exclusion Zone

Breaks in ASME Section III Code Classes 2 and 3 high-energy piping are postulated at the following locations in each piping and branch run outboard of break exclusion zones (except those portions of piping systems identified in Section 3.6.2.1.2, Items 1 and 2):

a. At terminal ends of the pressurized portions of the runs.

b. At the extremity of the break exclusion zone (if applicable), and

c. At intermediate locations selected by either of the following criteria:

- At each pipe fitting (e.g., elbow, tee, cross, and nonstandard fitting) welded attachment and valve or, if the run contains no fittings, at one location at each extreme of the run (a terminal end, if located within a protective structure, may substitute for one intermediate break), or

- At each location where the stresses exceed 0.8 (1.2 $S_h + S_A$).

Breaks in nonnuclear safety class high-energy piping are postulated at the following locations in each piping run or branch run:

1. At terminal ends of the pressurized portions of the runs, and

2. At intermediate locations selected by either of the following criteria:

a. At each pipe fitting, welded attachment, and valve, or

b. At each location where the stress ranges associated with normal and upset plant conditions and a 1/2 SSE event, calculated by Equations 9 and 10, paragraph NC-3652 of ASME Section III exceed 0.8 (1.2 $S_h + S_A$).

<u>Moderate-Energy Piping Systems</u>

For the purpose of satisfying the separation provisions of plant arrangement, a review of the piping layout and plant arrangement drawings is conducted to show that the effects of through-wall leakage cracks at any location are isolated or physically remote from essential systems, components, and structures.

For certain Moderate Energy Systems, leakage cracks are not postulated in those portions of ASME Section III Code Class 2 piping between the outboard containment penetration weld and the outboard containment isolation valve. A Crack Exclusion Zone is defined providing the piping is designed to the requirements of ASME Section III Subsection NE-1120 and that the maximum stress range (as calculated by Equations 9 and 10, Paragraph NC-3652 of Section III of the ASME Code) does not exceed 0.4 ($1.2S_h + S_A$).

Crack Exclusion Zones are invoked for the Hydrogen Recombiner (HCS) System Suction piping, the Containment Atmosphere Monitoring (CMS) System Pump Suction piping, and the Containment Vacuum (CVS) System Pump Suction piping. The Crack Exclusion Zones are designated to ensure containment integrity post DBA since postulating through wall leakage cracks in this region of piping does not maintain containment integrity. As discussed in Section 6.2.4.2, these systems take exception to GDC-56 by placing two isolation valves in series outside containment. For these systems, Augmented Inservice Inspection (ISI) is invoked to support the GDC exception.

Through-wall leakage cracks are postulated in moderate-energy piping except where exempted by Moderate-Energy Piping Systems under Section 3.6.2.1.2, or where the maximum stress range in these portions of ASME Section III Code Class 2 or 3 piping is less than 0.4 ($1.2S_h + S_A$). For nonnuclear, nonseismic piping systems, throughwall leakage cracks are postulated at any location that has the worst consequences for essential structures, systems, or components. Only environmental effects due to fluid spray and flooding are considered.

Cracks are not postulated in moderate-energy piping location in an area in which a break in high-energy piping occurs. Where a postulated leakage crack in the moderate-energy piping results in more limiting environmental conditions than the break in proximate high-energy piping, the provisions of the previous paragraph are applied.

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

An operational period is considered short if the fraction of time that the system operates within the pressure-temperature conditions specified for high-energy systems is less than 2 percent of the time that the system operates as a moderate-energy system (e.g., systems such as the reactor residual heat removal systems qualify as moderate-energy systems); however, systems such as auxiliary feedwater systems operated during reactor startup, hot standby, or shutdown, qualify as high-energy systems.

3.6.2.1.3 Design Basis Break/Crack Types and Orientation

Circumferential Pipe Breaks

The following circumferential breaks are postulated in high-energy piping at the locations specified in Sections 3.6.2.1.1 and 3.6.2.1.2:

1. Circumferential breaks are postulated in high-energy piping runs and branch runs exceeding a nominal pipe size of 1 inch. When the maximum stress range or usage factor exceeds the limits specified for break postulation, and if it is determined by detailed stress analysis, that the maximum stress range in the circumferential direction is at least 1.5 times that in the axial direction, then only longitudinal breaks are postulated.

2. Where break locations are selected at pipe fittings without the benefit of stress calculations, breaks are postulated at the piping weld to each fitting, valve, or welded attachment. If detailed stress analyses or tests are performed, the maximum stressed location in the fitting may be selected instead of the pipe-to-fitting weld.

3. Circumferential breaks are assumed to result in pipe severance and separation amounting to a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic analysis.

4. The dynamic force of the jet discharge at the break location is based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by a thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are taken into account as applicable, in the reduction of jet discharge.

5. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration, and is assumed to cause pipe movement in the direction of the jet reaction.

Longitudinal Pipe Breaks

The following longitudinal breaks are postulated in high-energy piping at the locations of each circumferential break specified under Circumferential Pipe Breaks in this section, except as noted:

1. Longitudinal breaks in piping runs and branch runs are postulated in nominal pipe sizes 4 inches and larger. However, when the maximum stress range or usage factor exceeds the limits specified for break postulation and if it is determined by detailed stress analysis that the maximum stress range in the axial direction is at

least 1.5 times that in the circumferential direction, then only a circumferential break is postulated.

2. Longitudinal breaks are not postulated at terminal ends.

3. Longitudinal breaks are assumed to result in an axial split without pipe severance. Splits are located (but not concurrently) at two diametrically opposed points on the piping circumference such that a jet reaction causing out-of-plane bending of the piping configuration results. Alternately, a single split may be assumed at the section of highest stress as determined by detailed stress analysis.

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

5. Pipe movement is assumed to occur in the directions defined by the stiffness of the piping configuration and jet reaction forces, unless limited by structural members or piping restraints.

Through-Wall Leakage Cracks (outside of containment only)

The following through-wall leakage cracks are postulated in moderate-energy piping at the locations specified under Moderate-Energy Piping Systems in Section 3.6.2.1.2, item 3.

1. Cracks are postulated in moderate-energy piping runs and branch runs exceeding a nominal pipe size of 1 inch.

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

3. The flow from the crack is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments. Flooding effects are determined on the basis of a conservatively estimated time period required to effect corrective actions.

3.6.2.1.4 Conformance with Regulatory Guide 1.46

Refer to Section 1.8 for this information.

3.6.2.2 Analytical Methods to Define Forcing Functions and Response Models

3.6.2.2.1 Introduction

Pipe rupture analyses include calculations to determine the fluid forces generated by blowdown of pressurized lines, complemented by dynamic or energy-balance methods to determine pipe motion and impact effects. Restraints for lines 6 inches and less in diameter are usually qualified on a generic basis using an energy balance method. However, restraints for larger lines are normally engineered individually for each system, using standard design concepts. The response of unrestrained lines is analyzed by either inelastic dynamic analysis or energy balance analysis. Figure 3.6–19 provides a flowchart for the pipe rupture analysis.

Criteria for the pipe rupture response analysis include:

1. An analysis of the pipe run or branch is performed for each postulated longitudinal and circumferential rupture or, alternatively, for a worst case. Worst cases are selected on the basis of gap, fluid force, and piping system stiffness.

2. The loading condition of a pipe run or branch prior to postulated rupture in terms of internal pressure, temperature, and stress state is that condition associated with reactor operation at 100-percent power.

3. For a circumferential rupture, pipe whip dynamic analyses are only performed for that end (or ends) of the pipe or branch that is (are) connected to a contained fluid energy reservoir having sufficient capacity to develop a jet stream.

4. Dynamic analytical methods, used for calculating the piping or piping/restraint system response to the jet thrust developed after a postulated rupture, account for the effects of the following:

    a. Mass, inertia, and stiffness properties of the system.

    b. Impact and rebound (if any) as permitted by gaps between piping and restraint.

    c. Elastic and inelastic deformation of piping and/or restraint.

    d. Support boundary conditions.

5. An allowable design strain limit of 0.5 ultimate uniform strain of the restraints is used for energy-absorbing components. For compressive energy absorbing components, the following deformation limits are used:

    a. The design limit for pipe crush bumpers shown on Figure 3.6–30 and metallic honeycomb is 80 percent of energy absorbing capacity.

b. The design limit for pipes crushed uniformly along their length is the lesser of:

1. one half of the pipe diameter, or

2. the maximum flattening limits as prescribed by ASTM A530.

The above deformation limits assure that the area under the essentially flat portion of the materials force-deflection curve is not exceeded.

6. A 10-percent increase of minimum specified yield strength (S_Y) may be used to account for strain rate effects in inelastic nonlinear analyses. Alternatively, experimental data may be used to determine the strain rate parameters for use in nonlinear codes which monitor strain rate.

3.6.2.2.2 Time Dependent Blowdown Force

Blowdown force calculations are based on methods suggested by Moody (1973) and include consideration of the transient pressures, velocities, and other thermodynamic properties of the fluid. To provide the time history of pressure, velocity, etc., the method of characteristics is used to solve the continuity and momentum equations simultaneously. A general description of the method can be found in gas dynamics textbooks (De Haller 1945, Rudinger 1969, Owzarek 1968). For these one dimensional fluid mechanics analyses, the pipe run is treated as an equivalent section of straight pipe. The calculated momentum and pressure forces are applied at changes in direction or cross section of the piping to provide time-dependent loads for pipe dynamic analysis.

The transient forces result from wave propagation and fluid momentum. It is assumed that pipe bends and elbows neither attenuate the traveling pressure waves nor cause reflections. Immediately following the rupture of a pipe, a decompression wave travels from the break at the speed of sound relative to the fluid. The fluid ahead of and behind the wave is at different thermodynamic states. This initial blowdown condition is maintained until a return signal from the pressure reservoir reaches the break. At this time, repeated wave reflections between the reservoir and break prevail until a steady state flow condition is established. Boundary conditions that govern the flow at the break are considered.

Fluid momentum changes will result in dynamic forces being exerted on pipe segments. The forcing function is calculated using Shapiro Volume I Equation 4.20 as follows:

$$F = (P - P_a)A + \frac{\dot{m}u}{g} = \left[(P - P_a) + \frac{Ru^2}{144g}\right]A \qquad (3.6.2.2\text{-}1)$$

where:

P = Local static pressure (psia)

P_a = Ambient pressure (psia)

R = Fluid density (lb_m/ft^3)

u = Velocity of blowdown fluid (fps)

A = Local flow cross-sectional area (in^2)

g = Dimensional Constant $\left(32.2\frac{ft-lb_m}{lb_f-sec^2}\right)$

$\dot{m}$ = Mass flow rate (lb /sec)

During the blowdown process, the local static pressure, mass flow rate, and other thermodynamic properties change with time; therefore, the forcing function varies with time.

3.6.2.2.2.1 Subcooled Nonflashing Waterline Blowdown

When a pipe rupture occurs, the blowdown flow rate and properties must go from an initial set of conditions to the final or steady state condition. A decompression wave travels upstream toward the pressure source. The initial blowdown velocity can be calculated by applying the momentum equation across the wave:

$$RC\Delta u = g\Delta P \tag{3.6.2.2-2}$$

$$(u\text{-}o) = 144g(P_o\text{-}P_a)/RC \tag{3.6.2.2-3}$$

where:

C = The speed of sound (fps)

P_o = The initial stagnation pressure (psia)

Therefore, the initial blowdown thrust for non-flashing water is:

$$F = \frac{Ru^2}{g(144)}A = \frac{g(P_o - P_a)(144)}{RC^2}P_oA \tag{3.6.2.2-4}$$

For transient flow analysis, the one dimensional equations of conservation of momentum and continuity are solved simultaneously to obtain transient pressures and velocities, which are used to calculate the transient blowdown thrust. The solutions are subject to the following boundary conditions:

$U(\chi,t) \approx O$ at $t = 0$

$P(\chi,t) \approx P_o$ at $t = 0$

$$P(\chi, t) = P_o - \frac{Ru^2(\chi, t)}{2g(144)} \text{ at } \chi = 0$$

The initial blowdown flow remains constant until the decompression wave, which is reflected from the pressure source, reaches the break end. It is then reflected again, causing a change of blowdown flow. These repeated wave transmissions and reflections continue until the steady state flow is established.

Steady State Flow with Friction

For steady state flow with friction, the blowdown forcing function calculation becomes:

$$F = \left[\frac{2(P_o - P_a)}{P_o} \frac{1}{1 + \frac{fL_e}{D}}\right] P_o A \tag{3.6.2.2-5}$$

which is derived by applying Bernoulli's equation across the pipe and by using the expression for the forcing function calculation, where:

L_e = Total equivalent length of pipe friction

f = Friction factor (Reynolds number and pipe surface roughness dependent), obtained from Moody's chart, (1961).

D = Pipe inside diameter. The friction parameter is calculated by considering all the possible losses on the line.

The friction parameter fL_e/D is calculated by considering all the possible losses on the line.

Transient Flow With Friction

With friction losses taken into consideration, transient pressures and velocities of a subcooled nonflashing water line blowdown can be obtained by simultaneously solving the continuity and momentum equations. The finite difference approximation using the method of characteristics is used as a principle for numerical solution of these two governing equations (Streeter 1967).

The computations proceed in the following manner. A grid is chosen in such a way that $\Delta\chi = C\Delta t$, where $\Delta\chi$ is a space increment, t a time increment, and C the propagation speed of the decompression wave. Starting from the initial conditions along the pipe, the pressure and velocity at $t = t + \Delta t$, and at any interior points of the pipe, can be calculated by using two characteristic equations. Whenever a boundary point is reached, the corresponding characteristic equation and boundary condition are used. The process proceeds until a steady state is reached.

The friction losses are expressed in terms of pressure drop of the system. To accurately model the system with friction losses, a smaller $\Delta\chi$ must be used. This method can be applied for flow with or without friction losses.

The transient pressures, velocities, etc., are then used to calculate the blowdown forces using the equation described previously.

Side Thrust for Longitudinal Split

For a longitudinal split, the blowdown flow comes from either upstream, downstream, or both pipe directions. Because of its geometry, the split is considered as an ideal nozzle with a discharge coefficient of unity.

A longitudinal break will cause a reaction force of 1 P_o A for an extremely short time interval, which allows decompression waves to move out of the break region into the adjacent upstream and downstream pipe sections. Then the reaction force drops to a value corresponding to the blowdown thrust until the reflected waves arrive from each direction. The steady state blowdown thrust is:

$$F = \frac{2(P_o - P_a)A}{1 + \frac{fL_e}{D}} \qquad (3.6.2.2\text{-}6)$$

where:

A ~ The break opening area of the split

3.6.2.2.2.2 Steamline Blowdown

Transient Flow without Friction

Steam is treated as an ideal, single-phase gas with a constant specific heat ratio, k, of 1.3. Except for the case of steady state blowdown flow, the flow is assumed to be isentropic with negligible pipe friction. The characteristic method (Jonssen et al., 1973, Hartree 1952), which is a finite difference approximation using the principle of characteristics, is used as a basis for the numerical solution of the continuity and momentum equations. The transient pressure, mass flow rate, and other thermodynamic properties are then used to calculate the transient-state forcing function.

Immediately following the break, a decompression wave travels into the pipe toward the pressure reservoir. The fluid in front of the wave is at a state:

$U_1 = 0$

$C_1 = C$

where:

U_1 = Velocity of fluid

C_o = Speed of sound in fluid

The fluid state at the exit is at the sonic condition (Shapiro 1953 Volume 1 Equation 4.6a).

$$\frac{U_e}{C_o} = \frac{C_e}{C_o} = \left[\frac{2}{K+1}\right]^{0.5} = 0.9325 \qquad (3.6.2.2\text{-}7)$$

The blowdown force can be calculated using Shapiro Volume I Equation 4.20 as:

$$F = \left[\frac{P_e}{P_o} + \frac{R_e C_e^2}{gP_o(144)}\right] P_{oA}$$

$$= \left[\frac{P_e}{P_o} + \frac{R_e}{R_o}\left(\frac{C_e}{C_o}\right)^2 \frac{R_o C_o^2}{gP_o(144)}\right] P_{oA} \qquad (3.6.2.2\text{-}8)$$

where:

$$C_o^2 = \frac{KgP_o(144)}{R}$$

The pressure ratio across the wave is:

$$\frac{P_e}{P_o} = \left(\frac{T_e}{T_o}\right)\frac{k}{k-1} = \left(\frac{C_e}{C_o}\right)\frac{2k}{k-1} = \left[\frac{2}{k+1}\right]^{\frac{k}{k-1}} = 0.55 \qquad (3.6.2.2\text{-}9)$$

where:

T = Temperature

The density ratio is:

$$\frac{R_e}{R_o} = \left(\frac{P_e}{P_o}\right)\frac{1}{k} = \left(\frac{C_e}{C_o}\right)\frac{2k}{k-1}\left(\frac{1}{k}\right) = \left[\frac{2}{K+1}\right]^{(1/k-1)} = 0.628 \qquad (3.6.2.2\text{-}10)$$

Therefore, the blowdown force can be reformulated as:

$$F = (1+K)\left(\frac{2}{K+1}\right)^{(2K/\mathrm{K}\text{-}1)} P_o A = 0.685\ P_o A \qquad (3.6.2.2\text{-}11)$$

This blowdown force is constant until a return signal from the pressure source reaches the break.

When the wave reaches the reservoir, it is reflected as a compression wave. The boundary condition at the reservoir lies on the steady state ellipse. From, Shapiro Volume II Equation 24.36.

$$\left(\frac{C_i}{C_o}\right)^2 + \frac{K-1}{2}\left(\frac{U_i}{C_o}\right)^2 = 1 \qquad (3.6.2.2\text{-}12)$$

which is the energy equation applying across the vessel-pipe inlet. The boundary condition for this case is from Shapiro Volume II Page 963:

$$T_o = T_i + \frac{U_i^2}{2C_p g(778)} \qquad (3.6.2.2\text{-}13)$$

where:

C_p = The constant pressure specific heat of a fluid $\frac{Btu}{lb_m - {}^\circ F}$

i = The state at the inlet to the pipe

If the steady state is reached, the flow in the pipe is uniform and, if the pressure in the pressure vessel remains high, then the boundary condition at the break always lies on the sonic line. For example:

$$\frac{U_*}{C_o} = \frac{C_*}{C_o} \qquad (3.6.2.2\text{-}14)$$

Then from the critical flow condition from Shapiro Volume I Equation 4.6a:

$$\frac{U_*}{C_o} = \frac{C_*}{C_o} = \sqrt{\frac{2}{k+1}} = 0.9325 \qquad (3.6.2.2\text{-}15)$$

where:

* = The critical flow condition

Then, the steady state blowdown force is:

$$F = \left(\frac{P_*}{P_o} + \frac{R^*(U^*)^2}{P_o g(144)}\right) P_o A$$

$$= (1+k)\left(\frac{2}{k+1}\right)^{(k/\mathrm{k}-1)} P_o A = 1.255\ P_o A \qquad (3.6.2.2\text{-}16)$$

Steady State with Friction

For steady state flow with friction losses, the analysis is based on the theory of compressible flow with friction (Shapiro 1953). The pipe friction is the chief factor bringing about the change of fluid properties in the flow. A curve which describes the variation of steady state steam blowdown force versus friction parameter fL_e/D is shown on Figure 3.6–20 (Moody 1973).

Transient Flow with Friction

Using a method similar to the transient flow analysis for a non-flashing waterline, a hybrid method of characteristics has been adopted from Jonsson et al. (1973) to solve the one dimensional governing equations of mass, momentum, and energy simultaneously for pipe with a constant cross-sectional area with friction effects taken into consideration. The governing equations are first transformed into a system of characteristic equations. Then the finite difference approximation is used to integrate the fluid variables which represent the pressure, velocity, and entropy along the characteristic lines and the path line. The equations are solved using computer program ME-143 One-Dimensional Unsteady Flow of a Compressible Fluid with Friction (UFLOW). These transient pressures, velocities, etc., are used to calculate the blowdown forcing functions using the equation described previously.

3.6.2.2.3 Simplified Blowdown Analysis

A conservative steady state forcing function may be used for calculations based on the energy balance method. The function has a magnitude of

T = KPA (3.6.2.2-17)

where:

P = System pressure prior to pipe break

A = Pipe break area

K = Thrust coefficient (theoretical maximum)

K values are 1.26 for saturated steam, water, and steam/water mixtures and 2.00 for nonflashing subcooled water.

An amplification factor between 1.1 and 1.2 has been demonstrated by testing. To account for rebound, it is applied to the above force. Alternatively, the maximum fluid force during the energy input phase, as determined by the detailed methods of Section 3.6.2.2.2, may be used.

3.6.2.2.4 Lumped-Parameter Dynamic Analysis

The piping system is modeled mathematically as a series of beam elements connected at nodes. Distributed mass of the pipe and contained fluid is modeled as a lumped mass located at the nodal points. Beam elements have the stiffness properties of the pipe in the elastic range and approximate the plastic behavior above yield.

Before a rupture, pipe is stressed by internal pressure and is in static equilibrium. When initial conditions have an effect on the parameters being calculated, such as stresses in break exclusion regions or loads on attached components, this effect is considered.

As a circumferential break propagates, the load-carrying metal area of the pipe decreases so that a force unbalance results. The force initially transmitted across the break is assumed to drop linearly to zero in 1 millisecond. After the break, the forces exerted on the pipe by the fluid are determined by the time-dependent blowdown force derived (Section 3.6.2.2.2). Similarly, for a longitudinal split, the crack propagation speed limits the rate at which the split opens, so a 1-millisecond force rise time is assumed. Other break opening times may be used if justified.

Subsequent to a postulated rupture, the inelastic system response is analyzed by the use of an elastic-plastic lumped-mass beam element computer code such as DINASAW or LIMITA (Appendix Sections 3A.2.6, 3A.2.10, and 3A.2.11. The analysis considers the free motion of the pipe through a gap, if one exists, using the appropriate initial conditions and the fluid blowdown forces as calculated in Section 3.6.2.2.2. The mathematical model includes the restraint or barrier, and sometimes a member simulating the local crush resistance of the pipe. Rebound effects are considered by automatically connecting and disconnecting that member for impact and rebound, respectively.

3.6.2.2.4.1 Sample Pipe Rupture Dynamic Analysis

Pipe rupture restraint 3FWS-PRR5S limits the motion of the feedwater line following a postulated circumferential rupture at the end of the reducer elbow (Figure 3.6–21). The restraint prevents the whipping pipe from impacting the steam generator.

The restraint is a U configuration stainless steel strap having a 7 inch strainable width and 1 inch thickness. The initial clearance between the hot pipe and restraint is 0.89 inch in the outward direction, resulting in a total acceleration gap, after slack takeup, of 2.04 inches. Figure 3.6–22 shows a typical laminated strap.

The analysis was conducted using the DINASAW computer code for the mathematical model of the pipe, restraint, and intermediate structure (Figure 3.6–21). The elastic-plastic characteristics of the pipe, strap, and intermediate structure were used in the computer solution in terms of bilinear engineering stress-strain relationships with the corresponding allowable stress not exceeding half the uniform ultimate strain. Strain rate sensitivities for materials below 400°F were considered. The fluid forcing function depicted on Figure 3.6–21 is applied at the elbow in terms of normal pressures per unit tangential length.

The restraint and intermediate structure reaction load are shown on Figures 3.6–23 through 3.6–25 which illustrate the intermediate structure loads and Figure 3.6–26 which illustrates the restraint reaction load. The depletion of the kinetic energy of the system and the steady work are indications of the convergence of the dynamic problem. This is shown on Figure 3.6–27.

3.6.2.2.5 Energy Balance Analysis

The energy balance technique for analyzing pipe impact equates the work done by the escaping fluid to the energy absorbed in deforming the ruptured pipe and the impacted target. A steady state blowdown force is used for the energy balance analysis. The magnitude of the force is described in Section 3.6.2.2.3.

The input energy of the system is determined by multiplying the pipe displacement at the break end by the component of the fluid blowdown force in the direction of the displacement.

The input energy is:

$$E = F_b(g + d)\left(\frac{L_h}{L_h - L}\right) \qquad (3.6.2.2\text{-}18)$$

where:

F_b = Component of blowdown force in direction of pipe displacement

L_h = Length from break to plastic hinge

L = Length from break to restraint

g = Pipe-target gap

d = Restraint deflection

The strain energy absorbed during pipe whip and impact consists of the energy absorbed by pipe bending, E_b, the energy absorbed by pipe crush during impact, E_c, and the energy absorbed by deformation of the target, E_t.

To determine post-impact target deformation and the peak reaction force, the input energy is equated to the strain energy absorbed by the pipe and target. The energy absorption characteristics of the pipe crush and target deformation are calculated on the basis of the displacement integral of the appropriate force-deformation curves.

Sample Energy Balance Analysis

Analyze the impact of a 4-inch, schedule 80 pipe into a pipe crush bumper following a circumferential break at an elbow (Figure 3.6–28). One source of energy input is recognized: the fluid blowdown force traveling through the distance moved by the ruptured end of the pipe. The input energy is:

$$E_{in} = F_b(g + d)\left(\frac{L_h}{L_h - L}\right) \qquad (3.6.2.2\text{-}19)$$

where:

F_b = Fluid blowdown force

g = Acceleration gap

d = Restraint deflection

L_h = Length from break to plastic hinge

L = Length from break to restraint

The ratio $L_h/(L_h\text{-}L)$ represents the increased pipe displacement at the break, compared to displacement at the restraint, due to the assumed pipe rotation about a plastic hinge.

The fluid force is calculated:

$$F_b = K_r\, K_f\, P_o A = 13.4 \text{ kips} \qquad (3.6.2.2\text{-}20)$$

where:

K_r = Rebound factor (1.2)

K_f = Thrust coefficient (0.88)

P_o = Initial pressure (1,106.7 psi)

A = Pipe flow area (11.497 square inches)

The maximum thrust coefficient in the period when the energy balance occurs is 1.0. However, this drops to 0.88 as soon as the decompression wave passes the elbow ($t \leq 0.001$ second) and occurs when the pipe is just starting to accelerate. Since the displacement and resulting energy input are negligible during this interval, 0.88 rather than 1.0 is used as the thrust coefficient. The duration of the entire energy balance event is evaluated after the restraint is sized. This permits a quick review of the fluid force history to assure that a higher thrust coefficient did not occur during the dynamic event.

Energy may be absorbed in plastic bending of the pipe and in crush of the restraint. The energy absorbed by bending at the plastic hinge is:

$$E_b = M_p\theta = M_p (g + d)/(L_h\text{-}L) \qquad (3.6.2.2\text{-}21)$$

where:

M_p = Plastic moment

θ = Hinge rotation

The value of M_p may be obtained from rigid, perfect-plastic limit theory, but, for this application, a strain hardening moment (Gerber 1974) is appropriate. This requires an estimate of the hinge rotation (i.e., g, d, L_h and L must be determined). The acceleration gap, g, is 1.63 inches. Let the bumper pipe have the same diameter as the process pipe. Set L at 20 inches to place the restraint on the straight pipe. Using the common expression for plastic hinge length, $L_h = 3\ M_p/F_b$, and the method described by Gerber (1974), an iterative solution shows that M equals 244.9 in-kips and L_h equals 119.6 inches.

Input energy:

$$E_{in} = F_b(g+d)x\frac{L_h}{(L_h - L})26.3 + 16.1 \quad \text{d in-kips} \qquad (3.6.2.2\text{-}22)$$

Energy absorbed by the process pipe bending:

$$E_b = M_p(g+d)/(L_h - L) = 4 + 2.5 \quad \text{d in-kips} \qquad (3.6.2.2\text{-}23)$$

The difference between E_{in} and E_b is the energy absorbed by the crush bumper and equals 22.3 + 13.6 d in-kips.

Size the bumper pipe thickness subject to the following constraint:

$$t_b \leq 0.75 \quad t_p\left(\frac{r_b}{r_p}\right)^{0.131} = 0.237 \text{ in for schedule 40} \qquad (3.6.2.2\text{-}24)$$

where:

t_b = Bumper pipe wall thickness

t_p = Process pipe wall thickness

r_b = Mean radius of bumper pipe

r_p = Mean radius of process pipe

The above identity assures that the bumper pipe will crush without crushing the process pipe upon impacting. Table 3.6–33 presents the energy absorbing capacity of a 4 inch schedule 80 pipe as functions of overall displacement and impact force (Peech et al., 1977). Interpolate to find the exact point of energy balance:

d = 2.34 in

E_p = 54.4 in-k

F = 32.4 kips

The impact force is thus 2.4 times the fluid blowdown force. A dynamic load factor of 2.0 is considered for the intermediate steel structure. Figures 3.6–29 and 3.6–30 illustrate the pipe crush bumpers.

Finally, determine the approximate time of peak restraint load to assure that the fluid force did not exceed 0.88 P_0A during the energy balance event:

$$t = \sqrt{\frac{2mL_h3}{3(L_n - L)}\frac{(g+d)^2}{g}\left(\frac{1}{F_bL_h - M_p}\right)} \qquad (3.6.2.2\text{-}25)$$

where:

m = The mass per unit length of the pipe

Thus:

t = 13.9 milliseconds

3.6.2.2.6 Local Pipe Indentation

The local shell indentation stiffness of the pipe is usually considered where other energy-absorbing mechanisms are not available at the point of impact. Examples include impacts into rigid displacement-limiting bumpers, concrete walls, and the omnidirectional restraint weldment (the latter interposes a significant mass between the impacting pipe and the energy absorbers).

Experimentally derived crush stiffnesses have been used. These were based on a series of pseudo-static pipe crush tests covering several crush geometries and a sufficient range of pipe thicknesses and diameters to develop parametric scaling laws (Peech et al., 1977). This was augmented by analyses to determine the sensitivity to material strength, dynamics, and variations in loading geometry. Computer Program LIDOP (ME-184) Local Indentation of Piping (Appendix Section 3A.2.14) augments the static crush tests.

3.6.2.2.7 Concrete Barrier Impact

In a pipe whip impact, the force on the barrier is a complex function of time depending primarily on the sudden deceleration of the pipe wall at the impact point (slug impact), the shell indentation of the pipe as it locally crushes against the wall, and the force transmitted to the impact point by the more gradual deceleration of the adjacent run of pipe. After impact, the pipe also transmits a more enduring force resulting from the continuing fluid blowdown. The concrete is affected by this much like any other missile impact, the only significant difference being the long term fluid force. To evaluate this postulated event, the pipe is transformed into an equivalent missile and the concrete is analyzed for scabbing and structural response using the procedure described in Section 3.5.3. The analysis for structural response includes the impulse of the initial impact, as well as the subsequent fluid blowdown force and other concurrent loads.

Four basic parameters must be determined to define the equivalent missile: the kinetic energy (or impulse), the impact velocity, the pipe crush stiffness, and the bearing area. The kinetic energy and velocity can be found by either of two methods:

1. Simplified Method - Use the total input energy (fluid blowdown force x distance of pipe travel) less the energy absorbed in pipe bending prior to impact. Compute the velocity using approximate formulae.

2. Lumped Parameter Dynamic Analysis (Section 3.6.2.2.4) - This method is especially suited for evaluating the impact of piping systems with complex geometries and can even consider multiple impact points. As an alternative to the kinetic energy, the impact force history (impulse) can be computed.

Regardless of which analysis method is used, the crush resistance of the equivalent missile and the bearing area are derived from the experimental data described in Section 3.6.2.2.6. These data are modified to account for the effect of dynamics and internal pressure.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

Required pipe rupture loads to determine the integrity of mechanical components are determined using the analytical methods described in Section 3.6.2.2. The load combinations for the components and for break exclusion regions are presented in Sections 3.9 and 3.6.2.1, respectively. Criteria for rupture restraints are presented in Section 3.6.2.3.1.

Jet impingement loadings are determined as follows:

1. Jet forces are represented by time-dependent forcing functions. The effects of the piping geometry, capacity of the upstream energy reservoir, source pressure, and fluid enthalpy are considered in utilizing these forcing functions.

2. The steady state jet force at the postulated break has a magnitude of:

$$T = KPA \qquad (3.6.2.3\text{-}1)$$

where:

P = System pressure prior to pipe break

A = Pipe break area

K = Jet coefficient (theoretical maximum)

The following K values are:

a. 1.26 for saturated steam, saturated water, and steam/water mixtures blowdown (presented in Section 3.6.2.2.2).

b. 2.00 for nonflashing subcooled water blowdown (presented in Section 3.6.2.2.2).

3. In calculating the jet impingement load on an object or target, the retarding action of the surrounding air along the jet path is neglected. The jet impingement pressure on the target is calculated by taking the jet force as being constant at all distances from, and normal to, the break area and by assuming that the jet stream diverges conically at a solid angle of 20 degrees for steam or water-steam mixtures. For those cases where the 20-degree divergence assumption is shown to be unnecessarily conservative for the blowdown of steam or steam-water mixtures, Moody's asymptotic jet expansion model is utilized (Moody 1969). Jet expansion is not applicable to cases involving saturated water or subcooled water blowdown which are below the saturation temperature at the corresponding ambient pressure beyond the break.

4. The proportion of the total jet force acting on a target is determined from the fraction of the jet intercepted and by the shape factor of the target. For a target with flat surface area normal to the center axis of the jet stream, the load is the product of the impingement pressure at the target and the intercepted jet area. In those cases where the target area is such that the intercepted jet stream is deflected rather than totally stopped, a shape factor which is less than unity and is a function of the target geometry is used in calculating the total jet impingement load. For a 20-degree divergence angle and target at a distance x, the pressure intensity at the target is:

$$P = P_1 d\left(\frac{d_1}{d_1 + 2x\tan 10}\right)^2 \qquad (3.6.2.3\text{-}2)$$

where:

P_1 = Pressure intensity at the source (psi)

d_1 = Diameter of the assumed circular break area (inch)

x = Normal distance between the break and the target (inch)

Since the jet impingement force is a dynamically applied load, the target is analyzed either by static methods using an appropriate dynamic load factor, or dynamically using elastic or inelastic structural response codes (Section 3.8.3.3). The load combinations and design allowables are given in Sections 3.8.3 and 3.9.

Attenuation of fluid jet impingement for target assessment of Design Basis Accident (DBA) mitigating systems utilizes the approach presented in NUREG/CR-2913 (Ref: Weigand, Thompson, and Tomasko).

3.6.2.3.1 Pipe Rupture Restraints

Two basic restraint types are used, elastic and energy-absorbing. Elastic restraints are generally used where displacements subsequent to a postulated pipe rupture must be minimized to restrict the break opening area, limit loads in the broken piping run, limit the pipe movement to protect some equipment, or to limit pressure buildup to minimize external loading on structures and equipment. Energy-absorbing restraints are used where the primary objective is to dissipate the kinetic energy of a ruptured pipe and prevent unrestricted pipe whip.

Elastic Restraints

Since elastic restraints are used to minimize displacements of the broken pipe, they are close gaped. For some applications, this requires that they contact the pipe during conditions other than a postulated rupture, in which case they are also designed as a pipe support. If an elastic restraint contacts the pipe following a rupture, it is designed according to the criteria for structural steel (Section 3.8.3) which in effect limits stresses to the elastic range.

Energy-Absorbing Restraints

Several approaches are used for energy absorption in pipe rupture restraints. In tension, stainless steel studs or straps are used, with a design limit of 50 percent of uniform ultimate strain. In compression, honeycomb panels or crushable pipes are used. The design limit for crushable energy absorbing components is defined in Section 3.6.2.2.1, Item 5.

Elastic intermediate structures of energy-absorbing restraints are designed to the criteria for structural steel (Section 3.8.3).

1. Pipe Crush Bumper - The pipe crush bumper absorbs impact energy in a direction toward the supporting structure. The energy absorber is a length of pipe placed normal to the axis of the process pipe. Subsequent to a rupture, the bumper pipe is crushed between its support structure and the moving process pipe. Energy is absorbed by deformation of the bumper pipe which forms a retaining recess in the bumper pipe. The bumper pipe is mechanically attached to its support by welding or bolting (Figures 3.6–29 and 3.6–30).

2. Laminated Strap Restraint - The laminated strap restraint is capable of absorbing impact loads in the outward direction from the supporting structure (Figure 3.6–22). The energy-absorbing component is a “U” shaped strap consisting of multiple strips (number and geometry depending on energy to be absorbed) of highly ductile material.

   This laminated design exhibits great flexibility in application. The design minimizes bending strains, permitting the strap to act mainly as a membrane during the postulated rupture event.

3. Omni-Directional Restraint - The omni-directional restraint is capable of absorbing impact loads applied in any direction in the plane of the restraint (Figure 3.6–31). This restraint consists of a base weldment, an arch, ductile stainless steel holddown studs on each side of the base weldment, and a crushable pipe. The primary function of the studs is to absorb impact energy in tension. The crushable pipe absorbs energy from impact loads acting in an inward direction. Side load impacts are absorbed by the combined action of the studs and crushable pipe.

   Combinations of pipe crush bumpers and laminated straps are used to achieve energy absorption over a range of impact directions up to a full 360 degrees.

3.6.2.4 Guard Pipe Assembly Design Criteria

Guard pipes were not used on Millstone 3.

3.6.3 REFERENCES FOR SECTION 3.6

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3.6-12 Letter from the NRC to Northeast Nuclear Energy Company (NNECO) approving NNECO’s Request for Exemption from a Portion of General Design Criterion 4 of Appendix A to 10 CFR 50 regarding the Need to Analyze Large Primary Loop Pipe Ruptures as the Structural Design Basis for Millstone Nuclear Power Station Unit 3. June 5, 1985.

3.6-13 Weigand, G. G., Thompson, S. L., Tomasko, D. - “Two Phase Jet Loads” - Sandia National Laboratories, Albuquerque, NM - January 1983 for the USNRC, NUREG/CR - 2913.

3.6-14 ASME Code Case N-716-1, Alternative Classification and Examination Requirements, Section XI, Division 1.

TABLE 3.6–1 HIGH-ENERGY SYSTEMS REMOTE FROM ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS

System	Code	Location	Maximum Operating Temperature (1) Approximately in Degrees Fahrenheit	Maximum Operating Pressure (1) Approximately in psig
Auxiliary Boiler Blowdown	ABD	Auxiliary Boiler Room, Turbine Building Condensate Polishing Area	228	7
Auxiliary Boiler Condensate	ABF	Auxiliary Boiler Room	228	165
Auxiliary Boiler Steam	ABM	Auxiliary Boiler Room	366	150
Chemical Feed Condensate (2)	CNC	Turbine Building	94	550
Chemical Feed Steam Generator	SGF	Turbine Building	446	1262
Cold Reheat	CRS	Turbine Building	374	183
Condensate Demineralizer Mixed Bed	CND	Condensate Polishing Area	94	550
Extraction Steam	ESS	Turbine Building	454	424
Feedwater Heater Relief Vents & Drains	SVH	Turbine Building	442	375
Feedwater Pump Recirculation and Balance Drum Leakoff	FWR	Turbine Building	367	1632
Feedwater Pump Turbine Steam & Exhaust	TFM	Turbine Building	557	1092
High Pressure Feedwater Heater Drains	HDH	Turbine Building	450	407
Hot Reheat	HRS	Auxiliary Boiler Room, Turbine Bldg.	510	106

TABLE 3.6–1 HIGH-ENERGY SYSTEMS REMOTE FROM ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

System	Code	Location	Maximum Operating Temperature (1) Approximately in Degrees Fahrenheit	Maximum Operating Pressure (1) Approximately in psig
Hot Water Heating	HVH	Condensate Polishing Area Turbine Building Service Building, Warehouse	270	185
Condenser Air Removal	ARC	Turbine Building	212	125
Main Feedwater	FWS	Turbine Building	446	1401
Radioactive Liquid Waste (3)	LWS	Turbine Building Waste Disposal Building, Yard	266	150
Instrument Air	IAS	Turbine Building	350	110
Low Pressure Feedwater Heater Drains	HDL	Turbine Building	374	519
Condensate Demineralizer Liquid Waste	LWC	Warehouse 5	250	63
Turbine Plant Miscellaneous Drains	DTM	Turbine Building	556	1092
Turbine Plant Sample	SST	Turbine Building	556	1106
Main Condensate (4)	CNM	Turbine Building	366	551
Nitrogen Supply	GSN	Turbine Building, Yard	Ambient	950
Condensate Make-Up & Draw Off	CNS	Turbine Building, Yard	250	531
Moisture Separator Reheater Vents & Drains	DSR	Turbine Building	536	936

TABLE 3.6–1 HIGH-ENERGY SYSTEMS REMOTE FROM ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

System	Code	Location	Maximum Operating Temperature (1) Approximately in Degrees Fahrenheit	Maximum Operating Pressure (1) Approximately in psig
Fire Protection Low Pressure CO_2	FPL	Auxiliary Boiler Room, Turbine Bldg.	60	300
Reactor Plant Aerated Vents	VAS	Waste Disposal Building	250	15
Moisture Separator Reheater Relief Valve				
Discharge and Bonnet Vent	HRS	Turbine Building	510	106
Moisture Separator Vents & Drains	DSM	Turbine Building	373	518
Turbine Generator Gland Seal & Exhaust Steam	TME	Turbine Building	557	1107

NOTES:

(1) Maximum Operating Temperature and Maximum Operating Pressure are tabulated for the Normal Plant Conditions. These are approximate values presented to classify a system as either high or moderate energy.

(2) The Chemical Feed Condensate High Energy piping extends from the Condensate Chemical Addition Feed Pump Discharge to the Condensate Lines.

(3) The High Energy portion of the Radioactive Liquid Waste piping consists of the Reboiler and the Distillate Piping from the Waste Evaporator to the Distillate Tank and back to the Waste Evaporator. This equipment is removed from service.

(4) The Condensate High Energy piping includes the Condensate Pump Discharge through all stages of the Low Pressure Heating System

TABLE 3.6–2 HIGH-ENERGY SYSTEMS IN PROXIMITY TO ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS

High Energy System	Code	Location	Maximum Operating Temperature (1) Approximately in Degrees Fahrenheit	Maximum Operating Pressure (1) Approximately in psig
Reactor Coolant - Normal Charging	RCS	Containment	557	2485
Normal Charging (2)	CHS	Containment, Auxiliary	190 (3)	2485
Reactor Coolant - Normal Letdown	RCS	Containment	557	2485
Normal Letdown (2)	CHS	Containment, Auxiliary	290	400
Seal Water Injection (2)	CHS	Containment, Auxiliary	130	2485
Reactor Coolant - Excess Letdown	RCS	Containment	557	2485
Reactor Coolant - Loop Fill	RCS	Containment	557	2485
Reactor Coolant - Loop Drains	RCS	Containment	557	2485
Reactor Coolant - High Pressure Safety Injection (4)	RCS	Containment	557	2485
High Pressure Safety Injection (5)	SIH	Auxiliary Building	Ambient	2485
Reactor Coolant - Low Pressure Safety Injection (4)	RCS	Containment	557	2485
Low Pressure Safety Injection Accumulator Discharge	SIL	Containment	Ambient	695
Reactor Coolant - Primary Coolant	RCS	Containment	557	2485

TABLE 3.6–2 HIGH-ENERGY SYSTEMS IN PROXIMITY TO ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS

High Energy System	Code	Location	Maximum Operating Temperature (1) Approximately in Degrees Fahrenheit	Maximum Operating Pressure (1) Approximately in psig
Reactor Coolant - Residual Heat Removal (6)	RCS	Containment	557	2485
Reactor Coolant - Accumulator Discharge	RCS	Containment	557	2485
Reactor Coolant - Pressurizer Surge	RCS	Containment	557	2485
Reactor Coolant - Pressurizer Spray	RCS	Containment	557	2485
Reactor Coolant - Pressurizer Safety	RCS	Containment	557	2485
Reactor Coolant - Pressurizer Relief	RCS	Containment	557	2485
Reactor Coolant - Loop Stop Valve Bypass	RCS	Containment	557	2485
Reactor Coolant - Loop Stop Valve Bypass - 1.5 inch Bypass Line	RCS	Containment	557	2485
Steam Generator Blowdown	BDG	Containment Main Steam Valve Building (MSVB)	543	975
Auxiliary Steam	ASS	Engineered Safety Features (ESF) Auxiliary Building	366	150
Hot Water Heating	HVH	Auxiliary Building Fuel Building Main Steam Valve Building (MSVB)	270	185

TABLE 3.6–2 HIGH-ENERGY SYSTEMS IN PROXIMITY TO ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS

High Energy System	Code	Location	Maximum Operating Temperature (1) Approximately in Degrees Fahrenheit	Maximum Operating Pressure (1) Approximately in psig
Boron Recovery (7)	BRS	Auxiliary Building	312	165
Containment Vacuum Pump Discharge	CVS	Auxiliary Building Containment	268	1.4
Auxiliary Condensate	CNA	Auxiliary Building	366	150
Reactor Plant Gaseous Drains	DGS	Containment Auxiliary Building	225	110
Radioactive Gaseous Waste (8)	GWS	Auxiliary Building	350	300
Turbine Plant Miscellaneous Drains	DTM	Main Steam Valve Building (MSVB) Engineered Safety Features (ESF)	575	1262
Control Building Air Conditioning (9)	HVC	Control Building	115	2450
Main Steam	MSS	Containment, Engineered Safety Features (ESF), Main Steam Valve Building (MSVB)	557	1092
Auxiliary Feedwater	FWA	Containment, ESF	100	1658
Main Feedwater	FWS	Containment, MSVB	446	1980

TABLE 3.6–2 HIGH-ENERGY SYSTEMS IN PROXIMITY TO ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS

High Energy System	Code	Location	Maximum Operating Temperature (1) Approximately in Degrees Fahrenheit	Maximum Operating Pressure (1) Approximately in psig
Nitrogen Supply (10)	GSN	Auxiliary Building Containment	90	950
Emergency Diesel Generator Air Start (11)	EGA	Emergency Generator Enclosure (EGE)	Ambient	500
Main Steam Safety Valve Vents & Drains	SVV	Main Steam Valve Building (MSVB)	510	741
Fire Protection Low Pressure CO_2	FPL	Emergency Generator Enclosure (EGE) Control Building	60	300
Emergency Diesel Generator Exhaust and Combustion (11)	EGD	Emergency Generator Enclosure (EGE)	820	0
Post Accident Sampling	SSP	Containment	617	2485
Reactor Plant Sampling	SSR	Auxiliary Building and Containment	680	2485
Reactor Plant Aerated Vents	VAS	Auxiliary Building	250	15
Steam Generator Chemical Feed	SGF	Containment, Main Steam Valve Building (MSVB)	446	1262

NOTES:

(1) Maximum Operating Pressure and Maximum Operating Temperature is tabulated for Normal Plant Conditions. These are approximate values presented only to classify the system as high or moderate energy.

(2) The Chemical & Volume Control (CHS) System High Energy piping extends from the Reactor Coolant System isolation valves to the Charging Pump suction (via the Letdown Line through the Regenerative and Non-Regenerative Heat Exchangers and the Volume Control Tank) and from the Charging Pump Discharge to the Reactor Coolant System (via the Charging Line). The portion of piping from the Charging Pump Discharge to the Reactor Coolant Pumps (for Seal Water Injection) also is High Energy. The remainder of the Chemical & Volume Control System is Moderate Energy piping.

(3) Normal Charging (CHS) Line Temperature at the outlet of the Regenerative Heat Exchanger may be as high as 500 Degrees F.

(4) The High Pressure Safety Injection (SIH) System, the Low Pressure Safety Injection (SIL) System, and the Containment Recirculation System are used only following a Loss of Coolant Accident (LOCA) thus these systems are excluded from pipe rupture analysis. Portions of piping included within the Reactor Coolant System pressure boundary are analyzed for pipe rupture effects (i.e. pipe whip and fluid jet impingement).

(5) The High Pressure Safety Injection (SIH) between the Charging Pump Discharge Header and the SIH Valves 3SIH*MV8801A, and 3SIH*MV8801B is High Energy.

(6) The Residual Heat Removal (RHS) System, although having pressure and temperature above the High Energy Threshold is classified as a Moderate Energy System as delineated in FSAR Section 3.6.2.1.2.3. Portions of piping included within the Reactor Coolant System pressure boundary are analyzed for pipe rupture effects (i.e. pipe whip and fluid jet impingement).

(7) Boron Recovery (BRS) System High Energy piping consists of the Boron Evaporator Recirculation and Distillation piping from the Boron Evaporator to the Boron Distillate Cooler and Boron Evaporator Bottom Cooler.

(8) Radioactive Gaseous Waste (GWS) System High Energy piping extends from the Degasifier Recovery Exchanger, the Degasifier, the Degasifier Recirculation Pumps and back to the Degasifier Recovery Exchanger.

(9) Control Building Air Conditioning (HVC) System High Energy components consists solely of the Control Room Air Bottle Supply.

(10) Portion of Nitrogen Gas System from Pressurized Nitrogen Gas Tubes to the Safety Injection Accumulator Tanks is High Energy piping. Pipe breaks are excluded in this line since the line is 1 inch diameter piping.

(11) The Emergency Diesel Generator (EGD) System and related auxiliaries is considered a single system for evaluating the effects of pipe break. A single failure is not postulated in the redundant system as discussed in FSAR Section 3.1.1.

TABLE 3.6–3 MODERATE-ENERGY SYSTEMS IN REMOTE FROM ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS

Moderate Energy System	Code	Location	Maximum Operating Temperature (1) Approximate Values Tabulated	Maximum Operating Pressure (1) Approximate Values Tabulated
Auxiliary Boiler Fuel Oil	FOA	Auxiliary Boiler Room	100°F	48 psig
Condensate Makeup & Drawoff	CNS	Turbine Building, Yard	120°F	136 psig
Condenser Air Removal	ARC	Turbine Building	170°F	125 psig
Condenser Tube Cleaning	CWA	Turbine Building	95°F	29 psig
Domestic Water	DWS	Auxiliary Boiler, Turbine Building, Service Building	140°F	80 psig
Feedwater Pump & Drive Lube Oil	FWL	Turbine Building	145°F	20 psig
Fire Protection - Water	FPW	Auxiliary Boiler Room, Turbine Building, Service Building, Waste Disposal Building, Transformer Area	110°F	122 psig
Generator Hydrogen (H_2) & Generator Carbon Dioxide (CO_2)	GMH	Turbine Building	55°F	70 psig
Instrument Air	IAS	Auxiliary Boiler Room, Turbine Building, Service Building, Waste Disposal Building	115°F	110 psig
Boron Recovery (2)	BRS	Waste Disposal Building, Yard	170°F	80 psig
Main Condensate (3)	CNM	Turbine Building	200°F	

TABLE 3.6–3 MODERATE-ENERGY SYSTEMS IN REMOTE FROM ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

Moderate Energy System	Code	Location	Maximum Operating Temperature (1) Approximate Values Tabulated	Maximum Operating Pressure (1) Approximate Values Tabulated
Hydrogen (H_2) Gas	GSH	Gas Storage Area, Turbine Building, Yard	90°F	100 psig
Nitrogen (N_2) Gas	GSN	Nitrogen Storage Pad, Auxiliary Boiler Room, Service Building	Ambient	200 psig
Primary Grade Water	PGS	Primary Grade Water Pump House, Yard	100°F	122 psig
Hot Water Pre-Heating	HVG	Service Building	200°F	
Service Air	SAS	Auxiliary Boiler Room, Turbine Building, Service Building, Waste Disposal Building, Circulating Water Pump House (CWPH), Warehouse 5	115°F	110 psig
Reactor Plant Aerated Vents	VAS	Waste Disposal Building	200°F	15 psig
Reactor Plant Gaseous Vents	VRS	Waste Disposal Building	200°F	75 psig
Reactor Plant Aerated Drains	DAS	Waste Disposal Building	200°F	70 psig
Service Water	SWP	Auxiliary Boiler Room, Turbine Building, Service Building	80°F	66 psig
Turbine Generator Lube Oil	TML	Turbine Building	100°F	0 psig

TABLE 3.6–3 MODERATE-ENERGY SYSTEMS IN REMOTE FROM ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

Moderate Energy System	Code	Location	Maximum Operating Temperature (1) Approximate Values Tabulated	Maximum Operating Pressure (1) Approximate Values Tabulated
Waste Oil Disposal	WOS	Turbine Building	100°F	15 psig
Waste Water Treatment	WTW	Turbine Building	Ambient	Atmospheric
Water Treating	WTS	Turbine Building, Yard	60°F	128 psig
Station Vacuum Priming	VPS	Turbine Building	115°F	20 psig
Reactor Plant Solid Waste	WSS	Waste Disposal Building	100°F	140 psig
Chemical Feed Chlorination	WTC	Circulating Water Pump House	80°F	248 psig
Turbine Plant Component Cooling Water	CCS	Turbine Building	95°F	125 psig
Auxiliary Feedwater (4)	FWA	Yard	100°F	60 psig
Condensate Demineralizer Component Cooling	CCD	Condensate Polishing Area	115°F	91 psig
Quench Spray	QSS	Yard	75°F	158 psig
Traveling Screen Wash & Disposal	SWT	Circulating Water Pump House	80°F	135 psig
Yard Vacuum Priming	VPS	Circulating Water Pump House	115°F	30 psig
Circulating Water	CWS	Turbine Building, Yard, Circulating Water Pump House	80°F	26 psig
Low Pressure Safety Injection	SIL	Yard	Ambient	30 psig

TABLE 3.6–3 MODERATE-ENERGY SYSTEMS IN REMOTE FROM ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

Moderate Energy System	Code	Location	Maximum Operating Temperature (1) Approximate Values Tabulated	Maximum Operating Pressure (1) Approximate Values Tabulated
Containment Recirculation Spray	RSS	Yard	115°F	225 psig

NOTES:

(1) Maximum Operating Pressure and Maximum Operating Temperature for the Normal Plant Conditions are tabulated. These values are approximate and are presented to classify the system as high or moderate energy.

(2) Boron Recovery High Energy piping consists of the Boron Evaporator Recirculation and Distillate piping from the Boron Evaporator to the Boron Distillate Cooler. Remainder of the Boron Recovery (BRS) System is Moderate Energy piping.

(3) Moderate Energy piping for the Main Condensate (CNM) System extends from the Condenser to the Condensate Pump Suction, and includes the Seal Water piping to the Feed Pump Exhaust Isolation Valves, Air Ejectors Condenser Loop Seal, Extraction Nonreturn Valves, 6th Point Drain Loop Seal, Condenser Vacuum Breaker Valves and Loop Seals.

(4) Auxiliary Feedwater and Recirculation (FWA) System is Moderate Energy piping from the Demineralized Water Storage Tank (DWST) to the suction side of the Auxiliary Feedwater pumps. This Moderate Energy piping includes the Recirculation piping.

TABLE 3.6–4 MODERATE-ENERGY SYSTEMS PROXIMATE TO ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS

Moderate Energy System	Code	Location	Maximum Operating Temperature (1) Approximate Values	Maximum Operating Pressure (1) Approximate Values
Auxiliary Feedwater & Recirculation (2)	FWA	Engineered Safety Features Building	100°F	48 psig
Residual Heat Removal (3)	RHS	Engineered Safety Features Building, Containment	350°F	575 psig
Service Water	SWP	Auxiliary Building, Engineered Safety Features Building, Circulating Water Pump House, Emergency Generator Enclosure, Control Building	95°F	66 psig
Fire Protection - Water	FPW	Auxiliary Building, Emergency Generator Enclosure, Control Building, Fuel Building, Main Steam Valve Building, Containment	110°F	122 psig
Component Cooling Water	CCP	Auxiliary Building, Engineered Safety Features Building, Fuel Building, Containment	137°F	186 psig
Control Building - Chilled Water	HVK	Control Building	55°F	70 psig
Instrument Air	IAS	Auxiliary Building, Engineered Safety Features Building, Fuel Building, Emergency Generator Enclosure, Control Building, Containment, Hydrogen Recombiner Building, Main Steam Valve Building	115°F	110 psig

TABLE 3.6–4 MODERATE-ENERGY SYSTEMS PROXIMATE TO ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

Moderate Energy System	Code	Location	Maximum Operating Temperature (1) Approximate Values	Maximum Operating Pressure (1) Approximate Values
Radioactive Solid Waste	WSS	Auxiliary Building, Fuel Building	100°F	140 psig
Charging Pump Cooling	CCE	Auxiliary Building	105°F	60 psig
Fuel Pool Cooling & Purification	SFC	Auxiliary Building, Engineered Safety Features Building, Containment, Fuel Building	150°F	135 psig
Chemical & Volume Control (4)	CHS	Auxiliary Building	190°F	220 psig
Reactor Plant Aerated Drains	DAS	Auxiliary Building, Engineered Safety Features Building, Containment, Fuel Building	200°F	70 psig
Domestic Water	DWS	Auxiliary Building, Control Building	140°F	80 psig
Fire Protection - Halon	FPG	Control Building	Ambient	36 psig
Containment Instrument Air	IAC	Containment	120°F	110 psig
Containment Atmosphere Monitoring	CMS	Auxiliary Building, Containment	100°F	Atmospheric
Safety Injection Pump Cooling	CCI	Engineered Safety Features Building	110°F	35 psig
Chilled Water, Containment Structure Ventilation	CDS	Auxiliary Building, Containment, Engineered Safety Features Building	90°F	140 psig
Piping for Containment Purge Air	HVU	Auxiliary Building, Containment	120°F	5 psig
Containment Leak Monitoring	LMS	Containment, Auxiliary Building	90°F	0 psig

TABLE 3.6–4 MODERATE-ENERGY SYSTEMS PROXIMATE TO ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

Moderate Energy System	Code	Location	Maximum Operating Temperature (1) Approximate Values	Maximum Operating Pressure (1) Approximate Values
Hydrogen Recombiner	HCS	Hydrogen Recombiner Building	135°F	0 psig
Chemical Feed Chlorination, Emergency Diesel Generator	WTC	Circulating Water Pumphouse	80°F	248 psig
Jacket & Intercooler Water	EGS	Emergency Generator Enclosure	180°F	50 psig
Hydrogen Gas	GSH	Auxiliary Building	90°F	100 psig
Nitrogen Supply	GSN	Auxiliary Building, Containment	Ambient	200 psig
Quench Spray	QSS	Engineered Safety Features Building, Containment	75°F	158 psig
Neutron Shield Tank Cooling	NSS	Containment	135°F	20 psig
Sanitary	PBS	Control Building	85°F	Static Head
Reactor Plant Gaseous Vents	VRS	Auxiliary Building, Containment	200°F	75 psig
Service Air	SAS	Auxiliary Building, Engineered Safety Features Building, Fuel Building, Emergency Generator Enclosure, Control Building, Hydrogen Recombiner Building	115°F	110 psig
Emergency Generator Fuel	EGF	Emergency Generator Enclosure	120°F	32 psig
Low Pressure Safety Injection	SIL	Engineered Safety Features Building, Containment	115°F	235 psig

TABLE 3.6–4 MODERATE-ENERGY SYSTEMS PROXIMATE TO ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

Moderate Energy System	Code	Location	Maximum Operating Temperature (1) Approximate Values	Maximum Operating Pressure (1) Approximate Values
Floor Equipment Drainage	DNF	Circulating Water Pumphouse, Engineered Safety Features Building, Auxiliary Building, Fuel Building	Ambient	Ambient
Roof Drainage	DNR	Engineered Safety Features Building, Auxiliary Building, Fuel Building, Main Steam Valve Building, Control Building	90°F	Ambient
Reactor Coolant Pump Oil Collection	FPR	Containment	160°F	Atmospheric
Primary Grade Water	PGS	Auxiliary Building, Engineered Safety Features Building, Fuel Building, Containment	100°F	122 psig
Containment Recirculation Spray	RSS	Engineered Safety Features Building, Containment	115°F	225 psig
Boron Recovery (5)	BRS	Auxiliary Building	170°F	80 psig
Emergency Generator Lube Oil	EGO	Emergency Generator Enclosure	160°F	150 psig
Condenser Air Removal	ARC	Auxiliary Building	170°F	125 psig
Reactor Plant Gaseous Drains	DGS	Auxiliary Building, Containment	165°F	104 psig
Reactor Plant Aerated Vents	VAS	Auxiliary Building	120°F	0 psig
Containment Vacuum (6)	CVS	Auxiliary Building, Containment	90°F	5 psig

TABLE 3.6–4 MODERATE-ENERGY SYSTEMS PROXIMATE TO ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS (CONTINUED)

Moderate Energy System	Code	Location	Maximum Operating Temperature (1) Approximate Values	Maximum Operating Pressure (1) Approximate Values
Feedwater Lube Oil	FWL	Engineered Safety Features Building	20°F	140 psig
Hot Water Pre-Heating	HVG	Auxiliary Building	185°F	170 psig

NOTES:

(1) Maximum Operating Pressure and Maximum Operating Temperature are tabulated for the Normal Plant Conditions. These are approximate values presented to classify the system as high and moderate energy.

(2) The Auxiliary Feedwater and Recirculation (FWA) System Moderate Energy piping extends from the Demineralized Water Storage Tank (DWST) to the suction side of the Auxiliary Feedwater Pumps which includes the Recirculation piping.

(3) Residual Heat Removal (RHS) System is a Moderate Energy System in accordance with the 2% Rule. The 2% Rule is defined in FSAR Section 3.6.2.1.2.3.

(4) The Moderate Energy piping for the Chemical Volume Control (CHS) System extends from the Letdown Heat Exchanger pressure reducing valve via the Mixed Bed Demineralizer and the Thermal Regeneration Remineralizer lines to the Volume Control Tank.

(5) Boron Recovery (BRS) System High Energy piping consists of the Boron Evaporator Recirculation and Distillate piping from the Boron Evaporator to the Boron Distillate Cooler. Remainder of the Boron Recovery (BRS) System is Moderate Energy piping.

(6) Containment Vacuum (CVS) System Moderate Energy piping extends from Inside Containment to the Containment Vacuum Pump suction. The remainder of Containment Vacuum (CVS) System piping is High Energy.

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN

System	Building	Required Safety Function
Reactor Coolant (RCS) (1) (2)	Containment	Reactor Core Integrity and Heat Removal
Reactor Vessel 3RCS*RV1	Containment	Pressure Boundary integrity needed to maintain fuel within acceptable temperature limits
Reactor Vessel 3RCS*RV1 Head Vent	Containment	Provide Letdown for Reactivity & Inventory Control and Reactor Coolant Pressure Control
Pressurizer 3RCS*TK1	Containment	Pressure Boundary - Maintain RCS Pressure Control
Steam Generators 3RCS*SG1A, SG1B, SG1C, and SG1D	Containment	Remove Heat from Core
Reactor Coolant Pumps 3RCS*P1A, P1B, P1C, and P1D	Containment	Maintain Pressure Boundary Integrity
Reactor Coolant Loop Stop Valves 3RCS*MV8002A, 3RCS*MV8002B, 3RCS*MV8002C, and 3RCS*MV8002D	Containment	Maintain Pressure Boundary Integrity - Primary Boundary
Reactor Coolant Loop Stop Valve Bypass	Containment	
Piping and Valves on the Reactor Coolant System Up to the Class 1/Class 2 Boundary as Delineated in FSAR Table 3.5–2	Containment	Maintain Pressure Boundary Integrity - Primary Boundary
Pressurizer Power Operated Relief Valves		Operability - Alternate Means of RCS Depressurization
(PORV's) 3RCS*PCV455A, and 3RCS*PCV456	Containment	
Control Rod Drive Mechanism	Containment	Insert Control Rods to Stabilize Plant at Hot Standby
Pressurizer Safety Valves 3RCS*SV8010A, B, C	Containment	Operability - Overpressure Protection for RCS System

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
PORV Block Valves 3RCS*MV8000A, and 3RCS*MV8000B	Containment	Operability - Support PORV Function
Letdown/RCS Isolation Valves 3RCS*LCV459 & LCV460	Containment	Isolates Reactor Coolant System for Letdown Line Break
Residual Heat Removal (RHS) [(1)] [(2)]	ESF	Low Pressure Safety Injection Path and Reactor Coolant Heat Removal
Residual Heat Removal Heat Exchanger Flow Control Valves		Operability - Supports RHS Function for Controlled Cooldown
3RHS*FCV610, and 3RHS*FCV611	ESF	Operability - Supports RHS Function for Controlled Cooldown
3RHS*FCV618, and 3RHS*FCV619	ESF	Operability - Supports RHS Function for Controlled Cooldown
3RHS*HCV606 and 3RHS*HCV607	ESF	Operability - Supports RHS Function for Controlled Cooldown
Three RHS Series Isolation Valves 3RHS*MV8701A, MV8701B, 8701C, 8702A, 8702B, and 8702C	Containment & ESF	Operability - Initiate Second Stage of RHS by Opening all three Isolation Valves on either Train
Residual Heat Removal (RHS) Pumps 3RHS*P1A,B	ESF	Operability - Takes Suction from RCS Hot Leg & RWST
RHS Heat Exchangers 3RHS*E1A, and 3RHS*E1B	ESF	Operability - Cools Reactor Coolant from Hot Leg and Discharges Reactor Coolant Back to Cold Leg via Safety Injection System
RHS Pump Suction 3SIL*MV8812A, and 3SIL*MV8812B	ESF	Operability - During Second Stage of Residual Heat Removal and Long Term Sump Recirculation

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
RHS Pump Discharge Containment Isolation Valves 3SIL*MV8809A, and 3SIL*MV8809B	ESF	Operability - Containment Isolation and Long Term Sump Recirculation
3SIL*MV8716A and 3SIL*MV8716B	ESF	Operability Required to Support Long Term Sump Recirculation
3SIL*MV8804A and 3SIL*MV8804B	ESF	Operability Required to Support Long Term Sump Recirculation
Chemical & Volume Control (CHS) [1] [2]		High Pressure Safety Injection, Reactivity & Inventory Control
Normal Charging Pumps 3CHS*P3A,B,CAuxiliary	Auxiliary	Operability - Reactivity & Inventory Control as well as Pressure Control, and Safety Injection
Boric Acid Transfer Pumps 3CHS*P2A,B	Auxiliary	Operability
Boric Acid Tanks 3CHS*TK5A,B	Auxiliary	Maintain Pressure Boundary Integrity
Boric Acid Blender 3CHS*BL1	Auxiliary	Maintain Pressure Boundary Integrity
Charging Pump Suction 3CHS*LCV112B, C, D, and E	Auxiliary	Operability - Reactivity & Inventory Control
Charging Pump Discharge Containment Isolation Valves 3CHS*MV8105, and 3CHS*MV8106	Auxiliary	Operability - Containment Isolation
Normal Letdown Line Containment Isolation Valve 3CHS*CV8152 and 3CHS*CV8160	Auxiliary	Operability - Containment Isolation
Letdown Line 3CHS*AV8149A,B,C	Containment	Maintain Pressure Boundary Integrity

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
Reactor Coolant Pump Seal Water Injection Containment Isolation Valves 3CHS*MV8109A, 3CHS*MV8109B, 3CHS*MV8109C, and 3CHS*MV8109D	Auxiliary	Operability - Containment Isolation
3CHS*MV8111A, 3CHS*MV8111B, and 3CHS*MV8111C	Auxiliary	Operability - Supports ECCS Function
3CHS*MV8512A, and 3CHS*MV8512B	Auxiliary	Operability - Supports ECCS Function
3CHS*MV8511A, and 3CHS*MV8511B	Auxiliary	Operability - Supports ECCS Function
3CHS*MV8110	Auxiliary	Operability - Supports ECCS Function
High Pressure Safety Injection (SIH) (1) (2)	ESF	Emergency Core Cooling System (ECCS)
High Pressure Safety Injection Pumps 3SIH*P1A,B	ESF	Operability-Supply Borated Water to all Four RCS Loops
Valves 3SIH*MV8801A, and 3SIH*MV8801B	Auxiliary	Operability - Containment Isolation
3SIH*MV8802A, 3SIH*MV8802B, and 3SIH*MV8835	ESF	Operability - Containment Isolation
3SIH*MV8813, 3SIH*MV8814, and 3SIH*MV8920	ESF	Operability - Supports ECCS Function
Low Pressure Safety Injection (SIL) (1) (2)		
Safety Injection Accumulators 3SIL*TK1A, 1B, 1C, and 1D	Containment	Inject Borated Water to RCS Cold Leg and Large LOCA
Safety Injection Accumulator Isolation Valves 3SIL*MV8808A, MV8808B, MV8808C, and MV8808D	Containment	Operability - Preclude Accumulator Injection during RCS Depressurization - Accumulators are Isolated or Vented
Safety Injection Accumulator Purge Valves 3SIL*SV8875A, B, C, D, E, F, G, and H	Containment	Operability - Accumulator Purge Valves

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
3SIL*HCV943A, and 3SIL*HCV943B	Containment	Operability - Purge Accumulator if Accumulator Isolation Valves Fail to Close
Main Feedwater (FWS) (1) (2)		Decay Heat Removal by Supplying Cooling Water to the Steam Generators
Main Feedwater Valves 3FWS*CTV41A, B, C, D	MSV	Operability - Containment Isolation
Main Feedwater Valves 3FWS*FCV510, 520, 530, and 540	MSV	Operability - Valves Close on Feedwater Isolation Signal
Main Feedwater Bypass Valves 3FWS*LV550, 560, 570, and 580	MSV	Operability - Valves Close on Feedwater Isolation Signal
Auxiliary Feedwater (FWA) (1) (2)		Decay Heat Removal by Supplying Cooling Water to the Steam Generators
Auxiliary Feedwater Motor Driven Pumps 3FWA*P1A, and 3FWA*P1B	ESF	Operability - Provide Redundant Water Supply to at Least Two Steam Generators Until Initiation of RHS
Auxiliary Feedwater Turbine Driven Pump 3FWA*P2	ESF	Operability - Alternate Path for Water Supply
Demineralized Water Storage Tank 3FWA*TK1	Yard	Pressure Boundary - Supply for Auxiliary Feedwater Pump Suction
Auxiliary Feedwater Supply Valves 3FWA*MOV35A, B, C, D and 3FWA*HV36A, B, C, D	ESF	Operability - Containment Isolation
Auxiliary Feedwater Supply Hand Control Valves 3FWA*HV31A, B, C, and D and 3FWA*HV32A, B, C, and D	ESF	Operability - Control Flow to Steam Generator Level to Support Cooldown
Containment Recirculation Spray (RSS) (1) (2)	ESF	Recirculation of Emergency Core Cooling System (ECCS) Water following a Design Basis Accident (DBA)

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
Containment Recirculation Spray Pumps 3RSS*P1A, 3RSS*P1B, 3RSS*P1C, and 3RSS*P1D	ESF	Operability Containment Recirculation Spray Pumps are started on receipt of a RWST Low-Low signal coincident with a CDA Signal
Containment Recirculation Spray Coolers 3RSS*E1A, 3RSS*E1B, 3RSS*E1C, and 3RSS*E1D	ESF	Operability Containment Recirculation Water Flows Shell side of the Heat Exchangers
Containment Recirculation Spray Valves 3RSS*MOV20A, B, C, and D	ESF	Operability - Containment Isolation
Containment Recirculation Spray Suction Isolation Valves 3RSS*MOV23A, B, C, D	ESF	Operability - Containment Isolation
3RSS*MOV8837A, and B and 3RSS*MOV8838A, and B	ESF	Operability - Switchover to Long Term Sump Recirculation
Quench Spray (QSS) (1) (2)	ESF	Maintain Integrity of the Containment by Removing Heat from Containment Atmosphere following a LOCA or a Main Steam Line or Feedwater Line Break
Quench Spray Pumps 3QSS*P3A, and 3QSS*P3B	ESF	Operability - Provide Borated Water to QSS Headers
Refueling Water Storage Tank 3QSS*TK1	Yard	Provide Chilled Water Supply to QSS Pump Suction and suction for the RHS, SIH, and CHS Pumps for the ECCS Function
Quench Spray Valves 3QSS*MOV34A,B	ESF	Operability - Containment Isolation and Open subsequent to a CDA Signal to Supply QSS Headers
Steam Generator Blowdown (BDG)		Maintain Pressure Boundary Integrity to preserve Heat Removal via Steam Generators

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
Steam Generator Blowdown Containment Isolation Valves 3BDG*CTV22A, CTV22B, CTV22C, and CTV22D	MSV	Operability - Containment Isolation
Main Steam (MSS) (1) (2)		Maintain Pressure Boundary of Steam Generators & Control Steam Release for Heat Removal
Main Steam Isolation Valves 3MSS*CTV27A,B,C,D	MSV	Operability - Containment Isolation
Main Steam Safety Valves 3MSS*RV22A, B, C, D	MSV	Operability & Over Pressure Protection
Main Steam Relief Valves 3MSS*RV23A, B, C, and D	MSV	Operability & Over Pressure Protection
3MSS*RV24A,B,C and D 3MSS*RV25A, B, C and D and 3MSS*RV26A, B, C and D	MSV	Operability & Over Pressure Protection
Main Steam to Turbine Driven Auxiliary Feedwater Pump 3MSS*MOV17A,B,D and 3MSS*AOV31A, 3MSS*AOV31B, and 3MSS*AOV31D	ESF	Operability - Containment Isolation Operability - Support Turbine Driven Auxiliary Feedwater Pump Operation
Main Steam Pressure Relieving Bypass Valves 3MSS*MOV74A, B, C, D	MSV	Operability - Provides Means to Reduce Steam Pressure during First Stage of RCS Cool Down
Main Steam Pressure Relieving Valves 3MSS*PV20A, 20B, 20C, and 20D	MSV	Operability - Primary Means to Reduce Steam Pressure during First Stage of RCS Cool Down
Main Steam Pressure Relieving Block Valves 3MSS*MOV 18A, B, C, D	MSV	Operability - Isolation
3MSS*HV28A, 3MSS*HV28B, 3MSS*HV28C, and 3MSS*HV28D	MSV	Operability - Containment Isolation

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
Reactor Plant Component Cooling Water (CCP) (1) (2)	Auxiliary	Remove Heat from Safety Components and Transfer Heat to the Ultimate Heat Sink
Reactor Plant Component Cooling Water Pumps 3CCP*P1A, 3CCP*P1B, and 3CCP*P1C	Auxiliary	Operability - Component Cooling Water Pumps Circulate Water Through the Various Closed Loops
Reactor Plant Component Cooling Water Heat Exchanger 3CCP*E1A, 3CCP*E1B, and 3CCP*E1C	Auxiliary	Operability - Transfer Heat to Service Water System
Residual Heat Removal Heat Exchanger Discharge Valves 3CCP*FV66A, and 3CCP*FV66B	ESF	Operability - Normally Closed Valves Prevent Flow Through RHS Heat Exchanger During Normal Plant Operation Valves Open Upon Residual Heat Removal System Initiation to Provide Component Cooling Water to RHS Heat Exchanger
Reactor Plant Component Cooling Water Surge Tank 3CCP*TK1	Auxiliary	Pressure Boundary Integrity - Provide Sufficient NPSH to CCP Pumps
Reactor Plant Component Cooling Water Valves 3CCP*MOV45A, and 3CCP*MOV45B	Auxiliary	Operability - Containment Isolation
Valves 3CCP*MOV48A, and 3CCP*MOV48B	Auxiliary	Operability - Containment Isolation
Valves 3CCP*MOV49A, and 3CCP*MOV49B	Auxiliary	Operability - Containment Isolation
Charging Pump Cooling (CCE) (1) (2)	Auxiliary	Operability - Supports CHS Pump Function
Safety Injection Pump Cooling (CCI) (1) (2)	ESF	Operability - Supports SIH Pump Function
Hot Water Heating (HVH) (3)	Auxiliary	None

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
Isolation Valves 3HVH-AOV135A, B and 3HVH-AOV136A, B	Service	Operability - A High Energy Line Break on HVH in the Auxiliary Building Trips a Pressure Switch which Closes these Valves to Isolate Flow to the Auxiliary Building
Auxiliary Steam (ASS) (3)	Auxiliary	None
Isolation Valves 3ASS-AOV102A and 3ASS-AOV102B	Turbine	Operability - A High Energy Line Break on Auxiliary Steam (ASS) System in the Auxiliary Building Trips a Pressure Switch which Closes these Valves to Isolate Flow to the Auxiliary Building
Service Water (SWP) (1) (2)		Transfer Heat from Reactor Cooling Systems to Ultimate Heat Sink (i.e., Ocean)
Service Water Pumps 3SWP*P1A, B, C, D	CWPH	Operability Provide Cooling Water to Safety Related Components
Containment Recirculation Coolers Service Water 3SWP*MOV54A, B, C, D and 3SWP*MOV57A, B, C, D	ESF	Operability - Provide Cooling Water for Containment Recirculation Coolers 3RSS*E1A, E1B, E1C, and E1D
Reactor Plant Component Cooling Water Heat Exchanger Service Water 3SWP*MOV50A, and MOV50B	Auxiliary	Operability - Normally Open Providing Cooling Water for CCP and Close Subsequent to a LOCA or a HELB Inside Containment
Service Water Valves 3SWP*AOV39A and 3SWP*AOV39B	EGE	Operability - Valves Open to Supply Cooling Water to Diesel Generators
Emergency Diesel Generator Fuel Oil (EGF) (1) (2)	EGE	Operation of the Diesel Generators
Emergency Diesel Generator Fuel Oil (EGF) Storage Motor Driven Fuel Pump 3EGF*P2A, and 3EGF*P2B	EGE	Operability - Supports Diesel Generator Function

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
Engine Driven Fuel Pump 3EGF*P3A, and 3EGF*P3B	EGE	Operability - Supports Diesel Generator Function
Fuel Oil Storage Tanks 3EGF*TK1A, and 3EGF*TK1B	Fuel Oil Vault	Pressure Boundary - Supports Diesel Generator Function
Fuel Oil Day Tank 3EGF*TK2A, and 3EGF*TK2B	EGE	Pressure Boundary - Supports Diesel Generator Function
Fuel Oil Transfer Pumps 3EGF*P1A,1B,1C, and 1D	EGE	Operability - Supports Diesel Generator Function
Emergency Diesel Generator Cooling Water (EGS) (1) (2)	EGE	Operator of the Diesel Generators
Emergency Diesel Generator Air Cooler Water Heat Exchanger 3EGS*E1A, and 3EGS*E1B	EGE	Operability - Supports Diesel Generator Function
Emergency Diesel Generator Jacket Water Electric Heaters 3EGS*H1A, and 3EGS*H1B	EGE	Pressure Boundary - Supports Diesel Generator Function
Emergency Diesel Generator Jacket Water Circulating Water Pump 3EGS*P2A, and 3EGS*P2B	EGE	Pressure Boundary - Supports Diesel Generator Function
Emergency Diesel Generator Fresh Water Expansion Tank 3EGS*TK1A, and 3EGS*TK1B	EGE	Pressure Boundary - Supports Diesel Generator Function
Emergency Diesel Generator Jacket Water Cooler 3EGS*E2A, and 3EGS*E2B	EGE	Operability - Supports Diesel Generator Function
Emergency Diesel Generator Cooling Water (EGS) Lube Oil Heat Exchanger 3EGS*E3A, and 3EGS*E3B	EGE	Operability - Supports Diesel Generator Function
Emergency Diesel Generator Cooling Water (EGS) Governor Lube Oil Cooler 3EGS*E4A, and 3EGS*E4B	EGE	Operability - Supports Diesel Generator Function

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
Emergency Diesel Generator Cooling Water (EGS) Engine Driven Pump 3EGS*P1A and 3EGS*P1B	EGE	Operability - Supports Diesel Generator Function
Emergency Diesel Generator Cooling Water (EGS) Engine Driven Intercooler Water Pump 3EGS*P3A and 3EGS*P3B	EGE	Operability - Supports Diesel Generator Function
Emergency Diesel Generator Lube Oil (EGO) (1) (2)	EGE	Operation of the Diesel Generators
Emergency Diesel Generator Lube Oil (EGO) Lube Oil Pumps 3EGO*P3A, and 3EGO*P3B	EGE	Operability - Supports Diesel Generator Function
Lube Oil Pumps 3EGO*P1A, and 3EGO*P1B	EGE	Operability - Supports Diesel Generator Function
Lube Oil Pumps 3EGO*P2A, and 3EGO*P2A, and 3EGO*P2B	EGE	Operability - Supports Diesel Generator Function
Lube Oil Pumps 3EGO*P4A, and 3EGO*P4B	EGE	Operability - Supports Diesel Generator Function
Emergency Diesel Generator Pre Lube Oil (EGO) Filter 3EGO*FLT1A, and 3EGO*FLT1B	EGE	Operability - Supports Diesel Generator Function
Emergency Diesel Generator Lube Oil (EGO) Strainers 3EGO*STR1A,5A and 3EGO*STR1B,5B	EGE	Operability - Supports Diesel Generator Function
Emergency Diesel Generator Lube Oil (EGO) Lube Oil Suction Strainer 3EGO*STR4A, and 3EGO*STR4B	EGE	Operability - Supports Diesel Generator Function
Emergency Diesel Generator Lube Oil (EGO) Pre Lube Oil and Filter Pump Suction Strainer 3EGO*STR2A, and 2B	EGE	Operability - Supports Diesel Generator Function
Emergency Diesel Generator Lube Oil (EGO) Pre Lube Oil Heater 3EGO*H1A, and 3EGO*H1B	EGE	Operability - Supports Diesel Generator Function

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
Emergency Diesel Generator Air Start (EGA) (2)	EGE	Operation of the Diesel Generators
Emergency Diesel Generator Air Start Air Receiver Tanks 3EGA*TK1A, 3EGA*TK1B and 3EGA*TK2A, 3EGA*TK2B	EGE	Pressure Boundary - Air Start Function
Emergency Diesel Generator Air Start (EGA) Air Tanks 3EGA*TK3A and 3EGA*TK3B	EGE	Pressure Boundary - Air Start Function
Diesel Generator Exhaust & Combustion (EGD) (1) (2)	EGE	Operation of the Diesel Generators
Emergency Diesel Generator Exhaust & Combustion (EGD) Exhaust Silencer (i.e. Muffler) 3EGD*SIL3A, 3B	EGE	Pressure Boundary - Supports Diesel Generator Function
Emergency Diesel Generator Exhaust & Combustion (EGD) Intake Air Filter and Silencer 3EGD*SIL1A, and 1B	EGE	Pressure Boundary - Supports Diesel Generator Function
Emergency Diesel Generator Exhaust & Combustion (EGD) Intake Air Silencer 3EGD*SIL2A, and 3EGD*SIL2B	EGE	Pressure Boundary - Supports Diesel Generator Function
Post DBA Hydrogen Recombiner Control Valves 3HCS*V2, V3, and V6, and 3HCS*V9, V10, and V13	Recombiner	Operability - Containment Isolation
Containment Isolation (1) (2)	Containment and Connecting Structures	Maintain Containment Integrity After an Accident
Containment Isolation Valves (4)	Containment and Connecting Structures	Operability - Powered Valves Required to Isolate Containment

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
Heating, Ventilation, and Air Conditioning and Cooling (HVAC) (1) (2)	Containment and Connecting Structures	Operability (5)
Instrument and Controls (I&C) (1) (2)	Containment and Connecting Structures	Operability (6)
Electrical Equipment (1) (2)	Containment and Connecting Structures	Operability (7)
Building Structures (1)		Support and Protect Safety Related Systems
Containment Structure	Containment	Secondary Barrier Providing Isolation of Reactor Coolant System and the Tertiary Barrier Between the Reactor Core and the Outside Atmosphere
Neutron Shield Tank 3NSS*TK2	Containment	Primary Barrier Providing Isolation of Reactor Core
Primary Shield Wall	Containment	Primary Barrier Providing Isolation of Reactor Core
Reinforced Concrete Internal Substructure	Containment	Secondary Barrier Providing Isolation of Reactor Core
Reinforced Concrete Internal Floors, Walls, and Enclosures	Containment	Secondary Barrier Providing Isolation of Reactor Coolant System
Containment Structure Liner	Containment	Provide a Leak Tight Membrane
Containment Penetrations	Containment	Maintain Pressure Boundary Integrity
Auxiliary Building		Supports Systems required for Safe Shutdown
Engineered Safety Features Building		Supports Systems required for ECCS and Safe Shutdown
Main Steam Valve Building		Supports Systems required for Safe Shutdown

TABLE 3.6–5 ESSENTIAL STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE REACTOR SHUTDOWN (CONTINUED)

System	Building	Required Safety Function
Emergency Diesel Generator Enclosures		Supports Systems required for Emergency Power
Control Building		Supports Systems required for Safe Shutdown
Circulating Water Pumphouse		Supports Systems required for Safe Shutdown
Containment Enclosure Building		Supports Systems required for Post Accident Mitigation

NOTES:

(1) Hazards Review Program addresses Safe Shutdown Requirements which are scenario dependent. Each scenario is evaluated on a case by case basis given an initiating event.

(2) Piping and Valves are required for Essential Pressure Boundary.

(3) The Hot Water Heating (HVH) System and the Auxiliary Steam (ASS) System are Non Safety Related Systems.

(4) Containment Isolation Valves are delineated in FSAR Table 6.2–65.

(5) HVAC Systems required to support Systems and Components delineated in this Table and HVAC Systems required for Accident Mitigation and Safe Shutdown are described in FSAR Chapter 9.

(6) Instruments and Control (I&C) Equipment required to support Systems and Components delineated in this Table and I&C Equipment required for Accident Mitigation and Safe Shutdown are described in FSAR Chapter 7.

(7) Electrical Equipment required to support the Systems and Components delineated in this Table and the Electrical Systems required for Accident Mitigation and Safe Shutdown are described in FSAR Chapter 8.

TABLE 3.6–6 POSTULATED BREAKS MAIN STEAM SYSTEM

Line Designation	Break #	Building	Elevation	Break Type [(1)]	Total Additive Stress	Figure
3-MSS-030-92-2	1	Containment	89'-1"	CB	N/A - Terminal End	3.6–8
3-MSS-030-92-2	2	Containment	76'-6"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-92-2	3	Containment	66'-3"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-104-3	17	MSVB	66'-3"	CB	N/A - Terminal End	3.6–8
3-MSS-030-25-4	4	Turbine	56'-9"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-25-4	5	Turbine	56'-5"	CB & LS	Above Threshold	3.6–8
3-MSS-030-25-4	6	Turbine	54'-8"	CB	N/A - Terminal End	3.6–8
3-MSS-030-67-4	7	Turbine	54'-6"	CB	(2)	3.6–8
3-MSS-030-67-4	8	Turbine	54'-6"	CB & LS	(2)	3.6–8
3-MSS-030-67-4	9	Turbine	54'-6"	CB	(2)	3.6–8
28" G.E. Piping	10	Turbine	40'-0"	CB	(2)	3.6–8
28" G.E. Piping	11	Turbine	36'-0"	CB & LS	(2)	3.6–8
28" G.E. Piping	12	Turbine	32'-6"	CB & LS	(2)	3.6–8
28" G.E. Piping	13	Turbine	31'-10"	CB & LS	(2)	3.6–8
28" G.E. Piping	14	Turbine	35'-4"	CB & LS	(2)	3.6–8
28" G.E. Piping	15	Turbine	72'-11"	CB & LS	(2)	3.6–8
28" G.E. Piping	16	Turbine	76'-5"	CB	(2)	3.6–8
3-MSS-030-93-2	1	Containment	89'-11"	CB	N/A - Terminal End	3.6–8

TABLE 3.6–6 POSTULATED BREAKS MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Total Additive Stress	Figure
3-MSS-030-93-2	2	Containment	93'-8"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-93-2	3	Containment	66'-3"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-105-3	16	MSVB	66'-3"	CB	N/A - Terminal End	3.6–8
3-MSS-030-26-4	4	Turbine	56'-9"	CB & LS	Above Threshold	3.6–8
3-MSS-030-26-4	5	Turbine	56'-6"	CB & LS	Above Threshold	3.6–8
3-MSS-030-26-4	6	Turbine	55'-9"	CB	N/A - Terminal End	3.6–8
3-MSS-030-68-4	7	Turbine	54'-6"	CB	N/A - Terminal End	3.6–8
3-MSS-030-68-4	8	Turbine	54'-6"	CB & LS	(2)	3.6–8
3-MSS-030-68-4	9	Turbine	54'-6"	CB	N/A - Terminal End	3.6–8
28" G.E. Piping	10	Turbine	40'-0"	CB	(2)	3.6–8
28" G.E. Piping	11	Turbine	36'-0"	CB & LS	(2)	3.6–8
28" G.E. Piping	12	Turbine	32'-6"	CB & LS	(2)	3.6–8
28" G.E. Piping	13	Turbine	31'-10"	CB & LS	(2)	3.6–8
28" G.E. Piping	14	Turbine	35'-4"	CB & LS	(2)	3.6–8
28" G.E. Piping	15	Turbine	63'-9"	CB	(2)	3.6–8
3-MSS-030-94-2	1	Containment	89'-8"	CB	N/A - Terminal End	3.6–8
3-MSS-030-94-2	2	Containment	93'-5"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-94-2	3	Containment	66'-3"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-106-3	18	MSVB	66'-3"	CB	N/A - Terminal End	3.6–8

TABLE 3.6–6 POSTULATED BREAKS MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure
3-MSS-030-59-4	4	Turbine	66'-3"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-59-4	5	Turbine	57'-0"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-59-4	6	Turbine	55'-7"	CB	N/A - Terminal End	3.6–8
3-MSS-030-69-4	7	Turbine	54'-6"	CB	N/A - Terminal End	3.6–8
3-MSS-030-69-4	8	Turbine	54'-6"	CB & LS	(2)	3.6–8
3-MSS-030-69-4	9	Turbine	54'-6"	CB	N/A - Terminal End	3.6–8
28" G.E. Piping	10	Turbine	40'-0"	CB	(2)	3.6–8
28" G.E. Piping	11	Turbine	36'-0"	CB & LS	(2)	3.6–8
28" G.E. Piping	12	Turbine	32'-6"	CB & LS	(2)	3.6–8
28"-G.E. Piping	13	Turbine	31'-10"	CB & LS	(2)	3.6–8
28" G.E. Piping	14	Turbine	35'-4"	CB & LS	(2)	3.6–8
28" G.E. Piping	15	Turbine	63'-9"	CB	(2)	3.6–8
3-MSS-030-95-2	1	Containment	89'-9"	CB	N/A - Terminal End	3.6–8
3-MSS-030-95-2	2	Containment	74'-4"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-95-2	3	Containment	66'-3"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-107-3	19	MSVB	66'-3"	CB	N/A - Terminal End	3.6–8
3-MSS-030-60-4	4	Turbine	65'-8"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-030-60-4	5	Turbine	56'-8"	CB & LS	Above Threshold	3.6–8
3-MSS-030-60-4	6	Turbine	55'-6"	CB	N/A - Terminal End	3.6–8

TABLE 3.6–6 POSTULATED BREAKS MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure
3-MSS-030-70-4	7	Turbine	54'-6"	CB	N/A - Terminal End	3.6–8
3-MSS-030-70-4	8	Turbine	54'-6"	CB & LS	(2)	3.6–8
3-MSS-030-70-4	9	Turbine	54'-6"	CB	N/A - Terminal End	3.6–8
28" G.E. Piping	10	Turbine	40'-0"	CB	(2)	3.6–8
28" G.E. Piping	11	Turbine	36'-0"	CB & LS	(2)	3.6–8
28" G.E. Piping	12	Turbine	32'-6"	CB & LS	(2)	3.6–8
28" G.E. Piping	13	Turbine	31'-10"	CB & LS	(2)	3.6–8
28" G.E. Piping	14	Turbine	35'-4"	CB & LS	(2)	3.6–8
28" G.E. Piping	15	Turbine	72'-11"	CB & LS	(2)	3.6–8
28" G.E. Piping	16	Turbine	76'-5"	CB	(2)	3.6–8
3-MSS-003-6-2	1	Containment	93'-6"	CB	N/A - Terminal End	3.6–8
3-MSS-003-6-2	2	Containment	86'-5"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-003-6-2	3	Containment	80'-5"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-003-6-2	4	Containment	31'-0"	CB	N/A - Terminal End	3.6–8
3-MSS-003-4-2	1	Containment	66'-3"	CB	N/A - Terminal End	3.6–8
3-MSS-003-4-2	2	Containment	56'-7"	CB	Above Threshold	3.6–8
3-MSS-003-4-2	3	Containment	43'-4"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-003-4-2	4	Containment	31'-0"	CB	N/A - Terminal End	3.6–8
3-MSS-003-33-2	1	Containment	93'-6"	CB	N/A - Terminal End	3.6–8

TABLE 3.6–6 POSTULATED BREAKS MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Total Additive Stress	Figure
3-MSS-003-33-2	2	Containment	80'-3"	CB	Above Threshold	3.6–8
3-MSS-003-33-2	3	Containment	71'-4"	CB	Above Threshold	3.6–8
3-MSS-003-33-2	4	Containment	62'-4"	CB	Above Threshold	3.6–8
3-MSS-003-33-2	5	Containment	42'-0"	CB	N/A - Arbitrary Intermediate	3.6–8
3-MSS-003-33-2	6	Containment	31'0"	CB	N/A - Terminal End	3.6–8

NOTES:

(1) Circumferential Pipe Break (CB) and Longitudinal Pipe Split (LS) are defined in FSAR Section 3.6.2.1.3.

(2) Pipe breaks on the Main Steam System downstream of the Main Steam Manifold, 3-MSS-042, are postulated at terminal ends, fittings, valves, and any integral welded attachments. This criteria is also applicable to the Main Steam Turbine Bypass piping and the Main Steam Moisture Separator piping shown on FSAR Figure 3.6–9.

TABLE 3.6–7 PIPE WHIP EFFECTS MAIN STEAM SYSTEM

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement
3-MSS-030-92-2	1	3-MSS-042-63-4	Polar Crane 3MHR*CRN1 Support System	3MSS-PRR7LA
3-MSS-030-92-2	2	3RCS*SG1A	None (2)	3MSS-PRR5LA
3-MSS-030-92-2	2	3-MSS-042-63-4	None (2)	3MSS-PRR2LA
3-MSS-030-92-2	3	3RCS*SG1A	None (2)	3MSS-PRR3LA
3-MSS-030-92-2	3	3-MSS-042-63-4	None (2)	3MSS-PRR1LA
3-MSS-030-104-3	17	3RCS*SG1A	None	None
3-MSS-030-104-3	17	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR4A
3-MSS-030-25-4	4	3RCS*SG1A	None (2)	3MSS-PRR2A & 3MSS-PRR3A
3-MSS-030-25-4	4	3-MSS-042-63-4	None (2)	3MSS-PRR4A & 3MSS-PRR5A
3-MSS-030-25-4	5	3RCS*SG1A	Limit Stress to Break Exclusion Zone	3MSS-PRR2A & 3MSS-PRR3A
3-MSS-030-25-4	5	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR4A & 3MSS-PRR5A
3-MSS-030-25-4	Split	(3)	Limit Stress to Break Exclusion Zone	3MSS-PRR2A & 3MSS-PRR3A
3-MSS-030-25-4	6	3RCS*SG1A	Control Building Boundary Wall	3MSS-PRR5A
3-MSS-030-67-4	7	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR7A & 3MSS-PRR13A
3-MSS-030-67-4	8	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR7A & 3MSS-PRR13A
3-MSS-030-67-4	9	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR13A
3-MSS-030-67-4	10	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR13A
3-MSS-030-67-4	Splits	3-MSS-042-63-4	None	None

TABLE 3.6–7 PIPE WHIP EFFECTS MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement
28" G.E. Piping	11	3-MSS-042-63-4	None	None
28" G.E. Piping	12	3-MSS-042-63-4	None	None
28" G.E. Piping	13	3-MSS-042-63-4	None	None
28" G.E. Piping	14	3-MSS-042-63-4	None	None
28" G.E. Piping	15	3-MSS-042-63-4	None	None
28" G.E. Piping	16	3-MSS-042-63-4	None	None
28" G.E. Piping	Splits	3-MSS-042-63-4	None	None
3-MSS-030-93-2	1	3-MSS-042-63-4	Polar Crane 3MHR*CRN1 Support System	3MSS-PRR7SB
3-MSS-030-93-2	2	3RCS*SG1B	None - (2)	None
3-MSS-030-93-2	2	3-MSS-042-63-4	None - (2)	3MSS-PRR6SB
3-MSS-030-93-2	3	3RCS*SG1B	None - (2)	3MSS-PRR5SB
3-MSS-030-93-2	3	3-MSS-042-63-4	None - (2)	None
3-MSS-030-105-3	16	3RCS*SG1B	None	None
3-MSS-030-105-3	16	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR4B
3-MSS-030-26-4	4	3RCS*SG1B	Limit Stress to Break Exclusion Zone	3MSS-PRR2B & 3MSS-PRR3B
3-MSS-030-26-4	4	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR4B
3-MSS-030-26-4	Split	(3)	Limit Stress to Break Exclusion Zone	3MSS-PRR2B & 3MSS-PRR3B
3-MSS-030-26-4	5	3RCS*SG1B	Limit Stress to Break Exclusion Zone	3MSS-PRR2B & 3MSS-PRR3B

TABLE 3.6–7 PIPE WHIP EFFECTS MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement
3-MSS-030-26-4	5	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR5B
3-MSS-030-26-4	5	(3)	Limit Stress to Break Exclusion Zone	3MSS-PRR2B & 3MSS-PRR3B
3-MSS-030-26-4	6	3RCS*SG1B	Control Building Boundary Wall	3MSS-PRR5B
3-MSS-030-68-4	7	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR7B & 3MSS-PRR13B
3-MSS-030-68-4	8	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR7B & 3MSS-PRR13B
3-MSS-030-68-4	9	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR7B & 3MSS-PRR13B
3-MSS-030-68-4	10	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR13B
3-MSS-030-68-4	Splits	3-MSS-042-63-4	None	None
28" G.E. Piping	11	3-MSS-042-63-4	None	None
28" G.E. Piping	12	3-MSS-042-63-4	None	None
28" G.E. Piping	13	3-MSS-042-63-4	None	None
28" G.E. Piping	14	3-MSS-042-63-4	None	None
28" G.E. Piping	15	3-MSS-042-63-4	None	None
28" G.E. Piping	Splits	3-MSS-042-63-4	None	None
3-MSS-030-094-2	1	3-MSS-042-63-4	Polar Crane 3MHR*CRN1 Support System	3MSS-PRR7SC
3-MSS-030-094-2	2	3RCS*SG1C	None (2)	None
3-MSS-030-094-2	2	3-MSS-042-63-4	None (2)	3MSS-PRR6SC

TABLE 3.6–7 PIPE WHIP EFFECTS MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # [1]	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement
3-MSS-030-094-2	3	3RCS*SG1C	None (2)	3MSS-PRR5SC
3-MSS-030-094-2	3	3-MSS-042-63-4	None (2)	None
3-MSS-030-106-3	18	3RCS*SG1C	None	None
3-MSS-030-106-3	18	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR4C
3-MSS-030-59-4	4	3RCS*SG1C	None (2)	3MSS-PRR1C & 3MSS-PRR3C
3-MSS-030-59-4	4	3-MSS-042-63-4	None (2)	3MSS-PRR4C
3-MSS-030-59-4	5	3RCS*SG1C	None (2)	3MSS-PRR1C & 3MSS-PRR3C
3-MSS-030-59-4	5	3-MSS-042-63-4	None (2)	3MSS-PRR5C
3-MSS-030-59-4	6	3RCS*SG1C	Control Building Boundary Wall	3MSS-PRR5C
3-MSS-030-69-4	7	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR7C & 3MSS-PRR13C
3-MSS-030-69-4	8	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR7C & 3MSS-PRR13C
3-MSS-030-69-4	9	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR7C & 3MSS-PRR13C
3-MSS-030-69-4	10	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR13C
3-MSS-030-69-4	Splits	3-MSS-042-63-4	None	None
28" G.E. Piping	11	3-MSS-042-63-4	None	None
28" G.E. Piping	12	3-MSS-042-63-4	None	None
28" G.E. Piping	13	3-MSS-042-63-4	None	None
28" G.E. Piping	14	3-MSS-042-63-4	None	None
28" G.E. Piping	15	3-MSS-042-63-4	None	None

TABLE 3.6–7 PIPE WHIP EFFECTS MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement
28" G.E. Piping	Splits	3-MSS-042-63-4	None	None
3-MSS-030-95-2	1	3-MSS-042-63-4	Polar Crane 3MHR*CRN1 Support System	3MSS-PRR7LD
3-MSS-030-95-2	2	3RCS*SG1D	None (2)	3MSS-PRR5LD
3-MSS-030-95-2	2	3-MSS-042-63-4	None (2)	3MSS-PRR2LD
3-MSS-030-95-2	3	3RCS*SG1D	None (2)	3MSS-PRR3LD
3-MSS-030-95-2	3	3-MSS-042-63-4	None (2)	3MSS-PRR1LD
3-MSS-030-107-3	19	3RCS*SG1D	None	None
3-MSS-030-107-3	19	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR4D
3-MSS-030-60-4	4	3RCS*SG1D	None (2)	3MSS-PRR1D & 3MSS-PRR3D
3-MSS-030-60-4	4	3-MSS-042-63-4	None (2)	3MSS-PRR4D
3-MSS-030-60-4	5	3RCS*SG1D	Limit Stress to Break Exclusion Zone	3MSS-PRR1D & 3MSS-PRR3D
3-MSS-030-60-4	5	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR5D
3-MSS-030-60-4	Split	(3)	Limit Stress to Break Exclusion Zone	3MSS-PRR1D & 3MSS-PRR3D
3-MSS-030-60-4	6	3RCS*SG1D	Control Building Boundary Wall	3MSS-PRR5D
3-MSS-030-70-4	7	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR7D & 3MSS-PRR13D
3-MSS-030-70-4	8	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR7D & 3MSS-PRR13D
3-MSS-030-70-4	9	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR7D & 3MSS-PRR13D
3-MSS-030-70-4	10	3-MSS-042-63-4	Control Building Boundary Wall	3MSS-PRR13D

TABLE 3.6–7 PIPE WHIP EFFECTS MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement
3-MSS-030-70-4	Splits	3-MSS-042-63-4	None	None
28" G.E. Piping	11	3-MSS-042-63-4	None	None
28" G.E. Piping	12	3-MSS-042-63-4	None	None
28" G.E. Piping	13	3-MSS-042-63-4	None	None
28" G.E. Piping	14	3-MSS-042-63-4	None	None
28" G.E. Piping	15	3-MSS-042-63-4	None	None
28" G.E. Piping	16	3-MSS-042-63-4	None	None
28" G.E. Piping	Splits	3-MSS-042-63-4	None	None
3-MSS-003-6-2	1	None (4)	None	None
3-MSS-003-6-2	2	3-MSS-030-92-2	None (2)	None
3-MSS-003-6-2	2	None (4)	None	None
3-MSS-003-6-2	3	3-MSS-030-92-2	None (2)	None
3-MSS-003-6-2	3	None (4)	None	None
3-MSS-003-6-2	4	3-MSS-030-92-2	None	None
3-MSS-003-6-2	4	None (4)	None	None
3-MSS-003-4-2	1	None (4)	None	None
3-MSS-003-4-2	2	3-MSS-030-93-2	None	None
3-MSS-003-4-2	2	None (4)	None	None
3-MSS-003-4-2	3	3-MSS-030-93-2	None (2)	None

TABLE 3.6–7 PIPE WHIP EFFECTS MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # [1]	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement
3-MSS-003-4-2	3	None (4)	None	None
3-MSS-003-4-2	4	3-MSS-030-93-2	None	None
3-MSS-003-4-2	4	None (4)	None	None
3-MSS-003-33-2	1	None (4)	None	None
3-MSS-003-33-2	2	3-MSS-030-95-2	None	None
3-MSS-003-33-2	2	None (4)	None	None
3-MSS-003-33-2	3	3-MSS-030-95-2	None	None
3-MSS-003-33-2	3	None (4)	None	None
3-MSS-003-33-2	4	3-MSS-030-95-2	None	None
3-MSS-003-33-2	4	None (4)	None	None
3-MSS-003-33-2	5	3-MSS-030-95-2	None (2)	None
3-MSS-003-33-2	5	None (4)	None	None
3-MSS-003-33-2	6	3-MSS-030-95-2	None	None
3-MSS-003-33-2	6	None (4)	None	None

TABLE 3.6–7 PIPE WHIP EFFECTS MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement

NOTES:

(1) Repetition of Break Numbers is used to identify separate fluid reservoirs which provide a constant pressure source to maintain system pressure subsequent to the postulated pipe break. These reservoirs maintain system blowdown during the transient event as well as in the steady state.

(2) USNRC Generic Letter 87-11 eliminates the need to evaluate the pipe whip effects subsequent to Arbitrary Intermediate Breaks (AIB's). Therefore pipe whip effects from AIB's are not evaluated.

(3) The Longitudinal Split 5 on 3-MSS-030-25-4 results in blowdown from both the Steam Generator 3RCS*SG1A and the Main Steam Header 3-MSS-042-63-4. The Longitudinal Split 5 on 3-MSS-030-25-4 does not result in pipe severance rather the split results in motion in and out of the plane of the pipe. This effect loads the pipe break exclusion zone but the load is much less severe than the circumferential break.

(4) The 3" Main Steam lines are used to run the Turbine Driven Auxiliary Feedwater Pump, 3FWA*P2, in the Engineered Safety Features Building. The limited capacity of the Turbine Driven Auxiliary Feedwater Pump does not sustain pipe whip and is not a contained fluid energy reservoir in accordance with FSAR Section 3.6.2.2.1, Item 3.

TABLE 3.6–8 JET IMPINGEMENT EFFECTS - MAIN STEAM SYSTEM

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-MSS-030-92-2	1	None	N/A	N/A	N/A	None
3-MSS-030-92-2	2	Crane Wall	4 Feet	340 psi	1033 Kips	None (2)
3-MSS-030-92-2	2	3RCS*SG1C	13 Feet	65 psi	564 Kips	None (2)
3-MSS-030-92-2	3	None	N/A	N/A	N/A	None (3)
3-MSS-030-92-2	3	None	N/A	N/A	N/A	None (3)
3-MSS-030-104-3	17	None	N/A	N/A	N/A	None
3-MSS-030-104-3	17	None	N/A	N/A	N/A	None
3-MSS-030-25-4	4	None	N/A	N/A	N/A	None
3-MSS-030-25-4	4	None	N/A	N/A	N/A	None
3-MSS-030-25-4	5	None	N/A	N/A	N/A	None (4)
3-MSS-030-25-4	5	None	N/A	N/A	N/A	None (4)
3-MSS-030-25-4	Split	None	N/A	N/A	N/A	None (4)
3-MSS-030-25-4	6	Control Building		(5)		None (6)
3-MSS-030-67-4	7	Control Building		(5)		None (6)
3-MSS-030-67-4	8	Control Building		(5)		None (6)
3-MSS-030-67-4	Split	None	N/A	N/A	N/A	None (4)
3-MSS-030-67-4	9	None	N/A	N/A	N/A	None (4)

TABLE 3.6–8 JET IMPINGEMENT EFFECTS - MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # [1]	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
28" G.E. Piping	10	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	11	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	12	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	13	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	14	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	15	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	16	None	N/A	N/A	N/A	None (4)
3-MSS-030-93-2	1	None	N/A	N/A	N/A	None
3-MSS-030-93-2	1	None	N/A	N/A	N/A	None
3-MSS-030-93-2	2 & 3	None	N/A	N/A	N/A	None (3)
3-MSS-030-105-3	16	None	N/A	N/A	N/A	None
3-MSS-030-105-3	16	None	N/A	N/A	N/A	None

TABLE 3.6–8 JET IMPINGEMENT EFFECTS - MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # [1]	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-MSS-030-26-4	4	None	N/A	N/A	N/A	None (4)
3-MSS-030-26-4	4	None	N/A	N/A	N/A	None (4)
3-MSS-030-26-4	Split	None	N/A	N/A	N/A	None (4)
3-MSS-030-26-4	5	None	N/A	N/A	N/A	None (4)
3-MSS-030-26-4	5	None	N/A	N/A	N/A	None (4)
3-MSS-030-26-4	Split	None	N/A	N/A	N/A	None (4)
3-MSS-030-26-4	6	Control Building		(5)		None (6)
3-MSS-030-68-4	7	Control Building		(5)		None (6)
3-MSS-030-68-4	8	Control Building		(5)		None (6)
3-MSS-030-68-4	Split	None	N/A	N/A	N/A	None (4)
3-MSS-030-68-4	9	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	10	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	11	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	12	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	13	None	N/A	N/A	N/A	None (4)

TABLE 3.6–8 JET IMPINGEMENT EFFECTS - MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	14	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	15	None	N/A	N/A	N/A	None (4)
3-MSS-030-94-2	1	None	N/A	N/A	N/A	None
3-MSS-030-94-2	2 & 3	None	N/A	N/A	N/A	None (3)
3-MSS-030-106-3	18	None	N/A	N/A	N/A	None
3-MSS-030-106-3	18	None	N/A	N/A	N/A	None
3-MSS-030-59-4	4	None	N/A	N/A	N/A	None
3-MSS-030-59-4	4	None	N/A	N/A	N/A	None
3-MSS-030-59-4	5	None	N/A	N/A	N/A	None
3-MSS-030-59-4	5	None	N/A	N/A	N/A	None
3-MSS-030-59-4	6	Control Building		(5)		None (6)
3-MSS-030-69-4	7	Control Building		(5)		None (6)
3-MSS-030-69-4	8	Control Building		(5)		None (6)
3-MSS-030-69-4	Split	None	N/A	N/A	N/A	None (4)
3-MSS-030-69-4	9	Control Building		(5)		None (6)

TABLE 3.6–8 JET IMPINGEMENT EFFECTS - MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
28" G.E. Piping	10	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	11	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	12	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	13	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	14	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	15	None	N/A	N/A	N/A	None (4)
3-MSS-030-95-2	1	None	N/A	N/A	N/A	None
3-MSS-030-95-2	1	None	N/A	N/A	N/A	None
3-MSS-030-95-2	2	Crane Wall	4 Feet	340 psi	1033 Kips	None (2)
3-MSS-030-95-2	2	3RCS*SG1C	13 Feet	65 psi	564 Kips	None (2)
3-MSS-030-95-2	3	None	N/A	N/A	N/A	None (3)
3-MSS-030-107-3	19	None	N/A	N/A	N/A	None
3-MSS-030-107-3	19	None	N/A	N/A	N/A	None

TABLE 3.6–8 JET IMPINGEMENT EFFECTS - MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-MSS-030-60-4	4	None	N/A	N/A	N/A	None (4)
3-MSS-030-60-4	4	None	N/A	N/A	N/A	None (4)
3-MSS-030-60-4	5	None	N/A	N/A	N/A	None (4)
3-MSS-030-60-4	5	None	N/A	N/A	N/A	None (4)
3-MSS-030-60-4	Split	None	N/A	N/A	N/A	None (4)
3-MSS-030-60-4	6	Control Building	10 Feet	86 psi	843 Kips	None (6)
3-MSS-030-70-4	7	Control Building	10 Feet	86 psi	843 Kips	None (6)
3-MSS-030-70-4	8	Control Building	10 Feet	86 psi	843 Kips	None (6)
3-MSS-030-70-4	Split	None	N/A	N/A	N/A	None (4)
3-MSS-030-70-4	9	Control Building	10 Feet	86 psi	843 Kips	None (6)
28" G.E. Piping	10	Control Building	10 Feet	86 psi	843 Kips	None (6)
28" G.E. Piping	11	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	12	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	13	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)

TABLE 3.6–8 JET IMPINGEMENT EFFECTS - MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
28" G.E. Piping	14	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	15	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	Split	None	N/A	N/A	N/A	None (4)
28" G.E. Piping	16	None	N/A	N/A	N/A	None (4)
3-MSS-003-6-2	1	Crane Wall		(7)		None
3-MSS-003-6-2	2 & 3	Crane Wall		(7)		None
3-MSS-003-6-2	4	Crane Wall		(7)		None
3-MSS-003-6-2	4	Crane Wall		(7)		None
3-MSS-003-4-2	1	Crane Wall		(7)		None
3-MSS-003-4-2	2 & 3	None	N/A	N/A	N/A	None
3-MSS-003-4-2	4	None	N/A	N/A	N/A	None
3-MSS-003-4-2	4	None	N/A	N/A	N/A	None
3-MSS-003-33-2	1	Crane Wall		(7)		None
3-MSS-003-33-2	2	None	N/A	N/A	N/A	None
3-MSS-003-33-2	3	None	N/A	N/A	N/A	None
3-MSS-003-33-2	4	None	N/A	N/A	N/A	None

TABLE 3.6–8 JET IMPINGEMENT EFFECTS - MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-MSS-003-33-2	5	None	N/A	N/A	N/A	None
3-MSS-003-33-2	6	None	N/A	N/A	N/A	None

NOTES:

(1) Repetition of Break Numbers is used to identify fluid reservoirs which maintain system pressure subsequent of the postulated pipe break. The reservoirs sustain blowdown during the transient event and in the steady state.

(2) Intermediate Break Number 2 is postulated at a location proximate to an Integral Welded Attachment (IWA) on Main Steam Lines 3-MSS-030-92-2 and 3-MSS-030-95-2. The Crane Wall remains structurally integral since this barrier is designed for larger loads over a smaller area at the pipe rupture restraint locations. The Steam Generation retain pressure boundary integrity and do not catastrophically fail as the upper support system is designed for much larger loads.

(3) USNRC Generic Letter 87-11 eliminates the need to evaluate jet impingement effects subsequent to Arbitrary Intermediate Breaks (AIB's). Therefore, jet impingement effects on the Main Steam System are not evaluated for AIB's.

(4) No essential systems are located in the Turbine Building. Jet impingement on an adjacent Main Steam pipe does not cause pressure boundary failure for the reasons provided in FSAR Section 3.6.2.1.

(5) Fluid jet impingement subsequent to this postulated break on this target is enveloped by Break Number 6 on 3-MSS-030-60-4 and Circumferential Breaks (CB) 7,8,9, and 10 on 3-MSS-030-70-4. The indicated break is partially shielded by adjacent Main Steam lines and is further from the Control Building than either 3-MSS-030-60-4 or 3-MSS-030-70-4.

(6) Fluid jet impingement on the Control Building Concrete Wall is not excessive based upon Generic Structural Evaluation of fluid jet impingement on concrete barriers described in NERM-069 Revision 1 Hazards Review Program Summary for Millstone Unit 3 Section 4.4.

(7) Fluid jet impingement subsequent to this postulated break on this target is enveloped by postulated break number 2 on either of the Main Steam System lines 3-MSS-030-92-2 or 3-MSS-030-95-2 loading the same target.

TABLE 3.6–9 POSTULATED BREAKS MAIN FEEDWATER SYSTEM

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure
3-FWS-020-18-2	1	Containment	64'-7"	CB	N/A - Terminal End	3.6–10
3-FWS-020-18-2	2	Containment	64'-7"	CB & LS	Above Threshold	3.6–10
3-FWS-020-18-2	3	Containment	61'-0"	CB	N/A - Arbitrary Intermediate	3.6–10
3-FWS-018-16-3	9	MSVB	44'-3"	CB	N/A - Terminal End	3.6–10
3-FWS-020-12-4	4 [2]	Turbine	51'-0"	CB	N/A - Arbitrary Intermediate	3.6–10
3-FWS-020-12-4	5 [2]	Turbine	51'-0"	CB	N/A - Arbitrary Intermediate	3.6–10
3-FWS-016-74-2	1	Containment	64'-7"	CB	N/A - Terminal End	3.6–10
3-FWS-016-74-2	2	Containment	64'-7"	CB & LS	Above Threshold	3.6–10
3-FWS-016-74-2	3	Containment	64'-7"	CB & LS	Above Threshold	3.6–10
3-FWS-018-20-3	9	MSVB	44'-3"	CB	N/A - Terminal End	3.6–10
3-FWS-016-73-2	1	Containment	64'-7"	CB	N/A - Terminal End	3.6–10
3-FWS-016-73-2	2	Containment	64'-7"	CB & LS	Above Threshold	3.6–10
3-FWS-016-73-2	3	Containment	64'-7"	CB & LS	Above Threshold	3.6–10
3-FWS-018-24-3	9	MSVB	44'-3"	CB	N/A - Terminal End	3.6–10
3-FWS-020-30-2	1	Containment	64'-7"	CB	N/A - Terminal End	3.6–10
3-FWS-020-30-2	2	Containment	64'-7"	CB & LS	Above Threshold	3.6–10
3-FWS-020-30-2	3	Containment	60'-10"	CB	N/A - Arbitrary Intermediate	3.6–10
3-FWS-018-28-3	9	MSVB	44'-3"	CB	N/A - Terminal End	3.6–10
3-FWS-020-15-4	6 [2]	Turbine	51'-0"	CB	N/A - Arbitrary Intermediate	3.6–10
3-FWS-020-15-4	7 [2]	Turbine	51'-0"	CB	N/A - Arbitrary Intermediate	3.6–10
3-FWS-036-11-4	8 [2]	Turbine	51'-0"	CB	N/A Terminal End	3.6–10

TABLE 3.6–9 POSTULATED BREAKS MAIN FEEDWATER SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure

NOTES:

(1) Circumferential Pipe Break (CB) and Longitudinal Pipe Split (LS) are defined in FSAR Section 3.6.2.1.3.

(2) The Main Feedwater piping in the Turbine Building is nonnuclear but is analyzed as part of the Main Feedwater piping in the Main Steam Valve Building (MSVB). The Main Feedwater piping in the Main Steam Valve Building is ASME Class 2 and 3. The criteria specified in FSAR Section 3.6.2.1.2.3.2.b is applied to the Main Feedwater piping in the Turbine Building. The approach results in fewer pipe break locations than if a fitting criteria is applied.

TABLE 3.6–10 PIPE WHIP EFFECTS - MAIN FEEDWATER SYSTEM

Line Designation	Break# (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-FWS-020-18-2	1	3-FWS-036-11-4	Steam Generator Cubicle Wall	3FWS-PRR12LA & 3FWS-PRR13LA
3-FWS-020-18-2	2	3RCS*SG1A	None	None
3-FWS-020-18-2	2	3-FWS-036-11-4	Operating Floor at Elevation 51'-4"	3FWS-PRR5LA (2)
3-FWS-020-18-2	Split	3RCS*SG1A	None	None
3-FWS-020-18-2	3	3RCS*SG1A	None (3)	3FWS-PRR5LA
3-FWS-020-18-2	3	3-FWS-036-11-4	None (3)	3FWS-PRR3LA
3-FWS-018-16-3	9	3RCS*SG1A	None	None
3-FWS-018-16-3	9	3-FWS-036-11-4	None	None
3-FWS-020-12-4	4	3RCS*SG1A	None (3)	None
3-FWS-020-12-4	4	3-FWS-036-11-4	None (3)	None
3-FWS-020-12-4	5	3RCS*SG1A	None (3)	None
3-FWS-020-12-4	5	3-FWS-036-11-4	None (3)	None
3FWS-016-74-2	1	3-FWS-036-11-4	Steam Generator Cubicle Wall	3FWS-PRR8SB
3-FWS-016-74-2	2	3RCS*SG1B	3RCS*SG1B	3FWS-PRR5SB
3-FWS-016-74-2	2	3-FWS-036-11-4	None	3FWS-PRR8SB
3-FWS-016-74-2	Split	3RCS*SG1B	None	None

TABLE 3.6–10 PIPE WHIP EFFECTS - MAIN FEEDWATER SYSTEM (CONTINUED)

Line Designation	Break# (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-FWS-016-74-2	3	3-FWS-036-11-4	None	3FWS-PRR8SB
3-FWS-016-74-2	3	3RCS*SG1B	None	3-FWS-PRR5SB
3-FWS-016-74-2	Split	3RCS*SG1B	None	None
3-FWS-018-20-3	9	3RCS*SG1B	None	None
3-FWS-018-20-3	9	3-FWS-036-11-4	None	None
3-FWS-016-73-2	1	3-FWS-036-11-4	Steam Generator Cubicle Wall	3FWS-PRR8SC
3-FWS-016-73-2	2	3RCS*SG1C	3RCS*SG1C	3FWS-PRR5SC
3-FWS-016-73-2	2	3-FWS-036-11-4	None	3FWS-PRR8SC
3-FWS-016-73-2	Split	3RCS*SG1C	None	None
3-FWS-016-73-2	3	3-FWS-036-11-4	None	3FWS-PRR8SC
3-FWS-016-73-2	3	3RCS*SG1C	None	3FWS-PRR5SC
3-FWS-016-73-2	Split	3RCS*SG1C	None	None
3-FWS-018-24-3	9	3RCS*SG1C	None	None
3-FWS-018-24-3	9	3-FWS-036-11-4	None	None
3-FWS-020-30-2	1	3-FWS-036-11-4	Steam Generator Cubicle Wall	3FWS-PRR12LD & 3FWS-PRR13LD
3-FWS-020-30-2	2	3RCS*SG1D	None	None
3-FWS-020-30-2	2	3-FWS-036-11-4	Operating Floor at Elevation 51'-4"	3FWS-PRR5LD (4)
3-FWS-020-30-2	Split	3RCS*SG1D	None	None

TABLE 3.6–10 PIPE WHIP EFFECTS - MAIN FEEDWATER SYSTEM (CONTINUED)

Line Designation	Break# (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-FWS-020-30-2	3	3-FWS-036-11-4	None (3)	None
3-FWS-020-30-2	3	3RCS*SG1D	None (3)	None
3-FWS-018-28-3	9	3RCS*SG1D	None	None
3-FWS-018-28-3	9	3-FWS-036-11-4	None	None
3-FWS-020-15-4	6	3-FWS-036-11-4	None (3)	None
3-FWS-020-15-4	6	3RCS*SG1D	None (3)	None
3-FWS-020-15-4	7	3-FWS-036-11-4	None (3)	None
3-FWS-020-15-4	7	3RCS*SG1D	None (3)	None
3-FWS-036-11-4	8	3RCS*SG1A & 1B	None	None
3-FWS-036-11-4	8	3RCS*SG1C & 1D	None	None

TABLE 3.6–10 PIPE WHIP EFFECTS - MAIN FEEDWATER SYSTEM (CONTINUED)

Line Designation	Break# [1]	Blowdown Source	Essential Pipe Whip Target	Protection Requirement

NOTES:

(1) Repetition of Break Numbers is used to identify separate fluid reservoirs which provide a constant pressure source to maintain system pressure subsequent to the postulated pipe break. These reservoirs maintain system blowdown during the transient event as well as in the steady state.

(2) The Main Feedwater piping in the Turbine Building is nonnuclear but is analyzed as part of the Main Feedwater piping in the Main Steam Valve Building (MSVB). The Main Feedwater piping in the Main Steam Valve Building is ASME Class 2 and 3. The criteria specified in FSAR Section 3.6.2.1.2.3.2.b is applied to the Main Feedwater piping in the Turbine Building. The approach results in fewer pipe break locations than if a fitting criteria is applied.

(3) USNRC Generic Letter 87-11 eliminates the requirement to evaluate pipe whip effects subsequent to Arbitrary Intermediate Breaks (AIB's). Therefore, pipe whip effects subsequent to AIB's on the Main Feedwater System are not evaluated.

(4) The Intermediate Break Number 2 is postulated to occur at any location along the reducing elbow. A Circumferential Break (CB) and a Longitudinal Split (LS) are postulated but not concurrently. The Circumferential Break (CB) results in an impact of the Main Feedwater Line 3-FWS-020-30-2 to the Operating Floor Slab at Elevation 51'-4". Unrestrained impact would result in catastrophic failure of the Operating Floor and result in secondary missile ejection into the Reactor Coolant System cubicles. This unrestrained impact to the Operating Floor is prevented by the rupture restraint.

TABLE 3.6–11 JET IMPINGEMENT EFFECTS - MAIN STEAM SYSTEM

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-FWS-020-18-2	1	None	N/A	N/A	N/A	None
3-FWS-020-18-2	2	None	N/A	N/A	N/A	None
3-FWS-020-18-2	2	None	N/A	N/A	N/A	None
3-FWS-020-18-2	Split	None	N/A	N/A	N/A	None
3-FWS-020-18-2	3	None (2)	N/A	N/A	N/A	None
3-FWS-020-18-2	3	None (2)	N/A	N/A	N/A	None
3-FWS-018-16-3	9	None	N/A	N/A	N/A	None
3-FWS-018-16-3	9	None	N/A	N/A	N/A	None
3-FWS-020-12-4	4	None (2)	N/A	N/A	N/A	None
3-FWS-020-12-4	4	None (2)	N/A	N/A	N/A	None
3-FWS-020-12-4	5	None (2)	N/A	N/A	N/A	None
3-FWS-020-12-4	5	None (2)	N/A	N/A	N/A	None
3-FWS-016-74-2	1	None	N/A	N/A	N/A	None
3-FWS-016-74-2	2	None	N/A	N/A	N/A	None
3-FWS-016-74-2	2	None	N/A	N/A	N/A	None
3-FWS-016-74-2	Split	None	N/A	N/A	N/A	None
3-FWS-016-74-2	3	None	N/A	N/A	N/A	None
3-FWS-016-74-2	3	None	N/A	N/A	N/A	None
3-FWS-016-74-2	Split	None	N/A	N/A	N/A	None

TABLE 3.6–11 JET IMPINGEMENT EFFECTS - MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-FWS-018-20-3	9	None	N/A	N/A	N/A	None
3-FWS-018-20-3	9	None	N/A	N/A	N/A	None
3-FWS-016-73-2	1	None	N/A	N/A	N/A	None
3-FWS-016-73-2	2	None	N/A	N/A	N/A	None
3-FWS-016-73-2	2	None	N/A	N/A	N/A	None
3-FWS-016-73-2	Split	None	N/A	N/A	N/A	None
3-FWS-016-73-2	3	None	N/A	N/A	N/A	None
3-FWS-016-73-2	3	None	N/A	N/A	N/A	None
3-FWS-016-73-2	Split	None	N/A	N/A	N/A	None
3-FWS-018-24-3	9	None	N/A	N/A	N/A	None
3-FWS-018-24-3	9	None	N/A	N/A	N/A	None
3-FWS-020-30-2	1	None	N/A	N/A	N/A	None
3-FWS-020-30-2	2	None	N/A	N/A	N/A	None
3-FWS-020-30-2	2	None	N/A	N/A	N/A	None
3-FWS-020-30-2	Split	None	N/A	N/A	N/A	None
3-FWS-020-30-2	3	None (2)	N/A	N/A	N/A	None
3-FWS-020-30-2	3	None (2)	N/A	N/A	N/A	None
3-FWS-018-28-3	9	None	N/A	N/A	N/A	None
3-FWS-018-28-3	9	None	N/A	N/A	N/A	None

TABLE 3.6–11 JET IMPINGEMENT EFFECTS - MAIN STEAM SYSTEM (CONTINUED)

Line Designation	Break # [1]	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-FWS-020-15-4	6	None [2]	N/A	N/A	N/A	None
3-FWS-020-15-4	6	None [2]	N/A	N/A	N/A	None
3-FWS-020-15-4	7	None [2]	N/A	N/A	N/A	None
3-FWS-020-15-4	7	None [2]	N/A	N/A	N/A	None
3-FWS-036-11-4	8	None	N/A	N/A	N/A	None
3-FWS-036-11-4	8	None	N/A	N/A	N/A	None

NOTES:

(1) Repetition of Break Numbers is used to identify separate fluid reservoirs which provide a constant pressure source to maintain system pressure subsequent to the postulated pipe break. These reservoirs maintain system blowdown during the transient event as well as in the steady state.

(2) USNRC Generic Letter 87-11 eliminates the requirement to evaluate the jet impingement effects subsequent to Arbitrary Intermediate Break (AIB's). Therefore jet impingement effects subsequent to AIB's are not evaluated.

TABLE 3.6–12 POSTULATED BREAKS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING. LETDOWN LINE, AND NORMAL CHARGING

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress Eqn 10	Threshold Stress Eqn 12 Or 13	Usage Factor	Figure
3-RCS-008-24-1	13	Containment	23'-0"	CB & LS		Above Threshold		3.6–12
3-RCS-008-25-1	14	Containment	31'-6"	CB & LS		Above Threshold		3.6–12
3-RCS-150-27-1	13	Containment	23'-0"	CB		Terminal End		3.6–12
3-RCS-150-27-1	15	Containment	23'-0"	CB		N/A - Arbitrary Intermediate		3.6–12
3-RCS-150-27-1	16	Containment	22'-2"	CB		N/A - Arbitrary Intermediate		3.6–12
3-RCS-150-27-1	17	Containment	17'-6"	CB		Terminal End		3.6–12
3-RCS-008-29-1	13	Containment	23'-0"	CB & LS		Above Threshold		3.6–12
3-RCS-008-30-1	14	Containment	31'-6"	CB & LS		Above Threshold		3.6–12
3-RCS-150-32-1	13	Containment	23'-0"	CB		Terminal End		3.6–12
3-RCS-150-32-1	15	Containment	23'-0"	CB		N/A - Arbitrary Intermediate		3.6–12
3-RCS-150-32-1	16	Containment	22'-2"	CB		N/A - Arbitrary Intermediate		3.6–12
3-RCS-150-32-1	17	Containment	17'-6"	CB		Terminal End		3.6–12
3-RCS-008-34-1	13	Containment	23'-0"	CB & LS		Above Threshold		3.6–12
3-RCS-008-35-1	14	Containment	31'-6"	CB & LS		Above Threshold		3.6–12
3-RCS-150-37-1	13	Containment	23'-0"	CB		Terminal End		3.6–12
3-RCS-150-37-1	15	Containment	23'-0"	CB		N/A - Arbitrary Intermediate		3.6–12

TABLE 3.6–12 POSTULATED BREAKS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING. LETDOWN LINE, AND NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress		Usage Factor	Figure
					Eqn 10	Eqn 12 Or 13		
3-RCS-150-37-7	16	Containment	22'-2"	CB		N/A - Arbitrary Intermediate		3.6–12
3-RCS-150-37-1	17	Containment	17'-6"	CB		Terminal End		3.6–12
3-RCS-008-39-1	13	Containment	23'-0"	CB & LS		Above Threshold		3.6–12
3-RCS-008-40-1	14	Containment	31'-6"	CB & LS		Above Threshold		3.6–12
3-RCS-150-42-1	13	Containment	23'-0"	CB		Terminal End		3.6–12
3-RCS-150-42-1	15	Containment	23'-0"	CB		N/A - Arbitrary Intermediate		3.6–12
3-RCS-150-42-1	16	Containment	22'-2"	CB		N/A - Arbitrary Intermediate		3.6–12
3-RCS-150-42-1	17	Containment	17'-6"	CB		Terminal End		3.6–12
3-RCS-002-127-1	TP	Containment	5'-5"	CB		N/A (2)		3.6–33
3-RCS-002-127-1	IP	Containment	4'-9"	CB		N/A (2)		3.6–33
3-RCS-002-127-1	TP	Containment	4'-9"	CB		N/A (2)		3.6–33
3-RCS-002-130-1	TP	Containment	5'-5"	CB		N/A (2)		3.6–33
3-RCS-002-130-1	IP	Containment	4'-9"	CB		N/A (2)		3.6–33
3-RCS-002-130-1	TP	Containment	4'-9"	CB		N/A (2)		3.6–33
3-RCS-002-135-1	TP	Containment	5'-5"	CB		N/A (2)		3.6–33

TABLE 3.6–12 POSTULATED BREAKS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING. LETDOWN LINE, AND NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type [1]	Threshold Stress		Usage Factor	Figure
					Eqn 10	Eqn 12 Or 13		
3-RCS-002-135-1	IP	Containment	4'-9"	CB		N/A [2]		3.6–33
3-RCS-002-135-1	TP	Containment	4'-9"	CB		N/A [2]		3.6–33
3-RCS-002-143-1	TP	Containment	5'-5"	CB		N/A [2]		3.6–33
3-RCS-002-143-1	IP	Containment	4'-9"	CB		N/A [2]		3.6–33
3-RCS-002-143-1	TP	Containment	4'-9"	CB		N/A [2]		3.6–33
3-RCS-002-126-1	TP	Containment	15'-10"	CB		N/A [3]		3.6–33
3-RCS-002-126-1	IP	Containment	15'-2"	CB		N/A [3]		3.6–33
3-RCS-002-126-1	TP	Containment	10'-3"	CB		N/A [3]		3.6–33
3-RCS-002-129-1	TP	Containment	15'-10"	CB		N/A [3]		3.6–33
3-RCS-002-129-1	IP	Containment	15'-2"	CB		N/A [3]		3.6–33
3-RCS-002-129-1	TP	Containment	10'-3"	CB		N/A [3]		3.6–33
3-RCS-002-134-1	TP	Containment	15'-10"	CB		N/A [3]		3.6–33
3-RCS-002-134-1	IP	Containment	15'-2"	CB		N/A [3]		3.6–33
3-RCS-002-134-1	TP	Containment	10'-3"	CB		N/A [3]		3.6–33

TABLE 3.6–12 POSTULATED BREAKS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING. LETDOWN LINE, AND NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress		Usage Factor	Figure
					Eqn 10	Eqn 12 Or 13		
3-RCS-002-142-1	TP	Containment	15'-10"	CB		N/A (3)		3.6–33
3-RCS-002-142-1	IP	Containment	15'-2"	CB		N/A (3)		3.6–33
3-RCS-002-142-1	TP	Containment	10'-3"	CB		N/A (3)		3.6–33
3-RCS-002-128-1	TP	Containment	8'-10"	CB		N/A (4)		3.6–33
3-RCS-002-128-1	IP	Containment	11'-7"	CB		N/A (4)		3.6–33
3-RCS-002-128-1	TP	Containment	11'-7"	CB		N/A (4)		3.6–33
3-RCS-002-131-1	TP	Containment	8'-10"	CB		N/A (4)		3.6–33
3-RCS-002-131-1	IP	Containment	11'-7"	CB		N/A (4)		3.6–33
3-RCS-002-131-1	TP	Containment	11'-7"	CB		N/A (4)		3.6–33
3-RCS-002-136-1	TP	Containment	8'-10"	CB		N/A (4)		3.6–33
3-RCS-002-136-1	IP	Containment	11'-7"	CB		N/A (4)		3.6–33
3-RCS-002-136-1	TP	Containment	11'-7"	CB		N/A (4)		3.6–33
3-RCS-002-144-1	TP	Containment	8'-10"	CB		N/A (4)		3.6–33
3-RCS-002-144-1	IP	Containment	11'-7"	CB		N/A (4)		3.6–33

TABLE 3.6–12 POSTULATED BREAKS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING. LETDOWN LINE, AND NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress		Usage Factor	Figure
					Eqn 10	Eqn 12 Or 13		
3-RCS-002-144-1	TP	Containment	11'-7"	CB		N/A (4)		3.6–33
3-RCS-003-137-1	1	Containment	15'-9"	CB		N/A Terminal End		3.6–16
3-RCS-003-137-1	2	Containment	10'-1"	CB		Above Threshold		3.6–16
3-RCS-003-137-1	3	Containment	8'-7"	CB		Above Threshold		3.6–16
3-RCS-003-137-1	4	Containment	7'-0"	CB		Above Threshold		3.6–16
3-RCS-003-137-1	5	Containment	7'-0"	CB		Above Threshold		3.6–16
3-RCS-003-171-1	6	Containment	7'-0"	CB		Above Threshold		3.6–16
3-RCS-003-171-1	7	Containment	7'-0"	CB		Above Threshold		3.6–16
3-RCS-003-171-1	8	Containment	7'-0"	CB		Above Threshold		3.6–16
3-RCS-003-145-1	1	Containment	17'-6"	CB		N/A Terminal End		3.6–15
3-RCS-003-145-1	2	Containment	17'-6"	CB		Above Threshold		3.6–15
3-RCS-003-145-1	3	Containment	17'-6"	CB		Above Threshold		3.6–15
3-RCS-003-145-1	4	Containment	17'-6"	CB		Above Threshold		3.6–15
3-RCS-003-145-1	5	Containment	17'-2"	CB		Above Threshold		3.6–15
3-RCS-003-145-1	6	Containment	16'-7"	CB		Above Threshold		3.6–15
3-RCS-003-145-1	7	Containment	8'-0"	CB		Above Threshold		3.6–15

TABLE 3.6–12 POSTULATED BREAKS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING. LETDOWN LINE, AND NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress		Usage Factor	Figure
					Eqn 10	Eqn 12 Or 13		
3-RCS-003-145-1	8	Containment	8'-0"	CB		Above Threshold		3.6–15
3-RCS-003-145-1	9	Containment	8'-0"	CB		Above Threshold		3.6–15
3-RCS-003-149-1	11	Containment	17'-6"	CB		N/A Terminal End		3.6–15
3-RCS-003-149-1	12	Containment	17'-6"	CB		Above Threshold		3.6–15
3-RCS-003-149-1	13	Containment	17'-6"	CB		Above Threshold		3.6–15
3-RCS-003-149-1	14	Containment	17'-0"	CB		Above Threshold		3.6–15
3-RCS-003-149-1	15	Containment	16'-7"	CB		Above Threshold		3.6–15
3-RCS-003-149-1	16	Containment	16'-2"	CB		Above Threshold		3.6–15
3-RCS-003-149-1	17	Containment	7'-1"	CB		Above Threshold		3.6–15
3-RCS-003-149-1	18	Containment	7'-1"	CB		Above Threshold		3.6–15
3-RCS-003-149-1	19	Containment	7'-1"	CB		Above Threshold		3.6–15

TABLE 3.6–12 POSTULATED BREAKS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING. LETDOWN LINE, AND NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress		Usage Factor	Figure
					Eqn 10	Eqn 12 Or 13		

NOTES:

(1) Circumferential Pipe Break (CB) and Longitudinal Pipe Split (LS) are defined in FSAR Section 3.6.2.1.3.

(2) Breaks on the Reactor Coolant System Excess Letdown Piping off each Crossover Leg (TP) are postulated at any location (IP) up to the normally closed valve, 3RCS*AV8037A (TP). The portion of the Reactor Coolant System Excess Letdown Piping downstream of the normally closed valve, 3RCS*AV8037A is moderate energy based upon the 2% rule. The 2% rule is defined in FSAR Section 3.6.2.1.2 under Moderate Energy Piping Systems.

(3) Breaks on the Reactor Coolant System Drain Lines off each Crossover Leg (TP) are postulated at any location (IP) along the piping up to the normally closed valve, 3RCS*V203 (TP). The portion of the Reactor Coolant System Drain Lines off each Crossover Leg downstream of the normally closed valve, 3RCS*V203 is moderate energy based upon the 2% rule. The 2% rule is defined in FSAR Section 3.6.2.1.2 under Moderate Energy Piping Systems.

(4) Breaks on the Reactor Coolant System Loop Fill Piping to each Crossover Leg (TP) are postulated at any location (IP) along the Reactor Coolant System portion of this line and at the Terminal Ends (TP) at each Crossover Leg connection and at the normally closed valves, 3RCS*AV8036A, 3RCS*AV8036B, 3RCS*AV8036C, and 3RCS*AV8036D. The portion of the Loop Fill Piping upstream of the normally closed valves, 3RCS*AV8036A, 3RCS*AV8036B, 3RCS*AV8036C, and 3RCS*AV8036D is moderate energy under the 2% rule and is part of the Chemical & Volume Control System. The 2% rule is defined in FSAR Section 3.6.2.1.2 under Moderate Energy Piping Systems.

TABLE 3.6–13 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING, LETDOWN AND NORMAL CHARGING

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Requirement
3-RCS-008-24-1	13	3-RCS-275-4-1	None	None
3-RCS-008-24-1	13	3-RCS-029-2-1	(2)	3RCS-PRRBA & 3RCS-PRRFA
3-RCS-008-24-1	Split	Hot & Cold Legs	(2)	3RCS-PRRCA & 3RCS-PRREA
3-RCS-008-25-1	14	3-RCS-275-4-1	(2)	3RCS-PRRA1A & 3RCS-PRRA2A
3-RCS-008-25-1	14	3-RCS-029-2-1	(2)	3RCS-PRR4A
3-RCS-008-25-1	Split	Hot & Cold Legs	(2)	3RCS-PRRCA & 3RCS-PRREA
3-RCS-150-27-1	13	Hot & Cold Legs	None	None
3-RCS-150-27-1	15	Hot & Cold Legs	None	None
3-RCS-150-27-1	16	Hot & Cold Legs	None	None
3-RCS-150-27-1	17	Hot & Cold Legs	None	None
3-RCS-008-29-1	13	3-RCS-275-9-1	None	None
3-RCS-008-29-1	13	3-RCS-029-7-1	(2)	3RCS-PRRBB & 3RCS-PRRFB
3-RCS-008-29-1	Split	Hot & Cold Legs	(2)	3RCS-PRRCB & 3RCS-PRREB
3-RCS-008-30-1	14	3-RCS-275-9-1	(2)	3RCS-PRRA1B & 3RCS-PRRA2B
3-RCS-008-30-1	14	3-RCS-029-7-1	(2)	3RCS-PRR48
3-RCS-008-30-1	Split	Hot & Cold Legs	(2)	3RCS-PRRCB & 3RCS-PRREB
3-RCS-150-32-1	13	Hot & Cold Legs	None	None

TABLE 3.6–13 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING, LETDOWN AND NORMAL CHARGING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Requirement
3-RCS-150-32-1	15	Hot & Cold Legs	None	None
3-RCS-150-32-1	16	Hot & Cold Legs	None	None
3-RCS-150-32-1	17	Hot & Cold Legs	None	None
3-RCS-008-34-1	13	3-RCS-275-14-1	None	None
3-RCS-008-34-1	13	3-RCS-029-12-1	(2)	3-RCS-PRRBC & 3RCS-PRRFC
3-RCS-008-34-1	Split	Hot & Cold Legs	(2)	3RCS-PRRCC & 3RCS-PRREC
3-RCS-008-35-1	14	3-RCS-275-14-1	(2)	3RCS-PRRA1C & 3RCS-PRRA2C
3-RCS-008-35-1	14	3-RCS-029-12-1	(2)	3RCS-PRR4C
3-RCS-008-35-1	Split	Hot & Cold Legs	(2)	3-RCS-PRRCC & 3RCS-PRREC
3-RCS-150-37-1	13	Hot & Cold Legs	None	None
3-RCS-150-37-1	15	Hot & Cold Legs	None	None
3-RCS-150-37-1	16	Hot & Cold Legs	None	None
3-RCS-150-37-1	17	Hot & Cold Legs	None	None
3-RCS-008-39-1	13	3-RCS-275-19-1	None	None
3-RCS-008-39-1	13	3-RCS-029-17-1	(2)	3RCS-PRRBD & 3RCS-PRRFD
3-RCS-008-39-1	Split	Hot & Cold Legs	(2)	3RCS-PRRCD & 3RCS-PRRED
3-RCS-008-40-1	14	3-RCS-275-19-1	(2)	3RCS-PRRA1D & 3RCS-PRRA2D

TABLE 3.6–13 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING, LETDOWN AND NORMAL CHARGING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Requirement
3-RCS-008-40-1	14	3-RCS-029-17-1	(2)	3RCS-PRR4D
3-RCS-008-40-1	Split	Hot & Cold Legs	(2)	3-RCS-PRRCD & 3RCS-PRRED
3-RCS-150-42-1	13	Hot & Cold Legs	None	None
3-RCS-150-42-1	15	Hot & Cold Legs	None	None
3-RCS-150-42-1	16	Hot & Cold Legs	None	None
3-RCS-150-42-1	17	Hot & Cold Legs	None	None
3-RCS-002-127-1	TP	3-RCS-031-3-1	None	None
3-RCS-002-127-1	IP	3-RCS-031-3-1	None	None
3-RCS-002-127-1	TP	3-RCS-031-3-1	None	None
3-RCS-002-130-1	TP	3-RCS-031-8-1	None	None
3-RCS-002-130-1	IP	3-RCS-031-8-1	None	None
3-RCS-002-130-1	TP	3-RCS-031-8-1	None	None
3-RCS-002-135-1	TP	3-RCS-031-13-1	None	None
3-RCS-002-135-1	IP	3-RCS-031-13-1	None	None
3-RCS-002-135-1	TP	3-RCS-031-13-1	None	None
3-RCS-002-143-1	TP	3-RCS-031-18-1	None	None
3-RCS-002-143-1	IP	3-RCS-031-18-1	None	None

TABLE 3.6–13 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING, LETDOWN AND NORMAL CHARGING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Requirement
3-RCS-002-143-1	TP	3-RCS-031-18-1	None	None
3-RCS-002-126-1	TP	3-RCS-029-2-1	None	None
3-RCS-002-126-1	IP	3-RCS-029-2-1	None	None
3-RCS-002-126-1	TP	3-RCS-029-2-1	None	None
3-RCS-002-129-1	TP	3-RCS-029-7-1	None	None
3-RCS-002-129-1	IP	3-RCS-029-7-1	None	None
3-RCS-002-129-1	TP	3-RCS-029-7-1	None	None
3-RCS-002-134-1	TP	3-RCS-029-12-1	None	None
3-RCS-002-134-1	IP	3-RCS-029-12-1	None	None
3-RCS-002-134-1	TP	3-RCS-029-12-1	None	None
3-RCS-002-142-1	TP	3-RCS-029-17-1	None	None
3-RCS-002-142-1	IP	3-RCS-029-17-1	None	None
3-RCS-002-142-1	TP	3-RCS-029-17-1	None	None
3-RCS-002-128-1	TP	3-RCS-031-3-1	None	None
3-RCS-002-128-1	IP	3-RCS-031-3-1	None	None
3-RCS-002-128-1	TP	3-RCS-031-3-1	None	None
3-RCS-002-131-1	TP	3-RCS-031-8-1	None	None

TABLE 3.6–13 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING, LETDOWN AND NORMAL CHARGING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Requirement
3-RCS-002-131-1	IP	3-RCS-031-8-1	None	None
3-RCS-002-131-1	TP	3-RCS-031-8-1	None	None
3-RCS-002-136-1	TP	3-RCS-031-13-1	None	None
3-RCS-002-136-1	IP	3-RCS-031-13-1	None	None
3-RCS-002-136-1	TP	3-RCS-031-13-1	None	None
3-RCS-002-144-1	TP	3-RCS-031-18-1	None	None
3-RCS-002-144-1	IP	3-RCS-031-18-1	None	None
3-RCS-002-144-1	TP	3-RCS-031-18-1	None	None
3-RCS-003-137-1	1	3-RCS-275-15-1	None	None
3-RCS-003-137-1	2	3-RCS-275-15-1	None	None
3-RCS-003-137-1	3	3-RCS-275-15-1	None	None
3-RCS-003-137-1	4	3-RCS-275-15-1	None	None
3-RCS-003-137-1	5	3-RCS-275-15-1	None	None
3-RCS-003-171-1	6	3-RCS-275-15-1	None	None
3-RCS-003-171-1	7	3-RCS-275-15-1	None	None
3-RCS-003-171-1	8	3-RCS-275-15-1	None	None
3-RCS-003-145-1	1	3-RCS-275-20-1	None	None

TABLE 3.6–13 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING, LETDOWN AND NORMAL CHARGING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Requirement
3-RCS-003-145-1	2	3-RCS-275-20-1	None	None
3-RCS-003-145-1	3	3-RCS-275-20-1	None	None
3-RCS-003-145-1	4	3-RCS-275-20-1	None	None
3-RCS-003-145-1	5	3-RCS-275-20-1	None	None
3-RCS-003-145-1	6	3-RCS-275-20-1	None	None
3-RCS-003-145-1	7	3-RCS-275-20-1	None	None
3-RCS-003-145-1	8	3-RCS-275-20-1	None	None
3-RCS-003-145-1	9	3-RCS-275-20-1	None	None
3-RCS-003-149-1	11	3-RCS-275-5-1	None	None
3-RCS-003-149-1	12	3-RCS-275-5-1	None	None
3-RCS-003-149-1	13	3-RCS-275-5-1	None	None
3-RCS-003-149-1	14	3-RCS-275-5-1	None	None
3-RCS-003-149-1	15	3-RCS-275-5-1	None	None
3-RCS-003-149-1	16	3-RCS-275-5-1	None	None
3-RCS-003-149-1	17	3-RCS-275-5-1	None	None
3-RCS-003-149-1	18	3-RCS-275-5-1	None	None
3-RCS-003-149-1	19	3-RCS-275-5-1	None	None

TABLE 3.6–13 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP FILL PIPING, LOOP DRAIN PIPING, LETDOWN AND NORMAL CHARGING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Requirement

NOTES:

(1) Repetition of Break Numbers is used to identify separate fluid reservoirs which provide a constant pressure source to maintain system pressure subsequent to the postulated pipe break. These reservoirs maintain system blowdown during the transient event as well as in the steady state.

(2) A Large Loss of Coolant Accident (LOCA) occurs subsequent to this postulated break. The NSSS System Standard Design Criteria defines the protection requirements subsequent to this break. Compliance with the NSSS System Standard Design Criteria is deemed essential for the effects of this postulated break. The NSSS System Standard Design Criteria requires that leg to leg propagation on the affected loop shall not exceed 20% of the flow area of the broken pipe. The pipe rupture restraints are provided to ensure compliance with this requirement. This requirement although not strictly defined as essential as stated in FSAR Section 3.6.1 is met such as not to increase the severity of the postulated LOCA.

TABLE 3.6–14 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP DRAIN PIPING, LOOP FILL PIPING, LETDOWN LINE, AND NORMAL CHARGING

Line Designation	Break #	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-RCS-008-24-1	13	None	N/A	N/A	N/A	None
3-RCS-008-25-1	14	None	N/A	N/A	N/A	None
3-RCS-150-27-1	13	None	N/A	N/A	N/A	None
3-RCS-150-27-1	15	None	N/A	N/A	N/A	None
3-RCS-150-27-1	16	None	N/A	N/A	N/A	None
3-RCS-150-27-1	17	None	N/A	N/A	N/A	None
3-RCS-008-29-1	13	None	N/A	N/A	N/A	None
3-RCS-008-30-1	14	None	N/A	N/A	N/A	None
3-RCS-150-32-1	13	None	N/A	N/A	N/A	None
3-RCS-150-32-1	15	None	N/A	N/A	N/A	None
3-RCS-150-32-1	16	None	N/A	N/A	N/A	None
3-RCS-150-32-1	17	None	N/A	N/A	N/A	None
3-RCS-008-34-1	13	None	N/A	N/A	N/A	None
3-RCS-008-35-1	14	None	N/A	N/A	N/A	None
3-RCS-150-37-1	13	None	N/A	N/A	N/A	None
3-RCS-150-37-1	15	None	N/A	N/A	N/A	None
3-RCS-150-37-1	16	None	N/A	N/A	N/A	None

TABLE 3.6–14 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP DRAIN PIPING, LOOP FILL PIPING, LETDOWN LINE, AND NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-RCS-150-37-1	17	None	N/A	N/A	N/A	None
3-RCS-008-39-1	13	None	N/A	N/A	N/A	None
3-RCS-008-40-1	14	None	N/A	N/A	N/A	None
3-RCS-150-42-1	13	None	N/A	N/A	N/A	None
3-RCS-150-42-1	15	None	N/A	N/A	N/A	None
3-RCS-150-42-1	16	None	N/A	N/A	N/A	None
3-RCS-150-42-1	17	None	N/A	N/A	N/A	None
3-RCS-002-127-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-127-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-127-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-130-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-130-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-130-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-135-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-135-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-135-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-143-1	TP	None	N/A	N/A	N/A	None

TABLE 3.6–14 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP DRAIN PIPING, LOOP FILL PIPING, LETDOWN LINE, AND NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-RCS-002-143-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-143-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-126-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-126-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-126-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-129-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-129-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-129-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-134-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-134-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-134-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-142-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-142-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-142-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-128-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-128-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-128-1	TP	None	N/A	N/A	N/A	None

TABLE 3.6–14 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP DRAIN PIPING, LOOP FILL PIPING, LETDOWN LINE, AND NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-RCS-002-131-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-131-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-131-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-136-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-136-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-136-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-144-1	TP	None	N/A	N/A	N/A	None
3-RCS-002-144-1	IP	None	N/A	N/A	N/A	None
3-RCS-002-144-1	TP	None	N/A	N/A	N/A	None
3-RCS-003-137-1	1	None	N/A	N/A	N/A	None
3-RCS-003-137-1	2	None	N/A	N/A	N/A	None
3-RCS-003-137-1	3	None	N/A	N/A	N/A	None
3-RCS-003-137-1	4	None	N/A	N/A	N/A	None
3-RCS-003-137-1	5	None	N/A	N/A	N/A	None
3-RCS-003-137-1	6	None	N/A	N/A	N/A	None
3-RCS-003-137-1	7	None	N/A	N/A	N/A	None
3-RCS-003-137-1	8	None	N/A	N/A	N/A	None

TABLE 3.6–14 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP DRAIN PIPING, LOOP FILL PIPING, LETDOWN LINE, AND NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-RCS-003-145-1	1	None	None	None	None	None
3-RCS-003-145-1	2	None	None	None	None	None
3-RCS-003-145-1	3	3RCS*V132 on 3-RCS-150-042-1		a		None
3-RCS-003-145-1	4	3RCS*V132 on 3-RCS-150-042-1		(a)		None
3-RCS-003-145-1	5	None	None	None	None	None
3-RCS-003-145-1	6	None	None	None	None	None
3-RCS-003-145-1	7	3RCS*V132 on 3-RCS-150-042-1		(a)		None
3-RCS-003-145-1	8	3RCS*V132 on 3-RCS-150-042-1		(a)		None
3-RCS-003-145-1	9	3RCS*V132 on 3-RCS-150-042-1		(a)		None
3-RCS-003-149-1	11	3RCS*V25 on 3-RCS-750-110-2		b		None
3-RCS-003-149-1	12	3RCS*V25 on 3-RCS-750-110-2		(b)		None

TABLE 3.6–14 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - LOOP STOP VALVE BYPASS PIPING, EXCESS LETDOWN PIPING, LOOP DRAIN PIPING, LOOP FILL PIPING, LETDOWN LINE, AND NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance To Target	Jet Intensity At The Target	Jet Load On The Target	Protection Requirement
3-RCS-003-149-1	13	None	None	None	None	None
3-RCS-003-149-1	14	None	None	None	None	None
3-RCS-003-149-1	15	None	None	None	None	None
3-RCS-003-149-1	16	None	None	None	None	None
3-RCS-003-149-1	17	None	None	None	None	None
3-RCS-003-149-1	18	None	None	None	None	None
3-RCS-003-149-1	19	None	None	None	None	None

a. The NSSS System Standard Design Criteria states that break propagation to an unaffected leg of an affected loop be prevented. The valve 3RCS*V25 and line 3-RCS-150-042-1 are part of the Loop Stop Valve Bypass System connecting to the Hot Leg. Loss of pressure boundary is admitted and since this connection is to the Hot Leg, the break propagation is transmitted to an unaffected leg of an affected loop. This result was transmitted via NEU-6039 Dated December 23, 1985. This condition is enveloped by Case 1.1 in NES-40190 and evaluated on Page 2 of NEU-6039. The break propagation for 3-RCS-150-042-1, namely 1.4 square inches, and the given break area of the ruptured pipe, 3-RCS-003-145-1, namely 5.4 square inches are enveloped by the Case 1.1 analyzed in NEU-6039. The conclusion of the NSSS vendor is that when breaks of equal size occur on the Hot Leg and the Cold Leg the calculated Emergency Core Cooling System (ECCS) performance is less severe than is predicted for a Cold Leg break with an area equal to the total of the two leg areas combined. In Case 1.1, the Hot Leg break area propagation is 37.3 square inches exceeds the Steam Generator Cold Leg break area, 20 square inches. For a combined Hot Leg/Cold Leg break case of 20 square inches in each loop, the total break areas is 40 square inches which corresponds to a 7.1" equivalent diameter break. Millstone 3 analysis demonstrates that a 6" equivalent diameter Cold leg break in calculated Emergency Core Cooling System (ECCS) performance is less severe than a 4" break, thus a 7.1" break is expected to be less limiting than the 6" or 4" breaks.

b. The NSSS System Standard Design Criteria states that break propagation to an unaffected leg of an affec valve, 3RCS*V25 and line 3-RCS-750-110-2 are part of the Reactor Plant Sampling System connection f pressure boundary is admitted and since this connection is to the Hot Leg, the break propagation is transm an affected loop. This result was transmitted to the NSSS Vendor via NES-40190 Dated November 27, 19 by the NSSS Vendor concerning this propagation and the results are transmitted via NEU-6039 Dated De condition is identified as Case 3.2 in NES-40190 and evaluated on Page 2 of NEU-6039. The NSSS Vend propagation break area, namely 0.3 square inches for 3-RCS-750-110-2 is trivial compared to the break a RCS-003-149-1, namely 5.4 square inches. Therefore phenomena associated at the initial location are not and calculated Emergency Core Cooling System (ECCS) performance is about the same as it would be w interaction is acceptable therefore protective hardware is not required.

TABLE 3.6–15 POSTULATED BREAKS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING

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Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress		Usage Factor	Figure
					Eqn 10	Eqn 12 Or 13		
3-RCS-014-64-1	11	Containment	19'-14"	CB	N/A Terminal End			3.6–14
3-RCS-014-64-1	37	Containment	20'-2"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	38	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	39	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	40	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	41	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	42	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	43	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	44	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	45	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	46	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	47	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	48	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	49	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	50	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	51	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	52	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	53	Containment	22'-4"	CB & LS	Above Threshold			3.6–14

TABLE 3.6–15 POSTULATED BREAKS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress		Usage Factor	Figure
					Eqn 10	Eqn 12 Or 13		
3-RCS-014-64-1	54	Containment	22-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	55	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	56	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	57	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	58	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	59	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	60	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	61	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	62	Containment	22'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	63	Containment	24'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-014-64-1	64	Containment	25'-11"	CB	N/A Terminal End			3.6–14
3-RCS-004-224-1	1	Containment	78'-6"	CB	N/A Terminal End			3.6–14
3-RCS-004-224-1	2	Containment	81'-3"	CB & LS	Above Threshold			3.6–14
3-RCS-004-224-1	3	Containment	82'-11"	CB & LS	Above Threshold			3.6–14
3-RCS-006-68-1	4	Containment	62'-3"	CB & LS	Above Threshold			3.6–14
3-RCS-004-61-1	5	Containment	52'-1"	CB & LS	Above Threshold			3.6–14
3-RCS-004-22-1	6	Containment	37'-8"	CB	N/A Terminal End			3.6–14
3-RCS-004-22-1	7	Containment	18'-0"	CB	N/A - Arbitrary Intermediate			3.6–14
3-RCS-004-22-1	8	Containment	17'-6"	CB	N/A - Arbitrary Intermediate			3.6–14

TABLE 3.6–15 POSTULATED BREAKS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type [1]	Threshold Stress		Usage Factor	Figure
					Eqn 10	Eqn 12 Or 13		
3-RCS-004-22-1	9	Containment	17'-6"	CB	N/A Terminal End			3.6–14
3-RCS-004-60-1	10	Containment	54'-4"	CB & LS	Above Threshold			3.6–14
3-RCS-004-21-1	11	Containment	52'-1"	CB & LS	Above Threshold			3.6–14
3-RCS-004-21-1	12	Containment	41'-6"	CB	N/A Terminal End			3.6–14
3-RCS-004-21-1	13	Containment	17'-6"	CB & LS	Above Threshold			3.6–14
3-RCS-004-21-1	14	Containment	17'-6"	CB	N/A - Arbitrary Intermediate			3.6–14
3-RCS-004-21-1	15	Containment	17'-6"	CB	N/A Terminal End			3.6–14
3-RCS-002-170-1	16	Containment	16'-6"	CB	N/A Terminal End			3.6–14
3-RCS-002-150-1	17	Containment	62'-3"	CB	N/A Terminal End			3.6–14
3-RCS-006-65-1	18	Containment	77'-1"	CB	N/A Terminal End			3.6–14
3-RCS-006-65-1	20	Containment	77'-9"	CB & LS	Above Threshold			3.6–14
3-RCS-003-66-1	21	Containment	75'-0"	CB	Above Threshold			3.6–14
3-RCS-003-69-1	22	Containment	75'-0"	CB	N/A Terminal End			3.6–14
3-RCS-003-67-1	23	Containment	75'-0"	CB	Above Threshold			3.6–14
3-RCS-003-70-1	24	Containment	75'-0"	CB	N/A Terminal End			3.6–14
3-RCS-003-69-1	37	Containment	75'-0"	CB	Above Threshold			3.6–14
3-RCS-003-69-1	38	Containment	75'-0"	CB	Above Threshold			3.6–14
3-RCS-003-70-1	39	Containment	75'-0"	CB	Above Threshold			3.6–14
3-RCS-006-82-1	25	Containment	77'-1"	CB	N/A Terminal End			3.6–14

TABLE 3.6–15 POSTULATED BREAKS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress		Usage Factor	Figure
					Eqn 10	Eqn 12 Or 13		
3-RCS-006-82-1	26	Containment	74'-11"	CB & LS	Above Threshold			3.6–14
3-RCS-006-82-1	27	Containment	74'-11"	CB	N/A - Arbitrary Intermediate			3.6–14
3-RCS-006-82-1	28	Containment	77'-5"	CB	N/A Terminal End			3.6–14
3-RCS-006-83-1	29	Containment	77'-1"	CB	N/A Terminal End			3.6–14
3-RCS-006-83-1	30	Containment	74'-11"	CB & LS	Above Threshold			3.6–14
3-RCS-006-83-1	31	Containment	74'-11"	CB	N/A - Arbitrary Intermediate			3.6–14
3-RCS-006-83-1	32	Containment	77'-5"	CB	N/A Terminal End			3.6–14
3-RCS-006-84-1	33	Containment	77'-1"	CB	N/A Terminal End			3.6–14
3-RCS-006-84-1	34	Containment	74'-11"	CB & LS	Above Threshold			3.6–14
3-RCS-006-84-1	35	Containment	74'-11"	CB	N/A - Arbitrary Intermediate			3.6–14
3-RCS-006-84-1	36	Containment	77'-5"	CB	N/A Terminal End			3.6–14

NOTE:

(1) Circumferential Pipe Break (CB) and Longitudinal Pipe Split (LS) are defined in FSAR Section 3.6.2.1.3.

TABLE 3.6–16 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING

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Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-014-64-1	11	3RCS*TK1	3RCS*SG1B Support System	3RCS-PRR1
3-RCS-014-64-1	37	3RCS*TK1	3RCS*SG1B Support System	3RCS-PRR1
3-RCS-014-64-1	37	3-RCS-029-6-1	None	None
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	38	3RCS*TK1	None	None
3-RCS-014-64-1	38	3-RCS-029-6-1	None	None
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	39	3RCS*TK1	None	None
3-RCS-014-64-1	39	3-RCS-029-6-1	None	None
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	40	3RCS*TK1	None	None
3-RCS-014-64-1	40	3-RCS-029-6-1	None	None
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	41	3RCS*TK1	None	None
3-RCS-014-64-1	41	3-RCS-029-6-1	None	None
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	42	3RCS*TK1	None	None
3-RCS-014-64-1	42	3-RCS-029-6-1	None	None

TABLE 3.6–16 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	43	3RCS*TK1	Pressurizer Cubicle Wall (3)	3RCS-PRR3
3-RCS-014-64-1	43	3-RCS-029-6-1	None	None
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	44	3RCS*TK1	Pressurizer Cubicle Wall (3)	3RCS-PRR3
3-RCS-014-64-1	44	3-RCS-029-6-1	None	None
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	45	3RCS*TK1	Pressurizer Cubicle Wall (3)	3RCS-PRR3
3-RCS-014-64-1	45	3-RCS-029-6-1	None	None
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	46	3RCS*TK1	Pressurizer Cubicle Wall (3)	3RCS-PRR3
3-RCS-014-64-1	46	3-RCS-029-6-1	None	None
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	47	3RCS*TK1	Pressurizer Cubicle Wall (3)	3RCS-PRR3
3-RCS-014-64-1	47	3-RCS-029-6-1	None	None
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	48	3RCS*TK1	Pressurizer Cubicle Wall (3)	3RCS-PRR3

TABLE 3.6–16 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-014-64-1	48	3-RCS-029-6-1	None	None
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	49	3RCS*TK1	Crane Wall (3)	3RCS-PRR5
3-RCS-014-64-1	49	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR3
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	50	3RCS*TK1	Crane Wall (3)	3RCS-PRR5
3-RCS-014-64-1	50	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR3
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	51	3RCS*TK1	Crane Wall (3)	3RCS-PRR5
3-RCS-014-64-1	51	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR3
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	52	3RCS*TK1	Crane Wall (3)	3RCS-PRR5
3-RCS-014-64-1	52	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR3
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	53	3RCS*TK1	Crane Wall - (3)	3RCS-PRR6
3-RCS-014-64-1	53	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR4

TABLE 3.6–16 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	54	3RCS*TK1	Crane Wall (3)	3RCS-PRR6
3-RCS-014-64-1	54	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR4
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	55	3RCS*TK1	Crane Wall (3)	3RCS-PRR6
3-RCS-014-64-1	55	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR4
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	56	3RCS*TK1	Crane Wall (3)	3RCS-PRR6
3-RCS-014-64-1	56	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR4
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	57	3RCS*TK1	Crane Wall (3)	3RCS-PRR6
3-RCS-014-64-1	57	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR4
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	58	3RCS*TK1	Crane Wall (3)	3RCS-PRR6
3-RCS-014-64-1	58	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR7
3-RCS-014-64-1	Split	(2)	None	None

TABLE 3.6–16 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-014-64-1	59	3RCS*TK1	Crane Wall (3)	3RCS-PRR6
3-RCS-014-64-1	59	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR7
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	60	3RCS*TK1	Crane Wall (3)	3RCS-PRR6
3-RCS-014-64-1	60	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR7
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	61	3RCS*TK1	Pressurizer Cubicle Wall (3)	3RCS-PRR8
3-RCS-014-64-1	61	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR7
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	62	3RCS*TK1	Pressurizer Cubicle Wall (3)	3RCS-PRR8
3-RCS-014-64-1	62	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR7
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	63	3RCS*TK1	None	None
3-RCS-014-64-1	63	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR8
3-RCS-014-64-1	Split	(2)	None	None
3-RCS-014-64-1	64	3-RCS-029-6-1	Pressurizer Cubicle Wall (3)	3RCS-PRR8

TABLE 3.6–16 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # [1]	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-004-224-1	1	[4]	None	None
3-RCS-004-224-1	2	[4]	None	None
3-RCS-004-224-1	2	3RCS*TK1	None	None
3-RCS-004-224-1	Split	[4]	None	None
3-RCS-004-224-1	3	[4]	None	None
3-RCS-004-224-1	3	3RCS*TK1	None	None
3-RCS-004-224-1	Split	[4]	None	None
3-RCS-006-68-1	4	[4]	None	None
3-RCS-006-68-1	4	3RCS*TK1	None	None
3-RCS-006-68-1	Split	[4]	None	None
3-RCS-004-61-1	5	[4]	None	None
3-RCS-004-61-1	5	3RCS*TK1	None	None
3-RCS-004-61-1	Split	[4]	None	None
3-RCS-004-22-1	6	[4]	None	None
3-RCS-004-22-1	6	3RCS*TK1	None	None
3-RCS-004-22-1	7	3RCS*TK1	None [5]	None
3-RCS-004-22-1	7	3-RCS-275-10-1	None [5]	None

TABLE 3.6–16 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-004-22-1	8	3RCS*TK1	None (5)	None
3-RCS-004-22-1	8	3-RCS-275-10-1	None (5)	None
3-RCS-004-22-1	9	3RCS*TK1	None	None
3-RCS-004-60-1	10	3RCS*TK1	None	None
3-RCS-004-60-1	10	(4)	None	None
3-RCS-004-60-1	Split	(4)	None	None
3-RCS-004-21-1	11	3RCS*TK1	None	None
3-RCS-004-21-1	11	3-RCS-275-10-1	None	None
3-RCS-004-21-1	Split	(4)	None	None
3-RCS-004-21-1	12	3RCS*TK1	None	None
3-RCS-004-21-1	12	3-RCS-275-5-1	None	None
3-RCS-004-21-1	13	3RCS*TK1	None	None
3-RCS-004-21-1	13	3-RCS-275-5-1	None	None
3-RCS-004-21-1	Split	(4)	None	None
3-RCS-004-21-1	14	3RCS*TK1	None (5)	None
3-RCS-004-21-1	14	3-RCS-275-5-1	None (5)	None
3-RCS-004-21-1	15	3RCS*TK1	None	None

TABLE 3.6–16 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-004-21-1	15	3-RCS-275-5-1	None	None
3-RCS-002-170-1	16	(4)	None	None
3-RCS-002-150-1	17	(4)	None	None
3-RCS-006-65-1	18	3RCS*TK1	None	None
3-RCS-006-65-1	Split	3RCS*TK1	None	None
3-RCS-006-65-1	20	3RCS*TK1	3-RCS-004-224-1(6)	3-RCS-PRR906
3-RCS-003-66-1	21	3RCS*TK1	None	None
3-RCS-003-69-1	22	3RCS*TK1	None	None
3-RCS-003-67-1	23	3RCS*TK1	None	None
3-RCS-003-70-1	24	3RCS*TK1	None	None
3-RCS-006-82-1	25	3RCS*TK1	None	None
3-RCS-006-82-1	26	3RCS*TK1	None	None
3-RCS-006-82-1	Split	3RCS*TK1	None	None
3-RCS-006-82-1	27	3RCS*TK1	None (5)	None
3-RCS-006-82-1	28	3RCS*TK1	None	None
3-RCS-006-83-1	29	3RCS*TK1	None	None
3-RCS-006-83-1	30	3RCS*TK1	None	None
3-RCS-006-83-1	Split	3RCS*TK1	None	None

TABLE 3.6–16 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-006-83-1	31	3RCS*TK1	None (5)	None
3-RCS-006-83-1	32	3RCS*TK1	None	None
3-RCS-006-84-1	33	3RCS*TK1	None	None
3-RCS-006-84-1	34	3RCS*TK1	3RCS-004-224-1 (6)	3RCS-PRR945
3-RCS-006-84-1	Split	3RCS*TK1	None	None
3-RCS-006-84-1	35	3RCS*TK1	None (5)	None
3-RCS-006-84-1	36	3RCS*TK1	None	None
3-RCS-003-69-1	37	3RCS*TK1	None	None
3-RCS-003-69-1	38	3RCS*TK1	None	None
3-RCS-003-70-1	39	3RCS*TK1	None	None

TABLE 3.6–16 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement

NOTES:

(1) Repetition of Break Numbers is used to identify separate fluid reservoirs which provide a constant pressure source to maintain system pressure subsequent to a postulated pipe break. These reservoirs maintain system blowdown during the transient event as well as in the steady state.

(2) Longitudinal Splits (LS) on the Pressurizer Surge Line result in blowdown from both the Hot Leg 3-RCS-029-6-1 and the Pressurizer 3RCS*TK1 since the piping remains intact for a LS as stated in FSAR Section 3.6.2.1.3.

(3) The structure identified if subjected to unrestrained pipe whip of the Pressurizer Surge Line would not remain structurally integral and as a result damage to essential systems subsequent to secondary missiles etc. is credible. The pipe rupture restraints are provided to prevent the unrestrained impact to the structure.

(4) Circumferential Breaks (CB) on the Pressurizer Spray Line have Cold Legs 3-RCS-275-5-1 and 3-RCS-275-10-1 as blowdown sources. Longitudinal Splits (LS) on the Pressurizer Spray Line have both Cold Legs 3-RCS-275-5-1 and 3-RCS-275-10-1 and the Pressurizer 3RCS*TK1 as blowdown sources.

(5) USNRC Generic Letter 87-11 eliminates the need to evaluate the effects of pipe whip for Arbitrary Intermediate Breaks (AIBs). therefore pipe whip effects for AIBs on the Pressurizer Safety Lines are not evaluated.

(6) The NSSS System Standard Design Criteria does not permit break propagation from the Pressurizer Relief Line (Steam Phase) to the Pressurizer Spray Line (Liquid Phase). The pipe rupture restraint is provided to prevent the interaction.

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING

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Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-014-64-1	11	3-RCS-008-30-1		(2)		None
3-RCS-014-64-1	11	3-RCS-006-119-1		(3)		None
3-RCS-014-64-1	37	3-RCS-008-30-1		(2)		None
3-RCS-014-64-1	37	3-RCS-006-119-1		(3)		None
3-RCS-014-64-1	Split	Primary Shield Wall	2 Feet	1274 Psi	774 Kips	None
3-RCS-014-64-1	Split	3-RCS-008-30-1		(4)		None
3-RCS-014-64-1	38	Primary Shield Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	38	None	N/A	N/A	N/A	None
3-RCS-014-64-1	Split	None	N/A	N/A	N/A	None
3-RCS-014-64-1	39	Primary Shield Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	39	None	N/A	N/A	N/A	None
3-RCS-014-64-1	Split	None	N/A	N/A	N/A	None
3-RCS-014-64-1	40	Primary Shield Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	40	None	N/A	N/A	N/A	None
3-RCS-014-64-1	Split	None	N/A	N/A	N/A	None
3-RCS-014-64-1	41	Primary Shield Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	41	None	N/A	N/A	N/A	None
3-RCS-014-64-1	Split	None	N/A	N/A	N/A	None

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-014-64-1	42	Primary Shield Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	42	None	N/A	N/A	N/A	None
3-RCS-014-64-1	Split	None	N/A	N/A	N/A	None
3-RCS-014-64-1	43	None	N/A	N/A	N/A	None
3-RCS-014-64-1	43	None	N/A	N/A	N/A	None
3-RCS-014-64-1	Split	None	N/A	N/A	N/A	None
3-RCS-014-64-1	44	North-South Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	44	None	N/A	N/A	N/A	None
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None (6)
3-RCS-014-64-1	45	Primary Shield Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	45	Steam Generator Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	9 Feet	15 psi	65 Kips	None (6)
3-RCS-014-64-1	46	North-South Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-014-64-1	46	Steam Generator Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	9 Feet	15 psi	101 Kips	None (6)
3-RCS-014-64-1	47	North-South Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	47	Steam Generator Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	9 Feet	15 psi	125 Kips	None
3-RCS-014-64-1	48	North-South Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	48	Steam Generator Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	9 Feet	15 psi	125 Kips	None
3-RCS-014-64-1	49	3-RCS-004-21-1	Less Severe than Longitudinal Split Number 49			None
3-RCS-014-64-1	49	North-South Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	Split	3-RCS-004-21-1	4 Feet	586 psi	6 Kips	None (7)

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-014-64-1	50	3-RCS-004-21-1	Less Severe than Longitudinal Split Number 50			None
3-RCS-014-64-1	50	North-South Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None
3-RCS-014-64-1	Split	3-RCS-004-21-1	4 Feet	586 psi	6 Kips	None (7)
3-RCS-014-64-1	51	3-RCS-004-21-1	Less Severe than Longitudinal Split Number 51			None (8)
3-RCS-014-64-1	51	North-South Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None (8)
3-RCS-014-64-1	Split	3-RCS-004-21-1	4 Feet	586 psi	6 Kips	None (7)
3-RCS-014-64-1	52	3-RCS-004-21-1	Less Severe than Longitudinal Split Number 52			None (8)
3-RCS-014-64-1	52	North-South Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None (8)
3-RCS-014-64-1	Split	3-RCS-004-21-1	4 Feet	586 psi	6 Kips	None (7)
3-RCS-014-64-1	53	Crane Wall	Less Severe than Longitudinal Split Number 37 (5)			None (8)
3-RCS-014-64-1	53	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None (8)
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None (6)
3-RCS-014-64-1	54	Crane Wall	Less Severe than Longitudinal Split Number 37 (5)			None (8)

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # [1]	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-014-64-1	54	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None [8]
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None [6]
3-RCS-014-64-1	55	Crane Wall	Less Severe than Longitudinal Split Number 37 [5]			None [8]
3-RCS-014-64-1	55	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None [8]
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	9 Feet	15 psi	66 Kips	None [6]
3-RCS-014-64-1	56	Crane Wall	Less Severe than Longitudinal Split Number 37 [5]			None [8]
3-RCS-014-64-1	56	Steam Generator Cubicle Wall	Less Severe than Longitudinal Split Number 37 [5]			None [8]
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None [6]
3-RCS-014-64-1	57	Steam Generator Cubicle Wall	Less Severe than Longitudinal Split Number 37 [5]			None [8]

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-014-64-1	57	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None (8)
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None (6)
3-RCS-014-64-1	58	Steam Generator Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None (8)
3-RCS-014-64-1	58	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None (8)
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None (6)
3-RCS-014-64-1	59	Steam Generator Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None (8)
3-RCS-014-64-1	59	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None (8)
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None (8)
3-RCS-014-64-1	60	Steam Generator Cubicle Wall	Less Severe than Longitudinal Split Number 37 (5)			None (8)

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # [1]	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-014-64-1	60	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None [8]
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None [6]
3-RCS-014-64-1	61	Crane Wall	Less Severe than Longitudinal Split Number 37 [5]			None [8]
3-RCS-014-64-1	61	Steam Generator Cubicle Wall	Less Severe than Longitudinal Split Number 37 [5]			None [8]
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 48			None [6]
3-RCS-014-64-1	62	Crane Wall	Less Severe than Longitudinal Split Number 37 [5]			None [8]
3-RCS-014-64-1	62	Removable Concrete Slab at Elevation 12'-9"	Less Severe than the Longitudinal Split Number 62			None [8]
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	9 Feet	15 psi	131 Kips	None [6]
3-RCS-014-64-1	63	Crane Wall	Less Severe than Longitudinal Split Number 37 [5]			None [8]
3-RCS-014-64-1	63	Steam Generator Cubicle Wall	Less Severe than Longitudinal Split Number 37 [5]			None [8]

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # [1]	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-014-64-1	Split	Removable Concrete Slab at Elevation 12'-9"	Less Severe than Longitudinal Split Number 37 [5]			None
3-RCS-014-64-1	64	None	N/A	N/A	N/A	None
3-RCS-004-224-1	1	None	N/A	N/A	N/A	None
3-RCS-004-224-1	1	None	N/A	N/A	N/A	None
3-RCS-004-224-1	2	None	N/A	N/A	N/A	None
3-RCS-004-224-1	2	None	N/A	N/A	N/A	None
3-RCS-004-224-1	Split	None	N/A	N/A	N/A	None
3-RCS-004-224-1	3	None	N/A	N/A	N/A	None
3-RCS-004-224-1	3	None	N/A	N/A	N/A	None
3-RCS-004-224-1	Split	None	N/A	N/A	N/A	None
3-RCS-006-68-1	4	None	N/A	N/A	N/A	None
3-RCS-006-68-1	4	None	N/A	N/A	N/A	None
3-RCS-006-68-1	Split	None	N/A	N/A	N/A	None
3-RCS-004-61-1	5	None	N/A	N/A	N/A	None
3-RCS-004-61-1	5	None	N/A	N/A	N/A	None
3-RCS-004-61-1	Split	None	N/A	N/A	N/A	None
3-RCS-004-22-1	6	None	N/A	N/A	N/A	None
3-RCS-004-22-1	6	None	N/A	N/A	N/A	None

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # [1]	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-004-22-1	7	None	N/A	N/A	N/A	None
3-RCS-004-22-1	7	None	N/A	N/A	N/A	None
3-RCS-004-22-1	8	None	N/A	N/A	N/A	None
3-RCS-004-22-1	8	None	N/A	N/A	N/A	None
3-RCS-004-22-1	9	None	N/A	N/A	N/A	None
3-RCS-004-22-1	9	None	N/A	N/A	N/A	None
3-RCS-004-60-1	10	None	N/A	N/A	N/A	None
3-RCS-004-60-1	10	None	N/A	N/A	N/A	None
3-RCS-004-60-1	Split	None	N/A	N/A	N/A	None
3-RCS-004-21-1	11	None	N/A	N/A	N/A	None
3-RCS-004-21-1	11	None	N/A	N/A	N/A	None
3-RCS-004-21-1	Split	None	N/A	N/A	N/A	None
3-RCS-004-21-1	12	None	N/A	N/A	N/A	None
3-RCS-004-21-1	12	None	N/A	N/A	N/A	None
3-RCS-004-21-1	13	North-South Cubicle Wall	Less Severe than Longitudinal Split Number 37 [5]			None
3-RCS-004-21-1	13	None	N/A	N/A	N/A	None
3-RCS-004-21-1	Split	None	N/A	N/A	N/A	None
3-RCS-004-21-1	14	None [9]	N/A	N/A	N/A	None
3-RCS-004-21-1	14	None	N/A	N/A	N/A	None

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-004-21-1	15	None	N/A	N/A	N/A	None
3-RCS-004-21-1	15	None	N/A	N/A	N/A	None
3-RCS-002-170-1	16	None	N/A	N/A	N/A	None
3-RCS-002-150-1	17	None	N/A	N/A	N/A	None
3-RCS-006-65-1	18	3-RCS-004-224-1	Less Severe than Either Longitudinal Split Numbers 49 Through 52			None (10)
3-RCS-006-65-1	20	None	N/A	N/A	N/A	None
3-RCS-006-65-1	Split	None	N/A	N/A	N/A	None
3-RCS-003-66-1	21	None	N/A	N/A	N/A	None
3-RCS-003-69-1	22	None	N/A	N/A	N/A	None
3-RCS-003-67-1	23	None	N/A	N/A	N/A	None
3-RCS-003-70-1	24	None	N/A	N/A	N/A	None
3-RCS-003-69-1	37	None	N/A	N/A	N/A	None
3-RCS-003-69-1	38	None	N/A	N/A	N/A	None
3-RCS-003-70-1	39	None	N/A	N/A	N/A	None
3-RCS-006-82-1	25	None	N/A	N/A	N/A	None
3-RCS-006-82-1	26	None	N/A	N/A	N/A	None
3-RCS-006-82-1	Split	None	N/A	N/A	N/A	None
3-RCS-006-82-1	27	None (9)	N/A	N/A	N/A	None
3-RCS-006-82-1	28	None	N/A	N/A	N/A	None

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-006-83-1	29	None	N/A	N/A	N/A	None
3-RCS-006-83-1	30	None	N/A	N/A	N/A	None
3-RCS-006-83-1	Split	None	N/A	N/A	N/A	None
3-RCS-006-83-1	31	None (9)	N/A	N/A	N/A	None
3-RCS-006-83-1	32	None	N/A	N/A	N/A	None
3-RCS-006-84-1	33	None	N/A	N/A	N/A	None
3-RCS-006-84-1	34	None	N/A	N/A	N/A	None
3-RCS-006-84-1	Split	None	N/A	N/A	N/A	None
3-RCS-006-84-1	35	None (9)	N/A	N/A	N/A	None
3-RCS-006-84-1	36	None	N/A	N/A	N/A	None

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement

NOTES:

(1) Repetition of Break Numbers is used to identify separate fluid reservoirs which provide a constant pressure source to maintain system pressure subsequent to the postulated pipe break. These reservoirs maintain system blowdown during the transient event as well as in the steady state.

(2) Terminal End break is a Large LOCA. The NSSS System Standard Design Criteria requirements are deemed essential. The requirement for a Large LOCA is to prevent leg to leg propagation within an affected loop to less than 20% of the flow area of the line that is broken. Jet impingement on the Loop Stop Valve Bypass Line 3-RCS-008-30-1 does not violate this criteria since the target pipe is loaded proximate to the Hot Leg nozzle and the remainder of the Loop Stop Valve Bypass: Line 3-RCS-008-30-1 is well supported. The jet impingement on the Loop Stop Valve Bypass, Line 3-RCS-008-30-1, is short duration and the initial break area is much smaller than a full double-ended rupture since pipe rupture restraint 3RCS-PRR1 limits separation and results in less than a full flow area exposed thus minimizing the jet loads on the Loop Stop Valve Bypass Line 3-RCS-008-30-1.

(3) The NSSS requirement delineated in Note 1 is applicable for this interaction. The Low Pressure Safety Injection Line 3-RCS-006-119-1 to the Hot Leg does not lose pressure boundary since the portion of the Low Pressure Safety Injection Line 3-RCS-006-119-1 to the Hot Leg is well supported by the Hot Leg. The pipe rupture restraint 3RCS-PRR1 limits separation and thus limits the amount of flow area exposed thus limiting the applied load on the Low Pressure Safety Injection Line 3-RCS-006-119-1 to the Hot Leg.

(4) Jet impingement subsequent to this postulated split targets the Loop Stop Valve Bypass Line. The criteria delineated in Note 1 must be complied with. The jet impingement loading is distributed as discussed in FSAR Section 3.6.2.1 on the Loop Stop Valve Bypass line. This line is well supported by pipe rupture restraints and by the Hot Leg to preclude pressure boundary failure.

(5) Jet impingement on structural barrier walls does not cause either barrier failure (i.e., back face scabbing or punching shear) or gross catastrophic failure of the barrier. This conclusion is based upon generic calculations which conclude that extremely high intensities are required to cause local barrier failure. The intensity, distance, and load for jet impingement effects on the Pressurizer Cubicle Piping is governed by the Longitudinal Split (LS) Number 37 which is the closest to the indicated barrier and produces the largest loads.

TABLE 3.6–17 JET IMPINGEMENT EFFECTS - REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement

(6) The barrier identified as essential for this break is a removable slab at Elevation 12'-9" in the Pressurizer Cubicle. This slab provides a radiation protection function and cannot fail catastrophically. The indicated jet intensity does not cause either local barrier failure (i.e., back face scabbing or punching shear) or gross catastrophic failure of the barrier. This is based upon generic calculations which conclude that much larger intensities are required to cause local barrier failure.

(7) This postulated Longitudinal Split (LS) results in fluid jet impingement on the Pressurizer Spray Line associated with another loop. The NSSS System Standard Design Criteria does not permit propagation of a pipe break to an unaffected loop. This NSSS requirement although not strictly defined as essential per FSAR Section 3.6.1 is complied with so as not to increase the severity of the LOCA. The jet impingement load in combination with other loads in accordance with FSAR Section 3.9B-10 results in a stress less than 3Sm. Compliance with this stress allowable demonstrates pressure boundary integrity.

(8) This circumferential break (CB) results in radial jet impingement. Radial jet impingement is less severe than the fluid jet impingement associated with Longitudinal Splits Numbers 37 or 48 since pipe rupture restraints on the Pressurizer Surge Line, 3-RCS-014-64-1, limit both axial and lateral displacement. Limiting the displacement limits the amount of flow area exposed to less than the flow area of the Pressurizer Surge Line, 3-RCS-014-64-1 thus reducing jet intensity on the target.

(9) USNRC Generic Letter 87-11 eliminates the requirement to evaluate jet impingement effects subsequent to Arbitrary Intermediate Breaks (AIBs). Therefore, jet impingement effects are not evaluated by AIBs on the Pressurizer Safety Lines.

(10) This postulated break results in a Large LOCA resulting in a release of saturated steam. The NSSS System Standard Design Criteria does not permit propagation from a steam phase to liquid phase LOCA. Liquid phase LOCA would occur if the Pressurizer Spray Line ruptures subsequent to fluid jet impingement from the Pressurizer Relief Line. This particular interaction is enveloped by the postulated Longitudinal Splits (LS) Numbers 49 through 52 on the Pressurizer Surge Line, 3-RCS-014-64-1. Pressure boundary integrity is maintained based upon the discussion provided in Note 5 since the Pressurizer Surge Line Splits are more severe.

TABLE 3.6–18 POSTULATED PIPE BREAKS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress Eqn 10 & Either Eqn 12 or 13 Usage Factor	Figure
3-RCS-010-122-1	10	Containment	18'-5"	CB	N/A Terminal End	3.6–13
3-RCS-010-122-1	Split	Containment	18'-9"	LS	(2)	3.6–13
3-SIL-010-44-2	11	Containment	-22'-0"	CB	N/A Terminal End	3.6–13
3-SIL-010-45-1	2	Containment	11'-10"	CB & LS	Above Threshold	3.6–13
3-SIL-010-45-1	3	Containment	17'-3"	CB & LS	Above Threshold	3.6–13
3-SIL-006-139-1	TP	Containment	13'-7"	CB	N/A Terminal End	3.6–13
3-RCS-010-132-1	10	Containment	18'-3"	CB	N/A Terminal End	3.6–13
3-RCS-010-132-1	Split	Containment	18'-9"	LS	(2)	3.6–13
3-SIL-010-46-2	11	Containment	-22'-0"	CB	N/A Terminal End	3.6–13
3-SIL-010-47-1	2	Containment	11'-10"	CB & LS	Above Threshold	3.6–13
3-SIL-010-47-1	3	Containment	17'-3"	CB & LS	Above Threshold	3.6–13
3-SIL-006-140-1	TP	Containment	13'-7"	CB	N/A Terminal End	3.6–13
3-RCS-010-138-1	10	Containment	18'-1"	CB	N/A Terminal End	3.6–13
3-RCS-010-138-1	Split	Containment	18'-9"	LS	(2)	3.6–13
3-SIL-010-48-2	11	Containment	-22'-0"	CB	N/A Terminal End	3.6–13
3-SIL-010-49-1	2	Containment	11'-10"	CB & LS	Above Threshold	3.6–13
3-SIL-010-49-1	3	Containment	17'-3"	CB & LS	Above Threshold	3.6–13

TABLE 3.6–18 POSTULATED PIPE BREAKS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress Eqn 10 & Either Eqn 12 or 13 Usage Factor	Figure
3-SIL-006-145-1	TP	Containment	13'-7"	CB	N/A Terminal End	3.6–13
3-RCS-010-146-1	10	Containment	18'-5"	CB	N/A Terminal End	3.6–13
3-RCS-010-146-1	Split	Containment	18'-9"	LS	(2)	3.6–13
3-SIL-010-50-2	11	Containment	-22'-0"	CB	N/A Terminal End	3.6–13
3-SIL-010-51-1	2	Containment	11'-10"	CB & LS	Above Threshold	3.6–13
3-SIL-010-51-1	3	Containment	17'-3"	CB & LS	Above Threshold	3.6–13
3-SIL-006-146-1	TP	Containment	13'-7"	CB	N/A Terminal End	3.6–13
3-RCS-012-123-1	9	Containment	15'-4"	CB	N/A Terminal End	3.6–13
3-RCS-012-123-1	4	Containment	13'-7"	CB & LS	Above Threshold	3.6–13
3-RCS-012-123-1	5	Containment	13'-7"	CB & LS	Above Threshold	3.6–13
3-RCS-012-123-1	6	Containment	13'-7"	CB & LS	Above Threshold	3.6–13
3-RCS-012-123-1	7	Containment	13'-7"	CB	N/A Terminal End	3.6–13
3-RCS-006-124-1	8	Containment	9'-6"	CB	N/A Terminal End	3.6–13
3-RCS-012-103-1	9	Containment	15'-4"	CB	N/A Terminal End	3.6–13
3-RCS-012-103-1	4	Containment	13'-7"	CB & LS	Above Threshold	3.6–13
3-RCS-012-103-1	5	Containment	13'-7"	CB & LS	Above Threshold	3.6–13
3-RCS-012-103-1	6	Containment	13'-7"	CB & LS	Above Threshold	3.6–13

TABLE 3.6–18 POSTULATED PIPE BREAKS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress Eqn 10 & Either Eqn 12 or 13	Usage Factor	Figure
3-RCS-012-103-1	7	Containment	13'-7"	CB	N/A Terminal End		3.6–13
3-RCS-006-140-1	8	Containment	9'-6"	CB	N/A Terminal End		3.6–13
3-RCS-006-119-1	TP	Containment	19'-1"	CB	N/A (3)	N/A (3)	3.6–13
3-RCS-006-119-1	IP	Containment	20'-10"	CB & LS	N/A (3)	N/A (3)	3.6–13
3-RCS-006-119-1	TP	Containment	20'-10"	CB	N/A (3)	N/A (3)	3.6–13
3-RCS-006-120-1	TP	Containment	19'-1"	CB	N/A (3)	N/A (3)	3.6–13
3-RCS-006-120-1	IP	Containment	20'-10"	CB & LS	N/A (3)	N/A (3)	3.6–13
3-RCS-006-120-1	TP	Containment	20'-10"	CB	N/A (3)	N/A (3)	3.6–13
3-SIH-004-16-2	12	Auxiliary	12'-6"	CB	N/A Terminal End		(4)
3-SIH-004-22-2	13	Auxiliary	12'-6"	CB	N/A Terminal End		(4)
3-SIH-004-16-2	14	Auxiliary	12'-6"	CB	N/A - Arbitrary Intermediate		(4)
3-SIH-004-16-2	15	Auxiliary	12'-6"	CB	N/A - Arbitrary Intermediate		(4)
3-SIH-004-16-2	16	Auxiliary	18'-5"	CB	N/A Terminal End		(4)
3-RCS-003-133-1	TP	Containment	17'-6"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-003-133-1	IP	Containment	17'-6"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-150-221-1	IP	Containment	17'-6"	CB	N/A (5)	N/A (5)	Not Shown

TABLE 3.6–18 POSTULATED PIPE BREAKS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress Eqn 10 & Either Eqn 12 or 13	Usage Factor	Figure
3-RCS-150-221-1	TP	Containment	17'-6"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-003-139-1	TP	Containment	17'-6"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-003-139-1	IP	Containment	17'-6"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-150-222-1	IP	Containment	17'-6"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-150-222-1	TP	Containment	17'-6"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-003-147-1	TP	Containment	17'-6"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-003-147-1	IP	Containment	16'-6"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-150-223-1	IP	Containment	9'-10"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-150-223-1	TP	Containment	9'-10"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-003-121-1	TP	Containment	17'-6"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-003-121-1	IP	Containment	17'-6"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-150-220-1	IP	Containment	16'-3"	CB	N/A (5)	N/A (5)	Not Shown
3-RCS-150-220-1	TP	Containment	16'-3"	CB	N/A (5)	N/A (5)	Not Shown

TABLE 3.6–18 POSTULATED PIPE BREAKS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Threshold Stress Eqn 10 & Either Eqn 12 or 13	Usage Factor	Figure

NOTES:

(1) Circumferential Pipe Break (CB) and Longitudinal Pipe Split (LS) are defined in FSAR Section 3.6.2.1.3.

(2) Only a Longitudinal Split (LS) is postulated along the intrados of the cut elbow of the Accumulator Discharge Line connection to each Cold Leg. A detailed stress analysis was performed as described in FSAR Section 3.6.2.1.3.1 and the stress analysis indicates that a LS is preferential at this location

(3) Breaks are postulated at any location (IP) along the Low Pressure Safety Injection System to the Hot Leg on Loops B & C. Both Circumferential Breaks (CB) and Longitudinal Splits (LS) are postulated. The Terminal Point (TP) at Check Valves 3RCS*V8949B and C and at each connection to the Hot Leg and Intermediate Point (IP) at any location for the Low Pressure Safety Injection System to the Hot Legs 3-RCS-029-6-1 (Loop B) and 3-RCS-029-11-1 (Loop C).

(4) High Pressure Safety Injection (SIH) System from the Normal Charging Pumps to the Containment Isolation Valves 3SIH*MV8801A and 3SIH*MV8801B is a High Energy System since the lines remain pressurized during Normal Plant Operating Conditions and are reviewed for postulated pipe break.

(5) Breaks are postulated at any location (IP) along the High Pressure Safety Injection System into each Cold Leg. Only Circumferential Breaks (CB) are postulated since the High Pressure Safety Injection System has a nominal pipe size less than 4". Terminal Points (TP) at the Cold Leg and Check Valves 3RCS*V8900A, B, C, or D and Intermediate Points (IP) at any location are postulated on the High Pressure Safety Injection System into each Cold Leg.

TABLE 3.6–19 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-010-122-1	10	3-RCS-275-5-1	3-RCS-008-24-1 (1)	3RCS-PRR6A
3-RCS-010-122-1	Split	3-RCS-275-5-1	None	None
3-RCS-010-44-1	11	3SIL*TK1A	None	None
3-SIL-010-45-1	2	3SIL*TK1A	None	None
3-SIL-010-45-1	Split	3SIL*TK1A	(2)	3SIL-PRR4A
3-SIL-010-45-1	3	3SIL*TK1A	None	None
3-SIL-010-45-1	Split	3SIL*TK1A	(2)	3SIL-PRR4A
3-SIL-006-139-1	TP	3SIL*TK1A	(2)	3SIL-PRR4A
3-RCS-010-132-1	10	3-RCS-275-10-1	3-RCS-008-30-1 (1)	3RCS-PRR6B
3-RCS-010-132-1	Split	3-RCS-275-10-1	None	None
3-RCS-010-46-1	11	3SIL*TK1B	None	None
3-SIL-010-47-1	2	3SIL*TK1B	None	None
3-SIL-010-47-1	Split	3SIL*TK1B	(2)	3SIL-PRR4B
3-SIL-010-47-1	3	3SIL*TK1B	None	None
3-SIL-010-47-1	Split	3SIL*TK1B	(2)	3SIL-PRR4B
3-SIL-006-140-1	TP	3SIL*TK1B	(2)	3SIL-PRR4B
3-RCS-010-138-1	10	3-RCS-275-15-1	3-RCS-008-35-1 (1)	3RCS-PRR6C

TABLE 3.6–19 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-010-138-1	Split	3-RCS-275-15-1	None	None
3-RCS-010-48-1	11	3SIL*TK1C	None	None
3-SIL-010-49-1	2	3SIL*TK1C	None	None
3-SIL-010-49-1	Split	3SIL*TK1C	(2)	3SIL-PRR4C
3-SIL-010-49-1	3	3SIL*TK1C	None	None
3-SIL-010-49-1	Split	3SIL*TK1C	(2)	3SIL-PRR4C
3-SIL-006-145-1	TP	3SIL*TK1C	(2)	3SIL-PRR4C
3-RCS-010-146-1	10	3-RCS-275-20-1	3-RCS-008-40-1 (1)	3RCS-PRR6D
3-RCS-010-146-1	Split	3-RCS-275-20-1	None	None
3-RCS-010-50-1	11	3SIL*TK1D	None	None
3-SIL-010-51-1	2	3SIL*TK1D	None	None
3-SIL-010-51-1	Split	3SIL*TK1D	(2)	3SIL-PRR4D
3-SIL-010-51-1	3	3SIL*TK1D	None	None
3-SIL-010-51-1	Split	3SIL*TK1D	(2)	3SIL-PRR4D
3-SIL-006-146-1	TP	3SIL*TK1D	(2)	3SIL-PRR4D
3-RCS-006-119-1	TP	3-RCS-029-6-1	None	None
3-RCS-006-119-1	IP	3-RCS-029-6-1	None	None
3-RCS-006-119-1	TP	3-RCS-029-6-1	None	None

TABLE 3.6–19 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-006-120-1	TP	3-RCS-029-11-1	None	None
3-RCS-006-120-1	IP	3-RCS-029-11-1	None	None
3-RCS-006-120-1	TP	3-RCS-029-11-1	None	None
3-RCS-003-133-1	TP	3-RCS-275-10-1	None	None
3-RCS-003-133-1	IP	3-RCS-275-10-1	None	None
3-RCS-150-221-1	IP	3-RCS-275-10-1	None	None
3-RCS-150-221-1	TP	3-RCS-275-10-1	None	None
3-RCS-003-139-1	TP	3-RCS-275-15-1	None	None
3-RCS-003-139-1	IP	3-RCS-275-15-1	None	None
3-RCS-150-222-1	IP	3-RCS-275-15-1	None	None
3-RCS-150-222-1	TP	3-RCS-275-15-1	None	None
3-RCS-003-147-1	TP	3-RCS-275-20-1	None	None
3-RCS-003-147-1	IP	3-RCS-275-20-1	None	None
3-RCS-150-223-1	IP	3-RCS-275-20-1	None	None
3-RCS-150-223-1	TP	3-RCS-275-20-1	None	None
3-RCS-003-121-1	TP	3-RCS-275-5-1	None	None
3-RCS-003-121-1	IP	3-RCS-275-5-1	None	None
3-RCS-150-220-1	IP	3-RCS-275-5-1	None	None

TABLE 3.6–19 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-RCS-150-220-1	TP	3-RCS-275-5-1	None	None
3-RCS-012-123-1	9	3-RCS-029-1-1	Containment Liner (3)	3RHS-PRS1A
3-RCS-012-123-1	4	3-RCS-029-1-1	Containment Liner (3)	3RHS-PRS2A
3-RCS-012-123-1	5	3-RCS-029-1-1	Containment Liner (3)	3RHS-PRS2A
3-RCS-012-123-1	6	3-RCS-029-1-1	Containment Liner (3)	3RHS-PRS2A
3-RCS-012-123-1	7	3-RCS-029-1-1	Containment Liner (3)	3RHS-PRS2A
3-RCS-012-123-1	Splits	3-RCS-029-1-1	None	None
3-RCS-006-124-1	8	3-RCS-029-1-1	East-West Cubicle Wall (4)	3RCS-PRR861
3-RCS-012-103-1	9	3-RCS-029-16-1	Containment Liner (3)	3RHS-PRS1D
3-RCS-012-103-1	4	3-RCS-029-16-1	Containment Liner (3)	3RHS-PRS2D
3-RCS-012-103-1	5	3-RCS-029-16-1	Containment Liner (3)	3RHS-PRS2D
3-RCS-012-103-1	6	3-RCS-029-16-1	Containment Liner (3)	3RHS-PRS2D
3-RCS-012-103-1	7	3-RCS-029-16-1	Containment Liner (3)	3RHS-PRS2D
3-RCS-012-103-1	Splits	3-RCS-029-16-1	None	None
3-RCS-006-124-1	8	3-RCS-029-16-1	East-West Cubicle Wall (4)	3RCS-PRR862
3-SIH-004-16-2	12	None (5)	None (6)	None

TABLE 3.6–19 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-SIH-004-22-2	13	None (5)	None (6)	None
3-SIH-004-16-2	14	None (5)	None (7)	None
3-SIH-004-16-2	15	None (5)	None (7)	None
3-SIH-004-16-2	16	None (5)	None (6)	None

TABLE 3.6–19 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement

NOTES:

(1) This terminal end break occurs at the Cold Leg Nozzle and results in a large LOCA. The NSSS System Standard Design Criteria requires that break propagation in the affected loop cannot exceed 20% of the flow area of the broken line. The Reactor Coolant System Loop Stop Valve Bypass Line identified has a nominal pipe size of 8" and therefore, if the 8" Loop Stop Valve Bypass Line is impacted pressure boundary integrity must be assured. The pipe rupture restraint indicated prevents the impact.

(2) The Terminal End (TP) break or Longitudinal Split (LS) does not results in a LOCA. The NSSS System Standard Design Criteria delineates requirements for a Large Line Break that does not result in a LOCA. The requirement is that a Non-LOCA cannot propagate into a LOCA. If unrestrained, the broken pipe would induce significant loading upon the Reactor Coolant System pressure boundary. The pipe rupture restraint is provided to prevent this effect.

(3) A break on the Reactor Coolant System Residual Heat Removal Piping is a Large LOCA. The NSSS System Standard design Criteria provides the acceptance criteria for Large LOCA breaks. The main criterion or this break is to ensure that Containment Leak Tight Integrity is maintained. The pipe whip of the Reactor Coolant System Residual Heat Removal Piping results in potential impact with the Containment Liner. This is an unacceptable interaction and is prevented by the pipe rupture restraints.

(4) Unrestrained pipe whip impact into the East - West Cubicle Wall would cause catastrophic failure of the wall. Therefore, pipe rupture restraints are provided to prevent this impact.

(5) The High Pressure Safety Injection (SIH) System in the Auxiliary Building is a High Energy System since the piping is maintained pressurized during Normal Plant Operation between the Normal Charging Pumps and the Normally Closed Containment Isolation Valves, 3SIH*MV8801A and 3SIH*MV8801B. The High Pressure Safety Injection (SIH) System in the Auxiliary Building connects to the Normal Charging Pumps which is not considered a constant pressure source (i.e., reservoir) to sustain system pressure subsequent to a pipe break on the SIH System based upon FSAR Section 3.6.2.2.1, Item 3. The portion between the Normally Closed Containment Isolation Valves, 3SIH*MV8801A and 3SIH*MV8801B and the Reactor Coolant System Pressure Boundary is normally empty.

TABLE 3.6–19 PIPE WHIP EFFECTS - REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement

(6) The High Pressure Safety Injection (SIH) System in the Auxiliary Building does not whip since the High Pressure Safety Injection (SIH) System in the Auxiliary Building connects t the Normal Charging Pumps which is not considered a constant pressure source (i.e., reservoir) to sustain system pressure subsequent to a pipe break on the SIH System based upon FSAR Section 3.6.2.2.1, Item 3. The portion between the Normally Closed Containment Isolation Valves, 3SIH*MV8801A and 3SIH*MV8801B and the Reactor Coolant System Pressure Boundary is normally empty.

(7) USNRC Generic Letter 87-11 eliminates the requirement to evaluate the dynamic effects (i.e., pipe whip) of Arbitrary Intermediate Breaks (AIBs). Therefore, pipe whip effects subsequent to AIBs are not evaluated.

TABLE 3.6–20 JET IMPINGEMENT EFFECTS - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-010-122-1	10	None	N/A	N/A	N/A	None
3-RCS-010-122-1	Split	None	N/A	N/A	N/A	None
3-RCS-010-44-1	11	None	N/A	N/A	N/A	None
3-SIL-010-45-1	2	None	N/A	N/A	N/A	None
3-SIL-010-45-1	Split	None	N/A	N/A	N/A	None
3-SIL-010-45-1	3	None	N/A	N/A	N/A	None
3-SIL-010-45-1	Split	None	N/A	N/A	N/A	None
3-SIL-006-139-1	TP	None	N/A	N/A	N/A	None
3-RCS-010-132-1	10	None	N/A	N/A	N/A	None
3-RCS-010-132-1	Split	None	N/A	N/A	N/A	None
3-RCS-010-46-1	11	None	N/A	N/A	N/A	None
3-SIL-010-47-1	2	None	N/A	N/A	N/A	None
3-SIL-010-47-1	Split	None	N/A	N/A	N/A	None
3-SIL-010-47-1	3	None	N/A	N/A	N/A	None
3-SIL-010-47-1	Split	None	N/A	N/A	N/A	None
3-SIL-006-140-1	TP	None	N/A	N/A	N/A	None
3-RCS-010-138-1	10	None	N/A	N/A	N/A	None

TABLE 3.6–20 JET IMPINGEMENT EFFECTS - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-010-138-1	Split	None	N/A	N/A	N/A	None
3-RCS-010-48-1	11	None	N/A	N/A	N/A	None
3-SIL-010-49-1	2	None	N/A	N/A	N/A	None
3-SIL-010-49-1	Split	None	N/A	N/A	N/A	None
3-SIL-010-49-1	3	None	N/A	N/A	N/A	None
3-SIL-010-49-1	Split	None	N/A	N/A	N/A	None
3-SIL-006-145-1	TP	None	N/A	N/A	N/A	None
3-RCS-010-146-1	10	None	N/A	N/A	N/A	None
3-RCS-010-146-1	Split	None	N/A	N/A	N/A	None
3-RCS-010-50-1	11	None	N/A	N/A	N/A	None
3-SIL-010-51-1	2	None	N/A	N/A	N/A	None
3-SIL-010-51-1	Split	None	N/A	N/A	N/A	None
3-SIL-010-51-1	3	None	N/A	N/A	N/A	None
3-SIL-010-51-1	Split	None	N/A	N/A	N/A	None
3-SIL-006-146-1	TP	None	N/A	N/A	N/A	None
3-RCS-006-119-1	TP	None	N/A	N/A	N/A	None
3-RCS-006-119-1	IP	None	N/A	N/A	N/A	None

TABLE 3.6–20 JET IMPINGEMENT EFFECTS - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-006-119-1	TP	None	N/A	N/A	N/A	None
3-RCS-006-120-1	TP	None	N/A	N/A	N/A	None
3-RCS-006-120-1	IP	None	N/A	N/A	N/A	None
3-RCS-006-120-1	TP	None	N/A	N/A	N/A	None
3-RCS-003-133-1	TP	None	N/A	N/A	N/A	None
3-RCS-003-133-1	IP	(1)		(1)		None
3-RCS-150-221-1	IP	3RCS*V984 and 3RCS*V985		(2)		None
3-RCS-150-221-1	TP	3RCS*V984 and 3RCS*V985		(2)		None
3-RCS-003-139-1	TP	None	N/A	N/A	N/A	None
3-RCS-003-139-1	IP	None	N/A	N/A	N/A	None
3-RCS-150-222-1	IP	None	N/A	N/A	N/A	None
3-RCS-150-222-1	TP	3RCS*V979		(2)		None
3-RCS-003-147-1	TP	None	N/A	N/A	N/A	None
3-RCS-003-147-1	IP	None	N/A	N/A	N/A	None
3-RCS-150-223-1	IP	None	N/A	N/A	N/A	None
3-RCS-150-223-1	TP	None	N/A	N/A	N/A	None
3-RCS-003-121-1	TP	None	N/A	N/A	N/A	None

TABLE 3.6–20 JET IMPINGEMENT EFFECTS - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-RCS-003-121-1	IP	None	N/A	N/A	N/A	None
3-RCS-150-220-1	IP	None	N/A	N/A	N/A	None
3-RCS-150-220-1	TP	None	N/A	N/A	N/A	None
3-RCS-012-103-1	9	None	N/A	N/A	N/A	None
3-RCS-012-103-1	4	3-SSR-375-026-2		(3)		None
3-RCS-012-103-1	5	3-SSR-375-026-2		(3)		None
3-RCS-012-103-1	6	3-SSR-375-026-2		(3)		None
3-RCS-012-103-1	7	3-SSR-375-026-2		(3)		None
3-RCS-012-103-1	Splits	3-SSR-375-026-2		(3)		None
3-RCS-006-124-1	8	None	N/A	N/A	N/A	None
3-RCS-012-123-1	9	None	N/A	N/A	N/A	None
3-RCS-012-123-1	4	3-SSR-375-035-2		(3)		None
3-RCS-012-123-1	5	3-SSR-375-035-2		(3)		None
3-RCS-012-123-1	6	3-SSR-375-035-2		(3)		None
3-RCS-012-123-1	7	3-SSR-375-035-2		(3)		None
3-RCS-012-123-1	Splits	3-SSR-375-035-2		(3)		None
3-RCS-006-124-1	8	None	N/A	N/A	N/A	None

TABLE 3.6–20 JET IMPINGEMENT EFFECTS - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-SIH-004-16-2	12	None (4)	N/A	N/A	N/A	None (5)
3-SIH-004-22-2	13	None (4)	N/A	N/A	N/A	None (5)
3-SIH-004-16-2	14	None (6)	N/A	N/A	N/A	None (5)
3-SIH-004-16-2	15	None (6)	N/A	N/A	N/A	None (5)
3-SIH-004-16-2	16	None (4)	N/A	N/A	N/A	None (5)

TABLE 3.6–20 JET IMPINGEMENT EFFECTS - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement

NOTES:

(1) The NSSS System Standard Design Criteria states for a Small LOCA that break propagation to the unaffected leg of the affected loop should be prevented. 3-RCS-150-032-1 Loop Stop Valve Bypass Line is the essential target and is not isolated to the Hot Leg. Loss of pressure boundary is admitted. In addition to this target, Disc Pressurization Valves 3RCS*V984 and 3RCS*V985 are targeted. Break propagation for an initiating event on the High Pressure Safety Injection Line is not permitted. These results were transmitted to the NSSS Vendor via NES-40190, dated November 27, 1985. A review was performed by the NSSS Vendor concerning these interactions and the results of the review transmitted via NEU-6039, dated December 23, 1985. This condition is identified as Case 3.3 in NES-40190 and evaluated on Page 2 of NEU-6039. The NSSS Vendor's conclusion is that Case 3.3 involves break propagation of a Small LOCA to other legs of the affected loop and that the break propagation areas are small compared to the break area of the ruptured line. Phenomena associated with the break at the original location are not affected by the propagation and calculated Emergency Core Cooling System (ECCS) performance will be basically the same as it would with no propagation. Therefore, calculated Emergency Core Cooling System (ECCS) performance conservatively bounds Case 3.3 in NES-40190 and the interactions are acceptable. Therefore, no protective hardware is required.

(2) The NSSS System Standard Design Criteria states for a Small LOCA that break propagation for an initiating break on the High Pressure Safety Injection Line is not permitted. Disc Pressurization Valves 3RCS*V984, 3RCS*V985, and 3RCS*V979 are targeted. Loss of pressure boundary is admitted. This interaction was transmitted to the NSSS Vendor via NES-40190, dated November 27, 1985. A review was performed by the NSSS Vendor concerning this interaction and the results are transmitted via NEU-6039, dated December 23, 1985. This condition is identified as Case 3.3 in NES-40190 and evaluated on Page 2 of NEU-6039. The NSSS Vendor's conclusion is that Case 3.3 involves break propagation of a Small LOCA to other legs of the affected loop. The break propagation areas are small compared to the break area of the ruptured line. Phenomena associated with the break at the original location are not affected by the propagation and calculated Emergency Core Cooling System (ECCS) performance will be basically the same as it would with no propagation. Therefore, calculated Emergency Core Cooling System (ECCS) performance conservatively bounds Case 3.3 in NES-40190 and the interaction is acceptable. Therefore, no protective hardware is required.

TABLE 3.6–20 JET IMPINGEMENT EFFECTS - LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement

(3) The NSSS System Standard Design Criteria states for a Large LOCA that damage to the Steam System be prevented. The 3-SSR-375-026-2 or 3-SSR-375-035-2 line is part of the Steam Generator Blowdown Sampling System. Loss of pressure boundary is admitted. This result was transmitted to the NSSS Vendor via NES-40190, dated November 27, 1985. A review was performed by the NSSS Vendor concerning this interaction and the results are transmitted via NEU-6039, dated December 23, 1985. This condition is identified as Case 2.1 in NES-40190 and evaluated on Page 1 of NEU-6039. The NSSS Vendor’s conclusion is that heat transfer to the secondary side is unimportant for 6: and larger LOCA breaks. For Millstone 3, the limiting case is the 4" equivalent diameter cold leg break which exhibits a 1400 second time interval between break inception and final core recovery. Over 1400 seconds only about 5% of the initial steam generator secondary side inventory would be lost through a Steam Generator Blowdown Sampling System break of this size. Moreover once Auxiliary Feedwater injection begins, following an “S” signal mass addition to the Steam Generator secondary will exceed the mass depletion through the broken Steam Generator Blowdown Sampling line. Since the Steam Generator inventory remains about the same Emergency Core Cooling System (ECCS) performance will be basically unaffected by the pressure boundary loss of the Steam Generator Blowdown Sampling System. Therefore this break is less limiting than the 4" equivalent diameter cold leg break and the interaction is acceptable. Therefore, no protective hardware is required.

(4) The High Pressure Safety Injection (SIH) System in the Auxiliary Building does not have sustained fluid jet impingement since the High Pressure Safety Injection (SIH) System in the Auxiliary Building connects to the Normal Charging Pumps which is not considered a constant pressure source (i.e., reservoir) to sustain system pressure subsequent to a pipe break on the SIH System based upon FSAR Section 3.6.2.2.1, Item 3. The portion between the Normally Closed Containment Isolation Valves, 3SIH*MV8801A and 3SIH*MV8801B and the Reactor Coolant System Pressure Boundary is normally empty.

(5) The High Pressure Safety Injection (SIH) System in the Auxiliary Building is a High Energy System since the piping is maintained pressurized during Normal Plant Operation between the Normal Charging Pumps and the Normally Closed Containment Isolation Valves, 3SIH*MV8801A and 3SIH*MV8801B. The High Pressure Safety Injection (SIH) System in the Auxiliary Building connects to the Normal Charging Pumps which is not considered a constant pressure source (i.e., reservoir) to sustain system pressure subsequent to a pipe break on the SIH System based upon FSAR Section 3.6.2.2.1, Item 3. The portion between the Normally Closed Containment Isolation Valves, 3SIH*MV8801A and 3SIH*MV8801B and the Reactor Coolant System Pressure Boundary is normally empty.

(6) USNRC Generic Letter 87-11 eliminates the requirement to evaluate the dynamic effects (i.e., fluid jet impingement) of Arbitrary Intermediate Breaks (AIBs). Therefore, fluid jet impingement effects subsequent to AIBs are not evaluated.

TABLE 3.6–21 POSTULATED BREAKS - CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING

Line Designation	Break # (1)	Building	Elevation	Break Type (2)	Total Additive Stress	Figure
3-CHS-003-662-2	10	Containment	-3'-6"	CB	N/A Terminal End	3.6–15
3-CHS-003-661-2	20	Containment	-3'-6"	CB	N/A Terminal End	3.6–15
3-CHS-003-662-2	21	Containment	-3'-6"	CB	N/A Arbitrary Intermediate	3.6–15
3-CHS-003-662-2	22	Containment	-3'-6"	CB	N/A Arbitrary Intermediate	3.6–15
3-CHS-003-662-2	23	Containment	-5'-10"	CB	N/A Terminal End	3.6–15
3-CHS-002-73-2	24	Containment	-10'-1"	CB	Above Threshold	3.6–15
3-CHS-002-73-2	25	Containment	-10'-1"	CB	N/A Terminal End	3.6–15
3-CHS-003-661-2	26	Containment	-2'-0"	CB	Above Threshold	3.6–15
3-CHS-003-661-2	27	Containment	-2'-0"	CB	N/A Arbitrary Intermediate	3.6–15
3-CHS-003-661-2	28	Containment	-2'-0"	CB	N/A Terminal End	3.6–15
3-CHS-003-661-2	29	Containment	-10'-2"	CB	Above Threshold	3.6–15
3-CHS-003-76-2	30	Containment	-9'-9"	CB	N/A Terminal End	3.6–15
3-CHS-003-72-2	31	Containment	-1'-5"	CB	N/A Terminal End	3.6–15
3-CHS-003-72-2	32	Containment	7'-6"	CB	N/A Arbitrary Intermediate	3.6–15
3-CHS-003-72-2	33	Containment	11'-3"	CB	N/A Arbitrary Intermediate	3.6–15
3-CHS-003-70-2	34	Auxiliary	12'-6"	CB	N/A Arbitrary Intermediate	3.6–15
3-CHS-003-70-2	35	Auxiliary	15'-4"	CB	N/A Arbitrary Intermediate	3.6–15
3-CHS-003-69-2	36	Auxiliary	16'-11"	CB	N/A Terminal End	3.6–15

TABLE 3.6–21 POSTULATED BREAKS - CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING

Line Designation	Break # (1)	Building	Elevation	Break Type (2)	Total Additive Stress	Figure
3-CHS-002-237-2	37	Auxiliary	18' - 8"	CB	N/A Terminal End	3.6–15
3-CHS-002-237-2	38	Auxiliary	20'-1"	CB	N/A Arbitrary Intermediate	3.6–15
3-CHS-002-237-2	39	Auxiliary	15'-10"	CB	N/A Arbitrary Intermediate	3.6–15
3-CHS-002-238-2	40	Auxiliary	12'-3"	CB	N/A Terminal End	3.6–15
3-CHS-002-238-2	41	Auxiliary	13'-0"	CB	N/A Arbitrary Intermediate	3.6–15
3-CHS-002-238-2	42	Auxiliary	18'-7"	CB	N/A Arbitrary Intermediate	3.6–15
3-CHS-003-72-2	43	Containment	12'-6"	CB	N/A Terminal End	3.6–15
3-CHS-003-72-2	44	Auxiliary	12'-6"	CB	N/A Terminal End	3.6–15
3-CHS-004-68-2	(3)	Auxiliary	Varies	CB and LS	N/A (3)	3.6–15
3-CHS-004-435-2	(3)	Auxiliary	Varies	CB and LS	N/A (3)	3.6–15
3-CHS-003-434-2	(3)	Auxiliary	Varies	CB	N/A (3)	3.6–15
3-CHS-004-28-2	(3)	Auxiliary	Varies	CB and LS	N/A (3)	3.6–15
3-CHS-004-30-2	(3)	Auxiliary	Varies	CB and LS	N/A (3)	3.6–15
3-CHS-004-29-2	(3)	Auxiliary	Varies	CB and LS	N/A (3)	3.6–15
3-CHS-003-65-2	(3)	Auxiliary	Varies	CB	N/A (3)	3.6–15
3-CHS-004-433-2	(3)	Auxiliary	Varies	CB and LS	N/A (3)	3.6–15

TABLE 3.6–21 POSTULATED BREAKS - CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING

Line Designation	Break # (1)	Building	Elevation	Break Type (2)	Total Additive Stress	Figure
3-CHS-004-682-2	(3)	Auxiliary	Varies	CB and LS	N/A (3)	3.6–15
3-CHS-004-678-2	(3)	Auxiliary	Varies	CB and LS	N/A (3)	3.6–15
3-CHS-004-677-2	(3)	Auxiliary	Varies	CB and LS	N/A (3)	3.6–15
3-CHS-004-679-2	(3)	Auxiliary	Varies	CB and LS	N/A (3)	3.6–15
3-CHS-003-67-2	(3)	Auxiliary	Varies	CB	N/A (3)	3.6–15
3-CHS-003-31-2	(3)	Auxiliary	Varies	CB	N/A (3)	3.6–15
3-CHS-003-32-2	(3)	Auxiliary	Varies	CB	N/A (3)	3.6–15
3-CHS-002-61-2	(3)	Auxiliary	Varies	CB	N/A (3)	3.6–15
3-CHS-002-59-2	(3)	Auxiliary	Varies	CB	N/A (3)	3.6–15
3-CHS-002-63-2	(3)	Auxiliary	Varies	CB	N/A (3)	3.6–15

NOTES:

(1) A portion of the Chemical & Volume Control System Normal Charging is part of the Reactor Coolant System. Breaks 1 through 9 and 11 through 19 are postulated on the Reactor Coolant System and are listed in FSAR Table 3.6–12.

(2) Circumferential Pipe Break (CB) and Longitudinal Pipe Split (LS) are defined in FSAR Section 3.6.2.1.3.

(3) Pipe breaks on the Discharge Lines of the Normal Charging System in each Charging Pump Cubicle are postulated at Terminal Ends (TPs), and at each Valve, Fitting, and any Integral Welded Attachment in accordance with the criteria delineated in FSAR Section 3.6.2.1.2.3.b.

TABLE 3.6–22 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-CHS-003-662-2	10	None (1)	None	None
3-CHS-003-661-2	20	None (1)	None	None
3-CHS-003-662-2	21	None (1)	None (2)	None
3-CHS-003-662-2	22	None (1)	None (2)	None
3-CHS-003-662-2	23	None (1)	None	None
3-CHS-002-73-2	24	None (1)	None	None
3-CHS-002-73-2	25	None (1)	None	None
3-CHS-003-661-2	26	None (1)	None	None
3-CHS-003-661-2	27	None (1)	None (2)	None
3-CHS-003-661-2	28	None (1)	None	None
3-CHS-003-661-2	29	None (1)	None	None
3-CHS-003-76-2	30	None (1)	None	None
3-CHS-003-72-2	31	None (1)	None	None
3-CHS-003-72-2	32	None (1)	None (2)	None
3-CHS-003-72-2	33	None (1)	None (2)	None
3-CHS-003-70-2	34	None (1)	None (2)	None

TABLE 3.6–22 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-CHS-003-70-2	35	None (1)	None (2)	None
3-CHS-003-69-2	36	None (1)	None	None
3-CHS-002-237-2	37	None (1)	None	None
3-CHS-002-237-2	38	None (1)	None (2)	None
3-CHS-002-237-2	39	None (1)	None (2)	None
3-CHS-002-238-2	40	None (1)	None	None
3-CHS-002-238-2	41	None (1)	None (2)	None
3-CHS-002-238-2	42	None (1)	None (2)	None
3-CHS-003-72-2	43	None (1)	None	None
3-CHS-003-72-2	44	None (1)	None	None
3-CHS-004-68-2	(3)	None (1)	None (4)	None
3-CHS-004-435-2	(3)	None (1)	None (4)	None
3-CHS-003-434-2	(3)	None (1)	None (4)	None
3-CHS-004-28-2	(3)	None (1)	None (4)	None
3-CHS-004-30-2	(3)	None (1)	None (4)	None
3-CHS-004-29-2	(3)	None (1)	None (4)	None

TABLE 3.6–22 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-CHS-003-65-2	(3)	None (1)	None (4)	None
3-CHS-004-433-2	(3)	None (1)	None (4)	None
3-CHS-004-682-2	(3)	None (1)	None (4)	None
3-CHS-004-678-2	(3)	None (1)	None (4)	None
3-CHS-004-677-2	(3)	None (1)	None (4)	None
3-CHS-004-679-2	(3)	None (1)	None (4)	None
3-CHS-003-67-2	(3)	None (1)	None (4)	None
3-CHS-003-31-2	(3)	None (1)	None (4)	None
3-CHS-003-32-2	(3)	None (1)	None (4)	None
3-CHS-002-61-2	(3)	None (1)	None (4)	None
3-CHS-002-59-2	(3)	None (1)	None (4)	None
3-CHS-002-63-2	(3)	None (1)	None (4)	None

TABLE 3.6–22 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement

NOTES:

(1) Sustained blowdown on the Chemical & Volume Control System Normal Charging Line does not occur. Dual inline check valves prevent backflow from the Reactor Coolant System and the Charging Pumps have limited capacity to sustain the system pressure. Therefore pipe whip does not occur.

(2) USNRC Generic Letter 87-11 eliminates the requirement to evaluate the pipe whip effects from Arbitrary Intermediate Breaks (AIBs). Therefore pipe whip effects for AIBs are not evaluated.

(3) Pipe breaks on the Discharge Lines of the Normal Charging System in each Charging Pump Cubicle are postulated at Terminal Ends (TPs), and at each Valve, Fitting, and any Integral Welded Attachments in accordance with the criteria delineated in FSAR Section 3.6.2.1.2.3.b.

(4) NERM-069 Revision 1, Attachment 5, Figure 10A shows the location of the Charging Pumps. NERM-069 Revision 1, Attachment 5, Figures 10A and 10B, show that Charging Pump 3CHS*P3A is located in Cubicle 092, Charging Pump 3CHS*P3B is located in Cubicle 093, and Charging Pump 3CHS*P3C is located in Cubicle 094. This layout demonstrates separation in accordance with FSAR Section 3.6.1.1.4.2. Therefore, protective hardware is not required as redundant safety-related systems, and components are separated from one another.

TABLE 3.6–23 JET IMPINGEMENT EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-CHS-003-662-2	10	None (1)	N/A	N/A	N/A	None
3-CHS-003-661-2	20	None (1)	N/A	N/A	N/A	None
3-CHS-003-662-2	21	None (1)	N/A	N/A	N/A	None (2)
3-CHS-003-662-2	22	None (1)	N/A	N/A	N/A	None (2)
3-CHS-003-662-2	23	None (1)	N/A	N/A	N/A	None
3-CHS-002-73-2	24	None (1)	N/A	N/A	N/A	None
3-CHS-002-73-2	25	None (1)	N/A	N/A	N/A	None
3-CHS-003-661-2	26	None (1)	N/A	N/A	N/A	None
3-CHS-003-661-2	27	None (1)	N/A	N/A	N/A	None (2)
3-CHS-003-661-2	28	None (1)	N/A	N/A	N/A	None
3-CHS-003-661-2	29	None (1)	N/A	N/A	N/A	None
3-CHS-003-76-2	30	None (1)	N/A	N/A	N/A	None
3-CHS-003-72-2	31	None (1)	N/A	N/A	N/A	None
3-CHS-003-72-2	32	None (1)	N/A	N/A	N/A	None (2)
3-CHS-003-72-2	33	None (1)	N/A	N/A	N/A	None (2)

TABLE 3.6–23 JET IMPINGEMENT EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-CHS-003-70-2	34	None (1)	N/A	N/A	N/A	None (2)
3-CHS-003-70-2	35	None (1)	N/A	N/A	N/A	None (2)
3-CHS-003-69-2	36	None (1)	N/A	N/A	N/A	None
3-CHS-002-237-2	37	None (1)	N/A	N/A	N/A	None
3-CHS-002-237-2	38	None (1)	N/A	N/A	N/A	None (2)
3-CHS-002-237-2	39	None (1)	N/A	N/A	N/A	None (2)
3-CHS-002-238-2	40	None (1)	N/A	N/A	N/A	None
3-CHS-002-238-2	41	None (1)	N/A	N/A	N/A	None (2)
3-CHS-002-238-2	42	None (1)	N/A	N/A	N/A	None (2)
3-CHS-003-72-2	43	None (1)	N/A	N/A	N/A	None
3-CHS-003-72-2	44	None (1)	N/A	N/A	N/A	None
3-CHS-004-68-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-004-435-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-003-434-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-004-28-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-004-30-2	(3)	None (1)	N/A	N/A	N/A	None

TABLE 3.6–23 JET IMPINGEMENT EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-CHS-004-29-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-003-65-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-004-433-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-004-682-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-004-678-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-004-677-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-004-679-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-003-67-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-003-31-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-003-32-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-002-61-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-002-59-2	(3)	None (1)	N/A	N/A	N/A	None
3-CHS-002-63-2	(3)	None (1)	N/A	N/A	N/A	None

TABLE 3.6–23 JET IMPINGEMENT EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement

NOTES:

(1) Sustained blowdown on the Chemical & Volume Control System Normal Charging Line does not occur. Dual inline check valves, 3RCS*V8378A and 3RCS*V8378B to Cold Leg 3-RCS-275-5-1 and 3RCS*V8379A and 3RCS*V8379B to Cold Leg 3-RCS-275-20-1 at the Chemical & Volume Control System/Reactor Coolant System (i.e., Class 1/Class 2) boundary prevent backflow from the Reactor Coolant System and the Charging Pumps have limited capacity and cannot sustain the system pressure. Since sustained blowdown does not occur, dynamic analyses for the effects of fluid jet impingement is not performed based upon FSAR Section 3.6.2.2.1, Item 3.

(2) USNRC Generic Letter 87-11 eliminates the need to evaluate jet impingement effects subsequent to Arbitrary Intermediate Breaks (AIBs). Therefore jet impingement effects are not evaluated for AIBs.

(3) Pipe breaks on the Discharge Lines of the Normal Charging System in each Charging Pump Cubicle are postulated at Terminal Ends (TPs), and at each Valve, Fitting, and any Integral Welded Attachments in accordance with the criteria delineated in FSAR Section 3.6.2.1.2.3.b.

TABLE 3.6–24 POSTULATED BREAKS - CHEMICAL VOLUME CONTROL SYSTEM - LETDOWN LINE

Line Designation	Break #[1]	Building	Elevation	Break Type [2]	Total Additive Stress	Figure
3-CHS-003-1-2	9	Containment	-9'-9"	CB	N/A Terminal End	3.6–16
3-CHS-003-416-2	10	Containment	-1'-5"	CB	N/A Terminal End	3.6–16
3-CHS-003-416-2	11	Containment	4'-7"	CB	Above Threshold	3.6–16
3-CHS-003-416-2	12	Containment	4'-7"	CB	Above Threshold	3.6–16
3-CHS-003-416-2	13	Containment	3'-7"	CB	Above Threshold	3.6–16
3-CHS-002-2-2	14	Containment	3'-3"	CB	Above Threshold	3.6–16
3-CHS-002-590-2	15	Containment	0'-8"	CB	N/A Terminal End	3.6–16
3-CHS-002-7-2	16	Containment	0'-8"	CB	N/A Terminal End	3.6–16
3-CHS-002-4-2	17	Containment	4'-4"	CB	Above Threshold	3.6–16
3-CHS-002-589-2	18	Containment	0'-8"	CB	N/A Terminal End	3.6–16
3-CHS-002-6-2	19	Containment	0'-8"	CB	N/A Terminal End	3.6–16
3-CHS-002-6-2	20	Containment	0'-6"	CB	Above Threshold	3.6–16
3-CHS-002-3-2	21	Containment	4'-4"	CB	Above Threshold	3.6–16
3-CHS-002-588-2	22	Containment	0'-8"	CB	N/A Terminal End	3.6–16
3-CHS-002-5-2	23	Containment	0'-8"	CB	N/A Terminal End	3.6–16
3-CHS-002-5-2	24	Containment	0'-6"	CB	Above Threshold	3.6–16
3-CHS-025-304-2	25	Containment	7'-8"	CB	N/A Terminal End	3.6–16
3-CHS-003-8-2	26	Auxiliary Bldg.	19'-9"	CB	N/A Arbitrary Intermediate	3.6–16

TABLE 3.6–24 POSTULATED BREAKS - CHEMICAL VOLUME CONTROL SYSTEM - LETDOWN LINE (CONTINUED)

Line Designation	Break #(1)	Building	Elevation	Break Type (2)	Total Additive Stress	Figure
3-CHS-003-8-2	27	Auxiliary Bldg.	20'-2"	CB	N/A Arbitrary Intermediate	3.6–16
3-CHS-003-8-2	28	Auxiliary Bldg.	11'-2"	CB	N/A Terminal End	3.6–16
3-CHS-003-8-2	29	Auxiliary Bldg.	12'-7"	CB	N/A Arbitrary Intermediate	3.6–16
3-CHS-003-78-2	30	Auxiliary Bldg.	32'-11"	CB	N/A Terminal End	3.6–16
3-CHS-003-77-2	31	Auxiliary Bldg.	32'-11"	CB	N/A Terminal End	3.6–16
3-CHS-003-9-2	32	Auxiliary Bldg.	22'-8"	CB	N/A Terminal End	3.6–16
3-CHS-002-81-2	33	Auxiliary Bldg.	17'-3"	CB	N/A Arbitrary Intermediate	3.6–16
3-CHS-002-81-2	34	Auxiliary Bldg.	16'-3"	CB	N/A Terminal End	3.6–16
3-CHS-002-80-2	35	Auxiliary Bldg.	6'-0"	CB	N/A Arbitrary Intermediate	3.6–16
3-CHS-002-80-2	36	Auxiliary Bldg.	6'-0"	CB	N/A Terminal End	3.6–16
3-CHS-002-80-2	37	Auxiliary Bldg.	17'-2"	CB	N/A Arbitrary Intermediate	3.6–16
3-CHS-002-80-2	38	Auxiliary Bldg.	20'-1"	CB	N/A Arbitrary Intermediate	3.6–16
3-CHS-002-80-2	39	Auxiliary Bldg.	20'-1"	CB	N/A Terminal End	3.6–16
3-CHS-002-80-2	40	Auxiliary Bldg.	12'-1"	CB	N/A Arbitrary Intermediate	3.6–16
3-CHS-002-80-2	41	Auxiliary Bldg.	11'-10"	CB	N/A Arbitrary Intermediate	3.6–16
3-CHS-002-80-2	42	Auxiliary Bldg.	-3'-8"	CB	N/A Terminal End	3.6–16
3-CHS-002-80-2	43	Auxiliary Bldg.	9'-10"	CB	Above Threshold	3.6–16
3-CHS-002-80-2	44	Auxiliary Bldg.	10'-1"	CB	Above Threshold	3.6–16

TABLE 3.6–24 POSTULATED BREAKS - CHEMICAL VOLUME CONTROL SYSTEM - LETDOWN LINE (CONTINUED)

Line Designation	Break #(1)	Building	Elevation	Break Type (2)	Total Additive Stress	Figure
3-CHS-002-80-2	45	Auxiliary Bldg.	13'-0"	CB	N/A Terminal End	3.6–16
3-CHS-002-80-2	46	Auxiliary Bldg.	13'-10"	CB	N/A Arbitrary Intermediate	3.6–16
3-CHS-002-80-2	47	Auxiliary Bldg.	14'-0"	CB	Above Threshold	3.6–16
3-CHS-002-80-2	48	Auxiliary Bldg.	14'-0"	CB	N/A Terminal End	3.6–16
3-CHS-002-80-2	49	Auxiliary Bldg.	14'-0"	CB	Above Threshold	3.6–16
3-CHS-002-80-2	50	Auxiliary Bldg.	14'-0"	CB	Above Threshold	3.6–16
3-CHS-002-80-2	51	Auxiliary Bldg.	14'-0"	CB	Above Threshold	3.6–16
3-CHS-002-80-2	52	Auxiliary Bldg.	13'-9"	CB	Above Threshold	3.6–16
3-CHS-002-80-2	53	Auxiliary Bldg.	12'-10"	CB	N/A Terminal End	3.6–16
3-CHS-003-8-2	54	Auxiliary Bldg.	12'-6"	CB	N/A Terminal End	3.6–16
3-CHS-003-659-2	55	Containment	12'-6"	CB	N/A Terminal End	3.6–16

NOTES:

(1) A portion of the Chemical & Volume Control System Normal Letdown Line is part of the Reactor Coolant System. Breaks 1 through 8 are postulated on the Reactor Coolant System portion of the Normal Letdown Line and are listed in FSAR Table 3.6–12.

(2) Circumferential Pipe Break (CB) is defined in FSAR Section 3.6.2.1.3.

TABLE 3.6–25 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - LETDOWN LINE

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-CHS-003-1-2	9	3-RCS-275-15-1	(2)	3CHS-PRR858 & 3RCS-PRR870
3-CHS-003-416-2	10	3-RCS-275-15-1	None	None
3-CHS-003-416-2	11	3-RCS-275-15-1	None	None
3-CHS-002-3-2	11	3-RCS-275-15-1	None	None
3-CHS-003-416-2	12	3-RCS-275-15-1	None	None
3-CHS-002-4-2	12	3-RCS-275-15-1	None	None
3-CHS-003-416-2	13	3-RCS-275-15-1	None	None
3-CHS-002-2-2	14	3-RCS-275-15-1	None	None
3-CHS-002-590-2	15	3-RCS-275-15-1	None	None
3-CHS-002-7-2	16	3-RCS-275-15-1	None	None
3-CHS-002-4-2	17	3-RCS-275-15-1	None	None
3-CHS-002-589-2	18	3-RCS-275-15-1	None	None
3-CHS-002-6-2	19	3-RCS-275-15-1	None	None
3-CHS-002-6-2	20	3-RCS-275-15-1	None	None
3-CHS-002-3-2	21	3-RCS-275-15-1	None	None
3-CHS-002-588-2	22	3-RCS-275-15-1	None	None
3-CHS-002-5-2	23	3-RCS-275-15-1	None	None
3-CHS-002-5-2	24	3-RCS-275-15-1	None	None
3-CHS-025-304-2	25	3-RCS-275-15-1	None	None
3-CHS-003-8-2	26	3-RCS-275-15-1	(3) (4)	3CHS-PRR967

TABLE 3.6–25 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - LETDOWN LINE (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-CHS-003-8-2	27	3-RCS-275-15-1	(4)	None
3-CHS-003-8-2	28	3-RCS-275-15-1	None	None
3-CHS-003-8-2	29	3-RCS-275-15-1	(4)	None
3-CHS-003-78-2	30	None	None	None
3-CHS-003-77-2	31	3-RCS-275-15-1	None	None
3-CHS-003-9-2	32	3-RCS-275-15-1	None	None
3-CHS-002-81-2	33	None	(4)	None
3-CHS-002-81-2	34	None	None	None
3-CHS-002-80-2	35	None	(4)	None
3-CHS-002-80-2	36	None	None	None
3-CHS-002-80-2	37	None	(4)	None
3-CHS-002-80-2	38	None	(4)	None
3-CHS-002-80-2	39	None	None	None
3-CHS-002-80-2	40	None	(4)	None
3-CHS-002-80-2	41	None	(4)	None
3-CHS-002-80-2	42	None	None	None
3-CHS-002-80-2	43	None	None	None
3-CHS-002-80-2	44	None	None	None
3-CHS-002-80-2	45	None	None	None
3-CHS-002-80-2	46	None	(4)	None

TABLE 3.6–25 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - LETDOWN LINE (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-CHS-002-80-2	47	None	None	None
3-CHS-002-80-2	48	None	None	None
3-CHS-002-80-2	49	None	None	None
3-CHS-002-80-2	50	None	None	None
3-CHS-002-80-2	51	None	None	None
3-CHS-002-80-2	52	None	None	None
3-CHS-002-80-2	53	None	None	None
3-CHS-003-8-2	54	3-RCS-275-15-1	None	None
3-CHS-003-659-2	55	3-RCS-275-15-1	None	None

NOTES:

(1) Repetition of Break Numbers is used to identify separate fluid reservoirs which provide a constant pressure source to maintain system pressure subsequent to the postulated pipe break. These reservoirs maintain system blowdown during the transient event as well as in the steady state.

(2) Pipe rupture restraints prevent propagation of a Non-LOCA into a LOCA. Restraints prevent excessive loads transmitted from the Class 2 line that is postulated to break to the Class 1 valve and piping.

(3) Pipe rupture restraint protects the Break Exclusion Zone for this system.

(4) USNRC Generic Letter 87-11 eliminates the need to evaluate the pipe whip effects of Arbitrary Intermediate Breaks (AIBs). Based upon this generic letter, pipe whip effects for these AIBs are not evaluated and no protection is necessary.

TABLE 3.6–26 JET IMPINGEMENT EFFECTS - CHEMICAL AND VOLUME CONTROL SYSTEM - LETDOWN LINE PIPING

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-CHS-003-1-2	9	None	N/A	N/A	N/A	None
3-CHS-003-416-2	10	None	N/A	N/A	N/A	None
3-CHS-003-416-2	11	None	N/A	N/A	N/A	None
3-CHS-003-416-2	12	None	N/A	N/A	N/A	None
3-CHS-003-416-2	13	None	N/A	N/A	N/A	None
3-CHS-002-2-2	14	None	N/A	N/A	N/A	None
3-CHS-002-590-2	15	None	N/A	N/A	N/A	None
3-CHS-002-7-2	16	None	N/A	N/A	N/A	None
3-CHS-002-4-2	17	None	N/A	N/A	N/A	None
3-CHS-002-589-2	18	None	N/A	N/A	N/A	None
3-CHS-002-6-2	19	None	N/A	N/A	N/A	None
3-CHS-002-6-2	20	None	N/A	N/A	N/A	None
3-CHS-002-3-2	21	None	N/A	N/A	N/A	None
3-CHS-002-588-2	22	None	N/A	N/A	N/A	None
3-CHS-002-5-2	23	None	N/A	N/A	N/A	None
3-CHS-002-5-2	24	None	N/A	N/A	N/A	None
3-CHS-025-304-2	25	None	N/A	N/A	N/A	None
3-CHS-003-8-2	26	None (1)	N/A	N/A	N/A	None
3-CHS-003-8-2	27	None (1)	N/A	N/A	N/A	None
3-CHS-003-8-2	28	None	N/A	N/A	N/A	None

TABLE 3.6–26 JET IMPINGEMENT EFFECTS - CHEMICAL AND VOLUME CONTROL SYSTEM - LETDOWN LINE PIPING

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-CHS-003-8-2	29	None [1]	N/A	N/A	N/A	None
3-CHS-003-78-2	30	None	N/A	N/A	N/A	None
3-CHS-003-77-2	31	None	N/A	N/A	N/A	None
3-CHS-003-9-2	32	None	N/A	N/A	N/A	None
3-CHS-002-81-2	33	None [1]	N/A	N/A	N/A	None
3-CHS-002-81-2	34	None	N/A	N/A	N/A	None
3-CHS-002-80-2	35	None [1]	N/A	N/A	N/A	None
3-CHS-002-80-2	36	None	N/A	N/A	N/A	None
3-CHS-002-80-2	37	None [1]	N/A	N/A	N/A	None
3-CHS-002-80-2	38	None [1]	N/A	N/A	N/A	None
3-CHS-002-80-2	39	None	N/A	N/A	N/A	None
3-CHS-002-80-2	40	None [1]	N/A	N/A	N/A	None
3-CHS-002-80-2	41	None [1]	N/A	N/A	N/A	None
3-CHS-002-80-2	42	None	N/A	N/A	N/A	None
3-CHS-002-80-2	43	None	N/A	N/A	N/A	None
3-CHS-002-80-2	44	None	N/A	N/A	N/A	None
3-CHS-002-80-2	45	None	N/A	N/A	N/A	None
3-CHS-002-80-2	46	None [1]	N/A	N/A	N/A	None
3-CHS-002-80-2	47	None	N/A	N/A	N/A	None

TABLE 3.6–26 JET IMPINGEMENT EFFECTS - CHEMICAL AND VOLUME CONTROL SYSTEM - LETDOWN LINE PIPING

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-CHS-002-80-2	48	None	N/A	N/A	N/A	None
3-CHS-002-80-2	49	None	N/A	N/A	N/A	None
3-CHS-002-80-2	50	None	N/A	N/A	N/A	None
3-CHS-002-80-2	51	None	N/A	N/A	N/A	None
3-CHS-002-80-2	52	None	N/A	N/A	N/A	None
3-CHS-002-80-2	53	None	N/A	N/A	N/A	None
3-CHS-003-8-2	54	None	N/A	N/A	N/A	None
3-CHS-003-659-2	55	None	N/A	N/A	N/A	None

NOTE:

(1) USNRC Generic Letter 87-11 eliminates the requirement to evaluate jet impingement effects for Arbitrary Intermediate Breaks (AIBs). Therefore, jet impingement effects subsequent to AIBs are not evaluated.

TABLE 3.6–27 POSTULATED BREAKS - SEAL WATER INJECTION SYSTEM

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure
3-CHS-003-32-2	1	Auxiliary Bldg.	27'-3"	CB	N/A Terminal End	3.6–17
3-CHS-003-32-2	2	Auxiliary Bldg.	18'-5"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-003-32-2	3	Auxiliary Bldg.	15'-2"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-003-32-2	4	Auxiliary Bldg.	19'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-446-2	5	Auxiliary Bldg.	21'-8"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-446-2	6	Auxiliary Bldg.	22'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-34-2	7	Auxiliary Bldg.	34'-11"	CB	N/A Terminal End	3.6–17
3-CHS-002-34-2	8	Auxiliary Bldg.	36'-4"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-34-2	9	Auxiliary Bldg.	36'-4"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-34-2	10	Auxiliary Bldg.	36'-4"	CB	N/A Terminal End	3.6–17
3-CHS-002-33-2	11	Auxiliary Bldg.	34'-3"	CB	N/A Terminal End	3.6–17
3-CHS-002-33-2	12	Auxiliary Bldg.	36'-4"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-33-2	13	Auxiliary Bldg.	36'-4"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-33-2	14	Auxiliary Bldg.	36'-4"	CB	N/A Terminal End	3.6–17
3-CHS-002-447-2	15	Auxiliary Bldg.	36'-4"	CB	N/A Terminal End	3.6–17
3-CHS-002-445-2	16	Auxiliary Bldg.	36'-4"	CB	N/A Terminal End	3.6–17
3-CHS-003-38-2	17	Auxiliary Bldg.	18'-0"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-003-38-2	18	Auxiliary Bldg.	18'-5"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-39-2	19	Auxiliary Bldg.	20'-1"	CB	N/A Terminal End	3.6–17
3-CHS-002-41-2	20	Auxiliary Bldg.	12'-9"	CB	N/A Arbitrary Intermediate	3.6–17

TABLE 3.6–27 POSTULATED BREAKS - SEAL WATER INJECTION SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Total Additive Stress	Figure
3-CHS-002-42-2	21	Auxiliary Bldg.	12'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-638-2	22	Containment	11'-8"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-638-2	23	Containment	12'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-638-2	24	Containment	12'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-638-2	25	Containment	9'-3"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-638-2	26	Containment	12'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-638-2	27	Containment	12'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-638-2	28	Containment	20'-3"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-638-2	29	Containment	20'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-003-38-2	33	Auxiliary Bldg.	18'-5"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-44-2	34	Auxiliary Bldg.	20'-0"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-44-2	35	Auxiliary Bldg.	18'-11"	CB	N/A Terminal End	3.6–17
3-CHS-002-46-2	36	Auxiliary Bldg.	12'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-637-2	37	Containment	11'-6"	CB	Above Threshold	3.6–17
3-CHS-002-637-2	38	Containment	11'-9"	CB	Above Threshold	3.6–17
3-CHS-003-38-2	41	Auxiliary Bldg.	18'-5"	CB	N/A Terminal End	3.6–17
3-CHS-002-51-2	42	Auxiliary Bldg.	13'-4"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-51-2	43	Auxiliary Bldg.	12'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-52-2	44	Containment	11'-8"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-651-2	45	Containment	12'-6"	CB	N/A Arbitrary Intermediate	3.6–17

TABLE 3.6–27 POSTULATED BREAKS - SEAL WATER INJECTION SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure
3-CHS-002-651-2	46	Containment	12'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-651-2	47	Containment	12'-9"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-651-2	48	Containment	20'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-54-2	52	Auxiliary	19'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-56-2	53	Auxiliary	12'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-57-2	54	Containment	12'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-650-2	55	Containment	11'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-650-2	56	Containment	11'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-650-2	57	Containment	11'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-650-2	58	Containment	11'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-650-2	59	Containment	11'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-650-2	60	Containment	13'-7"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-650-2	61	Containment	19'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-43-1	30	Containment	20'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-43-1	31	Containment	20'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-150-689-1	32	Containment	20'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-48-1	39	Containment	20'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-150-688-1	40	Containment	20'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-53-1	49	Containment	20'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-53-1	50	Containment	20'-6"	CB	N/A Arbitrary Intermediate	3.6–17

TABLE 3.6–27 POSTULATED BREAKS - SEAL WATER INJECTION SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure
3-CHS-150-687-1	51	Containment	20'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-58-1	62	Containment	20'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-002-58-1	63	Containment	20'-6"	CB	N/A Arbitrary Intermediate	3.6–17
3-CHS-150-686-1	64	Containment	20'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-42-2	65	Auxiliary	12'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-42-2	66	Containment	12'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-47-2	67	Auxiliary	12'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-47-2	68	Containment	12'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-57-2	69	Auxiliary	12'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-57-2	70	Containment	12'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-52-2	71	Auxiliary	12'-6"	CB	N/A Terminal End	3.6–17
3-CHS-002-52-2	72	Containment	12'-6"	CB	N/A Terminal End	3.6–17

NOTE:

(1) Circumferential Pipe Break (CB) is defined in FSAR Section 3.6.2.1.3.

TABLE 3.6–28 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - SEAL WATER INJECTION LINE

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-CHS-003-32-2	1	None (1)	None	None
3-CHS-003-32-2	2	None (1)	None	None (2)
3-CHS-003-32-2	3	None (1)	None	None (2)
3-CHS-003-32-2	4	None (1)	None	None
3-CHS-002-446-2	5	None (1)	None	None (2)
3-CHS-002-446-2	6	None (1)	None	None (2)
3-CHS-002-34-2	7	None (1)	None	None
3-CHS-002-34-2	8	None (1)	None	None (2)
3-CHS-002-34-2	9	None (1)	None	None (2)
3-CHS-002-34-2	10	None (1)	None	None
3-CHS-002-33-2	11	None (1)	None	None
3-CHS-002-33-2	12	None (1)	None	None (2)
3-CHS-002-33-2	13	None (1)	None	None (2)
3-CHS-002-33-2	14	None (1)	None	None
3-CHS-002-447-2	15	None (1)	None	None
3-CHS-002-445-2	16	None (1)	None	None

TABLE 3.6–28 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - SEAL WATER INJECTION LINE

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-CHS-003-38-2	17	None (1)	None	None (2)
3-CHS-003-38-2	18	None (1)	None	None (2)
3-CHS-002-39-2	19	None (1)	None	None
3-CHS-002-41-2	20	None (1)	None	None (2)
3-CHS-002-42-2	21	None (1)	None	None (2)
3-CHS-002-638-2	22	None (1)	None	None (2)
3-CHS-002-638-2	23	None (1)	None	None (2)
3-CHS-002-638-2	24	None (1)	None	None
3-CHS-002-638-2	25	None (1)	None	None (2)
3-CHS-002-638-2	26	None (1)	None	None (2)
3-CHS-002-638-2	27	None (1)	None	None
3-CHS-002-638-2	28	None (1)	None	None (2)
3-CHS-002-638-2	29	None (1)	None	None (2)
3-CHS-003-38-2	33	None (1)	None	None (2)
3-CHS-002-44-2	34	None (1)	None	None (2)
3-CHS-002-44-2	35	None (1)	None	None

TABLE 3.6–28 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - SEAL WATER INJECTION LINE

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-CHS-002-46-2	36	None (1)	None	None (2)
3-CHS-002-637-2	37	None (1)	None	None
3-CHS-002-637-2	38	None (1)	None	None
3-CHS-003-38-2	41	None (1)	None	None
3-CHS-002-51-2	42	None (1)	None	None (2)
3-CHS-002-51-2	43	None (1)	None	None (2)
3-CHS-002-52-2	44	None (1)	None	None (2)
3-CHS-002-651-2	45	None (1)	None	None (2)
3-CHS-002-651-2	46	None (1)	None	None
3-CHS-002-651-2	47	None (1)	None	None (2)
3-CHS-002-651-2	48	None (1)	None	None (2)
3-CHS-002-54-2	52	None (1)	None	None
3-CHS-002-56-2	53	None (1)	None	None (2)
3-CHS-002-57-2	54	None (1)	None	None (2)
3-CHS-002-650-2	55	None (1)	None	None (2)
3-CHS-002-650-2	56	None (1)	None	None

TABLE 3.6–28 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - SEAL WATER INJECTION LINE

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-CHS-002-650-2	57	None (1)	None	None (2)
3-CHS-002-650-2	58	None (1)	None	None (2)
3-CHS-002-650-2	59	None (1)	None	None
3-CHS-002-650-2	60	None (1)	None	None (2)
3-CHS-002-650-2	61	None (1)	None	None (2)
3-CHS-002-43-1	30	None (1)	None	None (2)
3-CHS-002-43-1	31	None (1)	None	None (2)
3-CHS-150-689-1	32	None (1)	None	None
3-CHS-002-48-1	39	None (1)	None	None (2)
3-CHS-150-688-1	40	None (1)	None	None
3-CHS-002-53-1	49	None (1)	None	None (2)
3-CHS-002-53-1	50	None (1)	None	None (2)
3-CHS-150-687-1	51	None (1)	None	None
3-CHS-002-58-1	62	None (1)	None	None (2)
3-CHS-002-58-1	63	None (1)	None	None (2)
3-CHS-150-686-1	64	None (1)	None	None

TABLE 3.6–28 PIPE WHIP EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - SEAL WATER INJECTION LINE

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-CHS-002-42-2	65	None [(1)]	None	None
3-CHS-002-42-2	66	None [(1)]	None	None
3-CHS-002-47-2	67	None [(1)]	None	None
3-CHS-002-47-2	68	None [(1)]	None	None
3-CHS-002-57-2	69	None [(1)]	None	None
3-CHS-002-57-2	70	None [(1)]	None	None
3-CHS-002-52-2	71	None [(1)]	None	None
3-CHS-002-52-2	72	None [(1)]	None	None

NOTES:

(1) Sustained blowdown of the Chemical & Volume Control System - Seal Water Injection Line does not occur. The Seal Water Injection Line connects the Charging Pump to the Reactor Coolant pump. The limited capacity of the pumps does not sustain the system pressure; therefore, pipe whip does not occur for the Seal Water Injection Line and is not evaluated in accordance with FSAR Section 3.6.2.2.1, Item 3.

(2) USNRC Generic Letter 87-11 eliminates the requirement to evaluate pipe whip for Arbitrary Intermediate Breaks (AIBs) therefore pipe whip is not evaluated for AIBs.

TABLE 3.6–29 JET IMPINGEMENT EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - SEAL WATER INJECTION LINE

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-CHS-003-32-2	1	None - See Note 1	N/A	N/A	N/A	None
3-CHS-003-32-2	2	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-003-32-2	3	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-003-32-2	4	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-446-2	5	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-446-2	6	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-34-2	7	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-34-2	8	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-34-2	9	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-34-2	10	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-33-2	11	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-33-2	12	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-33-2	13	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-33-2	14	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-447-2	15	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-445-2	16	None - See Note 1	N/A	N/A	N/A	None
3-CHS-003-38-2	17	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-003-38-2	18	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-39-2	19	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-41-2	20	None - See Note 1	N/A	N/A	N/A	None - See Note 2

TABLE 3.6–29 JET IMPINGEMENT EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - SEAL WATER INJECTION LINE (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-CHS-002-42-2	21	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-638-2	22	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-638-2	23	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-638-2	24	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-638-2	25	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-638-2	26	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-638-2	27	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-638-2	28	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-638-2	29	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-003-38-2	33	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-44-2	34	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-44-2	35	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-46-2	36	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-637-2	37	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-637-2	38	None - See Note 1	N/A	N/A	N/A	None
3-CHS-003-38-2	41	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-51-2	42	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-51-2	43	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-52-2	44	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-651-2	45	None - See Note 1	N/A	N/A	N/A	None - See Note 2

TABLE 3.6–29 JET IMPINGEMENT EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - SEAL WATER INJECTION LINE (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-CHS-002-651-2	46	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-651-2	47	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-651-2	48	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-54-2	52	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-56-2	53	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-57-2	54	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-650-2	55	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-650-2	56	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-650-2	57	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-650-2	58	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-650-2	59	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-650-2	60	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-650-2	61	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-43-1	30	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-43-1	31	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-150-689-1	32	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-48-1	39	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-150-688-1	40	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-53-1	49	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-53-1	50	None - See Note 1	N/A	N/A	N/A	None - See Note 2

TABLE 3.6–29 JET IMPINGEMENT EFFECTS - CHEMICAL VOLUME CONTROL SYSTEM - SEAL WATER INJECTION LINE (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-CHS-150-687-1	51	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-58-1	62	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-002-58-1	63	None - See Note 1	N/A	N/A	N/A	None - See Note 2
3-CHS-150-686-1	64	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-42-2	65	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-42-2	66	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-47-2	67	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-47-2	68	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-57-2	69	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-57-2	70	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-52-2	71	None - See Note 1	N/A	N/A	N/A	None
3-CHS-002-52-2	72	None - See Note 1	N/A	N/A	N/A	None

NOTES:

(1) Sustained blowdown does not occur on the Chemical & Volume Control System Seal Water Injection Line. The Seal Water Injection Line connects the Charging Pump to each Reactor Coolant Pump. In each case, the pump has limited capacity to sustain the system pressure. Therefore, jet impingement effects are not sustained and are not evaluated as stated in FSAR Section 3.6.2.2.1 Item 3.

TABLE 3.6–30 POSTULATED BREAKS - STEAM GENERATOR BLOWDOWN SYSTEM

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure
3-BDG-002-45-2	1	Containment	24'-7"	CB	N/A Terminal End	3.6–18
3-BDG-004-9-2	2	Containment	21'-1"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-79-2	3	Containment	20'-4"	CB	N/A Terminal End	3.6–18
3-BDG-004-9-2	4	Containment	22'-7"	CB & LS	Above Threshold	3.6–18
3-BDG-004-17-4	5	MSVB	20'-0"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-17-4	6	MSVB	19'-6"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-17-4	7	MSVB	19'-6"	CB	N/A Terminal End	3.6–18
3-BDG-002-44-2	8	Containment	24'-7"	CB	N/A Terminal End	3.6–18
3-BDG-002-44-2	9	Containment	24'-3"	CB	Above Threshold	3.6–18
3-BDG-004-10-2	10	Containment	20'-6"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-81-2	11	Containment	20'-5"	CB	N/A Terminal End	3.6–18
3-BDG-004-18-4	12	MSVB	20'-0"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-18-4	13	MSVB	19'-6"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-18-4	14	MSVB	19'-6"	CB	N/A Terminal End	3.6–18
3-BDG-002-46-2	15	Containment	24'-7"	CB	N/A Terminal End	3.6–18
3-BDG-002-46-2	16	Containment	24'-2"	CB	Above Threshold	3.6–18
3-BDG-004-11-2	17	Containment	21'-1"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-83-2	18	Containment	20'-3"	CB	N/A Terminal End	3.6–18
3-BDG-004-19-4	19	MSVB	20'-0"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-19-4	20	MSVB	19'-6"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-19-4	21	MSVB	19'-6"	CB	N/A Terminal End	3.6–18
3-BDG-002-47-2	22	Containment	24'-7"	CB	N/A Terminal End	3.6–18

TABLE 3.6–30 POSTULATED BREAKS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure
3-BDG-004-85-2	23	Containment	19'-9"	CB	N/A Terminal End	3.6–18
3-BDG-004-12-2	24	Containment	16'-6"	CB & LS	Above Threshold	3.6–18
3-BDG-004-12-2	25	Containment	20'-0"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-20-4	26	MSVB	20'-0"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-20-4	27	MSVB	19'-6"	CB	N/A Arbitrary Intermediate	3.6–18
3-BDG-004-20-4	28	MSVB	19'-5"	CB	N/A Terminal End	3.6–18
3-BDG-004-17-4	29	MSVB	20'-6"	CB	N/A Terminal End	3.6–18
3-BDG-004-18-4	30	MSVB	20'-6"	CB	N/A Terminal End	3.6–18
3-BDG-004-19-4	31	MSVB	20'-6"	CB	N/A Terminal End	3.6–18
3-BDG-004-20-4	32	MSVB	20'-6"	CB	N/A Terminal End	3.6–18
3-BDG-004-78-2	33	Containment	72'-4"	CB	N/A Terminal End	3.6–18
3-BDG-004-78-2	34	Containment	72'-4"	CB	N/A - Arbitrary Intermediate	3.6–18
2" LINE TO SG1A	35	Containment	69'-1"	CB	N/A - Arbitrary Intermediate	3.6–18
2" LINE TO SG1A	36	Containment	68'-10"	CB	N/A Terminal End	3.6–18
3-BDG-004-80-2	37	Containment	74'-7"	CB	N/A Terminal End	3.6–18
2" LINE TO SG1B	38	Containment	69'-1"	CB	Above Threshold	3.6–18
2" LINE TO SG1B	39	Containment	68'-11"	CB	Above Threshold	3.6–18
2" LINE TO SG1B	40	Containment	68'-8"	CB	N/A Terminal End	3.6–18
3-BDG-004-82-2	41	Containment	74'-8"	CB	N/A Terminal End	3.6–18
3-BDG-004-82-2	42	Containment	74'-8"	CB	N/A - Arbitrary Intermediate	3.6–18
2" LINE TO SG1C	43	Containment	69'-7"	CB	N/A - Arbitrary Intermediate	3.6–18
2" LINE TO SG1C	44	Containment	69'-4"	CB	N/A Terminal End	3.6–18

TABLE 3.6–30 POSTULATED BREAKS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure
3-BDG-004-84-2	45	Containment	72'-4"	CB	N/A Terminal End	3.6–18
2" LINE TO SG1D	46	Containment	69'-2"	CB	Above Threshold	3.6–18
2" LINE TO SG1D	47	Containment	68'-11"	CB	Above Threshold	3.6–18
2" LINE TO SG1D	48	Containment	68'-8"	CB	N/A Terminal End	3.6–18
3-BDG-004-55-2	49	Containment	20'-6"	CB	N/A Terminal End	3.6–18
3-BDG-004-54-2	50	Containment	20'-6"	CB	N/A Terminal End	3.6–18
3-BDG-004-56-2	51	Containment	20'-6"	CB	N/A Terminal End	3.6–18
3-BDG-004-57-2	52	Containment	20'-6"	CB	N/A Terminal End	3.6–18
3-BDG-004-17-4	IP	MSVB	38' to 56'	CB & LS	N/A [2]	3.6–18
3-BDG-004-18-4	IP	MSVB	38' to 56'	CB & LS	N/A [2]	3.6–18
3-BDG-004-19-4	IP	MSVB	38' to 56'	CB & LS	N/A [2]	3.6–18
3-BDG-004-20-4	IP	MSVB	38' to 56'	CB & LS	N/A [2]	3.6–18
3-BDG-004-21-4	IP	MSVB & Turbine	56'-0"	CB & LS	N/A [2]	3.6–18
3-BDG-004-22-4	IP	MSVB & Turbine	56'-0"	CB & LS	N/A [2]	3.6–18
3-BDG-004-23-4	IP	MSVB & Turbine	56'-0"	CB & LS	N/A [2]	3.6–18
3-BDG-004-24-4	IP	MSVB & Turbine	56'-0"	CB & LS	N/A [2]	3.6–18
3-BDG-003-49-4	IP	MSVB	19'-1"	CB	N/A [2]	3.6–18
3-BDG-003-48-4	IP	MSVB	19'-2"	CB	N/A [2]	3.6–18
3-BDG-003-50-4	IP	MSVB	18'-10"	CB	N/A [2]	3.6–18
3-BDG-003-51-4	IP	MSVB	18'-9"	CB	N/A [2]	3.6–18

TABLE 3.6–30 POSTULATED BREAKS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure
3-BDG-003-49-4	TP	MSVB	17'-3"	CB	N/A [3]	3.6–18
3-BDG-003-48-4	TP	MSVB	17'-3"	CB	N/A [3]	3.6–18
3-BDG-003-50-4	TP	MSVB	16'-11"	CB	N/A [3]	3.6–18
3-BDG-003-51-4	TP	MSVB	16'-10"	CB	N/A [3]	3.6–18
3-BDG-008-29-4	None	Turbine	56'-0"	None	N/A [4]	3.6–18

NOTES:

(1) Circumferential Pipe Break (CB) and Longitudinal Pipe Split (LS) are defined in FSAR Section 3.6.2.1.3.

(2) Pipe breaks are postulated on the Nonnuclear (i.e., Nonsafety and Nonseismic) portion of the Steam Generator Blowdown System in the Main Steam Valve Building at all valves, fittings, and integral welded attachments. Use of a fitting criteria is in accordance with FSAR Section 3.6.2.1.2.3.2.a. Break Number Designation IP is for an Intermediate Break Location and the Break Number Designation TP is a Terminal Point.

(3) Terminal End (TP) pipe breaks are postulated at the normally closed valves 3BDG-V978, 3BDG-V979, 3BDG-V977, and 3BDG-V976 in accordance with FSAR Section 3.6.2.1.2.3.1.

(4) The Steam Generator Blowdown Tank Manifold, 3-BDG-008-29-4, is located in the Turbine Building. No safety-related structures, systems, or components are located in the Turbine Building. Steam Generator Blowdown Tank Manifold, 3-BDG-008-29-4 is therefore remote from essential structures, system, or component. Postulated pipe break on the Steam Generator Blowdown Tank Manifold, 3-BDG-008-29-4 is not assessed.

TABLE 3.6–31 PIPE WHIP EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement	Figure
3-BDG-002-45-2	1	3-BDG-008-29-4	Steam Generator 3RCS*SG1A Shell	3BDG-PRR975A	3.6–18
3-BDG-004-9-2	2	3-BDG-008-29-4	East-West Concrete Wall	None	3.6–18
3-BDG-004-9-2	2	3RCS*SG1A	Steam Generator 3RCS*SG1A Shell	3BDG-PRR972A	3.6–18
3-BDG-004-79-2	3	3RCS*SG1A	None	None	3.6–18
3-BDG-004-9-2	4	3RCS*SG1A	None	None	3.6–18
3-BDG-004-9-2	4	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-17-4	5	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-17-4	5	3RCS*SG1A	Limit Stress to Break Exclusion Zone	3BDG-PRR886	3.6–18
3-BDG-004-17-4	6	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-17-4	6	3RCS*SG1A	Limit Stress to Break Exclusion Zone	3BDG-PRR941	3.6–18
3-BDG-004-17-4	7	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-17-4	7	3RCS*SG1A	Limit Stress to Break Exclusion Zone	3BDG-PRR948	3.6–18
3-BDG-004-17-4	29	3RCS*SG1A	Limit Stress to Break Exclusion Zone	3BDG-PRR885	3.6–18
3-BDG-004-17-4	29	3-BDG-008-29-4	None	None	3.6–18
3-BDG-002-44-2	8	3-BDG-008-29-4	Steam Generator 3RCS*SG1B Shell	3BDG-PRR975B	3.6–18
3-BDG-002-44-2	9	3-BDG-008-29-4	Steam Generator 3RCS*SG1B Shell	3BDG-PRR975B	3.6–18
3-BDG-002-44-2	9	3RCS*SG1B	None	None	3.6–18

TABLE 3.6–31 PIPE WHIP EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement	Figure
3-BDG-004-10-2	10	3RCS*SG1B	Crane Wall Thimble 18-1	None	3.6–18
3-BDG-004-10-2	10	3-BDG-008-29-4	Prevent Pipe Whip into Containment Liner	3BDG-PRR973	3.6–18
3-BDG-004-81-2	11	3RCS*SG1B	None	None	3.6–18
3-BDG-004-18-4	12	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-18-4	12	3RCS*SG1B	Limit Stress to Break Exclusion Zone	3BDG-PRR888	3.6–18
3-BDG-004-18-4	13	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-18-4	13	3RCS*SG1B	Limit Stress to Break Exclusion Zone	3BDG-PRR942	3.6–18
3-BDG-004-18-4	14	3RCS*SG1B	Limit Stress to Break Exclusion Zone	3BDG-PRR950	3.6–18
3-BDG-004-18-4	14	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-18-4	30	3RCS*SG1B	Limit Stress to Break Exclusion Zone	3BDG-PRR887	3.6–18
3-BDG-004-18-4	30	3-BDG-008-29-4	None	None	3.6–18
3-BDG-002-46-2	15	3-BDG-008-29-4	Steam Generator 3RCS*SG1C Shell	3BDG-PRR975C	3.6–18
3-BDG-002-46-2	16	3RCS*SG1C	None	None	3.6–18
3-BDG-002-46-2	16	3-BDG-008-29-4	Steam Generator 3RCS*SG1C Shell	3BDG-PRR975C	3.6–18
3-BDG-004-11-2	17	3RCS*SG1C	Steam Generator 3RCS*SG1C Shell	3BDG-PRR972C	3.6–18
3-BDG-004-11-2	17	3-BDG-008-29-4	East-West Concrete Wall	None	3.6–18
3-BDG-004-83-2	18	3RCS*SG1C	None	None	3.6–18

TABLE 3.6–31 PIPE WHIP EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement	Figure
3-BDG-004-19-4	19	3RCS*SG1C	Limit Stress to Break Exclusion Zone	3BDG-PRR883	3.6–18
3-BDG-004-19-4	19	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-19-4	20	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-19-4	20	3RCS*SG1C	Limit Stress to Break Exclusion Zone	3BDG-PRR943	3.6–18
3-BDG-004-19-4	21	3RCS*SG1C	Limit Stress to Break Exclusion Zone	3BDG-PRR947	3.6–18
3-BDG-004-19-4	21	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-19-4	31	3RCS*SG1C	Limit Stress to Break Exclusion Zone	3BDG-PRR884	3.6–18
3-BDG-004-19-4	31	3-BDG-008-29-4	None	None	3.6–18
3-BDG-002-47-2	22	3-BDG-008-29-4	Steam Generator 3RCS*SG1D Shell	3BDG-PRR975D	3.6–18
3-BDG-004-85-2	23	3RCS*SG1D	None	None	3.6–18
3-BDG-004-12-2	24	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-12-2	24	3RCS*SG1D	None	None	3.6–18
3-BDG-004-12-2	25	3-BDG-008-29-4	Prevent Pipe Whip into Containment Liner	3BDG-PRR974	3.6–18
3-BDG-004-12-2	25	3RCS*SG1D	None	None	3.6–18
3-BDG-004-20-4	26	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-20-4	26	3RCS*SG1D	Limit Stress to Break Exclusion Zone	3BDG-PRR881	3.6–18
3-BDG-004-20-4	27	3-BDG-008-29-4	None	None	3.6–18

TABLE 3.6–31 PIPE WHIP EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break # (1)	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement	Figure
3-BDG-004-20-4	27	3RCS*SG1D	Limit Stress to Break Exclusion Zone	3BDG-PRR944	3.6–18
3-BDG-004-20-4	28	3RCS*SG1D	Limit Stress to Break Exclusion Zone	3BDG-PRR949	3.6–18
3-BDG-004-20-4	28	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-20-4	32	3RCS*SG1D	Limit Stress to Break Exclusion Zone	3BDG-PRR882	3.6–18
3-BDG-004-20-4	32	3-BDG-008-29-4	None	None	3.6–18
3-BDG-004-55-2	49	3RCS*SG1A	None	None	3.6–18
3-BDG-004-54-2	50	3RCS*SG1B	None	None	3.6–18
3-BDG-004-56-2	51	3RCS*SG1C	Prevent Pipe Whip into Steam Generator Support	3BDG-PRR970	3.6–18
3-BDG-004-57-2	52	3RCS*SG1D	None	None	3.6–18

TABLE 3.6–31 PIPE WHIP EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break # [1]	Blowdown Source	Essential Pipe Whip Target	Primary Protection Requirement	Figure

NOTES:

(1) Repetition of Break Numbers is used to identify separate fluid reservoirs which provide a constant pressure source to maintain system pressure subsequent to the postulated pipe break. These reservoirs maintain system blowdown during the transient event as well as in the steady state. The Steam Generator Blowdown Line runs from each Steam Generator Inside Containment to a common header 3-BDG-008-29-4 in the Turbine Building and Eventually to the Steam Generator Blowdown Tank 3BDG*TK1 in the Turbine Building. The common header is assumed to be a constant pressure source. The Steam Generator Blowdown Tank is low pressure (i.e., Relief Valve 3BDG-RV30 set at 75 psig) and cannot sustain the system pressure.

(2) Steam Generator Blowdown (BDG) System - Wet Lay Up Piping is a High Energy System from each connection to the Steam Generators to the Normally Closed Valves 3BDG*V887, 3BDG*V889, 3BDG*V891, and 3BDG*V893. The West Lay Up Piping is separated from adjacent Steam Generators so any pipe whip in one Steam Generator Cubicle does not damage essential structures, systems, or components associated with an adjacent Steam Generator based upon the Millstone Unit 3 Hazards Review Program Summary, NERM-069 Revision 1. The Wet Lay Up Piping is above the Floor Slab at Elevation 51'-4" which provides separation between the Reactor Coolant Pump Cubicles and the Upper Steam Generator Cubicles. Consequently pipe whip effects on the Wet Lay Up Piping cannot propagate into a Loss of Coolant Accident (LOCA) based upon separation documented in the Millstone Unit 3 Hazards Review Program Summary, NERM-069, Revision 1.

TABLE 3.6–32 JET IMPINGEMENT EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-BDG-002-45-2	1	None	N/A	N/A	N/A	None
3-BDG-004-9-2	2	None (2)	N/A	N/A	N/A	None
3-BDG-004-9-2	2	None (2)	N/A	N/A	N/A	None
3-BDG-004-79-2	3	None	N/A	N/A	N/A	None
3-BDG-004-9-2	4	None	N/A	N/A	N/A	None
3-BDG-004-9-2	4	None	N/A	N/A	N/A	None
3-BDG-004-9-2	Split	None	N/A	N/A	N/A	None
3-BDG-004-17-4	5	None (2)	N/A	N/A	N/A	None
3-BDG-004-17-4	5	None (2)	N/A	N/A	N/A	None
3-BDG-004-17-4	6	None (2)	N/A	N/A	N/A	None
3-BDG-004-17-4	6	None (2)	N/A	N/A	N/A	None
3-BDG-004-17-4	7	None	N/A	N/A	N/A	None
3-BDG-004-17-4	7	None	N/A	N/A	N/A	None
3-BDG-004-17-4	29	None	N/A	N/A	N/A	None
3-BDG-004-17-4	29	None	N/A	N/A	N/A	None
3-BDG-002-44-2	8	None	N/A	N/A	N/A	None
3-BDG-002-44-2	9	None	N/A	N/A	N/A	None

TABLE 3.6–32 JET IMPINGEMENT EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-BDG-002-44-2	9	None	N/A	N/A	N/A	None
3-BDG-004-10-2	10	None (2)	N/A	N/A	N/A	None
3-BDG-004-10-2	10	None (2)	N/A	N/A	N/A	None
3-BDG-004-81-2	11	None	N/A	N/A	N/A	None
3-BDG-004-18-4	12	None (2)	N/A	N/A	N/A	None
3-BDG-004-18-4	12	None (2)	N/A	N/A	N/A	None
3-BDG-004-18-4	13	None (2)	N/A	N/A	N/A	None
3-BDG-004-18-4	13	None (2)	N/A	N/A	N/A	None
3-BDG-004-18-4	14	None	N/A	N/A	N/A	None
3-BDG-004-18-4	14	None	N/A	N/A	N/A	None
3-BDG-004-18-4	30	None	N/A	N/A	N/A	None
3-BDG-004-18-4	30	None	N/A	N/A	N/A	None
3-BDG-002-46-2	15	None	N/A	N/A	N/A	None
3-BDG-002-46-2	16	None	N/A	N/A	N/A	None
3-BDG-002-46-2	16	None	N/A	N/A	N/A	None
3-BDG-004-11-2	17	None (2)	N/A	N/A	N/A	None
3-BDG-004-11-2	17	None (2)	N/A	N/A	N/A	None

TABLE 3.6–32 JET IMPINGEMENT EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-BDG-004-83-2	18	None	N/A	N/A	N/A	None
3-BDG-004-19-4	19	None (2)	N/A	N/A	N/A	None
3-BDG-004-19-4	19	None (2)	N/A	N/A	N/A	None
3-BDG-004-19-4	20	None (2)	N/A	N/A	N/A	None
3-BDG-004-19-4	20	None (2)	N/A	N/A	N/A	None
3-BDG-004-19-4	21	None	N/A	N/A	N/A	None
3-BDG-004-19-4	21	None	N/A	N/A	N/A	None
3-BDG-004-19-4	31	None	N/A	N/A	N/A	None
3-BDG-004-19-4	31	None	N/A	N/A	N/A	None
3-BDG-002-47-2	22	None	N/A	N/A	N/A	None
3-BDG-004-85-2	23	None	N/A	N/A	N/A	None
3-BDG-004-12-2	24	None	N/A	N/A	N/A	None
3-BDG-004-12-2	24	None	N/A	N/A	N/A	None
3-BDG-004-12-2	Split	None	N/A	N/A	N/A	None
3-BDG-004-12-2	25	None (2)	N/A	N/A	N/A	None
3-BDG-004-12-2	25	None (2)	N/A	N/A	N/A	None
3-BDG-004-20-4	26	None (2)	N/A	N/A	N/A	None

TABLE 3.6–32 JET IMPINGEMENT EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-BDG-004-20-4	26	None (2)	N/A	N/A	N/A	None
3-BDG-004-20-4	27	None (2)	N/A	N/A	N/A	None
3-BDG-004-20-4	27	None (2)	N/A	N/A	N/A	None
3-BDG-004-20-4	28	None	N/A	N/A	N/A	None
3-BDG-004-20-4	28	None	N/A	N/A	N/A	None
3-BDG-004-20-4	32	None	N/A	N/A	N/A	None
3-BDG-004-20-4	32	None	N/A	N/A	N/A	None
3-BDG-004-55-2	49	None	N/A	N/A	N/A	None
3-BDG-004-54-2	50	None	N/A	N/A	N/A	None
3-BDG-004-56-2	51	None	N/A	N/A	N/A	None
3-BDG-004-57-2	52	None	N/A	N/A	N/A	None
3-BDG-004-78-2	33	None (3)	N/A	N/A	N/A	None
3-BDG-004-78-2	34	None (3)	N/A	N/A	N/A	None
2" LINE TO SG1A	35	None (3)	N/A	N/A	N/A	None
2" LINE TO SG1A	36	None (3)	N/A	N/A	N/A	None
3-BDG-004-80-2	37	None (3)	N/A	N/A	N/A	None
2" LINE TO SG1B	38	None (3)	N/A	N/A	N/A	None

TABLE 3.6–32 JET IMPINGEMENT EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
2" LINE TO SG1B	39	None (3)	N/A	N/A	N/A	None
2" LINE TO SG1B	40	None (3)	N/A	N/A	N/A	None
3-BDG-004-82-2	41	None (3)	N/A	N/A	N/A	None
3-BDG-004-82-2	42	None (3)	N/A	N/A	N/A	None
2" LINE TO SG1C	43	None (3)	N/A	N/A	N/A	None
2" LINE TO SG1C	44	None (3)	N/A	N/A	N/A	None
3-BDG-004-84-2	45	None (3)	N/A	N/A	N/A	None
2" LINE TO SG1D	46	None (3)	N/A	N/A	N/A	None
2" LINE TO SG1D	47	None (3)	N/A	N/A	N/A	None
2" LINE TO SG1D	48	None (3)	N/A	N/A	N/A	None
3-BDG-004-17-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-17-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-17-4	Split	None	N/A	N/A	N/A	None
3-BDG-004-18-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-18-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-18-4	Split	None	N/A	N/A	N/A	None
3-BDG-004-19-4	IP	None	N/A	N/A	N/A	None

TABLE 3.6–32 JET IMPINGEMENT EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break # [(1)]	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-BDG-004-19-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-19-4	Split	None	N/A	N/A	N/A	None
3-BDG-004-20-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-20-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-20-4	Split	None	N/A	N/A	N/A	None
3-BDG-004-21-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-21-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-21-4	Split	None	N/A	N/A	N/A	None
3-BDG-004-22-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-22-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-22-4	Split	None	N/A	N/A	N/A	None
3-BDG-004-23-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-23-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-23-4	Split	None	N/A	N/A	N/A	None
3-BDG-004-24-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-24-4	IP	None	N/A	N/A	N/A	None
3-BDG-004-24-4	Split	None	N/A	N/A	N/A	None
3-BDG-003-49-4	IP	None	N/A	N/A	N/A	None

TABLE 3.6–32 JET IMPINGEMENT EFFECTS - STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Line Designation	Break # (1)	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-BDG-003-48-4	IP	None	N/A	N/A	N/A	None
3-BDG-003-50-4	IP	None	N/A	N/A	N/A	None
3-BDG-003-51-4	IP	None	N/A	N/A	N/A	None
3-BDG-003-49-4	TP	None	N/A	N/A	N/A	None
3-BDG-003-48-4	TP	None	N/A	N/A	N/A	None
3-BDG-003-50-4	TP	None	N/A	N/A	N/A	None
3-BDG-003-51-4	TP	None	N/A	N/A	N/A	None

NOTES:

(1) Repetition of Break Numbers is used to identify separate fluid reservoirs which provide a constant pressure source to maintain system pressure subsequent to the postulated pipe break. These reservoirs maintain system blowdown during the transient event as well as in the steady state.

(2) USNRC Generic Letter 87-11 eliminates the need to study jet impingement effects subsequent to Arbitrary Intermediate Breaks (AIBs). Therefore, no targets are evaluated.

(3) Steam Generator Blowdown (BDG) System - Wet Lay Up Piping is a High Energy System from each connection to the Steam Generators to the Normally Closed Valves 3BDG*V887, 3BDG*V889, 3BDG*V891, and 3BDG*V893. The Wet Lay Up Piping is separated from adjacent Steam Generators so any fluid jet impingement in one Steam Generator Cubicle does not damage essential structures, systems, or components associated with an adjacent Steam Generator based upon the Millstone Unit 3 Hazards Review Program Summary, NERM-069 Revision 1. The Wet Lay Up Piping is above the Floor Slab at Elevation 51'-4" which provides separation between the Reactor Coolant Pump Cubicles and the Upper Steam Generator Cubicles. Consequently fluid jet impingement effects on the Wet Lay Up Piping cannot propagate into a Loss of Coolant Accident (LOCA) based upon separation documented in the Millstone Unit 3 Hazards Review Program Summary, NERM-069, Revision 1.

TABLE 3.6–33 ENERGY ABSORBING CAPACITY OF A 4 INCH SCHEDULE 80 PIPE

Overall Displacement d (in)	Energy Absorbed Ep (in-k)	Impact Force F (kips)
0.0	0.0	0.0
0.52	6.60	18.15
1.05	17.68	23.69
1.57	31.18	27.74
2.09	46.45	30.75
2.62	63.43	34.32

TABLE 3.6–34 POSTULATED BREAKS AUXILIARY FEEDWATER SYSTEM

Line Designation	Break #	Building	Elevation	Break Type (1)	Total Additive Stress	Figure
3FWA*P1A Discharge Flange	1	ESF	26'-5"	CB	N/A Terminal End	3.6–32
3-FWA-006-2-3	2	ESF	26'-5"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-006-2-3	3	ESF	23'-2"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-006-2-3	4	ESF	22'-5"	CB	N/A Terminal End	3.6–32
3-FWA-006-8-3	5	ESF	22'-6"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-003-32-3	6	ESF	25'-11"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-003-39-3	7	ESF	25'-11"	CB	N/A Terminal End	3.6–32
3-FWA-003-39-3	8	ESF	25'-1"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-004-121-3	9	ESF	22'-6"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-004-121-3	10	ESF	22'-6"	CB	N/A Terminal End	3.6–32
3-FWA-004-42-2	13	Containment	43'-0"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-004-42-2	14	Containment	43'-0"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-006-135-2	15	Containment	58'-4"	CB	N/A Terminal End	3.6–32
3-FWA-003-33-3	16	ESF	25'-11"	CB	N/A Terminal End	3.6–32
3-FWA-003-33-3	17	ESF	25'-1"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-004-123-3	18	ESF	22'-6"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-004-123-3	19	ESF	22'-6"	CB	N/A Terminal End	3.6–32
3-FWA-004-36-2	22	Containment	43'-0"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-004-36-2	23	Containment	43'-0"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-006-100-2	24	Containment	58'-5"	CB	N/A Terminal End	3.6–32
3-FWA-004-125-3	25	ESF	22'-6"	CB	N/A Terminal End	3.6–32

TABLE 3.6–34 POSTULATED BREAKS AUXILIARY FEEDWATER SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type [1]	Total Additive Stress	Figure
3-FWA-004-120-2	28	Containment	20'-0"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-004-120-2	29	Containment	20'-0"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-004-120-2	30	Containment	20'-0"	CB	N/A Terminal End	3.6–32
3-FWA-004-120-2	31	Containment	20'-0"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-004-120-2	32	Containment	21'-6"	CB & LS	Above Threshold	3.6–32
3-FWA-004-120-2	33	Containment	21'-6"	CB	N/A Terminal End	3.6–32
3-FWA-004-54-2	34	Containment	21'-6"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-006-102-2	35	Containment	52'-11"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-006-138-2	36	Containment	58'-4"	CB	N/A Terminal End	3.6–32
3-FWA-004-127-3	37	ESF	22'-6"	CB	N/A Terminal End	3.6–32
3-FWA-004-119-2	40	Containment	20'-5"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-004-119-2	41	Containment	21'-0"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-004-119-2	42	Containment	21'-0"	CB	N/A Terminal End	3.6–32
3-FWA-004-48-2	43	Containment	21'-0"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-006-101-2	44	Containment	21'-9"	CB	N/A Arbitrary Intermediate	3.6–32
3-FWA-006-137-2	45	Containment	58'-4"	CB	N/A Terminal End	3.6–32
3-FWA-006-79-3	46	ESF	22'-6"	CB	N/A Terminal End	3.6–32
3-FWA-004-117-2	47	Containment	43'-0"	CB	N/A Terminal End	3.6–32
3-FWA-004-118-2	48	Containment	43'-0"	CB	N/A Terminal End	3.6–32
3-FWA-004-119-2	49	Containment	20'-0"	CB	N/A Terminal End	3.6–32
3-FWA-004-120-2	50	Containment	20'-0"	CB	N/A Terminal End	3.6–32
3-FWA-004-41-2	51	ESF	43'-0"	CB	N/A Terminal End	3.6–32

TABLE 3.6–34 POSTULATED BREAKS AUXILIARY FEEDWATER SYSTEM (CONTINUED)

Line Designation	Break #	Building	Elevation	Break Type (1)	Total Additive Stress	Figure
3-FWA-004-35-2	52	ESF	43'-0"	CB	N/A Terminal End	3.6–32
3-FWA-004-47-2	53	ESF	20'-0"	CB	N/A Terminal End	3.6–32
3-FWA-004-53-2	54	ESF	20'-0"	CB	N/A Terminal End	3.6–32
3-FWA-004-122-3	55	ESF	12'-0"	CB	N/A Terminal End	3.6–32
3-FWA-003-66-3	56	ESF	26'-0"	CB	N/A Terminal End	3.6–32
3-FWA-004-124-3	57	ESF	12'-0"	CB	N/A Terminal End	3.6–32
3-FWA-003-71-3	58	ESF	26'-0"	CB	N/A Terminal End	3.6–32
3-FWA-006-8-3	59	ESF	26'-6"	CB	N/A Terminal End	3.6–32
3-FWA-004-126-3	60	ESF	22'-0"	CB	N/A Terminal End	3.6–32
3-FWA-003-56-3	61	ESF	26'-0"	CB	N/A Terminal End	3.6–32
3-FWA-004-128-3	62	ESF	22'-0"	CB	N/A Terminal End	3.6–32
3-FWA-003-61-3	63	ESF	26'-0"	CB	N/A Terminal End	3.6–32
3-FWA-008-14-3	64	ESF	22'-6"	CB	N/A Terminal End	3.6–32
3-FWA-008-14-3	65	ESF	26'-5"	CB	N/A Terminal End	3.6–32
3-FWA-004-160-3	66	ESF	25'-5"	CB	N/A Terminal End	3.6–32

NOTE:

(1) Circumferential Pipe Break (CB) and Longitudinal Pipe Split (LS) are defined in FSAR Section 3.6.2.1.3.

TABLE 3.6–35 PIPE WHIP EFFECTS AUXILIARY FEEDWATER SYSTEM

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3FWA*P1A Discharge Flange	1	None [(1)]	None	None
3-FWA-006-2-3	2	None [(1)]	None	None
3-FWA-006-2-3	3	None [(1)]	None	None
3-FWA-006-2-3	4	None [(1)]	None	None
3-FWA-006-8-3	5	None [(1)]	None	None
3-FWA-003-32-3	6	None [(1)]	None	None
3-FWA-003-39-3	7	None [(1)]	None	None
3-FWA-003-39-3	8	None [(1)]	None	None
3-FWA-004-121-3	9	None [(1)]	None	None
3-FWA-004-121-3	10	None [(1)]	None	None
3-FWA-004-42-2	13	None [(1)]	None	None
3-FWA-004-42-2	14	None [(1)]	None	None
3-FWA-006-135-2	15	None [(1)]	None	None
3-FWA-003-33-3	16	None [(1)]	None	None
3-FWA-003-33-3	17	None [(1)]	None	None
3-FWA-004-123-3	18	None [(1)]	None	None
3-FWA-004-123-3	19	None [(1)]	None	None

TABLE 3.6–35 PIPE WHIP EFFECTS AUXILIARY FEEDWATER SYSTEM (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-FWA-004-36-2	22	None (1)	None	None
3-FWA-004-36-2	23	None (1)	None	None
3-FWA-004-36-2	23	3-FWS-020-18-2	None	None
3-FWA-006-100-2	24	None (1)	None	None
3-FWA-004-125-3	25	None (1)	None	None
3-FWA-004-120-2	28	None (1)	None	None
3-FWA-004-120-2	29	None (1)	None	None
3-FWA-004-120-2	30	None (1)	None	None
3-FWA-004-120-2	31	None (1)	None	None
3-FWA-004-120-2	32 Split	None (1)	None	None
3-FWA-004-120-2	32	None (1)	None	None
3-FWA-004-120-2	33	None (1)	None	None
3-FWA-004-54-2	34	None (1)	None	None
3-FWA-004-54-2	34	3-FWS-020-26-2	None	None
3-FWA-006-102-2	35	None (1)	None	None
3-FWA-006-102-2	35	3-FWS-020-26-2	None	None
3-FWA-006-138-2	36	None (1)	None	None
3-FWA-004-127-3	37	None (1)	None	None

TABLE 3.6–35 PIPE WHIP EFFECTS AUXILIARY FEEDWATER SYSTEM (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-FWA-004-119-2	40	None [(1)]	None	None
3-FWA-004-119-2	41	None [(1)]	None	None
3-FWA-004-119-2	42	None [(1)]	None	None
3-FWA-004-48-2	43	None [(1)]	None	None
3-FWA-004-48-2	43	3-FWS-020-22-2	None	None
3-FWA-006-101-2	44	None [(1)]	None	None
3-FWA-006-101-2	44	3-FWS-020-22-2	None	None
3-FWA-006-137-2	45	None [(1)]	None	None
3-FWA-006-79-3	46	None [(1)]	None	None
3-FWA-004-117-2	47	None [(1)]	None	None
3-FWA-004-118-2	48	None [(1)]	None	None
3-FWA-004-119-2	49	None [(1)]	None	None
3-FWA-004-120-2	50	None [(1)]	None	None
3-FWA-004-41-2	51	None [(1)]	None	None
3-FWA-004-35-2	52	None [(1)]	None	None
3-FWA-004-47-2	53	None [(1)]	None	None
3-FWA-004-53-2	54	None [(1)]	None	None
3-FWA-004-122-3	55	None [(1)]	None	None

TABLE 3.6–35 PIPE WHIP EFFECTS AUXILIARY FEEDWATER SYSTEM (CONTINUED)

Line Designation	Break #	Blowdown Source	Essential Pipe Whip Target	Protection Requirement
3-FWA-003-66-3	56	None (1)	None	None
3-FWA-004-124-3	57	None (1)	None	None
3-FWA-003-71-3	58	None (1)	None	None
3-FWA-006-8-3	59	None (1)	None	None
3-FWA-004-126-3	60	None (1)	None	None
3-FWA-003-56-3	61	None (1)	None	None
3-FWA-004-128-3	62	None (1)	None	None
3-FWA-003-61-3	63	None (1)	None	None
3-FWA-008-14-3	64	None (1)	None	None
3-FWA-008-14-3	65	None (1)	None	None
3-FWA-004-160-3	66	None (1)	None	None

NOTE:

(1) Sustained blowdown of the Auxiliary Feedwater System does not occur. Dual inline check valves prevent backflow from the Main Feedwater System and the Auxiliary Feedwater Pumps have limited capacity to sustain the system pressure. Therefore, pipe whip does not occur.

TABLE 3.6–36 JET IMPINGEMENT EFFECTS AUXILIARY FEEDWATER SYSTEMS

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3FWA*P1A Discharge Flange	1	None (1)	N/A	N/A	N/A	None
3-FWA-006-2-3	2	None (1)	N/A	N/A	N/A	None (2)
3-FWA-006-2-3	3	None (1)	N/A	N/A	N/A	None (2)
3-FWA-006-2-3	4	None (1)	N/A	N/A	N/A	None
3-FWA-006-8-3	5	None (1)	N/A	N/A	N/A	None (2)
3-FWA-003-32-3	6	None (1)	N/A	N/A	N/A	None (2)
3-FWA-003-39-3	7	None (1)	N/A	N/A	N/A	None
3-FWA-003-39-3	8	None (1)	N/A	N/A	N/A	None (2)
3-FWA-004-121-3	9	None (1)	N/A	N/A	N/A	None (2)
3-FWA-004-121-3	10	None (1)	N/A	N/A	N/A	None
3-FWA-004-42-2	13	None (1)	N/A	N/A	N/A	None (2)
3-FWA-004-42-2	14	None (1)	N/A	N/A	N/A	None (2)
3-FWA-006-135-2	15	None (1)	N/A	N/A	N/A	None
3-FWA-003-33-3	16	None (1)	N/A	N/A	N/A	None
3-FWA-003-33-3	17	None (1)	N/A	N/A	N/A	None (2)

TABLE 3.6–36 JET IMPINGEMENT EFFECTS AUXILIARY FEEDWATER SYSTEMS (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-FWA-004-123-3	18	None (1)	N/A	N/A	N/A	None (2)
3-FWA-004-123-3	19	None (1)	N/A	N/A	N/A	None
3-FWA-004-36-2	22	None (1)	N/A	N/A	N/A	None (2)
3-FWA-004-36-2	23	None	N/A	N/A	N/A	None (2)
3-FWA-006-100-2	24	None (1)	N/A	N/A	N/A	None
3-FWA-004-125-3	25	None (1)	N/A	N/A	N/A	None
3-FWA-004-120-2	28	None (1)	N/A	N/A	N/A	None (2)
3-FWA-004-120-2	29	None (1)	N/A	N/A	N/A	None (2)
3-FWA-004-120-2	30	None (1)	N/A	N/A	N/A	None
3-FWA-004-120-2	31	None (1)	N/A	N/A	N/A	None (2)
3-FWA-004-120-2	32 Split	None (1)	N/A	N/A	N/A	None
3-FWA-004-120-2	32	None (1)	N/A	N/A	N/A	None
3-FWA-004-120-2	33	None (1)	N/A	N/A	N/A	None
3-FWA-004-54-2	34	None	N/A	N/A	N/A	None (2)
3-FWA-006-102-2	35	None	N/A	N/A	N/A	None (2)
3-FWA-006-138-2	36	None (1)	N/A	N/A	N/A	None

TABLE 3.6–36 JET IMPINGEMENT EFFECTS AUXILIARY FEEDWATER SYSTEMS (CONTINUED)

Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-FWA-004-127-3	37	None [(1)]	N/A	N/A	N/A	None
3-FWA-004-119-2	40	None [(1)]	N/A	N/A	N/A	None [(2)]
3-FWA-004-119-2	41	None [(1)]	N/A	N/A	N/A	None [(2)]
3-FWA-004-119-2	42	None [(1)]	N/A	N/A	N/A	None
3-FWA-004-48-2	43	None	N/A	N/A	N/A	None [(2)]
3-FWA-006-101-2	44	None	N/A	N/A	N/A	None [(2)]
3-FWA-006-137-2	45	None [(1)]	N/A	N/A	N/A	None
3-FWA-006-79-3	46	None [(1)]	N/A	N/A	N/A	None
3-FWA-004-117-2	47	None [(1)]	N/A	N/A	N/A	None
3-FWA-004-118-2	48	None [(1)]	N/A	N/A	N/A	None
3-FWA-004-119-2	49	None [(1)]	N/A	N/A	N/A	None
3-FWA-004-120-2	50	None [(1)]	N/A	N/A	N/A	None
3-FWA-004-41-2	51	None [(1)]	N/A	N/A	N/A	None
3-FWA-004-35-2	52	None [(1)]	N/A	N/A	N/A	None
3-FWA-004-47-2	53	None [(1)]	N/A	N/A	N/A	None
3-FWA-004-53-2	54	None [(1)]	N/A	N/A	N/A	None

TABLE 3.6–36 JET IMPINGEMENT EFFECTS AUXILIARY FEEDWATER SYSTEMS (CONTINUED)

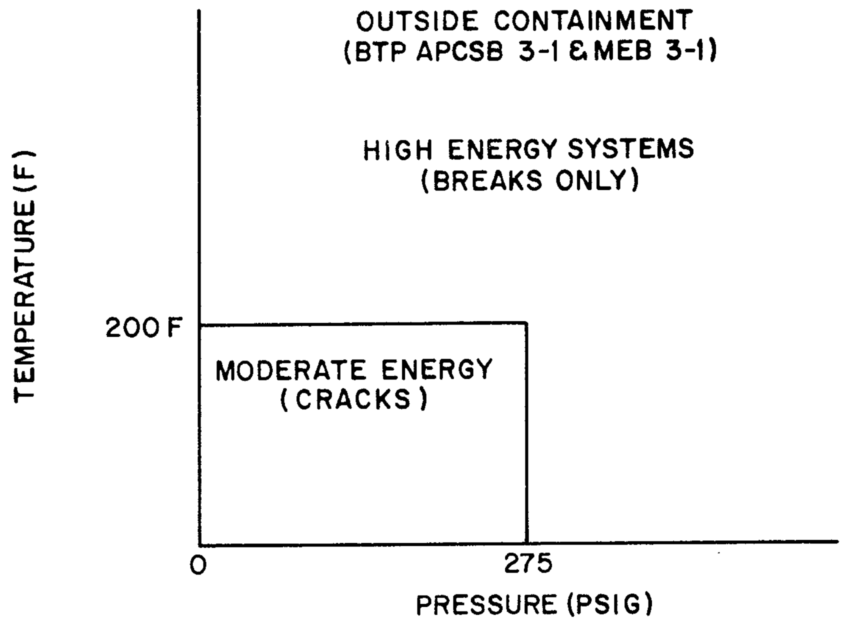
Line Designation	Break #	Essential Jet Impingement Target	Distance to Target	Jet Intensity at the Target	Jet Load on the Target	Protection Requirement
3-FWA-004-122-3	55	None (1)	N/A	N/A	N/A	None
3-FWA-003-66-3	56	None (1)	N/A	N/A	N/A	None
3-FWA-004-124-3	57	None (1)	N/A	N/A	N/A	None
3-FWA-003-71-3	58	None (1)	N/A	N/A	N/A	None
3-FWA-006-8-3	59	None (1)	N/A	N/A	N/A	None
3-FWA-004-126-3	60	None (1)	N/A	N/A	N/A	None
3-FWA-003-56-3	61	None (1)	N/A	N/A	N/A	None
3-FWA-004-128-3	62	None (1)	N/A	N/A	N/A	None
3-FWA-003-61-3	63	None (1)	N/A	N/A	N/A	None
3-FWA-008-14-3	64	None (1)	N/A	N/A	N/A	None
3-FWA-008-14-3	65	None (1)	N/A	N/A	N/A	None
3-FWA-004-160-3	66	None (1)	N/A	N/A	N/A	None

NOTES:

(1) Sustained blowdown of the Auxiliary Feedwater System does not occur. Dual inline check valves inside containment prevent backflow from the Main Feedwater System and the Auxiliary Feedwater Pumps have limited capacity to sustain the system pressure.

(2) USNRC Generic Letter 87-11 eliminates the requirement to evaluate the jet impingement effects of Arbitrary Intermediate Breaks (AIBs). Therefore, jet impingement effects subsequent to AIBs are not evaluated.

FIGURE 3.6–1 FLUID SYSTEMS SUBJECTED TO BREAK AND CRACK ANALYSIS



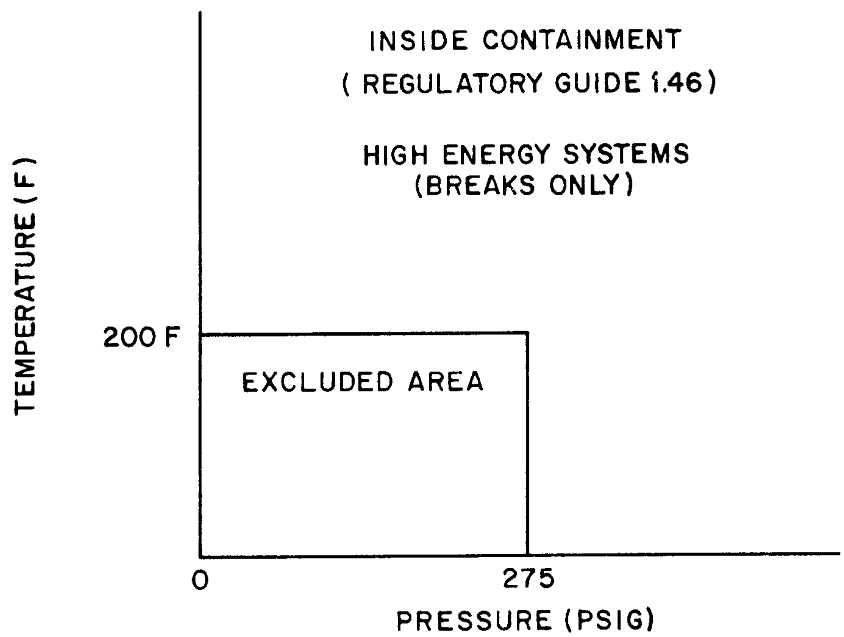


FIGURE 3.6–2 RELATIVE LOCATIONS OF SAFETY RELATED EQUIPMENT, ELEVATION 3 FEET 8 INCHES

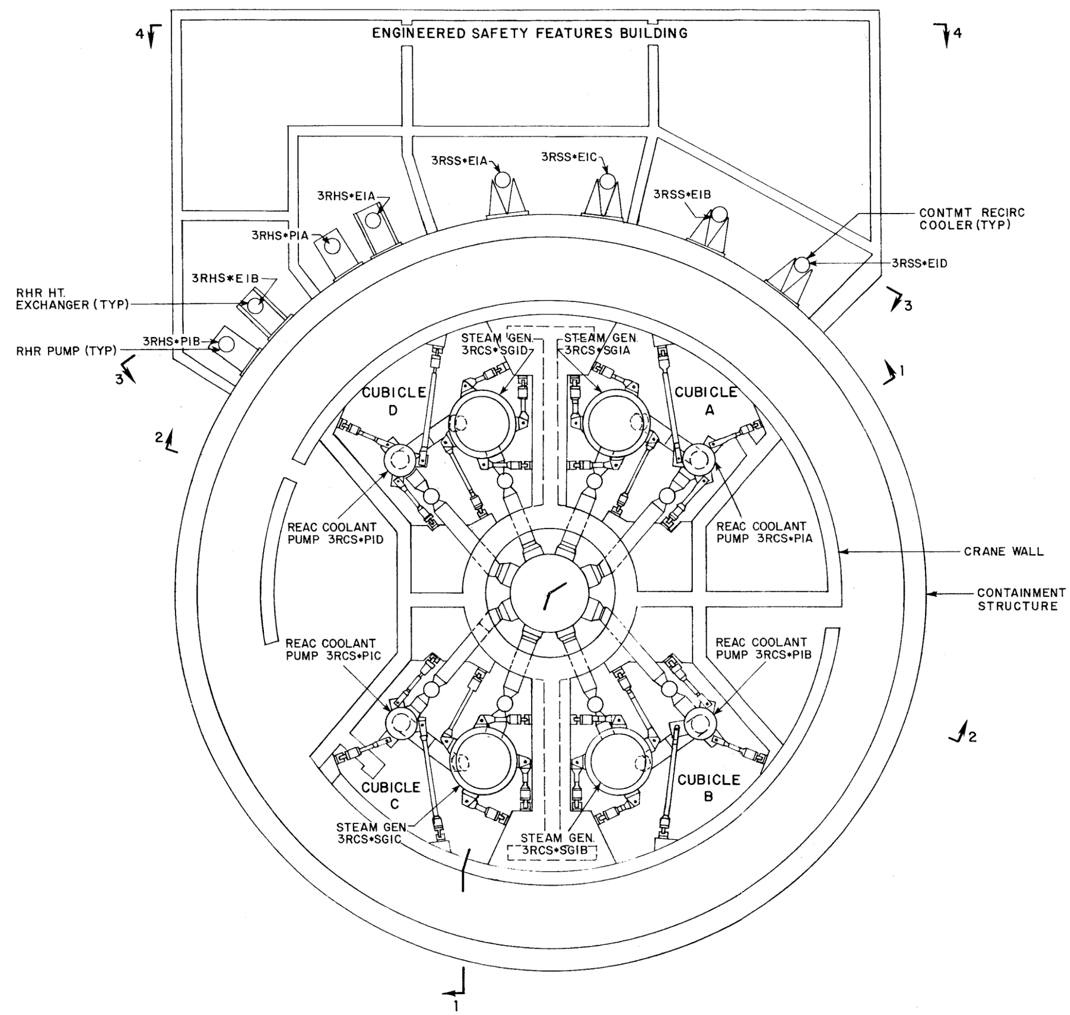


FIGURE 3.6-2
RELATIVE LOCATIONS
OF SAFETY - RELATED EQUIPMENT-EL. 3'-8"
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–3 RELATIVE LOCATIONS OF SAFETY RELATED EQUIPMENT, ELEVATION 24 FEET 6 INCHES

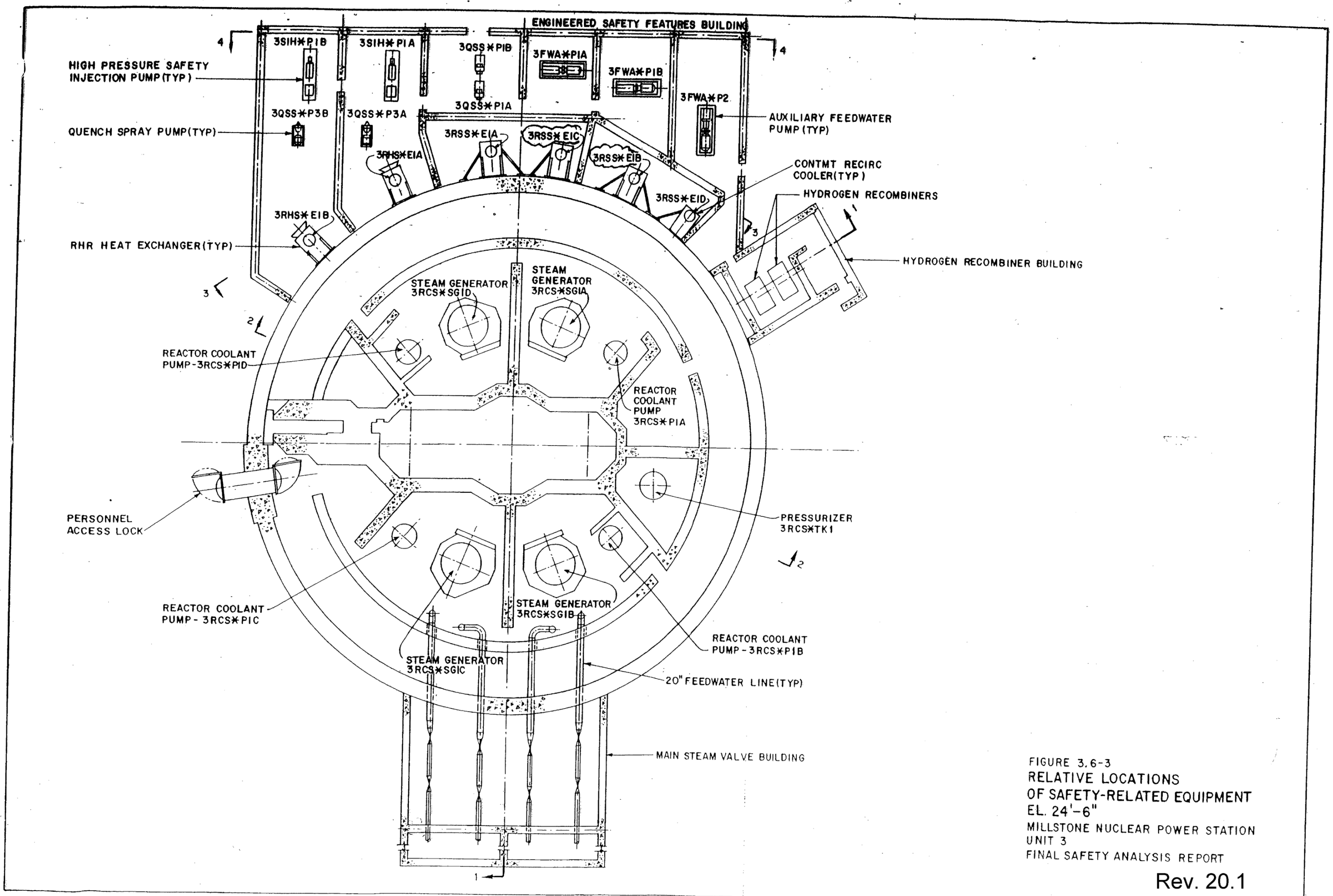


FIGURE 3.6-3
RELATIVE LOCATIONS
OF SAFETY-RELATED EQUIPMENT
EL. 24'-6"
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
Rev. 20.1

FIGURE 3.6–4 RELATIVE LOCATIONS OF SAFETY RELATED EQUIPMENT, ELEVATION 51 FEET 4 INCHES

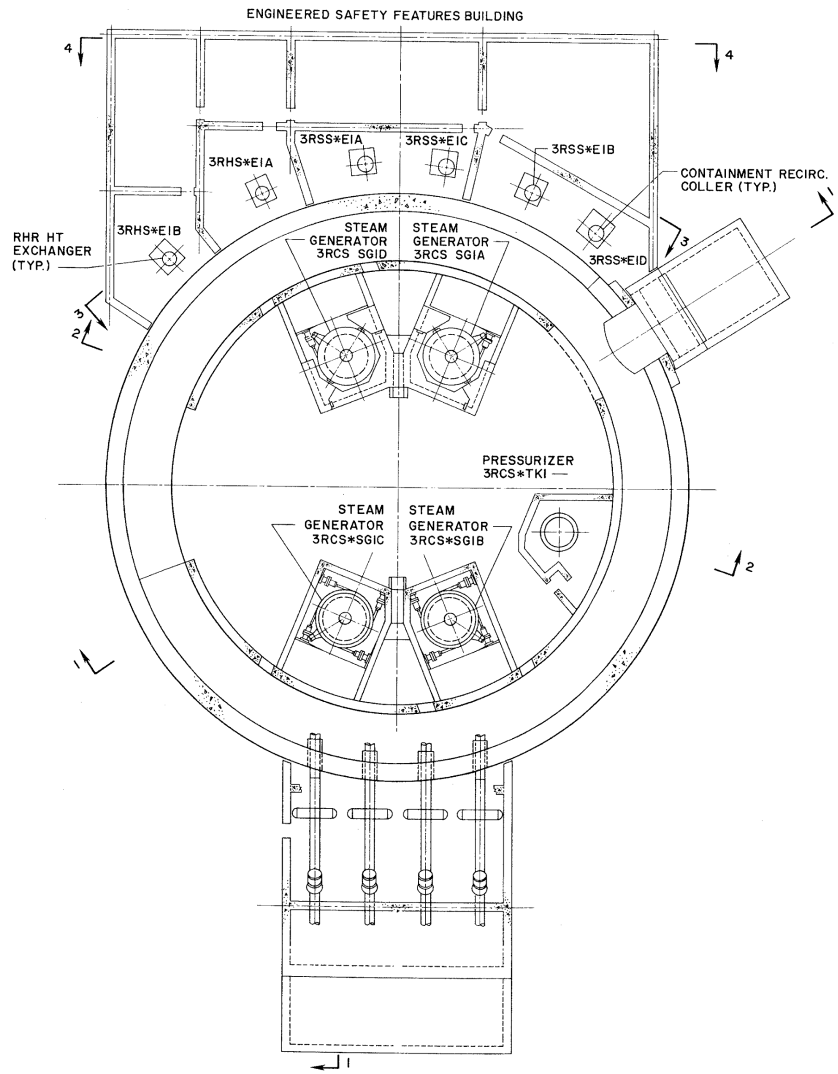


FIGURE 3.6-4

RELATIVE LOCATIONS
SAFETY RELATED EQUIPMENT
EL. 51' - 4"
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–5 RELATIVE LOCATIONS OF SAFETY RELATED EQUIPMENT, SECTION 1-1

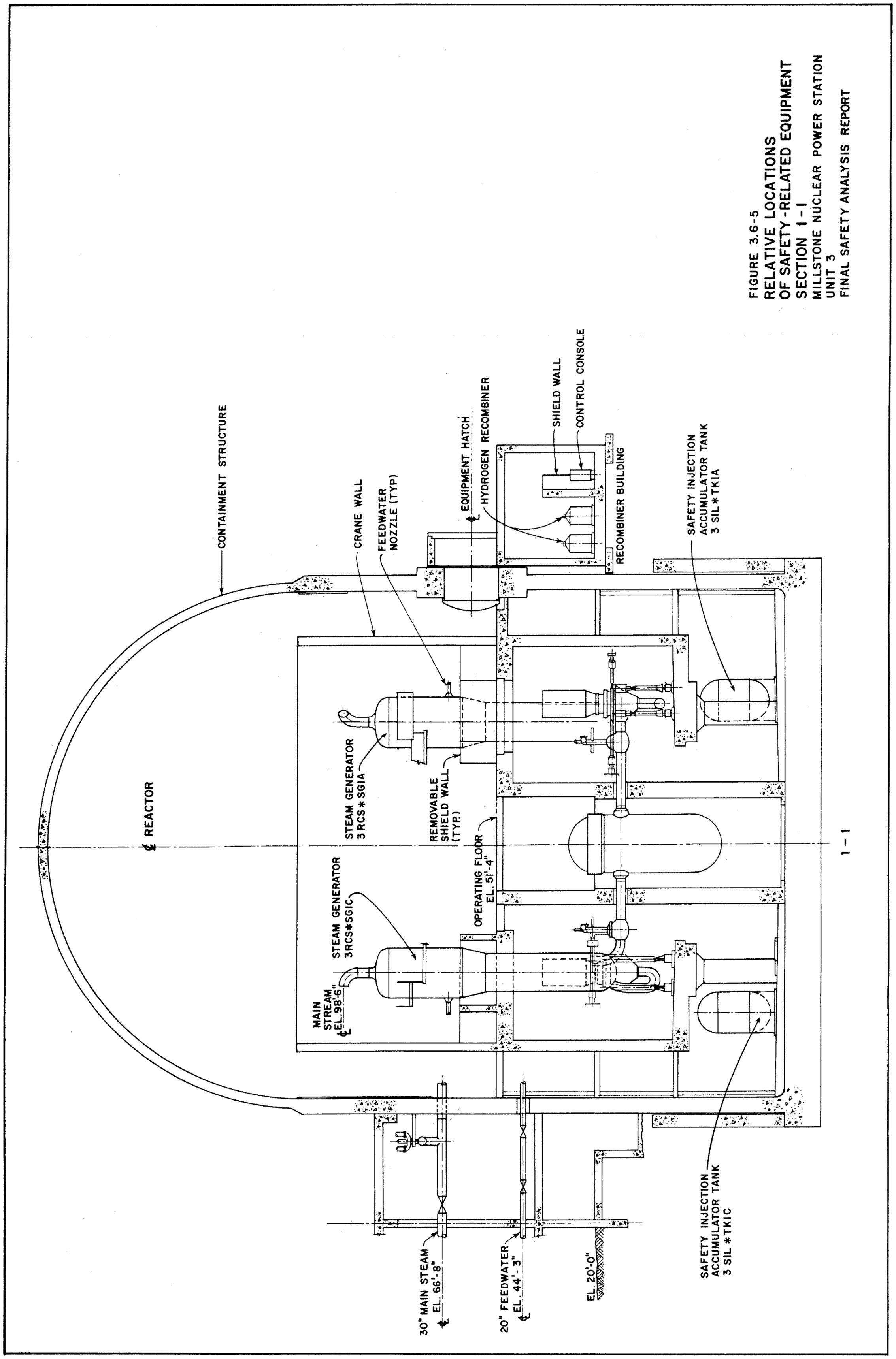


FIGURE 3.6–6 RELATIVE LOCATIONS OF SAFETY RELATED EQUIPMENT, SECTION 2-2

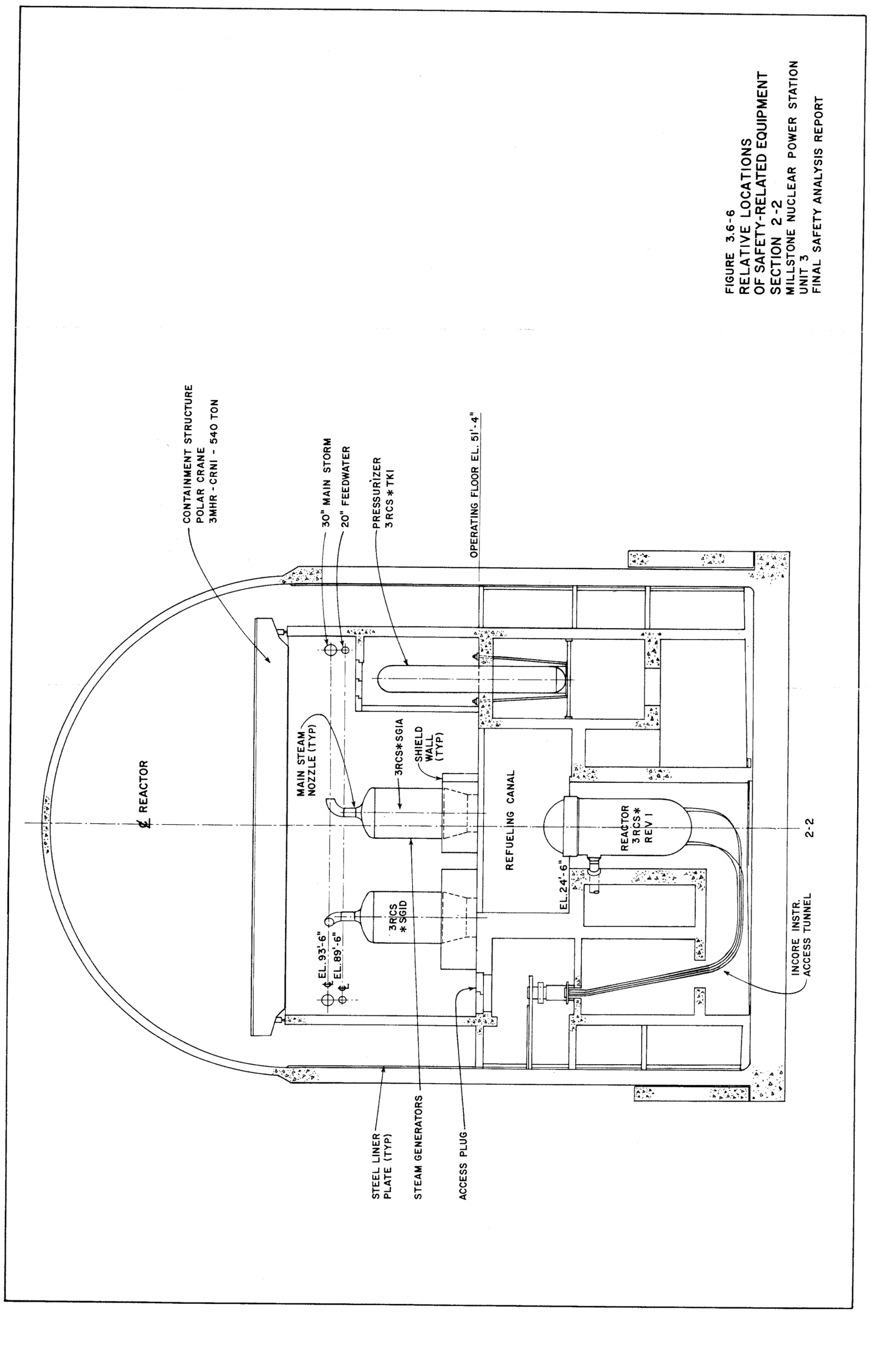


FIGURE 3.6–7 RELATIVE LOCATIONS OF SAFETY RELATED EQUIPMENT, SECTIONS 3-3, 4-4

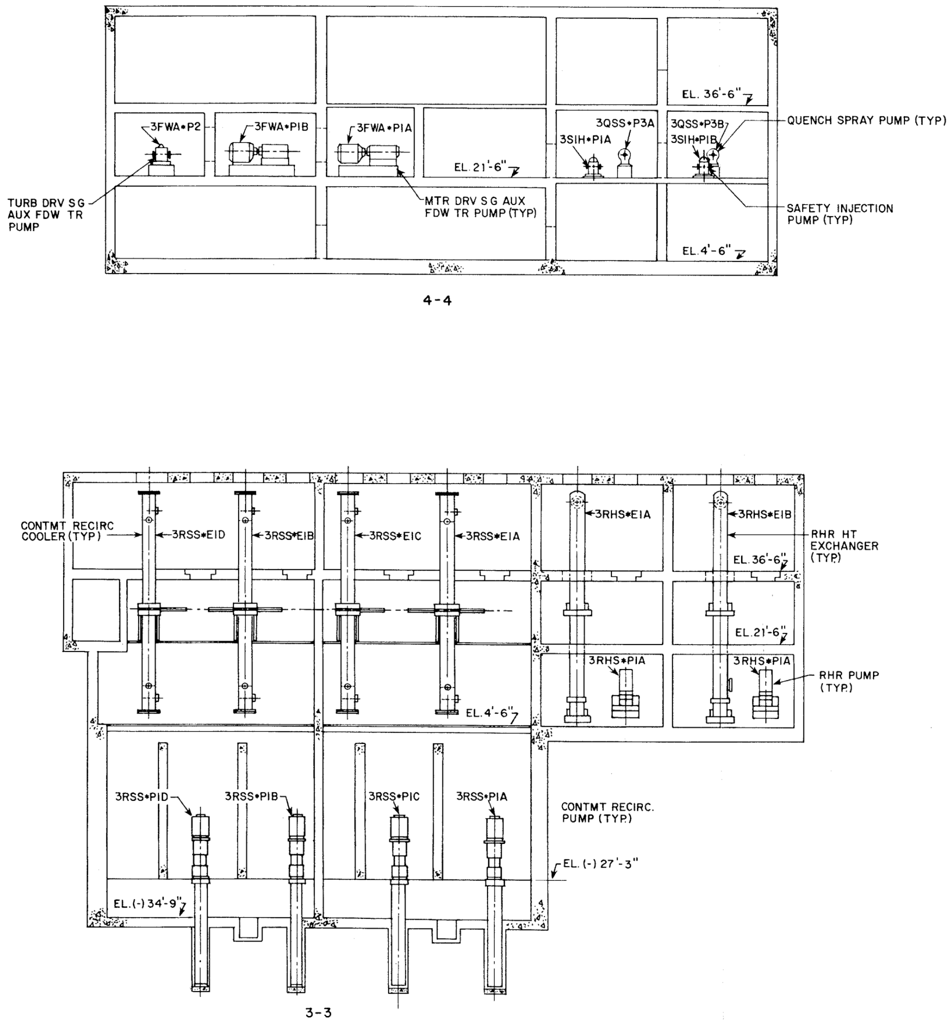
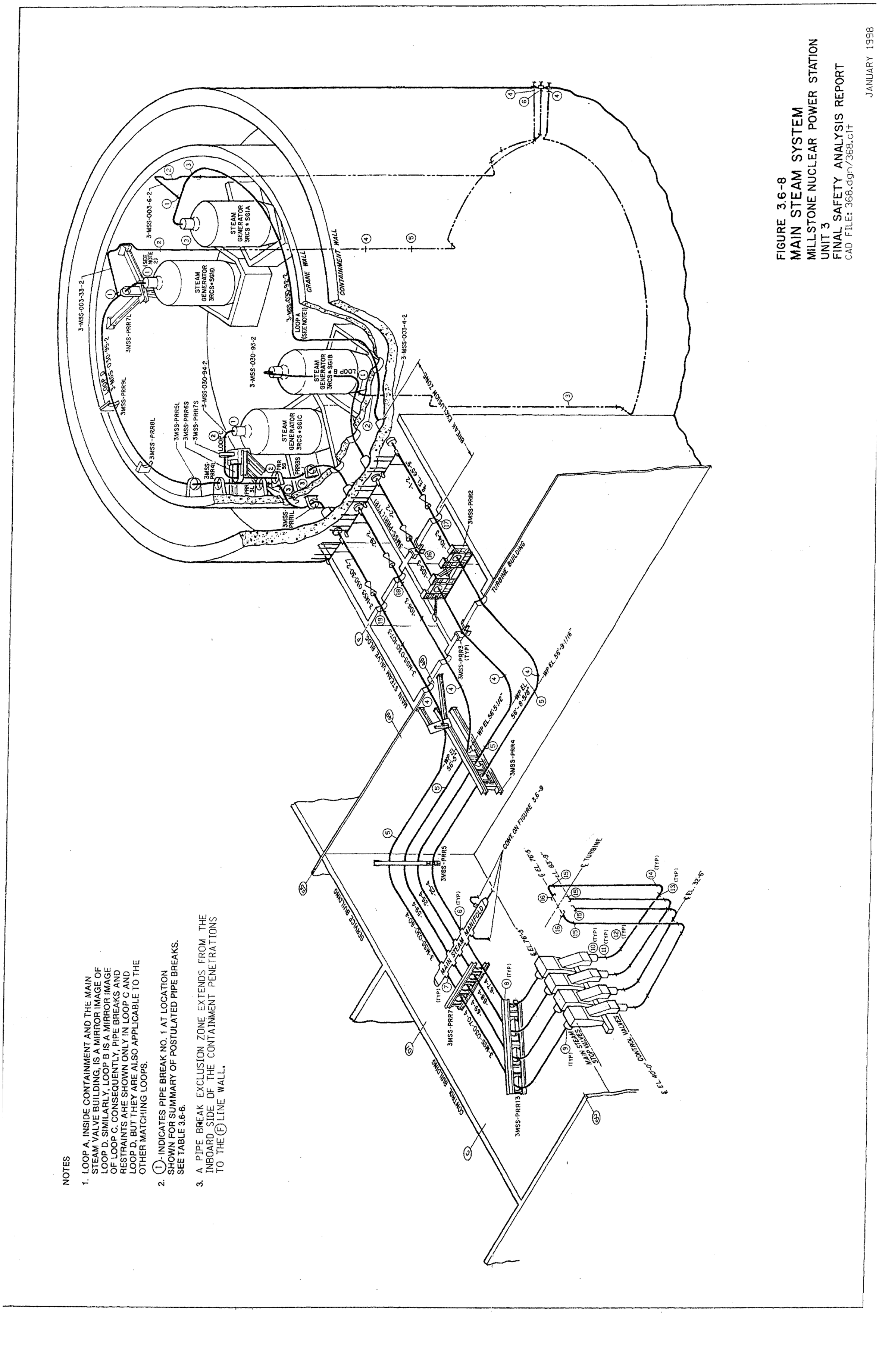


FIGURE 3.6-7
RELATIVE LOCATIONS
OF SAFETY-RELATED EQUIPMENT
SECTION 3-3, 4-4
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–8 MAIN STEAM SYSTEM



NOTES TO FIGURE 3.6–8

DATA FOR PIPE RUPTURE RESTRAINTS - MAIN STEAM SYSTEM

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3MSS-PRR1LA	(1)	66'-3"	53'-3" West	21'-8" South	Limit Stop & Strap	Prevent pipe whip into Crane Wall & Liner
3MSS-PRR2LA	2	66'-3"	45'-10" West	32'-2" South	Sleeve in Crane Wall	Protect Crane Wall
3MSS-PRR3LA	3	71'-8"	42'-6" West	34'-10" South	Omni-Directional	(2)
3MSS-PRR4LA	1 & 2	87'-6"	42'-6" West	34'-10" South	Omni-Directional	Prevent whip into Crane Wall
3MSS-PRR5LA	2	93'-6"	39'-3" West	38'-6" South	Omni-Directional	Prevent whip into Polar Crane Support
3MSS-PRR7LA	1	93'-6"	35'-11" East	20'-11" South	Laminated Strap	Prevent whip into Polar Crane Support
3MSS-PRR8LA	1, 2, & 3	93'-6"	14'-3" West	53'-2" South	Limit Stop	Prevent whip into Crane Wall
3MSS-PRR9LA	1, 2, & 3	93'-6"	14'-3" East	53'-2" South	Limit Stop	Prevent whip into Crane Wall
3MSS-PRR3SB	(3)	66'-3"	52'-2" West	17'-5" South	Omni-Directional	Prevent whip into Steam Generator Supports
3MSS-PRR5SB	3	72'-6"	48'-11" West	25'-2" South	Omni-Directional	Prevent whip into Crane Wall
3MSS-PRR6SB	2	88'-9"	39'-3" West	18'-6" South	Laminated Strap	Prevent whip into Crane Wall
3MSS-PRR7SB	1	88'-9"	39'-3" West	18'-6" South	Laminated Strap	Prevent whip into Polar Crane Support
3MSS-PRR3SC	(3)	66'-3"	52'-2" West	17'-5" North	Omni-Directional	Prevent whip into Steam Generator Supports

NOTES TO FIGURE 3.6–8 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS - MAIN STEAM SYSTEM (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3MSS-PRR5SC	3	72'-6"	48'-11" West	25'-2" North	Omni-Directional	Prevent whip into Crane Wall
3MSS-PRR6SC	2	88'-9"	39'-3" West	18'-6" North	Laminated Strap	Prevent whip into Crane Wall
3MSS-PRR7SC	1	88'-9"	39'-3" West	18'-6" North	Laminated Strap	Prevent whip into Polar Crane Support
3MSS-PRR1LD	(1)	66'-3"	53'-3" West	21'-8" North	Limit Stop & Strap	Prevent pipe whip into Crane Wall & Liner
3MSS-PRR2LD	2	66'-3"	45'-10" West	32'-2" North	Sleeve in Crane Wall	Protect Crane Wall
3MSS-PRR3LD	3	71'-8"	42'-6" West	34'-10" North	Omni-Directional	
3MSS-PRR4LD	1 & 2	87'-6"	42'-6" West	34'-10" North	Omni-Directional	Prevent whip into Crane Wall
3MSS-PRR5LD	2	93'-6"	39'-3" West	38'-6" North	Omni-Directional	Prevent whip into Polar Crane Support
3MSS-PRR7LD	1	93'-6"	35'-11" East	20'-11" North	Laminated Strap	Prevent whip into Polar Crane Support
3MSS-PRR8LD	1, 2, & 3	93'-6"	14'-3" West	53'-2" North	Limit Stop	Prevent whip into Crane Wall
3MSS-PRR9LD	1, 2, & 3	93'-6"	14'-3" East	53'-2" North	Limit Stop	Prevent whip into Crane Wall
3MSS-PRR1A	4 & 5	66'-3"	105'-2" West	21'-0" South	Dual Function	Protects the Break Exclusion Zone
3MSS-PRR2A	4 & 5	66'-3"	122'-5" West	21'-0" South	Dual Function	Protects the Break Exclusion Zone
3MSS-PRR3A	4 & 5	66'-3"	141'-2" West	21'-0" South	Limit Stop	Protects the Break Exclusion Zone
3MSS-PRR4A	4	56'-3"	173'-8" West	29'-0" North	Limit Stop	Protects the Control Building

NOTES TO FIGURE 3.6–8 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS - MAIN STEAM SYSTEM (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3MSS-PRR5A	5 & 6	55'-10"	177'-0" West	66'-6" North	Limit Stop	Protects the Control Building
3MSS-PRR7A	7 or 8	55'-4"	214'-6" West	70'-7" North	Limit Stop	Protects the Control Building
3MSS-PRR13A	7, 8, or 9	54'-8"	234'-10" West	65'-9" North	Limit Stop	Protects the Control Building
3MSS-PRR1B	4 & 5	66'-3"	105'-2" West	7'-0" South	Dual Function	Protects the Break Exclusion Zone
3MSS-PRR2B	4 & 5	66'-3"	122'-7" West	7'-0" South	Dual Function	Protects the Break Exclusion Zone
3MSS-PRR3B	4 & 5	66'-3"	141'-2" West	7'-0" South	Limit Stop	Protects the Break Exclusion Zone
3MSS-PRR4B	4	56'-4"	168'-2" West	29'-0" North	Limit Stop	Protects the Control Building
3MSS-PRR5B	5 & 6	55'-10"	171'-7" West	71'-7" North	Limit Stop	Protects the Control Building
3MSS-PRR7B	7 or 8	55'-4"	214'-6" West	75'-7" North	Limit Stop	Protects the Control Building
3MSS-PRR13B	7, 8, or 9	54'-8"	241'-10" West	71'-2" North	Limit Stop	Protects the Control Building
3MSS-PRR1C	5	66'-3"	105'-2" West	7'-0" North	Dual Function	Protects the Break Exclusion Zone
3MSS-PRR3C	5	66'-3"	141'-2" West	7'-0" North	Limit Stop	Protects the Break Exclusion Zone
3MSS-PRR4C	4	56'-5"	162'-8" West	29'-0" North	Limit Stop	Protects the Control Building
3MSS-PRR5C	5 & 6	55'-10"	166'-2" West	76'-8" North	Limit Stop	Protects the Control Building
3MSS-PRR7C	7 or 8	55'-4"	214'-6" West	80'-7" North	Limit Stop	Protects the Control Building
3MSS-PRR13C	7, 8, or 9	54'-8"	248'-9" West	76'-7" North	Limit Stop	Protects the Control Building
3MSS-PRR1D	5	66'-3"	105'-2" West	21'-0" North	Dual Function	Protects the Break Exclusion Zone
3MSS-PRR3D	5	66'-3"	141'-2" West	21'-0" North	Limit Stop	Protects the Break Exclusion Zone
3MSS-PRR4D	4 & 5	62'-5"	150'-1" West	29'-0" North	Laminated Strap	Protects the Control Building

NOTES TO FIGURE 3.6–8 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS - MAIN STEAM SYSTEM (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3MSS-PRR5D	5 & 6	55'-10"	160'-9" West	81'-9" North	Limit Stop	Protects the Control Building
3MSS-PRR7D	7 or 8	55'-4"	214'-6" West	85'-7" North	Limit Stop	Protects the Control Building
3MSS-PRR13D	7, 8, or 9	54'-8"	255'-8" West	82'-0" North	Limit Stop	Protects the Control Building

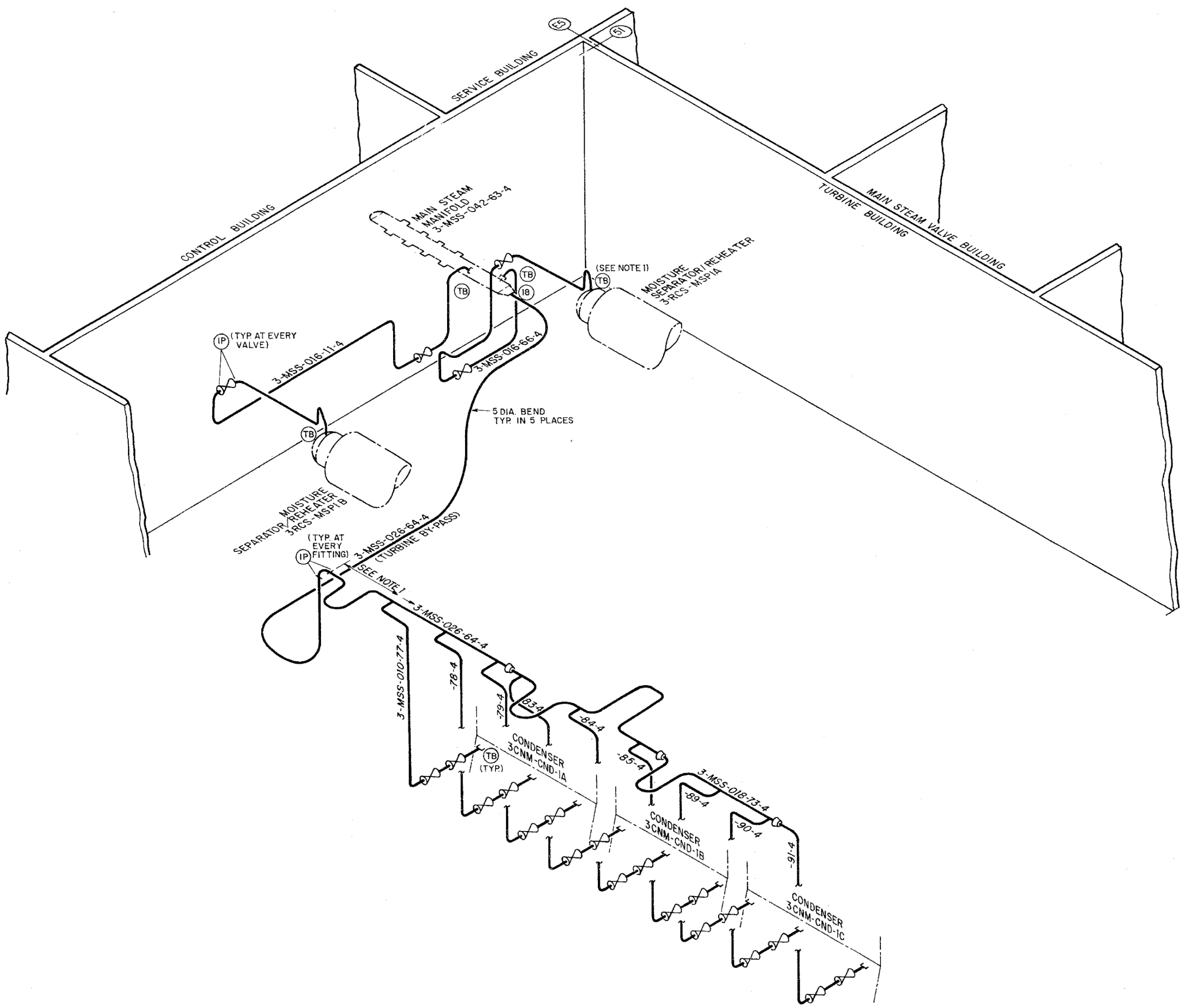
Notes

(1) Pipe Rupture Restraints 3MSS-PRR1LA and 3MSS-PRR1LD are laminated straps which prevent the Main Steam line from impacting the Containment Liner subsequent to Break Number 3 on Main Steam Lines 3-MSS-030-92-2 and 3-MSS-030-95-2. The Limit Stop function was provided to prevent the Main Steam line from impacting the Crane Wall subsequent to a Terminal End Break at the Containment Penetration. This Terminal End Break was deleted as each Main Steam Line Break Exclusion Zone extends from the pipe to flued head weld at each containment penetration inside containment to each Main Steam Containment Isolation Valve Outside Containment.

(2) Pipe rupture restraints 3MSS-PRR3LA and 3MSS-PRR3LD prevent pipe whip within the adjacent Steam Generator Cubicles 3RCS*SG1B and 3RCS*SG1C. The NSSS System Standard Design Criteria does not permit break propagation from a broken loop to an unbroken loop. Therefore, the restraints prevent pipe whip within the unbroken loop cubicles.

(3) Pipe rupture restraints 3MSS-PRR3SB and 3MSS-PRR3SC prevent the pipe whip effects subsequent to a Terminal End break at the Main Steam Line pipe weld to the Containment Penetration flued head. The Main Steam System Break Exclusion Zone includes the pipe to flued head weld inside containment to protect the containment liner from jet impingement effects.

FIGURE 3.6–9 MAIN STEAM SYSTEM (CONTINUED)



NOTES

1. FOR THE TURBINE BY-PASS STEAM PIPING BREAKS WERE POSTULATED AT TERMINAL ENDS, AND AT EVERY PIPING WELD TO FITTING AND VALVE. HOWEVER, THE PORTION OF THE TURBINE BY-PASS PIPING FROM THE MAIN STEAM MANIFOLD WHICH RUN IN PARALLEL WITH THE CONTROL BUILDING WALL IS PROVIDED WITH 5-DIAMETER BENDS IN LIEU OF FITTINGS TO PRECLUDE BREAKS FROM OCCURRING WITHIN THIS SECTION OF THE PIPING, THEREBY PROTECTING THE CONTROL BUILDING WALL FROM PIPE WHIP. SIMILARLY, FOR THE MOISTURE SEPARATOR STEAM PIPING. BREAKS WERE POSTULATED AT TERMINAL ENDS, AND AT EVERY PIPING WELD TO FITTING AND VALVE.

2. EXAMINATION OF THE MOISTURE SEPARATOR STEAM PIPING FOR EFFECTS OF PIPE RUPTURE AND JET IMPINGEMENT HAS DETERMINED THAT NO PROTECTIVE MEASURES ARE REQUIRED.

3. NOMENCLATURE:

   (TP) - INDICATES BREAK AT TERMINAL ENDS.

   (IP) - INDICATES BREAK AT INTERMEDIATE LOCATIONS.

FIGURE 3.6-9
MAIN STEAM SYSTEM (CONTINUED)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–10 FEEDWATER SYSTEM POSTULATED PIPE RUPTURE LOCATIONS

NOTES

1. LOOP A INSIDE CONTAINMENT AND MAIN STEAM VALVE BUILDING IS A MIRROR IMAGE OF LOOP D. SIMILARLY, LOOP B IS A MIRROR IMAGE OF LOOP C. CONSEQUENTLY, PIPE BREAK LOCATIONS ARE SHOWN ONLY IN LOOP B AND LOOP A ON MAIN FEEDWATER PIPING, BUT THEY ARE ALSO APPLICABLE TO THE OTHER MATCHING LOOPS.
2. (1) INDICATES PIPE BREAK NO. 1 FOR SUMMARY OF POSTULATED PIPE BREAKS SEE TABLE 3.6-9.
3. A PIPE BREAK EXCLUSION ZONE EXTENDS FROM EACH CONTAINMENT PENETRATION ON THE INBOARD SIDE TO THE F LINE WALL IN THE MAIN STEAM VALVE BUILDING.
4. FOR SUMMARY OF RUPTURE RESTRAINTS INSIDE AND OUTSIDE CONTAINMENT STRUCTURE, SEE FIGURE 3.6-11.
5. A TERMINAL END BREAK IS POSTULATED ON EACH MAIN FEEDWATER LINE AT THE F LINE WALL IN THE MAIN STEAM VALVE BUILDING. THIS TERMINAL END BREAK DEFINES THE EXTREMITY OF THE BREAK EXCUSION ZONE.

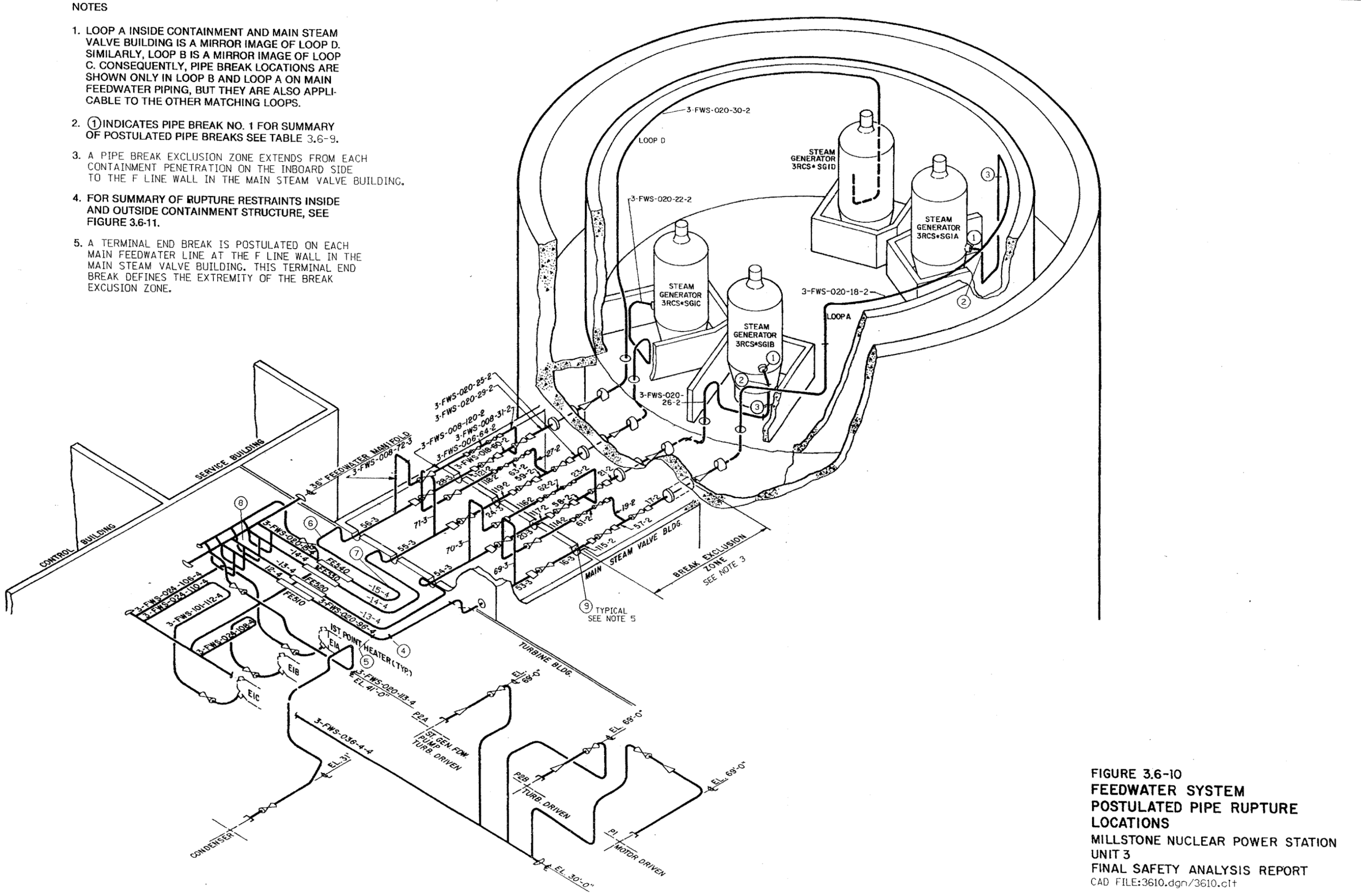


FIGURE 3.6-10
FEEDWATER SYSTEM
POSTULATED PIPE RUPTURE LOCATIONS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
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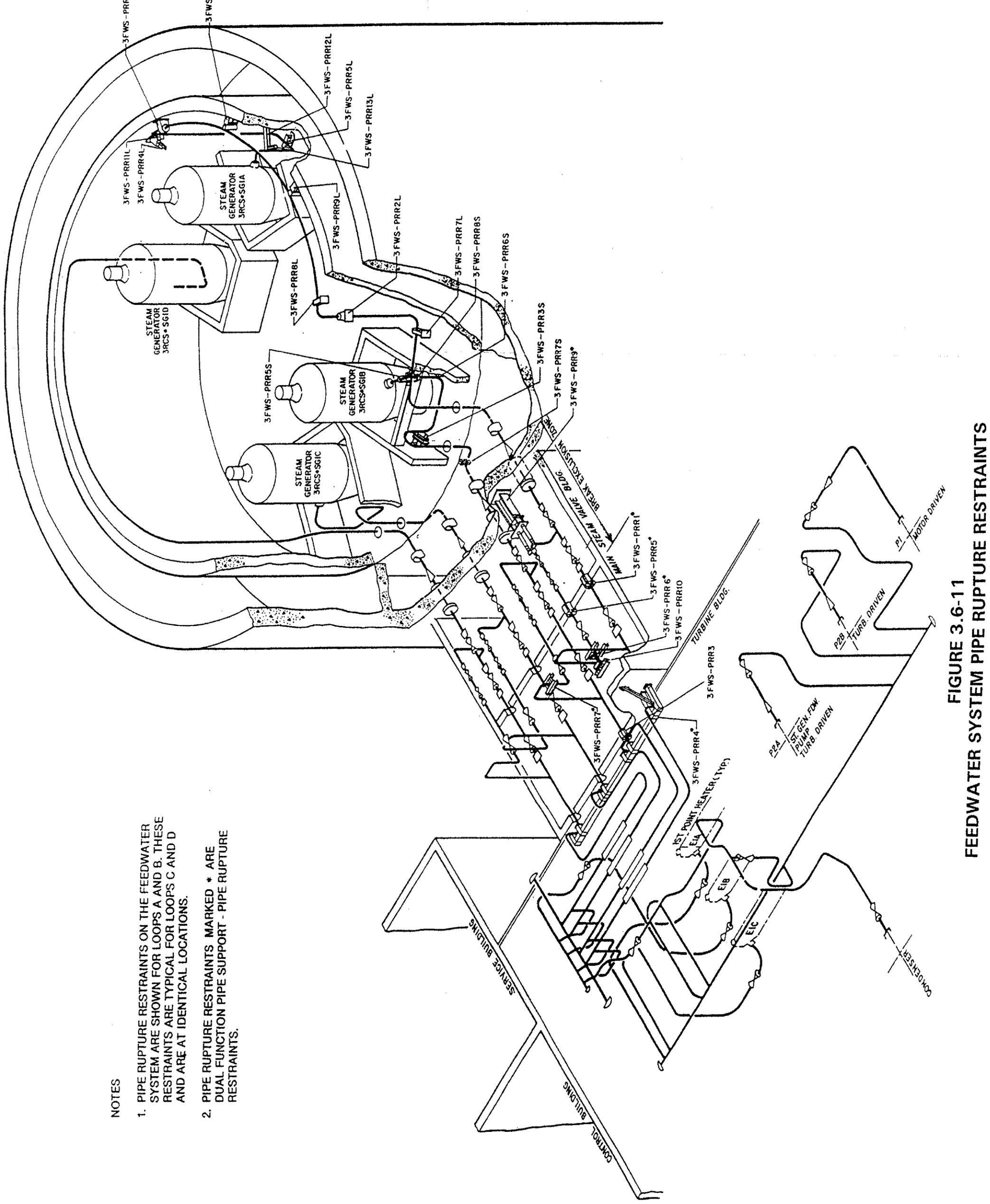
JANUARY 1998

FIGURE 3.6–11 FEEDWATER SYSTEM PIPE RUPTURE RESTRAINTS

MNPS-3 FSAR



NOTES

1. PIPE RUPTURE RESTRAINTS ON THE FEEDWATER SYSTEM ARE SHOWN FOR LOOPS A AND B. THESE RESTRAINTS ARE TYPICAL FOR LOOPS C AND D AND ARE AT IDENTICAL LOCATIONS.
2. PIPE RUPTURE RESTRAINTS MARKED * ARE DUAL FUNCTION PIPE SUPPORT - PIPE RUPTURE RESTRAINTS.

FIGURE 3.6-11
FEEDWATER SYSTEM PIPE RUPTURE RESTRAINTS

JANUARY 1998

NOTES TO FIGURE 3.6–11

DATA FOR PIPE RUPTURE RESTRAINTS - MAIN FEEDWATER SYSTEM

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3FWS-PRR2LA	(1)	85'-1"	41'-3" West	36'-4" South	Omni-Directional	Prevent pipe whip into Crane Wall
3FWS-PRR3LA	3	89'-6"	47'-10" East	27'-2" South	Omni-Directional	Prevent pipe whip into Crane Wall
3FWS-PRR4LA	(1)	86'-11"	47'-6" East	19'-6" South	Pipe Crush Bumper	Prevent pipe whip into Crane Wall
3FWS-PRR5LA	2 & 3	54'-3"	44'-1" East	21'-11" South	Flat Stop	Prevent pipe whip into Operating Floor
3FWS-PRR7LA	(1)	63'-0"	39'-6" West	38'-3" South	Omni-Directional	Prevent pipe whip into Crane Wall
3FWS-PRR8LA	(1)	89'-6"	32'-8" West	44'-3" South	Omni-Directional	Prevent pipe whip into Crane Wall
3FWS-PRR9LA	(1)	89'-6"	3'-10" West	54'-10" South	Pipe Crush Bumper	Prevent pipe whip into Crane Wall
3FWS-PRR10LA	(1)	89'-6"	25'-10" East	48'-7" South	Pipe Crush Bumper	Prevent pipe whip into Crane Wall
3FWS-PRR11LA	(1)	87'-0"	46'-4" East	21'-8" South	Flat Stop	Prevent excessive load to steam generator nozzle
3FWS-PRR12LA	1 & 2	61'-4"	42'-9" East	25'-0" South	Brace on 3MSS-PRR7L	(2)
3FWS-PRR13LA	1 & 2	61'-4"	42'-9" East	25'-0" South	Flat Stop	(2)
3FWS-PRR3SB	(1)	56'-4"	47'-2" West	9'-3" South	Pipe Crush Bumper	(2)
3FWS-PRR5SB	3	64'-7"	39'-4" West	23'-7" South	Laminated Strap	Prevent pipe whip into 3RCS*SG1B
3FWS-PRR6SB	(1)	56'-4"	40'-1" West	26'-9" South	Flat Stop	Prevent excessive load to steam generator nozzle
3FWS-PRR7SB	(1)	44'-3"	50'-9" West	9'-11" South	Laminated Strap	Prevent excessive load on containment penetration

NOTES TO FIGURE 3.6–11 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS - MAIN FEEDWATER SYSTEM (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3FWS-PRR8SB	1, 2 & 3	59'-10"	39'-4" West	23'-7" South	Laminated Strap	(2)
3FWS-PRR3SC	(1)	56'-4"	47'-2" West	9'-3" North	Pipe Crush Bumper	(2)
3FWS-PRR5SC	3	64'-7"	39'-4" West	23'-7" North	Laminated Strap	Prevent pipe whip into 3RCS*SG1C
3FWS-PRR6SC	(1)	56'-4"	40'-1" West	26'-9" North	Flat Stop	Prevent excessive load to steam generator nozzle
3FWS-PRR7SC	(1)	44'-3"	50'-9" West	9'-11" North	Laminated Strap	Prevent excessive load on containment penetration
3FWS-PRR8SC	1, 2 & 3	59'-10"	39'-4" West	23'-7" North	Laminated Strap	(2)
3FWS-PRR2LD	(1)	85'-1"	41'-3" West	36'-4" North	Omni-Directional	Prevent pipe whip into Crane Wall
3FWS-PRR3LD	3	89'-6"	47'-10" East	27'-2" North	Omni-Directional	Prevent pipe whip into Crane Wall
3FWS-PRR4LD	(1)	86'-11"	47'-6" East	19'-6" North	Pipe Crush Bumper	Prevent pipe whip into Crane Wall
3FWS-PRR5LD	2 & 3	54'-3"	44'-1" East	21'-11" North	Flat Stop	Prevent pipe whip into Operating Floor
3FWS-PRR7LD	(1)	63'-0"	39'-6" West	38'-3" North	Omni-Directional	Prevent pipe whip into Crane Wall
3FWS-PRR8LD	(1)	89'-6"	32'-8" West	44'-3" North	Omni-Directional	Prevent pipe whip into Crane Wall
3FWS-PRR9LD	(1)	89'-6"	3'-10" West	54'-10" North	Pipe Crush Bumper	Prevent pipe whip into Crane Wall
3FWS-PRR10LD	(1)	89'-6"	25'-10" East	48'-7" North	Pipe Crush Bumper	Prevent pipe whip into Crane Wall
3FWS-PRR11LA	(1)	87'-0"	46'-8" East	22'-1" North	Flat Stop	Prevent excessive load to steam generator nozzle
3FWS-PRR12LA	1 & 2	61'-4"	42'-9" East	25'-0" North	Brace on 3MSS-PRR7L	(2)

NOTES TO FIGURE 3.6–11 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS - MAIN FEEDWATER SYSTEM (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3FWS-PRR13LA	1 & 2	61'-4"	42'-9" East	25'-0" North	Flat Stop	(2)
3FWS-PRR1A	(3)	44'-3"	105'-2" West	21'-0" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR3A	(3)	44'-3"	141'-2" West	21'-0" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR4A	(3)	47'-7"	146'-6" West	21'-0" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR5A	(3)	49'-11"	105'-2" West	16'-6" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR6A	(3)	49'-11"	124'-1" West	16'-6" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR7A	(3)	60'-9"	124'-1" West	16'-6" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR9SA	(3)	49'-11"	83'-5" West	16'-6" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR10SA	(3)	51'-2"	124'-11" West	16'-6" South	Pipe Crush Bumper	Prevent interaction with adjacent feedwater line
3FWS-PRR1B	(3)	44'-3"	105'-2" West	7'-0" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR3B	(3)	44'-3"	141'-2" West	7'-0" South	Dual Function	Limit stress within Break Exclusion Zone

NOTES TO FIGURE 3.6–11 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS - MAIN FEEDWATER SYSTEM (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3FWS-PRR4B	(3)	47'-7"	146'-6" West	7'-0" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR5B	(3)	49'-11"	105'-2" West	11'-6" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR6B	(3)	49'-11"	124'-1" West	11'-6" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR7B	(3)	60'-9"	124'-1" West	11'-6" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR9SB	(3)	49'-11"	87'-10" West	11'-6" South	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR10SB	(3)	51'-2"	124'-11" West	11'-6" South	Pipe Crush Bumper	Prevent interaction with adjacent feedwater line
3FWS-PRR1C	(3)	44'-3"	105'-2" West	7'-0" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR3C	(3)	44'-3"	141'-2" West	7'-0" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR4C	(3)	47'-7"	146'-6" West	7'-0" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR5C	(3)	49'-11"	105'-2" West	11'-6" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR6C	(3)	49'-11"	124'-1" West	11'-6" North	Dual Function	Limit stress within Break Exclusion Zone

NOTES TO FIGURE 3.6–11 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS - MAIN FEEDWATER SYSTEM (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3FWS-PRR7C	(3)	60'-9"	124'-1" West	11'-6" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR9NC	(3)	49'-11"	87'-10" West	11'-6" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR10NC	(3)	51'-2"	124'-11" West	11'-6" North	Pipe Crush Bumper	Prevent interaction with adjacent feedwater line
3FWS-PRR1D	(3)	44'-3"	105'-2" West	21'-0" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR3D	(3)	44'-3"	141'-2" West	21'-0" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR4D	(3)	47'-7"	146'-6" West	21'-0" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR5D	(3)	49'-11"	105'-2" West	16'-6" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR6D	(3)	49'-11"	124'-1" West	16'-6" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR7D	(3)	60'-9"	124'-1" West	16'-6" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR9ND	(3)	49'-11"	83'-5" West	16'-6" North	Dual Function	Limit stress within Break Exclusion Zone
3FWS-PRR10ND	(3)	51'-2"	124'-11" West	16'-6" North	Pipe Crush Bumper	Prevent interaction with adjacent feedwater line

NOTES TO FIGURE 3.6–11 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS - MAIN FEEDWATER SYSTEM (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint

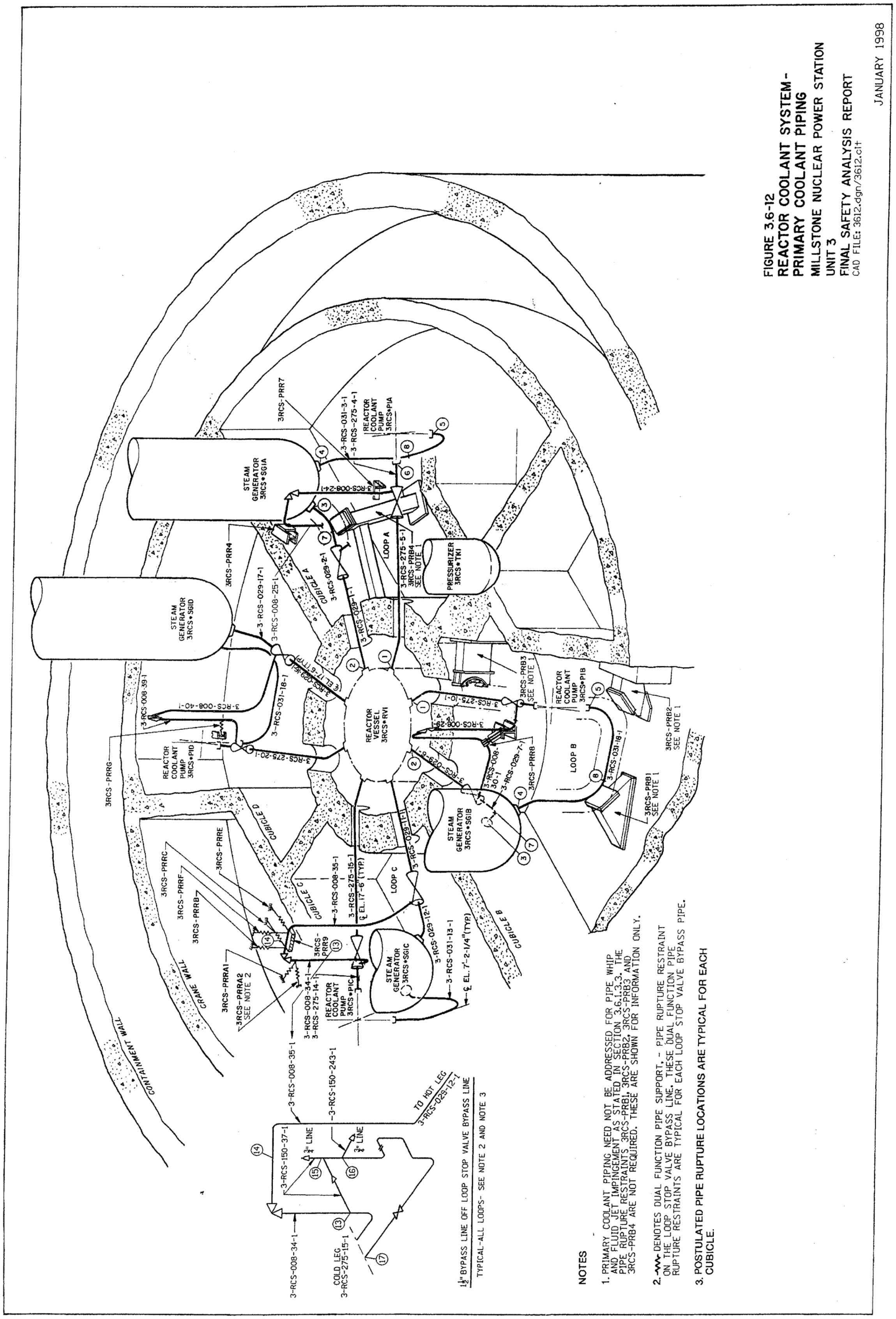
Notes

(1) Pipe Rupture Restraints 3MSS-PRR1LA and 3MSS-PRR1LD are laminated straps which prevent the Main Steam line from impacting the Containment Liner subsequent to Break Number 3 on Main Steam Lines Pipe rupture restraint provided to mitigate the effects of Arbitrary Intermediate Breaks (AIBs). USNRC Generic Letter 87-11 eliminates the requirement to analyze the effects of pipe whip subsequent to AIBs. Therefore, these restraints are installed but do not perform a safety function.

(2) Pipe rupture restraint is provided to prevent an impact to the Steam Generator Cubicle Wall. Unrestrained pipe whip impact to this concrete barrier would result in catastrophic failure of the barrier. The effects of catastrophic failure of the barrier are unacceptable therefore the pipe whip into the barrier is prevented.

(3) Pipe rupture restraints provided to limit stress in the Main Feedwater System Break Exclusion Zone. The pipe rupture restraints 3FWS-PRR1, 3FWS-PRR3, and 3FWS-PRR4 are on the Main Feedwater piping and the restraints 3FWS-PRR5, 3FWS-PRR6, 3FWS-PRR7, 3FWS-PRR9N, 3FWS-PRR9S, 3FWS-PRR10N, and 3FWS-PRR10S are on the Main Feedwater Bypass piping. Postulated break locations are on the Main Feedwater Piping in the Turbine Building and the Main Feedwater Bypass Piping in the Main Steam Valve Building where a fitting criteria was used as the basis to select the break locations in accordance with FSAR Section 3.6.2.1.2.3.b. Subsequent to stress reconciliation, the pipe break locations on the Main Feedwater Piping in the Turbine Building are located in accordance with the criteria delineated in FSAR Section 3.6.2.1.2.3.2.b. The latter criteria greatly reduced the number of postulated pipe breaks and the final postulated breaks, other than the terminal end break at the "F" line wall, were remote from the Break Exclusion Zone. No other postulated breaks were located in the Main Steam Valve Building and none were postulated on the Main Feedwater Bypass Piping as this piping was analyzed as part of the Main Feedwater System and resulted in stresses below the threshold stress defined in FSAR Section 3.6.2.1.2.3.b.

FIGURE 3.6–12 REACTOR COOLANT SYSTEM - PRIMARY COOLANT PIPING



NOTES TO FIGURE 3.6–12

DATA FOR PIPE RUPTURE RESTRAINTS - REACTOR COOLANT SYSTEM - PRIMARY COOLANT PIPING

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3RCS-PRB1A	(1)	7'-0"	32'-11" East	18'-10" South	Limit Stop	None (1)
3RCS-PRB2A	(1)	6'-10"	25'-10" East	30'-0" South	Limit Stop	None (1)
3RCS-PRB3A	(1)	17'-6"	13'-11" East	12'-2" South	Limit Stop	None (1)
3RCS-PRB4A	(1)	15'-6"	27'-6" East	11'-1" South	Limit Stop	None (1)
3RCS-PRR4A	14	27'-8"	19'-9" East	17'-1" South	Limit Stop	(2)
3RCS-PRR7A	(3)	18'-6"	22'-1" East	20'-8" South	Limit Stop	(2)
3RCS-PRR8A	(3)	18'-6"	19'-10" East	15'-9" South	Limit Stop	(2)
3RCS-PRR9A	(4)	31'-6"	20'-1" East	18'-8" South	Limit Stop	(2)
3RCS-PRRGA	(3)	24'-10"	22'-1" East	20'-8" South	Elastic Sway Strut	(2)
3RCS-PRRFA	13	31'-6"	20'-6" East	18'-2" South	Shock Suppressor	(2)
3RCS-PRREA	Split	31'-6"	20'-6" East	18'-2" South	Shock Suppressor	(2)
3RCS-PRRCA	Split	31'-6"	21'-2" East	19'-4" South	Shock Suppressor	(2)
3RCS-PRRBA	13	31'-6"	21'-2" East	19'-4" South	Shock Suppressor	(2)
3RCS-PRRA1A	14	30'-4"	22'-1" East	20'-9" South	Shock Suppressor	(2)
3RCS-PRRA2A	14	30'-4"	22'-1" East	20'-9" South	Shock Suppressor	(2)
3RCS-PRB1B	(1)	7'-0"	32'-11" West	18'-10" South	Limit Stop	None (1)
3RCS-PRB2B	(1)	6'-10"	25'-10" West	30'-0" South	Limit Stop	None (1)
3RCS-PRB3B	(1)	17'-6"	13'-11" West	12'-2" South	Limit Stop	None (1)
3RCS-PRB4B	(1)	15'-6"	27'-6" West	11'-1" South	Limit Stop	None (1)

NOTES TO FIGURE 3.6–12 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS - REACTOR COOLANT SYSTEM - PRIMARY COOLANT PIPING (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3RCS-PRR4B	14	27'-8"	19'-9" West	17'-1" South	Limit Stop	(2)
3RCS-PRR7B	(3)	18'-6"	22'-1" West	20'-8" South	Limit Stop	(2)
3RCS-PRR8B	(3)	18'-6"	19'-10"West	15'-9" South	Limit Stop	(2)
3RCS-PRR9B	(4)	31'-6"	20'-1" West	18'-8" South	Limit Stop	(2)
3RCS-PRRGB	(3)	24'-10"	22'-1" West	20'-8" South	Elastic Sway Strut	(2)
3RCS-PRRFB	13	31'-6"	20'-6" West	18'-2" South	Shock Suppressor	(2)
3RCS-PRREB	Split	31'-6"	20'-6" West	18'-2" South	Shock Suppressor	(2)
3RCS-PRRCB	Split	31'-6"	21'-2" West	19'-4" South	Shock Suppressor	(2)
3RCS-PRRBB	13	31'-6"	21'-2" West	19'-4" South	Shock Suppressor	(2)
3RCS-PRRA1B	14	30'-4"	22'-1" West	20'-9" South	Shock Suppressor	(2)
3RCS-PRRA2B	14	30'-4"	22'-1" West	20'-9" South	Shock Suppressor	(2)
3RCS-PRB1C	(1)	7'-0"	32'-11" West	18'-10" North	Limit Stop	None (1)
3RCS-PRB2C	(1)	6'-10"	25'-10" West	30'-0" North	Limit Stop	None (1)
3RCS-PRB3C	(1)	17'-6"	13'-11" West	12'-2" North	Limit Stop	None (1)
3RCS-PRB4C	(1)	15'-6"	27'-6" West	11'-1" North	Limit Stop	None (1)
3RCS-PRR4C	14	27'-8"	19'-9" West	17'-1" North	Limit Stop	(2)
3RCS-PRR7C	(3)	18'-6"	22'-1" West	20'-8" North	Limit Stop	(2)
3RCS-PRR8C	(3)	18'-6"	19'-10"West	15'-9" North	Limit Stop	(2)
3RCS-PRR9C	(4)	31'-6"	20'-1" West	18'-8" North	Limit Stop	(2)

NOTES TO FIGURE 3.6–12 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS - REACTOR COOLANT SYSTEM - PRIMARY COOLANT PIPING (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3RCS-PRRGC	(3)	24'-10"	22'-1" West	20'-8" North	Elastic Sway Strut	(2)
3RCS-PRRFC	13	31'-6"	20'-6" West	18'-2" North	Shock Suppressor	(2)
3RCS-PRREC	Split	31'-6"	20'-6" West	18'-2" North	Shock Suppressor	(2)
3RCS-PRRCC	Split	31'-6"	21'-2" West	19'-4" North	Shock Suppressor	(2)
3RCS-PRRBC	13	31'-6"	21'-2" West	19'-4" North	Shock Suppressor	(2)
3RCS-PRRA1C	14	30'-4"	22'-1" West	20'-9" North	Shock Suppressor	(2)
3RCS-PRRA2C	14	30'-4"	22'-1" West	20'-9" North	Shock Suppressor	(2)
3RCS-PRB1D	(1)	7'-0"	32'-11" East	18'-10" North	Limit Stop	None (1)
3RCS-PRB2D	(1)	6'-10"	25'-10" East	30'-0" North	Limit Stop	None (1)
3RCS-PRB3D	(1)	17'-6"	13'-11" East	12'-2" North	Limit Stop	None (1)
3RCS-PRB4D	(1)	15'-6"	27'-6" East	11'-1" North	Limit Stop	None (1)
3RCS-PRR4D	14	27'-8"	19'-9" East	17'-1" North	Limit Stop	(2)
3RCS-PRR7D	(3)	18'-6"	22'-1" East	20'-8" North	Limit Stop	(2)
3RCS-PRR8D	(3)	18'-6"	19'-10" East	15'-9" North	Limit Stop	(2)
3RCS-PRR9D	(4)	31'-6"	20'-1" East	18'-8" North	Limit Stop	(2)
3RCS-PRRGD	(3)	24'-10"	22'-1" East	20'-8" North	Elastic Sway Strut	(2)
3RCS-PRRFD	13	31'-6"	20'-6" East	18'-2" North	Shock Suppressor	(2)
3RCS-PRRED	Split	31'-6"	20'-6" East	18'-2" North	Shock Suppressor	(2)
3RCS-PRRCD	Split	31'-6"	21'-2" East	19'-4" North	Shock Suppressor	(2)

NOTES TO FIGURE 3.6–12 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS - REACTOR COOLANT SYSTEM - PRIMARY COOLANT PIPING (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3RCS-PRRBD	13	31'-6"	21'-2" East	19'-4" North	Shock Suppressor	(2)
3RCS-PRRA1D	14	30'-4"	22'-1" East	20'-9" North	Shock Suppressor	(2)
3RCS-PRRA2D	14	30'-4"	22'-1" East	20'-9" North	Shock Suppressor	(2)

Notes

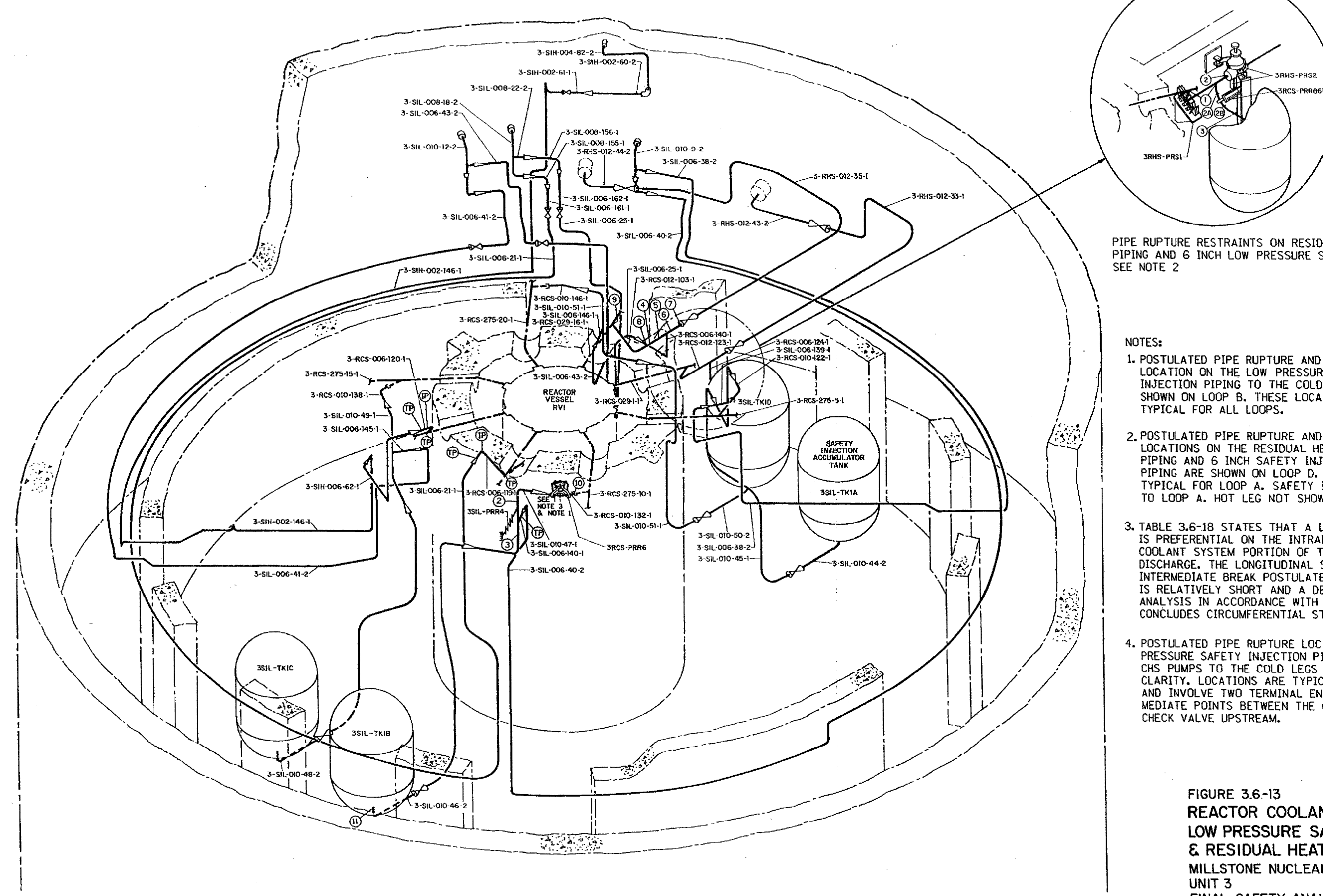
(1) Subsequent to the exemption regarding the need to analyze Large Primary Loop Pipe Ruptures as the Structural Design Basis for Millstone 3 from a portion of GDC-4 of Appendix A to 10 CFR Part 50, these pipe rupture restraints are no longer required.

(2) The NSSS System Standard Design Criteria does not permit break propagation in excess of 20% of the flow area of the ruptured pipe to an unaffected leg within the broken loop. Therefore, these pipe rupture restraints limit displacement subsequent to a postulated break on the 8" Reactor Coolant Bypass line to meet the NSSS System Standard Design Criteria.

(3) Pipe rupture restraints provided to mitigate the effects of Terminal End (TP) postulated breaks at the Reactor Coolant Bypass nozzle on either the Hot Leg and the Cold Leg. The Reactor Coolant Bypass piping was analyzed as part of the Primary Reactor Coolant Piping analysis and the Reactor Coolant Bypass piping has a significant effect on the main Primary Coolant line behavior. Therefore, in accordance with FSAR Section 3.6.2.1.1.1 the Reactor Coolant Bypass line connection to each Hot Leg and Cold Leg is not a Terminal End (TP) of the Reactor Coolant Bypass piping.

(4) 3RCS-PRR9A through D limit the steady state displacement of the shock suppressors.

FIGURE 3.6–13 REACTOR COOLANT SYSTEM - LOW PRESSURE SAFETY INJECTION AND RESIDUAL HEAT REMOVAL PIPING



NOTES:

1. POSTULATED PIPE RUPTURE AND RESTRAINT LOCATION ON THE LOW PRESSURE SAFETY INJECTION PIPING TO THE COLD LEGS ARE SHOWN ON LOOP B. THESE LOCATIONS ARE TYPICAL FOR ALL LOOPS.

2. POSTULATED PIPE RUPTURE AND RESTRAINT LOCATIONS ON THE RESIDUAL HEAT REMOVAL PIPING AND 6 INCH SAFETY INJECTION BRANCH PIPING ARE SHOWN ON LOOP D. LOCATIONS ARE TYPICAL FOR LOOP A. SAFETY INJECTION PIPING TO LOOP A. HOT LEG NOT SHOWN FOR CLARITY.

3. TABLE 3.6-18 STATES THAT A LONGITUDINAL SPLIT IS PREFERENTIAL ON THE INTRADOS OF THE REACTOR COOLANT SYSTEM PORTION OF THE ACCUMULATOR DISCHARGE. THE LONGITUDINAL SPLIT IS THE ONLY INTERMEDIATE BREAK POSTULATED SINCE THE PIPING IS RELATIVELY SHORT AND A DETAILED STRESS ANALYSIS IN ACCORDANCE WITH FSAR SECTION 3.6.2.1.3 CONCLUDES CIRCUMFERENTIAL STRESS IS DOMINANT.

4. POSTULATED PIPE RUPTURE LOCATIONS FOR HIGH PRESSURE SAFETY INJECTION PIPING FROM THE CHS PUMPS TO THE COLD LEGS IS NOT SHOWN FOR CLARITY. LOCATIONS ARE TYPICAL FOR EACH LOOP AND INVOLVE TWO TERMINAL ENDS AND TWO INTERMEDIATE POINTS BETWEEN THE COLD LEG AND FIRST CHECK VALVE UPSTREAM.

FIGURE 3.6-13
REACTOR COOLANT SYSTEM-
LOW PRESSURE SAFETY INJECTION
& RESIDUAL HEAT REMOVAL PIPING
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
CAD FILE: flg3613.dgn/3613.clt

JANUARY 1998

NOTES TO FIGURE 3.6–13

DATA FOR PIPE RUPTURE RESTRAINTS REACTOR COOLANT SYSTEM, LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3RCS-PRR861	8	10'-3"	25'-1" East	6'-3" South	Laminated Strap	(1)
3RHS-PRS1A	9	13'-7"	20'-7" East	4'-6" South	Limit Stop	(1)
3RHS-PRS2A	4,5,6, or 7	14'-7"	29'-2" East	4'-6" South	Laminated Strap	(2)
3RCS-PRR6A	10	18'-9"	17'-2" East	17'-2" South	Laminated Strap	(3)
3SIL-PRR4A	TP & Splits	16'-2"	20'-2" East	11'-10" South	Shock Suppressor	(4)
3RCS-PRR6B	10	18'-9"	17'-2" West	17'-2" South	Laminated Strap	(3)
3SIL-PRR4B	TP & Splits	16'-2"	20'-2" West	11'-10" South	Shock Suppressor	(4)
3RCS-PRR6C	10	18'-9"	17'-2" West	17'-2" North	Laminated Strap	(3)
3SIL-PRR4C	TP & Splits	16'-2"	20'-2" West	11'-10" North	Shock Suppressor	(4)
3RCS-PRR862	8	10'-3"	25'-1" East	6'-3" North	Laminated Strap	(1)
3RHS-PRS1D	9	13'-7"	20'-7" East	4'-6" North	Limit Stop	(1)
3RHS-PRS2D	4,5,6, or 7	14'-7"	29'-2" East	4'-6" North	Laminated Strap	(2)
3RCS-PRR6D	10	18'-9"	17'-2" East	17'-2" North	Laminated Strap	(3)
3SIL-PRR4D	TP & Splits	16'-2"	20'-2" East	11'-10" North	Shock Suppressor	(4)

NOTES TO FIGURE 3.6–13 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS REACTOR COOLANT SYSTEM, LOW PRESSURE SAFETY INJECTION, HIGH PRESSURE SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PIPING (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint

Notes

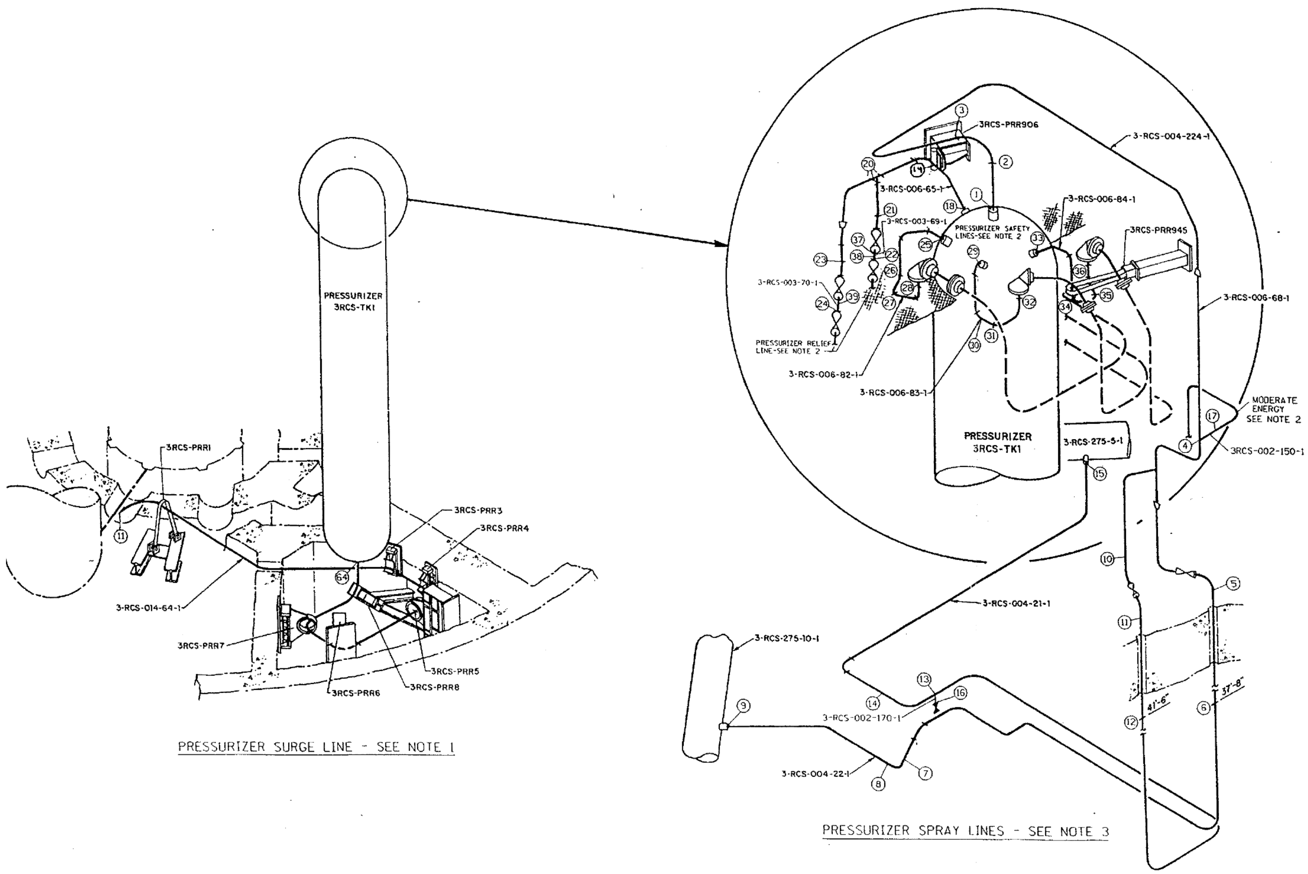
(1) Pipe rupture restraint prevents unrestrained pipe whip into the East-West Cubicle Wall. Catastrophic failure of the East-West Wall is credible if subjected to unrestrained pipe whip impact by either the Residual Heat Removal Piping or the Low Pressure Safety Injection piping to the Hot Leg.

(2) Pipe rupture restraint is provided to prevent the Residual Heat Removal line from impacting the Containment Liner. Leak tight integrity is required subsequent to a Large LOCA and the unrestrained pipe whip subsequent to a postulated break on the Residual Heat Removal line would violate this criteria.

(3) This terminal end break occurs at the Cold Leg Nozzle and results in a large LOCA. The NSSS System Standard Design Criteria requires that break propagation in the affected loop cannot exceed 20% of the flow area of the broken line. The Reactor Coolant System Loop Stop Valve Bypass Line identified has a nominal pipe size of 8" and therefore, if the 8" Loop Stop Valve Bypass Line is impacted, pressure boundary integrity must be assured. The pipe rupture restraint indicated prevents the impact.

(4) The Terminal End (TP) break or Longitudinal Split (LS) does not result in a LOCA. The NSSS System Standard Design Criteria delineates requirements for a Large Line Break that does not result in a LOCA. The requirement is that a Non-LOCA cannot propagate into a LOCA. If unrestrained, the broken pipe would induce significant loading upon the Reactor Coolant System pressure boundary. The pipe rupture restraint is provided to prevent this effect.

FIGURE 3.6–14 REACTOR COOLANT SYSTEM - PRESSURIZER CUBICLE PIPING



NOTES

1. POSTULATED PIPE RUPTURE LOCATION ON 3-RCS-014-64-1 PRESSURIZER SURGE LINE ARE LISTED IN TABLE 3.6-15 AS BREAKS 37 THROUGH 64 AND 11 FOR CLARITY, DUE TO THE QUANTITY OF BREAKS, LOCATIONS ONLY FOR TERMINAL ENDS ARE SHOWN.
2. POSTULATED PIPE RUPTURE LOCATIONS ON THE SAFETY AND RELIEF LINE PIPING ARE BREAKS 18 THROUGH 39 IN TABLE 3.6-15 ON THE HIGH ENERGY PORTION OF SAFETY AND RELIEF PIPING. PIPING DESIGNATED AS — — — IS MODERATE ENERGY BEYOND THE VALVES.
3. POSTULATED PIPE RUPTURE LOCATIONS ON THE PRESSURIZER SPRAY LINE ARE BREAKS 1 THROUGH 17 IN TABLE 3.6-15.

FIGURE 3.6-14
REACTOR COOLANT SYSTEM
PRESSURIZER CUBICLE PIPING
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
CAD FILE: 3614.dgn/3614.cl+
APRIL 1998

NOTES TO FIGURE 3.6–14

DATA FOR PIPE RUPTURE RESTRAINTS REACTOR COOLANT SYSTEM PRESSURIZER CUBICLE PIPING

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3RCS-PRR1	11	22'-4"	15'-7" West	13'-5" South	Laminated Strap	(1)
3RCS-PRR3	(2)	22'-6"	5'-5" West	40'-6" South	Limit Stop	(3)
3RCS-PRR4	(2)	22'-6"	5'-0" West	44'-7" South	Limit Stop	(3)
3RCS-PRR5	(2)	22'-6"	8'-4" West	47'-8" South	Omni-Directional	(3)
3RCS-PRR6	(2)	22'-7"	19'-8" West	47'-8" South	Omni-Directional	(3)
3RCS-PRR7	(2)	22'-7"	19'-9" West	41'-6" South	Omni-Directional	(3)
3RCS-PRR8	(2)	23'-1"	11'-6" West	41'-0" South	Limit Stop	Prevents Impact to 3RCS*TK1 Ring Girder
3RCS-PRR906	(2)	78'-0"	9'-3" West	36'-3" South	Pipe Crush Bumper	(4)
3RCS-PRR945	(2)	76'-2"	10'-2" West	46'-6" South	Laminated Strap	(4)

Notes

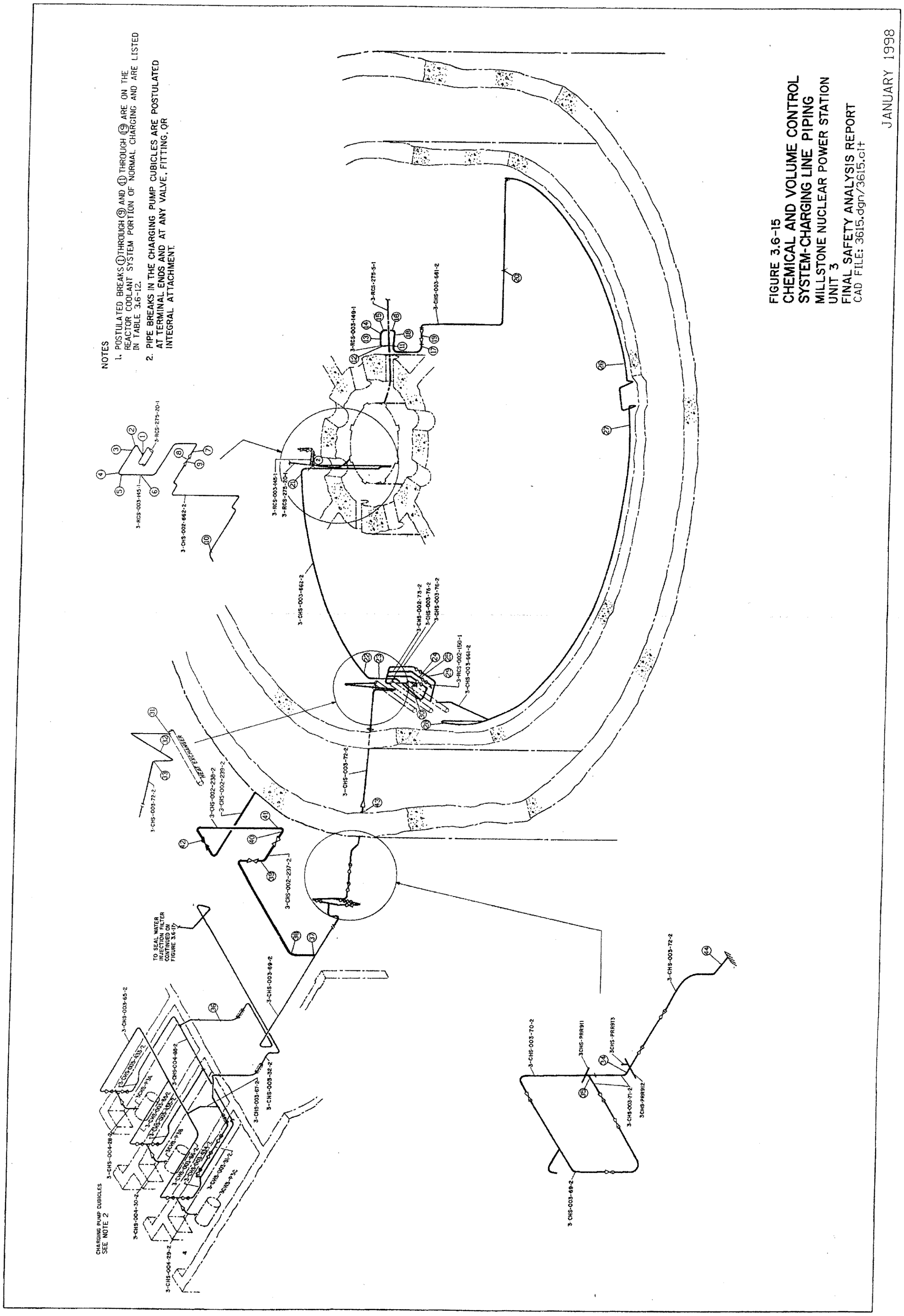
(1).Pipe rupture restraint prevents the Pressurizer Surge Line from impacting the support system for the Steam Generator, 3RCS*SG1B.

(2).See FSAR Table 3.6-16 for the specific break location.

(3).The Crane Wall, Steam Generator Cubicle Wall and Pressurizer Cubicle Wall if subjected to unrestrained pipe whip of the Pressurizer Surge Line would catastrophically fail. As a result, damage to essential systems subsequently to secondary missiles, etc., is credible. The pipe rupture restraints are provided to prevent the unrestrained impact to these structures.

(4).The NSSS System Standard Design Criteria does not permit break propagation from the Pressurizer Relief Line (Steam Phase) to the Pressurizer Spray Line (Liquid Phase). The pipe rupture restraint is provided to prevent the interaction.

FIGURE 3.6–15 CHEMICAL AND VOLUME CONTROL SYSTEM - CHARGING LINE PIPING



NOTES TO FIGURE 3.6–15

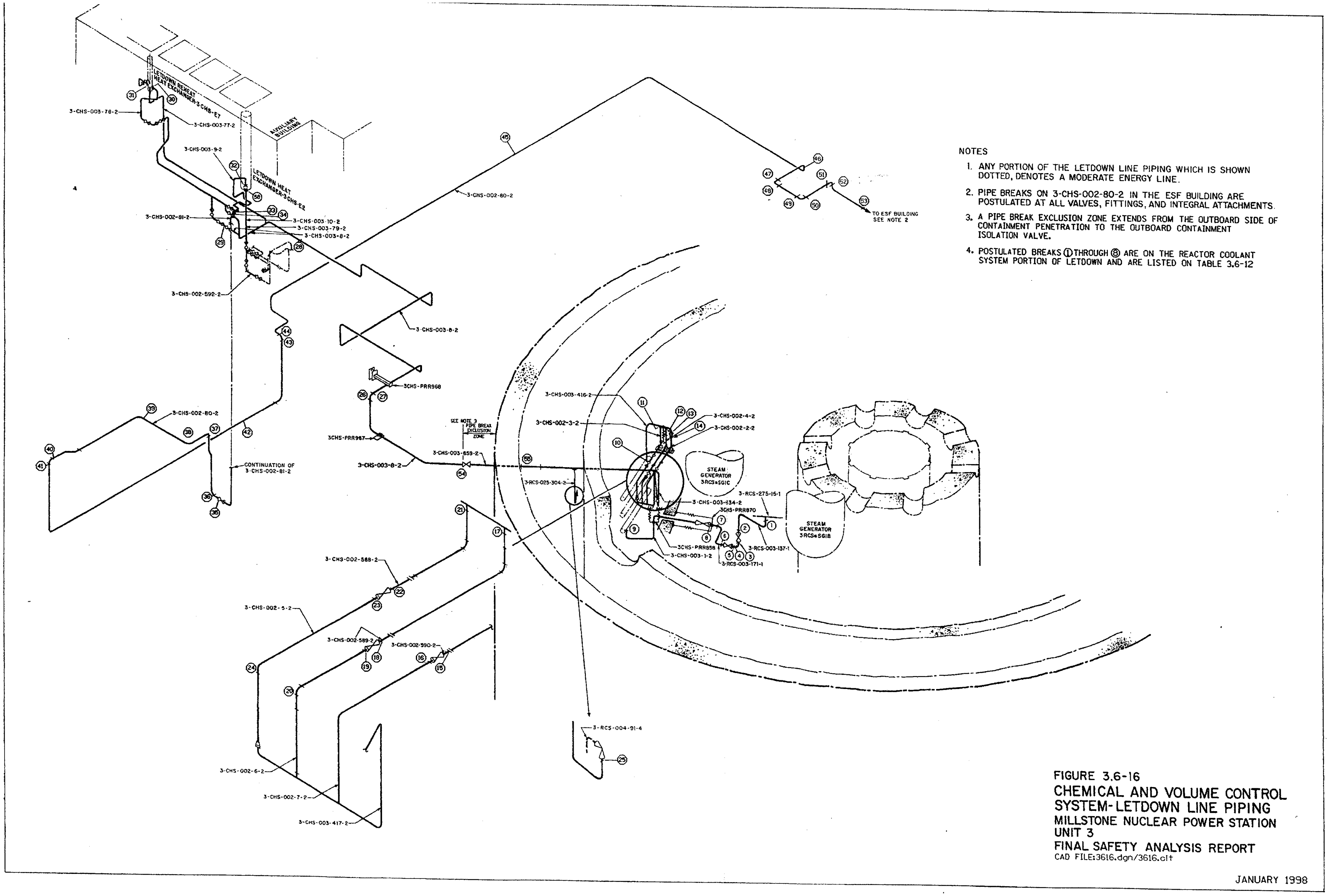
DATA FOR PIPE RUPTURE RESTRAINTS CHEMICAL VOLUME CONTROL SYSTEM - NORMAL CHARGING

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3CHS-PRR911	35	13'-2"	52'-4" West	69'-7" North	Pipe Crush Bumper	(1)
3CHS-PRR912	35	12'-3"	52'-0" West	69'-0" North	Pipe Crush Bumper	(1)
3CHS-PRR913	34	12'-6"	51'-8" West	68'-7" North	Solid Bumper	(1)

Note

(1) Pipe rupture restraints were originally specified to limit stress in the containment penetration area and containment isolation valve. However, this break exclusion zone was eliminated based on technical justification provided in calculation 97-ENG-1329-M3, since a charging line break outside containment is a non-LOCA break which does not require containment isolation.

FIGURE 3.6–16 CHEMICAL AND VOLUME CONTROL SYSTEM - LETDOWN LINE PIPING



NOTES

1. ANY PORTION OF THE LETDOWN LINE PIPING WHICH IS SHOWN DOTTED, DENOTES A MODERATE ENERGY LINE.
2. PIPE BREAKS ON 3-CHS-002-80-2 IN THE ESF BUILDING ARE POSTULATED AT ALL VALVES, FITTINGS, AND INTEGRAL ATTACHMENTS.
3. A PIPE BREAK EXCLUSION ZONE EXTENDS FROM THE OUTBOARD SIDE OF CONTAINMENT PENETRATION TO THE OUTBOARD CONTAINMENT ISOLATION VALVE.
4. POSTULATED BREAKS ① THROUGH ⑧ ARE ON THE REACTOR COOLANT SYSTEM PORTION OF LETDOWN AND ARE LISTED ON TABLE 3.6-12

FIGURE 3.6-16
CHEMICAL AND VOLUME CONTROL SYSTEM-LETDOWN LINE PIPING
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT
CAD FILE:3616.dgn/3616.clt

JANUARY 1998

NOTES TO FIGURE 3.6–16

DATA FOR PIPE RUPTURE RESTRAINTS CHEMICAL VOLUME CONTROL SYSTEM - LETDOWN LINE

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3CHS-PRR870	(1)	7'-0"	17'-0" West	36'-6" North	Shock Suppressor	(2)(2)
3CHS-PRR858	(1)	5'-2"	22'-0" West	41'-0" North	Shock Suppressor	(2)
3CHS-PRR967	26	12'-6"	60'-2" West	75'-3" North	Dual Function	Limits Stress to Pipe Break Exclusion Zone
3CHS-PRR968	(3)	20'-2"	55'-10" West	78'-0" North	Dual Function	None

Notes

(1) The Terminal End (TP) postulated pipe break number 9 is the only break on the Class 2 portion of the Chemical & Volume Control System Letdown Line piping. This TP is at the inlet to the Regenerative Heat Exchanger, 3CHS*E1, and is remote from the Class 1 portion of the Chemical & Volume Control System Letdown Line piping. Therefore, break propagation from the Class 2 portion of the Chemical & Volume Control System Letdown Line piping to the Class 1 portion (i.e., Propagating a Non-LOCA into a LOCA) does not occur.

(2) Pipe rupture restraints 3CHS-PRR870 and 3CHS-PRR858 are provided to prevent propagation of a Non-LOCA into a LOCA. The NSSS System Standard Design Criteria does not permit break propagation of a Non-LOCA into a LOCA. The pipe rupture restraints 3CHS-PRR870 and 3CHS-PRR858 limit displacement of the Class 1 piping subsequent to a pipe break on the Class 2 portion of the Chemical & Volume Control System Letdown Line piping.

(3) Pipe Rupture Restraint 3CHS-PRR968 is provided to mitigate the effects of an Arbitrary Intermediate Break (AIB). USNRC Generic Letter 87-11 eliminates the requirement to evaluate the dynamic effects of AIBs. Therefore, this pipe rupture restraint has no safety-related function.

FIGURE 3.6–17 CHEMICAL AND VOLUME CONTROL SYSTEM - SEAL WATER INJECTION AND PIPING

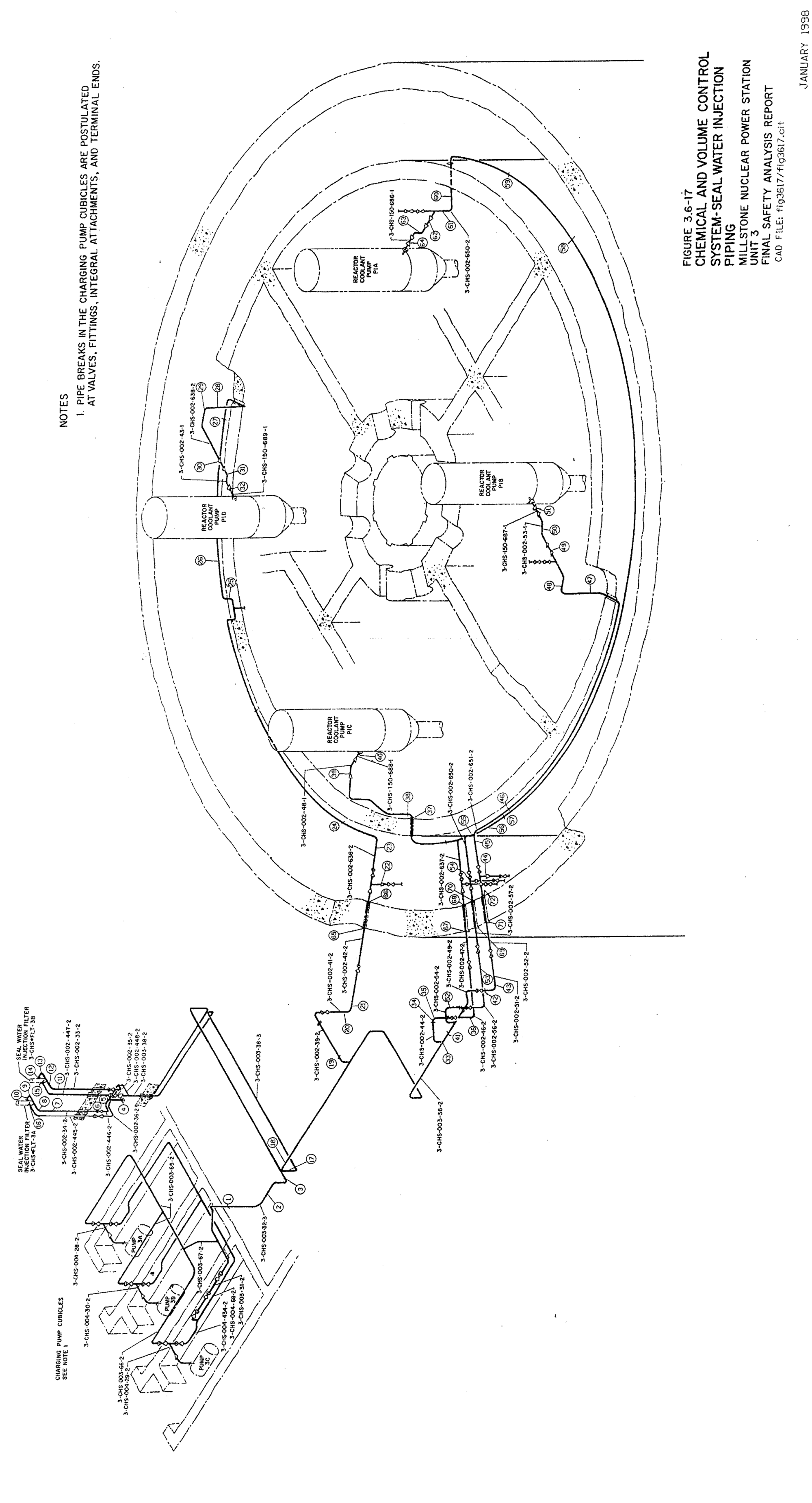
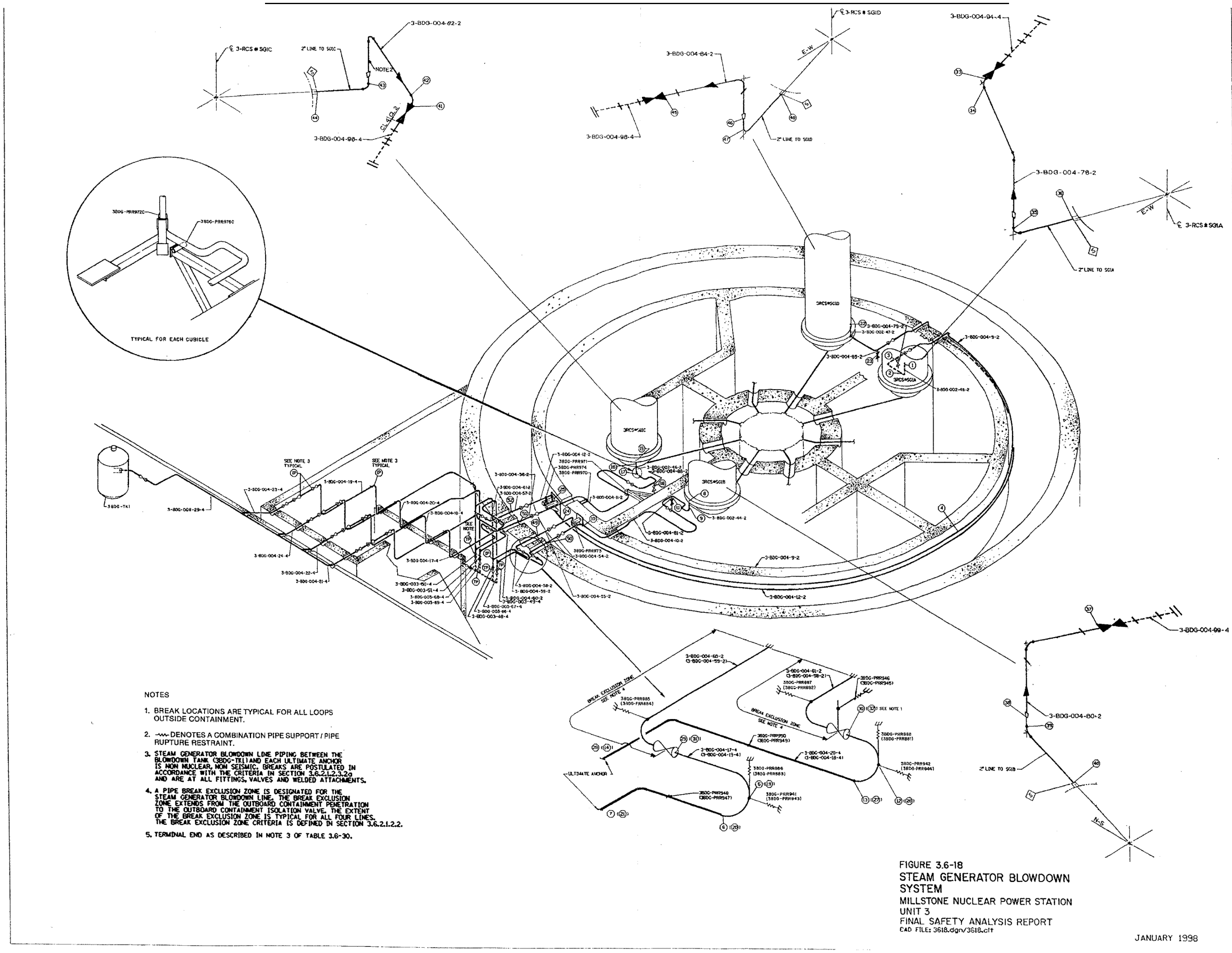


FIGURE 3.6–18 STEAM GENERATOR BLOWDOWN SYSTEM



NOTES TO FIGURE 3.6–18

DATA FOR PIPE RUPTURE RESTRAINTS STEAM GENERATOR BLOWDOWN SYSTEM

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3BDG-PRR972A	2	22'-2"	37'-6" East	8'-10" South	Laminated Strap	Prevent impact to Steam Generator Shell
3BDG-PRR975A	1	21'-1"	37'-6" East	7'-4" South	Pipe Crush Bumper	Prevent impact to Steam Generator Shell
3BDG-PRR885	29	20'-6"	76'-3" West	10'-1" North	Dual Function	Limits Stress to the Break Exclusion Zone
3BDG-PRR886	5	20'-0"	77'-3" West	7'-2" North	Shock Suppressor	Limits Stress to the Break Exclusion Zone
3BDG-PRR941	6	20'-0"	77'-4" West	6'-9" North	Shock Suppressor	Limits Stress to the Break Exclusion Zone
3BDG-PRR948	7	19'-6"	76'-2" West	10'-7" North	Solid Bumper	Limits Stress to the Break Exclusion Zone
3BDG-PRR972B	(1)	22'-2"	39'-1" West	11'-7" South	Laminated Strap	Prevent impact to Steam Generator Shell
3BDG-PRR975B	8 & 9	21'-1"	39'-9" West	9'-5" South	Pipe Crush Bumper	Prevent impact to Steam Generator Shell
3BDG-PRR973	10	20'-6"	59'-9" West	5'-4" North	Dual Function	Prevent impact to Containment Liner
3BDG-PRR887	30	20'-6"	75'-6" West	6'-9" North	Dual Function	Limits Stress to the Break Exclusion Zone
3BDG-PRR888	12	20'-0"	76'-5" West	3'-4" North	Shock Suppressor	Limits Stress to the Break Exclusion Zone

NOTES TO FIGURE 3.6–18 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3BDG-PRR942	13	20'-0"	76'-5" West	2'-11" North	Shock Suppressor	Limits Stress to the Break Exclusion Zone
3BDG-PRR946	12,13, or 14	21'-7"	76'-4" West	5'-0" North	Dual Function	Limits Stress to the Isolation Valve Stem
3BDG-PRR950	14	19'-6"	74'-11" West	10'-6" North	Solid Bumper	Limits Stress to the Break Exclusion Zone
3BDG-PRR972C	17	22'-2"	37'-5" West	8'-10" North	Laminated Strap	Prevent impact to Steam Generator Shell
3BDG-PRR975C	15 & 16	21'-1"	37'-11" West	7'-3" North	Pipe Crush Bumper	Prevent impact to Steam Generator Shell
3BDG-PRR970	51	20'-6"	52'-1" West	10'-7" North	Laminated Strap	Prevent impact to Steam Generator Supports
3BDG-PRR971	(1)	21'-1"	45'-0" West	4'-9" North	Pipe Crush Bumper	Prevent impact to Steam Generator Supports
3BDG-PRR884	31	20'-6"	74'-11" West	17'-8" North	Dual Function	Limits Stress to the Break Exclusion Zone
3BDG-PRR883	19	20'-0"	74'-9" West	20'-8" North	Shock Suppressor	Limits Stress to the Break Exclusion Zone
3BDG-PRR943	20	20'-0"	74'-8" West	21'-2" North	Shock Suppressor	Limits Stress to the Break Exclusion Zone
3BDG-PRR947	21	19'-6"	74'-11" West	12'-2" North	Solid Bumper	Limits Stress to the Break Exclusion Zone

NOTES TO FIGURE 3.6–18 (CONTINUED)

DATA FOR PIPE RUPTURE RESTRAINTS STEAM GENERATOR BLOWDOWN SYSTEM (CONTINUED)

Restraint Identification	Break #	Elevation	Distance from N-S Containment	Distance from E-W Containment	Restraint Type	Function of the Pipe Rupture Restraint
3BDG-PRR972D	(1)	22'-2"	39'-1" East	11'-7" North	Laminated Strap	Prevent impact to Steam Generator Shell
3BDG-PRR975D	22	21'-1"	39'-7" East	9'-5" North	Pipe Crush Bumper	Prevent impact to Steam Generator Shell
3BDG-PRR974	25	20'-6"	58'-10" West	16'-7" North	Laminated Strap	Prevent impact to Containment Liner
3BDG-PRR882	32	20'-6"	72'-11" West	20'-7" North	Dual Function	Limits Stress to the Break Exclusion Zone
3BDG-PRR881	26	20'-0"	72'-9" West	23'-9" North	Shock Suppressor	Limits Stress to the Break Exclusion Zone
3BDG-PRR944	27	20'-0"	72'-8" West	24'-3" North	Shock Suppressor	Limits Stress to the Break Exclusion Zone
3BDG-PRR945	26,27,or 28	21'-7"	73'-1" West	22'-7" North	Dual Function	Limits Stress to the Isolation Valve Stem
3BDG-PRR949	28	19'-6"	73'-9" West	11'-10" North	Solid Bumper	Limits Stress to the Break Exclusion Zone

Note

(1) Pipe Rupture Restraint mitigates the effects of an Arbitrary Intermediate Break (AIB). USNRC Generic Letter 87-11 eliminates the need to evaluate the dynamic effects of AIBs.

FIGURE 3.6–19 PIPE RUPTURE ANALYSIS FLOW CHART

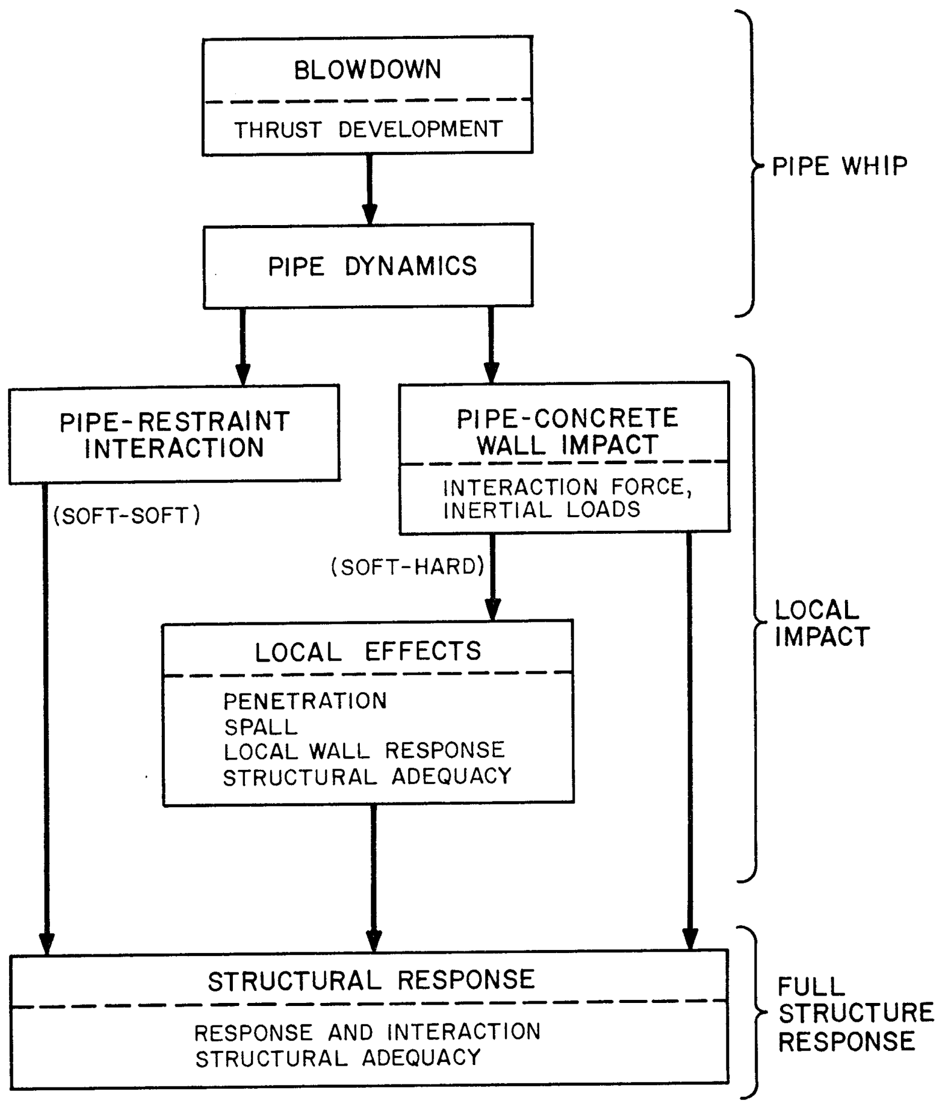


FIGURE 3.6–20 VARIATION OF STEADY STATE STEAM BLOWDOWN VS FRICTION

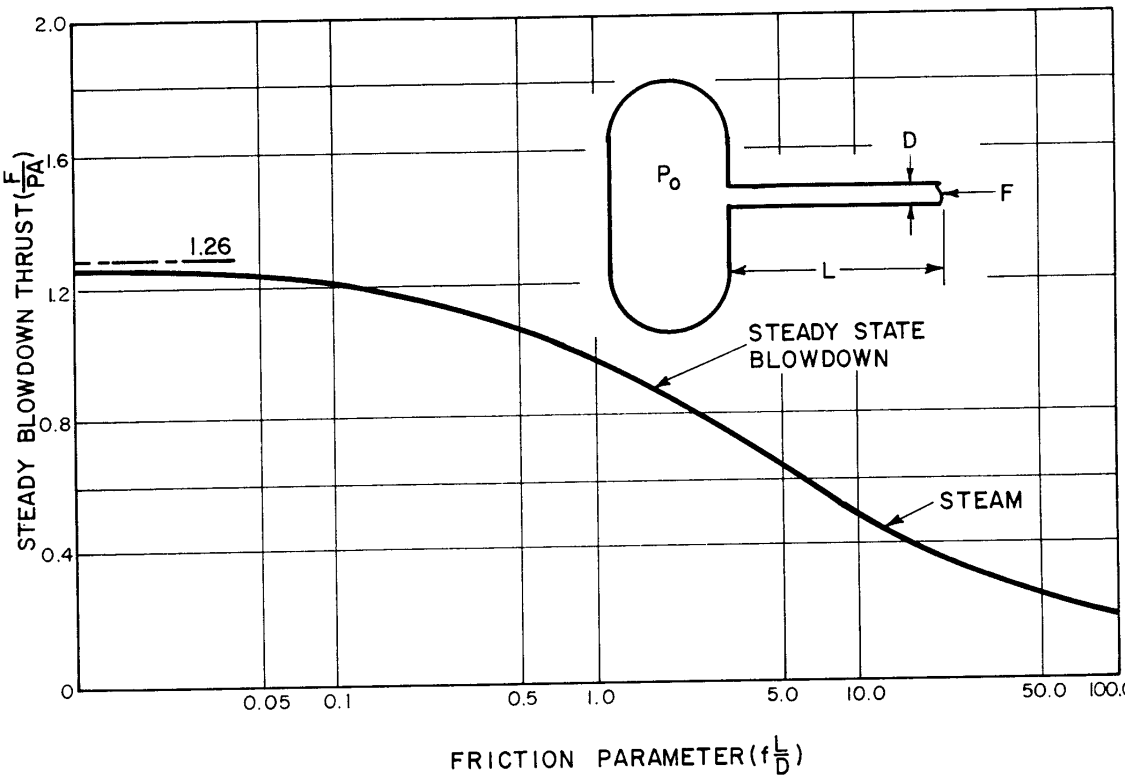


FIGURE 3.6–21 PIPE RESTRAINT INTERMEDIATE STRUCTURE SYSTEM, MATHEMATICAL MODEL

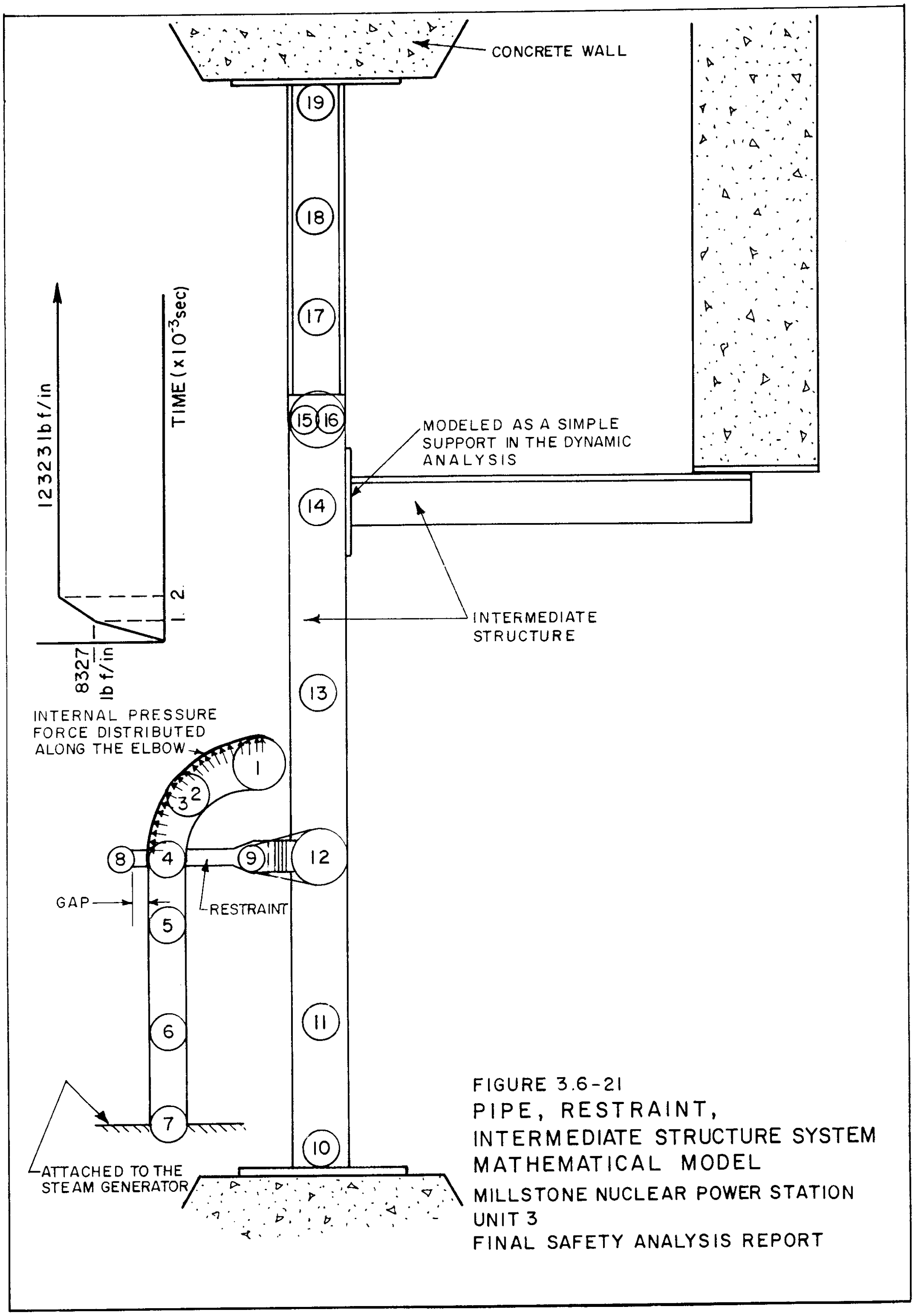


FIGURE 3.6-21
PIPE, RESTRAINT, INTERMEDIATE STRUCTURE SYSTEM MATHEMATICAL MODEL
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–22 LAMINATED STRAP RESTRAINT FOR SMALL LINES

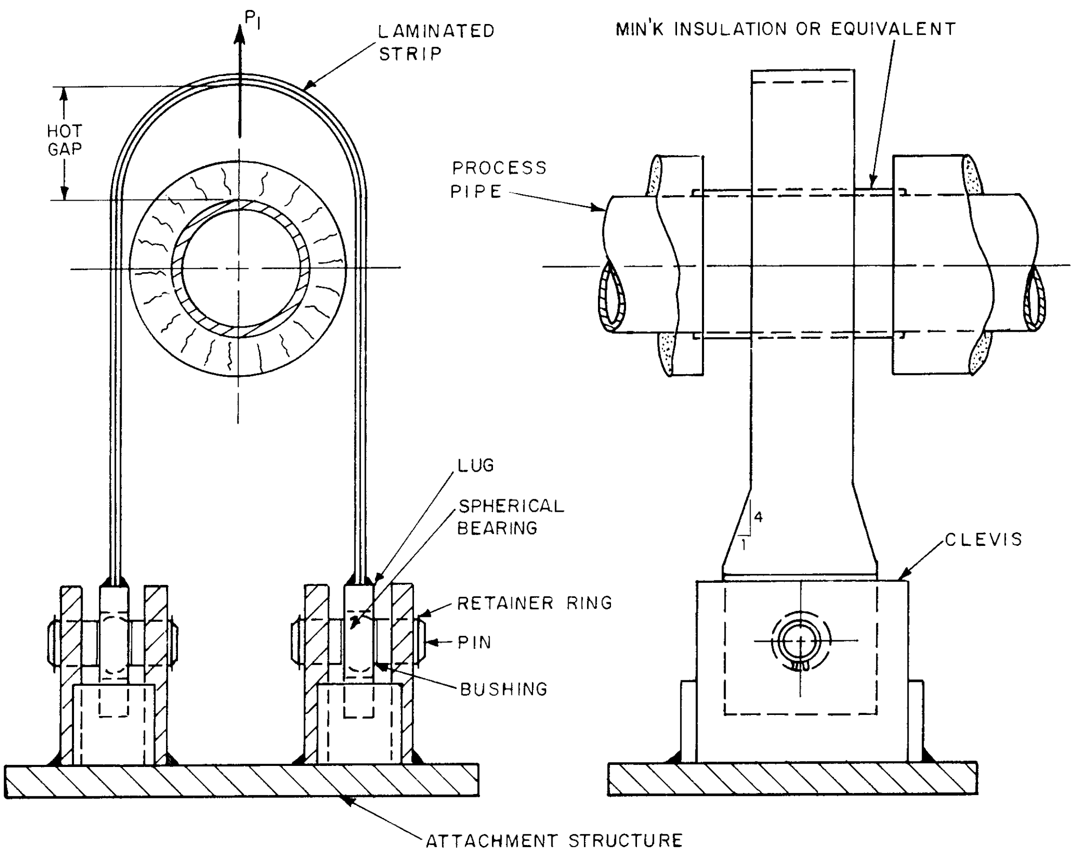


FIGURE 3.6-22
LAMINATED STRAP RESTRAINT FOR SMALL LINES
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–23 FEEDWATER SYSTEM SHORT LOOP STRAP NEAR THE STEAM GENERATOR, NET SHEAR FORCE AT NODE 10

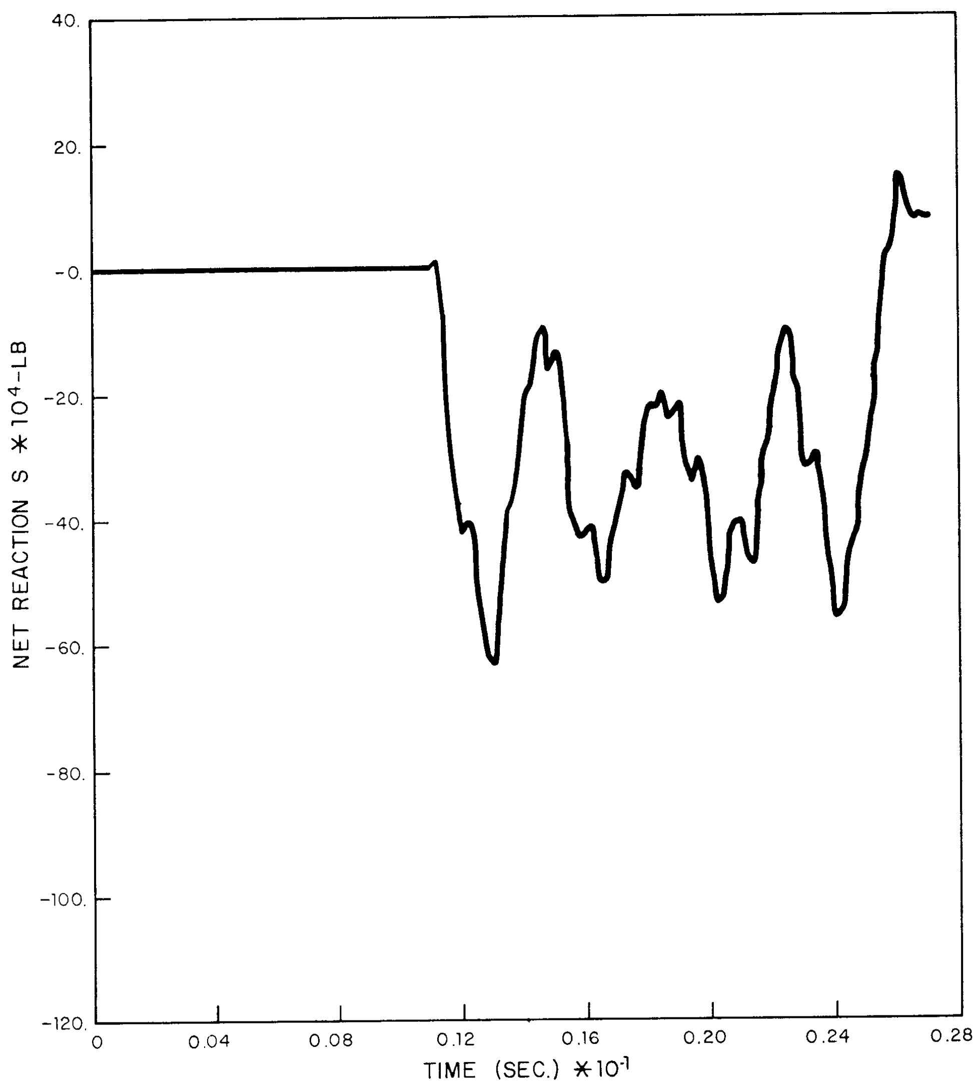


FIGURE 3.6-23
FEEDWATER SYSTEM SHORT LOOP STRAP NEAR THE STEAM GENERATOR, NET SHEAR FORCE AT NODE 10
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–24 FEEDWATER SYSTEM SHORT LOOP STRAP NEAR THE STEAM GENERATOR, NET SHEAR FORCE AT NODE 14

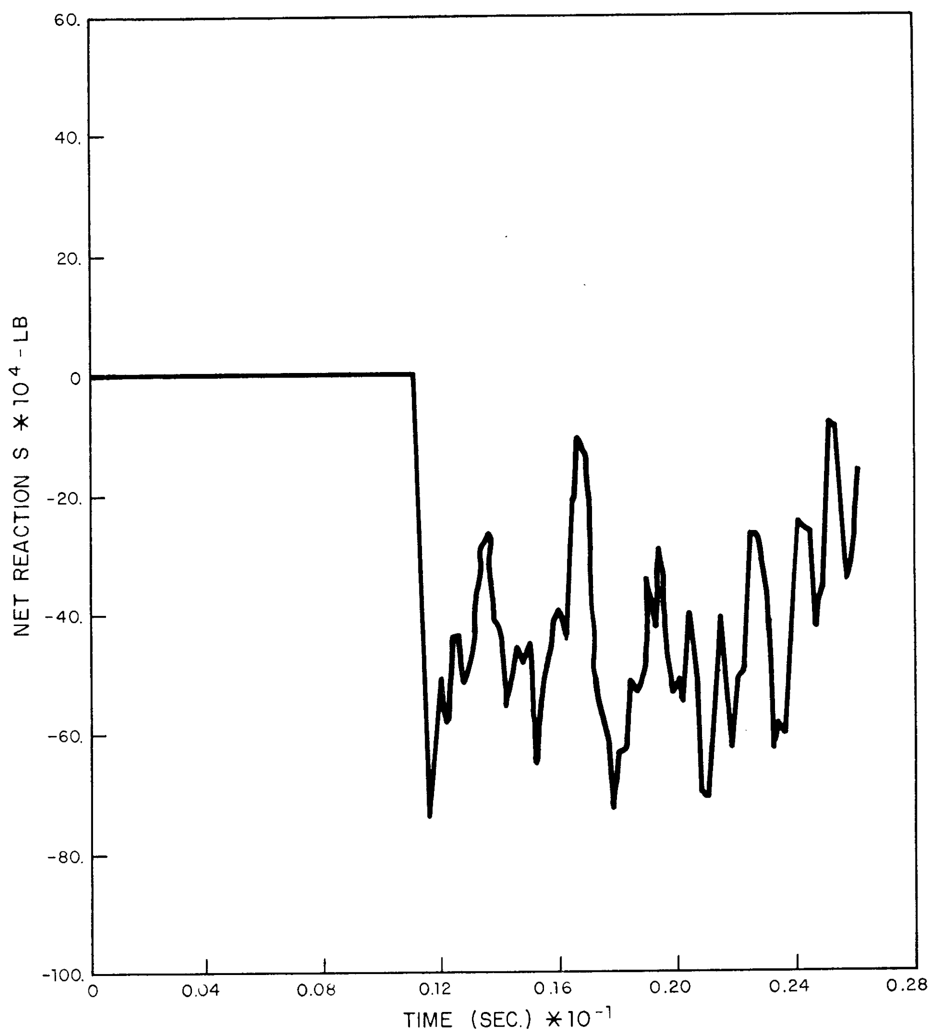


FIGURE 3.6–24
FEEDWATER SYSTEM SHORT LOOP STRAP NEAR THE STEAM GENERATOR,
NET SHEAR FORCE AT NODE 14
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–25 FEEDWATER SYSTEM SHORT LOOP STRAP NEAR THE STEAM GENERATOR, NET MOMENT FORCE AT NODE 14

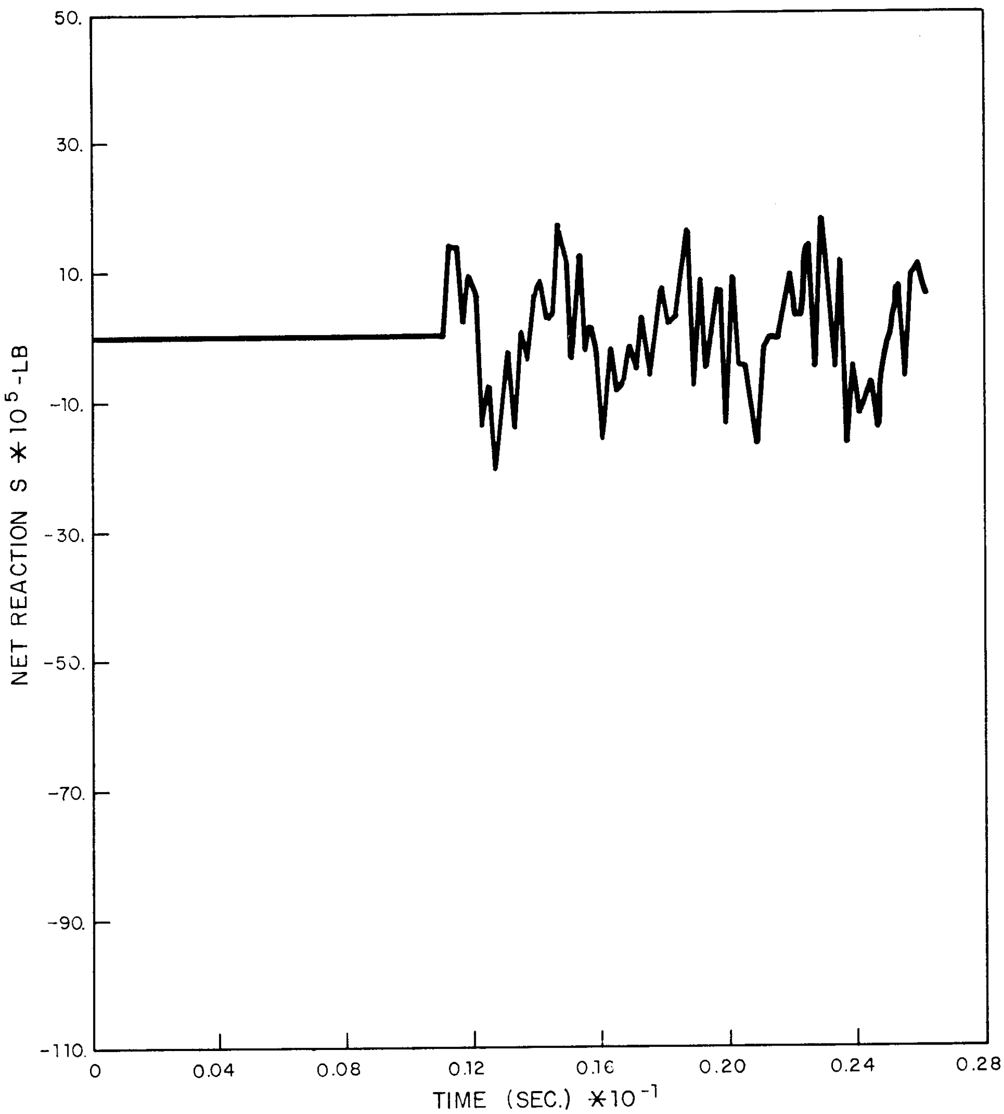


FIGURE 3.6-25
FEEDWATER SYSTEM SHORT LOOP STRAP NEAR THE STEAM GENERATOR, NET MOMENT FORCE AT NODE 14
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–26 FORCE ACTING ON THE STRAP NODE 8

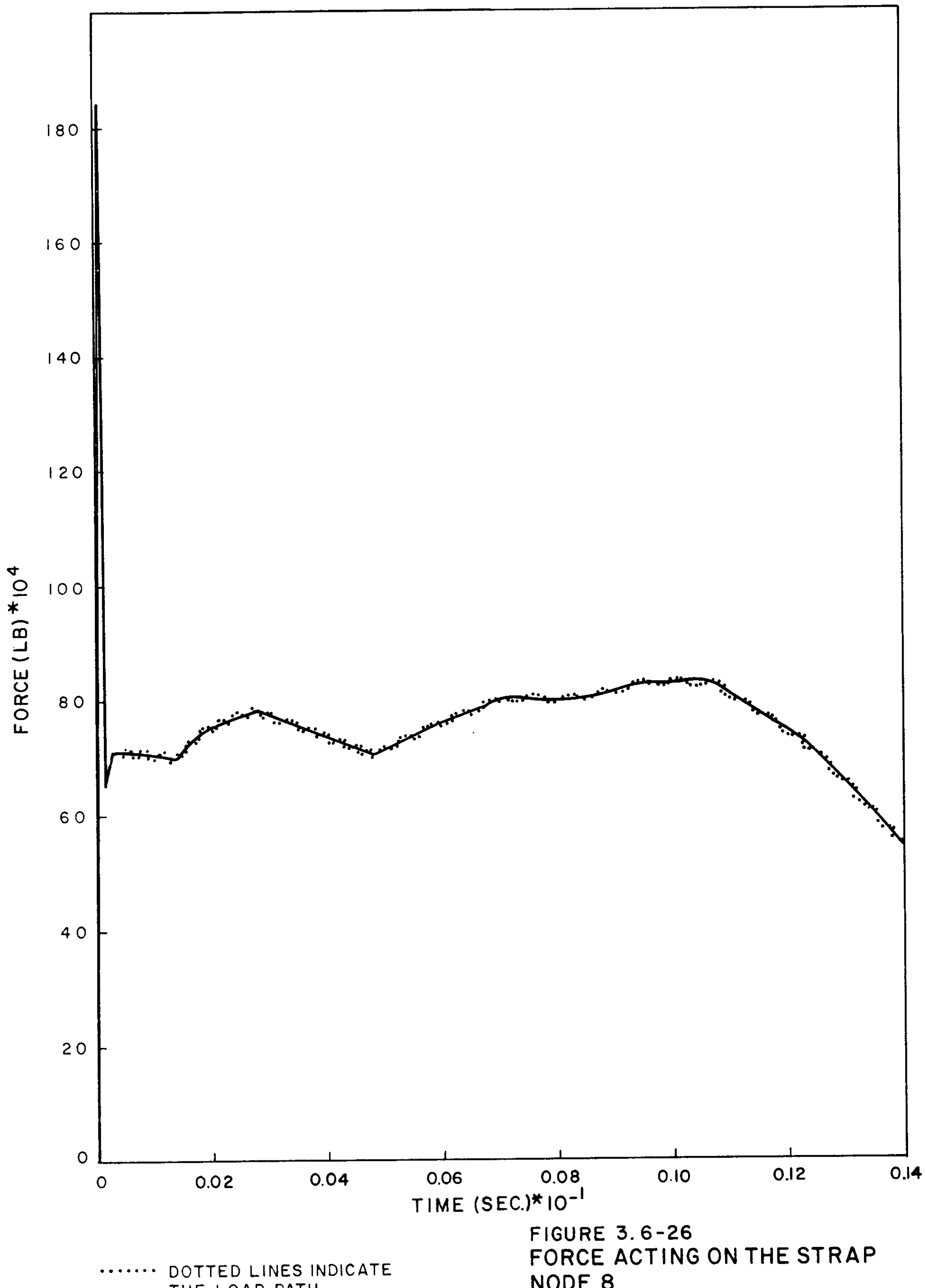


FIGURE 3.6-26
FORCE ACTING ON THE STRAP NODE 8
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–27 FEEDWATER SYSTEM SHORT LOOP STRAP NEAR THE STEAM GENERATOR, ENERGY

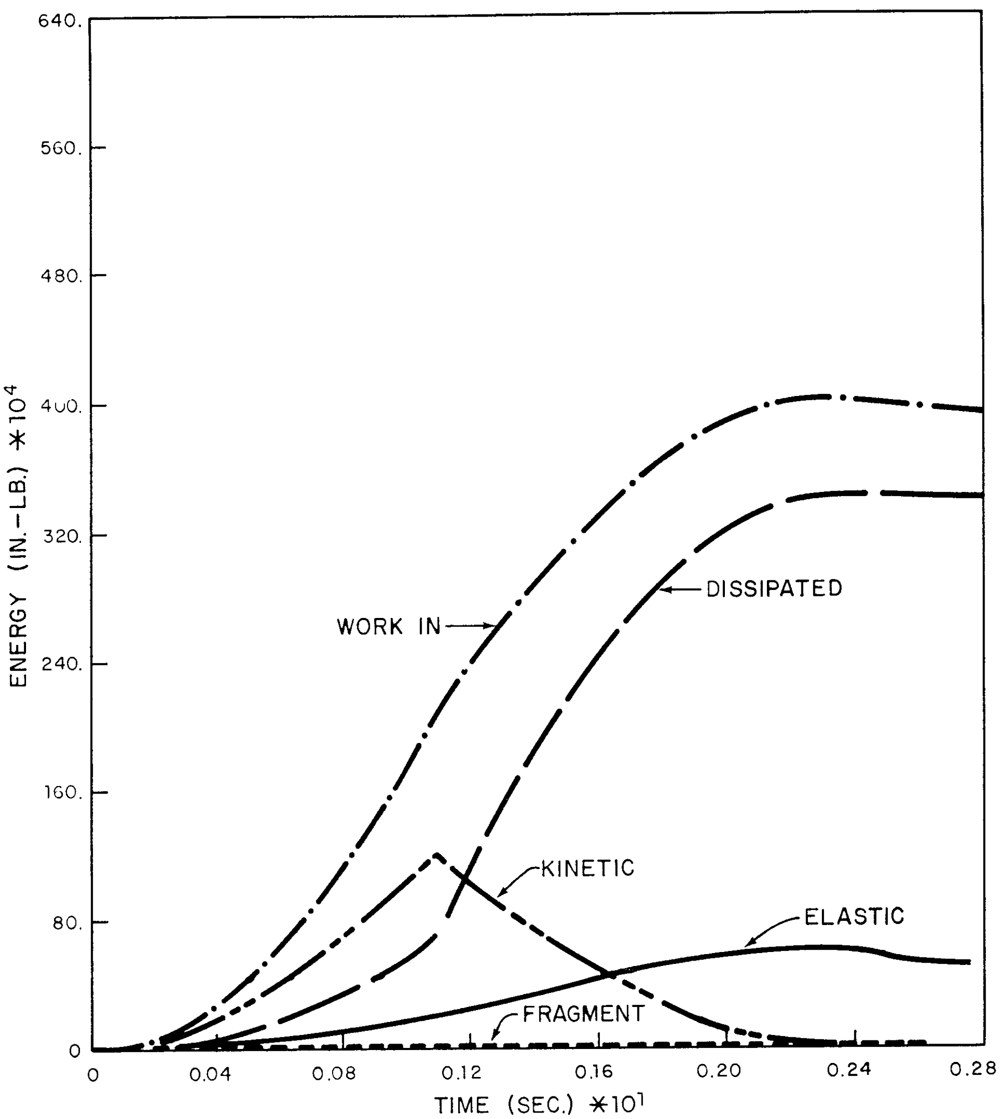


FIGURE 3.6-27
FEEDWATER SYSTEM SHORT LOOP STRAP NEAR THE STEAM GENERATOR, ENERGY
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–28 ENERGY BALANCE ANALYSIS MODEL

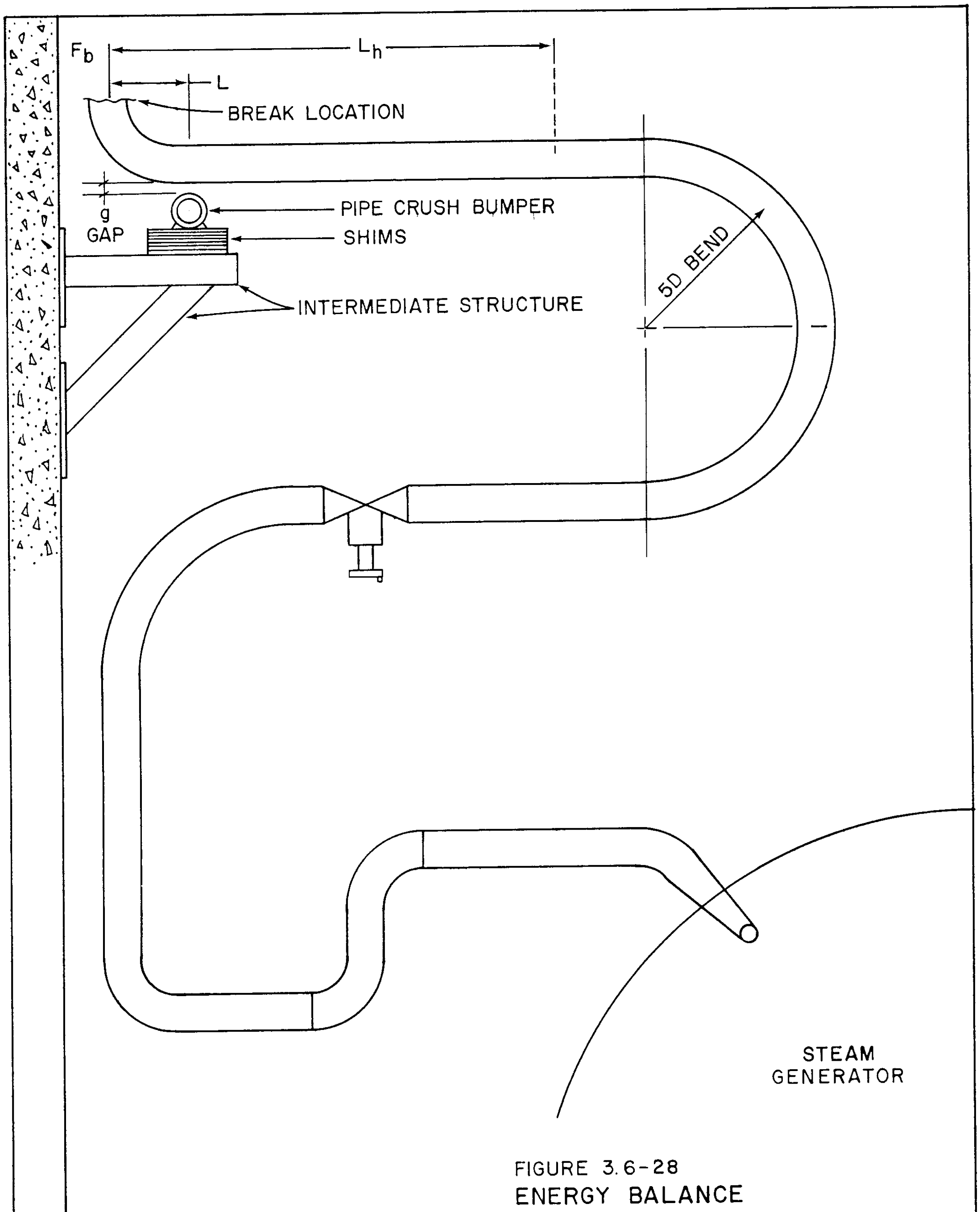


FIGURE 3.6-28
ENERGY BALANCE
ANALYSIS MODEL
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–29 PIPE CRUSH BUMPER

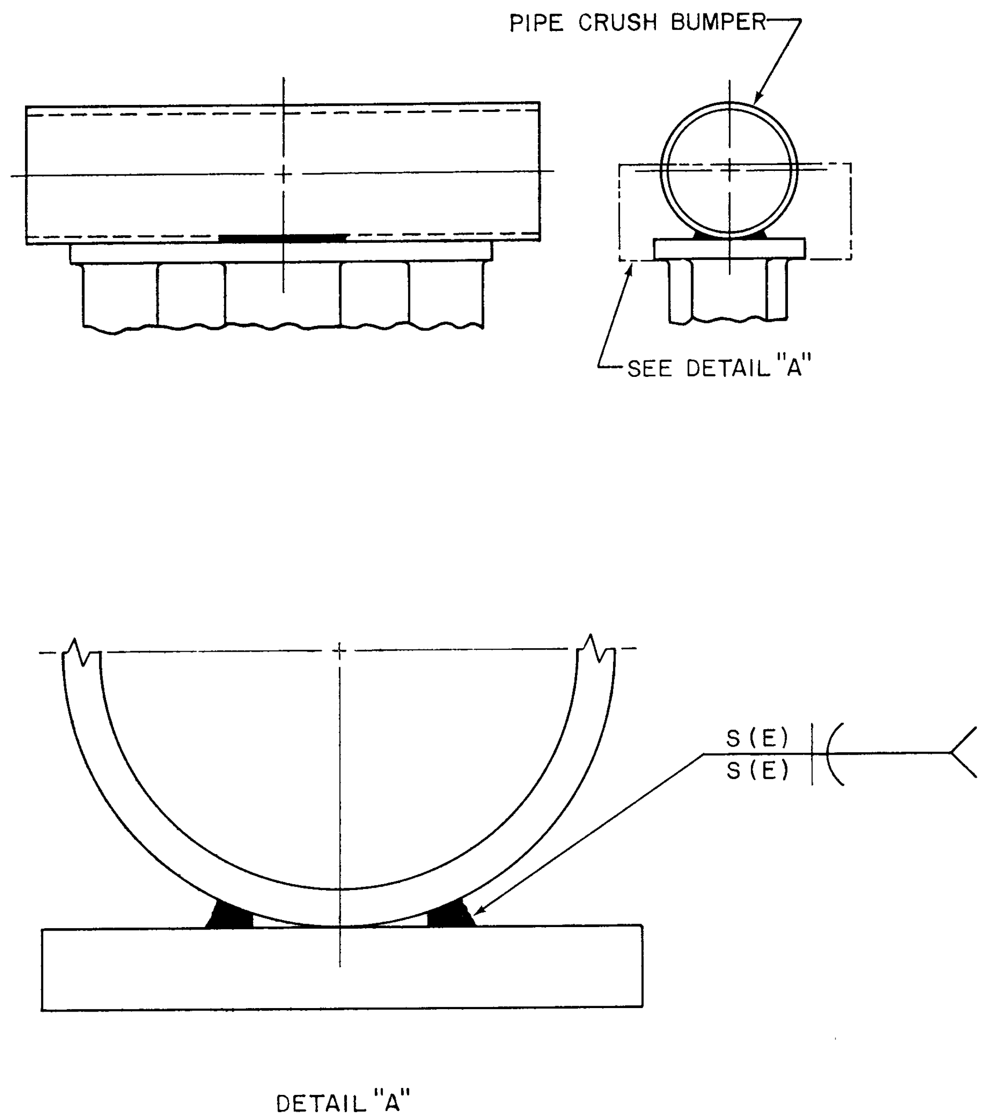


FIGURE 3.6-29
PIPE CRUSH BUMPER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–30 PIPE CRUSH BUMPER

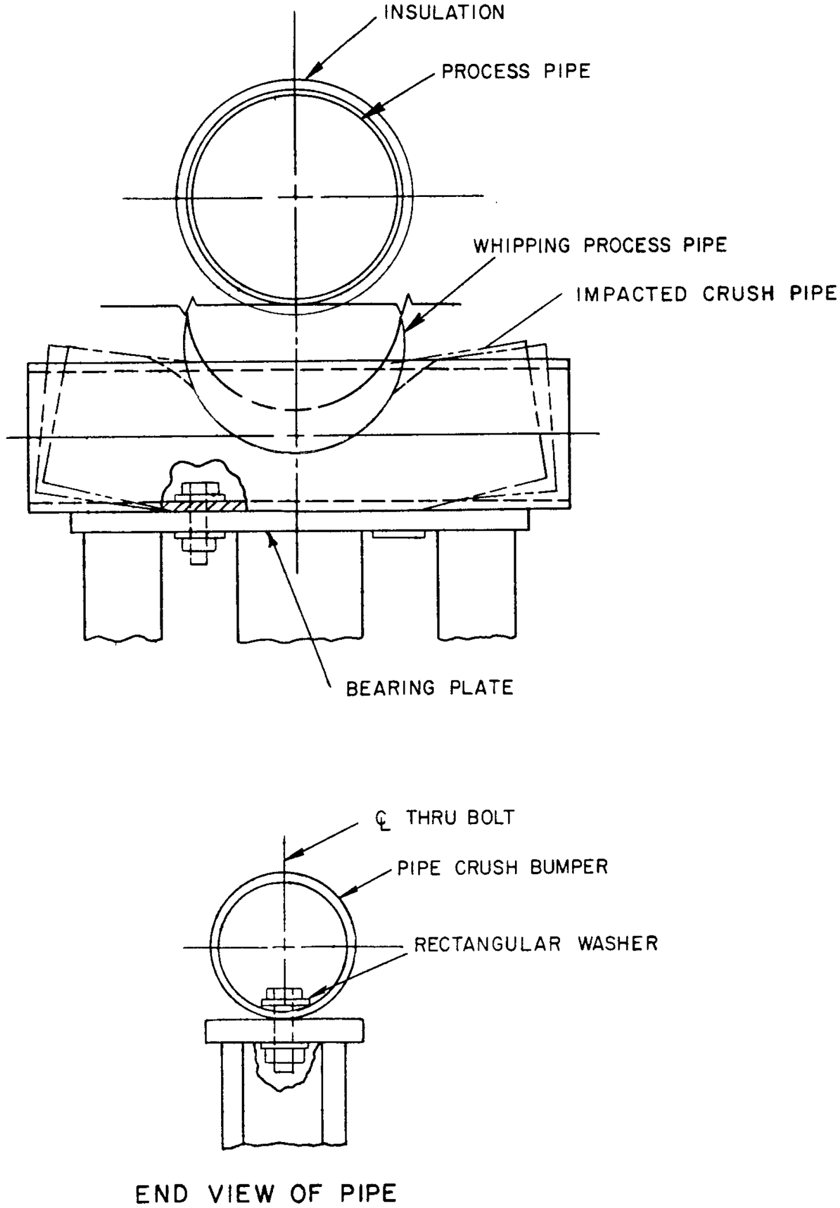


FIGURE 3.6-30
PIPE CRUSH BUMPER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–31 OMNI DIRECTIONAL PIPE RUPTURE RESTRAINT

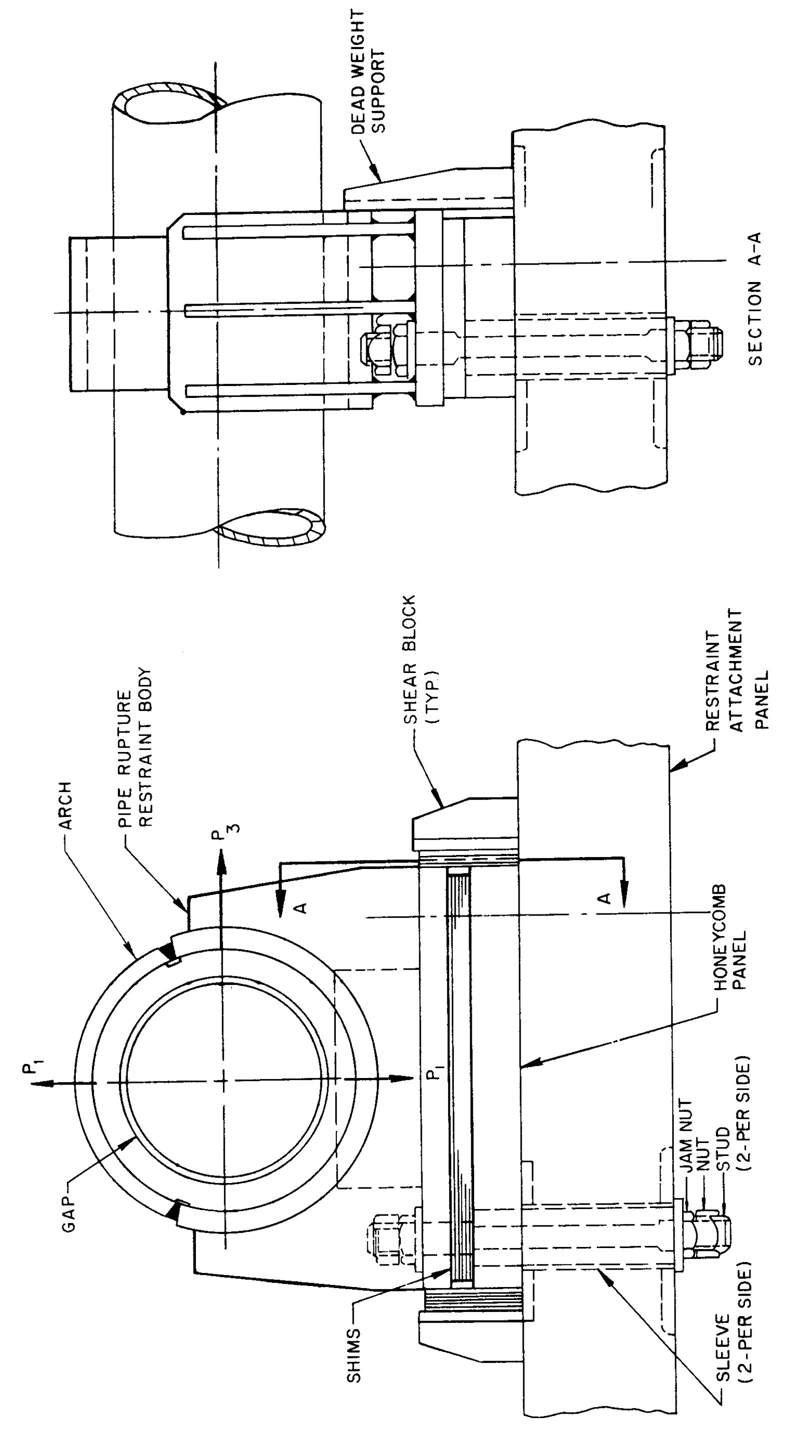


FIGURE 3.6-31
OMNI-DIRECTIONAL PIPE RUPTURE RESTRAINT
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6–32 AUXILIARY FEEDWATER SYSTEM

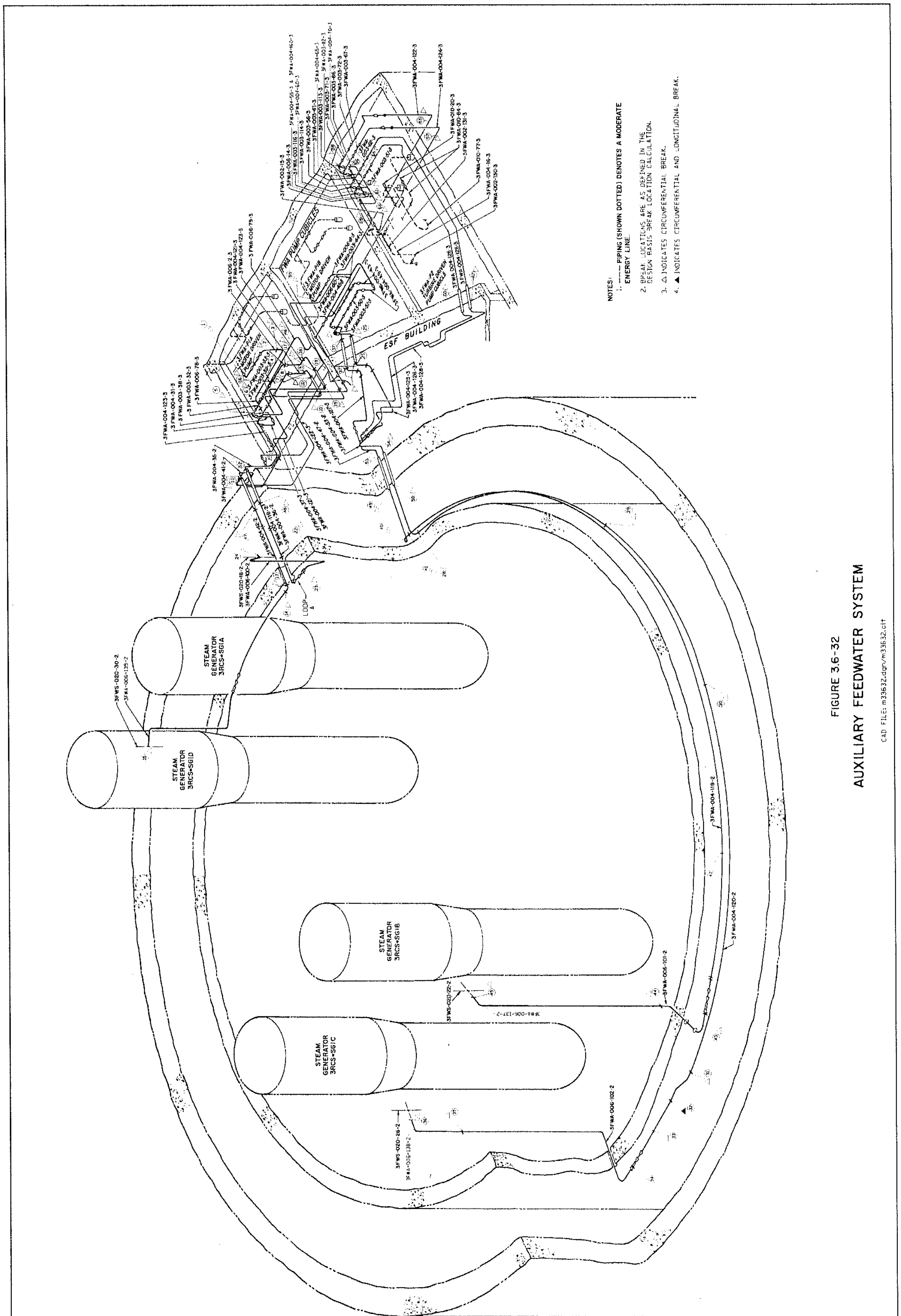
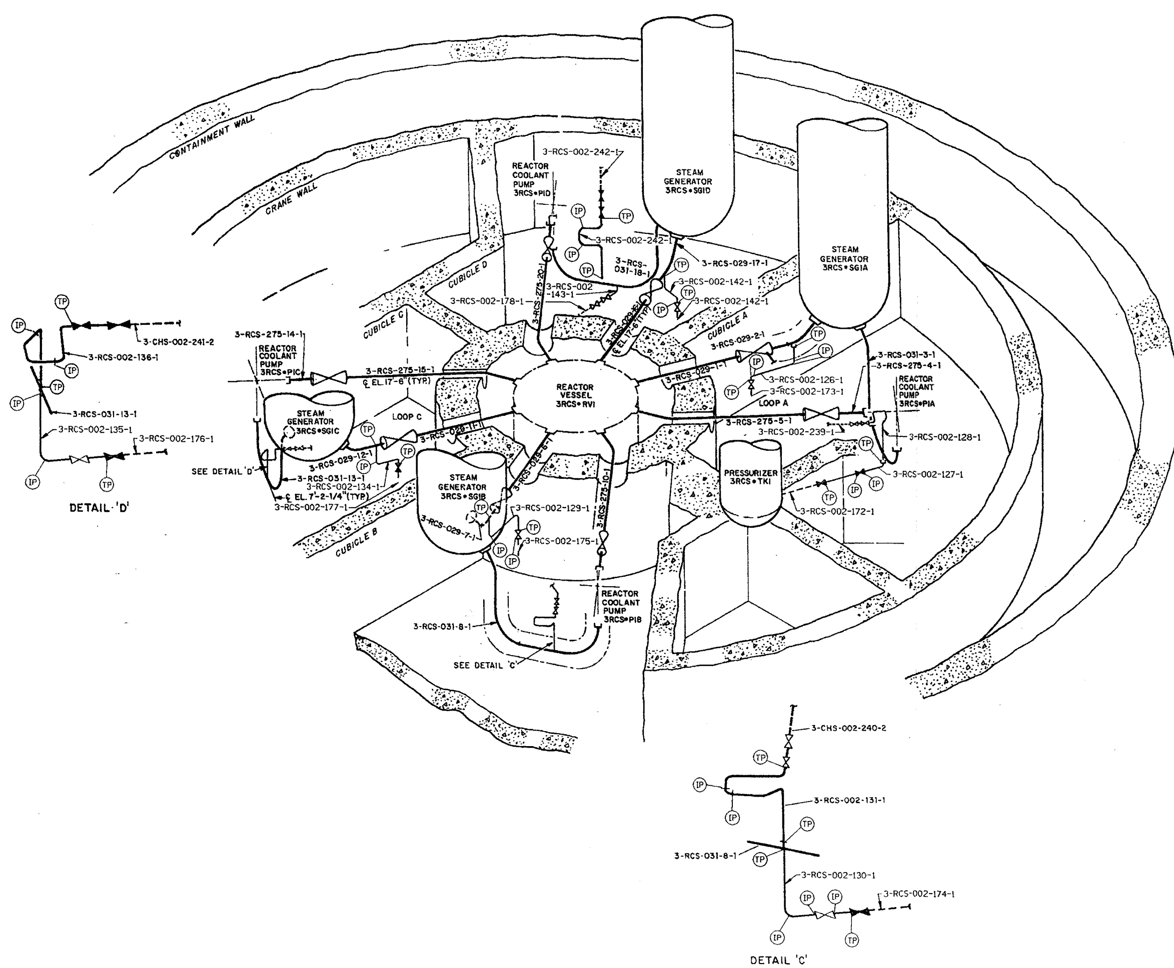


FIGURE 3.6–33 REACTOR COOLANT SYSTEM SMALL BORE PIPING IN STEAM GENERATOR CUBICLES



NOTES:

1. PIPING SHOWN– – –DENOTES A MODERATE ENERGY PIPING SYSTEM.
2. POSTULATED PIPE RUPTURE LOCATIONS AND THE GEOMETRY ARE TYPICAL FOR THE REACTOR COOLANT SYSTEM PRIMARY COOLANT PIPING DRAIN LINES (INCLUDING EXCESS LETDOWN) IN EACH STEAM GENERATOR CUBICLE. PIPE RUPTURE LOCATIONS ON THIS PIPING ARE POSTULATED AT ALL TERMINAL ENDS. VALVES AND FITTINGS. INTEGRAL WELDED ATTACHMENTS ARE NOT UTILIZED ON THE DRAIN LINES.
3. PIPING GEOMETRY FOR THE REACTOR COOLANT SYSTEM LOOP FILL LINES ARE TYPICAL IN EACH STEAM GENERATOR CUBICLE. PIPE RUPTURE LOCATIONS ON THIS PIPING ARE POSTULATED TO OCCUR AT ANY TERMINAL END VALVE OR FITTING. INTEGRAL WELDED ATTACHMENTS ARE NOT UTILIZED ON THIS PIPING SYSTEM. THE LOOP FILL LINE GEOMETRY IS SHOWN AS DETAIL C AND D.
4. REFER TO FIGURE 3.6-15 FOR PIPE RUPTURE LOCATIONS FOR NORMAL CHARGING LINE PIPING WITHIN RCS BOUNDARY.
5. REFER TO FIGURE 3.6-16 FOR PIPE RUPTURE LOCATIONS FOR LETDOWN LINE PIPING WITHIN RCS BOUNDARY.

FIGURE 3.6-33
REACTOR COOLANT SYSTEM SMALL BORE PIPING IN STEAM GENERATOR CUBICLES
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
CAD FILE: 3633.dgn/3633.cit

JANUARY 1998

3.7 SEISMIC DESIGN

Sections whose identification number includes the letter B contain material within balance-of-plant (BOP) scope, while sections whose identification number includes the letter N contain material within the nuclear steam supply system (NSSS) scope (including material pertaining to the reactor coolant loop piping out to the first weld of any connecting piping).

3.7B.1 SEISMIC INPUT

3.7B.1.1 Design Response Spectra

The horizontal design response spectra used for seismic analysis are shown on Figures 3.7B–1 and 3.7B–2. The spectra for the safe shutdown earthquake (SSE) correspond to a maximum rock acceleration of 0.17g, and the spectra for the operating basis earthquake (OBE) correspond to a maximum rock acceleration of 0.09g. These spectra are based on seismic analysis data determined during the construction permit stage. The vertical design response spectra are taken to be two-thirds of the horizontal design response spectra. Regulatory Guide 1.60 does not apply as described in Section 1.8.

3.7B.1.2 Design Time History

The horizontal design time histories produce spectra which envelop the horizontal design response spectra. These spectra are shown on Figures 3.7B–3 through 3.7B–8. The vertical design time histories are taken to be two-thirds of the horizontal design time histories for both the SSE and the OBE.

Where soil structure interaction analysis is performed, the time history is applied at bedrock in the free field. Where analysis is performed using subgrade springs to represent rock stiffness, the time history is applied at the foundation level.

The horizontal design time history is checked by comparing its spectral values with those of the site design response spectra at 250 oscillator periods between 0.0167 second and 5.0 seconds. The oscillator periods are distributed logarithmically according to the following expression:

$$T_i = \lambda T_{i-1} \qquad (3.7B\text{-}1)$$

$$\lambda = (T_{250}/T_1)^{1/249} = (5/0.0167)^{1/249}$$

$$\lambda = 1.023$$

where:

T_{i-1} and T_i = Any two consecutive oscillator periods

T_i/T_{i-1} = The period ratio (ratio between the i^{th} period, T_i, and the (i-1) period, T_{i-1}).

3.7B.1.3 Critical Damping Values

The values of the percentage of critical damping used in the analysis of Seismic Category I structures, systems, and components depends on the stress levels resulting from the seismic input motion (SSE or OBE) used in the analysis.

The values of damping used for the input motion are listed in Table 3.7B–1. Structural damping is assigned to be 2.0 percent for the OBE and 5.0 percent for the SSE.

The higher damping values (Table 3.7B–1) are used only where justified by detailed study (either testing or calculated stress levels). Damping values utilized in the analysis of Seismic Category I structures, systems, and components meet the intent of Regulatory Guide 1.61 which is not applicable as described in Section 1.8.

3.7B.1.4 Supporting Media for Seismic Category I Structures

The founding materials for major plant structures are listed in Table 2.5.4–14. Most of the major safety related structures are founded on bedrock, with the exception of the control building, emergency diesel generator building, and the hydrogen recombiner building. The control building is founded on 1 to 4 feet of compacted structural backfill overlying basal till of thickness varying between 1 foot on the east side and 15 feet on the west. The emergency diesel generator building is founded on basal till varying in thickness from less than 10 feet to 30 feet overlying bedrock. The hydrogen recombiner is founded on concrete fill overlying bedrock.

A large portion of the circulating water discharge tunnel and the service water intake lines are founded on bedrock. However, some sections are founded on soil, particularly near the intake and discharge points in the vicinity of Niantic Bay. When soil was encountered as a founding material, all unsuitable overburden was removed to sound ablation or basal till. In the event that the invert elevation was higher than the excavated grade, compacted structural backfill was placed in thin lifts to the subgrade elevation in accordance with procedures described in Section 2.5.4.5.2.

A description of each of the founding materials at the site is presented in Section 2.5.4.2. The seismic velocities and moduli values of these materials are presented in Section 2.5.4.4.3. The properties of structural backfill materials are described in Section 2.5.4.5.2. The use of these properties in the analysis of soil-structure interaction is discussed in Section 3.7B.2.4.

3.7B.2 SEISMIC SYSTEM ANALYSIS

Seismic systems discussed in this section are those major Category I structures that are considered in conjunction with foundation media in forming a soil-structure interaction model. Other systems and components are considered seismic subsystems (Sections 3.7B.3 and 5.4.14).

3.7B.2.1 Seismic Analysis Methods

Generally, analysis of seismic systems were performed using either the modal analysis response spectra method or the modal analysis time history method. Response spectrum analysis uses the natural frequencies, mode shapes, and weighted modal dampings to determine the maximum seismic response of multi-degree-of-freedom systems with lumped masses and elastic connecting members. Time history modal analysis uses the same free vibration characteristics and damping factors as the spectrum analysis, primarily to determine acceleration response spectra at selected locations within the structure. Those buildings that required finite element soil-structure interaction were analyzed using the frequency domain time history method (Section 3.7B.2.4). The methods used for seismic analysis of particular Seismic Category I structures are summarized in Table 3.7B–2.

Mathematical models of three buildings representative of Seismic Category I structures are discussed in detail. Models selected are the containment structure (Figure 3.7B–9), the main steam valve building (Figure 3.7B–10), and the emergency generator enclosure (Figures 3.7B–11 and 3.7B–12). These buildings are described in Sections 3.8.1 and 3.8.3.

Foundation effects, namely torsion, rocking, and translation, were included in the dynamic analysis as described in Section 3.7B.2.4. For structures founded on shallow soil, a finite element soil-structure interaction analysis was performed. In this case, a planar model was employed in each of the horizontal direction. All other Seismic Category I structures were analyzed considering six degrees-of-freedom; three translation, two rocking, and one torsional.

Those buildings that were modeled using finite element soil structure interaction techniques included the control building and the emergency generator enclosure. The computer program for these analyses was PLAXLY. Structures founded on rock, which are discussed in Section 3.7B.2.4, were analyzed using discrete springs suggested by Richart et al., 1970, Whitman et al., 1967, Wass 1972, and Kausel et al., 1975. Springs associated with the degrees-of-freedom in a global orthogonal coordinate system were used.

In developing the lumped-mass models for Seismic Category I structures, an adequate number of degrees-of-freedom was ensured by providing a lumped mass at the mat, floor, and roof elevations. Additional masses were used when dictated by changes in geometry or stiffness. In the cases of the containment shell, eight lumped masses were selected at intervals uniformly distributed along the height of the shell. Studies have been conducted which show the difference in base shear and overturning moment between a 3-mass and a 15-mass model to be less than 3 percent. Comparisons between 3-, 5-, 10-, and 15-mass models have shown a difference in natural frequency for the first two modes of less than 1 percent.

For analysis of the models, twice the number of modes with frequencies less than 33 Hz were considered.

Each Seismic Category I building has a foundation mat which serves as a single structural support, except the emergency generator enclosure which has spread footings that serve the same purpose. Two inch (above grade) and 1 inch (below grade) shake spaces, larger than any predicted structure-to-structure displacements, are provided between all Seismic Category I structures. In addition, 4 inch shake spaces are provided between all Seismic Category I structures and the containment structure. Relative displacements were considered for all Category I systems and components which run between structures. Seismic effects such as hydrodynamic loads and nonlinear responses were considered where appropriate (Section 2.5.4).

3.7B.2.2 Natural Frequencies and Response Loads

Response spectrum analyses for the containment structure and the main steam valve building, are summarized below, as well as the finite element soil structure interaction analysis of the emergency generator enclosure. They are representative samples of Category I structures.

3.7B.2.2.1 Containment Summary

The response spectrum analysis of the containment and internal structures included the enveloping of two discrete analyses. The dynamic model of the containment (Figure 3.7B–9) was analyzed with cracked and uncracked concrete properties.

The significant natural frequencies and modal participation factors resulting from the uncracked and cracked analyses are presented in Tables 3.7B–3 and 3.7B–4. Significant mode shapes resulting from these analyses are presented in Tables 3.7B–5 and 3.7B–6 and the response loads obtained by the square root of the sum of the squares method (SRSS) are summarized in Tables 3.7B–7 and 3.7B–8. Table 3.7B–9 presents the degrees-of-freedom corresponding to x, y, and z direction and rotation about the x, y, and z axis for the containment and internal structures response loads obtained by the SRSS method. Final lumped mass accelerations and displacements are obtained by taking the absolute sum of the SRSS modal responses due to each direction of excitation and enveloping the cracked and uncracked analysis (Tables 3.7B–10 and 3.7B–11). The amplified response spectra (ARS) were produced using the modal analysis time history method. Typical ARS resulting from the enveloping of the cracked and uncracked analyses are provided at the mat, steam generators support slab, top of the primary shield wall, operating floor and crane wall of internal structure, as well as the springline and dome apex of containment structure for horizontal and vertical safe shutdown earthquake (SSE) excitations (Figures 3.7B–13 through 3.7B–33).

3.7B.2.2.2 Main Steam Valve Building Summary

The dynamic model of the main steam valve building (Figure 3.7B–10) was excited separately by one vertical and two orthogonal horizontal excitations. The resulting natural frequencies, participation factors, and mode shapes are listed in Tables 3.7B–12 and 3.7B–13. The closely spaced modes (CSM) modal responses are given in Tables 3.7B–14 and 3.7B–15. The CSM modal combination method is explained in Section 3.7B.2.7. Final lumped mass accelerations and displacements are obtained by taking the absolute sum of the CSM modal responses due to each direction of excitation (Table 3.7B–16). For the main steam valve building analysis, Table 3.7B–17 presents the relationship between the degrees-of-freedom and the motion of the lumped masses given in Tables 3.7B–14, 3.7B–15, and 3.7B–16.
Examples of the ARS resulting from each direction of excitation produced using the modal analysis time history method are provided at the mat (Figures 3.7B–34 thru 3.7B–36), the floor slab at el 41 feet 0 inch (Figures 3.7B–37 thru 3.7B–39), and the roof slab at el 85 feet 4 inches (Figures 3.7B–40 through 3.7B–42).

3.7B.2.2.3 Emergency Generator Enclosure Summary

The seismic analysis of the emergency generator enclosure used a lumped mass finite element model (Section 3.7B.2.4). The dynamic model shown on Figures 3.7B–11 and 3.7B–12 was excited separately by one vertical and two orthogonal horizontal excitations. Since the method used for the analysis was a frequency domain solution rather than a modal analysis, no natural frequencies, participation factors, or mode shapes are tabulated. The solution gives transfer functions which represent the ratio of the amplitude of any particular mass point of the structure to the amplitude of the input motion at bedrock. Typical transfer functions (including imaginary part) for each mass point of the dynamic model are presented on Figures 3.7B–43 thru 3.7B–51 for each direction of excitation. Final absolute sum response loads are presented in Table 3.7B–18 and 3.7B–19. Also, since diesel generators are located on independent foundations, a separate analysis was performed for them.
Typical ARS at elevation 24 feet 6 inches, 51 feet 0 inches, and 66 feet 0 inches, produced using the frequency domain time history method, are shown on Figures 3.7B–52 thru 3.7B–60. ARS resulting from the independent analysis of the diesel generator isolation mats are presented on Figures 3.7B–61 and 3.7B–62. Examples of the input time history at bedrock of the plant site and the resulting time history at the base of the structure are shown on Figures 3.7B–63 and 3.7B–64.

3.7B.2.3 Procedures Used for Analytical Modeling

The dynamic model of a Seismic Category I structure is constructed to obtain a satisfactory representation of the dynamic behavior of the structure. Major Seismic Category I structures that are considered in conjunction with foundation media in forming a soil-structure interaction model are defined as seismic systems. Other Seismic Category I systems and components that are not designated as seismic systems are considered as seismic subsystems (R). In most cases, equipment and components come under the definition of seismic subsystems and are analyzed as a decoupled system from the primary structure. To define criteria for decoupling subsystems, R_m, the mass ratio and R_f, the frequency ratio are significant. R_m and R_f are defined as:

$$R_m = \frac{\text{Total mass of the supported subsystem}}{\text{Mass that supports the subsystem}} \tag{3.7B-2}$$

$$R_f = \frac{\text{Fundamental frequency of the support subsystem}}{\text{Frequency of the dominant support motion}} \tag{3.7B-3}$$

The following criteria are used.

1. If $R_m < 0.01$, decoupling is acceptable for any R_f,

2. If $0.01 \le R_m \le 0.1$, decoupling is acceptable if $R_f \le 0.8$ or if $R_f \ge 1.25$,

3. If $R_m > 0.1$, an approximate model of the subsystem is included in the primary system model.

If the subsystem is comparatively rigid and also rigidly connected to the primary system, only the mass of the subsystem is included at the support point in the primary system model. In the case of a subsystem supported by very flexible connections (e.g., a pipe supported by hangers), the subsystem is not included in the primary model; however additional mass is considered.

In most cases, the equipment and components which come under the definition of subsystems were analyzed separately from the primary structure, and the seismic input for the subsystem was obtained from the analysis of the structure. One important exception to this procedure is the analysis of the reactor coolant system (RCS), which was considered to be a subsystem but was analyzed using a coupled model of the RCS and primary structure with ground motion as the seismic input (Section 5).

The dynamic model of a seismic system consists of a set of lumped masses, generally having six degrees-of-freedom per mass connected by weightless elastic members. Masses are usually lumped at floor levels and include the masses of the floors, walls, columns, equipment, and piping. The floors are treated as rigid diaphragms that transfer the earthquake inertia forces to frames and shearwalls, which in turn transfer the loads to the foundation mat and the subgrade. Beam theory, combining the effects of shear, flexure, torsion, and axial deformation, is used to establish the stiffness characteristics of the frame-wall systems. Eccentricities between the centers of mass (CM) and centers of rigidity (CR) are considered.

As an example, the lumped mass model of the containment is shown on Figure 3.7B–9. The model was constructed so that it properly represents the free vibration of a cantilevered structure in shear and flexure. The model consists of a system of lumped masses, each with six degrees-of-freedom, connected by weightless members. The base of the model is connected to the subgrade represented by springs as discussed in Section 3.7B.2.4. Masses M_{10} through M_{17} represent the dome and the cylindrical portion of the containment shell and the dome. M_1 consists of the mat and portions of the walls and columns which are attached to the mat. The internal structure, consisting of equipment, primary shield wall, cubicle walls, crane wall, etc., were modeled by masses M_2 through M_9.

3.7B.2.4 Soil-Structure Interaction

The supporting media for Seismic Category I structures has been discussed in Section 3.7B.1.4. As indicated in Table 2.5.4–14, Seismic Category I structures are founded directly on bedrock with a few structures founded on shallow soil overburden over rock.

Finite element soil-structure interaction analysis is required for shallowly embedded structures on shallow soil overburden over rock (NRC 1975). The control building and the emergency generator enclosure were analyzed using the finite element method.

For the finite element analysis, the emergency generator enclosure lumped mass model (Section 3.7B.2.3) was attached to a finite element representation of the soil subgrade (Figures 3.7B–11 and 3.7B–12). These models were then subjected to horizontal and vertical excitations by computer program, PLAXLY. The bottom boundary was taken at bedrock and the side boundaries used PLAXLY's energy transmitting boundary corresponding to layers of soil extending laterally to infinity. The material properties of the soil were assumed to be linearly viscoelastic, and the dynamic equations were solved in the frequency domain using Fourier transformation techniques. Nonlinear behavior in the subgrade was accounted for by the use of the computer program SHAKE, which determines the stain corrected soil properties.

Section 3A.1.7 of Appendix 3A give a detailed discussion of PLAXLY.

For structures founded directly on bedrock analyses using discrete subgrade springs based on equations suggested by Richart et al. (1970), Whitman et al. (1967), Wass (1972), and Kausel et al. (1975) were performed. A value of 10 percent (translational) and 5 percent (rotational) maximum subgrade damping assured the conservatism of the discrete spring method.

3.7B.2.5 Development of Floor Response Spectra

Floor response spectra were developed separately for one vertical and two mutually perpendicular horizontal earthquake motions using the modal analysis time history method discussed in Section 3.7B.2.1 for all buildings except the emergency generator enclosure. Floor response spectra for the emergency generator enclosure were developed using the modal analysis method as in PLAXLY (Section 3.7B.2.4).

Combination of the three components of floor response spectra for use in subsystem seismic analysis is discussed in Section 3.7B.3.

3.7B.2.6 Three Components of Earthquake Motion

Procedures to combine three components of earthquake motion to determine the response of seismic systems meet the recommendations of Regulatory Guide 1.92, Rev. 1, to the extent described in Section 1.8.1.92.

3.7B.2.7 Combination of Modal Responses

In general, for those Seismic Category I systems analyzed by the modal analysis response spectrum method, the influence of closely spaced modes was considered by use of the "double sum method," described in Regulatory Guide 1.92, Rev. 1, to the extent described in Section 1.8.1.92 and in Singh et al. (1973).

3.7B.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

Adjacent Seismic Category I structures are separated from Non-Category I structures by a 1-inch Shake space below grade and a 2-inch Shake space above grade.

Non-Category I structures are sufficiently remote from Seismic Category I structures to preclude interaction or are designed (Section 3.8.3) to avoid collapse during an SSE, thus preventing significant damage to adjacent Category I structures.

3.7B.2.9 Effects of Parameter Variations on Floor Response Spectra

In order to consider the effects of variations of structural properties, dampings, and soil properties, peak resonant period values of the floor response spectra were spread +15 and -15 percent. For the containment structure which experiences substantial cracking during the structural acceptance test, floor response spectra for the cracked and uncracked cases were enveloped.

The use of floor time histories in subsystem analysis is discussed in Section 5.4.14.

3.7B.2.10 Use of Constant Vertical Static Factors

Dynamic vertical responses were calculated in all seismic system analyses, precluding the need to use constant vertical static factors.

3.7B.2.11 Method Used to Account for Torsional Effects

The emergency generator building was analyzed using planar models for two horizontal directions using three degrees-of-freedom per mass. Since this building is basically symmetrical, this approach is justified. All other Seismic Category I structures were analyzed dynamically using six degrees-of-freedom per mass. Effects of eccentricity between centers of mass and centers of stiffness or soil contact were included in determining appropriate member stiffnesses in the dynamic models.

3.7B.2.12 Comparison of Responses

As discussed in Section 3.7B.2.1, the results of the response spectrum analysis were used for the design of most Seismic Category I structures and the results of the modal time history analyses provided the floor time histories used to create floor response spectra. The maximum responses attained from the time history analysis are compared with those resulting from the response spectrum technique in Tables 3.7B-20 through 3.7B-22 for the containment structure and main steam valve building.

The higher time history results are attributable to the fact that the artificial ground time history created to envelop the Millstone 3 ground response spectra is quite conservative. The extent of this conservatism is illustrated on Figures 3.7B–3 through 3.7B–8 which compares the Millstone 3 ground response spectra with the response spectra generated by using the site artificial time history as the forcing function. However, because of the method used in combining structural responses from the response spectrum analysis due to each of the three components of earthquake motion (absolute sum), a majority of the response accelerations from the response spectrum method are larger than those from the time history analysis.

3.7B.2.13 Methods for Seismic Analysis of Category I Dams

There are no Category I dams at this site.

3.7B.2.14 Determination of Seismic Category I Structure Overturning Moments

Overturning moments for Seismic Category I structures were calculated using inertia forces based on the accelerations summarized in Section 3.7B.2.2 as follows:

$$OM = \sum_{i=1}^{n} M_i a_{hi} H_i + \sum_{i=1}^{n} M_i a_{vi} D_i + M_a \qquad (3.7B\text{-}4)$$

where

OM = Overturning moment

i = Mass point on structure

M_i = Mass at i

a_{hi} = Horizontal acceleration at i in the direction being investigated

a_{vi} = Vertical acceleration at i

H_i = Height of point i above bottom of foundation mat

D_i = Distance from point i to edge of foundation mat in the direction of acceleration

n = Number of Mass points in structure

Four directions of horizontal acceleration were considered: north, south, east, and west. The overturning moment was calculated for each direction of horizontal acceleration assuming the vertical acceleration to act either up or down. The maximum moment calculated from the different combinations of the horizontal and vertical accelerations was then divided into the resisting moment based on the deadweight of the structure to determine the factor of safety against overturning. The minimum factors of safety allowed are summarized in Section 3.8.5.5.

3.7B.2.15 Analysis Procedure for Damping

An equivalent viscous modal damping, reflecting the different damping rates in various portions of the structure, was computed for use in structural dynamic analysis. The modal damping ratio is a weighted average of member and support spring damping, based on the contribution of each to the total strain energy of the mode shape. The method is based on work by Roesset et al. (1973). The following discussion is meant to serve as a general description. The more theoretical derivations (Roesset et al., 1973, Whitman 1970) are found in the literature.

Damping and Strain Energy Methods

An important factor in determining structural response is the damping phenomenon. Two types of damping are generally recognized: viscous (in which the energy dissipated per cycle is proportional to frequency) and hysteretic (in which no frequency dependence is seen). Most structural elements display hysteretic behavior, while supporting soils appear to combine both hysteretic and viscous damping mechanisms.
For certain applications (e.g., pipe breaks) in which foundation motion can be neglected, only hysteretic damping need be considered. Whitman's (1973) analysis of Biggs' formula gives a useful approximation for the damping of each mode when material damping varies from element to element. His expression for the equivalent viscous modal damping is obtained by a strain-energy weighting of element damping:

$$B^j_{eqv} = \frac{\sum_{i=1}^{N} D_i E^j_i}{\sum_{i=1}^{N} E^j_i} \qquad (3.7B\text{-}5)$$

where:

B^j_{eqv} = Equivalent viscous damping ratio (fraction of critical) for structure vibrating in mode j,

N = Number of elements,

D_i = Hysteretic damping ratio for element i,

${}^{j}E_i$ = Strain energy in element i when deflected into mode shape j.

In particular, when damping is uniform (i.e., $D_i = D$) then $B_{eqv} = D$ for all modes. When damping is not uniform, modal damping is weighted toward those elements which make the largest contribution to the energy of each mode. In other applications (e.g., earthquakes) in which foundation motion is significant, viscous damping also must be considered. Current practice treats the soil damping as viscous for translational motion, and hysteretic for rotational motion. Roesset et al. (1973) extended Biggs' formula to include the viscous damping contributions of the soil:

$$B^j_{eqv} = \frac{\sum_{i=1}^{N} D_i E^j_i + \sum_{k=1}^{N_v} \frac{W^j}{W_k} B_k E^j_k}{\sum_{i=1}^{N} E^j_i + \sum_{i=1}^{N_v} E^j_k} \qquad (3.7B\text{-}6)$$

where:

N_H = Number of hysteretically damped elements,

N_V = Number of viscously damped elements,

W^j = Frequency of structure mode j (radians per second),

W_K = Frequency of element k (radians per second),

B_K = Critical damping ratio of element k at frequency k.

The square of the natural frequency w of the soil element, is equal to the ratio of the soil spring stiffness to total mass of the structure plus foundation.

This formula reflects the fact that the energy dissipation per cycle by the viscous mechanism is proportional to frequency of motion. Any element which displays both hysteretic and viscous damping appears in both summations of the numerator, but is not repeated in the denominator. Each element strain energy appearing in Equation (3.7B-4) is evaluated from the element stiffness matrix and the displacement of the element's boundary joints. After comparing the Biggs and Roesset modal damping ratios calculated from Equations 3.7B-5 and 3.7B-6, the lower value for each mode is selected for use in the dynamic analysis, thus assuring that the composite modal damping value never exceeds the hysteretic damping value. In no case do modal damping values exceed 10 percent.

3.7B.3 SEISMIC SUBSYSTEM ANALYSIS

3.7B.3.1 Seismic Analysis Methods

3.7B.3.1.1 Equipment and Components

Seismic Category I equipment and components are documented for seismic adequacy. The basic source of seismic design data is either the ground response spectra, the amplified response spectra, floor, or mat time history, derived through a dynamic analysis of the relevant structure (Sections 3.7B.2.5 and 3.7B.2.9).

Three principal methods, which include combinations of these methods, are used for documenting adequacy for Seismic Category I equipment and components:

1. Static analysis

2. Dynamic analysis

3. Testing

The effects of supports are reflected in the input to seismically qualified equipment and components. General stress limits are given in Section 3.9B.3.1. Specific component stress allowables are given in Tables 3.9B–5, 3.9B–6, and 3.9B–7. Such limits either conform to, or are more conservative than, those of ASME Section III, Subsection NF.

Laboratory tests are performed on Seismic Category I mechanical and electrical equipment, and complex instrumentation that cannot be modeled to predict response correctly. Conversely, analytical methods described in this section are employed when mathematical modeling techniques are used or when equipment characteristics (e.g., size, rating) preclude laboratory testing. Combinations of analysis and testing are also employed to assure adequacy of Seismic Category I equipment.

The safe shutdown earthquake (SSE) in combination with the faulted loads are the load basis which assure the structural integrity of Seismic Category I equipment. The operating basis earthquake (1/2 SSE) in conjunction with the operating loadings are used to assure continued operation of the equipment and components.

Equipment vendors and suppliers are required to formulate programs for qualification of Seismic Category I equipment in accordance with specification requirements. Documentation of the seller's qualification program is reviewed and approved by the Applicant.

3.7B.3.1.1.1 Static Analysis

Static analysis is used for equipment and components that can be characterized as relatively simple structures. This type of analysis involves the multiplication of the equipment, or component, total weight by the specified seismic acceleration (direction dependent loading), to produce forces that are applied at the center of gravity in the horizontal and vertical directions. A stress analysis of equipment components, such as supports, holddown bolts, and other structural members, is performed to determine their adequacy.

In the specification of equipment for static analysis, two or more sets of acceleration data are provided. The choice of which set to use is dependent upon the equipment's fundamental natural frequency. The relevant response curves are reviewed to determine a cutoff frequency which bounds the rigid range from the resonance range of the response curves. Equipment and components having fundamental natural frequencies above the cutoff frequency of the relevant response curve are analyzed to rigid range response accelerations.

For components or equipment having a fundamental natural frequency below the cutoff frequency, the accelerations used in static analysis are 1.3 times the peak acceleration value, as indicated in the amplified response curve for simply supported conditions. For equipment having multi-support points in a single plane, the resonant response acceleration is multiplied by 1.5. Each of the three defined directions of earthquake input (two horizontal and one vertical taken orthogonally) is evaluated separately. Horizontal and vertical seismic loads are combined using the square root of the sum of the squares. Equipment is designed to withstand the combined effects of normal operating loads, acting simultaneously with the earthquake loadings, without loss of safety function or structural integrity.

3.7B.3.1.1.2 Dynamic Analysis

A detailed dynamic analysis is performed when equipment complexity, or dynamic interaction, precludes static analysis, or when static analysis is too conservative. Dynamic analysis methods include:

1. Response spectrum modal analysis
2. Time-history by modal superposition
3. Time-history by numerical integration

The response spectrum modal analysis technique is used most commonly.

Modeling

The lumped mass, or the consistent mass, approach is employed in the dynamic analysis. In the lumped and the consistent 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 be associated with each point. In accordance with minimum spacing requirements, the analyst can then choose, for the model, particular mass points which reflect predominant masses of subcomponents which contribute significantly to the total response.

Modeling of equipment for dynamic analysis starts with the calculation of lumped masses at discrete stations and the evaluation of the elastic properties (or stiffness) of connecting members. The number of discrete mass points is selected to adequately describe the dynamic characteristics and natural frequencies of the equipment. General modeling guidelines indicate that good natural frequency characteristics are obtained when the number of discrete mass or node points selected is twice the number of the highest mode of interest. Papers prepared by Lin and Hadjian (1976), Johnson (1977), and Lin (1974), along with Sections 3.7B.2.1 and 3.7B.2.3, provide supplemental modeling guidance.

Response Spectrum Modal Analysis

The normal mode approach is employed for seismic analysis of equipment and components. Natural frequencies, eigenvectors, participation factors, and modal member-end forces and moments of the undamped structure are calculated. The system of equations which describes the free vibrations of an n-degree of freedom, undamped structure is:

$$[M]\{\ddot{X}\} + [K]\{X\} = 0 \qquad (3.7B.3\text{-}1)$$

where:

[M] = Mass matrix for assembled system

[K] = Stiffness matrix for assembled system

$\{\ddot{X}\}$ = Nodal acceleration vector

{X} =Nodal displacement vector

The mode shapes and frequencies are solved in accordance with:

$$([K] - \omega_n^2\ [M])\ [\phi]_n = 0 \qquad n = 1, 2, 3, \ldots N \qquad (3.7B.3\text{-}2)$$

where:

ωn = Natural frequency of the nth mode

$[\phi]_n$ = Mode shape vector for the nth mode

n = Number of significant modes considered

Eigenvector-eigenvalue extraction techniques, 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.
The modal participation factor for the nth mode specific direction, i, is defined by

$$\Gamma_{ni} = \frac{[\phi]^T\ [M][\Delta]}{[\phi]^T\ [M][\phi]} \qquad (3.7B.3\text{-}3)$$

where:

$[\phi]^T$ = Transpose of mode shape vector for the nth mode

$[\Delta]$ = Earthquake direction vector referring to direction i

The modal member-end forces and moments are determined by

$$[F_n] = [K_m]\,[\phi]_n \qquad (3.7B.3\text{-}4)$$

where:

$[K_m]$ = Member stiffness matrix

For each modal frequency, the corresponding response acceleration is determined for a given level of equipment damping from the applicable response curve.
The maximum response for each mode is found by computing:

$$\begin{aligned} \{\ddot{X}\}_n &= \Gamma_{ni} R_{ni} [\phi]_n \\ \{\dot{X}\} &= \frac{1}{\omega_n}\{\ddot{X}\}_n \\ \{X\}_n &= \frac{1}{\omega_n^2}\{\ddot{X}\}_n \\ \{F\}_n &= \frac{\Gamma_n R_{ni}}{\omega_n^2}\{F_n\}_n \end{aligned} \qquad (3.7B.3\text{-}5)$$

where:

$\{\ddot{X}\}$ = The modal acceleration

$\{\dot{X}\}$ = The velocity

$\{X\}$ = The displacement

$\{F\}$ = The member-end force and moment vectors

R_{ni} = The spectral acceleration for the nth mode

The basis for the combination of maximum modal response is discussed in Section 3.7B.3.7.

Time-History Methods

There are two separate approaches to the solution of the equations of dynamic equilibrium for time-history motions. The modal superposition method involves the modal solution of the free vibration response of the system, and transformation to normal coordinates using the mode shapes of the system. This procedure uncouples the equations of motion so that the response of the system in each individual mode may be evaluated independently. Total system response is determined by combining responses of individual modes oscillating simultaneously. The second method of time-history analysis is the direct integration solution that includes numerical integration of the simultaneous differential equations of dynamic equilibrium without transformation to normal coordinates. System response at each time point is evaluated by this technique.

Time-history solutions, due to their analytical complexity, are only used when results of other methods are too conservative and when acceptable numerical solutions are available in a computer program. Equations of motion for time-history solutions are contained in Section 3.7B.2.1. Computer program capabilities for time-history solutions are outlined in Appendix 3A, Section 3.2.

3.7B.3.1.1.3 Testing

Equipment and components that use seismic testing as their qualification basis conform to the following general instructions for earthquake testing. These requirements conform with other applicable industry standards such as IEEE 344-1975, Section 3.10, or provide guidance for testing where no such standards are available. Equipment packages or components are shown to be seismically adequate by being tested individually, as part of a simulated structural section, or part of an assembled module or unit. In any case, the minimum acceptance criteria include:

1. No loss of safety function or ability to function before, during, or after the test.

2. No structural/electrical failure (i.e., connections and anchorages) that would compromise safety related component integrity.

3. No adverse or maloperation before, during, or after the proposed test that could result in an improper safety action.

Equipment vendors and suppliers are required to formulate programs for qualifying the equipment in accordance with the conditions specified in the seismic design requirements contained in the equipment specifications. The vendor must submit a summary of the proposed effort for review and approval.

The characteristics of the testing input at the equipment mounting locations are defined by amplified response spectrum curves, zero period response curve levels, time-history motions, or combinations of these as applicable.

The use of single and multifrequency input testing is accepted as a method of seismic qualification, based upon the particular plant site, 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 and response characteristics of the equipment generally dictate 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 envelop multiple peak floor responses, as well as single peak, providing sufficiently high forcing is used. For equipment exhibiting multiple response modes, single frequency input may be used, providing the input has sufficient intensity to envelop the floor response spectra of the individual modes of the equipment.

The testing machine (fixture) setup is arranged so that the equipment tested is mounted to simulate, to the extent possible, the actual service mounting, with no dynamic coupling to the test item. Equipment is tested in the operating condition wherever possible, and functions are monitored and verified both during and after testing. However, a true operating environment is sometimes not obtainable for equipment such as pumps, etc.

The in situ application of vibratory devices, to superimpose the seismic vibratory loadings on the complex active device for operability testing, is acceptable when application is justifiable.

The test program may be based upon selectively testing a representative number of mechanical components according to type, load level, size, etc., on a prototype basis.

In addition to the single and multifrequency testing programs outlined, laboratory shock results, in-shipment shock data, or adequate historical dynamic adequacy data (i.e., previous relevant test or environmental data) are also given consideration. The test method selected must demonstrate the adequacy of principal structural and functional capability of the equipment.

General testing guidance criteria specified for equipment include the following:

1. Single Frequency Testing

   Testing is performed for as much of the range between 1 and 33 Hz as practicable or justified. Input for qualification should, as a minimum, equal the zero period response curve level.

2. Sinusoidal Input

   a. A frequency scan (two octaves per minute, maximum), at a constant acceleration level, is performed over the frequency range of interest. 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 (damping independent).

   b. A dwell test of the equipment at its fundamental natural frequency is included at the acceleration values specified in item (a) above. Additionally, other frequencies are selected if amplification factors of 2.0 or more are indicated. A minimum 20 second duration is considered acceptable for each dwell.

   c. Other methods of sinusoidal testing may be employed as justified. Included are exploratory tests, per item 2(a), which employ a low acceleration level input to identify equipment response characteristics, and to aid in selecting requirements for further testing. A geometrically spaced constant frequency input may be employed for further testing. Intervals of one-half octave or less are employed for this spacing.

3. Sine Beat Input

   A sine beat test may be performed in conjunction with a sine scan, as an alternative to the dwell portion of the program outlined in item 2(b). 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 produce a magnitude of equipment response most nearly 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 a lower number of cycles is shown sufficient to duplicate or exceed the response spectra for the equipment at the appropriate location.

An alternative qualification program consists of applying a series of sine beats at geometrically-spaced frequency intervals of one-half octave or less over the frequency range of interest. The peak amplitude of the beat employed is, as a minimum, the maximum rigid range acceleration indicated by the relevant response spectrum curve.

4. Multi-frequency Testing

Multi-frequency testing is applicable as a general qualification method. Input excitation in this category includes time history, random, power spectral density, complex wave shapes, and others as justified. For the type of input applied, the testing machine input must equal, as a minimum, the zero period acceleration of the applicable response curve. A frequency range of 1 to 33 Hz is normally considered.

5. Time-History Input

An acceleration time-history of the equipment support location, or one based on a synthesized response curve, may be used as testing machine input. A 15- to 30-second time history input is normally employed. The test table input must develop a response curve which envelops the relevant response spectrum curve when a synthesized record is used.

6. Random Motion Input

Random input testing is performed so that the applicable response spectrum curve is enveloped by that produced by the table motion. The input is controlled by one-third octave (or less) bandwidth filters over the frequency range of interest with a minimum of 15 seconds or greater test duration. Normally, random tests are performed to produce a response curve based on test machine input which envelops the relevant response spectrum curve. A special case random test may be performed when a power spectral density equivalent of the applicable response curve is specified.

Tests combining random input in conjunction with other waveforms may be employed as justified.

7. Complex Wave Test

   A complex wave test may be performed by subjecting the equipment to an input motion, generated by summing a group of decaying sinusoids spaced at one-third octave, or narrower frequency intervals, over the frequency range of interest. Individual decay rate controls of from 0.5 to 10 percent are used. Response curves based on test table input must be shown to envelop the relevant response curve.

3.7B.3.1.2 Piping Systems

Analyses of Seismic Category I (including all ASME Code Classes 1, 2, and 3 piping systems) piping are performed by the modal analysis response spectra method. Nonseismic piping is seismically analyzed when its failure could result in unacceptable damage to a Seismic Category I system. Either a modal analysis response spectra method or equivalent static load method of analysis is utilized for the latter.

The criteria and procedures used for modeling are described in Section 3.7B.3.3.2. A typical mathematical model of a piping system is shown on Figure 3.7B–65. Defined boundaries, such as equipment and pipe anchors, may be considered as isolating the piping system when it is undergoing earthquake excitation.

The modal analysis response spectra method is used for dynamic analysis of Seismic Category I piping. Input for these piping systems is described as follows:

1. Amplified Response Spectra (ARS)

   Obtained for discrete locations in the structure where the piping systems are supported. Enveloping and peak broadening procedures are applied on these ARS curves before input, as described below. Damping values used for piping are 0.5 percent for OBE and 1 percent for SSE (Section 3.7B.1.3), except that increased damping values may be applied on an as-needed basis for final stress reconciliation (or piping system backfits) in accordance with ASME Code Case N-411 (Figure 3.7B–71).

2. Seismic Piping Anchor Movements

   Seismic piping anchor movements are obtained from seismic displacements of structures at piping anchor and support locations. These movements are used as static input to calculate the resulting internal forces and moments throughout the piping system. The methods used to consider differential piping support movements at different support points are discussed in Section 3.7B.3.8.

Where a piping system is subjected to more than one response spectrum, as when support points are located in different parts of the structure or in separate structures, an enveloping procedure as well as peak broadening is applied to generate a composite, or worst-case, spectrum for analysis. Peak broadening of minus 15 percent and plus 15 percent of peak frequencies is provided to account for uncertainties in the calculated values of structural frequencies. Accordingly, piping systems designed using those amplified response spectra having natural frequencies within ±15 percent of the peak resonant frequency are assigned the peak response value(s). Outside this range, the amplified response spectra is used exactly as stated. The response spectra modal analysis provides peak response quantities for each mode which are then combined according to Section 3.7B.3.7. All significant dynamic modes of responses under seismic excitation with frequencies less than 50 cps or modes less than 50, whichever is reached first, are included in the dynamic analysis described in Section 3.7B.3.8. The combined seismic responses, together with internal forces and moments due to seismic anchor movements, are then combined with other loadings according to ASME Section III Code, Articles NB 3600 (Class 1 piping), NC 3600 (Class 2 piping), or ND 3600 (Class 3 piping).

Time-history modal superposition analysis is employed for fluid-induced transient dynamic problems (e.g., water hammer and steam hammer), but is not normally used for piping seismic analysis.

Small size seismic Category I piping systems (Section 3.7B.3.5.2) are seismically qualified, in part, by the application of standard span procedures. The standard span procedures are restricted to small bore piping systems (one inch and below ASME Class 1 and two inch and below ASME Class 2, 3, and ANSI B31.1) which meet the criteria of the prequalified analysis.

No tests or empirical methods are used in lieu of analytical methods for all Seismic Category I piping.

Detailed descriptions of analytical procedures and design criteria for Seismic Category I piping can be found in Section 3.7B.3.8. The type of seismic analysis used and the criteria used are summarized in Table 3.7B–26.

3.7B.3.2 Determination of Number of Earthquake Cycles

3.7B.3.2.1 Equipment and Components

ASME III (NB 3112,3b) requires that the number of earthquake cycles to be used in the analysis of ASME Code Class 1 components be specified as part of the design mechanical loads. The following criteria are used for all components within the jurisdication of this code:

1. A total of five OBE and one SSE is assumed.

2. A minimum of 10 maximum stress cycles per earthquake is assumed. Alternatively, the number of cycles per earthquake may be obtained from the structural time-history analysis.

3.7B.3.2.2 Piping Systems

All ASME Class 1 piping systems are designed for a minimum of 10 maximum stress cycles per seismic event in the analysis. A total of five OBE and one SSE is assumed.

3.7B.3.3 Procedures Used for Modeling

3.7B.3.3.1 Equipment and Components

The procedures used for modeling of equipment and components are contained in Section 3.7B.3.1.

3.7B.3.3.2 Piping Systems

The basic method of analysis used is the finite element stiffness method. In accordance with this method, the continuous piping is mathematically idealized as an assembly of elastic structural members connecting discrete nodal points. Nodal points are placed in such a manner as to isolate particular types of piping elements, such as straight runs of pipe, elbows, valves, etc., for which force-deformation characteristics can be categorized. Nodal points are also placed at all discontinuities, such as piping supports, concentrated weights, branch lines, and changes in cross section. Inertial characteristics of the piping system are simulated by discrete masses of pipe and pipe components (including all concentrated and eccentric masses such as valves and valve operators) lumped at selected node points. System loads other than weights, such as thermal forces and earthquake inertial forces, are also applied at the nodal points. The stiffness matrix of the piping system is calculated based upon the elastic properties of the pipe and pipe components, to include the effects of bending, torsional, axial, and shear deformations. The stiffness of piping elbows, and certain branch connections, is modified to account for local deformation effects by the flexibility factors suggested in the ASME Section III Code, Articles NB 3600 (Class 1 piping), NC 3600 (Class 2 piping), and ND 3600 (Class 3 piping).

3.7B.3.4 Basis for Selection of Frequencies

3.7B.3.4.1 Equipment and Components

Amplified response spectra (floor) developed for two orthogonal horizontal and vertical direction earthquakes are the basic sources of seismic design accelerations. As noted in Section 3.7B.3.1.1, seismic accelerations are selected from the amplified response spectra based on natural frequency calculations for the equipment or component.

3.7B.3.4.2 Piping Systems

In the seismic design and multi-mass modal analysis of Seismic Category I piping systems (Section 3.7B.3.8), the practice of selecting piping fundamental natural frequencies to preclude resonance is not used.

3.7B.3.5 Use of Equivalent Static Load Method of Analysis

3.7B.3.5.1 Equipment and Components

Those components which are considered relatively simple or rigid are designed, by virtue of natural frequency calculations, to withstand the effects of amplified seismic acceleration values dependent upon frequency and amplitude ranges associated with the installation, location, and corresponding relevant amplified response spectrum. Analysis of components to the peak value of resonant response is considered conservative, since fundamental natural frequencies do not generally coincide with the frequency at resonance of the relevant response curve. Components having fundamental natural frequencies within the broadened response peak are designed to peak acceleration values, increased by a factor of 1.3, or as justified, to account for the contribution of all significant dynamic modes under a resonant condition. Generally, the vibratory characteristics of the components, qualified by resonant static analysis, are such that no possibility exists for adjacent or multiple modes to exist within the relatively narrow peak of a typical response spectrum.

The discussion which follows justifies the use of a factor of 1.3 as a conservative multiple to be applied to single or multiple degree-of-freedom systems having fundamental frequencies within the broadened resonant response peak. Multiply supported, or continuous type, span components are not part of this proof. When such cases arise, a factor of 1.5 times the peak resonant response is used.

3.7B.3.5.1.1 Single Degree-of-Freedom Systems

Peak broadening is intended to reflect a range of uncertainty in the precise location of the resonant peak of the response curve, and not to indicate that the multiple peak resonant response is possible within this broadened range. What is concluded is that there is a fairly equal chance that the peak of the curve (singular) would fall in the specified range and, thus, what exists, in fact, is a family of resonant response curves, each having only one point of peak resonant response (Figure 3.7B–66). If more than one system or component mode of vibration falls within the broadened peak, one and only one mode (a presumed worst case) can be presumed at an actual response peak value (Figure 3.7B–67). All other possible modes would realistically respond to lower values. Using the simple vibration theory and some simplifying assumptions, it is shown that a factor of 1.3 is conservative.

A simple damped oscillator responds with a transmissibility:

$$TR = \frac{\sqrt{1 + \left(2\beta\frac{\omega}{\omega_n}\right)^2}}{\sqrt{\left[1 - \left(\frac{\omega}{\omega_n}\right)^2\right]^2 + \left(2\beta\frac{\omega}{\omega_n}\right)^2}} \qquad (3.7B.3\text{-}6)$$

where:

ω_n = The undamped natural circular frequency

w = The frequency of the exciting force

The value of TR is dependent on the damping value, and the ratio of exciting frequency to oscillator natural frequency. When the exciting frequency equals the oscillator natural frequency, the steady state input is amplified by the value of TR and the response amplitude is maximum. In a seismic environment, maximum response is equal to the peak of the amplified response spectrum curve.

If additional modes are assumed around the peak of the response curve (Figure 3.7B–67), values of TR can be determined for each mode and the square root of the sum of the squares (SRSS) of these values computed. It is shown that TR increases as the number of modes increase, and that the most conservative placement of assumed modes is with one mode at the peak and other modes centered around this peak.

Data shown on Figure 3.7B–68 justify the use of a factor of 1.3 as conservative for all potential equipment applications. The curves are developed for two planes representing five modes and nine modes assumed acting within the broadened resonant peak. These numbers are intended to show an upper bound for general equipment application. Equipment damping values of 2.0 and 3.0 percent are used for static analysis. Higher damping is shown to indicate the trend and the conservatism of this method.

As further conservatism, all modes are considered participating equally. This is never the case in dynamic analysis. The higher frequencies of the component are given equal weight to the fundamental resonant frequency and the modes are centered on the nominal response curve. If the fundamental frequency were placed on the peak of the nominal curve, the results would show even lower transmissibilities.

The factor 1.3 is applicable only for those components whose fundamental natural frequency falls within the broadened response peak.

It has been shown that, for the range of values associated with component and system static analysis, use of the 1.3 factor is conservative. In fact, for a predominant number of likely cases, a value far less than this could be justified on the basis of the data.

For example, a value of 1.1 could be justified for most components which present only a few significant modes of vibration within the broadened response peak. It is further emphasized that, in reaching these conclusions, the most conservative assumptions regarding location of the nominal response curve and the placement of response modes for the arbitrary component have been made.

3.7B.3.5.1.2 Multi-Degree-of-Freedom Systems

As a conclusive supplement to the previous discussion, a study was performed utilizing rigorous dynamic analysis of models closely representative of typical components.

This investigation consists of computing the ratio of maximum dynamic stress to maximum static stress (i.e., the factor denoted by K) for several model beams subject to a flat response and typical amplified response spectra. Since bending stress is dominant for frame/equipment construction, the actual ratio employed equals:

$$K = \frac{\text{maximum dynamic moment}}{\text{maximum static moment}} \qquad (3.7B.3\text{-}7)$$

Both SRSS and absolute (ABS) moments are computed for comparison purposes, but conclusions are based solely on SRSS moments because they most closely represent actual dynamic stress. Maximum static moment corresponds, in the case of the 1 g flat response, to 1 g static load. In the case of a typical amplified response, the maximum static load is based upon the following frequency relationships (Figure 3.7B–69):

$f_o \leq f_p$, $g = g_{max}$ (peak acceleration)

$f_o > f_p$, g = acceleration at f_o (3.7B.3-8)

where:

f_o = The fundamental frequency of the model beam

f_p = The frequency at which the peak acceleration occurs

The effect of peak spreading is investigated by using a flat response, thus giving all modes the same acceleration. This is equivalent to infinite peak spreading. The importance of the uncertainty in the location of the peak acceleration with respect to the fundamental mode of the model beams is examined by adjusting the fundamental frequency from well below to well above the peak resonant frequency of a typical response spectrum.

The model beams selected for this study are shown on Figure 3.7B–70. These beams are typical of the frames and equipment combinations used in nuclear power plants. All dynamic analyses were conducted using the STRUDL-SW computer program described in Appendix 3A. Static analyses were carried out by hand, except for the simple/fixed beam with overhang. Consistent with design practice, all mountings in this study are assumed rigid.

3.7B.3.5.1.3 Results for Flat Response

Table 3.7B–23 summarizes the results for a 1 g flat response applied to the model beams of Figure 3.7B–70. Three K factors were computed for comparison purposes:

$$K_{s/c} = \frac{\begin{array}{c}\text{Maximum SRSS}\\ \text{dynamic moment}\end{array}}{\begin{array}{c}\text{Maximum static moment}\\ \text{from concentrated load}\end{array}}$$

$$K_{s/u} = \frac{\begin{array}{c}\text{Maximum SRSS}\\ \text{dynamic moment}\end{array}}{\begin{array}{c}\text{Maximum static moment}\\ \text{from uniform load}\end{array}} \qquad (3.7B.3\text{-}9)$$

$$K_{a/u} = \frac{\begin{array}{c}\text{Maximum ABS}\\ \text{dynamic moment}\end{array}}{\begin{array}{c}\text{Maximum static moment}\\ \text{from uniform load}\end{array}}$$

All conclusions in this study are based on $K_{s/u}$ because it most closely represents the actual ratio of dynamic moment to static moment. $K_{a/u}$ was not chosen because, as Table 3.7B–24 illustrates, modes are so widely spaced that no more than one modal frequency lies within a ±10 percent frequency band. $K_{s/c}$ is shown since this is the K factor which represents a typical simplification used in component analysis (concentrated static loads at component center of gravity).

The 1 g flat response was selected to give infinite peak spreading. As can be seen, $K_{s/u}$ was never greater than unity.

3.7B.3.5.1.4 Results for Amplified Response

Table 3.7B–25 presents the results for the simply supported/fixed model beam with 33 percent overhang subjected to the response spectra of Figure 3.7B–69. The first mode column on this table gives the fundamental frequency, f_o, and response acceleration, g, at f_o. Note that f was adjusted (by density variation) from well below to well above the peak frequency, f_p, of the response spectra to determine the effect on K of the uncertainty in the location of the peak frequency with respect to the fundamental frequency of the model beam. Since all values of $K_{s/u}$ were less than unity, it is concluded that this uncertainty has no important effects on the K factor.

3.7B.3.5.1.5 Conclusions

1. Peak acceleration times 1.3 applied as a static load to equipment whose fundamental natural frequency is within the broadened peak of the amplified response spectra curve is conservative for simply supported systems.

2. No amount of peak spreading can itself result in a $K_{s/u}$ factor significantly greater than unity.

3. Uncertainty in the frequency at which the peak response acceleration occurs itself has no important effects on the K factor.

4. Multiple supported continuous spans are not included in the scope of this study. Components or equipment which make up a system of continuous multiple span supports utilize a factor no less than 1.5 times peak acceleration as in item 1 above, if applicable.

3.7B.3.5.2 Piping Systems

Equivalent static load method is not generally used in the seismic analysis of Seismic Category I piping. However, seismic and certain nonseismic small bore piping and instrument tubing (Section 3.7B.3.1.2) are seismically supported at standard intervals based on maximum spans established from modal response spectra analysis of representative small bore piping configurations.

In some cases, when deemed appropriate, the analysis of small bore piping and instrument tubing is performed using methods based on simplified dynamic engineering formulations to ascertain their code adequacy.

The simplified dynamic formulation essentially involves the application of a factor of 1.3 to the peak value of ARS when the fundamental frequency of the piping or tubing configuration is less than 33 Hz while accounting for the seismic effects. The factor of 1.0 is applied when the fundamental frequency of the piping or tubing configuration is greater than or equal to 33 Hz.

3.7B.3.6 Three Components of Earthquake Motion

3.7B.3.6.1 Equipment and Components

In the seismic analysis of equipment and components, each of the three defined directions of earthquake input (two horizontal and one vertical taken orthogonally) is evaluated separately. The stresses resulting from the orthogonal earthquake inputs are combined by the square root of the sum of squares (SRSS).

3.7B.3.6.2 Piping Systems

In the seismic analysis of piping systems, the effects of simultaneous action of three spatial components of earthquake motion are considered. When the response spectrum modal analysis method is used for seismic analysis, the maximum piping modal responses (e.g., moments and displacements) due to the three spatial components of earthquake motion are obtained by the modified SRSS method. In this method, when considering a particular vibration mode, responses in a particular direction due to the two horizontal direction excitations are combined first by the SRSS method and then combined with response (in this same direction) due to the vertical direction excitation by absolute sum method. It has been demonstrated that this modified SRSS method always results in conservative answers as compared with the SRSS method described in position C.2.1 of Regulatory Guide 1.92 (Chang 1973).
In mathematical terms, the modified SRSS method of response combination due to three spatial components of an earthquake at a particular node point can be expressed as:

$$(R_x)_N = \sqrt{(R_{xx})^2{}_N + (R_{xz})^2{}_N} + |R_{xy}|_N \qquad (3.7B.3\text{-}10)$$

where:

(R_{xx})
(R_{xy}) = Response in the x, y, or z direction, respectively for mode N
(R_{xz})

$(R_x)^N$ = Combined response in the x direction due to seismic excitation for mode N

Once this is accomplished, $(R_x)_N$, $(R_y)_N$, and $(R_z)_N$ are then combined independently for significant modes N from 1 to K by methods described in Section 3.7B.3.7, where K is the number of significant modes considered in the modal response combination.

3.7B.3.7 Combination of Modal Responses

The following methods are applicable to piping system analysis employing the modal analysis response spectra method.

1. If the modes are not closely spaced, the maximum responses of piping are obtained by taking the SRSS of corresponding maximum modal responses of the piping

attributed to individual significant modes. This is in agreement with Position C.1.1 of Regulatory Guide 1.92, dated February 1976.

2. If closely spaced modes exist, then the grouping method is employed to combine various modal responses from a dynamic modal analysis. This is in conformance with Position C.1.2.1 of Regulatory Guide 1.92.

For equipment and components, the following methods of combination of modal responses are employed for each direction of earthquake motion:

1. When performing response spectrum modal analysis, the representative maximum value of a particular response earthquake is obtained by taking the SRSS of corresponding maximum values of the response of the element attributed to individual significant modes of the structure, system, or component. Mathematically, this can be expressed as:

$$R = \sqrt{\sum_{K=1}^{N} R_k^2} \qquad (3.7B.3\text{-}11)$$

where:

R = The representative maximum value of a particular response of a given element to a given component of an earthquake,

R_k = The peak value of the response of the element due to the kth mode,

N = The number of significant modes considered in the modal response combination.

2. When performing response spectrum modal analysis, if closely spaced modes exist, the grouping method as described in Regulatory Guide 1.92 is employed to combine modal responses. Mathematically, this can be expressed as follows:

$$R = \left[\sum_{k=1}^{N} R_k^2 + \sum_{q=1}^{P} \sum_{l=i}^{j} \sum_{m=i}^{j} \left| R_{lq} R_{mq} \right| \right]^{1/2} \qquad (3.7B.3\text{-}12)$$

where:

R_{lq} and R_{mq} = Modal responses, R_{λ} and R_m within the qth group, respectively;

i = The number of the mode where a group starts;

j = The number of the mode where a group ends;

R, R_k, and N = Definition in position 1.1 of Regulatory Guide 1.92;

P = The number of groups of closely spaced modes, excluding individual separated modes.

3. The method used for combination of three components of earthquake motion is described in Section 3.7B.3.6.

3.7B.3.8 Analytical Procedures for Piping Systems

The general analytical procedure of the modal analysis response spectra method for piping systems is described in Section 3.7B.3.1.2. Basic steps and equations used in the analytical procedure are described below.

For the dynamic analysis, the piping is represented by a lumped mass, multi-degree-of-freedom mathematical model. The distributed piping mass is lumped at the system nodal points. The equation of motion for the system is:

$$[M]\{\ddot{X}\} + [C]\{\dot{X}\} + [K]\{X\} = \{F\} \qquad (3.7B.3\text{-}13)$$

where:

[M] = Mass matrix for assembled system

[C] = Damping matrix for assembled system

[K] = Stiffness matrix for assembled system

{X} = Nodal displacement vector = [X(t)]

$\{\dot{X}\}$ = Nodal velocity vector = $\{\dot{X}(t)\}$

$\{\ddot{X}\}$ = Nodal acceleration vector = $\{\ddot{X}(t)\}$

{F} = Applied dynamic force vector = {F(t)}, or = [M] $\{\ddot{U}g\}$ for seismic analysis

$\{\ddot{U}g\}$ = Seismic acceleration vector for points of pipe support

Equation 3.7B.3-13 is solved for the system dynamic response as follows: First, the frequency equation, obtained by removing the forcing and damping terms from the above equation, is solved for the system natural frequencies and mode shapes. Next, the natural mode shapes are used to effect an orthogonal transformation of the equation yielding a series of independent equations of motion uncoupled in the system modes. Then, the uncoupled equations are solved by the response spectrum method to obtain system response in each mode, and the individual modal results are combined to determine the total system dynamic response. The mathematical formulation of these steps is described in the following subsections.

3.7B.3.8.1 Natural Frequencies and Mode Shapes

First, the eigenvalues (natural frequencies) and the eigenvectors (mode shapes) for each of the natural modes are calculated by solving the frequency equation:

$$([K] - \omega_n^2\ [M])[\phi]_n = \{0\} \qquad (3.7B.3\text{-}14)$$
$$n = 1, 2, 3, \ldots, N$$

where:

W_n = Natural frequency of the nth mode,

$\{\phi\}_n$ = Mode shape vector of the nth mode,

$\{0\}$ = Null vector,

N = Number of significant modes considered

The eigenvalues and eigenvectors are obtained using the Householder-QR algorithm, Jacobi Reduction or Inverse iteration.

3.7B.3.8.2 Dynamic Response

Next, let {n(t)] be the generalized coordinate vector, substitute {X} = [ϕ] {n} into the equation of motion and pre-multiply by [ϕ]T; an orthogonal transformation results, from which the uncoupled equations of motion shown below are obtained

$$\ddot{\eta} + 2\beta_n\omega_n\dot{\eta} + \omega_n^2\ \eta_n = P_n \qquad (3.7B.3\text{-}15)$$
$$n = 1, 2, 3, \ldots, N$$

where:

$[\phi]$ = The square matrix of mode shape vectors

η_n = Generalized coordinate for the nth mode = η_{nt}

β_h = Damping ratio of the nth mode expressed as percent of critical damping

P_n = Generalized force of the nth mode

$P_n = \{\phi\}_n^T\ \{F\}/M_n$ for applied dynamic force {F}

$P_n = \{\phi\}_n^T\ [M]\ \{\ddot{U}g\}\ /M_n$ for seismic analysis

M_n = Generalized mass of the nth mode = $\{\phi\}_n^T\ [M]\ \{\phi\}_n$

Solutions to these differential equations are obtained by the method of amplified floor response spectrum (ARS) superposition, as described below.

3.7B.3.8.3 Response Spectrum Modal Analysis

The response of a piping system to seismic excitations is obtained using the method of response spectrum superposition. At any pipe support point, seismic input is produced by a set of three

ARS, one in each global coordinate direction. These ARS are generated from the application of time-history acceleration responses obtained from the structure or equipment time-history analysis. These ARS are peak broadened (Section 3.7B.2.9) to reflect variations in structure properties. Where a piping system is subjected to more than one set of (three) ARS, such as support points located in different structures or different parts of the same structure, the enveloping and peak broadening are applied to all sets of ARS at support points (Section 3.7B.3.1.2). Thus, after the enveloping and peak broadening process, a set of three ARS, one in each global coordinate direction, results. The maximum acceleration for the nth mode of the piping system in jth global coordinate direction is then given by

$$\{\ddot{X}\}_{nj} = \{\phi\}_n \ddot{\eta}_{nj}\max$$
$$\{\ddot{X}\}_{nj} = \{\phi\}_n \Gamma_{nj}/M_n \qquad (3.7B.3\text{-}16)$$

n = 1, 2, 3, ...N

j = 1,2,3 corresponds to response in X, Y, or Z global coordinate direction, respectively.

where:

$\{\ddot{X}\}_{nj}$ = Maximum acceleration vector of mode n

$\eta_{nj\ max}$ = Maximum generalized coordinate acceleration of mode n

Γ_{nj} = Modal participation factor for the nth mode in jth global coordinate direction

$\Gamma_{nj} = \{\phi\}_n^T [M] \{e\}_j$

$\{e\}_j$ = A vector with components of unity in all directions parallel to jth global coordinate direction and zero otherwise $(S_a)_{nj}$ = Spectral acceleration for the nth mode, in jth global coordinate direction (from enveloped and peak broadened ARS)

The maximum inertia force vector for the nth mode and in jth global coordinate direction is given by:

$$\{F\}_{nj}\max = M_N \{\phi\}_n \ddot{\eta}_{nj}\max$$
$$\{F\}_{nj}\max = \{\phi\}_n \Gamma_{nj} (S_a)_{nj} \qquad (3.7B.3\text{-}17)$$

These inertia forces are calculated for each of the system natural modes in all three global coordinate directions and applied as static forces in the same manner as the deadweight or equivalent thermal forces, to find internal moments and forces in each mode. The total maximum responses due to seismic excitation are then obtained by combining the modal responses described in Section 3.7B.3.7. The calculated primary stress range due to seismic inertial responses for ASME Code Class 1 piping components are added absolutely to the secondary stresses due to seismic anchor displacement calculated in the following manner.

Maximum relative displacements in two horizontal and the vertical direction between piping supports and anchor points (i.e., between floor penetrations and equipment supports at different elevations within a building, and also between buildings) are used as equivalent static displacement boundary conditions in order to calculate the secondary stresses of the piping system. Relative seismic displacements used are obtained from a dynamic analysis of the structures, and are always considered to be out-of-phase between different buildings to obtain the most conservative piping responses.

These seismic member moments and forces are then combined with loads from deadweight, pressure, thermal, and other mechanical loads to complete the stress analysis of all Seismic Category I, and some Non- Seismic Category piping. For ASME Code Class 1 piping, stress intensities, and cumulative usage factors of the piping system are computed based on the formulation specified in Subarticle NB 3600, ASME Section III for ASME Code Class 2 and 3 piping, the formulations in Subarticle NC 3600 and ND 3600, respectively, are used.

The design criteria, loading combination, and stress limits for BOP Seismic Category I piping systems are described in Section 3.9B.3.

3.7B.3.9 Multiply Supported Equipment and Components with Distinct Inputs

To calculate the maximum inertial response of multiply supported subsystems, an upper bound envelope of all the individual response spectra for the support locations is used. In addition, the relative displacements at the support points are considered. Support displacements are imposed statically on the subsystem in the most conservative combinations. For support locations within a Category I structure, relative displacements are determined algebraically and imposed. Displacements between Category I structures are considered to be out-of-phase, and the maximum relative displacements between Category I structures are thus determined from absolute sums of the support displacements. The stresses due to seismic inertia and relative displacements are added to those due to other appropriate loadings such as deadweight, pressure, etc., and the resulting stresses are limited by allowable stresses defined in applicable codes.

3.7B.3.10 Use of Constant Vertical Static Factors

3.7B.3.10.1 Equipment and Components

Constant load factors are not used for vertical floor response in the seismic design of Seismic Category I equipment and components.

3.7B.3.10.2 Piping Systems

The method of applying constant static factors as vertical response loads, based on the assumption of vertically rigid structures, for the seismic design of Seismic Category I piping is not used. However, a simplified analysis (equivalent static load method) using constant load factors for the vertical and horizontal directions based on the peaks of applicable amplified response spectra is used for seismic analysis of certain nonseismic piping systems (Section 3.7B.3.5.2).

3.7B.3.11 Torsional Effects of Eccentric Masses

The effect of eccentric masses such as valve operators is considered in the seismic piping analysis described in Section 3.7B.3.1.2. These eccentric masses are included in the mathematical model for the piping analysis and the torsional effects caused by them are evaluated and included in the total piping response.

3.7B.3.12 Buried Seismic Category I Piping Systems

In performing stress analysis of buried Seismic Category I piping systems, the loadings considered are:

1. Internal pressure

2. Soil pressure (includes dead load, and live loads due to traffic when applicable)

3. Thermal expansion

4. Differential movements between structures and adjacent soil due to settlement and seismic motion

5. Seismic wave effects

Effects of loadings (1) and (2) are assessed by well known methods (e.g., Terzaghi 1955; King, R.C. and Crocker, C. 1967, Section 21-29 to 21-33); effects of loadings (3) and (4) are accounted for by a static analysis considering piping modeled together with soil springs. This is basically a beam on elastic foundation approach. Loadings (4) and (5) are discussed in detail in the sections below.

The basic assumptions concerning buried piping stress analysis are:

1. piping satisfies elementary theory of beams;

2. soil is linear elastic, homogeneous and isotropic; and

3. soil strain is fully transferred to the pipe; i.e., there is no slippage between the pipe and the soil.

3.7B.3.12.1 Seismic Wave Effect in the Free Field

Various seismic waves develop during an earthquake. There are compression waves (P-waves), shear waves (S-waves), and different kinds of surface waves such as Rayleigh waves (R-waves). When seismic waves propagate through the soil, responses of buried Seismic Category I piping are calculated by making use of the analytical approach proposed by Goodling (1978, 1979, 1980).

Straight portions of buried piping far from the effect of external supports, bends and tees are assumed to move with the soil when seismic waves propagate through it. Since Rayleigh waves induce the highest axial strains in buried piping, only these waves are considered in the analysis. The strains of the soil have been conservatively established as follows:

$$\text{Axial strain } \varepsilon_m = \frac{V_m}{C_R} \tag{3.7B.3-18}$$

Bending curvature $\chi = a_m/C^2_R$

where:

V_m = peak ground velocity, in/sec

a_m = peak ground acceleration, in/sec^2

C_R = Rayleigh wave velocity, in/sec

Therefore, the stresses on the straight portions of buried piping mentioned above are given by:

$$\text{Axial Stress } \sigma_a = \frac{EV_m}{C_R} = \frac{F_{max}}{A_m} = \varepsilon_m E \tag{3.7B.3-20}$$

$$\textit{Bending} \text{ Stress } \sigma_b = \frac{ED_o a_m}{2C_R^2} = \frac{ED_o X}{2} \tag{3.7B.3-21}$$

where:

E = Young's modulus of pipe, psi

D_o = outside diameter of pipe, in

F_{max} = maximum axial force, lb

A_m = cross-sectional area of pipe, in^2

Seismic wave effect on bends and tees in the free field are considered separately. For a bend, the maximum stresses are determined by assuming that its longitudinal leg is in the direction of maximum soil strain and its transverse leg is in the perpendicular direction.

The longitudinal leg may terminate into another bend, an anchor, or a free end. The bend is classified accordingly and the actual slippage length (L') along which slippage between pipe and soil occurs is determined as outlined by Goodling (1978).

The net relative displacement Δ, between soil and pipe at the bend is given by:

$$\Delta_1 = \varepsilon_m L' - \frac{fL^2}{2A_m E} - \frac{S_1 L'}{A_m E} \tag{3.7B.3-22}$$

where:

$\varepsilon_m L'$ = theoretical unrestrained relative movement at the elbow over length L'.

S_1L'/A_mE = the pipe elongation due to friction along the soil/pipe interface.

$fL'^2/2A_mE$ = the pipe elongation due to friction along the soil/pipe interface.

f = frictional force per unit length of pipe, lb/in

The bend legs are considered as beams on elastic foundation for which its parameter λ is given by:

$$\lambda = 4\sqrt{\frac{k}{4EI}} \qquad (3.7B.3\text{-}23)$$

where:

k = soil spring constant, per unit length, lb/in^2

I = moment of inertia of pipe cross section, in^4

In case of long-transverse leg (its length is greater than 3π/4λ), the following equations are derived by incorporating the interdependence of forces, moments, soil deformation, and rotation of the pipe in the immediate vicinity of the bend (Goodling 1980).

$$M = \frac{\lambda\Delta_1}{R\phi/K'EI} \qquad (3.7B.3\text{-}24)$$

$$S_1 = \frac{k\Delta_1}{2\lambda} + \lambda M \qquad (3.7B.3\text{-}25)$$

where:

M = bending moment, in-lb/in

f = elbow angle, radians

R = radius of elbow, in

$$K' = 1 - \frac{9}{10 + 12\left(\frac{tR}{a^2}\right)^2}$$

t = actual pipe wall thickness, in

a = outside radius of pipe, in

In case of short transverse leg, the following conservative equations derived by Goodling (1978) are used:

$$\Delta_1 = \frac{\varepsilon_m L' - \dfrac{fL'^2}{2A_m E}}{1 + \dfrac{KL'}{A_m E\lambda} \dfrac{C_3}{2C_1C_3 - C_2^2}} \quad (3.7B.3\text{-}26)$$

$$S_1 = \frac{k\Delta_1}{\lambda} \frac{C_3}{2C_1C_3 - C_2^2} \quad (3.7B.3\text{-}27)$$

$$M = \frac{k\Delta_1}{2\lambda^2} \frac{C_2}{2C_1C_3 - C_2^2} \quad (3.7B.3\text{-}28)$$

where:

C_1, C_2, and C_3 are coefficients given by Goodling (1978)

The maximum axial force in the longitudinal leg, F_{max}, induced by the seismic motion is given by:

$F_{max} = S_1 + fL'$ (3.7B.3-29)

Having determined the values of S, M and F, the stresses due to local deformation at the bend can be evaluated. These stresses are superimposed on stresses caused by the curvature of the pipe during seismic wave propagation. The combined stresses at the bend are multiplied by an intensification factor (0.75 i) to account for the higher intensity of stresses at the elbow. The following expressions for stress result:

(1) Stress at an Elbow
S_{o1} (elbow) = $0.75i\,[ED_o\,\chi/2+M/Z] + S_1/A_m$ (3.7B.3-30)

(2) Stress in the Longitudinal Run
S_{o1} (long) = $F_{max}/A_m + ED_o\,\chi/2$ (3.7B.3-31)

where:

Z = section modulus of pipe, in^3

Occasionally, when the conservative equations used in case of short transverse leg result in unacceptable stresses, the technique incorporating the passive resistance of soil, as presented by Goodling (1978), may be used.
A similar approach is used for analyzing tees (Goodling 1978).

3.7B.3.12.2 Effects of Differential Movements Between Structure and Adjacent Soil Due to Seismic Motion

During an earthquake, differential seismic motions occur between structures and adjacent soil. These differential motions are obtained by the seismic analysis of structures with soil-structure interaction taken into account. The effect on the buried piping systems due to these differential motions can be evaluated by considering separately the effects of different components, namely, differential motion components transverse to the direction of the piping axis, and differential motion components parallel to the direction of the piping axis.

1. Differential motion components transverse to the direction of piping axis.

   The subgrade reaction approach is used here to simulate the effect of soil on the deformation and stress of the buried pipe due to differential motion components transverse to the direction of piping axis. The approach is based on Terzaghi's theory (Terzaghi 1955) that soil subjected to pressure behaves like a system of uniformly spaced elastic springs with predetermined stiffness. The soil is thus represented by a series of orthogonal pairs of elastic springs in directions transverse to the piping axis and attached to the piping in the mathematical model. The elastic springs in the vertical direction are calculated in accordance with Terzaghi (1955). The elastic springs in the horizontal direction are calculated based on the method described in Audibert and Nyman (1975). The maximum expected transverse seismic displacements at the structural penetration are used as input in the calculation. The computation is done by the computer program NUPIPE-SW, which is listed in Appendix 3A. In principle, this approach is basically a beam analysis on an elastic foundation (Hetenyi 1946).

2. Differential motion components parallel to the direction of piping axis.

   The effect on buried long straight piping due to differential seismic motions along the direction of piping axis at penetration is assessed by considering the frictional force between pipe and soil, and the maximum axial stress due to this effect is (Yeh 1974):

$$\sigma_a = \sqrt{\frac{2EF\delta_a}{t}}$$

where:

σ_a = Stress in piping at penetration point due to axial displacement, δ_a, of the structure at the penetration point (psi)

E = Young's modulus of pipe (psi)

F = Frictional force between pipe and soil (psi)

δ_a = Axial displacement of pipe at penetration point through structure (inch) t = Pipe thickness (inch)

When bends or tees exist close to the penetration, the effect of differential motion parallel to direction of piping axis is analyzed again by the method described in Shah and Chu (1974).

3.7B.3.12.3 Effects of Differential Movements Due to Structural Settlement

Settlement of a structure can happen either due to its own weight over a period of time or due to an earthquake. The settlement of the structure where the piping is connected, as well as the soil adjacent to the structure where the piping is buried, are imposed on the buried piping, and the approach outlined in “differential motion components transverse to the direction of piping axis” given above is used to evaluate stresses in the buried piping.

3.7B.3.12.4 Accommodations for Buried Piping Structural Penetrations

The resultant loadings imposed by thermal, structural, and seismic distortions may cause severe local stresses in buried piping at structure penetration points. The piping, if anchored at the structural wall, may be too stiff to accommodate these distortions for such locations. In such cases, the buried piping design includes a structural penetration, consisting of a concrete box or a conduit which is not attached to the structure, and is free to move with the soil rather than with the structure. Within the box or conduit, the piping may (if necessary) be provided with expansion joints or piping loops to accommodate relative displacements in both axial and transverse directions.

3.7B.3.13 Interaction of Other Systems (Piping and Equipment) with Seismic Category I Systems (Piping and Equipment)

Nonseismic category systems (piping and equipment) are designed to be isolated from Seismic Category I systems (piping and equipment). Isolation may be accomplished by physical restraints, barriers, or separation. If it is not practical to isolate the Seismic Category I system from the nonseismic, then the potential for damage due to seismic interactions is evaluated through the following program.
The Seismic Interaction Program consists of three distinct tasks:

- demonstrating the adequacy of equipment anchorages for nonseismic equipment in Seismic Category I buildings,

- demonstrating the structural integrity of piping and supports for selected subsystems, and

- performing walkdowns to identify swing/sway interactions between non-Seismic Category I piping and equipment and Seismic Category I piping and equipment.

Non-Seismic Category I equipment in Seismic Category I buildings is reviewed to ensure the seismic adequacy of its anchorage by one of the following methods:

1. verify that the equipment anchorage has been explicitly qualified;

2. compare the anchorage detail to explicitly qualified anchorages; and

3. for anchorages which are not seismically designed and are not similar to seismic anchorages, calculations are performed to demonstrate adequacy.

For equipment anchorages evaluated by Method 3, an effort is made to group typical anchorage details and perform bounding calculations. The manner in which structural integrity is demonstrated for each piece of interacting equipment is documented by the calculations. Acceptance criteria for equipment anchorage evaluations is given in J.F. Opeka to B.J. Youngblood Letter B11844, dated October 31, 1985, Docket No. 50-423.

The structural integrity of piping and supports is addressed through a program developed and implemented by Sargent and Lundy. A set of piping subsystems was selected to be representative of pipe sizes, hanger configurations, and operating conditions. As discussed in more detail in the above-referenced letter, these selected subsystems are bounding.

Restraint loads and pipe stresses were calculated using dynamic analysis results. They have been compared to the failure capacities associated with each subsystem to assess the inherent margin of safety in the design. Maximum dynamic lateral displacements are used to confirm interaction criteria utilized during plant walkdowns.

Seismic interaction walkdowns are conducted to identify swing/sway interactions between non-Seismic Category I piping and equipment and Seismic Category I piping and equipment. All interactions are evaluated considering the local flexibility of the interacting equipment/piping. These reviews address restrictions such as penetrations and interferences with structures which would limit displacements. In all cases, interactions with active Seismic Category I components is prevented.

Information gained from the experience database is used in conjunction with the above efforts. The database shows that properly supported equipment and piping maintains its structural integrity during strong motion earthquakes. Further, this program benefits from the database information regarding the severity of seismic interactions and the knowledge of configurations which have not performed well in past earthquakes.

For nonseismic category piping systems attached to Seismic Category I piping systems, the dynamic effects of the nonseismic category piping are simulated in the analysis modeling of the Seismic Category I piping. The nonseismic category piping is modeled in a manner consistent with the accuracy of the Category I piping analysis. The attached nonseismic category piping is also designed to ensure that, during an earthquake of SSE intensity, it does not cause a failure of the Seismic Category I piping system.

3.7B.3.14 Seismic Analysis for Reactor Internals

See Section 3.7N.3.

3.7B.3.15 Analysis Procedure for Damping

Damping values of equipment, components and piping systems are given in Section 3.7B.1.3.

FIGURE 3.7B–1 HORIZONTAL DESIGN RESPONSE SPECTRA 0.17G SSE

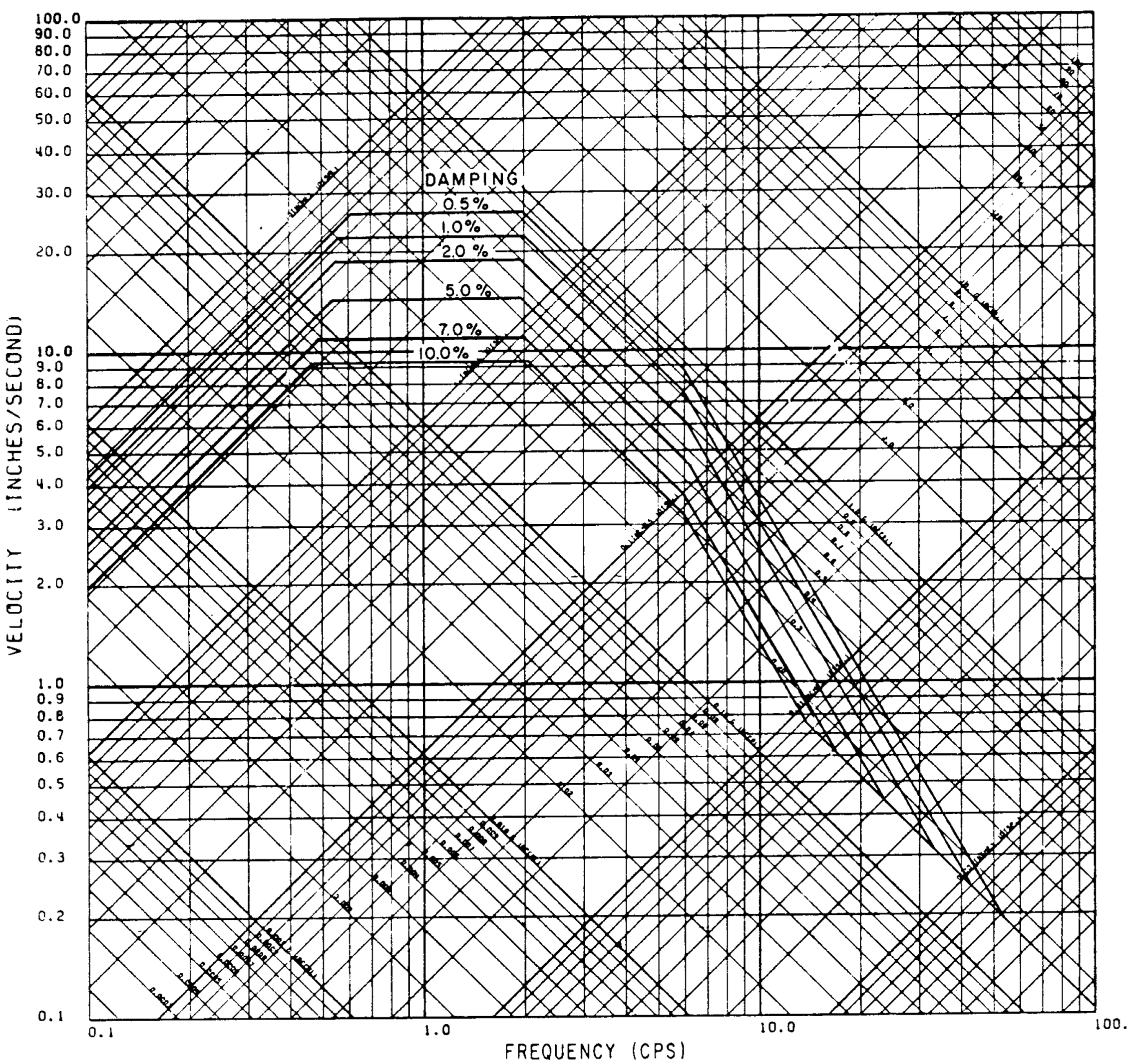


FIGURE 3.7B–2 HORIZONTAL DESIGN RESPONSE SPECTRA 0.09G 1/2 OBE

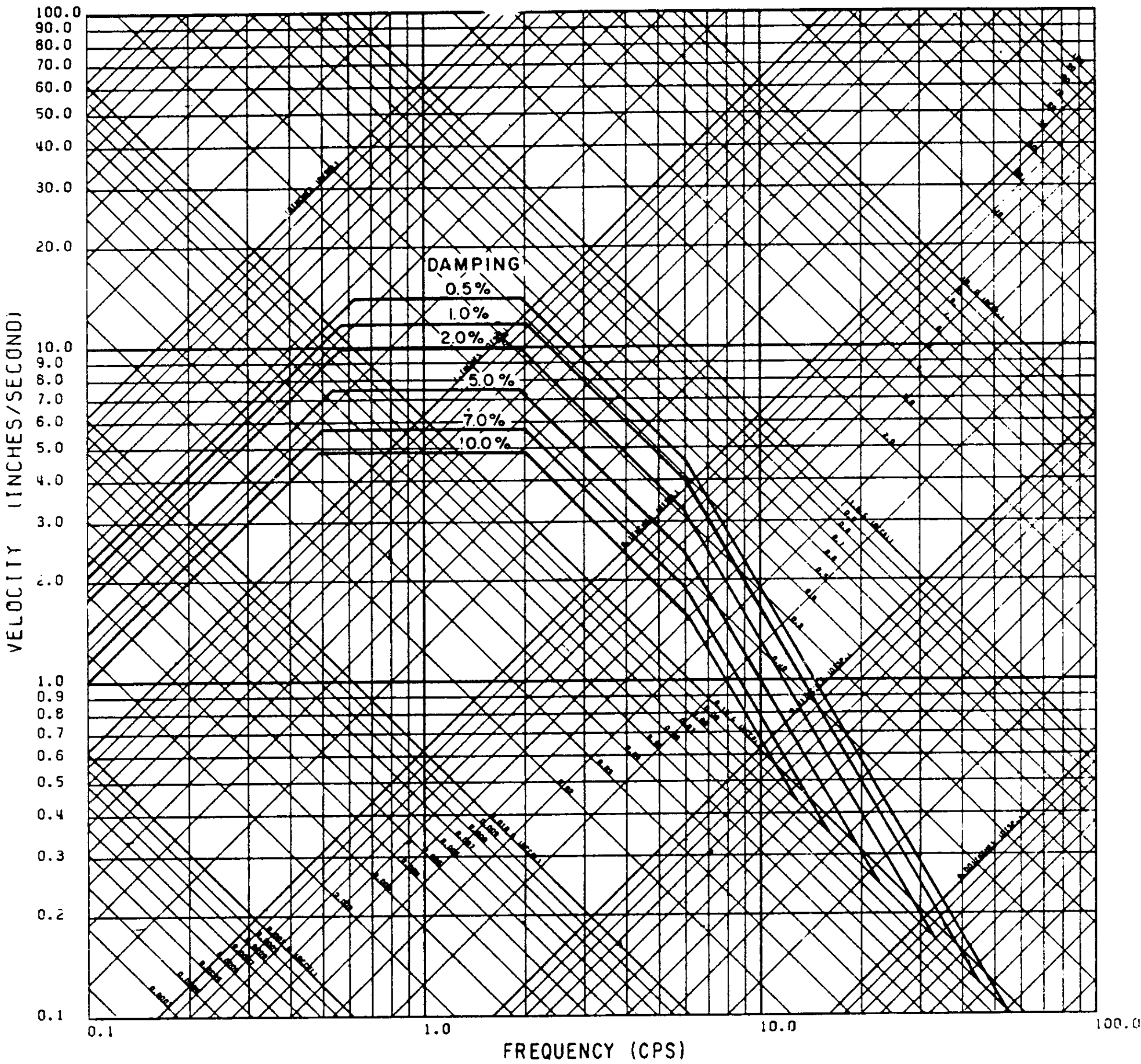


FIGURE 3.7B–3 HORIZONTAL TIME HISTORY RESPONSE SPECTRA 0.5 PERCENT DAMPING VALUE

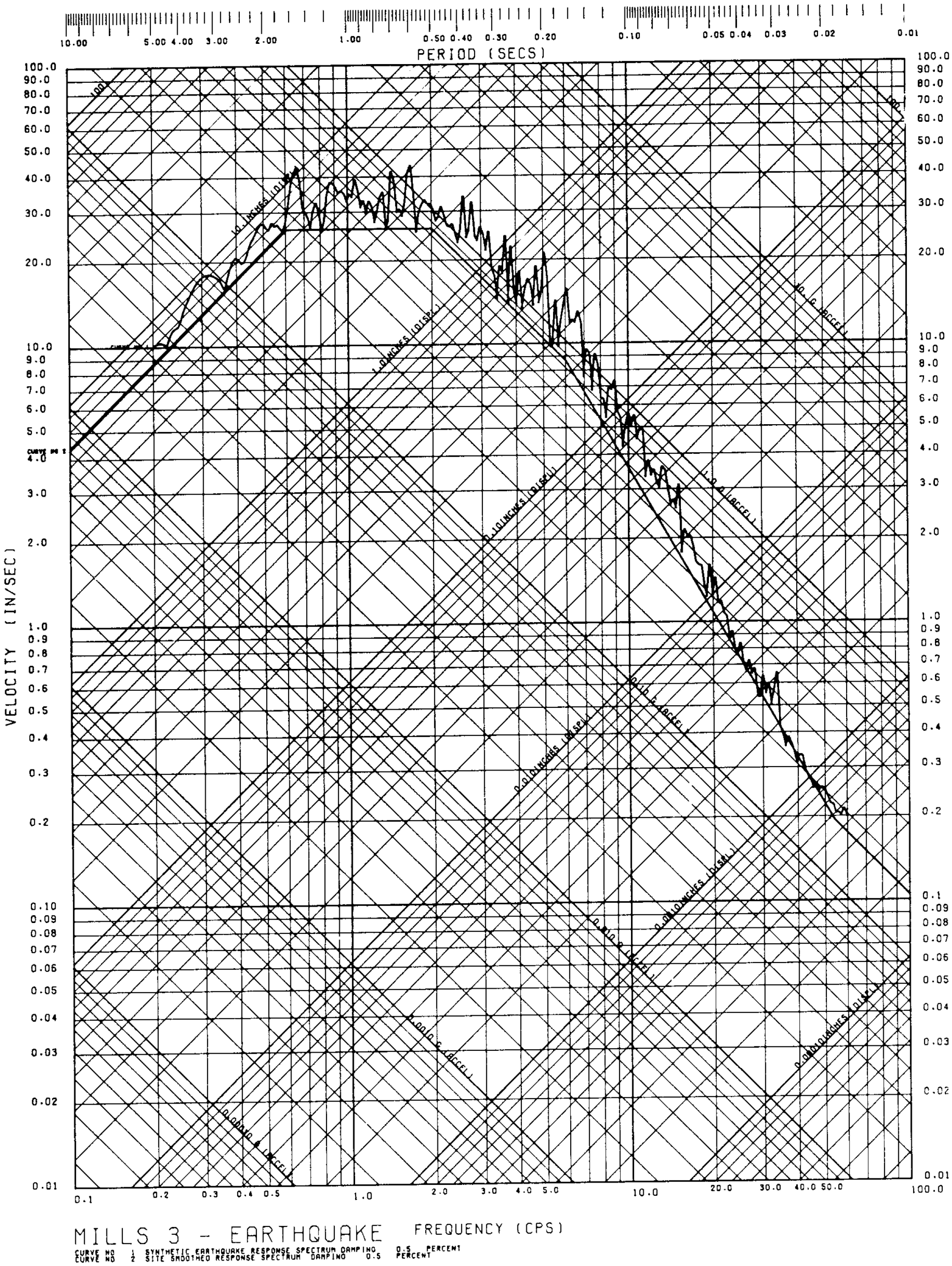


FIGURE 3.7B–4 HORIZONTAL TIME HISTORY RESPONSE SPECTRA 1.0 PERCENT DAMPING VALUE

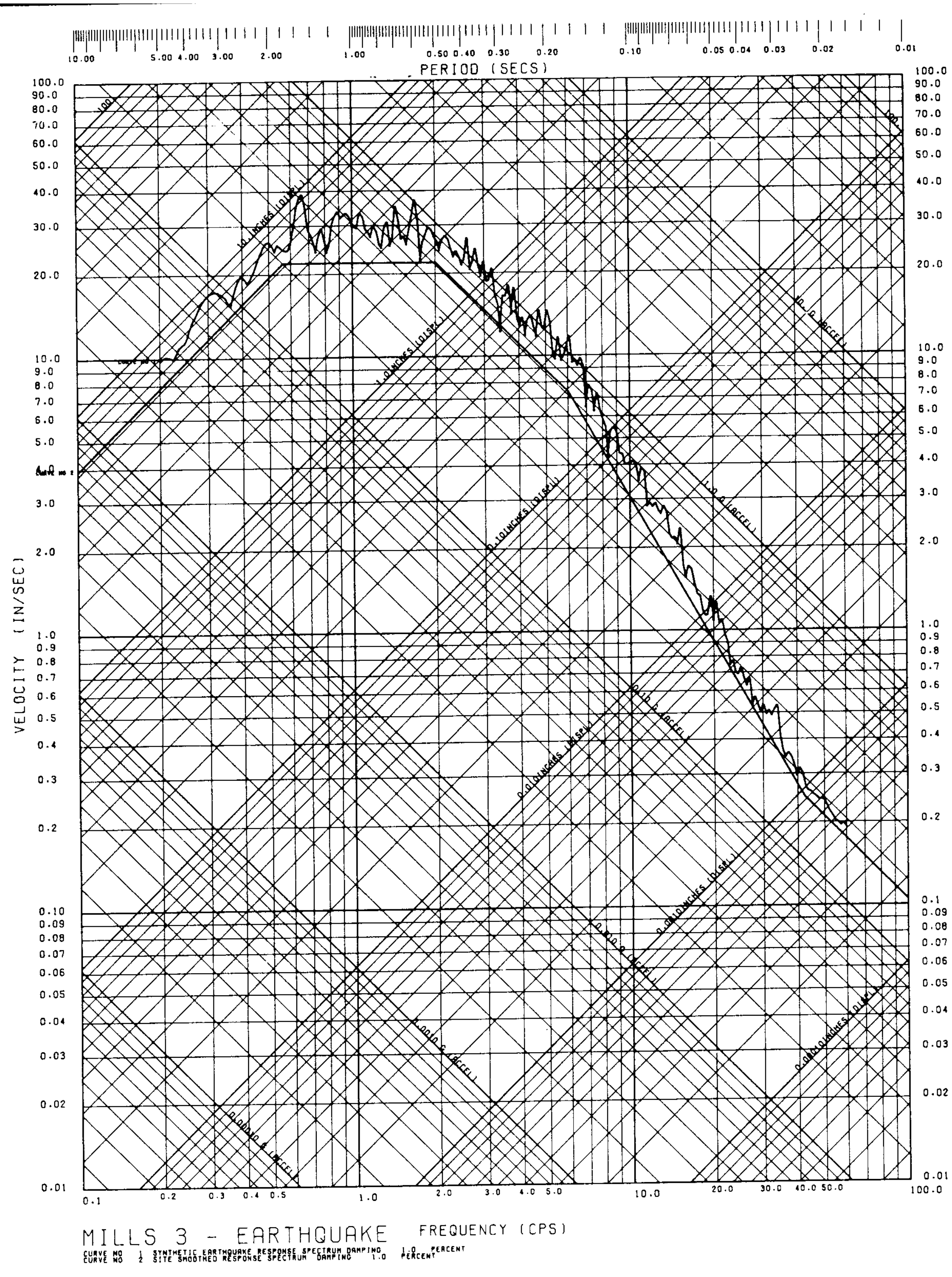


FIGURE 3.7B–5 HORIZONTAL TIME HISTORY RESPONSE SPECTRA 2.0 PERCENT DAMPING VALUE

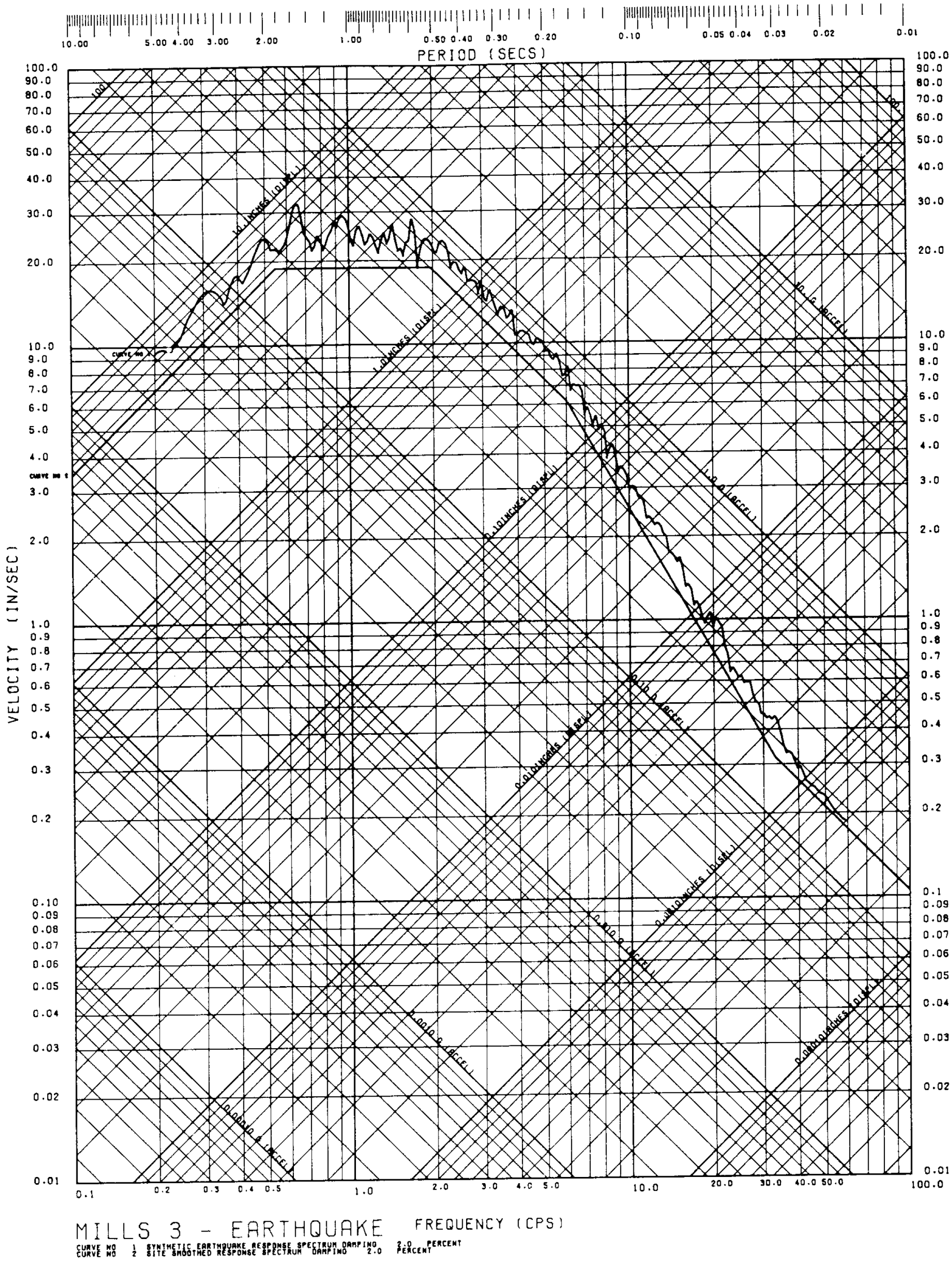
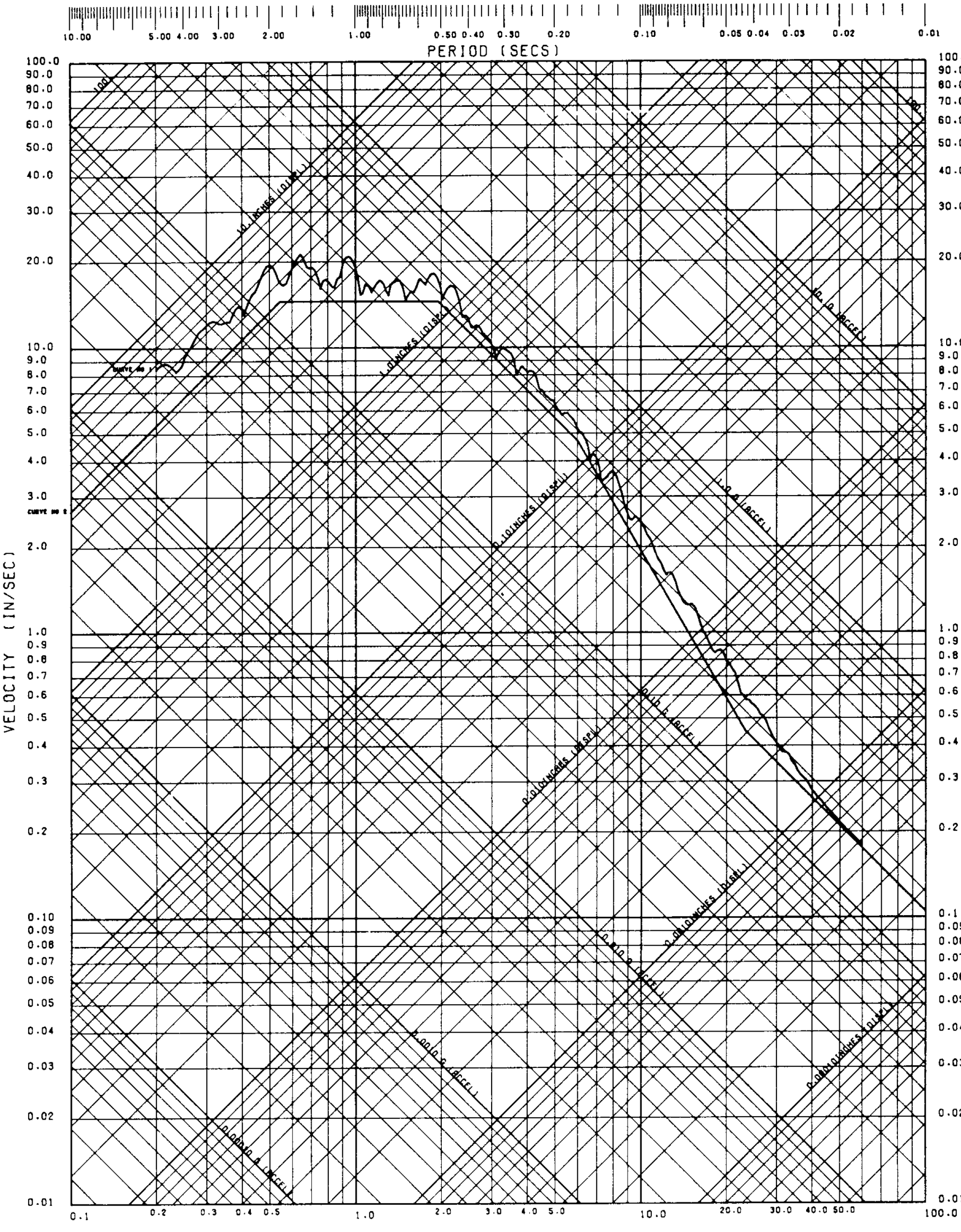


FIGURE 3.7B–6 HORIZONTAL TIME HISTORY RESPONSE SPECTRA 5.0 PERCENT DAMPING VALUE



MILLS 3 - EARTHQUAKE

CURVE NO 1 SYNTHETIC EARTHQUAKE RESPONSE SPECTRUM DAMPING 5.0 PERCENT
CURVE NO 2 SITE SMOOTHED RESPONSE SPECTRUM DAMPING 5.0 PERCENT

FIGURE 3.7B–7 HORIZONTAL TIME HISTORY RESPONSE SPECTRA 7.0 PERCENT DAMPING VALUE

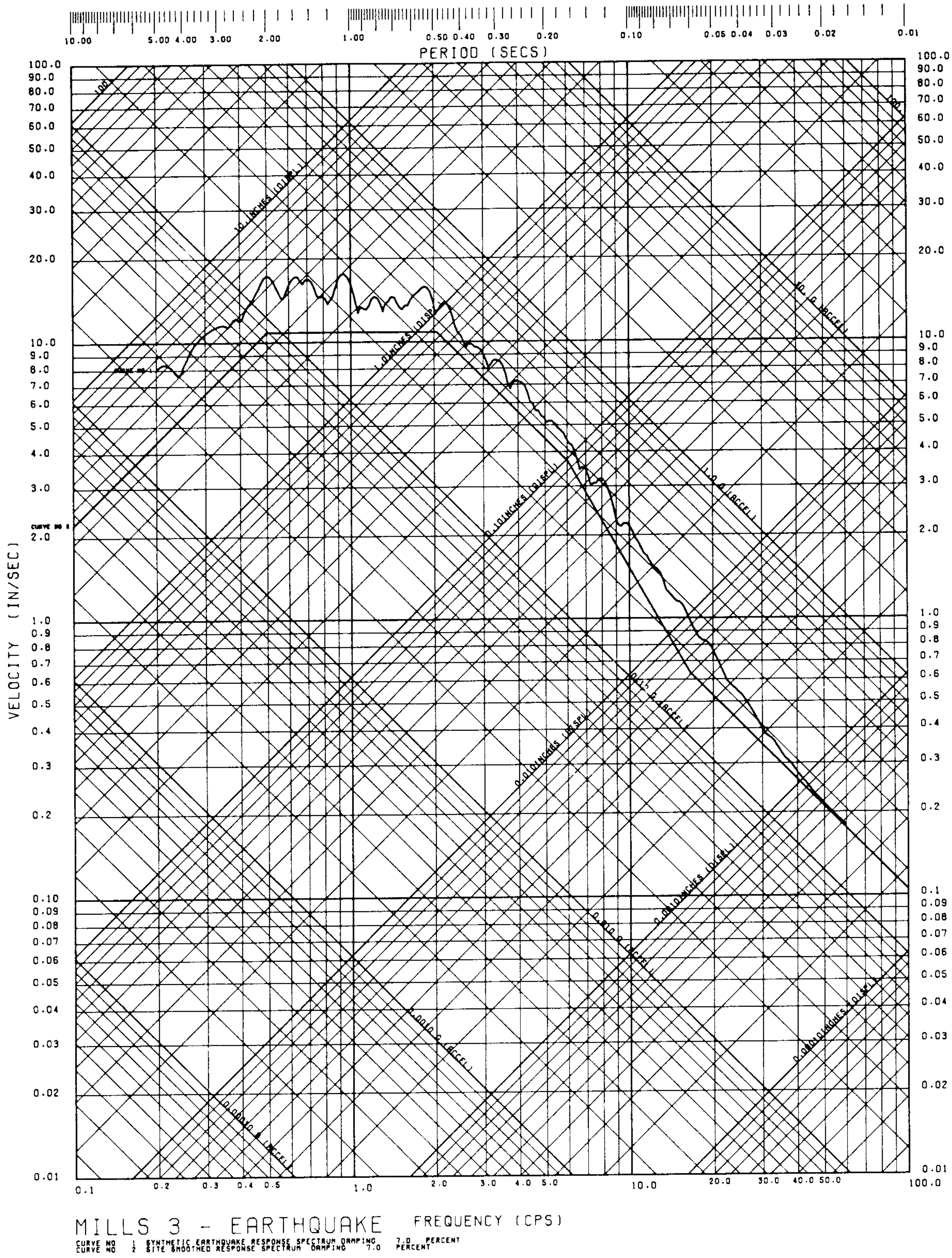


FIGURE 3.7B–8 HORIZONTAL TIME HISTORY RESPONSE SPECTRA 10 PERCENT DAMPING VALUE

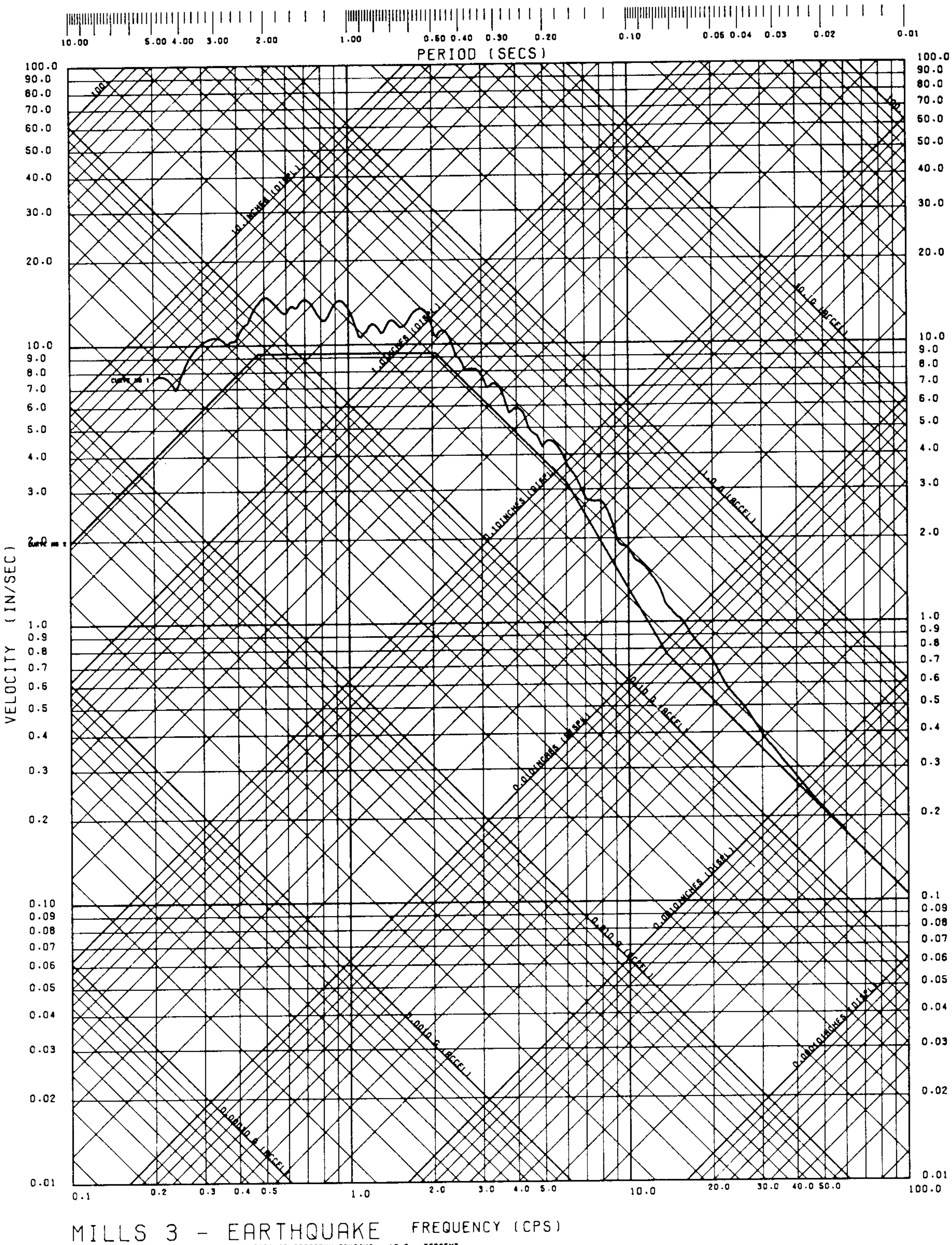


FIGURE 3.7B–9 DYNAMIC MODEL OF THE CONTAINMENT STRUCTURE

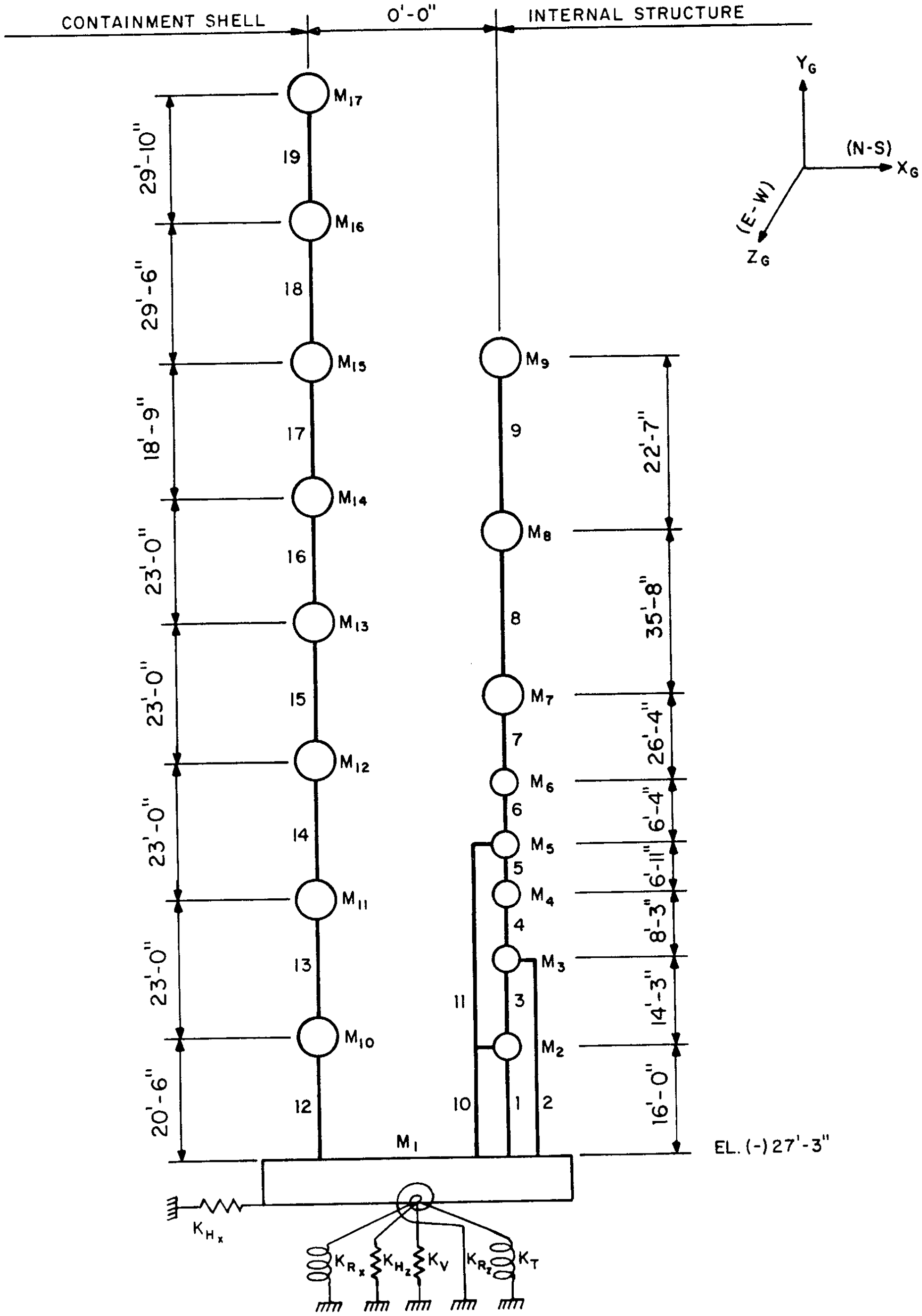


FIGURE 3.7B-9
DYNAMIC MODEL OF
THE CONTAINMENT STRUCTURE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

K_H, K_V, K_R, K_T =
HORIZONTAL, VERTICAL, ROCKING AND
TORSIONAL SUBGRADE SPRINGS

FIGURE 3.7B–10 DYNAMIC MODEL OF THE MAIN STEAM VALVE BUILDING

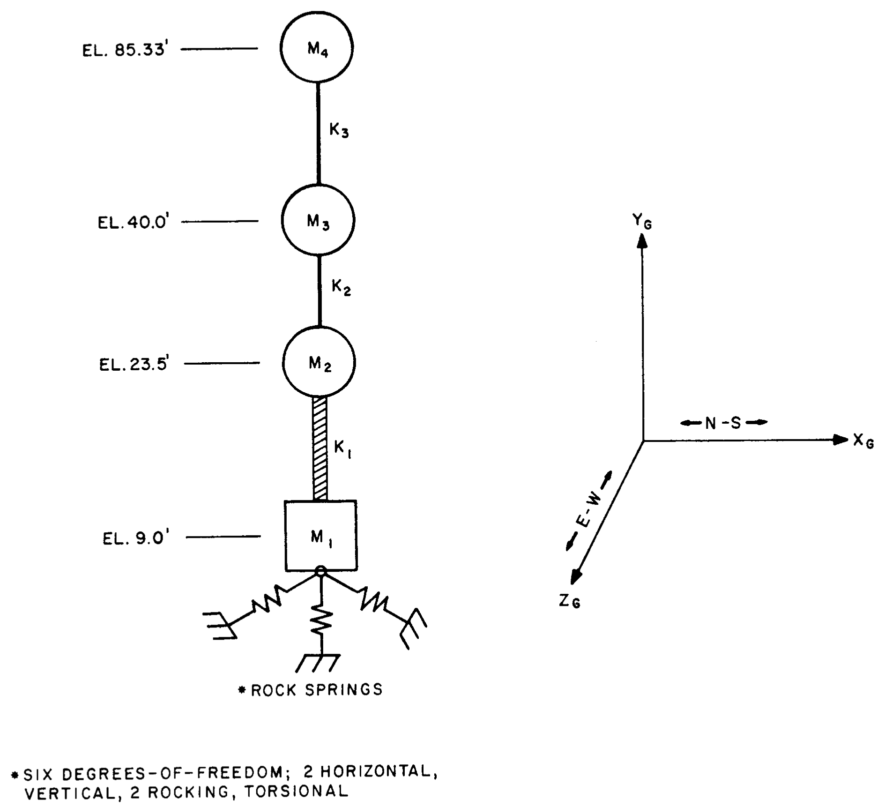


FIGURE 3.7B-10
DYNAMIC MODEL OF THE MAIN STEAM VALVE BUILDING
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–11 DYNAMIC MODEL OF THE EMERGENCY GENERATOR ENCLOSURE N-S VIEW

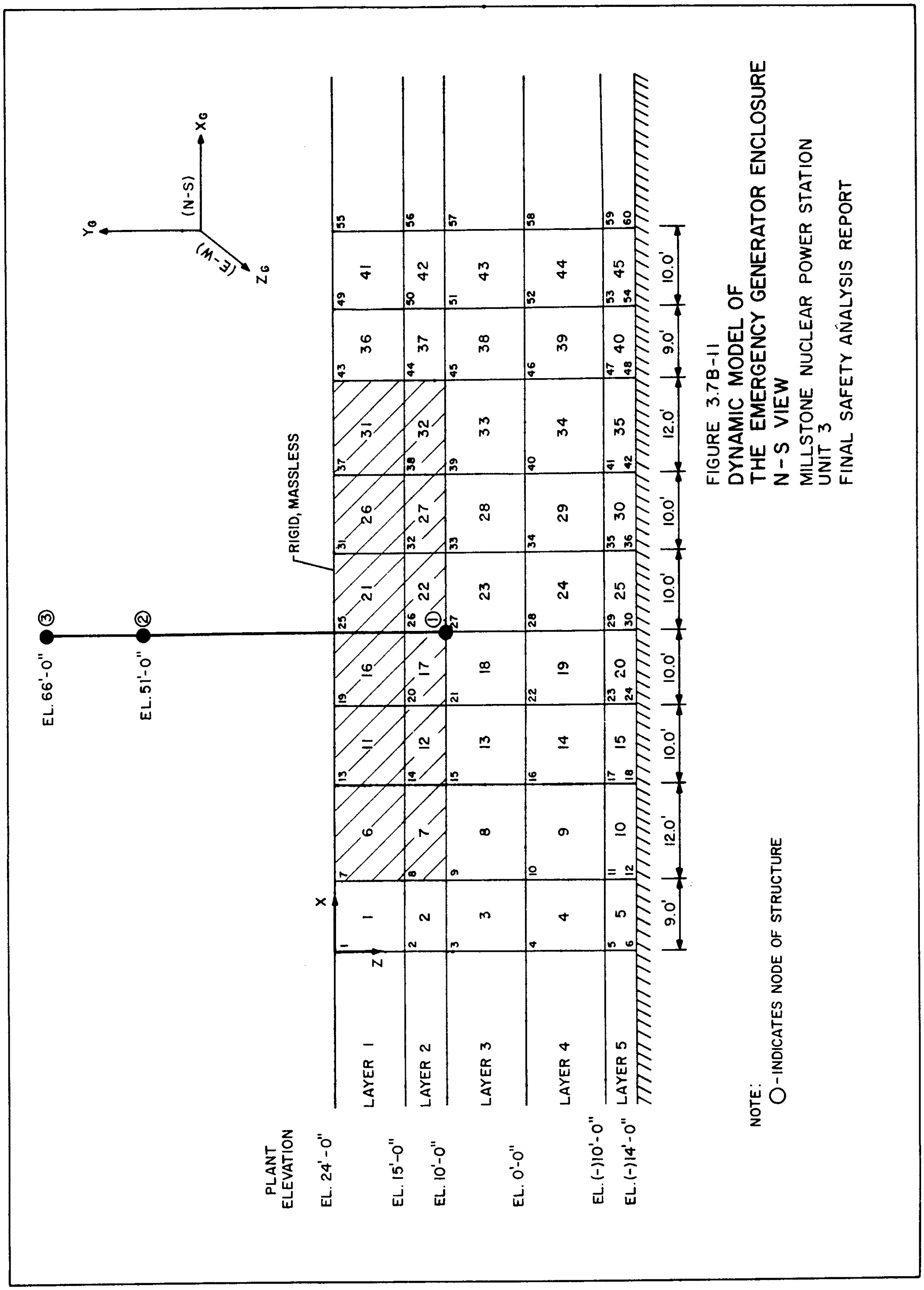


FIGURE 3.7B–12 DYNAMIC MODEL OF THE EMERGENCY GENERATOR ENCLOSURE E-W VIEW

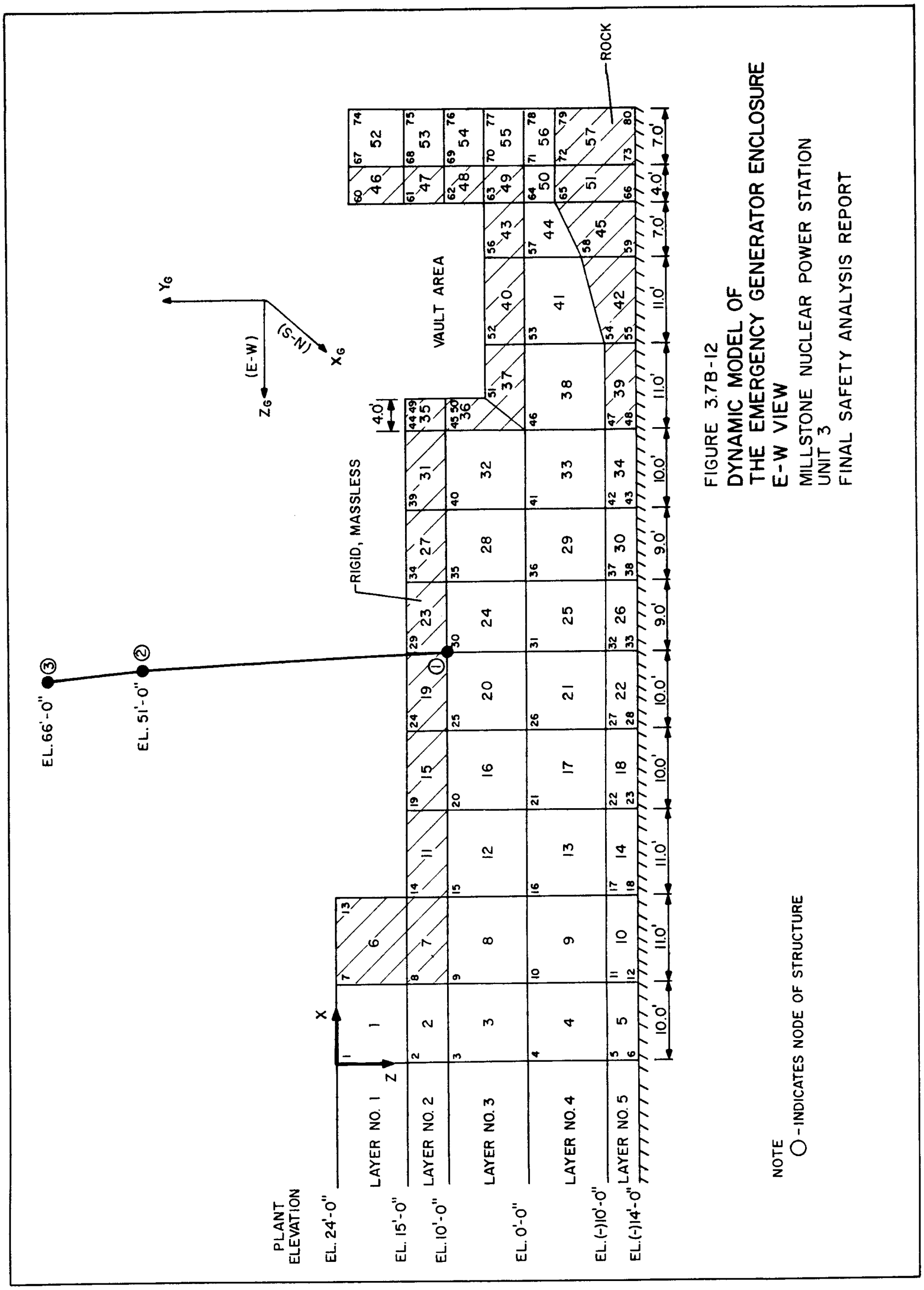


FIGURE 3.7B-12
DYNAMIC MODEL OF THE EMERGENCY GENERATOR ENCLOSURE E-W VIEW
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–13 REACTOR CONTAINMENT MAT ELEVATION -37 FEET 3 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE

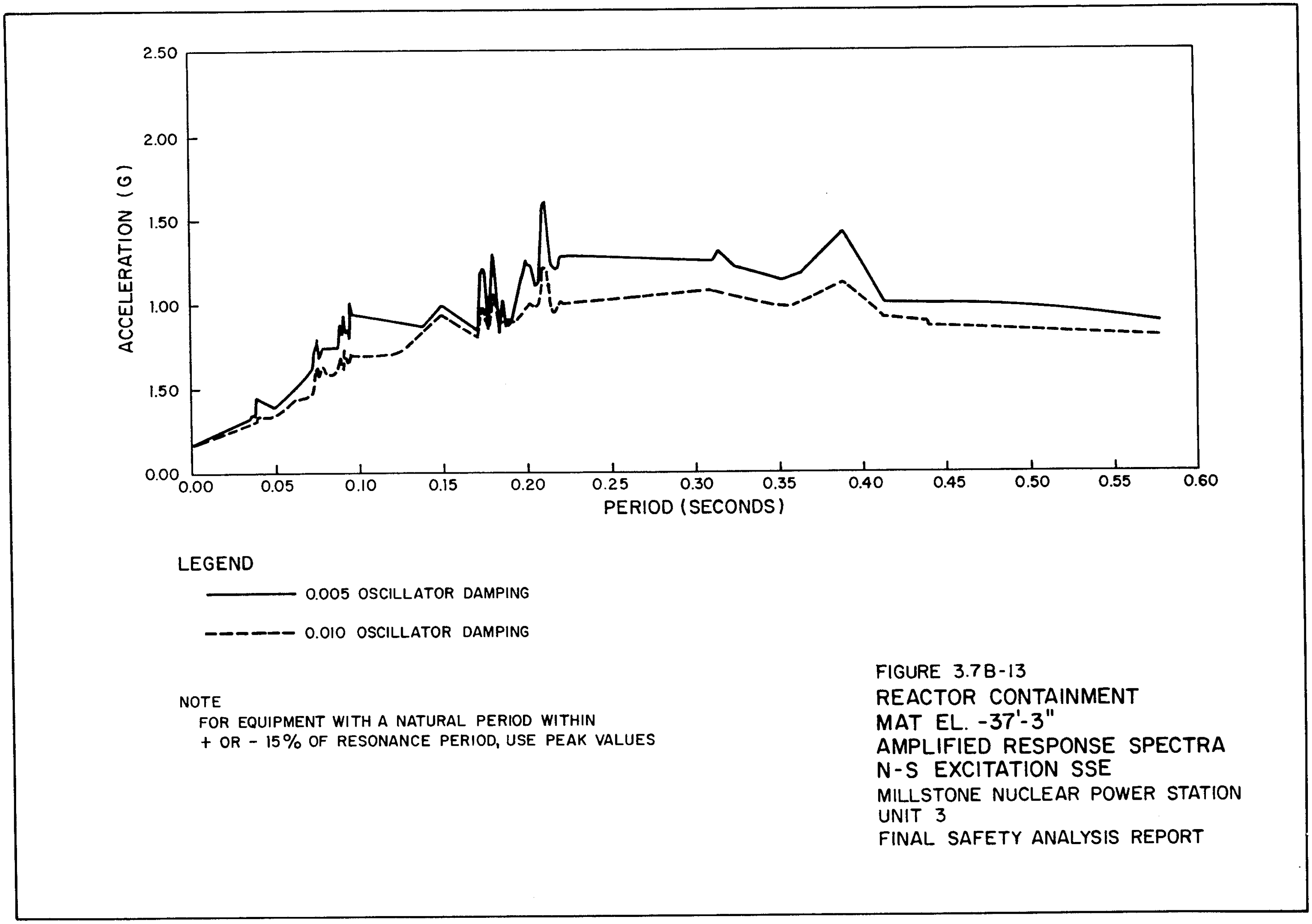
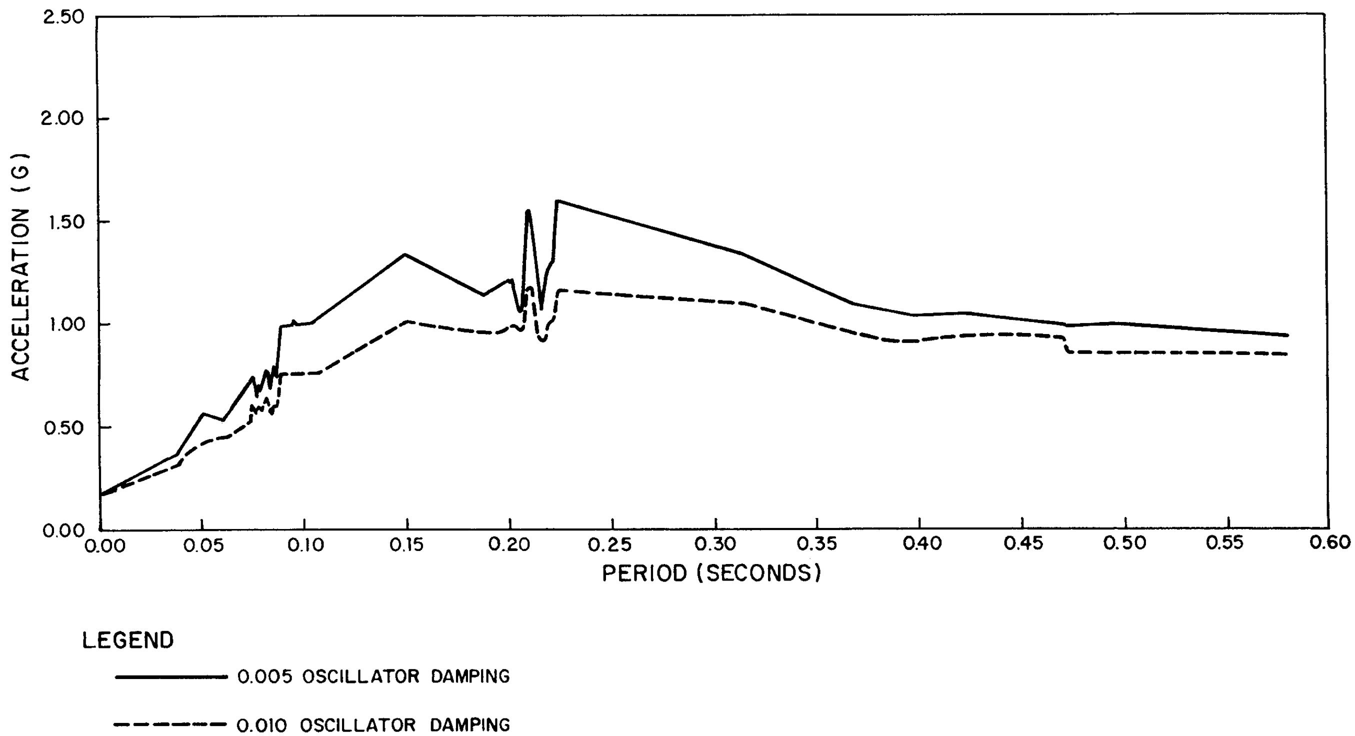


FIGURE 3.7B–14 REACTOR CONTAINMENT MAT ELEVATION -37 FEET 3 INCHES AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE



NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN
+ OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-14
REACTOR CONTAINMENT
MAT EL. -37'-3"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–15 REACTOR CONTAINMENT MAT ELEVATION -37 FEET 3 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE

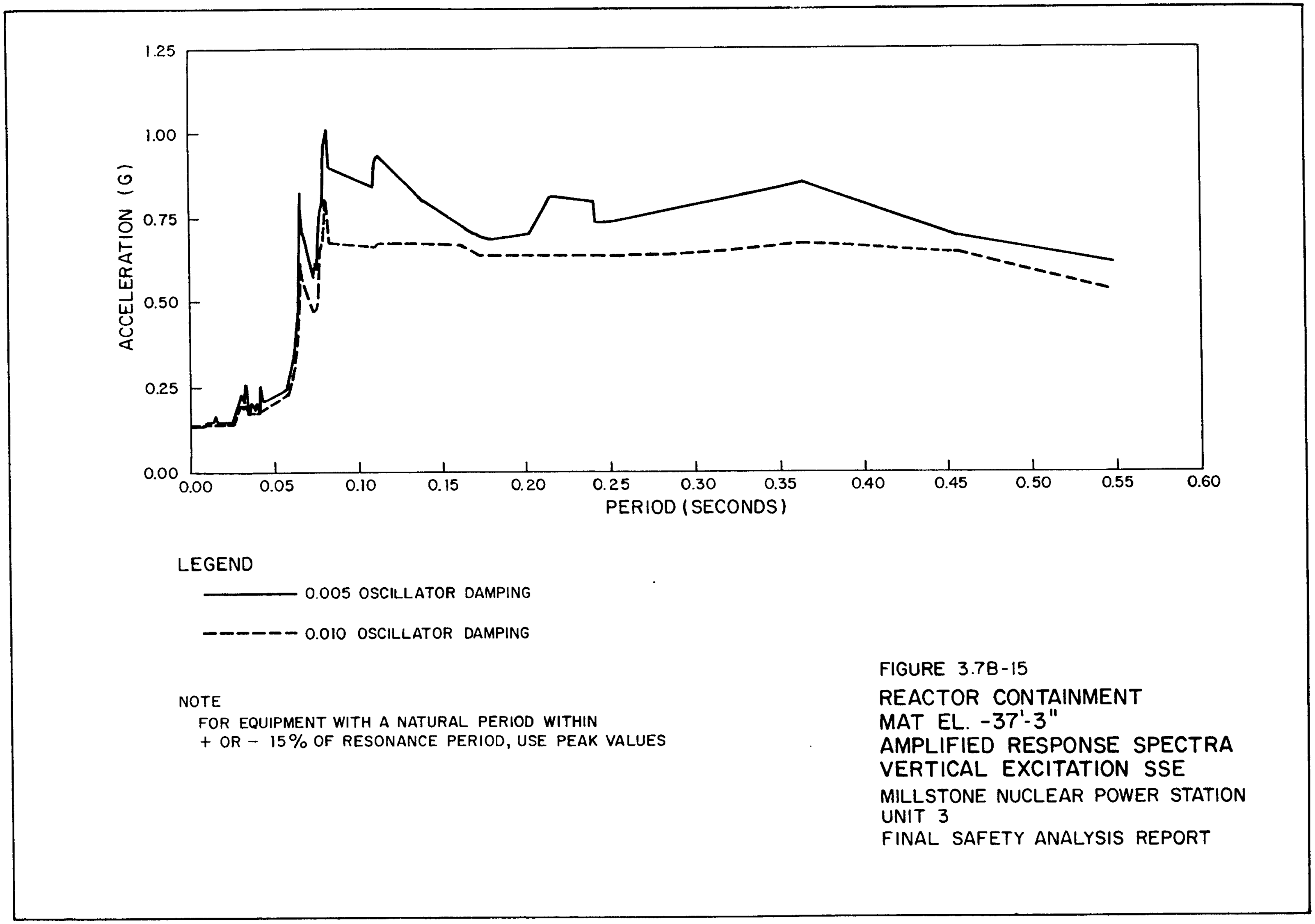
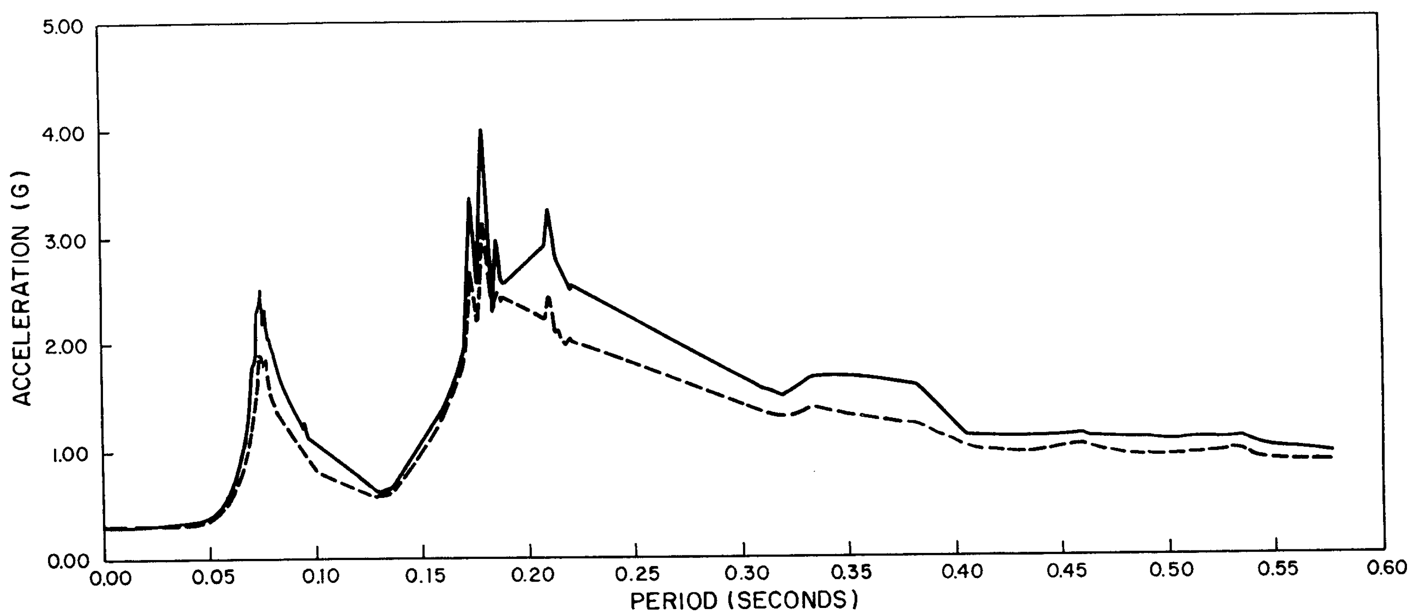


FIGURE 3.7B–16 REACTOR CONTAINMENT STEAM GENERATORS SUPPORT SLAB ELEVATION 3 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE



LEGEND

——— 0.005 OSCILLATOR DAMPING

- - - - 0.010 OSCILLATOR DAMPING

NOTE

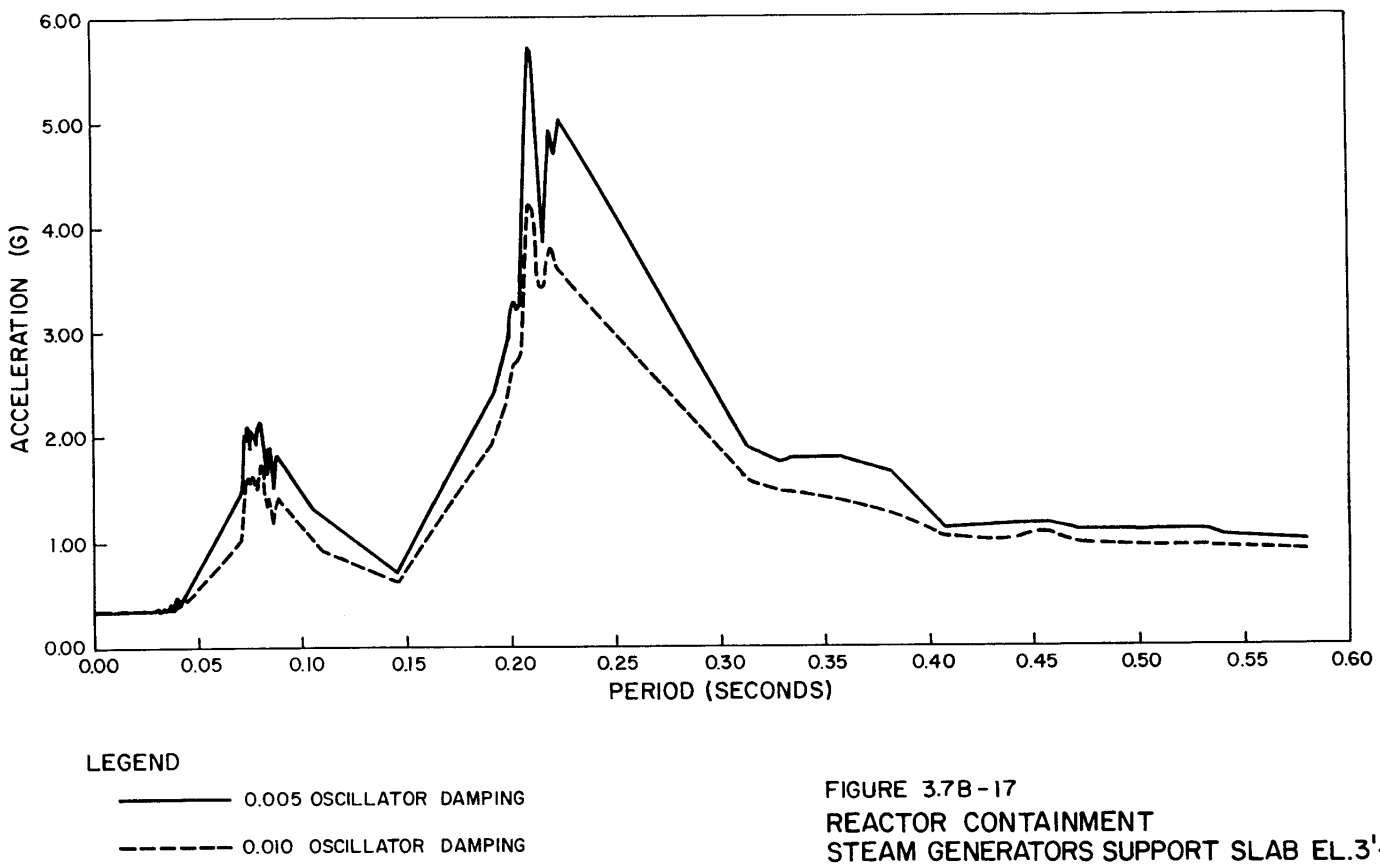
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR – 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-16

REACTOR CONTAINMENT
STEAM GENERATORS SUPPORT SLAB EL.3'-0"
AMPLIFIED RESPONSE SPECTRA
N-S EXCITATION SSE

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

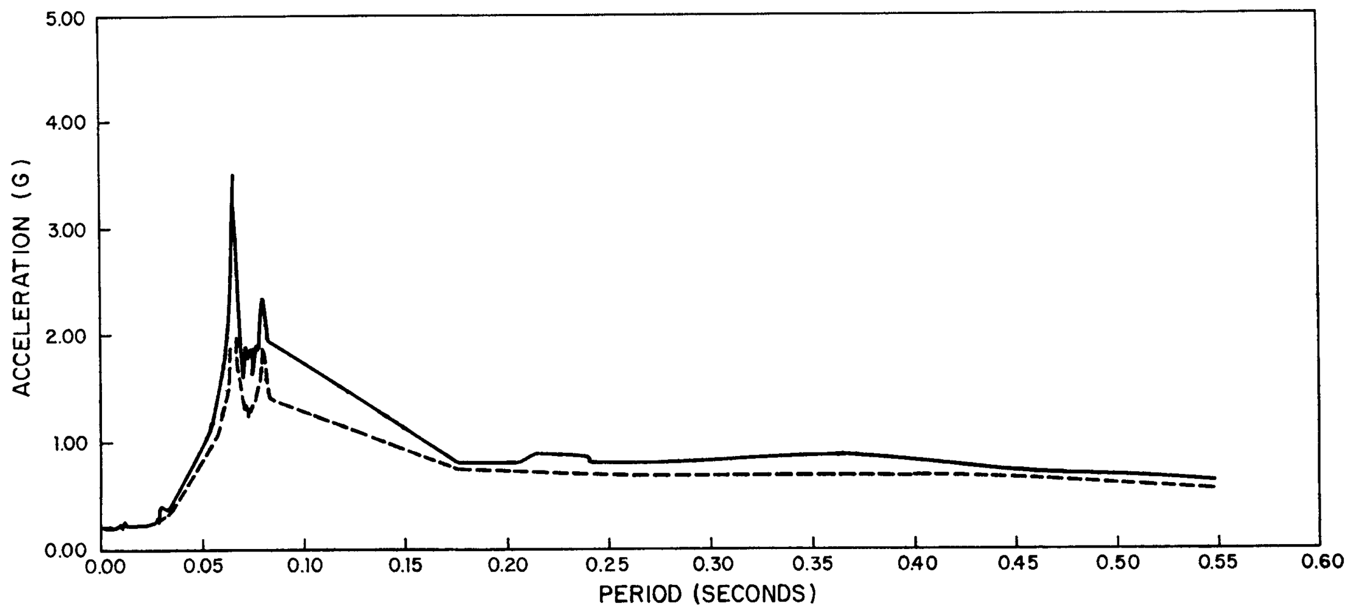
FIGURE 3.7B–17 REACTOR CONTAINMENT STEAM GENERATORS SUPPORT SLAB ELEVATION 3 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE



NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-17
REACTOR CONTAINMENT
STEAM GENERATORS SUPPORT SLAB EL.3'-0"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–18 REACTOR CONTAINMENT STEAM GENERATORS SUPPORT SLAB ELEVATION 3 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE



LEGEND

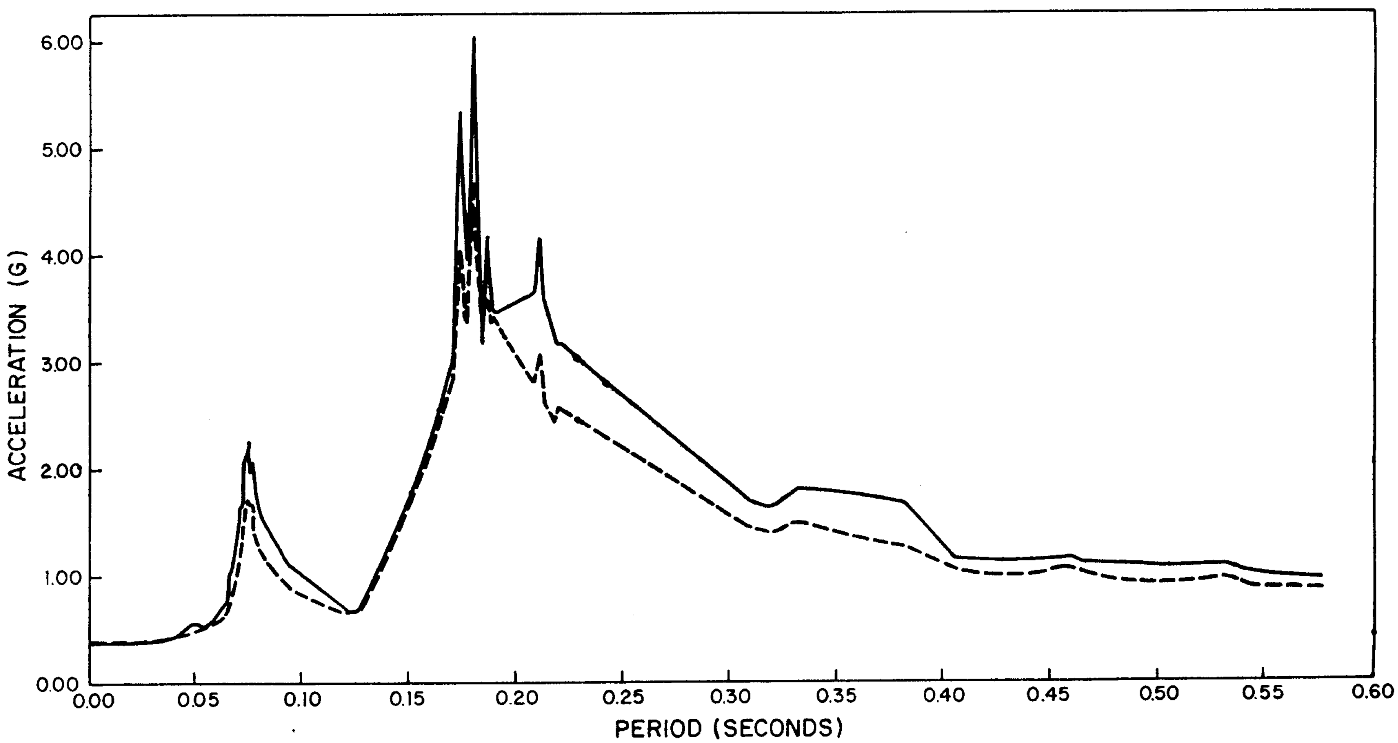
——— 0.005 OSCILLATOR DAMPING

- - - - - 0.010 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-18
REACTOR CONTAINMENT
STEAM GENERATORS SUPPORT SLAB EL.3'-0"
AMPLIFIED RESPONSE SPECTRA
VERTICAL EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–19 REACTOR CONTAINMENT TOP OF PRIMARY SHIELD WALL ELEVATION 24 FEET 6 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE



LEGEND

——— 0.005 OSCILLATOR DAMPING

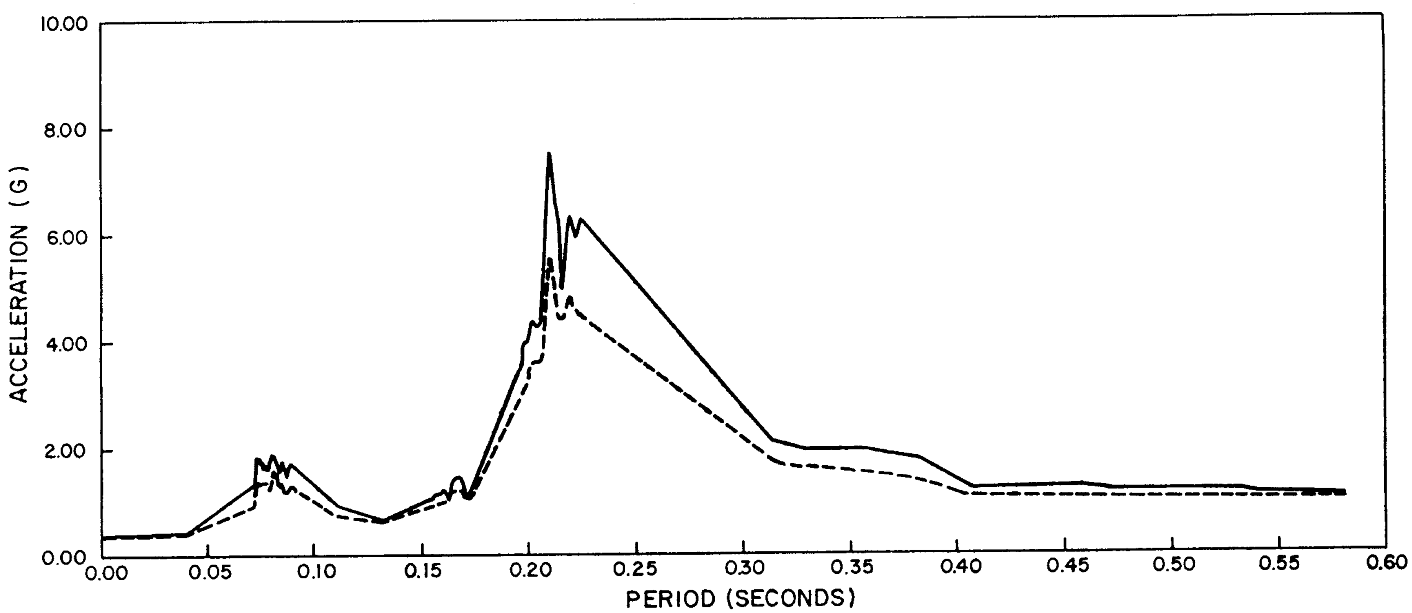
– – – – 0.010 OSCILLATOR DAMPING

NOTE

FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-19
REACTOR CONTAINMENT
TOP OF PRIMARY SHIELD WALL EL. 24'-6"
AMPLIFIED RESPONSE SPECTRA
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–20 REACTOR CONTAINMENT TOP OF PRIMARY SHIELD WALL EL 24 FEET 6 INCHES AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE



LEGEND

——— 0.005 OSCILLATOR DAMPING

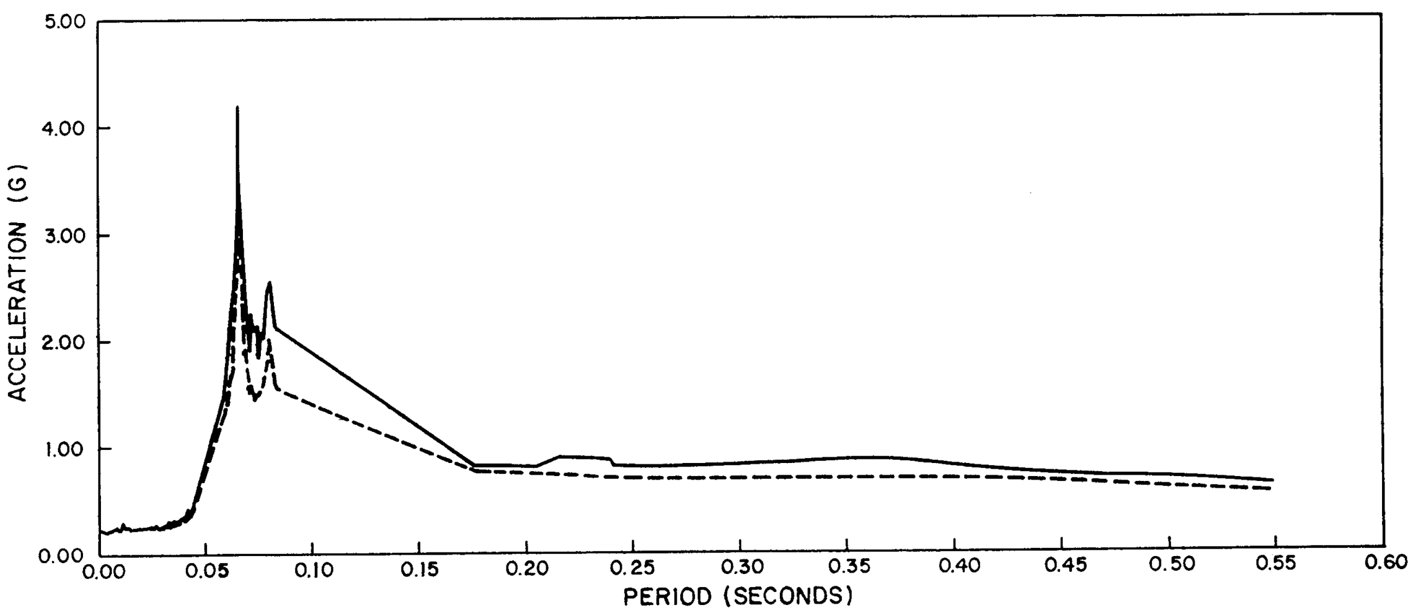
– – – – 0.010 OSCILLATOR DAMPING

NOTE

FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15 % OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-20
REACTOR CONTAINMENT
TOP OF PRIMARY SHIELD WALL EL. 24'-6"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–21 1 REACTOR CONTAINMENT TOP OF PRIMARY SHIELD WALL ELEVATION 24 FEET 6 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE



LEGEND

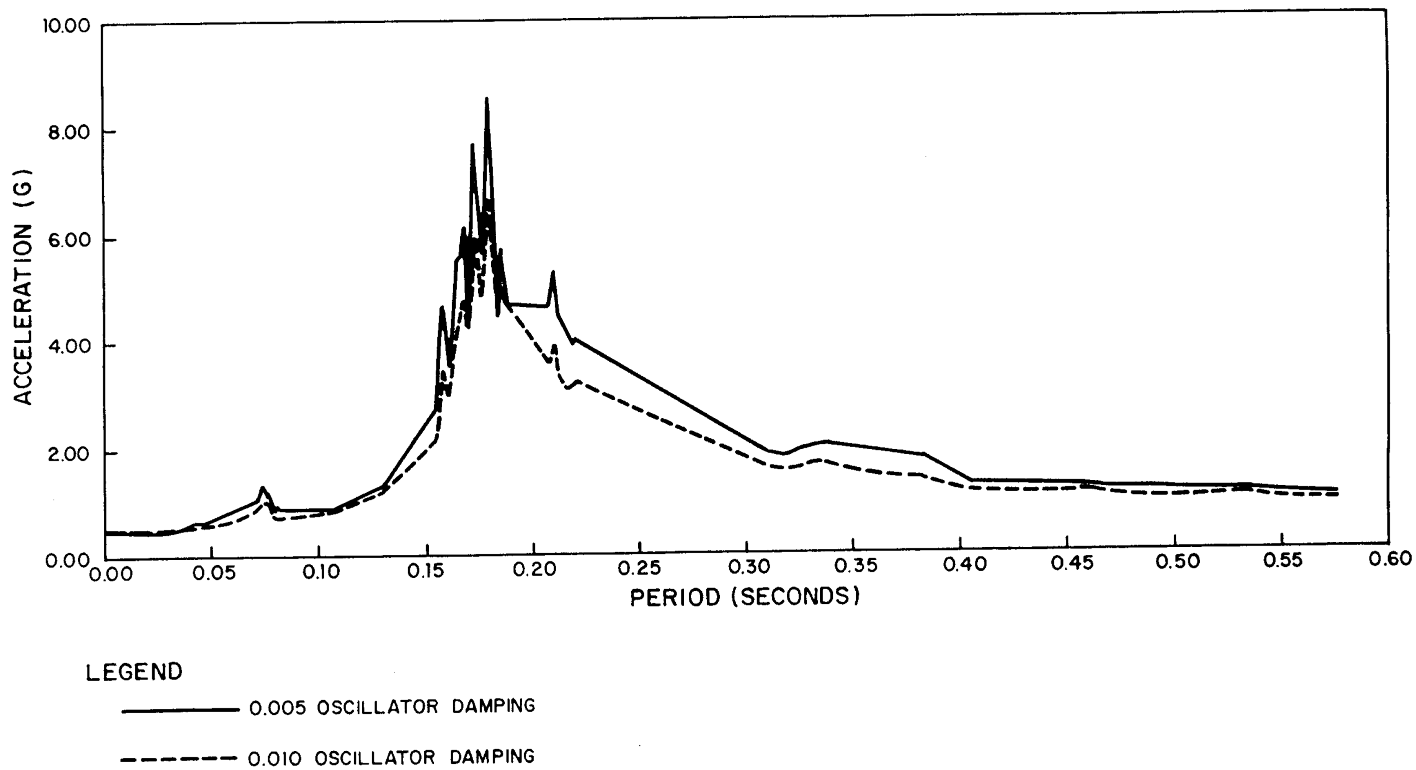
——— 0.005 OSCILLATOR DAMPING

- - - - 0.010 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN
+ OR – 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-21
REACTOR CONTAINTAINMENT
TOP OF PRIMARY SHIELD WALL EL. 24'-6"
AMPLIFIED RESPONSE SPECTRA
VERTICAL EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

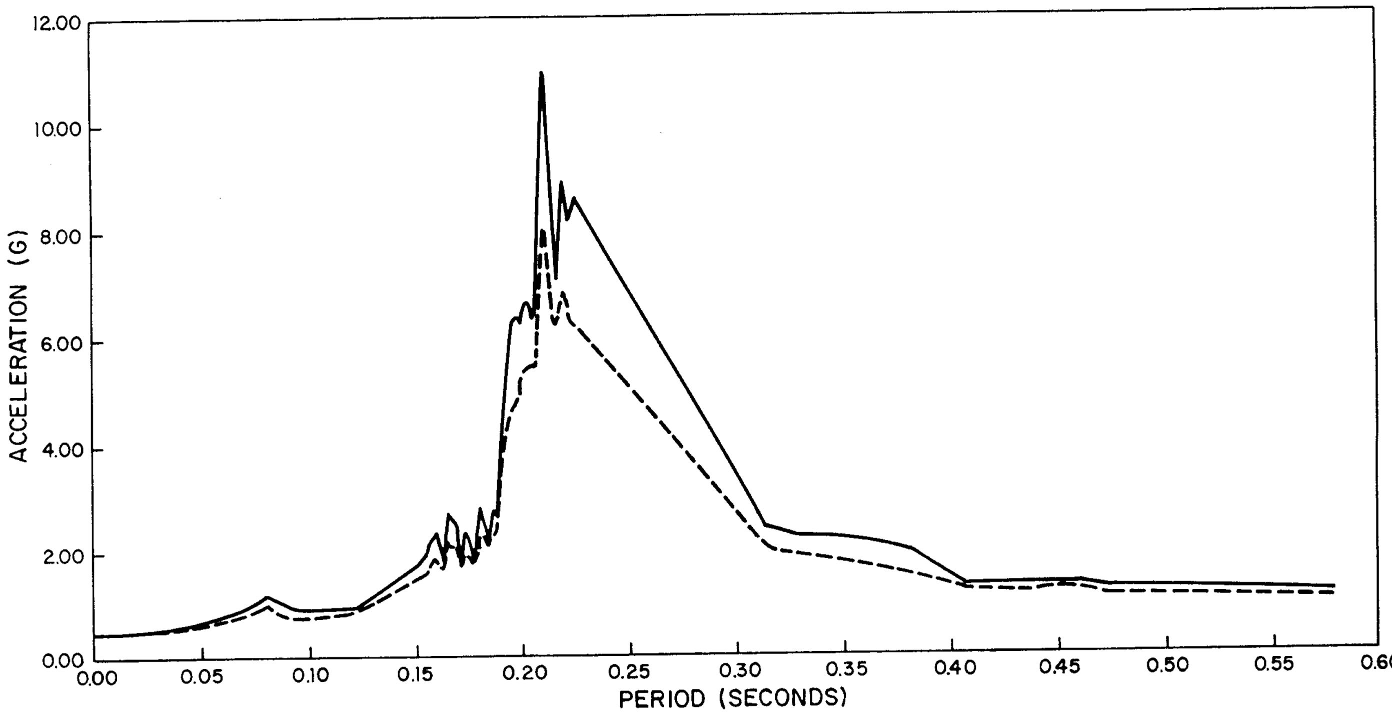
FIGURE 3.7B–22 REACTOR CONTAINMENT OPERATION FLOOR ELEVATION 50 FEET 10 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE



NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15 % OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-22
REACTOR CONTAINMENT
OPERATING FLOOR EL. 50'-10"
AMPLIFIED RESPONSE SPECTRA
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–23 REACTOR CONTAINMENT OPERATING FLOOR ELEVATION 50 FEET 10 INCHES AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE



LEGEND

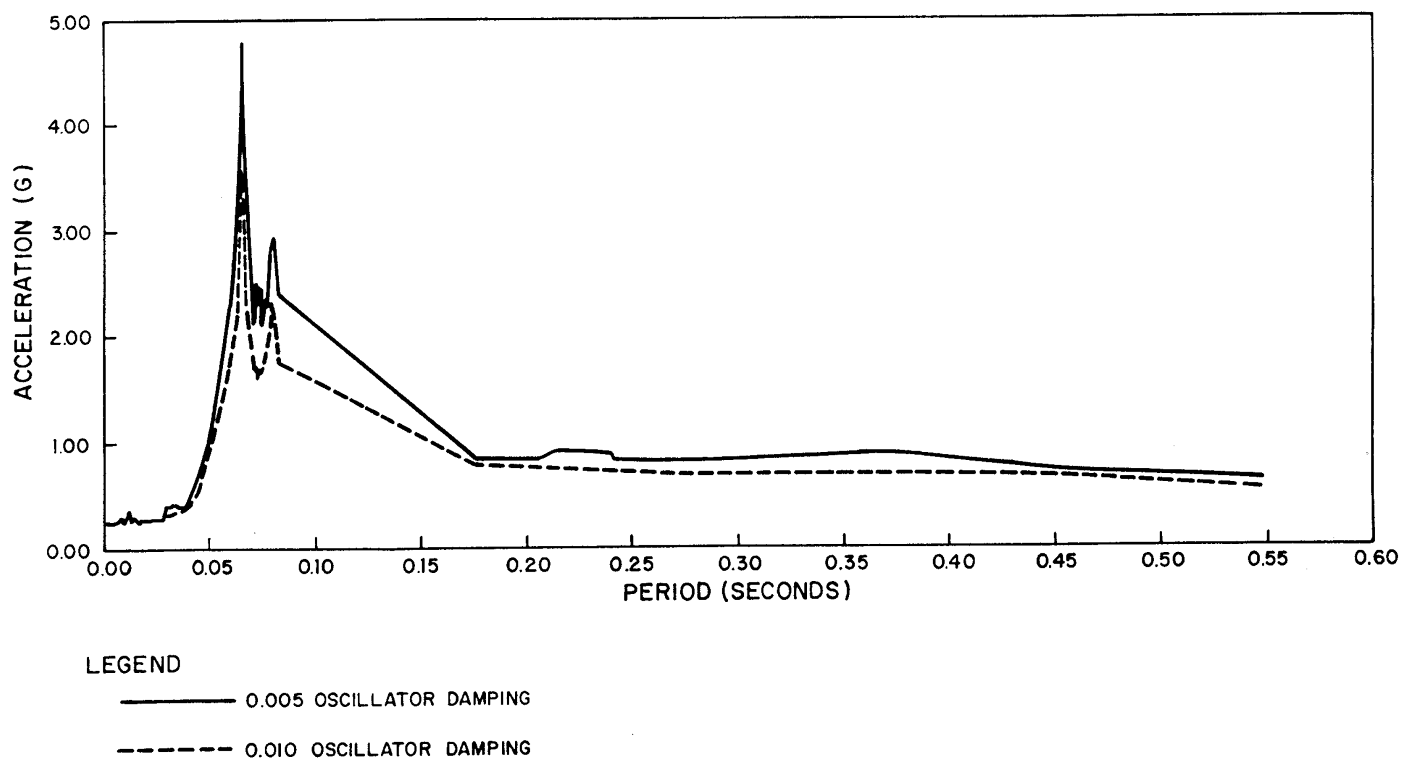
——— 0.005 OSCILLATOR DAMPING

– – – – – 0.010 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-23
REACTOR CONTAINMENT
OPERATING FLOOR EL. 50'-10"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

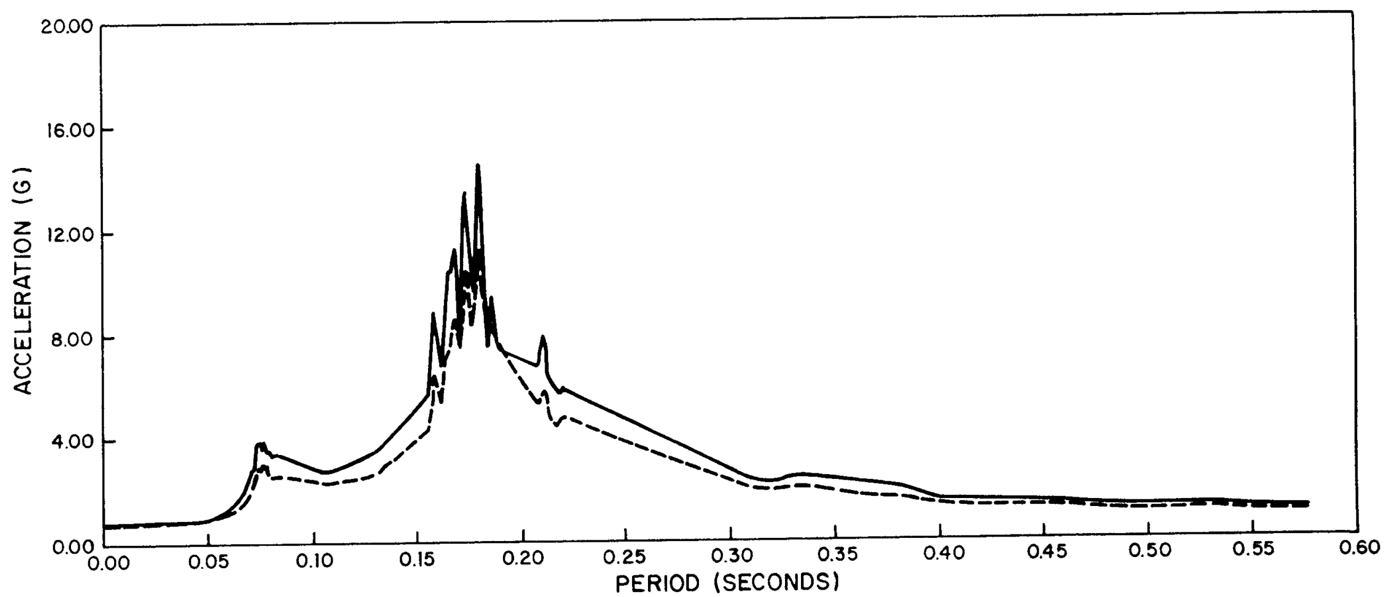
FIGURE 3.7B–24 REACTOR CONTAINMENT OPERATING FLOOR ELEVATION 50 FEET 10 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION



NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN
+ OR - 15 % OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-24
REACTOR CONTAINMENT
OPERATING FLOOR EL.50'-10"
AMPLIFIED RESPONSE SPECTRA
VERTICAL EXCITATION
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–25 REACTOR CONTAINMENT TOP OF CRANE WALL ELEVATION 109 FEET 1 INCH AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE



LEGEND

——— 0.005 OSCILLATOR DAMPING

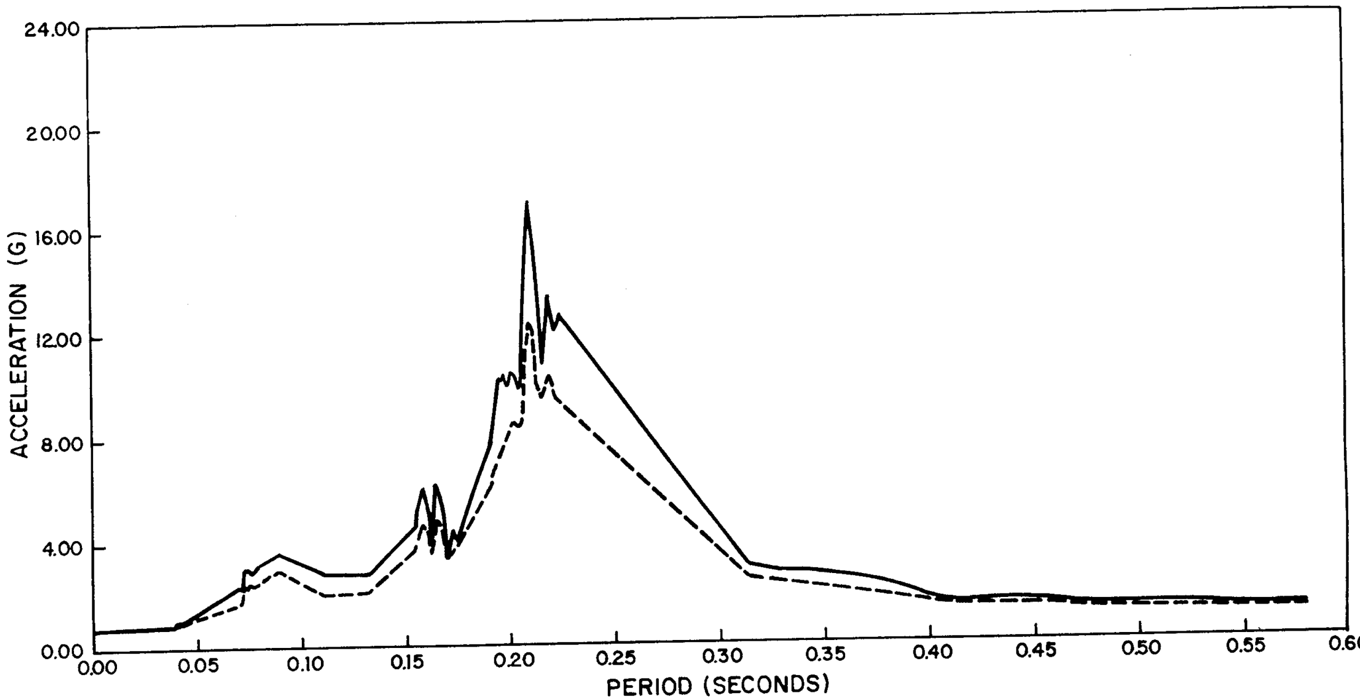
– – – – 0.010 OSCILLATOR DAMPING

NOTE

FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR -15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-25
REACTOR CONTAINMENT
TOP OF CRANE WALL EL. 109'-1"
AMPLIFIED RESPONSE SPECTRA
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–26 REACTOR CONTAINMENT TOP OF CRANE WALL ELEVATION 109 FEET 1 INCH AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE



LEGEND

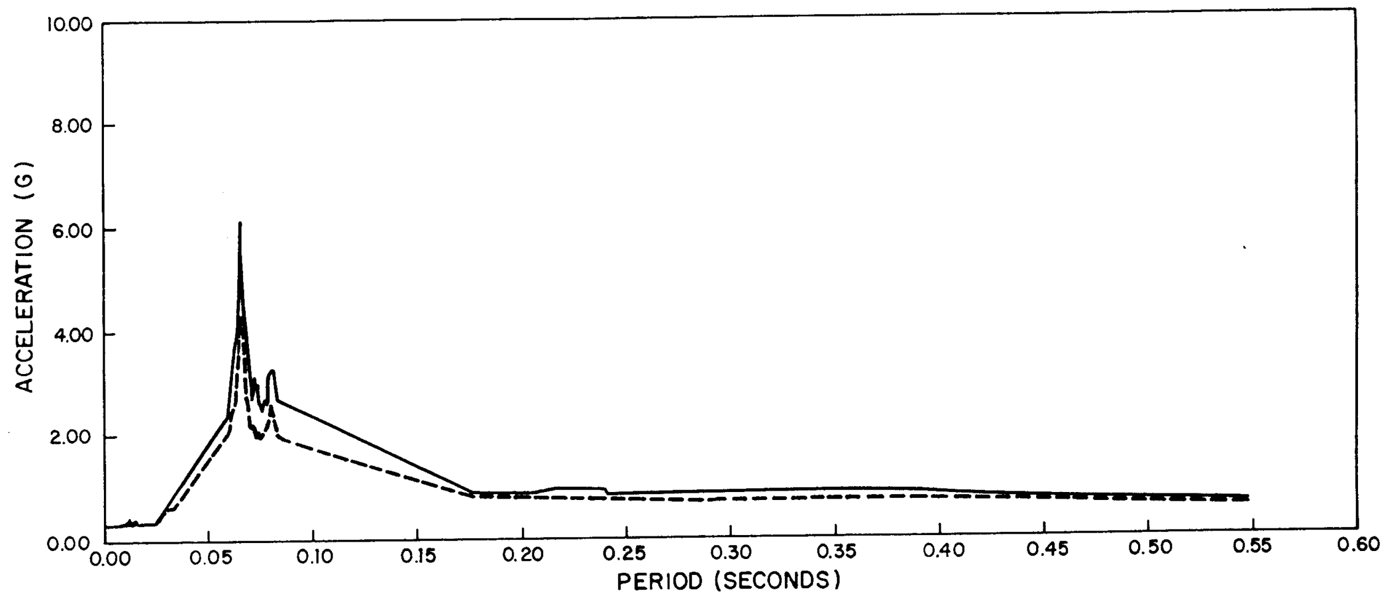
——— 0.005 OSCILLATOR DAMPING

– – – – 0.010 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN
+ OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE. 3.7B-26
REACTOR CONTAINMENT
TOP OF CRANE WALL EL.109'-1"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–27 REACTOR CONTAINMENT TOP OF CRANE WALL ELEVATION 109 FEET 1 INCH AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE



LEGEND

——— 0.005 OSCILLATOR DAMPING

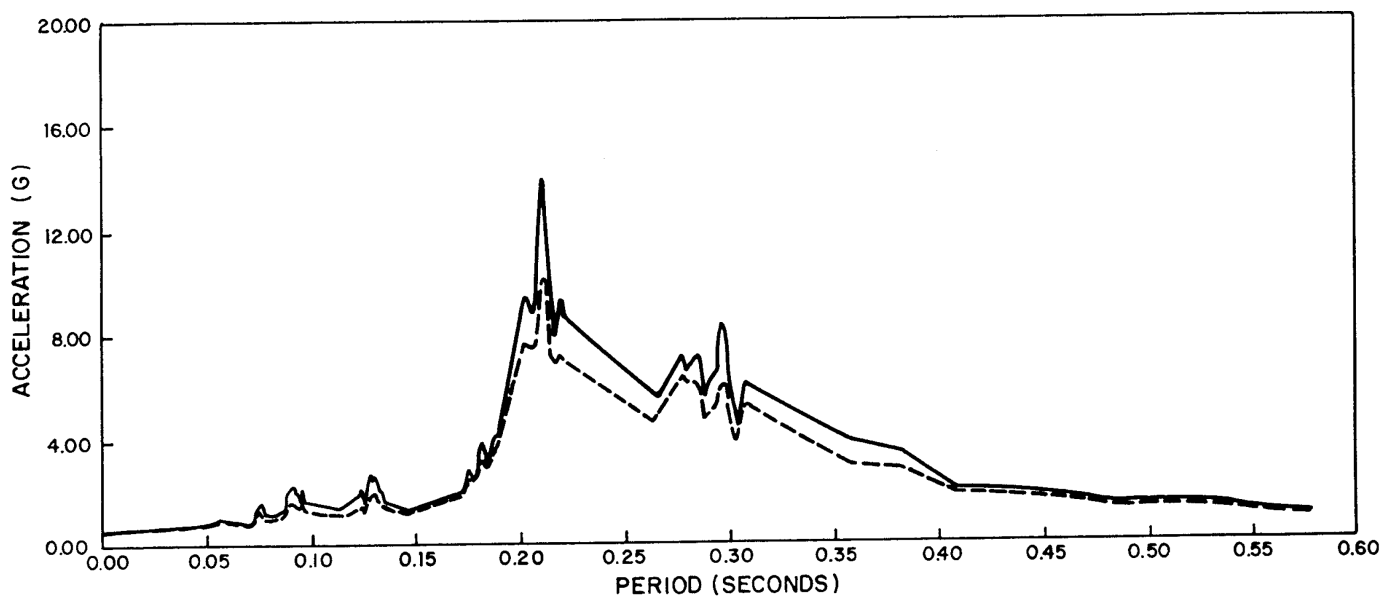
- - - - - 0.010 OSCILLATOR DAMPING

NOTE

FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-27
REACTOR CONTAINMENT
TOP OF CRANE WALL EL. 109'-1"
AMPLIFIED RESPONSE SPECTRA
VERTICAL EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–28 REACTOR CONTAINMENT SPRINGLINE ELEVATION 104 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE



LEGEND

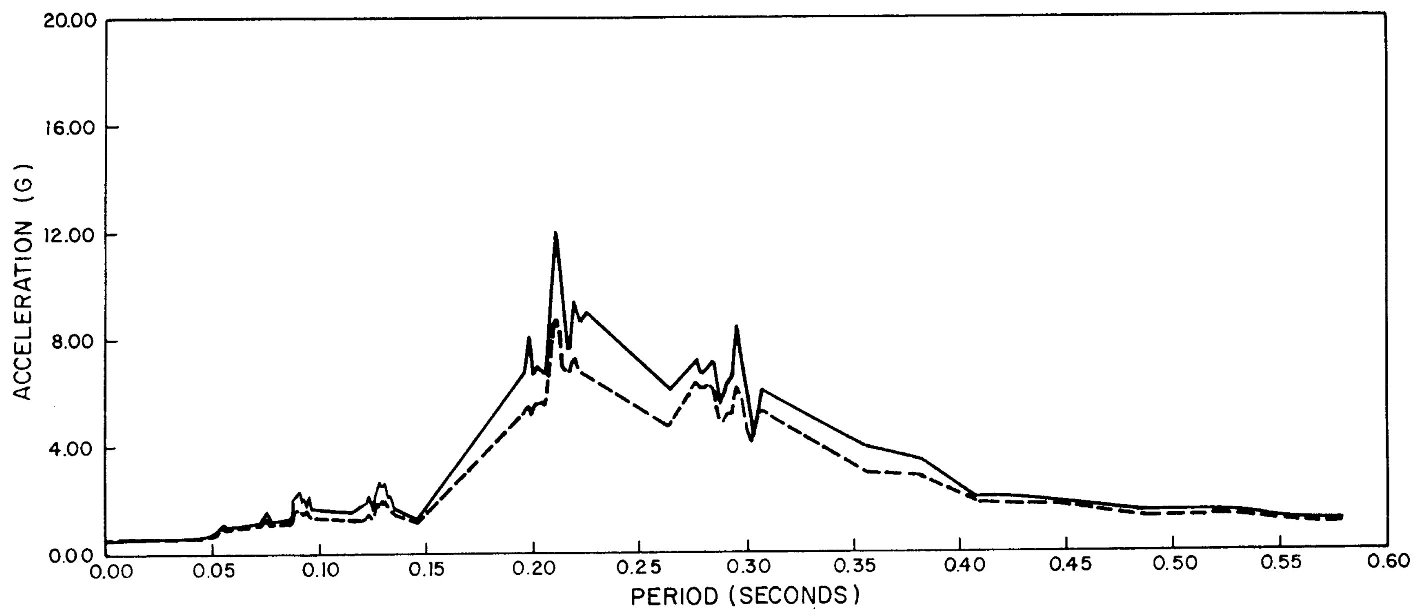
——— 0.005 OSCILLATOR DAMPING

– – – – 0.010 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-28
REACTOR CONTAINMENT SPRINGLINE EL. 104'-0"
AMPLIFIED RESPONSE SPECTRA
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–29 REACTOR CONTAINMENT SPRINGLINE ELEVATION 104 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE



LEGEND

——— 0.005 OSCILLATOR DAMPING

- - - - 0.010 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN
+ OR – 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-29
REACTOR CONTAINMENT
SPRINGLINE EL. 104'-0"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–30 REACTOR CONTAINMENT SPRINGLINE ELEVATION 104 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION

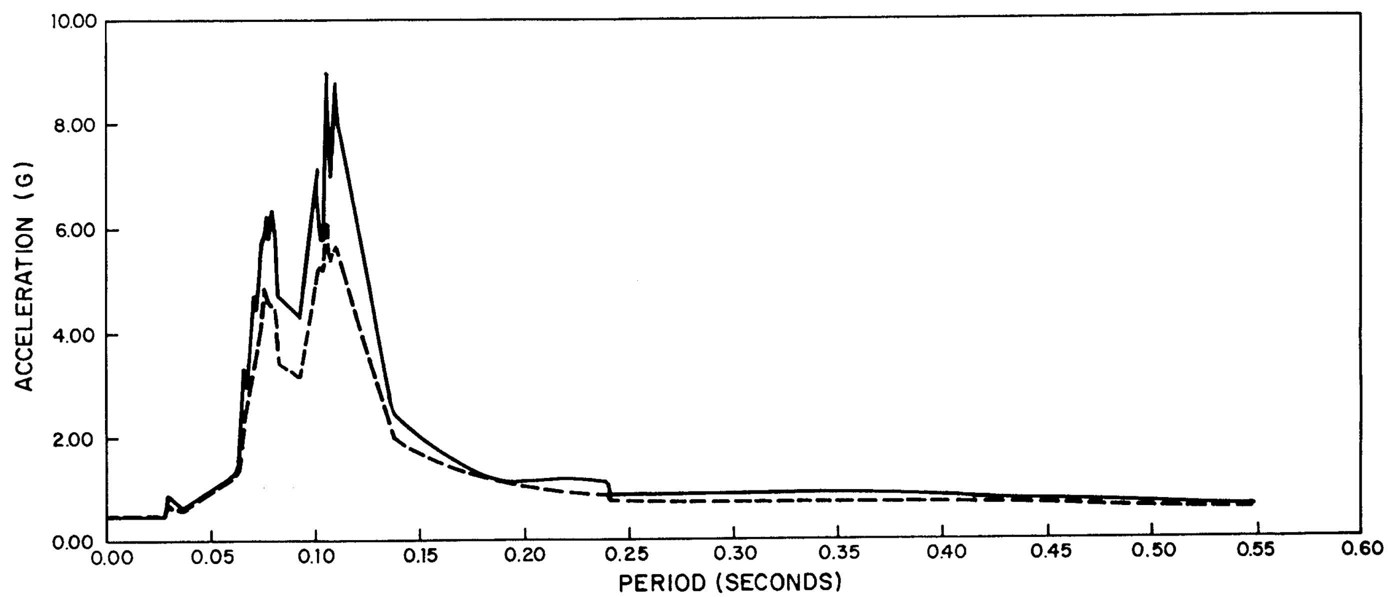
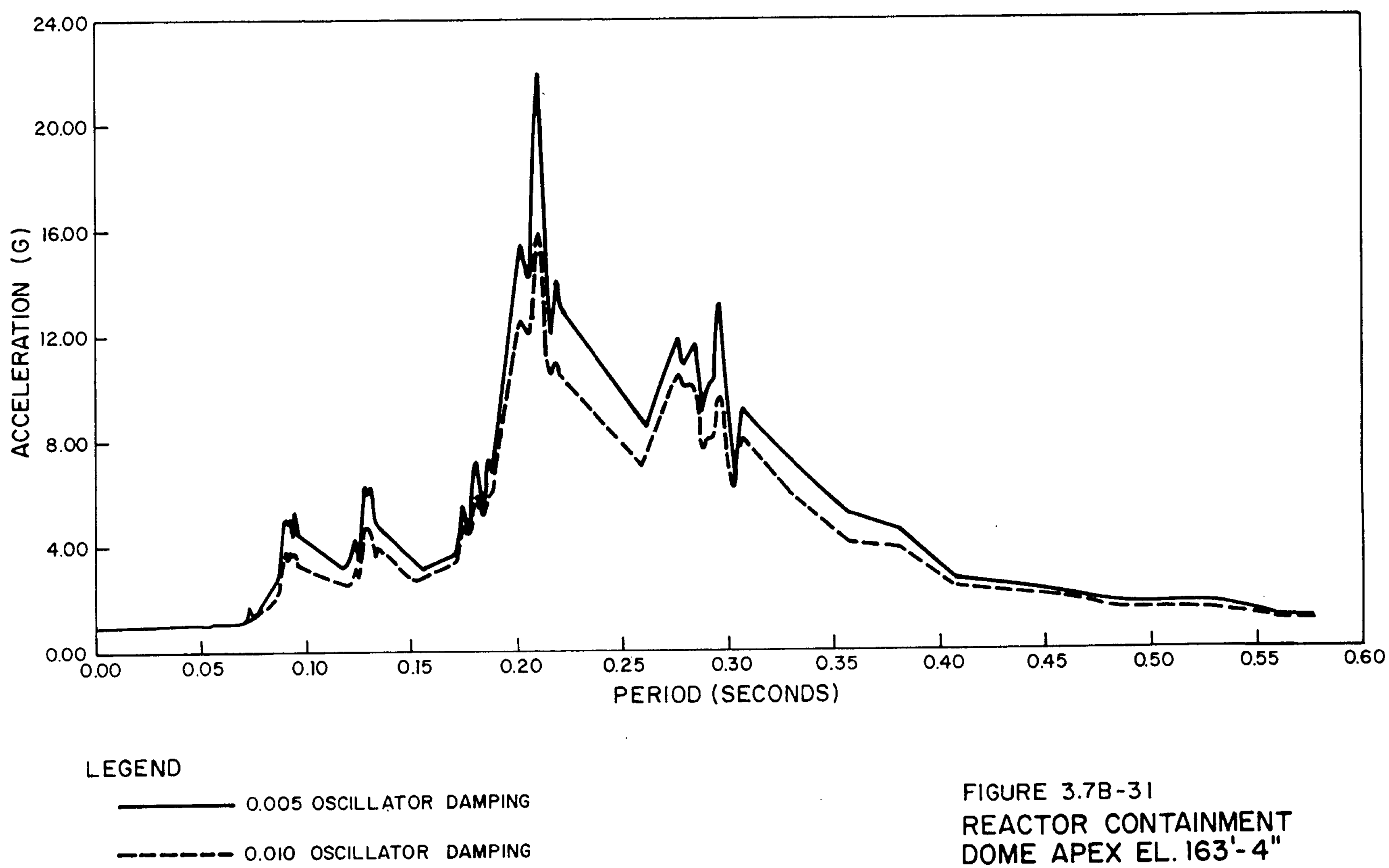


FIGURE 3.7B-30
REACTOR CONTAINMENT
SPRINGLINE EL. 104'-0"
AMPLIFIED RESPONSE SPECTRA
VERTICAL EXCITATION
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

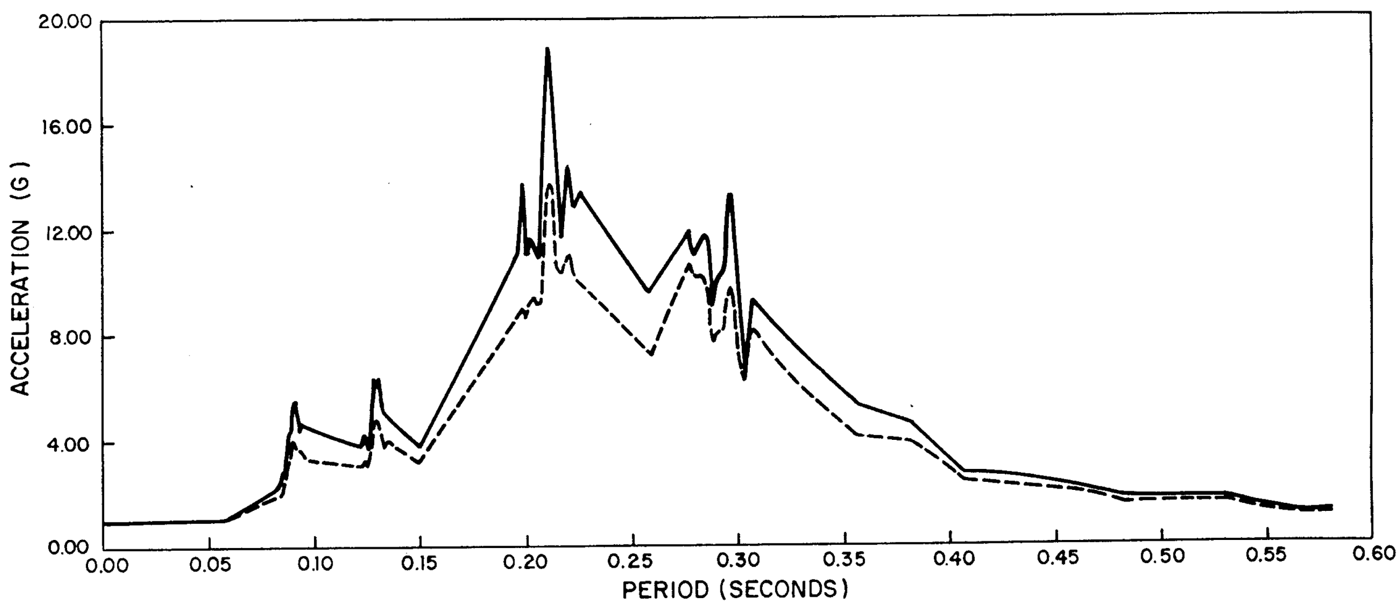
FIGURE 3.7B–31 REACTOR CONTAINMENT DOME APEX ELEVATION 163 FEET 4 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE



NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR – 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-31
REACTOR CONTAINMENT
DOME APEX EL. 163'-4"
AMPLIFIED RESPONSE SPECTRA
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–32 REACTOR CONTAINMENT DOME APEX ELEVATION 163 FEET 4 INCHES AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE



LEGEND

——— 0.005 OSCILLATOR DAMPING

- - - - 0.010 OSCILLATOR DAMPING

NOTE

FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-32
REACTOR CONTAINMENT
DOME APEX EL. 163'-4"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–33 REACTOR CONTAINMENT DOME APEX ELEVATION 163 FEET 4 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE

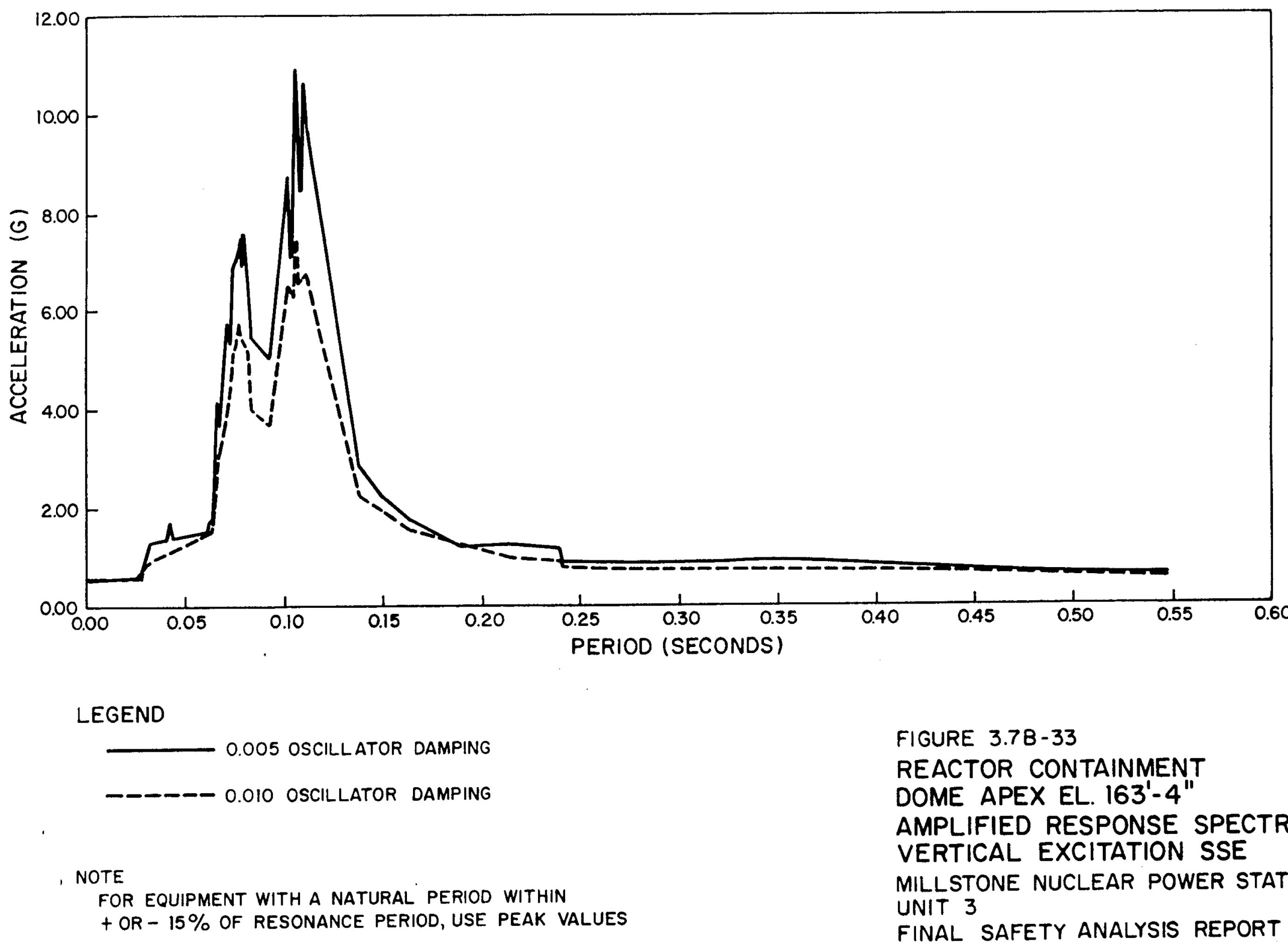


FIGURE 3.7B-33
REACTOR CONTAINMENT
DOME APEX EL. 163'-4"
AMPLIFIED RESPONSE SPECTRA
VERTICAL EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–34 MAIN STEAM VALVE BUILDING MAT ELEVATION 9 FEET -0 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE

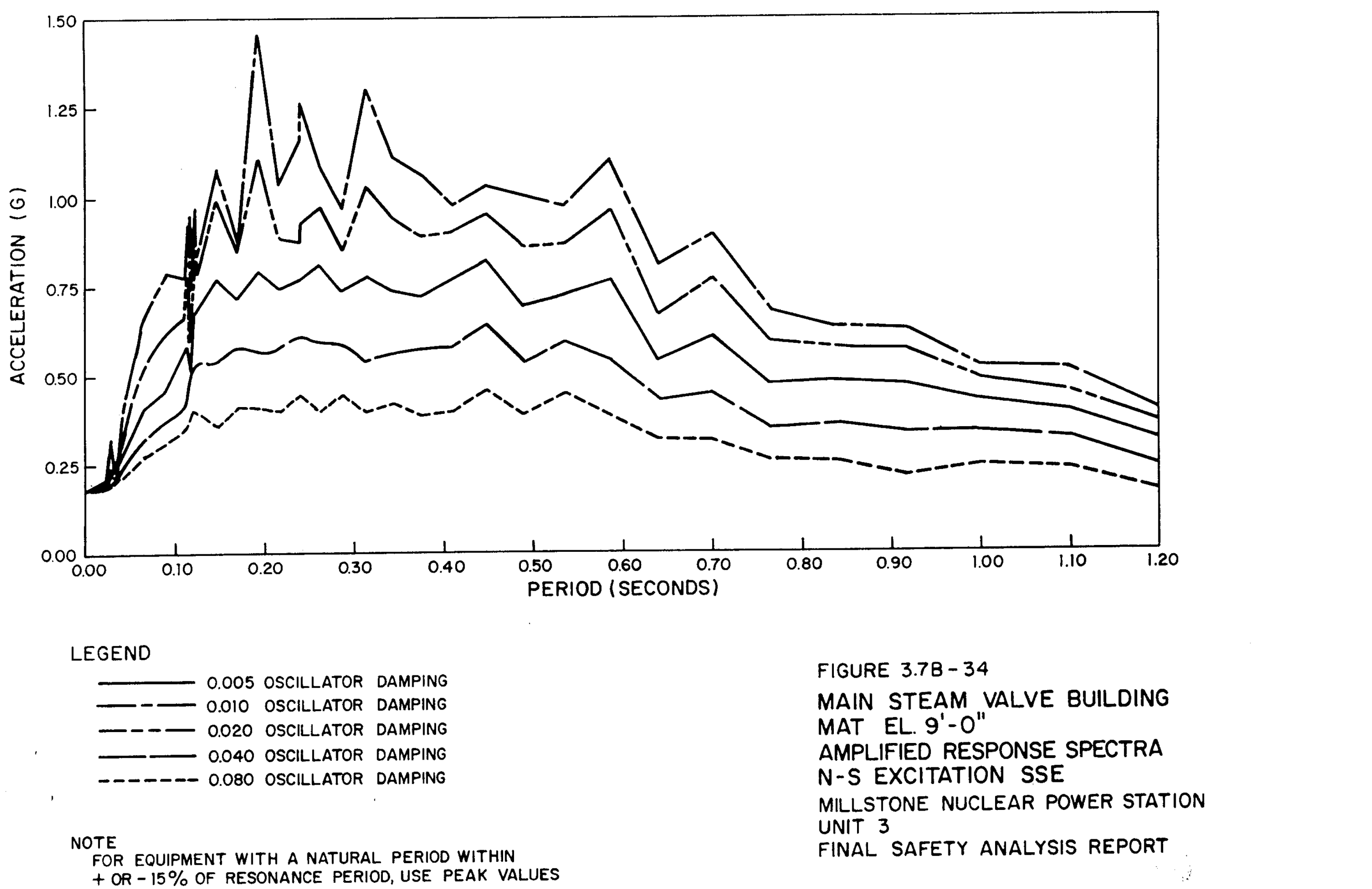


FIGURE 3.7B–35 MAIN STEAM VALVE BUILDING MAT ELEVATION 9 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE

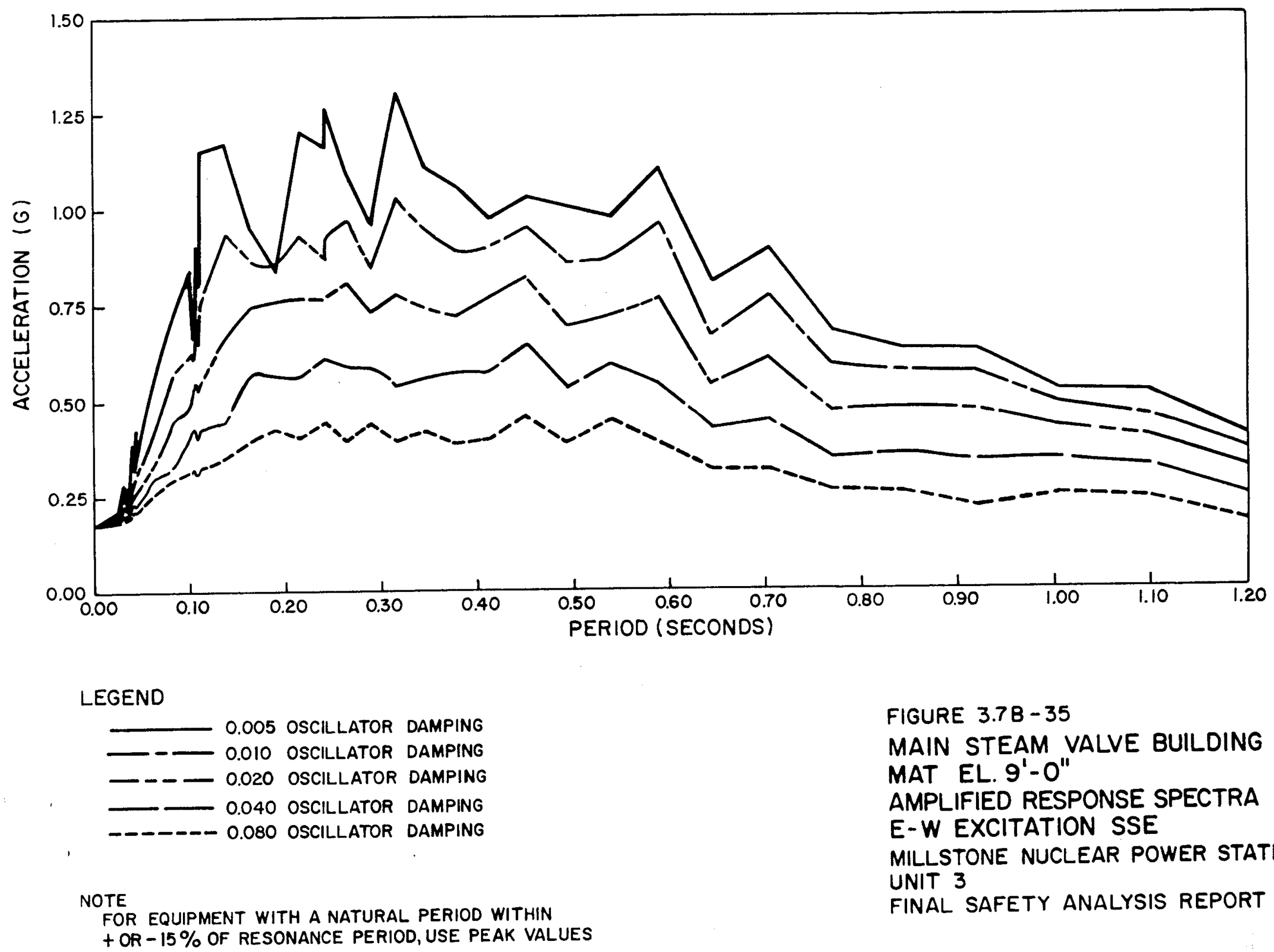
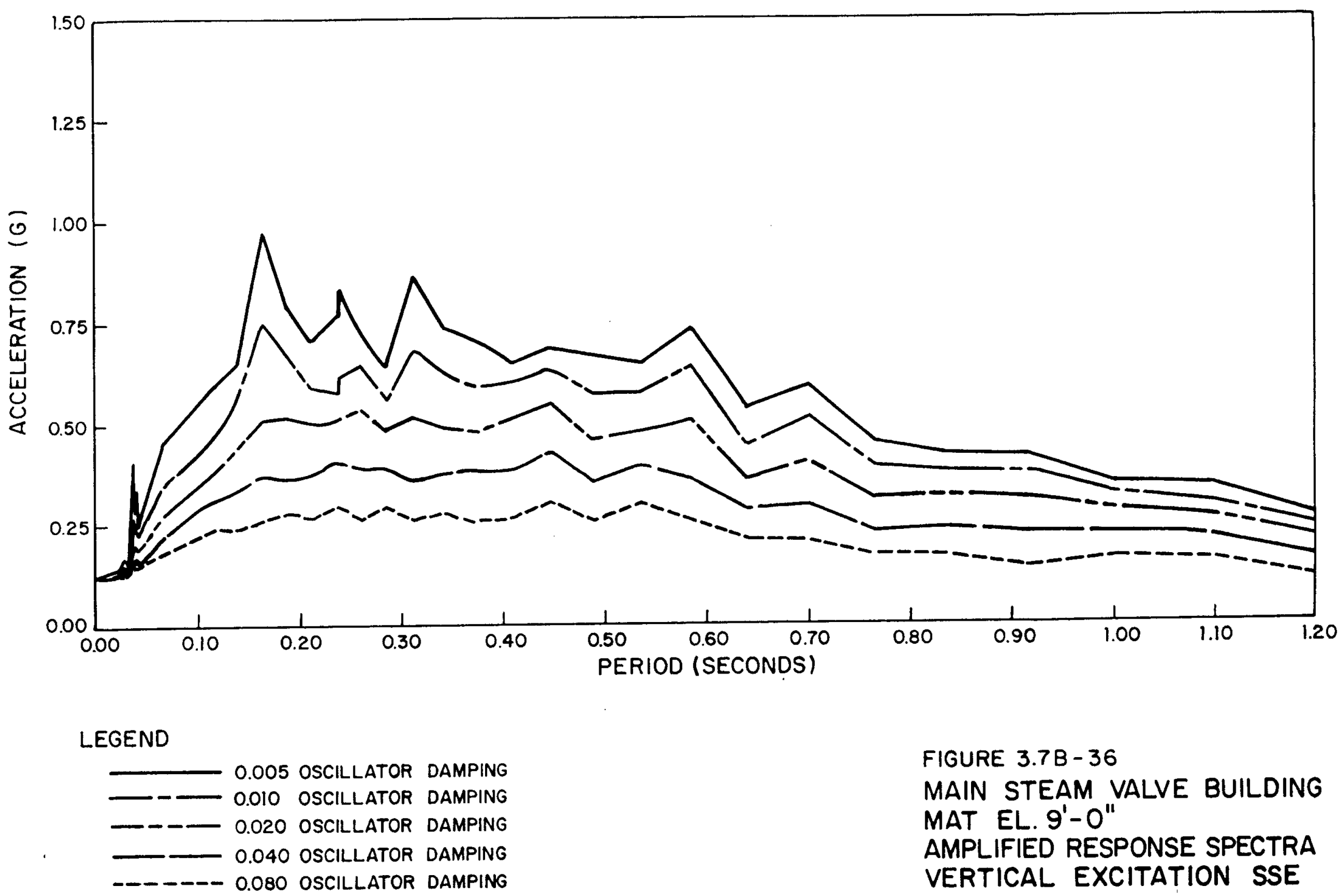


FIGURE 3.7B-35
MAIN STEAM VALVE BUILDING
MAT EL. 9'-0"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

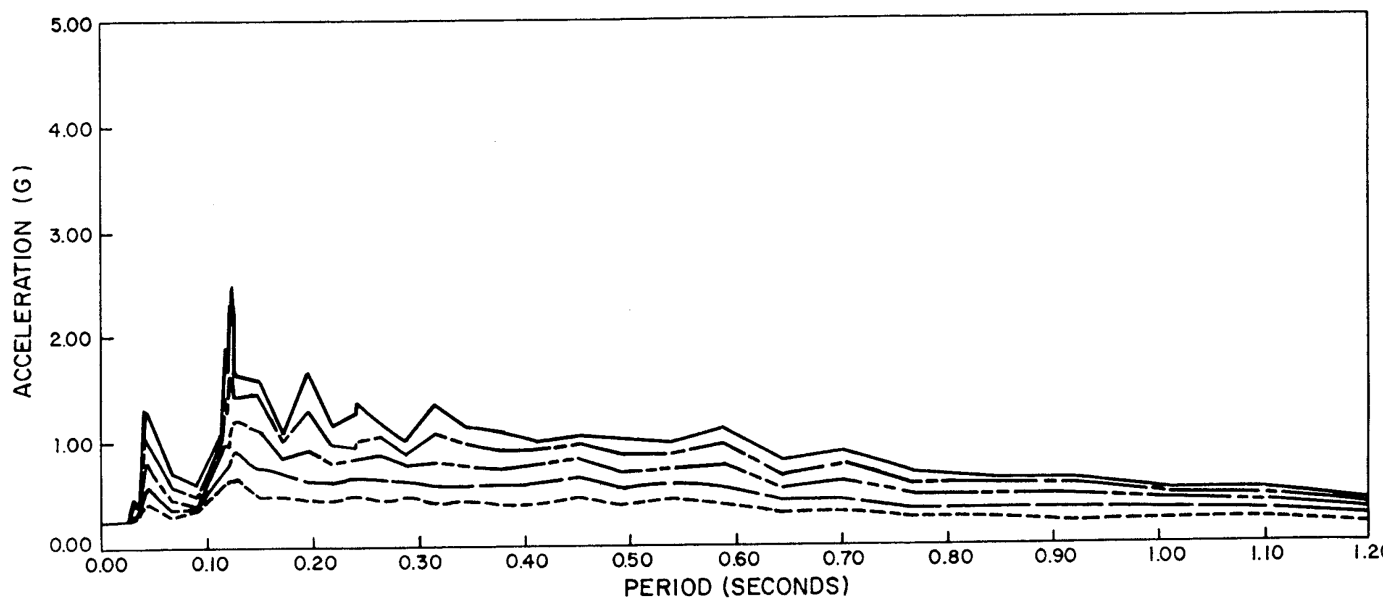
FIGURE 3.7B–36 MAIN STEAM VALVE BUILDING MAT ELEVATION 9 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE



NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN
+ OR – 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-36
MAIN STEAM VALVE BUILDING
MAT EL. 9'-0"
AMPLIFIED RESPONSE SPECTRA
VERTICAL EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–37 MAIN STEAM VALVE BUILDING FLOOR SLAB ELEVATION 41 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE



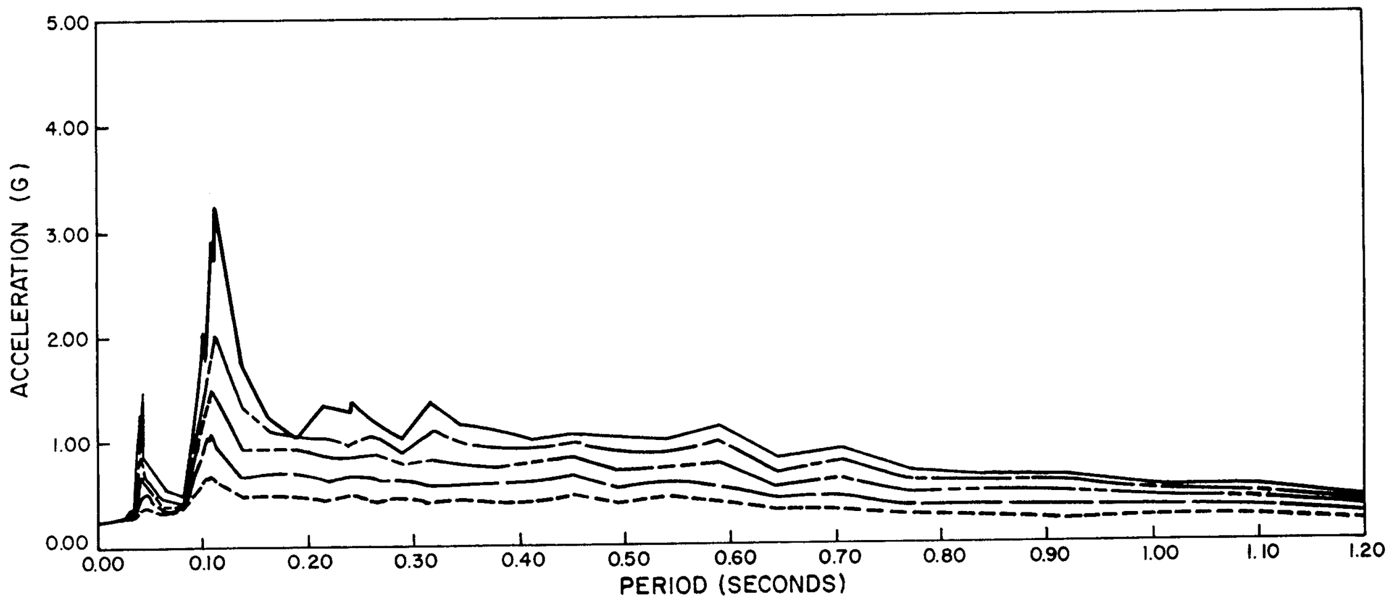
LEGEND

- 0.005 OSCILLATOR DAMPING
- 0.010 OSCILLATOR DAMPING
- 0.020 OSCILLATOR DAMPING
- 0.040 OSCILLATOR DAMPING
- 0.080 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B - 37
MAIN STEAM VALVE BUILDING
FLOOR SLAB EL.41'-0"
AMPLIFIED RESPONSE SPECTRA
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–38 MAIN STEM VALVE BUILDING FLOOR SLAB ELEVATION 41 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE



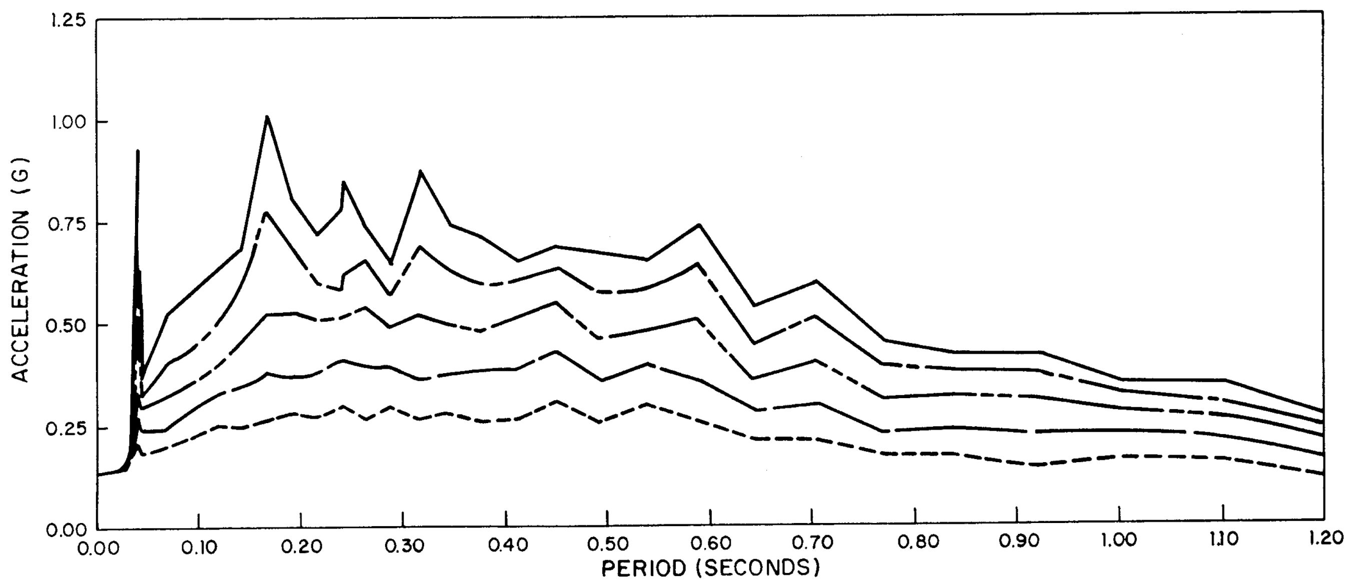
LEGEND

- 0.005 OSCILLATOR DAMPING
- 0.010 OSCILLATOR DAMPING
- 0.020 OSCILLATOR DAMPING
- 0.040 OSCILLATOR DAMPING
- 0.080 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B - 38
MAIN STEAM VALVE BUILDING
FLOOR SLAB EL. 41'-0"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–39 MAIN STEAM VALVE BUILDING FLOOR SLAB ELEVATION 41 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE



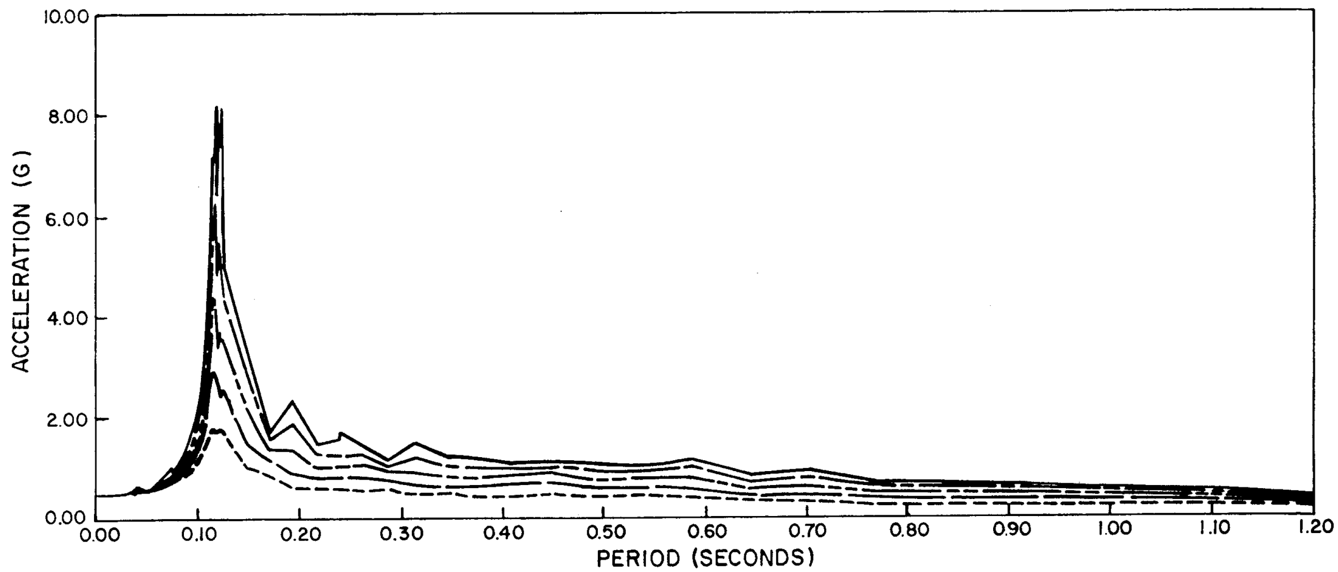
LEGEND

- 0.005 OSCILLATOR DAMPING
- 0.010 OSCILLATOR DAMPING
- 0.020 OSCILLATOR DAMPING
- 0.040 OSCILLATOR DAMPING
- 0.080 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN
+ OR – 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-39
MAIN STEAM VALVE BUILDING
FLOOR SLAB EL. 41'-0"
AMPLIFIED RESPONSE SPECTRA
VERTICAL EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–40 MAIN STEAM VALVE BUILDING ROOF SLAB ELEVATION 85 FEET 4 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE



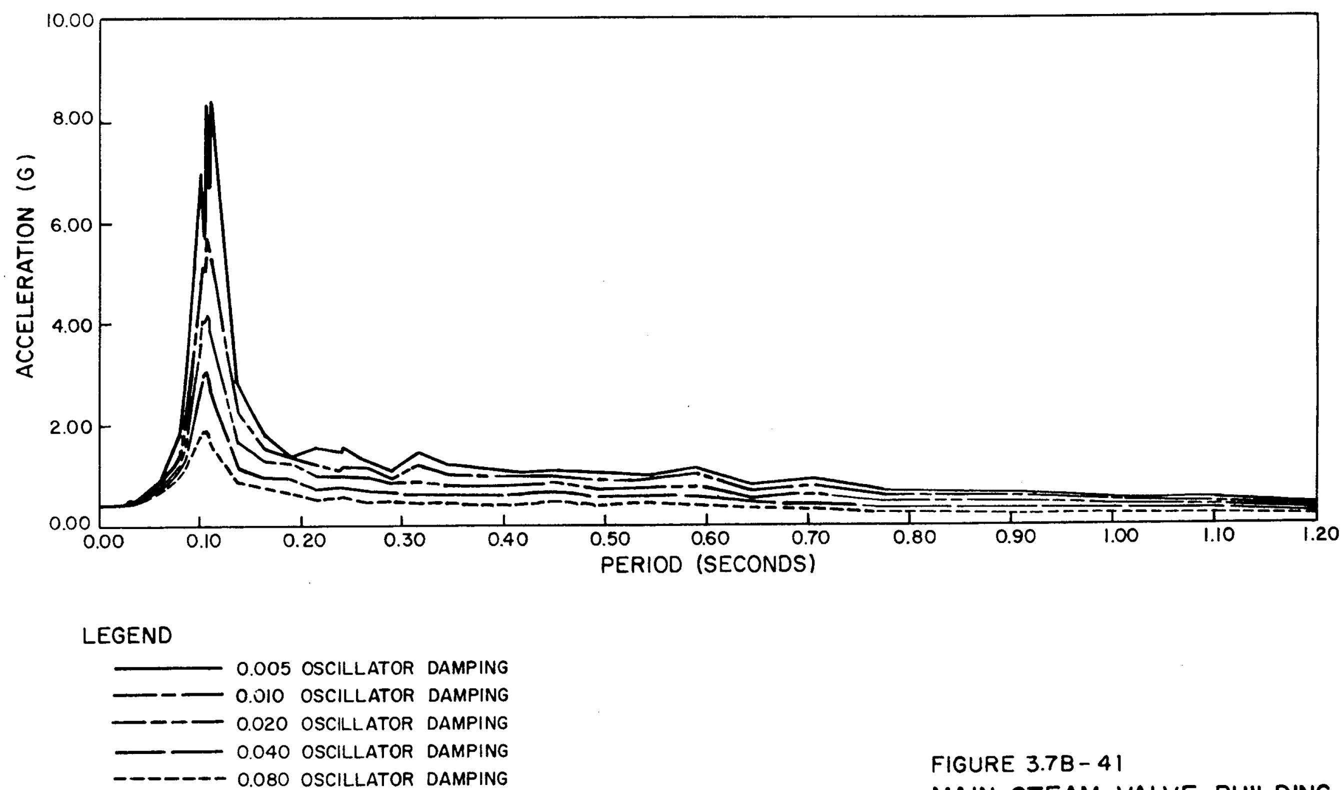
LEGEND

- 0.005 OSCILLATOR DAMPING
- 0.010 OSCILLATOR DAMPING
- 0.020 OSCILLATOR DAMPING
- 0.040 OSCILLATOR DAMPING
- 0.080 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN
+ OR − 15 % OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-40
MAIN STEAM VALVE BUILDING
ROOF SLAB EL. 85'-4"
AMPLIFIED RESPONSE SPECTRA
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–41 MAIN STEAM VALVE BUILDING ROOF SLAB ELEVATION 85 FEET 4 INCHES AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE



NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-41
MAIN STEAM VALVE BUILDING
ROOF SLAB EL. 85'-4"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–42 MAIN STEAM VALVE BUILDING ROOF SLAB ELEVATION 85 FEET 4 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE

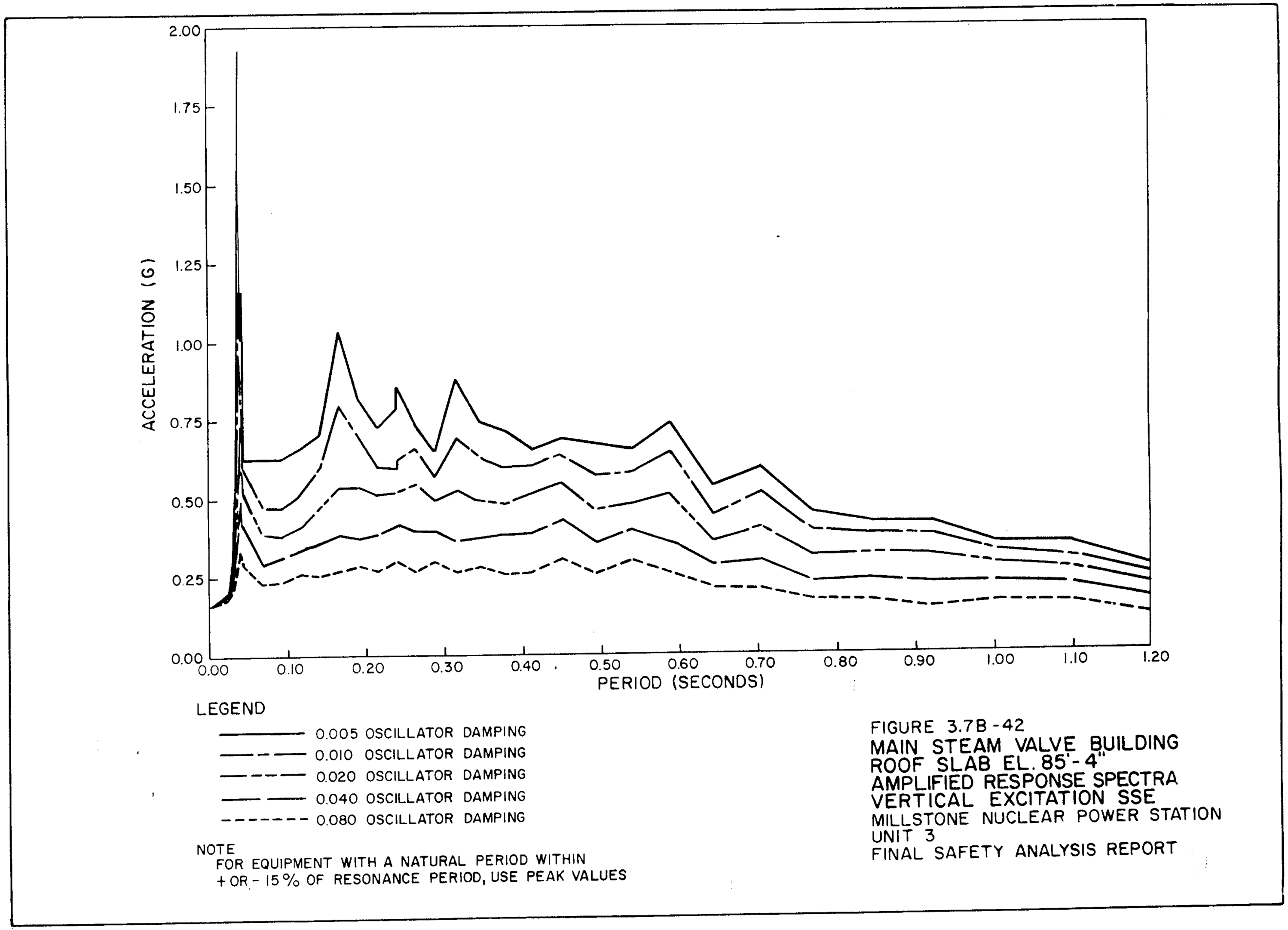


FIGURE 3.7B–43 EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION OF BASE OF STRUCTURE ELEVATION 9 FT-0 IN FOR HORIZONTAL ACCELERATION N-S EXCITATION SSE

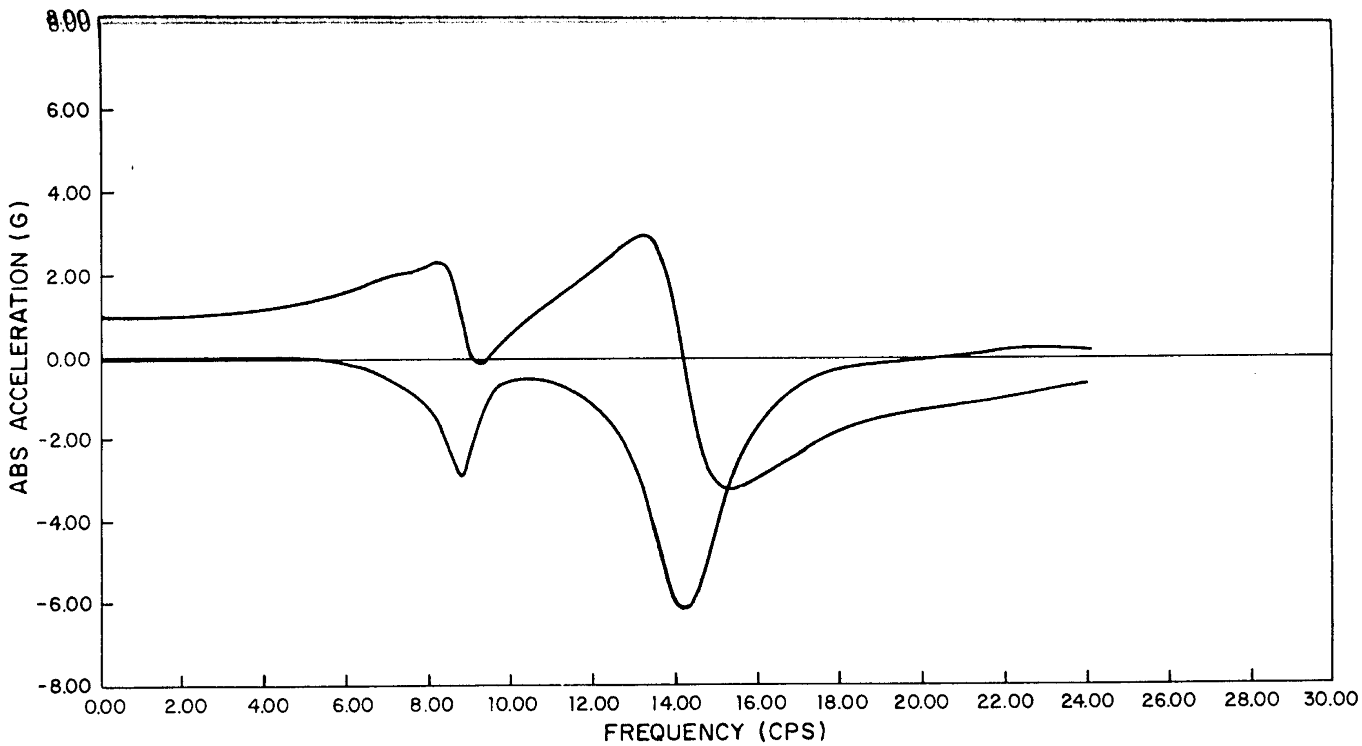


FIGURE 3.7B–43
EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION AT BASE OF STRUCTURE EL. 9'-0"
FOR HORIZONTAL ACCELERATION
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–44 EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION AT FLOOR SLAB ELEVATION 51 FEET 0 INCHES FOR HORIZONTAL ACCELERATION N-S EXCITATION SSE

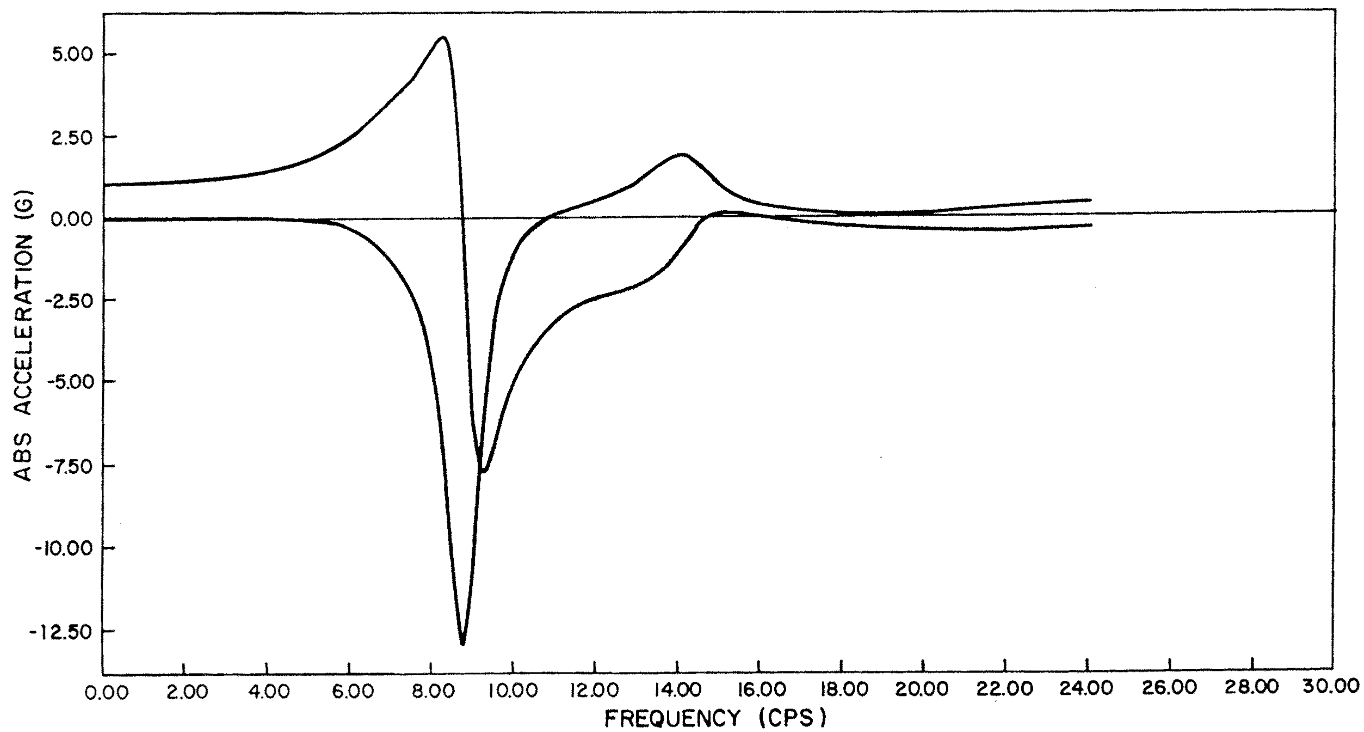


FIGURE 3.7B-44
EMERGENCY GENERATOR ENCLOSURE
TRANSFER FUNCTION AT FLOOR SLAB EL. 51'-0"
FOR HORIZONTAL ACCELERATION
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–45 EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION AT ROOF SLAB ELEVATION 66 FEET 0 INCHES FOR HORIZONTAL ACCELERATION N-S EXCITATION SSE

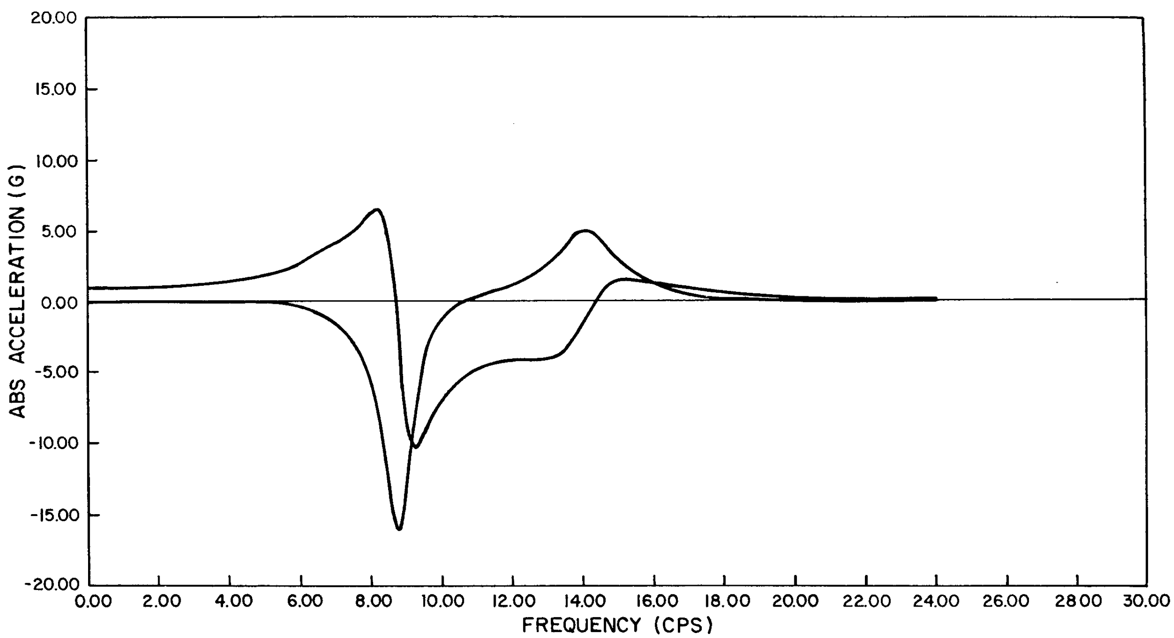


FIGURE 3.7B-45
EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION AT ROOF SLAB EL. 66'-0" FOR HORIZONTAL ACCELERATION N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

AMENDMENT 3 AUGUST 1983

FIGURE 3.7B–46 EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION AT BASE OF STRUCTURE ELEVATION 9 FEET 0 INCHES FOR VERTICAL ACCELERATION VERTICAL EXCITATION SSE

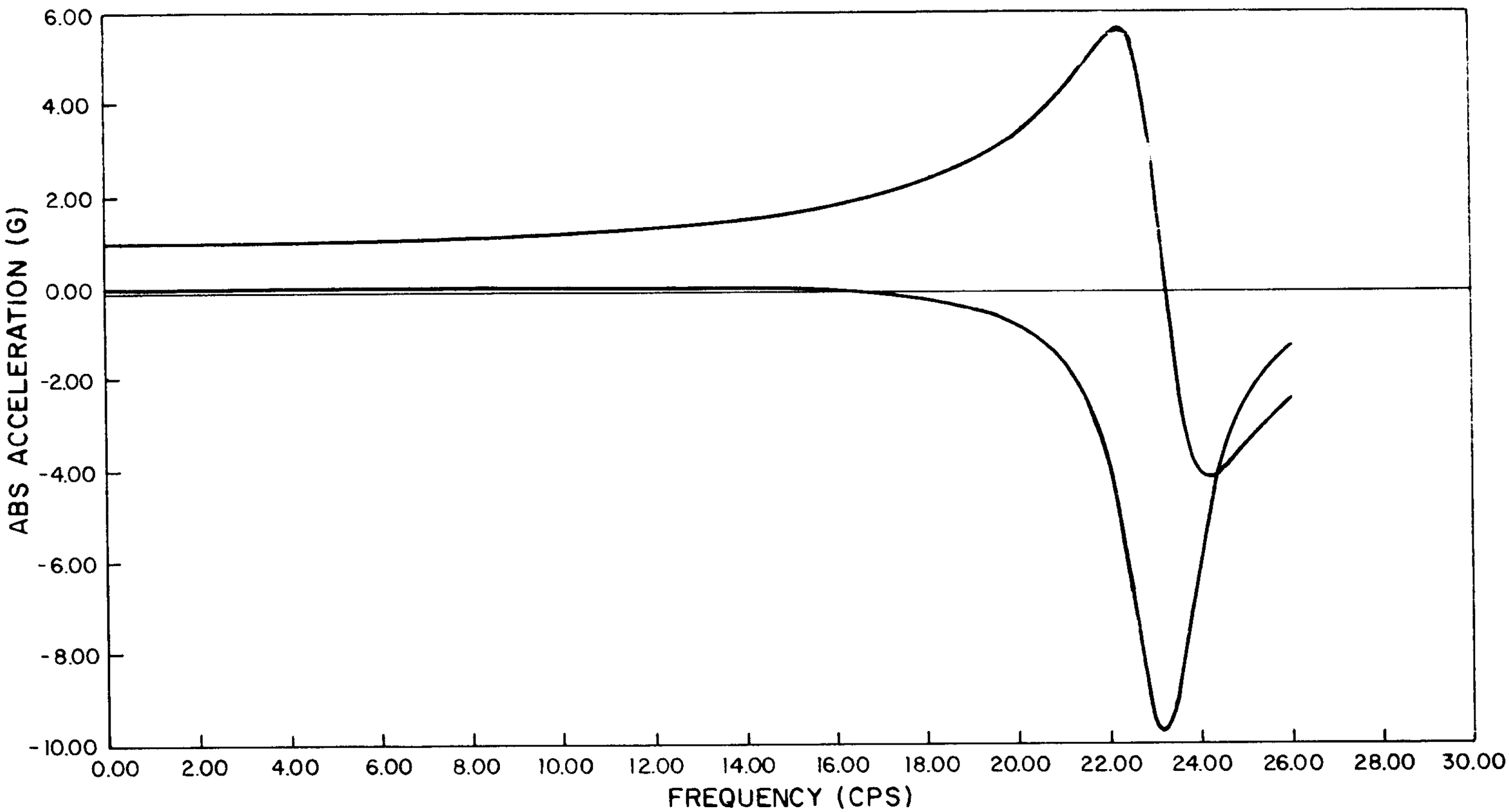


FIGURE 3.7B–47 EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION AT FLOOR SLAB ELEVATION 51 FEET 0 INCHES FOR VERTICAL ACCELERATION VERTICAL EXCITATION SSE

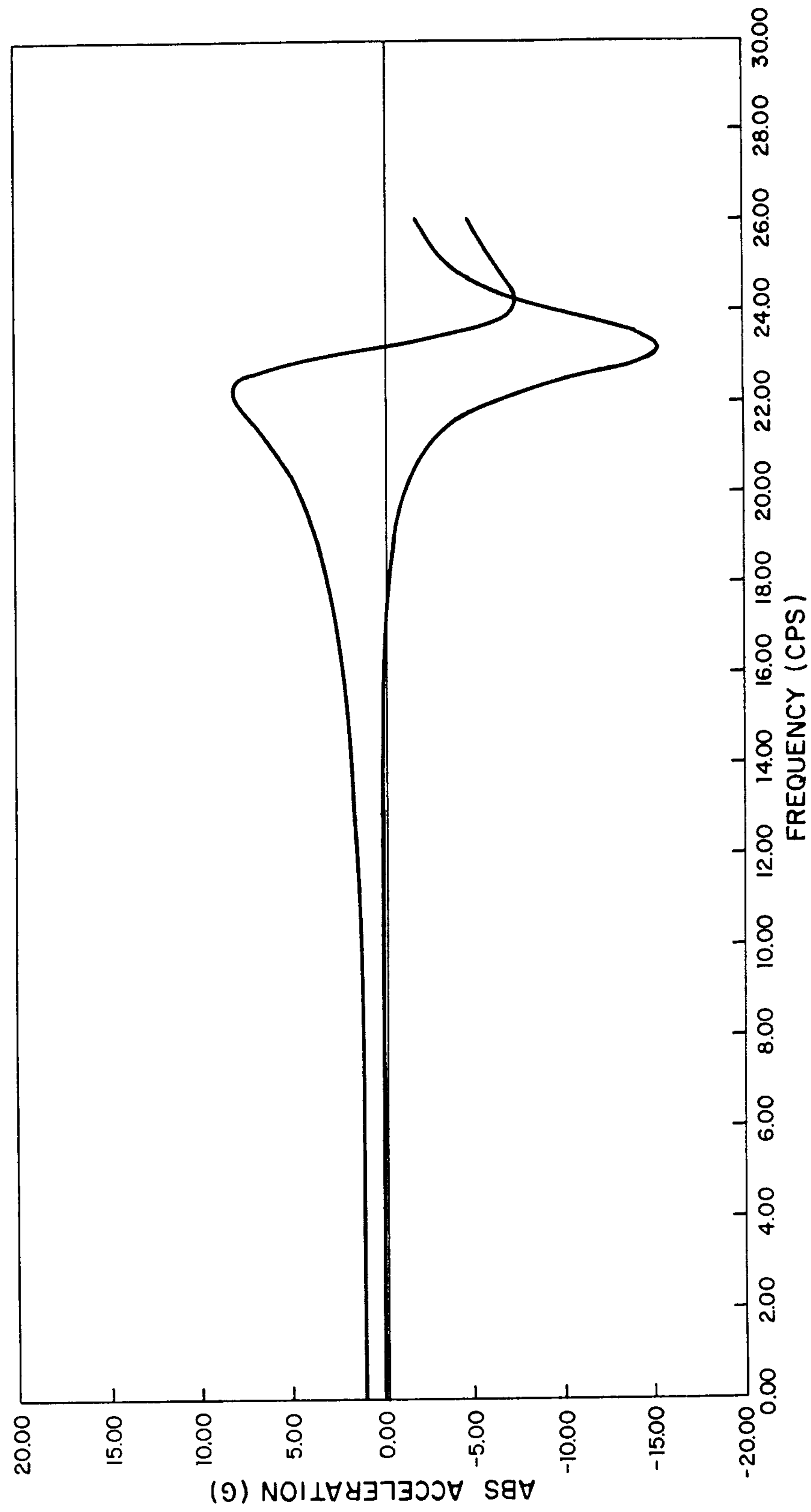


FIGURE 3.7B–48 EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION AT ROOF SLAB ELEVATION 66 FEET 0 INCHES FOR VERTICAL ACCELERATION VERTICAL EXCITATION SSE

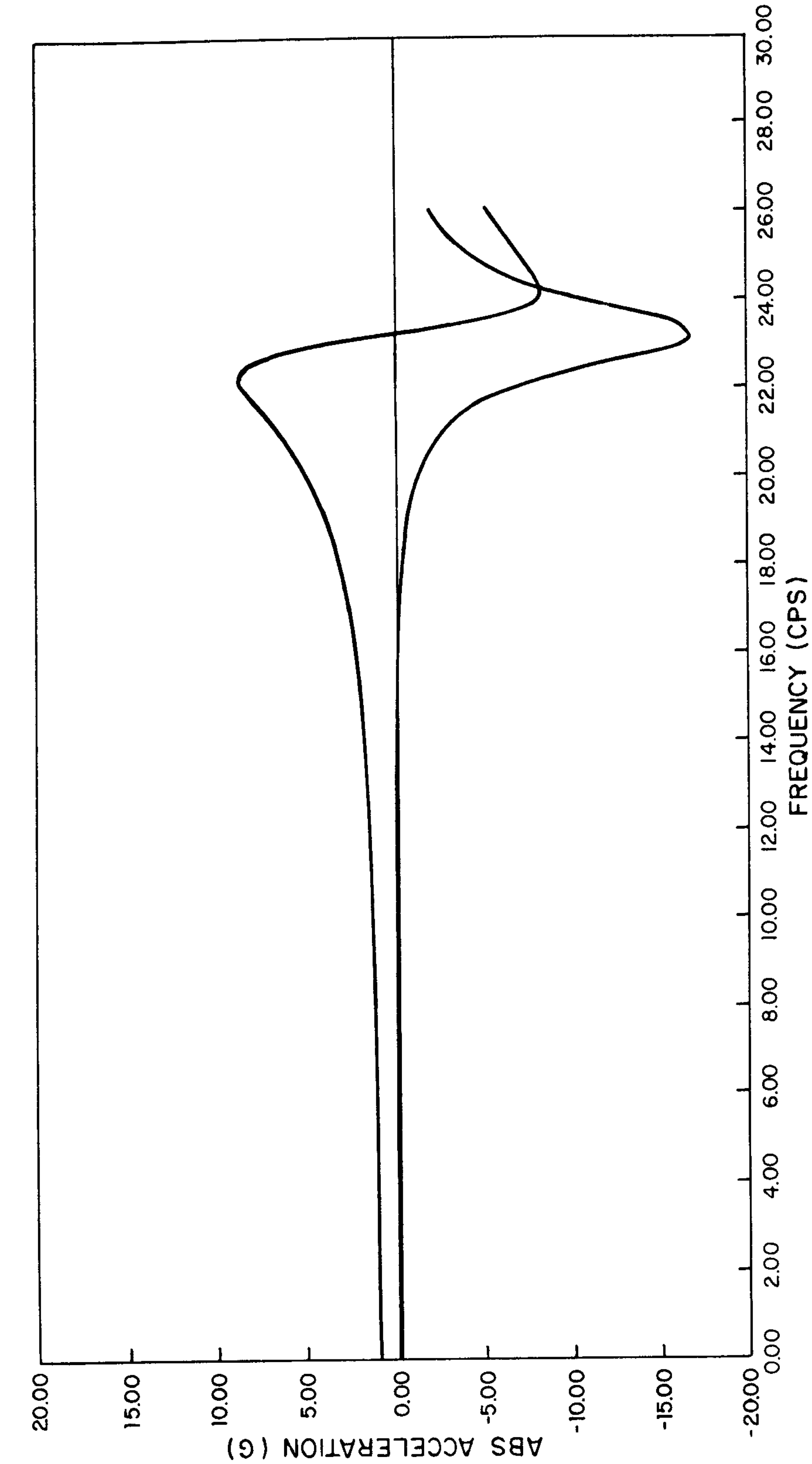


FIGURE 3.7B–49 EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION AT BASE OF STRUCTURE ELEVATION 9 FEET 0 INCHES FOR HORIZONTAL ACCELERATION E-W EXCITATION SSE

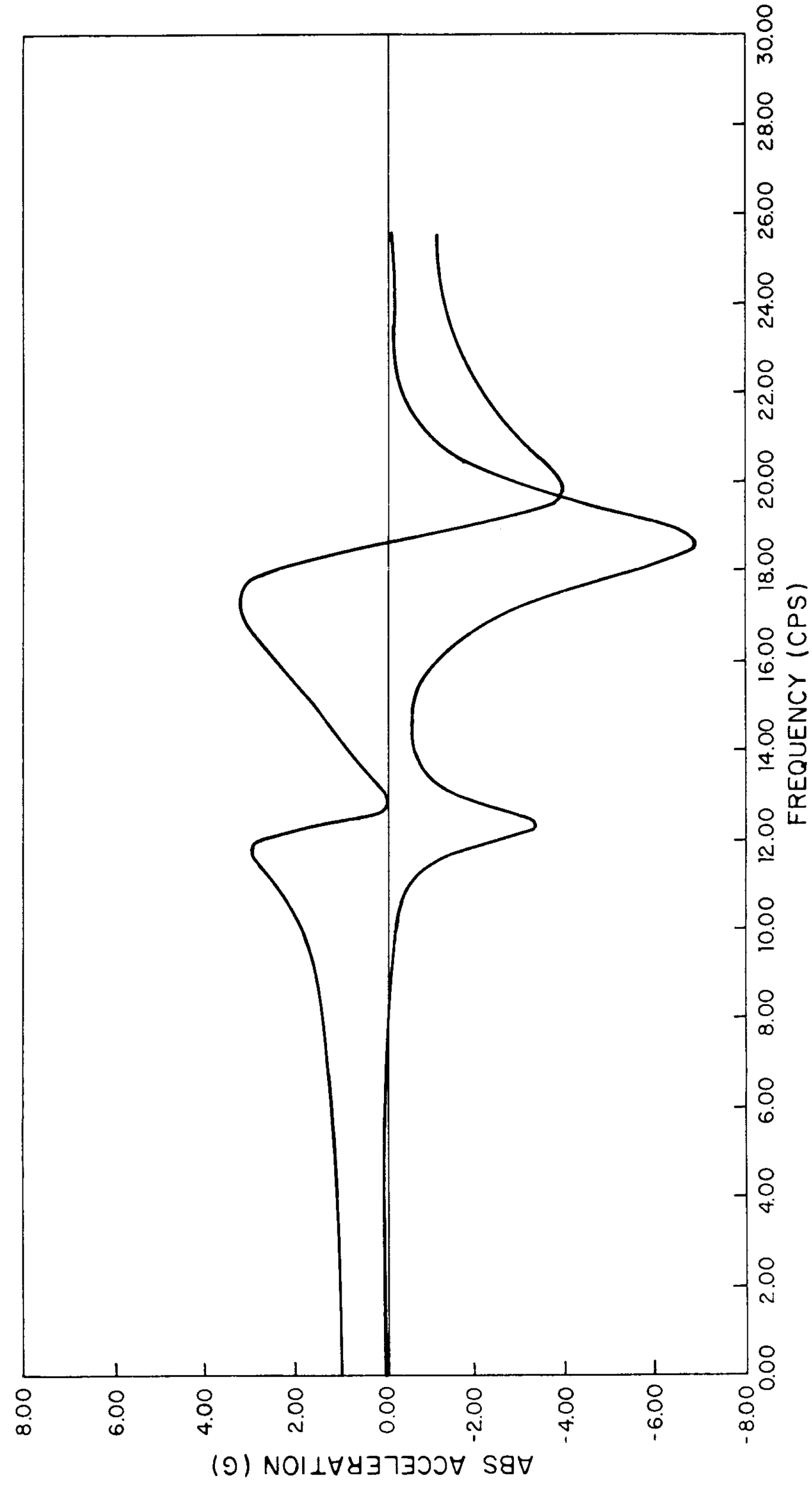


FIGURE 3.7B–50 EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION AT FLOOR SLAB ELEVATION 51 FEET 0 INCHES FOR HORIZONTAL ACCELERATION E-W EXCITATION SSE

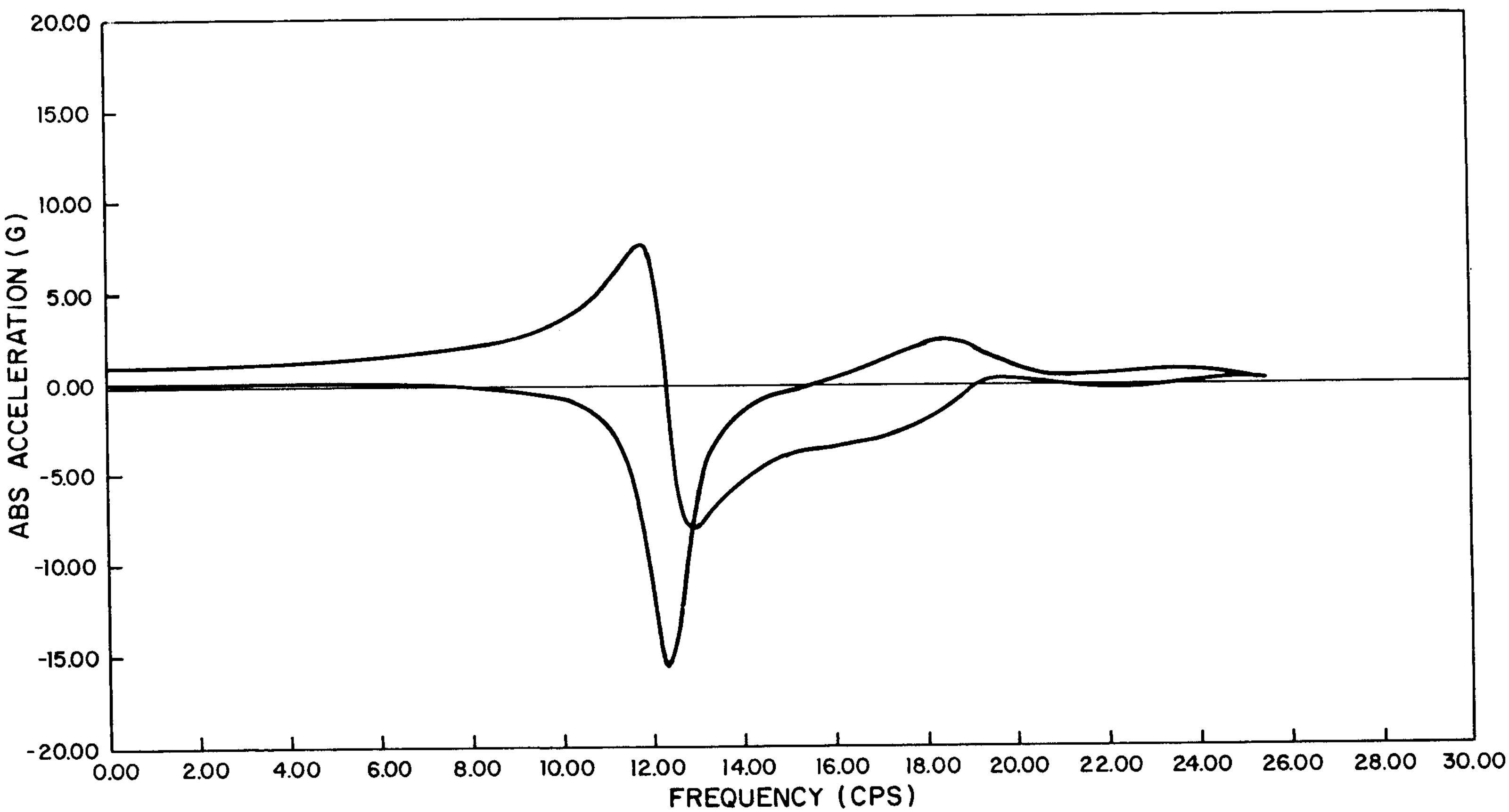


FIGURE 3.7B–51 EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION AT ROOF SLAB ELEVATION 66 FEET 0 INCHES FOR HORIZONTAL ACCELERATION E-W EXCITATION SSE

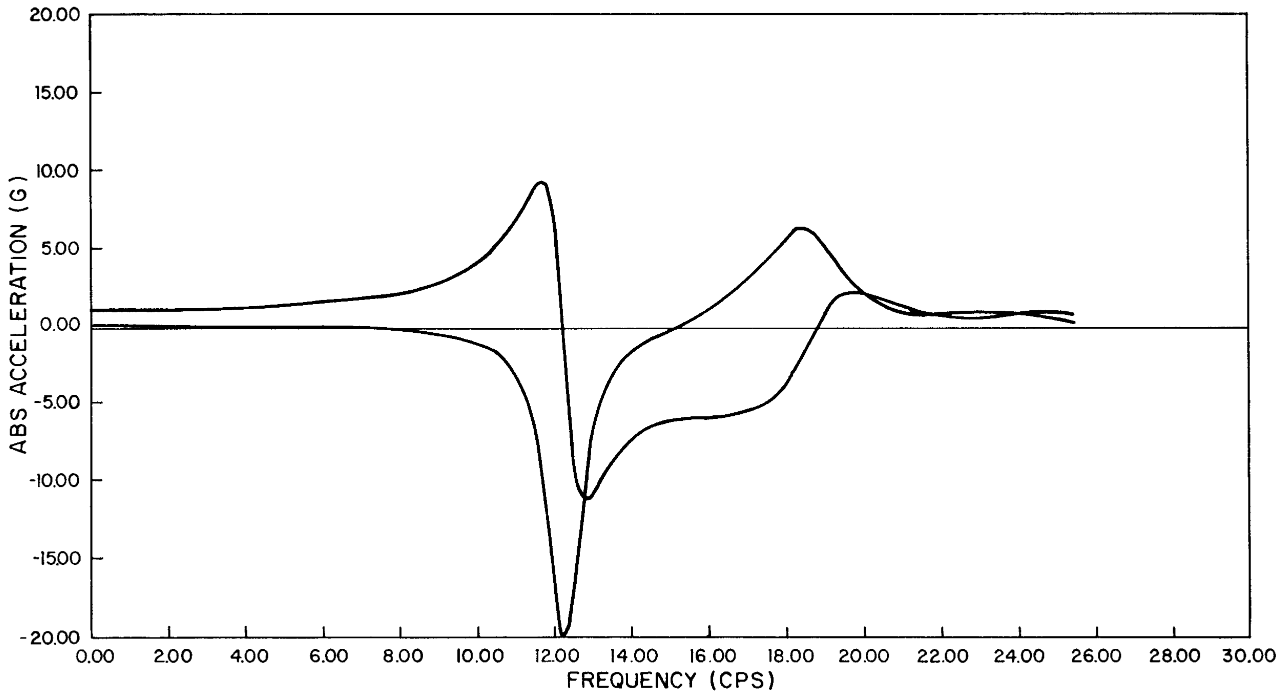
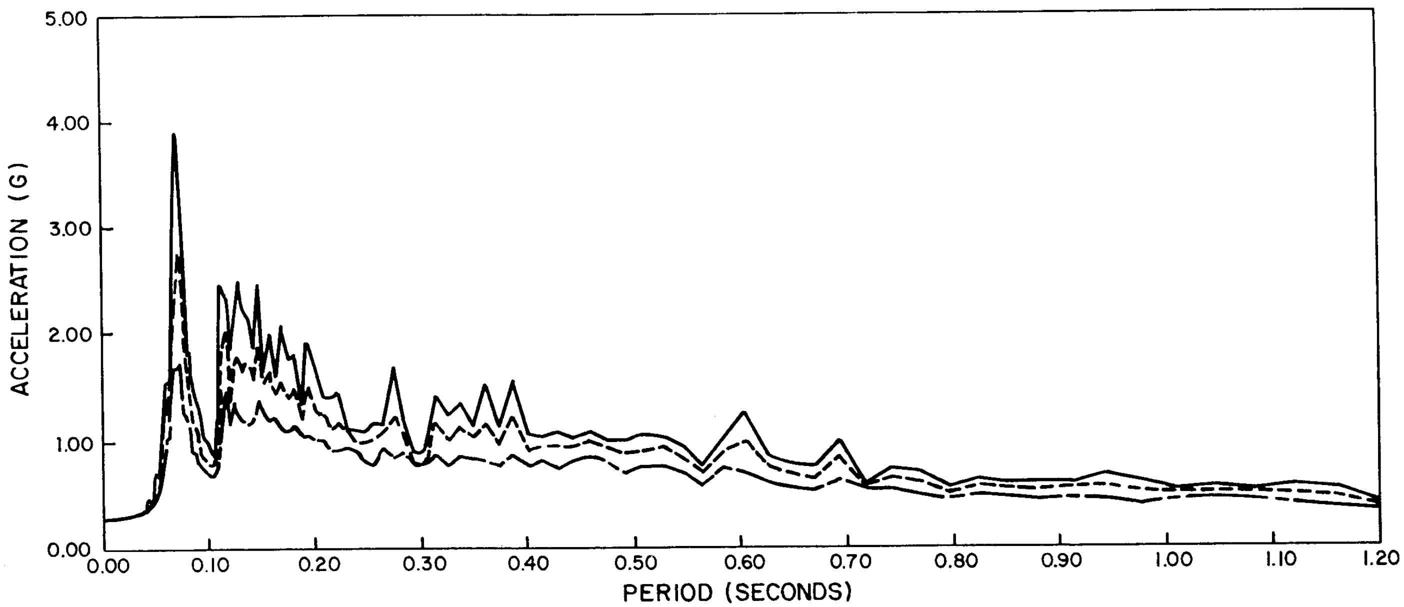


FIGURE 3.7B – 51
EMERGENCY GENERATOR ENCLOSURE TRANSFER FUNCTION AT ROOF SLAB EL. 66'-0"
FOR HORIZONTAL ACCELERATION
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–52 EMERGENCY GENERATOR ENCLOSURE FLOOR SLAB ELEVATION 24 FEET 6 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE



LEGEND

0.005 OSCILLATOR DAMPING

0.010 OSCILLATOR DAMPING

0.020 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN
+ OR - 15 % OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B - 52
EMERGENCY GENERATOR ENCLOSURE
FLOOR SLAB EL. 24'-6"
AMPLIFIED RESPONSE SPECTRA
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–53 EMERGENCY GENERATOR ENCLOSURE FLOOR SLAB ELEVATION 24 FEET 6 INCHES AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE

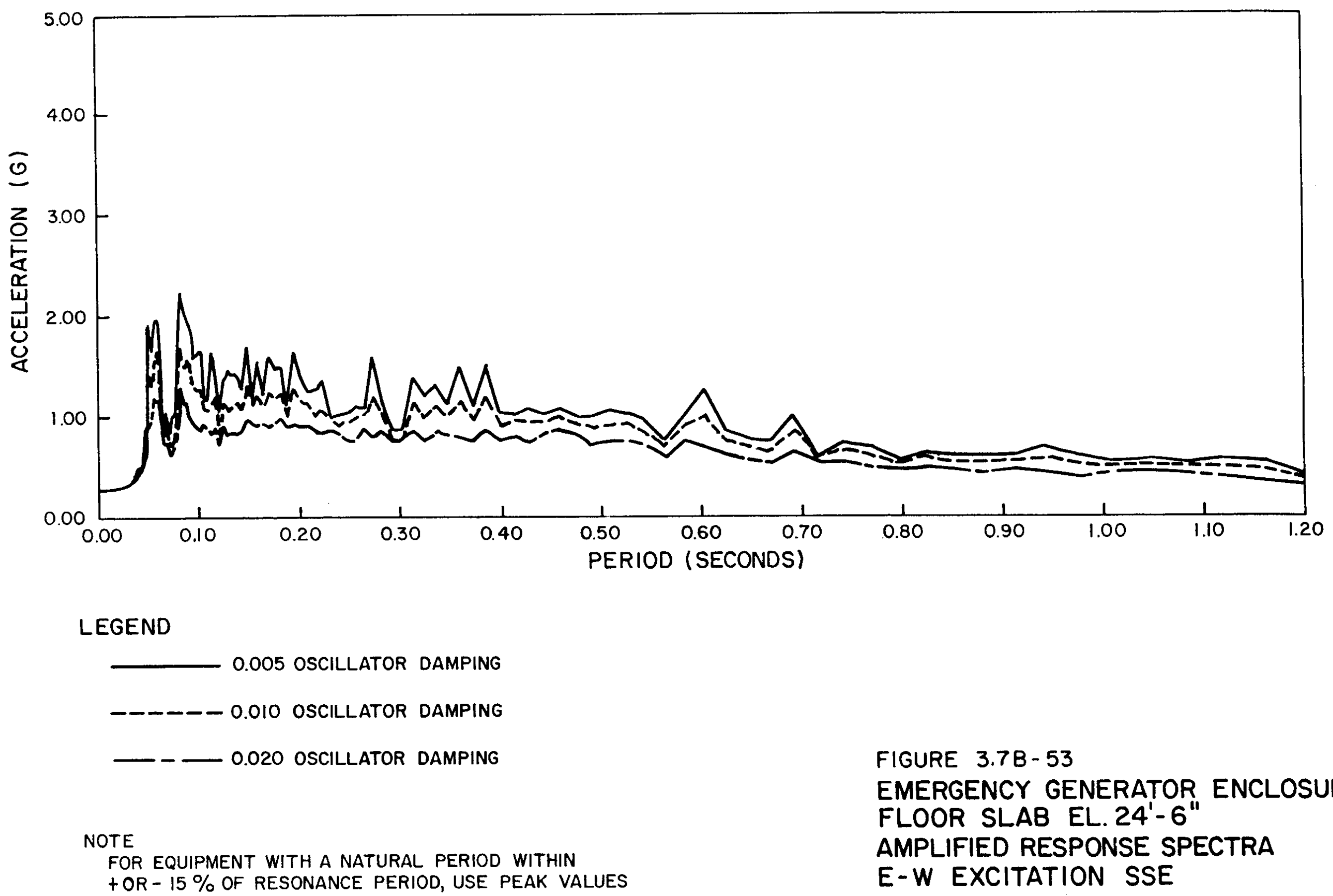


FIGURE 3.7B-53
EMERGENCY GENERATOR ENCLOSURE
FLOOR SLAB EL. 24'-6"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–54 EMERGENCY GENERATOR ENCLOSURE FLOOR SLAB ELEVATION 24 FEET 6 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE

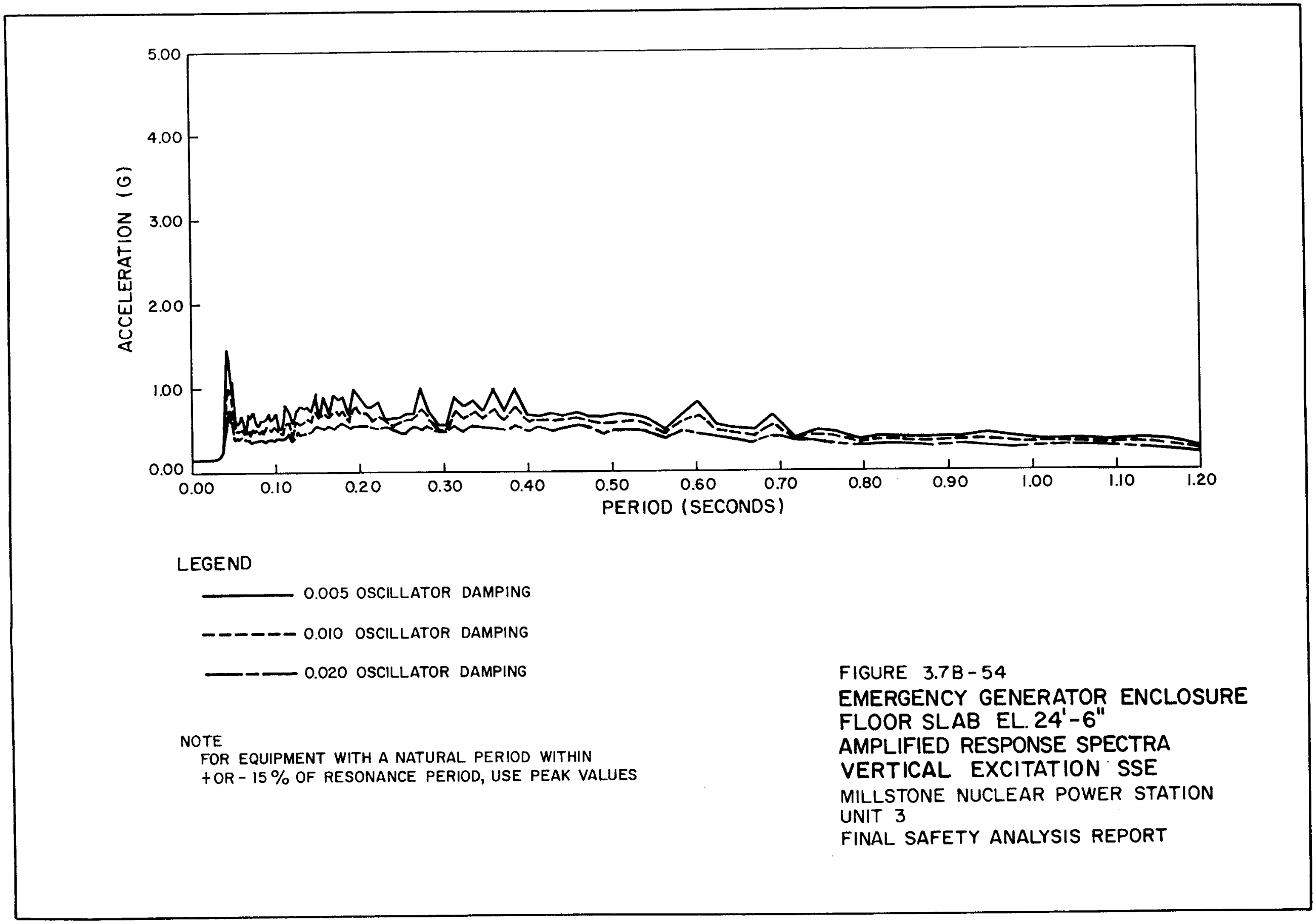


FIGURE 3.7B–55 EMERGENCY GENERATOR ENCLOSURE FLOOR SLAB ELEVATION 51 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE

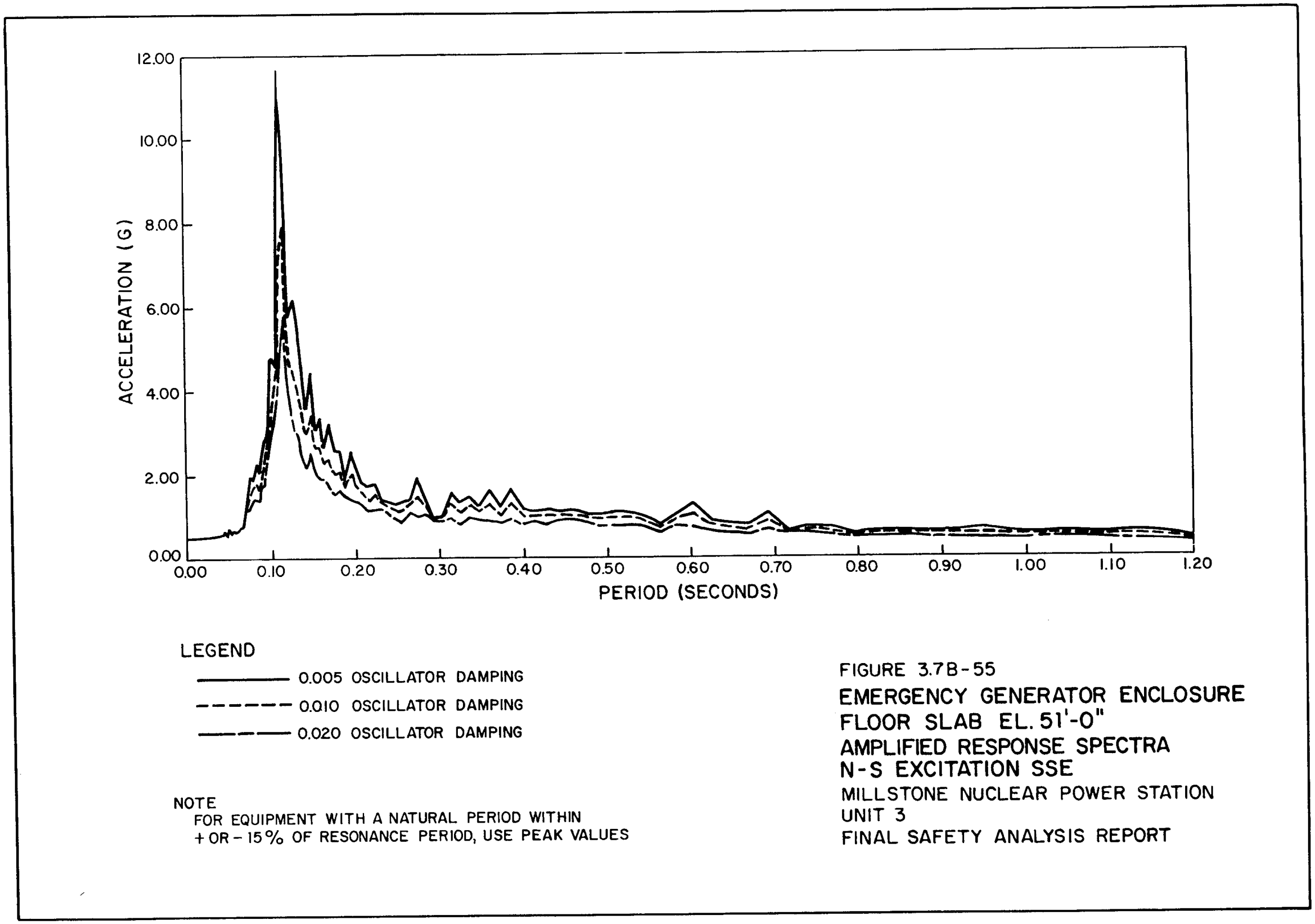
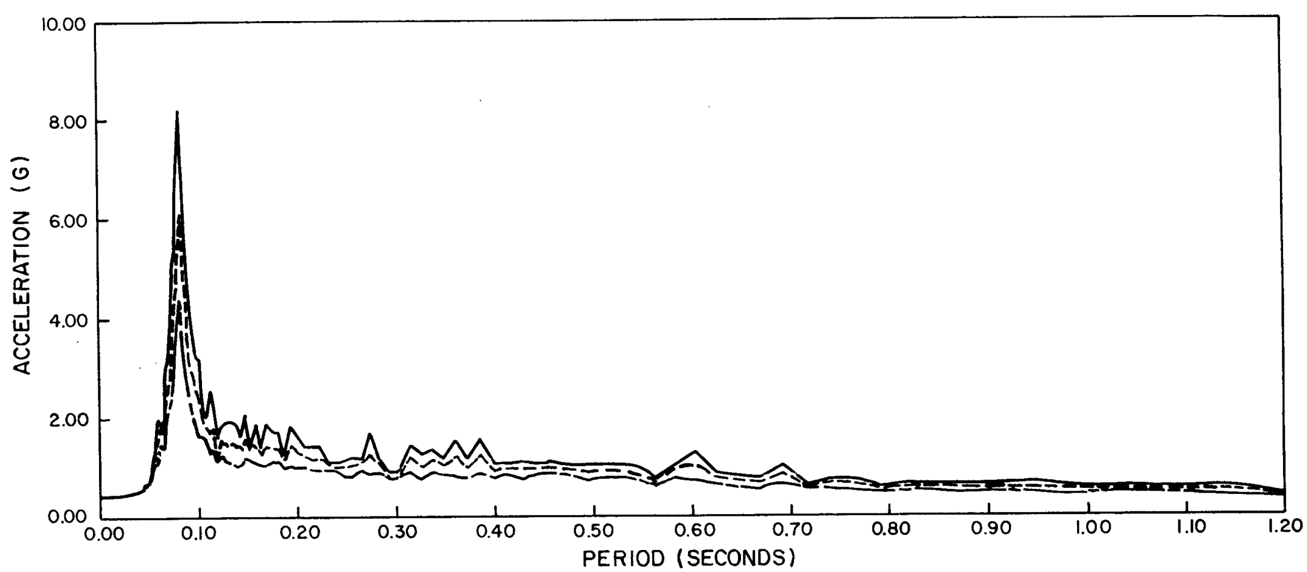


FIGURE 3.7B–56 EMERGENCY GENERATOR ENCLOSURE FLOOR SLAB ELEVATION 51 FT-0 IN AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE



LEGEND

0.005 OSCILLATOR DAMPING

0.010 OSCILLATOR DAMPING

0.020 OSCILLATOR DAMPING

NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN
+ OR - 15 % OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-56
EMERGENCY GENERATOR ENCLOSURE
FLOOR SLAB EL.51'-0"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–57 EMERGENCY GENERATOR ENCLOSURE FLOOR SLAB ELEVATION 51 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE

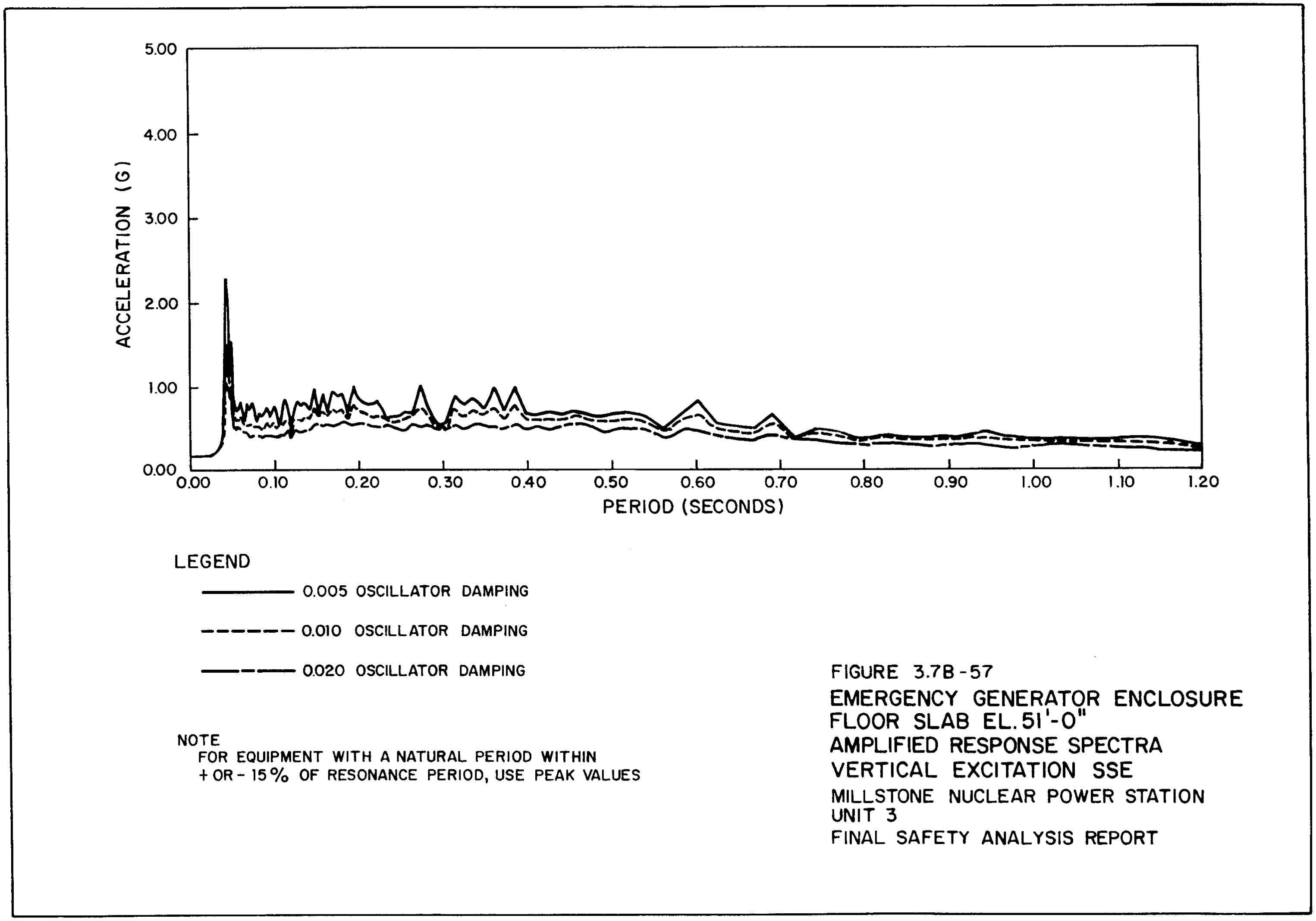


FIGURE 3.7B–58 EMERGENCY GENERATOR ENCLOSURE ROOF SLAB ELEVATION 66 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE

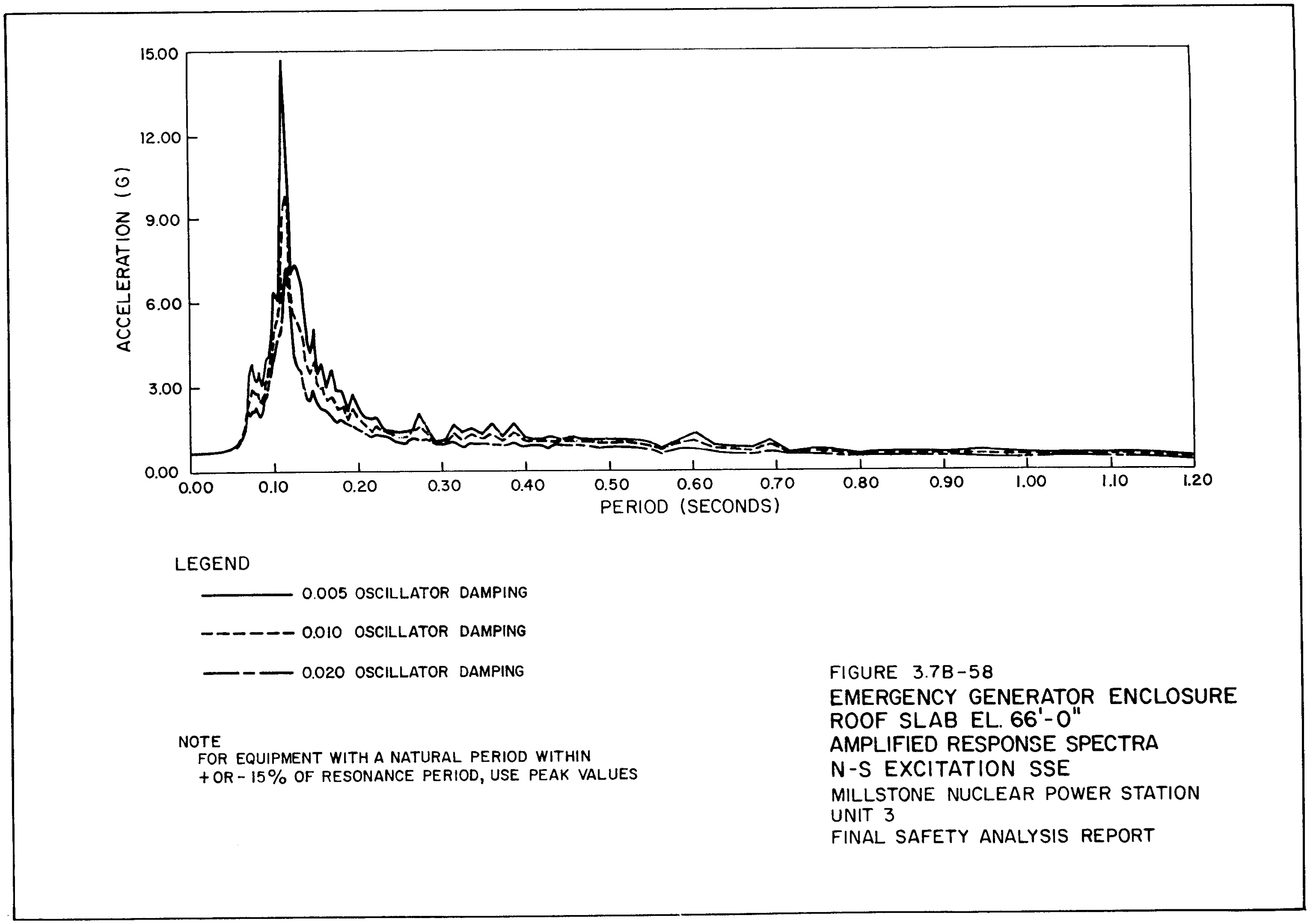


FIGURE 3.7B–59 EMERGENCY GENERATOR ENCLOSURE ROOF SLAB ELEVATION 66 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA E-W EXCITATION SSE

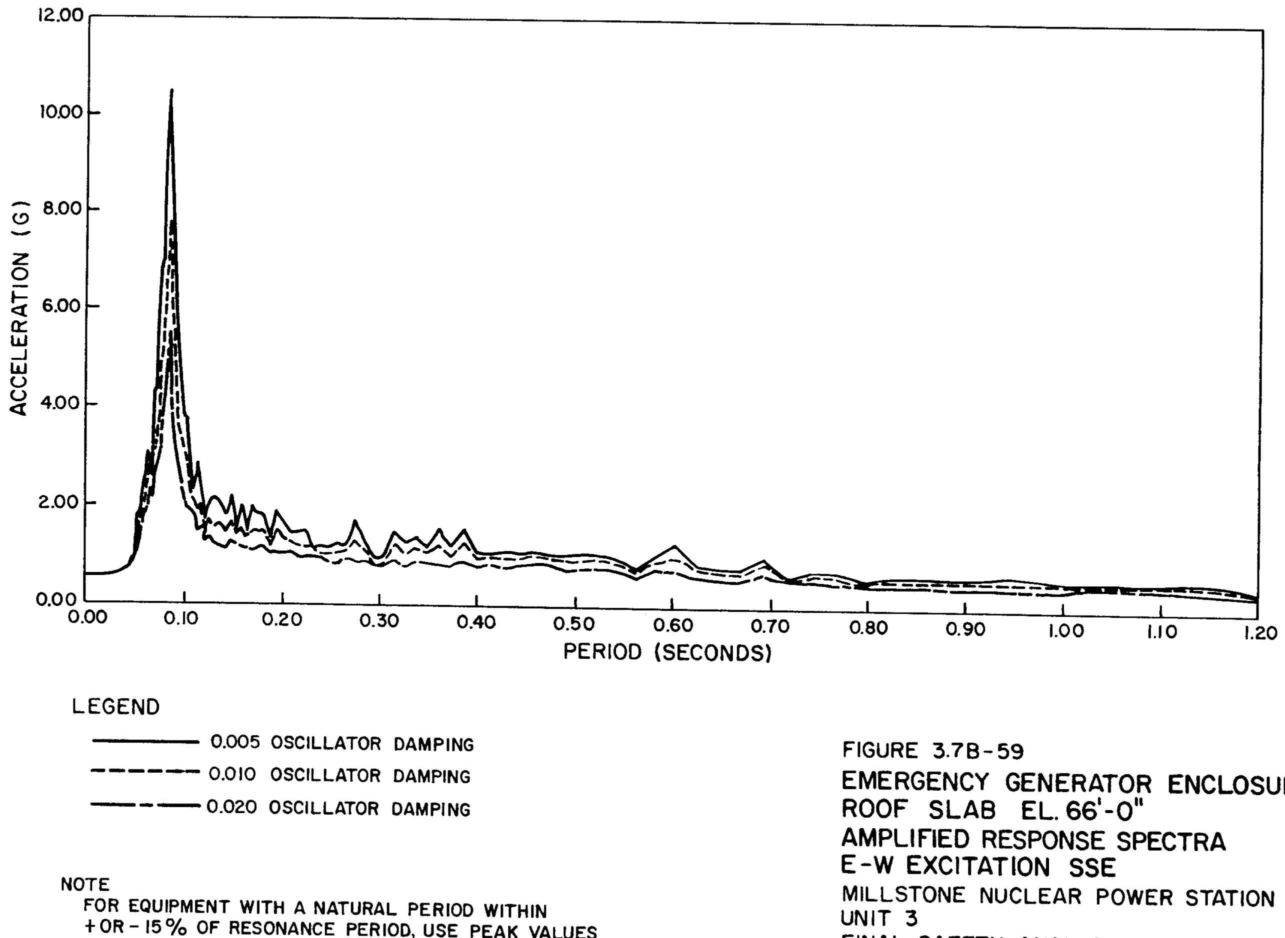


FIGURE 3.7B-59
EMERGENCY GENERATOR ENCLOSURE
ROOF SLAB EL. 66'-0"
AMPLIFIED RESPONSE SPECTRA
E-W EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–60 EMERGENCY GENERATOR ENCLOSURE ROOF SLAB ELEVATION 66 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE

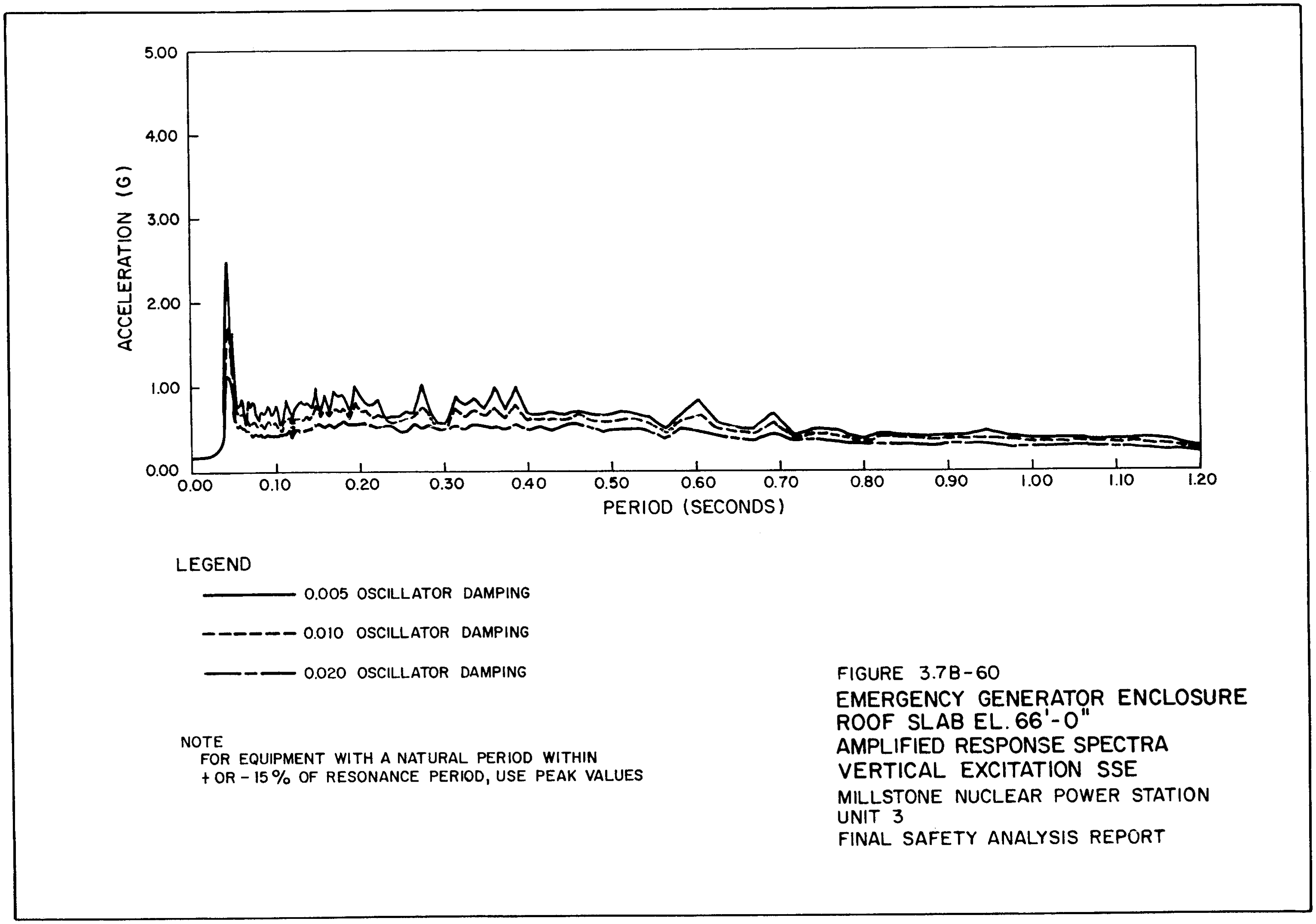


FIGURE 3.7B–61 EMERGENCY GENERATOR ENCLOSURE DIESEL GENERATOR MAT ELEVATION 24 FEET 0 INCHES AMPLIFIED RESPONSE SPECTRA N-S EXCITATION SSE

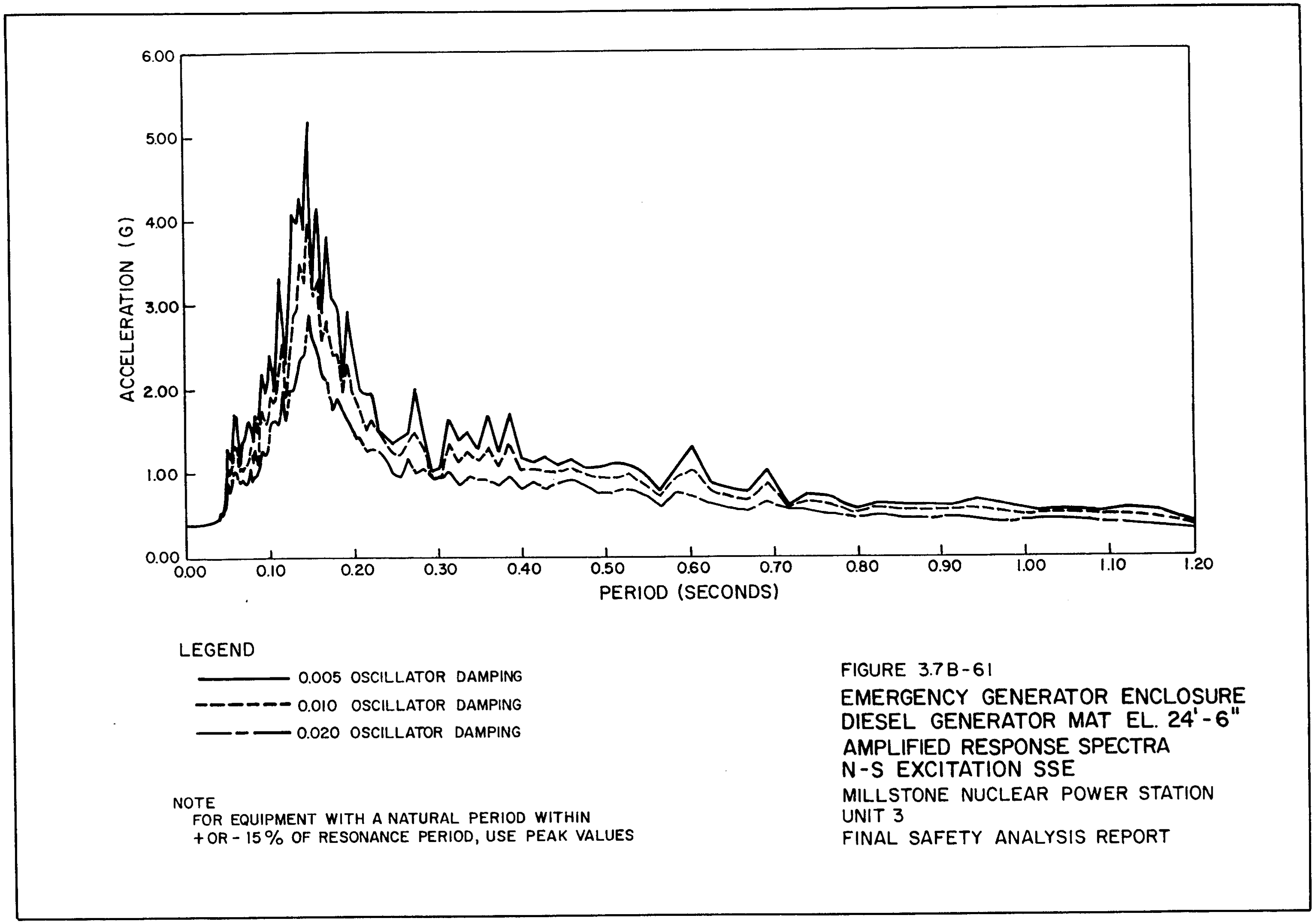
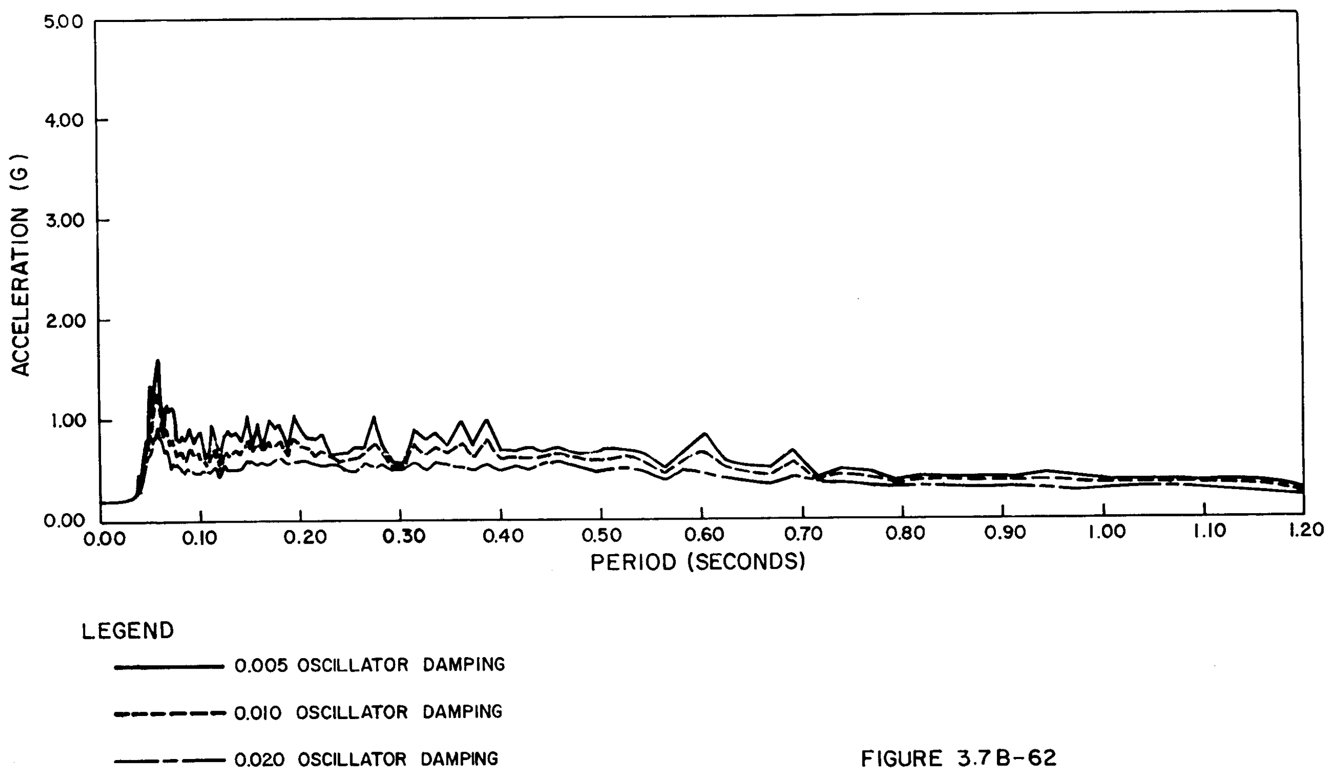


FIGURE 3.7B–62 EMERGENCY GENERATOR ENCLOSURE DIESEL GENERATOR MAT ELEVATION 24 FEET 6 INCHES AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE



NOTE
FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN +OR- 15% OF RESONANCE PERIOD, USE PEAK VALUES

FIGURE 3.7B-62
EMERGENCY GENERATOR ENCLOSURE DIESEL GENERATOR MAT EL. 24'-6" AMPLIFIED RESPONSE SPECTRA VERTICAL EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–63 EMERGENCY GENERATOR ENCLOSURE HORIZONTAL ACCELERATION TIME HISTORY AT BEDROCK N-S EXCITATION SSE

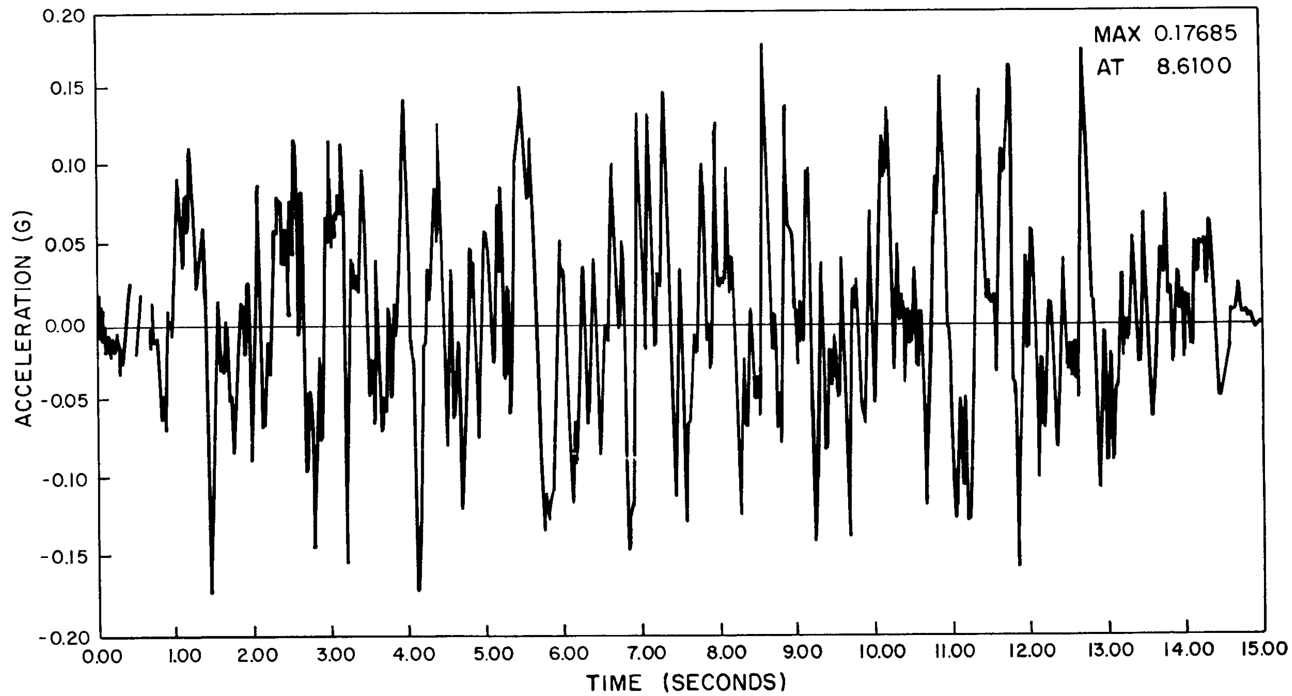


FIGURE 3.7B-63
EMERGENCY GENERATOR ENCLOSURE
HORIZONTAL ACCELERATION TIME HISTORY
AT BEDROCK
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–64 EMERGENCY GENERATOR ENCLOSURE HORIZONTAL ACCELERATION TIME HISTORY AT BASE OF STRUCTURE N-S EXCITATION SSE

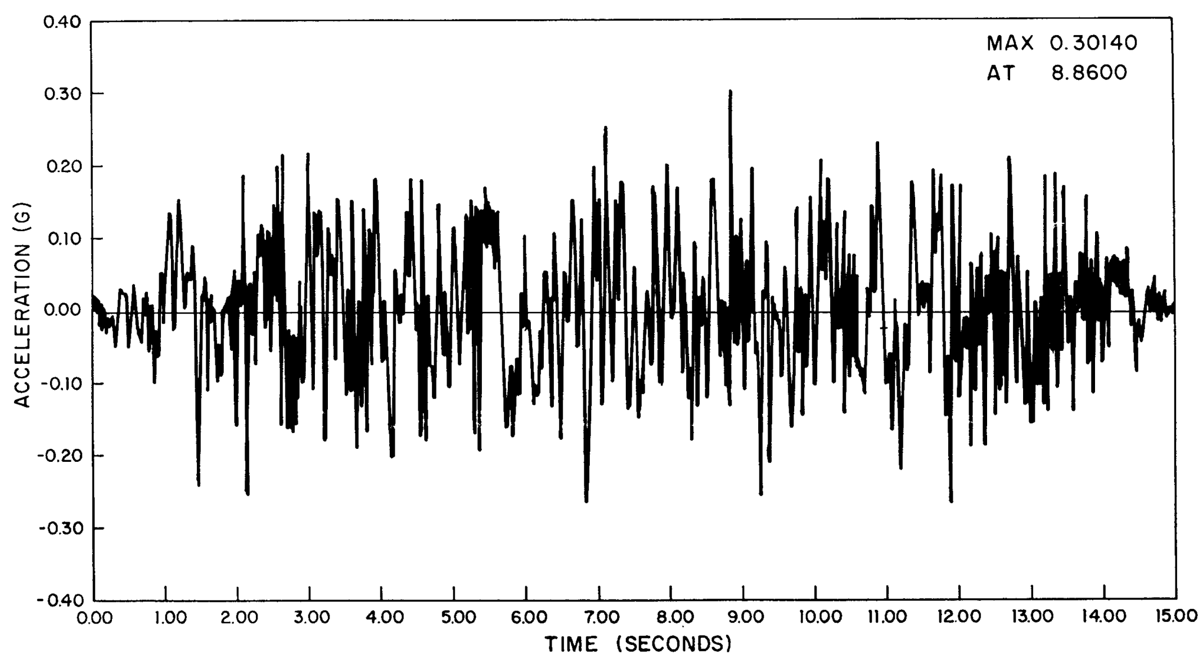


FIGURE 3.7B-64
EMERGENCY GENERATOR ENCLOSURE
HORIZONTAL ACCELERATION TIME HISTORY
AT BASE OF STRUCTURE
N-S EXCITATION SSE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.7B–65 TYPICAL MATHEMATICAL MODEL OF A PIPING SYSTEM

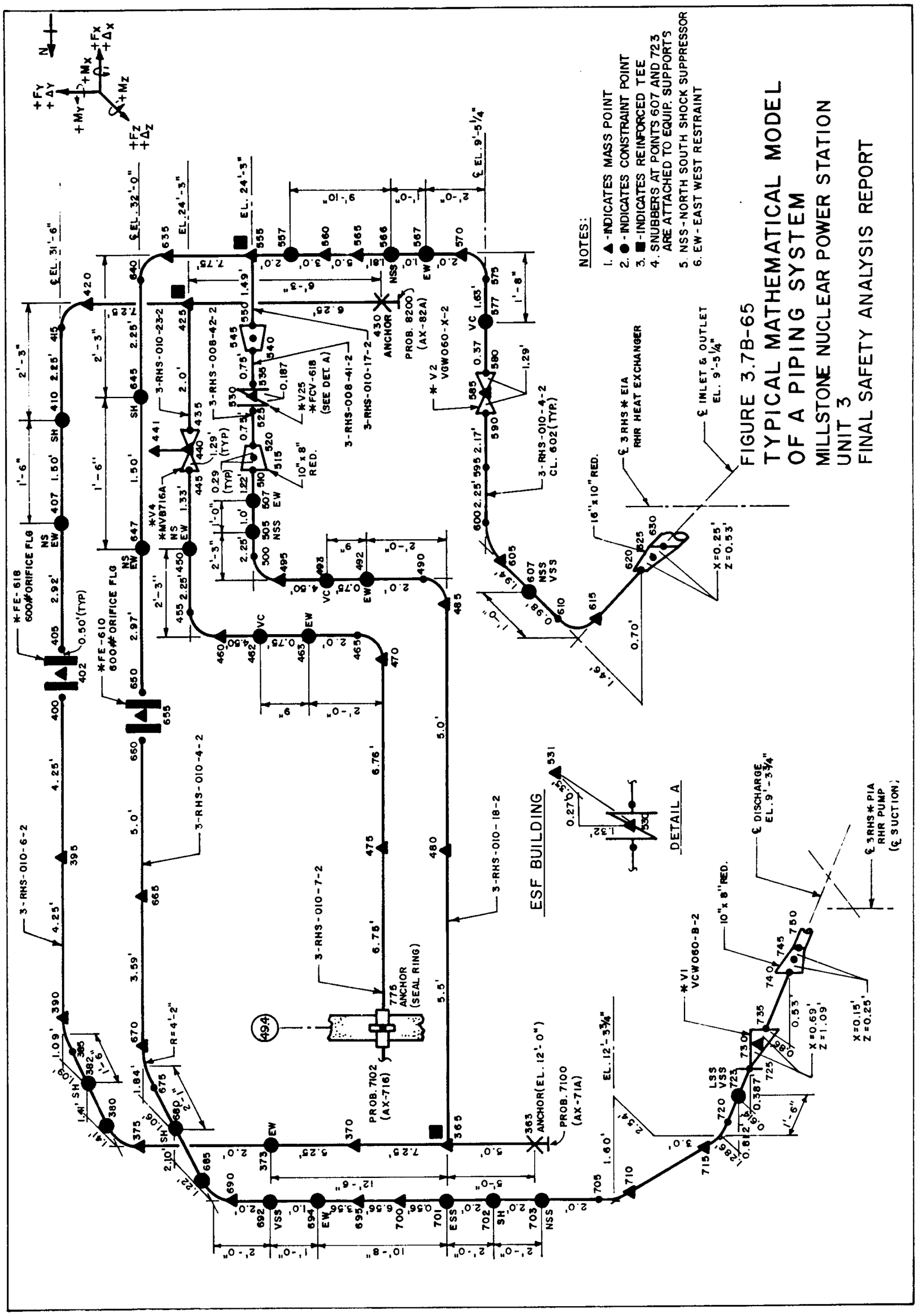


FIGURE 3.7B–66 REPRESENTATION OF FAMILY OF PEAK RESPONSES CURVES WITHIN BROADENED RESONANT PEAK

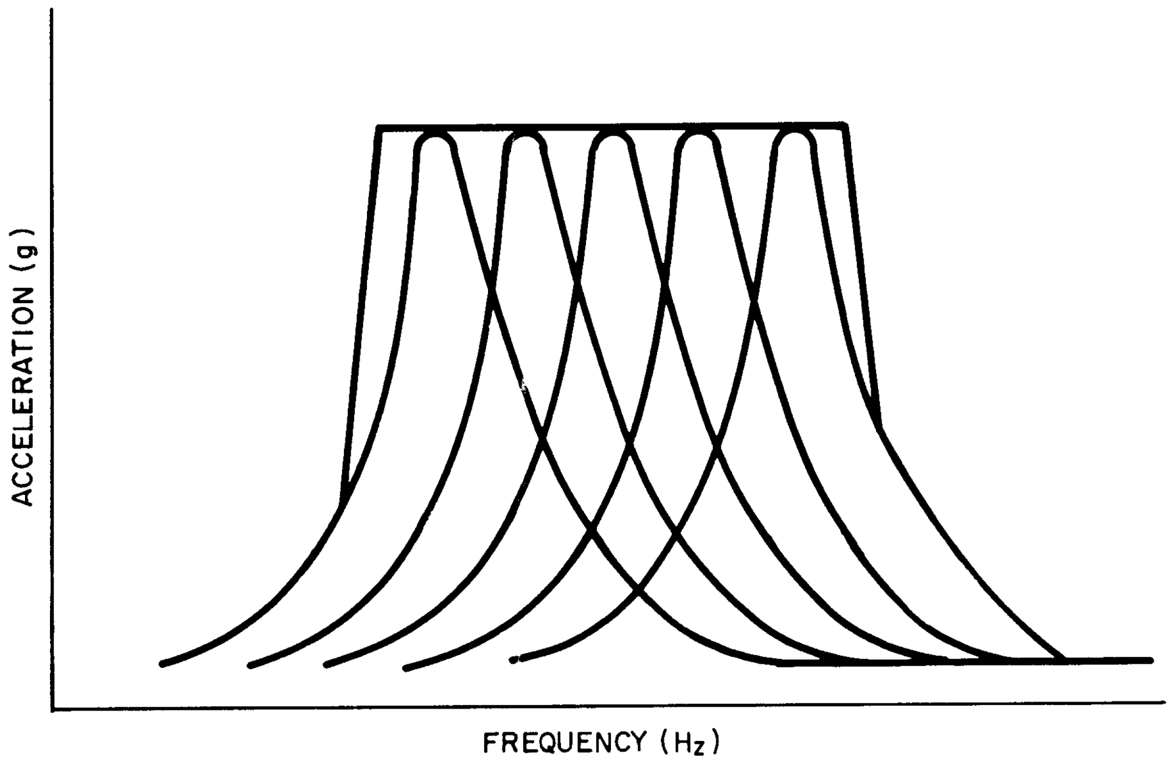


FIGURE 3.7B–67 HYPOTHETICAL VS ACTUAL RESPONSE OF MULTIPLE MODES WITHIN BROADENED RESPONSE PEAK

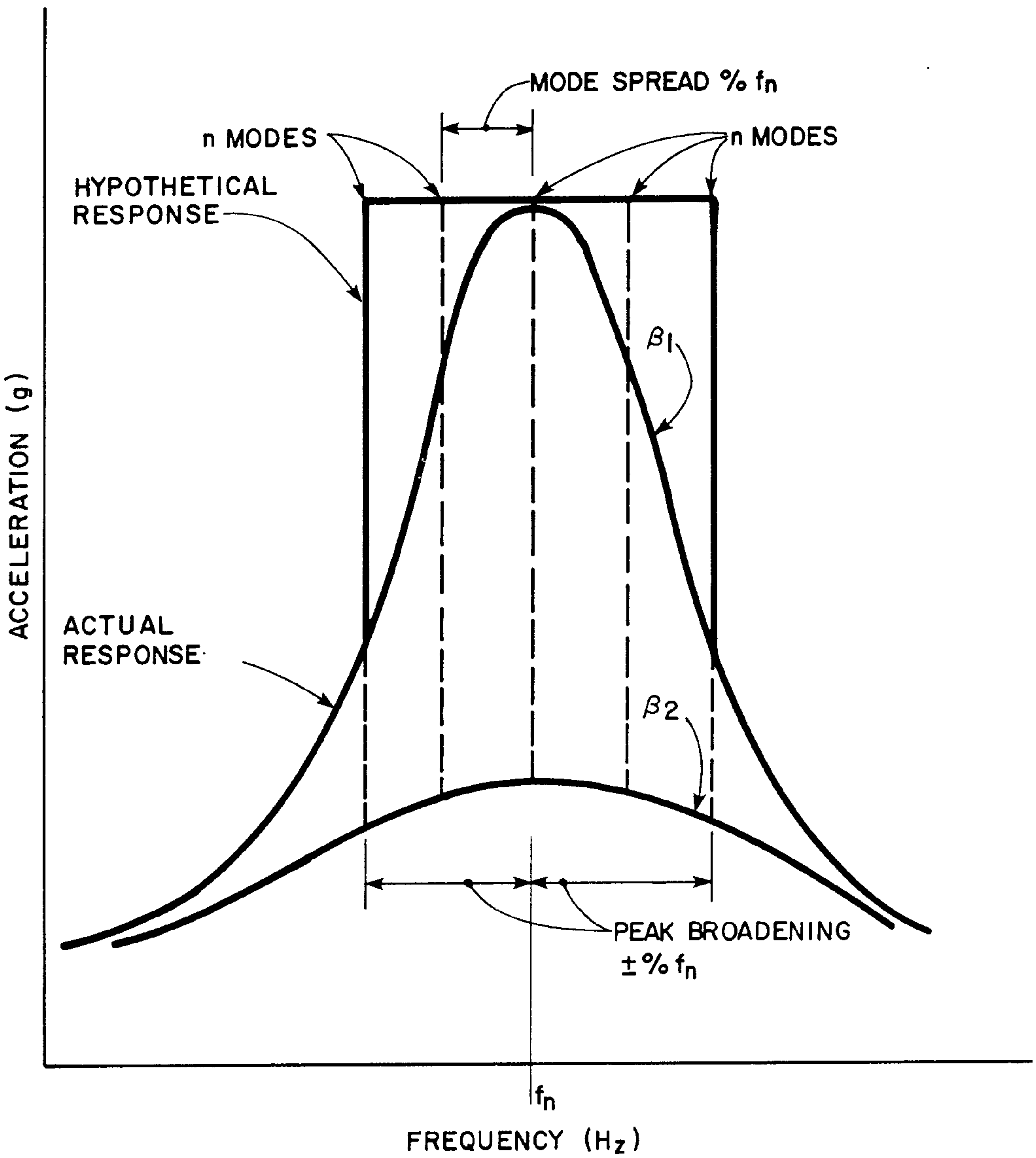


FIGURE 3.7B–68 JUSTIFICATION OF STATIC LOAD FACTOR

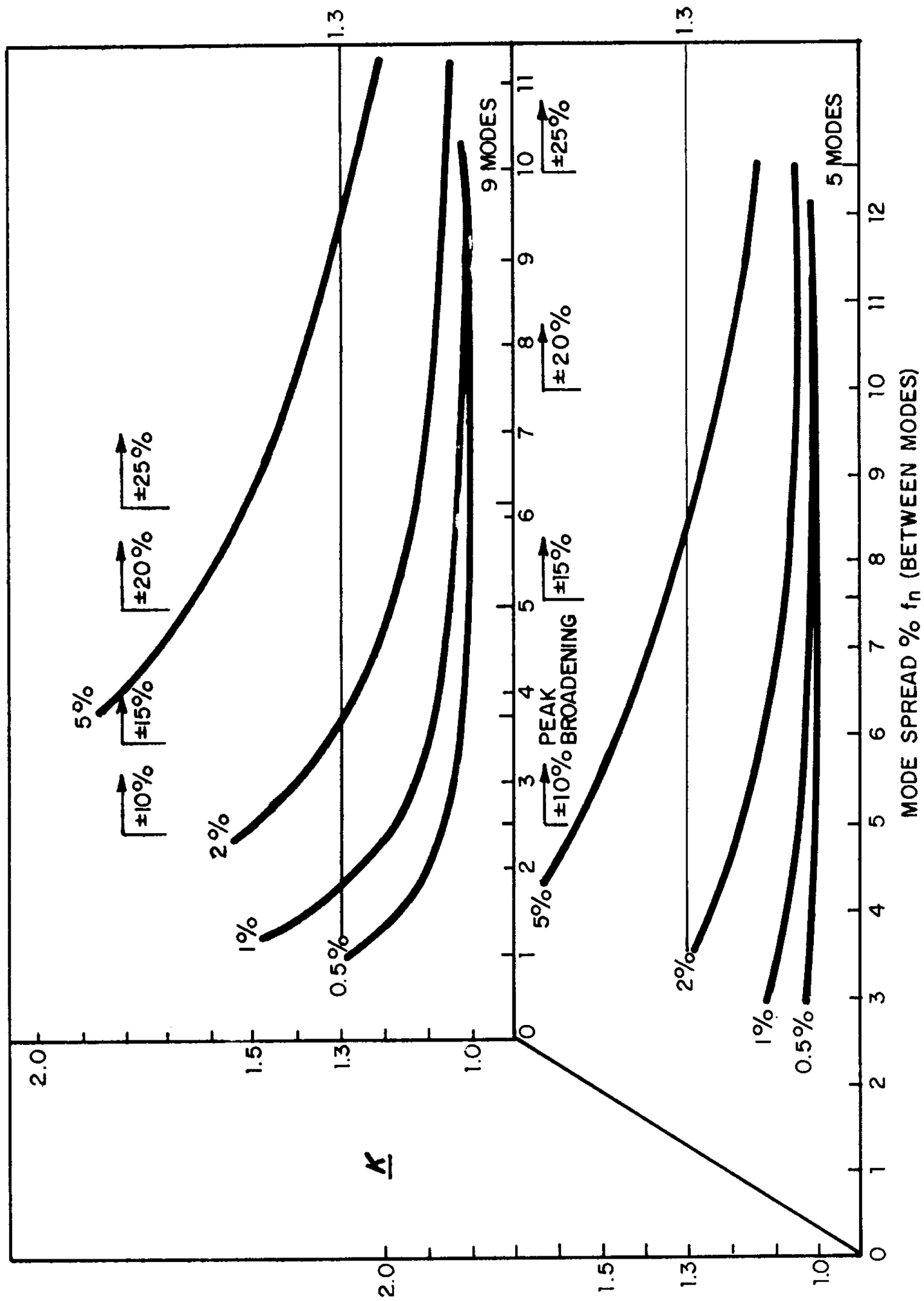


FIGURE 3.7B–69 TYPICAL AMPLIFIED RESPONSE SPECTRA

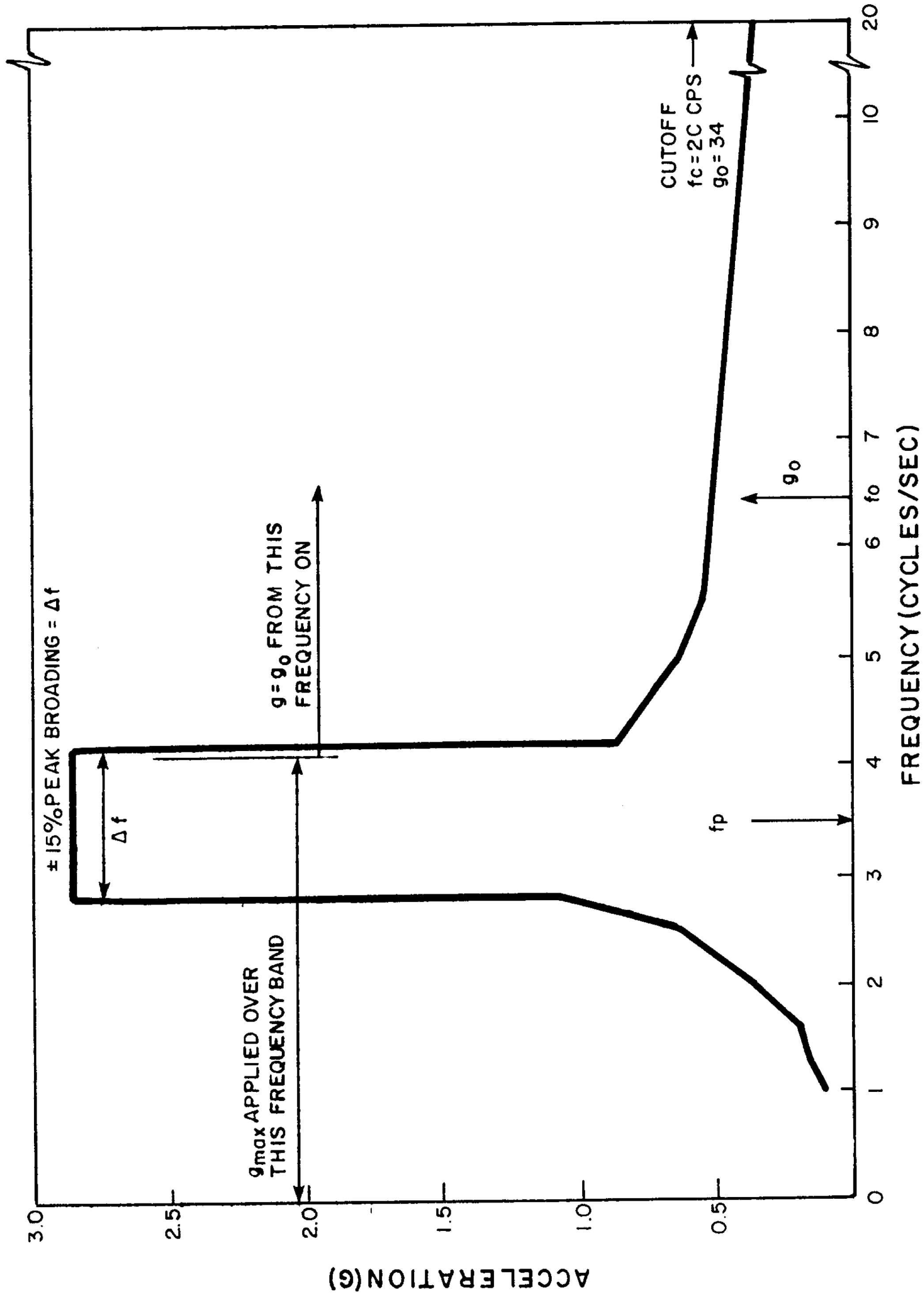


FIGURE 3.7B–70 MODEL BEAMS

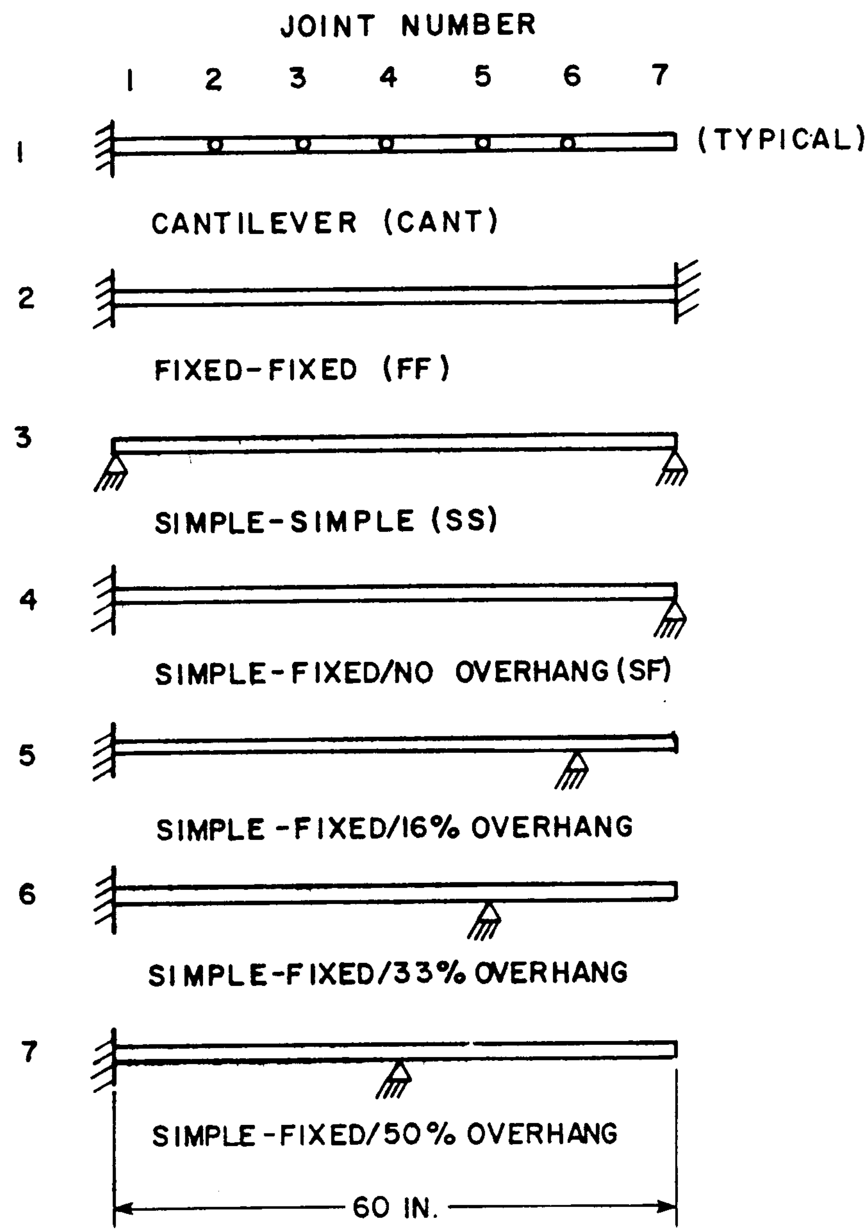
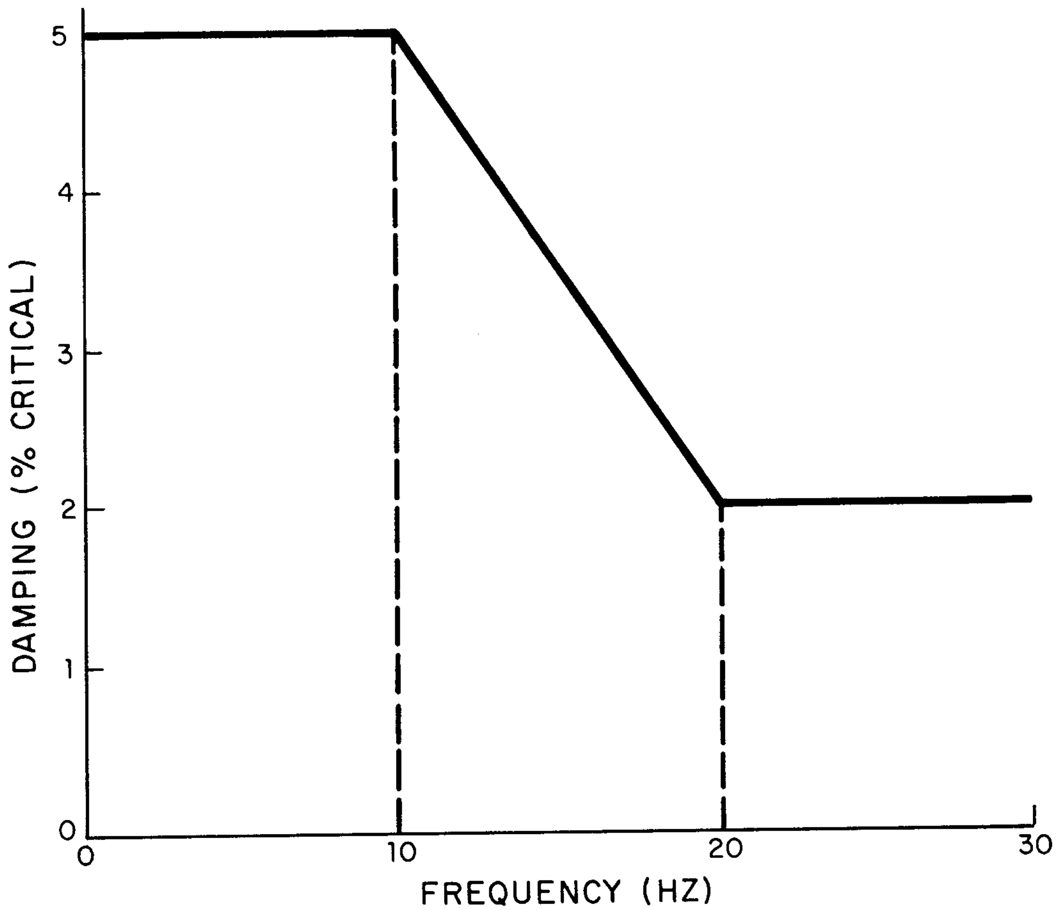


FIGURE 3.7B–71 DAMPING VALUE FOR SEISMIC ANALYSIS OF PIPING



NOTE:
APPLICABLE TO BOTH OBE & SSE, INDEPENDENT OF PIPE DIAMETER

3.7N SEISMIC DESIGN

In addition to the steady state loads imposed on the system under normal operating conditions, the design of the equipment requires that consideration also be given to abnormal loading conditions, such as earthquakes. Seismic loadings are considered for earthquakes of two magnitudes: safe shutdown earthquake (SSE) and operating basis earthquake (OBE). The SSE is defined as the maximum vibratory ground motion at the plant site that can reasonably be predicted from geologic and seismic evidence. The OBE is that earthquake which, considering the local geology and seismology, can be reasonably expected to occur during the plant life.

For the OBE loading condition, the reactor coolant system is designed to be capable of continued safe operation. The design for the SSE is intended to assure:

1. That the integrity of the reactor coolant pressure boundary is not compromised
2. That the capability to shut down the reactor and maintain it in a safe condition is not compromised
3. That the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100 is not compromised

The seismic requirements for safety related instrumentation and electrical equipment are covered in Section 3.10. The safety class definitions, classification lists, operating condition categories, and the methods used for seismic qualification of mechanical equipment are given in Section 3.2.

3.7N.1 SEISMIC INPUT

3.7N.1.1 Design Response Spectra

Refer to Section 3.7B.1.1.

3.7N.1.2 Design Time History

Refer to Section 3.7B.1.2.

3.7N.1.3 Critical Damping Values

The damping values given in Table 3.7N–1 are used in the analysis of Westinghouse equipment. The values for RCS components are based on testing programs, as reported in WCAP-7921-AR, which have been accepted by the staff. This WCAP defines the Westinghouse NSSS position on Regulatory Guide 1.61.

Tests on fuel assembly bundles justified conservative component damping, values of 7 percent for OBE and 10 percent for SSE to be used in the fuel assembly component qualification. Documentation of the fuel assembly tests is found in WCAP-8236 (1973) and WCAP-8288 (1974).

The damping values used in component analysis of CRDM's and their seismic supports were developed through a testing program performed by Westinghouse. The test conducted was on a full size CRDM complete with rod position indicator coils, attachment to a simulated vessel head, and variable gap between the top of the pressure housing support plate and a rigid bumper representing the support.

The program consisted of transient vibration tests in which the CRDM was deflected a specified initial amount and suddenly released. A logarithmic decrement analysis of the decaying transient provides the effective damping of the assembly. The effect on damping of variations in the drive shaft axial position, upper seismic support clearance, and initial deflection amplitude was investigated.

The upper support clearance had the largest affect on the CRDM damping, with the damping increasing with increasing clearance. With an upper clearance of 0.06 inches, the measured damping was approximately 8 percent. The clearances in a typical upper seismic CRDM support are a minimum of 0.10 inches. The increasing damping with increasing clearances trend from the test results, indicated that the damping would be greater than 8 percent for both the OBE and the SSE, based on a comparison between typical deflections during these seismic events and the initial deflections of mechanisms in the test. Component damping values of 5 percent are, therefore, conservative for both the OBE and SSE.

3.7N.1.4 Supporting Media for Seismic Category I Structures

Refer to Section 3.7B.1.4.

3.7N.2 SEISMIC SYSTEM ANALYSIS

Refer to Section 3.7B.2.

3.7N.3 SEISMIC SUBSYSTEM ANALYSIS

This section describes the seismic analysis performed on subsystems within Westinghouse’s scope of responsibility.

3.7N.3.1 Seismic Analysis Methods

Those components that must remain functional in the event of the SSE (Seismic Category I) are identified by applying the criteria of Section 3.2.1.

In general, the dynamic analyses are performed using a modal analysis in combination with a response spectrum analysis.

3.7N.3.1.1 Dynamic Analysis - Mathematical Model

The first step in any dynamic analysis is to model the structure or component, i.e., convert the real structure or component into a system of masses, springs, and dashpots suitable for mathematical analysis. The essence of this step is to select a model so that the displacements obtained are a good representation of the motion of the structure or component. Stated differently, the true inertia forces should not be altered so as to appreciably affect the internal stresses in the structure or component. Some typical modeling techniques are presented in Lin (1974).

Equations of Motion

Consider the multi-degree of freedom system shown in Figure 3.7N–1. Making a force balance on each mass point r, the equations of motion can be written in the form:

$$m_r\ddot{y}_r + \sum^{i} c_{ri}\dot{u}_i + \sum^{i} k_{ri}u_i = 0 \qquad (3.7N\text{-}1)$$

where:

m_r = The value of the mass or mass moment of rotational inertia at mass point r

$\ddot{y}_r$ = Absolute translational or angular acceleration of mass point r

c_{ri} = Damping coefficient - external force or moment required at mass point r to produce a unit translational or angular velocity at mass point i, maintaining zero translational or angular velocity at all other mass points. Force or moment is positive in the direction of positive translational or angular velocity.

$\dot{u}_i$ = Translational or angular velocity of mass point i relative to the base

K_{ri} = Stiffness coefficient - the external force (moment) required at mass point r to produce a unit deflection (rotation) at mass point i, maintaining zero displacement (rotation) at all other mass points Force (moment) is positive in the direction of positive displacement (rotation).

u_i = Displacement (rotation) of mass point i relative to the base

since:

$$\ddot{y} = \ddot{u}_r + \ddot{y}_s \qquad (3.7N\text{-}2)$$

where:

$\ddot{y}_s$ = Absolute translational (angular) acceleration of the base

$\ddot{u}_r$ = Translational (angular) acceleration of mass point r relative to the base

Equation 3.7N-1 can be written as:

$$m_r\ddot{u}_r + \sum^{i} c_{ri}\dot{u}_i + \sum^{i} k_{ri}u_i = -m_r\ddot{y}_s \qquad (3.7N\text{-}3)$$

For a single degree of freedom system with displacement u, mass m, damping c, and stiffness k, the corresponding equation of motion is:

$$m\ddot{u} + c\dot{u} + ku = -m\ddot{y}_s \qquad (3.7N\text{-}4)$$

3.7N.3.1.2 Modal Analysis

Natural Frequencies and Mode Shapes

The first step in the modal analysis method is to establish the normal modes, which were determined by eigen solution of Equation 3.7N-3. The right hand side and the damping term are set equal to zero for this purpose, as illustrated in Biggs (1964) (Pages 83 thru 111). Thus, Equation 3.7N-3 becomes:

$$m_r\ddot{u}_r + \sum^{i} k_{ri}u_i = 0 \qquad (3.7N\text{-}5)$$

The equation given for each mass point r in Equation 3.7N-5 can be written as a system of equations in matrix form as

$$M\{\Delta\} + K\{\Delta\} \qquad (3.7N\text{-}6)$$

where:

M = Mass and rotational inertia matrix

$\{\Delta\}$ = Column matrix of the general displacment and rotation at each mass point relative to the base

K = Square stiffness matrix

$\{\Delta\}$= Column matrix of general translational and angular accelerations at each mass point relative to the base, $d^2\{\Delta\}/dt^2$

Harmonic motion is assumed and the $\{\Delta\}$ is expressed as:

$$\{\Delta\}=\{\delta\} \sin wt \qquad (3.7N\text{-}7)$$

where:

$\{\delta\}$ = Column matrix of the spatial displacement and rotation at each mass point relative to the base

ω = Natural frequency of harmonic motion in radians per second

The displacement function and its second derivative are substituted into Equation 3.7N-6 and yield:

$$[K]\{\delta\} = \omega^2 [M]\{\delta\} \quad (3.7N\text{-}8)$$

The determinant $[K] - \omega^2 [M]$ is set equal to zero and is then solved for the natural frequencies. The associated mode shapes are then obtained from Equation 3.7N-8. This yields n natural frequencies and mode shapes where n equals the number of dynamic degrees of freedom of the system. The mode shapes are all orthogonal to each other and are sometimes referred to as normal mode vibrations. For a single degree of freedom system, the stiffness matrix and mass matrix are single terms and the determinant $[K] - \omega^2 [M]$ when set equal to zero yields simply:

$$k - \omega^2 m = 0$$

$$\omega = \sqrt{\frac{k}{m}} \quad (3.7N\text{-}9)$$

where ω is the natural angular frequency in radians per second.
The natural frequency in cycles per second is therefore:

$$f = \frac{1}{2\pi}\sqrt{\frac{k}{m}} \quad (3.7N\text{-}10)$$

To find the mode shapes, the natural frequency corresponding to a particular mode, ω_n, can be substituted in Equation 3.7N-8.

1. Modal Equations

The response of a structure or component is always some combination of its normal modes. Good accuracy can usually be obtained by using only the first few modes of vibration. In the normal mode method, the mode shapes are used as principal coordinates to reduce the equations of motion to a set of uncoupled differential equations that describe the motion of each mode n. These equations may be written as (Biggs 1964, pages 116 thru 125):

$$\ddot{A} + 2\omega_n p_n \dot{A}_n + \omega_n^2 A_n = -\Gamma_n \ddot{y}_s \quad (3.7N\text{-}11)$$

where the modal displacement or rotation, A_n, is related to the displacement or rotation of mass point r in mode n, u_{rn}, by the equation:

$$u_{rn} = A_n \phi_{rn} \quad (3.7N\text{-}12)$$

where:

ω_n = Natural frequency of mode n in radians per seconds

p_n = Critical damping ratio of mode n

Γ_n = Modal participation factor of mode n given by:

$$\Gamma_n = \frac{\sum_r m_r \phi_{rn}^{1}}{\sum_r m_r \phi_{rn}^{2}} \qquad (3.7N\text{-}13)$$

where:

ϕ'_{rn} = Value of ϕ_{rn} in the direction of the earthquake

The essence of the modal analysis lies in the fact that Equation 3.7N-11 is analogous to the equation of motion for a single degree of freedom system that is developed from Equation 3.7N-4. Dividing Equation 3.7N-4 by m gives:

$$\ddot{u} + \frac{c}{m}\dot{u} + \frac{k}{m}u = -\ddot{y}_s \qquad (3.7N\text{-}14)$$

The critical damping ratio of the single degree of freedom system, p, is defined by the equation:

$$p = \frac{c}{c_c} \qquad (3.7N\text{-}15)$$

where the critical damping coefficient is given by the expression:

$$c_c = 2m\omega \qquad (3.7N\text{-}16)$$

Substituting Equation 3.7N-16 into Equation 3.7N-15 and solving for c/m gives:

$$\frac{c}{m} = 2\omega p \qquad (3.7N\text{-}17)$$

Substituting this expression and the expression for k/m given by Equation 3.7N-9 into Equation 3.7N-14 gives:

$$\ddot{u} + 2\omega p\dot{u} + \omega^2 u = -\ddot{y}_s \qquad (3.7N\text{-}18)$$

Note the similarity of Equations 3.7N-11 and 3.7N-18. Thus, each mode may be analyzed as though it were a single degree of freedom system and all modes are independent of each other. By this method, a fraction of critical damping, i.e., c/c_c, may be assigned to each mode and it is not necessary to identify or evaluate individual damping coefficients, i.e., c. However, assigning only a single damping ratio to each mode is not appropriate for a slightly damped structure supported by a massive moderately damped structure. There are several methods which can be used to incorporate damping in the model.

One method is to develop and analyze separate mathematical models for both structures, using their respective damping values. The massive moderately damped support structure is analyzed first. The calculated response at the support points for the slightly damped structures is used as a forcing function for the subsequent detailed analysis. Another method is to inspect the mode shapes to determine which modes correspond to the slightly damped structure and then use the damping associated with the structure having predominant motion. A third method is to use the Rayleigh damping method based on computed modal energy distribution.

3.7N.3.1.3 Response Spectrum Analysis

The response spectrum is a plot showing the variation in the maximum response (Thomas et al., 1963, pages 24 thru 51) (displacement, velocity, and acceleration) of a single degree of freedom system versus its natural frequency of vibration when subjected to a time history motion at its base.

The response spectrum concept can be best explained by outlining the steps involved in developing a spectrum curve. Determination of a single point on the curve requires that the response (displacement, velocity, and acceleration) of a single degree of freedom system with a given damping and natural frequency is calculated for a given base motion.

The variations in response are established and the maximum absolute value of each is plotted as an ordinate with the natural frequency used as the abscissa. The process is repeated for other assumed values of frequency in sufficient detail to establish the complete curve. Other curves corresponding to different fractions of critical damping are obtained in a similar fashion. Thus, the determination of each point of the curve requires a complete dynamic response analysis; the determination of a complete spectrum may involve hundreds of such analyses. However, once a response spectrum plot is generated for the particular base motion, it may be used to analyze each structure and component with the base motion. The spectral acceleration, velocity, and displacement are related by the equation:

$$S_{a_n} = \omega_n S_{v_n} = \omega_n^2 s_{d_n}$$

Response spectra developed for Millstone 3 are discussed in Section 3.7B.3.

3.7N.3.2 Determination of Number of Earthquake Cycles

Where fatigue analyses of mechanical systems and components are required, Westinghouse specifies in the equipment specification the number of cycles of the operating basis earthquake (OBE) to be considered. The number of cycles for NSSS components is given in Table 3.9-1. The fatigue analyses are performed and presented as part of the components stress report.

3.7N.3.3 Procedure Used for Modeling

Refer to Section 3.7N.3.1 for modeling procedures for subsystems in Westinghouse's scope of responsibility.

3.7N.3.4 Basis for Selection of Frequencies

The analysis of equipment subjected to seismic loading involves several basic steps, the first of which is the establishment of the intensity of the seismic loading. Considering that the seismic input originates at the point of support, the response of the equipment and its associated supports

based upon the mass and stiffness characteristics of the system determine the seismic accelerations which the equipment must withstand.
Three ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible:

1. If the equipment is rigid, relative to the structure, the maximum acceleration of the equipment mass approaches that of the structure at the point of equipment support. The equipment acceleration value in this case corresponds to the low period region of the floor response spectra.

2. If the equipment is very flexible, relative to the structure, the internal distortion of the structure is unimportant and the equipment behaves as though supported on the ground.

3. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

3.7N.3.5 Use of Equivalent Static Load Method of Analysis

The equivalent static load or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components which can be adequately characterized as single degree of freedom systems are considered to have a modal participation factor of one. Seismic acceleration coefficients for multi-degree of freedom systems, which may be in the resonance region of the amplified response spectra curves, are increased by 50 percent to account conservatively for the increased modal participation where the equivalent static load method is used.

3.7N.3.6 Three Components of Earthquake Motion

The seismic design of the RCS equipment includes the effect of the seismic response of the supports, equipment, structures, and components. Floor response spectra are generated for two perpendicular horizontal directions (i.e., N-S, E-W) and the vertical direction. The equipment response is determined using horizontal and vertical spectra which envelope the appropriate floor response spectra. The total seismic response is obtained by combining the unidirectional responses in the two horizontal directions and the vertical direction using the square root of the sum of the squares method.
Time history analysis was not used on Millstone 3.

3.7N.3.7 Combination of Modal Responses

The total unidirectional seismic response is obtained by combining the individual modal responses utilizing the square root of the sum of the squares method. For systems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. The groups of closely spaced modes are chosen such that the difference between the frequencies of the first mode and the last mode in the group does not exceed 10 percent of the lower frequency. Groups are formed starting from the lowest frequency and working towards successively higher frequencies. No one frequency is in more than one group. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root of the sum of the squares of all modes the product of the responses of the modes in each group of closely spaced modes and a coupling factor. This can be represented mathematically as:

$$R_T^2 = \sum_{i=1}^{N} R_i^2 + 2\sum_{j=1}^{S} \sum_{K=M_j}^{N_j-1} \sum_{l=K+1}^{N_j} R_K R_l E_{K_l} \qquad (3.7N\text{-}20)$$

where:

R_T = Total unidirectional response

R_i = Absolute value of response of mode i

N = Total number of modes considered

S = Number of groups of closely spaced modes

M_j = Lowest modal number associated with group j of closely spaced modes

N_j = Highest modal number associated with group j of closely spaced modes

ε_{Kl} = Coupling factor with

$$\varepsilon_{Kl} = \left\{1 + \left[\frac{\omega'_K - \omega'_l}{(\beta'_K\omega_K + \beta_l\omega_l)}\right]^2\right\}^{-1} \qquad (3.7N\text{-}21)$$

and

$$\omega'_K = \omega_K[1 - (\beta'_K)^2]^{1/2} \qquad (3.7N\text{-}22$$

$$\beta'_K = \beta_K + \frac{2}{\omega_K t_d} \qquad (3.7N\text{-}23)$$

where:

ω_K = Frequency of closely spaced mode K

β_K = Fraction of critical damping in closely spaced mode K

t_d = Duration of the earthquake

3.7N.3.8 Analytical Procedures for Piping

Refer to Section 3.7B.3.9.

3.7N.3.9 Multiply Supported Equipment Components with Distinct Inputs

When response spectrum methods are used to evaluate reactor coolant system primary components interconnected between floors, the procedures of the following paragraphs are used. The primary components of the reactor coolant system are supported at no more than two floor elevations.

A dynamic response spectrum analysis is first made assuming no relative displacement between support points. The response spectra used in this analysis are the most severe floor response spectra.

Secondly, the effect of differential seismic movement of components interconnected between floors is considered statically in the detailed component analysis. The differential motion is evaluated as a free end displacement in accordance with ASME III, NB-3213.19.

The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses.

3.7N.3.10 Use of Constant Vertical Static Factors

Constant vertical load factors are not used as the vertical floor response load for the seismic design of safety related components and equipment within Westinghouse’s scope of responsibility.

3.7N.3.11 Torsional Effects of Eccentric Masses

Refer to Section 3.7B.3.11.

3.7N.3.12 Buried Seismic Category I Piping Systems and Tunnels

Refer to Section 3.7B.3.12.

3.7N.3.13 Interaction of Other Piping with Seismic Category I Piping

Refer to Section 3.7B.3.13.

3.7N.3.14 Seismic Analyses for Reactor Internals

Fuel assembly component stresses induced by horizontal seismic disturbances are analyzed through the use of finite element computer modeling.

The time history floor response, based on a standard seismic time history normalized to SSE levels, is used as the seismic input. The reactor internals and the fuel assemblies are modeled as spring and lumped mass systems or beam elements. The component seismic response of the fuel assemblies is analyzed to determine design adequacy. A detailed discussion of the analyses performed for typical fuel assemblies is contained in WCAP-8236 (1973), WCAP-8288 (1974); and Lin (1974), ASME Paper 74-NE-7.

Fuel assembly lateral structural damping obtained experimentally is presented in WCAP-8236 Addendum 1 (1974) and WCAP-8288 Addendum 1 (1974) (Figure 3-4). The data indicate that no damping values less than 10 percent were obtained for fuel assembly displacements greater than 0.11 inches.

The distribution of fuel assembly amplitudes decreases as one approaches the center of the core. The average amplitude for the minimum displacement fuel assembly is well above 0.11 inches for the SSE.

Fuel assembly displacement time history for the SSE seismic input is illustrated in WCAP-8236 Addendum 1 (1974) and WCAP-8288 Addendum 1 (1974) (Figure 2-3).

The CRDM's are seismically analyzed to confirm that system stresses under the combined loading conditions, as described in Section 3.9N.1, do not exceed allowable levels as defined by the ASME Code, Section III. The CRDM is mathematically modeled as a system of lumped and distributed masses. The model is analyzed under appropriate seismic excitation and the resultant seismic bending moments along the length of the CRDM are calculated. The corresponding stresses are then combined with the stresses from the other loadings required and the combination is shown to meet ASME Code, Section III requirements.

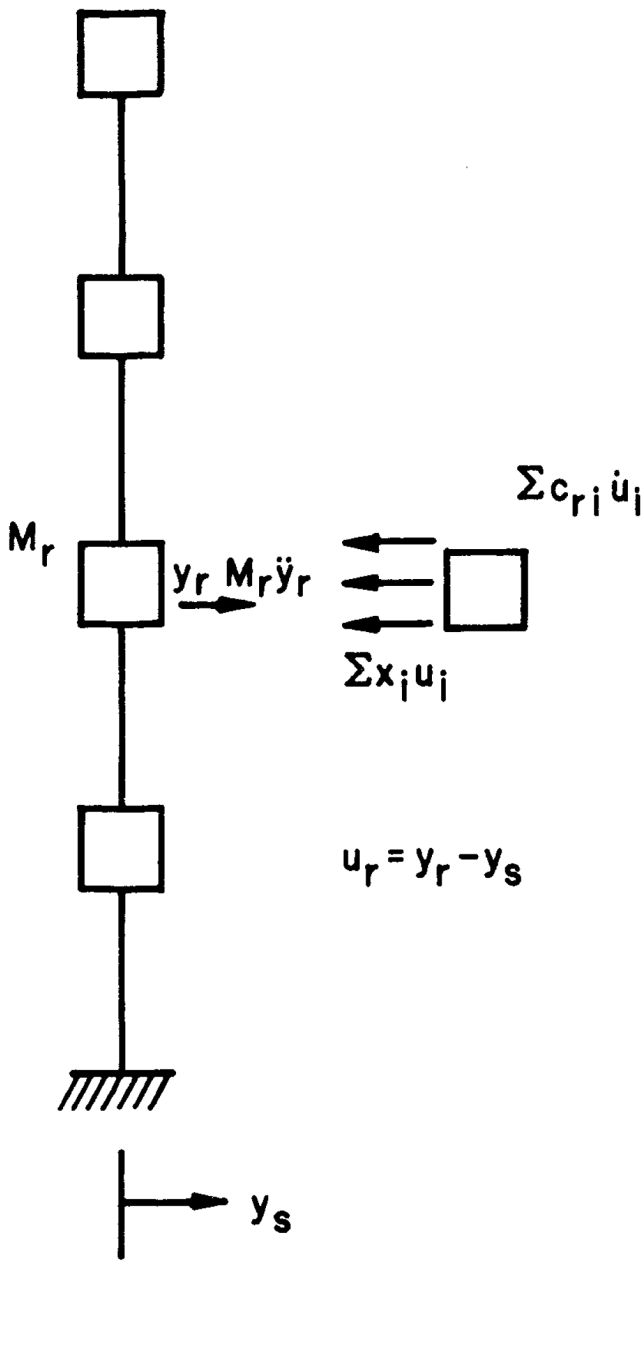
3.7N.3.15 Analysis Procedure for Damping

The damping values and procedures used for Westinghouse scope of supply and analysis are discussed in Sections 3.7N.1.3 and 3.7N.3.1.

3.7N.4 SEISMIC INSTRUMENTATION

Refer to Section 3.7.4.

FIGURE 3.7N–1 MULTI-DEGREE OF FREEDOM SYSTEM



3.7.4 SEISMIC INSTRUMENTATION

3.7.4.1 Comparison with Regulatory Guide 1.12

Millstone 3 complies with Regulatory Guide 1.12 (Section 1.8) with the following exceptions.
ANSI/ANS 2.2 is used instead of ANSI N18.5 referenced by the Regulatory Guide, since ANSI/ANS 2.2 is the final issued version of proposed standard ANSI N18.5.
Only solid state digital instrumentation that will enable the processing of data at the plant site within 4 hours of the seismic event is used. No peak recording instrumentation is used.
The sensor required by section C.1.c.(3) is mounted on the wall of the emergency generator enclosure approximately 3 feet above the mat near a corner to reduce the potential for equipment damage due to flooding. This location will provide representative indication based on the stiffness of the supporting structure.
Regulatory Position C.3 of the guide is not applicable because the safe shutdown earthquake for Millstone 3 is less than 0.3g (Section 2.5.2.6).

3.7.4.2 Location and Description of Instrumentation

Table 3.7–1 lists the location of the seismic instrumentation. Seismic instrumentation is located in areas where it can be serviced during periods of unit shutdown. All instruments are oriented to the same azimuths used in the mathematical model to permit direct use of the data within the model. All accelerometers, recorders, and central controllers have instrument characteristics that meet or exceed the requirements in Section 5 of ANSI/ANS 2.2-1978.

Triaxial Accelerograph (five provided)

Five triaxial time-history accelerographs capable of measuring and permanently recording absolute acceleration versus time are provided. The instrumentation consists of local triaxial force balance accelerometers, solid state recorders, and a central controller located in the control room. Four accelerometers input to the two centrally located solid state dual panel recorders. The fifth accelerometer inputs to a local solid state recorder located in the Auxiliary Building.
Accelerometers are installed at two locations on the outside of the containment structure in the engineering safety features building. One accelerometer is mounted on the base mat of the containment at elevation (-) 24 feet - 3 inches and the other directly above the first on the containment shell at elevation 40 feet - 6 inches.
The third accelerometer is installed on the wall of the emergency generator enclosure at elevation 4 feet - 6 inches. The vault is part of the emergency generator enclosure building. This location is chosen because the emergency generator enclosure building has dynamic response characteristics different from that of the containment structure. Table 2.5.4–14 describes the differing founding conditions.
The fourth accelerometer is located at elevation 46 feet - 6 inches near the charging pump surge tank located in the Auxiliary Building. This accelerometer replaces the response spectrum recorder required at this location by Regulatory Guide 1.12 revision 1 since alternate triaxial instrumentation may be provided at this location in accordance with Part 1b of Section 4.1.6 of ANSI/ANS 2.2-1978.

The fifth accelerometer is located on the elevation 51 foot - 4 inch slab of the containment structure internal adjacent to a steam generator support. This accelerometer inputs to the local solid state recorder located in the Auxiliary Building.

Solid State Recorders

Two solid state dual panel recorders capable of recording motion in all three axes are provided. These recorders are centrally located in the control room and continuously monitor the input from the first four accelerometers described above. Each axis has an independently adjustable trigger. If any one axis is triggered all three axes of all four sensors are recorded. Event data is stored on flash storage cards for later analysis. Each solid state recorder has its own battery to ensure the recorder can record events after a loss of AC power. The internal battery provides approximately 30 hours of power autonomy. A main control board alarm annunciates via the alarm relay panel if the containment mat accelerometer is triggered. The alarm relay panel is not powered from the UPS such that the main control board alarm will not annunciate following a loss of external power. A main control board alarm annunciates on a loss of external power.

The local solid state recorder located in the Auxiliary Building records the data from the accelerometer located on the elevation 51 foot - 4 inch slab of the containment structure internal adjacent to a steam generator support. The local solid state recorder is similar to the solid state dual panel recorders described above except it is stand alone so it will only record data when one of its monitored axes exceeds the trigger and it will not trigger the other recorders. Operations will verify power available on a daily basis and the internal battery will ensure the recorder can record events after a loss of AC power. The internal battery provides approximately 30 hours of power autonomy. Data collected by the local solid state recorder may be evaluated using the central controller to determine the response spectrum.

Central Controller

The central controller automatically retrieves data from the two dual panel solid state recorders shortly after the recorders are triggered and determines the response spectrum of the event. If the response spectrum recorded at the containment mat exceeds the OBE criteria an LED is lit on the alarm relay panel and the central controller display will indicate “OBE”. The central controller is powered from a UPS such that data can be analyzed after a loss of power. The UPS is sized to power the central controller for more than 25 minutes. The alarm relay panel is not powered from the UPS and will not indicate OBE exceedance following a loss of power. A main control board alarm annunciates on a loss of external power.

The central controller may be used to determine the response spectrum of data recorded off the local solid state recorder.

3.7.4.3 Control Room Operator Notification

The recorders located in the control room and the local solid state recorder located in the Auxiliary Building record the signals generated by the accelerometers on flash memory cards. If the signals from any axis of any of the first four sensors exceed the trigger value all axes of all four sensor record. If the signals from any axis of the sensor mounted on the containment steam

generator support exceed the trigger value all axes of this sensor record on the local solid state recorder. Trigger values will be set as required by ANSI/ANS 2.2-1978.
If any of the first four sensors trigger or if power is lost to the seismic monitoring system (except the local solid state recorder) a main control board "seismic monitoring warning/trouble" alarm annunciates.
If the containment mat accelerometer triggers, a main control board alarm will annunciate via the alarm relay panel. The alarm relay panel is not powered from the UPS such that the main control board will not annunciate following a loss of external power.
The central controller will automatically retrieve the data from the two solid state dual panel recorders and determine the response spectrum. If the data from the containment mat exceeds OBE criteria a LED is lit on the alarm relay panel and the central controller display will indicate "OBE". Data from the local solid state recorder must be collected and brought to the central controller or another personal computer with correct software installed to determine the response spectrum.
The central controller will continue to operate on UPS supply for approximately 25 minutes following a loss of external power. The operator may determine OBE exceedance from the central controller during this time. The alarm relay panel is not powered from the UPS and will not indicate OBE exceedance following a loss of power.

3.7.4.4 Comparison of Measured and Predicted Responses

The criteria and procedures used to compare recorded data obtained from seismic instrumentation to plant design parameters are based on ANSI/ANS 2.10-1979 (Section 3.7.5). Instrumentation requirements are in accordance with ANSI/ANS 2.2-1978 and the supplemental provisions of Regulatory Guide 1.12, Revision 1 (Section 3.7.4.1).

3.7.5 REFERENCES FOR SECTION 3.7

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TABLE 3.7–1 SEISMIC INSTRUMENTATION

Instrument Location	Triaxial Force Balance Accelerometer	Main Control Board Annunciation
Containment Structure - Outside		
Mat, elevation (-) 24 feet 3 inches	X (1)	X(2)
Containment shell, elevation 40 feet 6 inches	X (2)	
Containment Structure - Inside		
Steam Generator Support, elevation 51 feet 4 inches	X (3)	
Emergency Generator Enclosure Located on Mat in Diesel Fuel Oil Vault, elevation 4 feet 6 inches	X (1) (4)	
Auxiliary Building Surge Tank Support, elevation 46 feet 6 inches	X (1)	

NOTES:

(1) All four sensors record event sensed by any axis of any of the four sensors.
(2) Main control board annunciation of OBE exceedance on containment mat sensor only.
(3) Data must be collected from Etna recorder in Auxiliary Building and may be evaluated on central controller.
(4) This sensor is mounted to the wall approximately 3 feet above the mat near a corner to reduce the potential for equipment damage due to flooding. This location will provide representative indication based on the stiffness of the supporting structure.

TABLE 3.7B–1 DAMPING FACTORS

	Stress Level		Type of Condition of Structure, System or Component	Percent of Critical Damping
1.	Low stress, well below proportional limit. Stresses below 0.25 yield point stress		Steel, reinforced concrete; no cracking and no slipping at joints, piping or components	0.5 to [(1)]
2.	Working stress limited to 0.5 yield point stress	a.	Welded steel, well reinforced concrete (with only slight cracking)	2
		b.	Bolted steel	5
3.	At or just below yield point	a.	Welded steel	5
		b.	Reinforced concrete	5
		c.	Bolted steel	7
4.	At all stress levels	a.	Rock (translation)	10
		b.	Rock (rotation)	5

NOTE:

1. For final reconciliation of pipe stress analysis or piping system backfits, damping values as defined in ASME Code Case N-411 (Figure 3.7B–71) may be utilized for both OBE and SSE.

TABLE 3.7B–2 METHODS OF SEISMIC ANALYSIS USED FOR SEISMIC CATEGORY I STRUCTURES

Structure	Response Spectrum Analysis	Modal Time History Analysis	Frequency Domain Time History Analysis
Reactor containment and internals	X	X	
Main steam valve building	X	X	
Hydrogen recombiner building	X	X	
Safeguards area	X	X	
Auxiliary building	X	X	
Fuel building	X	X	
Control building			X
Service building	X	X	
Emergency generator enclosure			X
Service water pumphouse	X	X	
Refueling water storage and chemical addition tanks	X	X	
Demineralized water storage tank and enclosure	X	X	
Fuel building canopy	X	X	
Turbine building	X	X	
Waste disposal building	X		

TABLE 3.7B–3 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODAL FREQUENCIES AND PARTICIPATION FACTORS UNCRACKED MODEL

Mode	Frequency (Hz)	Participation Factors		
		Horizontal		Vertical
		N-S	E-W	
1	4.664	33.88	424.20	4.67
2	4.751	353.06	-30.44	-3.28
3	4.854	3.98	37.96	-4.09
4	5.565	252.24	-42.17	-32.00
5	6.135	53.40	86.74	-1.22
6	8.244	0.11	0.89	0.10
7	10.917	3.42	-177.61	3.11
8	10.921	170.86	3.10	8.60
9	11.809	13.83	-99.23	17.97
10	12.630	78.31	-58.48	399.46
11	12.881	10.41	-117.44	-228.28
12	13.295	157.36	61.05	-62.87
13	13.607	8.36	8.53	0.17
14	15.592	42.13	4.40	-241.13
15	18.751	44.54	-110.37	-0.04
16	18.755	108.26	43.75	1.45
17	22.152	2.60	-0.14	-1.10
18	22.157	0.48	0.95	-0.06

TABLE 3.7B–4 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODAL FREQUENCIES AND PARTICIPATION FACTORS CRACKED MODEL

Mode	Frequency (Hz)	Participation Factors		
		Horizontal Earthquake		Vertical Earthquake
		N-S	E-W	
1	3.4236	14.97	317.58	0.22
2	3.4239	315.07	-14.92	-0.25
3	4.750	22.93	277.82	6.24
4	5.544	295.89	-48.38	-32.19
5	5.863	9.33	10.31	-0.51
6	6.134	57.44	95.54	-1.06
7	7.761	15.69	155.83	-0.35
8	7.762	153.44	-15.40	2.02
9	9.371	4.21	-1.29	376.49
10	9.695	0.25	2.02	-0.03
11	11.801	15.39	-118.13	10.09
12	12.758	87.27	-136.84	79.36
13	13.135	190.02	98.59	94.36
14	13.375	37.78	-72.22	-17.73
15	13.428	4.91	3.73	-63.01
16	15.388	44.63	4.92	-324.04
17	15.763	0.88	1.23	-0.20
18	15.766	2.09	0.49	-29.84

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL

Mode	Joint	Displacement		
		X	Y	Z
1	1	0.001	0.0	0.016
	2	0.012	0.006	0.153
	3	0.023	0.009	0.336
	4	0.016	0.005	0.338
	5	0.021	0.003	0.418
	6	0.043	0.018	0.442
	7	0.054	0.013	0.646
	8	0.075	0.011	0.916
	9	0.078	0.007	1.0
	10	0.008	0.0	0.098
	11	0.016	0.0	0.20
	12	0.025	0.0	0.310
	13	0.035	0.0	0.425
	14	0.044	0.0	0.536
	15	0.051	0.0	0.621
	16	0.063	0.0	0.776
	17	0.079	0.0	0.972
2	1	0.009	0.0	0.0
	2	0.031	0.005	-0.005
	3	0.051	-0.006	-0.005
	4	0.060	0.0	-0.007
	5	0.067	-0.005	-0.006
	6	0.074	-0.003	-0.007
	7	0.102	-0.007	-0.009
	8	0.143	-0.013	-0.011
	9	0.166	-0.006	-0.014

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	10	0.093	0.0	-0.008
	11	0.198	0.0	-0.017
	12	0.312	0.0	-0.027
	13	0.430	0.0	-0.037
	14	0.545	0.0	-0.047
	15	0.633	0.0	-0.054
	16	0.795	0.0	-0.068
	17	1.0	0.0	-0.086
3	1	0.0	0.0	0.001
	2	-0.010	-0.006	-0.104
	3	-0.021	-0.007	-0.264
	4	-0.014	-0.004	-0.257
	5	-0.017	-0.002	-0.326
	6	-0.038	-0.014	-0.342
	7	-0.048	-0.010	-0.508
	8	-0.066	-0.009	-0.722
	9	-0.066	-0.006	-0.776
	10	0.008	0.0	0.083
	11	0.017	0.0	0.188
	12	0.028	0.0	0.303
	13	0.038	0.0	0.421
	14	0.049	0.0	0.538
	15	0.057	0.0	0.626
	16	0.072	0.0	0.790
	17	0.091	0.0	1.0
4	1	0.009	0.0	-0.002
	2	0.132	0.007	-0.040

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	3	0.270	-0.053	-0.039
	4	0.327	-0.016	-0.063
	5	0.370	-0.048	-0.057
	6	0.417	-0.038	-0.070
	7	0.590	-0.062	-0.093
	8	0.858	-0.094	-0.142
	9	1.0	-0.053	-0.191
	10	-0.002	-0.001	0.0
	11	-0.020	-0.001	0.003
	12	-0.040	-0.001	0.007
	13	-0.062	-0.001	0.010
	14	-0.082	-0.001	0.014
	15	-0.098	-0.001	0.016
	16	-0.129	-0.001	0.022
	17	-0.171	-0.001	0.028
5	1	0.004	0.0	0.006
	2	0.010	0.012	0.107
	3	0.052	-0.006	-0.010
	4	0.104	0.002	0.147
	5	0.121	-0.012	0.10
	6	0.094	0.010	0.173
	7	0.182	-0.004	0.290
	8	0.326	-0.018	0.640
	9	0.437	-0.008	1.0
	10	0.001	0.0	0.002
	11	-0.003	0.0	-0.005
	12	-0.007	0.0	-0.013

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	13	-0.013	0.0	-0.022
	14	-0.018	0.0	-0.031
	15	-0.022	0.0	-0.038
	16	-0.030	0.0	-0.052
	17	-0.041	0.0	-0.071
6	1	0.0	0.0	-0.004
	2	-0.081	0.007	-0.285
	3	-0.036	0.006	-0.300
	4	-0.024	0.005	-0.240
	5	-0.020	0.001	-0.235
	6	-0.030	0.017	-0.193
	7	-0.005	0.011	-0.061
	8	0.089	0.006	0.582
	9	0.166	0.005	1.0
	10	0.0	0.0	-0.004
	11	-0.001	0.0	-0.005
	12	-0.001	0.001	-0.005
	13	-0.001	0.001	-0.005
	14	-0.001	0.001	-0.005
	15	-0.001	0.001	-0.005
	16	-0.001	0.001	-0.004
	17	-0.001	0.001	-0.004
7	1	0.001	0.0	-0.028
	2	0.005	0.001	-0.058
	3	0.005	0.004	-0.055
	4	0.003	0.002	-0.059
	5	0.002	0.003	-0.051

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	6	0.003	0.005	-0.049
	7	-0.001	0.006	-0.019
	8	-0.007	0.007	0.076
	9	-0.011	0.004	0.116
	10	0.004	0.001	-0.235
	11	0.008	0.001	-0.429
	12	0.010	0.001	-0.552
	13	0.011	0.001	-0.583
	14	0.009	0.001	-0.516
	15	0.007	0.001	-0.391
	16	-0.001	0.001	0.084
	17	-0.018	0.001	1.0
8	1	-0.027	-0.001	0.0
	2	-0.036	-0.002	0.0
	3	-0.038	-0.012	0.0
	4	-0.034	-0.007	0.0
	5	-0.030	-0.012	0.0
	6	-0.025	-0.010	0.0
	7	-0.002	-0.015	0.0
	8	0.049	-0.021	-0.001
	9	0.076	-0.014	-0.001
	10	-0.234	-0.001	-0.004
	11	-0.429	-0.002	-0.008
	12	-0.552	-0.002	-0.010
	13	-0.584	-0.003	-0.011
	14	-0.517	-0.003	-0.010
	15	-0.393	-0.003	-0.007

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL (CONTINUED)

Mode	Joint	Displacement X	Displacement Y	Displacement Z
	16	0.083	-0.003	0.002
	17	1.0	-0.004	0.019
9	1	0.003	0.003	-0.024
	2	0.090	-0.002	-0.389
	3	0.110	0.025	-0.337
	4	0.065	0.010	-0.496
	5	0.051	0.027	-0.423
	6	0.083	0.022	-0.481
	7	-0.012	0.038	-0.412
	8	-0.107	0.057	0.658
	9	-0.165	0.029	1.0
	10	0.0	0.006	0.002
	11	-0.003	0.008	0.033
	12	-0.005	0.011	0.058
	13	-0.007	0.013	0.073
	14	-0.007	0.014	0.075
	15	-0.005	0.015	0.065
	16	0.001	0.016	0.011
	17	0.016	0.017	-0.104
10	1	0.026	0.102	-0.020
	2	0.253	0.198	-0.273
	3	0.408	0.350	-0.411
	4	0.418	0.314	-0.331
	5	0.396	0.365	-0.327
	6	0.355	0.388	-0.267
	7	0.193	0.459	-0.001

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	8	-0.315	0.551	0.227
	9	-0.603	0.482	0.441
	10	0.013	0.252	-0.009
	11	-0.006	0.412	0.003
	12	-0.023	0.556	0.015
	13	-0.036	0.681	0.023
	14	-0.041	0.780	0.027
	15	-0.039	0.837	0.025
	16	-0.012	0.913	0.005
	17	0.054	1.0	-0.045
11	1	0.004	-0.060	-0.041
	2	0.058	-0.077	-0.542
	3	0.077	-0.070	-0.784
	4	0.091	-0.10	-0.670
	5	0.086	-0.095	-0.653
	6	0.061	-0.054	-0.557
	7	0.035	-0.065	-0.056
	8	-0.157	-0.069	0.555
	9	-0.259	-0.116	1.0
	10	0.002	-0.183	-0.023
	11	0.0	-0.314	0.001
	12	-0.002	-0.433	0.024
	13	-0.003	-0.536	0.042
	14	-0.004	-0.618	0.051
	15	-0.004	-0.666	0.049
	16	0.0	-0.729	0.013
	17	0.008	-0.801	-0.080

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
12	1	-0.043	0.013	-0.016
	2	-0.368	-0.010	-0.191
	3	-0.615	-0.136	-0.250
	4	-0.609	-0.087	-0.238
	5	-0.575	-0.147	-0.229
	6	-0.536	-0.118	-0.210
	7	-0.255	-0.189	-0.065
	8	0.553	-0.272	0.211
	9	1.0	-0.181	0.361
	10	-0.028	0.079	-0.011
	11	-0.006	0.150	-0.002
	12	0.018	0.214	0.007
	13	0.037	0.270	0.015
	14	0.049	0.316	0.019
	15	0.049	0.342	0.019
	16	0.019	0.377	0.007
	17	-0.064	0.417	-0.025
13	1	-0.041	0.0	-0.042
	2	-0.255	-0.047	-0.364
	3	-0.529	-0.168	-0.489
	4	-0.549	-0.133	-0.566
	5	-0.525	-0.184	-0.531
	6	-0.472	-0.159	-0.545
	7	-0.257	-0.225	-0.324
	8	0.448	-0.305	0.636
	9	0.831	-0.240	1.0
	10	-0.030	0.046	-0.031

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	11	-0.010	0.096	-0.010
	12	0.012	0.141	0.014
	13	0.032	0.181	0.034
	14	0.044	0.213	0.047
	15	0.046	0.232	0.049
	16	0.021	0.257	0.023
	17	-0.052	0.286	-0.051
14	1	-0.017	0.073	-0.002
	2	-0.142	0.348	0.005
	3	-0.246	0.543	0.035
	4	-0.251	0.599	0.022
	5	-0.243	0.627	0.022
	6	-0.228	0.659	0.013
	7	-0.10	0.767	-0.032
	8	0.543	0.909	-0.091
	9	0.854	1.0	-0.128
	10	-0.017	0.010	-0.002
	11	-0.015	-0.061	-0.001
	12	-0.008	-0.129	0.0
	13	0.001	-0.190	0.001
	14	0.010	-0.240	0.002
	15	0.015	-0.269	0.003
	16	0.007	-0.309	0.003
	17	-0.027	-0.357	0.002
15	1	-0.034	0.0	0.084
	2	-0.012	0.002	0.019
	3	0.009	-0.001	-0.022

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	4	0.014	0.0	-0.029
	5	0.017	0.0	-0.036
	6	0.018	0.0	-0.039
	7	0.019	0.001	-0.036
	8	-0.026	0.0	0.045
	9	-0.048	-0.005	0.086
	10	-0.263	0.0	0.650
	11	-0.404	0.0	1.0
	12	-0.367	0.0	0.909
	13	-0.169	0.0	0.419
	14	0.105	0.0	-0.259
	15	0.297	0.0	-0.736
	16	0.329	0.0	-0.815
	17	-0.194	0.0	0.481
16	1	0.082	0.001	0.033
	2	0.029	-0.002	0.005
	3	-0.030	0.003	-0.011
	4	-0.043	0.003	-0.014
	5	-0.049	0.004	-0.018
	6	-0.053	0.004	-0.019
	7	-0.051	0.004	-0.017
	8	0.070	0.007	0.020
	9	0.129	0.016	0.037
	10	0.649	0.001	0.262
	11	1.0	-0.0	0.404
	12	0.910	-0.0	0.368
	13	0.420	-0.001	0.170

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	14	-0.259	-0.001	-0.104
	15	-0.736	-0.001	-0.297
	16	-0.815	-0.001	-0.329
	17	0.481	-0.002	-0.195
17	1	0.002	-0.001	0.0
	2	-0.002	-0.017	-0.001
	3	0.004	-0.001	-0.001
	4	0.002	-0.009	0.0
	5	-0.002	-0.002	0.001
	6	-0.005	-0.004	0.002
	7	-0.012	0.001	0.002
	8	0.060	0.010	-0.009
	9	0.075	0.007	-0.011
	10	-0.059	0.0	0.010
	11	-0.155	0.0	0.027
	12	-0.230	0.0	0.039
	13	-0.230	0.0	0.039
	14	-0.141	0.0	0.024
	15	-0.020	0.0	0.004
	16	0.234	0.001	-0.040
	17	1.0	0.001	-0.172
18	1	0.0	0.0	0.001
	2	-0.002	0.001	-0.004
	3	0.0	0.0	-0.008
	4	0.001	0.0	-0.006
	5	0.001	0.001	-0.006
	6	0.0	-0.001	-0.004

TABLE 3.7B–5 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) UNCRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	7	0.002	0.0	0.014
	8	0.005	0.0	0.024
	9	0.004	0.0	0.020
	10	-0.010	0.0	-0.059
	11	-0.027	0.0	-0.154
	12	-0.039	0.0	-0.228
	13	-0.040	0.0	-0.229
	14	-0.024	0.0	-0.141
	15	-0.004	0.0	-0.021
	16	0.040	0.0	0.233
	17	0.172	0.0	1.0

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL

Mode	Joint	Displacement		
		X	Y	Z
1	1	0.0	0.0	0.004
	2	0.0	0.0	0.010
	3	0.001	0.0	0.016
	4	0.001	0.0	0.017
	5	0.001	0.0	0.019
	6	0.001	0.001	0.020
	7	0.001	0.0	0.028
	8	0.002	0.0	0.038
	9	0.002	0.0	0.043
	10	0.004	0.0	0.086
	11	0.009	0.0	0.191
	12	0.014	0.0	0.306
	13	0.020	0.0	0.424
	14	0.026	0.0	0.540
	15	0.030	0.0	0.628
	16	0.037	0.0	0.791
	17	0.047	0.0	1.0
2	1	0.004	0.0	0.0
	2	0.009	0.002	-0.001
	3	0.013	-0.001	-0.001
	4	0.014	0.001	-0.001
	5	0.016	0.0	-0.001
	6	0.017	0.0	-0.001
	7	0.023	-0.001	-0.001
	8	0.030	-0.002	-0.001

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	9	0.035	-0.001	-0.002
	10	0.086	0.0	-0.004
	11	0.190	0.0	-0.009
	12	0.305	0.0	-0.014
	13	0.424	0.0	-0.020
	14	0.540	0.0	-0.026
	15	0.628	0.0	-0.030
	16	0.791	0.0	-0.037
	17	1.0	0.0	-0.047
3	1	0.001	0.0	0.008
	2	0.012	0.007	0.145
	3	0.025	0.010	0.338
	4	0.017	0.005	0.335
	5	0.021	0.003	0.419
	6	0.045	0.018	0.441
	7	0.056	0.013	0.650
	8	0.079	0.011	0.922
	9	0.081	0.008	1.0
	10	0.001	0.0	0.006
	11	0.0	0.0	0.001
	12	-0.001	0.0	-0.006
	13	-0.001	0.0	-0.014
	14	-0.002	0.0	-0.022
	15	-0.002	0.0	-0.028
	16	-0.004	0.0	-0.041
	17	-0.005	0.0	-0.060

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
4	1	0.010	-0.001	-0.002
	2	0.134	0.008	-0.040
	3	0.271	-0.053	-0.038
	4	0.328	-0.015	-0.062
	5	0.371	-0.047	-0.057
	6	0.416	-0.038	-0.069
	7	0.590	-0.061	-0.091
	8	0.858	-0.094	-0.138
	9	1.0	-0.052	-0.186
	10	0.011	-0.001	-0.002
	11	0.007	-0.001	-0.001
	12	0.002	-0.001	0.0
	13	-0.005	-0.001	0.001
	14	-0.013	-0.001	0.002
	15	-0.019	-0.001	0.003
	16	-0.033	-0.001	0.005
	17	-0.054	-0.001	0.009
5	1	0.006	0.0	0.007
	2	0.052	0.014	0.151
	3	0.119	-0.017	0.043
	4	0.184	-0.001	0.195
	5	0.211	-0.022	0.148
	6	0.195	0.002	0.224
	7	0.322	-0.018	0.345
	8	0.521	-0.039	0.666

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	9	0.659	-0.019	1.0
	10	0.007	0.0	0.008
	11	0.006	0.0	0.006
	12	0.003	0.0	0.003
	13	-0.001	0.0	-0.002
	14	-0.005	0.0	-0.007
	15	-0.009	0.0	-0.011
	16	-0.018	0.0	-0.021
	17	-0.032	0.0	-0.037
6	1	0.004	0.0	0.006
	2	0.010	0.012	0.107
	3	0.050	-0.006	-0.011
	4	0.102	0.002	0.146
	5	0.119	-0.011	0.094
	6	0.092	0.010	0.172
	7	0.178	-0.004	0.288
	8	0.320	-0.018	0.638
	9	0.430	-0.007	1.0
	10	0.005	0.0	0.008
	11	0.004	0.0	0.007
	12	0.003	0.0	0.004
	13	0.0	0.0	0.0
	14	-0.002	0.0	-0.004
	15	-0.004	0.0	-0.008
	16	-0.010	0.0	-0.018
	17	-0.020	0.0	-0.034

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
7	1	-0.001	0.0	-0.013
	2	-0.001	0.0	-0.011
	3	-0.001	0.0	-0.007
	4	-0.001	0.0	-0.005
	5	0.0	0.001	-0.003
	6	0.0	0.001	
	7	0.001	0.001	0.008
	8	0.003	0.0	0.026
	9	0.004	0.0	0.036
	10	-0.023	0.0	-0.222
	11	-0.042	0.0	-0.422
	12	-0.055	0.0	-0.552
	13	-0.059	0.0	-0.589
	14	-0.053	0.0	-0.526
	15	-0.041	0.0	-0.403
	16	0.007	0.0	0.074
	17	0.101	0.0	1.0
8	1	-0.012	0.0	0.001
	2	-0.009	0.0	0.0
	3	-0.004	-0.003	0.0
	4	-0.001	-0.002	0.0
	5	0.001	-0.003	-0.001
	6	0.004	-0.003	-0.001
	7	0.013	-0.004	-0.001
	8	0.030	-0.006	-0.002

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	9	0.039	-0.004	-0.002
	10	-0.222	0.0	0.022
	11	-0.422	0.0	0.042
	12	-0.552	0.0	0.056
	13	-0.589	0.0	0.059
	14	-0.526	0.0	0.053
	15	-0.403	0.0	0.041
	16	0.074	0.0	-0.007
	17	1.0	0.0	-0.101
9	1	0.001	0.035	0.0
	2	0.005	0.043	-0.002
	3	0.009	0.050	-0.003
	4	0.009	0.050	-0.003
	5	0.009	0.052	-0.003
	6	0.009	0.052	-0.003
	7	0.008	0.056	-0.002
	8	0.007	0.060	-0.002
	9	0.006	0.060	-0.002
	10	0.0	0.193	0.0
	11	0.0	0.363	0.0
	12	0.0	0.517	0.0
	13	-0.001	0.651	0.0
	14	-0.001	0.759	0.0
	15	-0.001	0.821	0.0
	16	0.0	0.904	0.0
	17	0.0	1.0	0.0

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
10	1	-0.002	0.0	-0.014
	2	-0.063	0.002	-0.426
	3	-0.027	0.006	-0.378
	4	-0.040	0.002	-0.410
	5	-0.041	0.002	-0.361
	6	-0.028	0.014	-0.357
	7	-0.033	0.012	-0.220
	8	0.035	0.012	0.605
	9	0.085	0.005	1.0
	10	-0.001	-0.001	-0.010
	11	0.0	-0.003	-0.004
	12	0.001	-0.005	0.003
	13	0.001	-0.006	0.010
	14	0.002	-0.007	0.014
	15	0.002	-0.008	0.015
	16	0.001	-0.009	0.006
	17	-0.002	-0.010	-0.018
11	1	0.004	0.002	-0.029
	2	0.090	-0.004	-0.392
	3	0.108	0.023	-0.341
	4	0.064	0.008	-0.497
	5	0.050	0.024	-0.424
	6	0.082	0.019	-0.480
	7	-0.012	0.035	-0.406
	8	-0.106	0.054	0.657

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	9	-0.164	0.025	1.0
	10	0.005	0.001	-0.039
	11	0.005	-0.001	-0.036
	12	0.004	-0.002	-0.021
	13	0.001	-0.003	0.001
	14	-0.002	-0.004	0.025
	15	-0.004	-0.004	0.037
	16	-0.003	-0.005	0.032
	17	0.007	-0.006	-0.018
12	1	0.031	0.022	-0.050
	2	0.270	0.081	-0.571
	3	0.424	0.211	-0.840
	4	0.440	0.157	-0.696
	5	0.417	0.204	-0.683
	6	0.363	0.247	-0.569
	7	0.193	0.299	-0.027
	8	-0.410	0.372	0.541
	9	-0.746	0.277	1.0
	10	0.077	0.012	-0.123
	11	0.097	0.001	-0.158
	12	0.078	-0.010	-0.128
	13	0.027	-0.021	-0.046
	14	-0.036	-0.030	0.056
	15	-0.077	-0.035	0.123
	16	-0.076	-0.043	0.121
	17	0.051	-0.052	-0.095

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
13	1	-0.061	-0.023	-0.032
	2	-0.395	-0.069	-0.281
	3	-0.648	-0.219	-0.376
	4	-0.638	-0.174	-0.346
	5	-0.601	-0.241	-0.334
	6	-0.562	-0.206	-0.299
	7	-0.266	-0.291	-0.071
	8	0.547	-0.392	0.308
	9	1.0	-0.304	0.530
	10	-0.287	-0.014	-0.151
	11	-0.418	-0.003	-0.220
	12	-0.368	0.008	-0.194
	13	-0.159	0.019	-0.084
	14	0.120	0.028	0.063
	15	0.312	0.033	0.164
	16	0.337	0.041	0.177
	17	-0.199	0.051	-0.107
14	1	-0.015	0.005	0.030
	2	0.040	0.022	-0.104
	3	0.077	0.047	-0.162
	4	0.080	0.039	-0.148
	5	0.076	0.048	-0.145
	6	0.070	0.055	-0.130
	7	0.040	0.067	-0.036
	8	-0.070	0.083	0.126

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	9	-0.132	0.065	0.228
	10	-0.328	0.004	0.619
	11	-0.530	0.001	1.0
	12	-0.495	-0.002	0.933
	13	-0.239	-0.004	0.450
	14	0.126	-0.006	-0.238
	15	0.386	-0.007	-0.728
	16	0.440	-0.010	-0.830
	17	-0.240	-0.012	0.451
15	1	0.002	-0.022	0.002
	2	-0.283	-0.075	-0.214
	3	-0.506	-0.199	-0.297
	4	-0.505	-0.167	-0.279
	5	-0.480	-0.220	-0.270
	6	-0.451	-0.195	-0.247
	7	-0.228	-0.267	-0.071
	8	0.418	-0.351	0.257
	9	0.777	-0.284	0.441
	10	0.603	-0.015	0.323
	11	1.0	-0.004	0.534
	12	0.945	0.006	0.505
	13	0.466	0.016	0.248
	14	-0.227	0.024	-0.121
	15	-0.724	0.030	-0.387
	16	-0.835	0.037	-0.446
	17	0.449	0.047	0.237

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
16	1	-0.017	0.094	-0.002
	2	-0.159	0.367	0.005
	3	-0.275	0.553	0.035
	4	-0.279	0.611	0.021
	5	-0.270	0.635	0.021
	6	-0.252	0.668	0.012
	7	-0.109	0.770	-0.034
	8	0.566	0.905	-0.090
	9	0.897	1.0	-0.127
	10	0.007	0.074	-0.004
	11	0.034	0.043	-0.006
	12	0.048	0.008	-0.007
	13	0.041	-0.028	-0.006
	14	0.015	-0.061	-0.003
	15	-0.012	-0.083	0.0
	16	-0.048	-0.114	0.008
	17	-0.105	-0.154	0.035
17	1	0.0	0.0	0.0
	2	-0.001	0.0	-0.001
	3	0.001	0.0	-0.002
	4	0.002	0.0	0.0
	5	0.002	0.0	0.0
	6	0.002	0.0	0.001
	7	0.002	0.0	0.008
	8	0.002	0.0	0.008

TABLE 3.7B–6 CONTAINMENT AND INTERNAL STRUCTURES SIGNIFICANT MODE SHAPES (NORMALIZED EIGENVECTORS) CRACKED MODEL (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	9	0.002	0.0	0.010
	10	-0.013	0.0	-0.044
	11	-0.040	0.0	-0.134
	12	-0.064	0.0	-0.215
	13	-0.069	0.0	-0.229
	14	-0.046	0.0	-0.154
	15	-0.012	0.0	-0.039
	16	0.064	0.0	0.214
	17	0.298	0.0	1.0
18	1	-0.001	0.007	0.0
	2	-0.014	0.022	0.001
	3	-0.019	0.041	0.004
	4	-0.018	0.044	0.003
	5	-0.018	0.047	0.003
	6	-0.016	0.049	0.002
	7	-0.004	0.058	-0.003
	8	0.058	0.070	-0.011
	9	0.085	0.077	-0.016
	10	-0.044	0.006	0.013
	11	-0.133	0.004	0.040
	12	-0.214	0.001	0.064
	13	-0.228	-0.002	0.068
	14	-0.154	-0.004	0.046
	15	-0.040	-0.006	0.012
	16	0.212	-0.009	-0.064
	17	1.0	-0.012	-0.298

TABLE 3.7B–7 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE UNCRACKED MODEL, RESPONSE SPECTRUM ANALYSIS

	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
Degree of * Freedom	N-S	E-W		N-S	E-W	
1	0.058	0.026	0.012	0.031xE-3	0.05xE-3	0.031xE-3
2	0.013	0.007	0.048	0.035xE-3	0.034xE-3	0.112xE-3
3	0.026	0.063	0.006	0.050xE-3	0.396xE-3	0.030xE-3
4	0.627xE-4	0.254xE-3	0.646xE-4	0.010xE-4	0.084xE-4	0.001xE-4
5	0.874xE-4	0.559xE-4	0.335xE-4	0.003xE-4	0.007xE-4	0.0
6	0.232xE-3	0.531xE-4	0.973xE-4	0.073xE-4	0.010xE-4	0.002xE-4
7	0.163	0.046	0.063	0.224xE-2	0.047xE-2	0.032xE-2
8	0.028	0.013	0.068	0.201xE-3	0.172xE-3	0.251xE-3
9	0.054	0.163	0.080	0.740xE-3	0.382xE-2	0.403xE-3
10	0.862xE-3	0.211xE-2	0.102xE-2	0.165xE-4	0.702xE-4	0.045xE-4
11	0.242xE-2	0.421xE-2	0.111xE-2	0.280xE-4	0.911xE-4	0.047xE-4
12	0.176xE-2	0.416xE-3	0.646xE-3	0.446xE-4	0.077xE-4	0.043xE-4
13	0.192	0.053	0.091	0.443xE-2	0.094xE-2	0.056xE-2
14	0.052	0.021	0.101	0.850xE-3	0.287xE-3	0.425xE-3
15	0.059	0.238	0.118	0.929xE-3	0.834xE-2	0.594xE-3
16	0.926xE-3	0.287xE-2	0.150xE-2	0.140xE-4	0.911xE-4	0.076xE-4
17	0.148xE-2	0.403xE-2	0.134xE-2	0.347xE-4	0.134xE-3	0.071xE-4

TABLE 3.7B–7 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE UNCRACKED MODEL, RESPONSE SPECTRUM ANALYSIS (CONTINUED)

	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
Degree of * Freedom	N-S	E-W		N-S	E-W	
18	0.262xE-2	0.720xE-3	0.107xE-2	0.635xE-4	0.148xE-4	0.068xE-4
19	0.222	0.056	0.092	0.532xE-2	0.101xE-2	0.061xE-2
20	0.041	0.019	0.101	0.277xE-3	0.150xE-3	0.408xE-3
21	0.061	0.239	0.098	0.123xE-2	0.839xE-2	0.50xE-3
22	0.971xE-3	0.305xE-2	0.170xE-2	0.152xE-4	0.976xE-4	0.086xE-4
23	0.153xE-2	0.415xE-2	0.129xE-2	0.364xE-4	0.137xE-3	0.069xE-4
24	0.285xE-2	0.789xE-3	0.115xE-2	0.686xE-4	0.160xE-4	0.075xE-4
25	0.244	0.059	0.089	0.601xE-2	0.116xE-2	0.064xE-2
26	0.052	0.021	0.110	0.767xE-3	0.173xE-3	0.459xE-3
27	0.062	0.289	0.096	0.125xE-2	0.104xE-1	0.049xE-2
28	0.101xE-2	0.321xE-2	0.189xE-2	0.162xE-4	0.103xE-3	0.095xE-4
29	0.159xE-2	0.427xE-2	0.124xE-2	0.380xE-4	0.141xE-3	0.068xE-4
30	0.298xE-2	0.831xE-3	0.122xE-2	0.713xE-4	0.165xE-4	0.079xE-4
31	0.269	0.064	0.082	0.675xE-2	0.299xE-2	0.066xE-2
32	0.048	0.023	0.115	0.620xE-3	0.467xE-3	0.481xE-3
33	0.071	0.305	0.081	0.144xE-2	0.110xE-1	0.042xE-2
34	0.106xE-2	0.333xE-2	0.207xE-2	0.170xE-4	0.107xE-3	0.104xE-4
35	0.165xE-2	0.436xE-2	0.120xE-2	0.391xE-4	0.144xE-3	0.066xE-4

TABLE 3.7B–7 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE UNCRACKED MODEL, RESPONSE SPECTRUM ANALYSIS (CONTINUED)

	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
Degree of * Freedom	N-S	E-W		N-S	E-W	
36	0.309xE-2	0.870xE-3	0.129xE-2	0.735xE-4	0.170xE-4	0.083xE-4
37	0.358	0.054	0.075	0.951xE-2	0.210xE-3	0.079xE-2
38	0.060	0.026	0.138	0.991xE-3	0.380xE-3	0.569xE-3
39	0.071	0.433	0.013	0.20xE-2	0.160xE-1	0.019xE-2
40	0.122xE-2	0.364xE-2	0.261xE-2	0.192xE-4	0.118xE-3	0.131xE-4
41	0.193xE-2	0.481xE-2	0.658xE-3	0.456xE-4	0.161xE-3	0.046xE-4
42	0.349xE-2	0.101xE-2	0.158xE-2	0.804xE-4	0.185xE-4	0.097xE-4
43	0.523	0.115	0.111	0.138xE-1	0.305xE-2	0.119xE-2
44	0.101	0.036	0.166	0.152xE-2	0.402xE-3	0.685xE-3
45	0.120	0.623	0.078	0.307xE-2	0.228xE-1	0.047xE-2
46	0.154xE-2	0.416xE-2	0.311xE-2	0.210xE-4	0.125xE-3	0.156xE-4
47	0.307xE-2	0.609xE-2	0.314xE-2	0.697xE-4	0.171xE-3	0.164xE-4
48	0.414xE-2	0.127xE-2	0.212xE-2	0.886xE-4	0.272xE-4	0.119xE-4
49	0.622	0.144	0.180	0.161xE-1	0.348xE-2	0.149xE-2
50	0.101	0.035	0.175	0.857xE-3	0.262xE-3	0.652xE-3
51	0.170	0.698	0.142	0.398xE-2	0.249xE-1	0.077xE-2
52	0.162xE-2	0.448xE-2	0.319xE-2	0.213xE-4	0.126xE-3	0.160xE-4
53	0.312xE-2	0.619xE-2	0.328xE-2	0.703xE-4	0.172xE-3	0.171xE-4

TABLE 3.7B–7 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE UNCRACKED MODEL, RESPONSE SPECTRUM ANALYSIS (CONTINUED)

	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
Degree of * Freedom	N-S	E-W		N-S	E-W	
54	0.425xE-2	0.133xE-2	0.224xE-2	0.895xE-4	0.204xE-4	0.122xE-4
55	0.129	0.045	0.015	0.268xE-2	0.031xE-2	0.027xE-3
56	0.024	0.020	0.078	0.093xE-3	0.096xE-3	0.270xE-3
57	0.045	0.121	0.005	0.030xE-2	0.246xE-2	0.028xE-3
58	0.233xE-3	0.776xE-3	0.829xE-4	0.032xE-4	0.258xE-4	0.003xE-4
59	0.179xE-3	0.807xE-4	0.421xE-4	0.004xE-4	0.008xE-4	0.001xE-4
60	0.864xE-3	0.196xE-3	0.127xE-3	0.282xE-4	0.032xE-4	0.002xE-4
61	0.205	0.044	0.012	0.571xE-2	0.064xE-2	0.050xE-3
62	0.036	0.033	0.110	0.160xE-3	0.163xE-3	0.443xE-3
63	0.044	0.188	0.003	0.063xE-2	0.501xE-2	0.048xE-3
64	0.355xE-3	0.127xE-2	0.390xE-4	0.053xE-4	0.419xE-4	0.004xE-4
65	0.108xE-3	0.938xE-4	0.286xE-4	0.006xE-4	0.010xE-4	0.001xE-4
66	0.147xE-2	0.314xE-3	0.789xE-4	0.480xE-4	0.054xE-4	0.004xE-4
67	0.291	0.055	0.013	0.90xE-2	0.100xE-2	0.085xE-3
68	0.047	0.045	0.133	0.222xE-3	0.224xE-3	0.60xE-3
69	0.055	0.255	0.006	0.099xE-2	0.775xE-2	0.078xE-3
70	0.391xE-3	0.166xE-2	0.798xE-4	0.070xE-4	0.545xE-4	0.005xE-4
71	0.179xE-3	0.107xE-3	0.537xE-4	0.007xE-4	0.011xE-4	0.001xE-4

TABLE 3.7B–7 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE UNCRACKED MODEL, RESPONSE SPECTRUM ANALYSIS (CONTINUED)

Degree of * Freedom	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
	N-S	E-W		N-S	E-W	
72	0.194xE-2	0.377xE-3	0.115xE-3	0.636xE-4	0.071xE-4	0.006xE-4
73	0.373	0.049	0.017	0.124xE-1	0.138xE-2	0.012xE-2
74	0.059	0.056	0.154	0.276xE-3	0.276xE-3	0.736xE-3
75	0.049	0.320	0.008	0.136xE-2	0.109xE-3	0.106xE-1
76	0.454xE-3	0.199xE-2	0.640xE-4	0.083xE-4	0.639xE-4	0.006xE-4
77	0.192xE-3	0.126xE-3	0.481xE-4	0.008xE-4	0.011xE-4	0.001xE-4
78	0.233xE-2	0.455xE-3	0.102xE-3	0.751xE-4	0.081xE-4	0.007xE-4
79	0.458	0.059	0.013	0.157xE-1	0.174xE-2	0.015xE-2
80	0.068	0.065	0.176	0.319xE-3	0.319xE-3	0.845xE-3
81	0.059	0.381	0.009	0.173xE-2	0.133xE-1	0.014xE-2
82	0.508xE-3	0.225xE-2	0.425xE-4	0.092xE-4	0.704xE-4	0.007xE-4
83	0.142xE-3	0.113xE-3	0.354xE-4	0.008xE-4	0.012xE-4	0.002xE-4
84	0.262xE-2	0.495xE-3	0.101xE-3	0.831xE-4	0.093xE-4	0.008xE-4
85	0.519	0.065	0.017	0.182xE-1	0.202xE-2	0.018xE-2
86	0.072	0.069	0.188	0.344xE-3	0.343xE-3	0.909xE-3
87	0.064	0.428	0.009	0.20xE-2	0.154xE-1	0.016xE-2
88	0.513xE-3	0.238xE-2	0.797xE-4	0.096xE-4	0.738xE-4	0.007xE-4
89	0.198xE-3	0.125xE-3	0.491xE-4	0.008xE-4	0.012xE-4	0.001xE-4

TABLE 3.7B–7 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE UNCRACKED MODEL, RESPONSE SPECTRUM ANALYSIS (CONTINUED)

	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
Degree of * Freedom	N-S	E-W		N-S	E-W	
90	0.277xE-2	0.489xE-3	0.140xE-3	0.872xE-4	0.097xE-4	0.009xE-4
91	0.646	0.089	0.010	0.229xE-1	0.254xE-2	0.022xE-2
92	0.079	0.076	0.206	0.377xE-3	0.375xE-3	0.993xE-3
93	0.089	0.526	0.006	0.251xE-2	0.193xE-1	0.019xE-2
94	0.119xE-2	0.558xE-2	0.276xE-3	0.150xE-4	0.116xE-3	0.017xE-4
95	0.168xE-3	0.155xE-3	0.462xE-4	0.011xE-4	0.017xE-4	0.002xE-4
96	0.590xE-2	0.120xE-2	0.471xE-3	0.138xE-3	0.151xE-4	0.021xE-4
97	0.832	0.093	0.017	0.288xE-1	0.319xE-2	0.029xE-2
98	0.093	0.084	0.255	0.415xE-3	0.413xE-3	0.109xE-2
99	0.092	0.686	0.014	0.316xE-2	0.242xE-1	0.025xE-2
100	0.267xE-2	0.829xE-2	0.450xE-3	0.168xE-4	0.130xE-3	0.022xE-4
101	0.290xE-3	0.253xE-3	0.780xE-4	0.016xE-4	0.020xE-4	0.004xE-4
102	0.832xE-2	0.268xE-2	0.117xE-2	0.155xE-3	0.170xE-4	0.027xE-4

NOTES:

* Degrees of freedom corresponding to x-, y-, and z-direction translation and rotation about the x, y, and z axis are identified in Table 3.7B–9.

** Tabulated accelerations are in units of g for translational degrees of freedom, and g/feet for rotational degrees of freedom.

*** Tabulated displacements are in units of feet for translational degrees of freedom, and rads for rotational degrees of freedom.

TABLE 3.7B–8 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE CRACKED MODEL, RESPONSE SPECTRUM ANALYSIS

Degree of *Freedom	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
	N-S	E-W		N-S	E-W	
1	0.068	0.022	0.015	0.296xE-3	0.049xE-3	0.024xE-3
2	0.017	0.004	0.046	0.027xE-3	0.014xE-3	0.126xE-3
3	0.022	0.068	0.004	0.049xE-3	0.284xE-3	0.012xE-3
4	****0.418xE-4	0.136xE-3	0.284xE-4	0.007xE-4	0.065xE-4	0.001xE-4
5	0.882xE-4	0.579xE-4	0.869xE-4	0.004xE-4	0.007xE-4	0.0
6	0.168xE-3	0.358xE-4	0.761xE-4	0.066xE-4	0.007xE-4	0.003xE-4
7	0.180	0.054	0.060	0.252xE-2	0.516xE-3	0.232xE-3
8	0.023	0.011	0.075	0.20xE-3	0.157xE-3	0.263xE-3
9	0.072	0.160	0.026	0.838xE-3	0.307xE-2	0.145xE-3
10	0.904xE-3	0.206xE-2	0.745xE-3	0.184xE-4	0.568xE-4	0.026xE-4
11	0.245xE-2	0.392xE-2	0.122xE-2	0.316xE-4	0.841xE-4	0.028xE-4
12	0.201xE-2	0.543xE-2	0.502xE-3	0.506xE-4	0.089xE-4	0.038xE-4
13	0.217	0.075	0.065	0.505xE-2	0.102xE-2	0.438xE-3
14	0.057	0.028	0.103	0.986xE-3	0.284xE-3	0.383xE-3
15	0.085	0.216	0.039	0.983xE-3	0.699xE-2	0.216xE-3
16	0.119xE-2	0.297xE-2	0.512xE-3	0.151xE-4	0.750xE-4	0.026xE-4
17	0.171xE-2	0.405xE-2	0.417xE-3	0.386xE-4	0.124xE-3	0.034xE-4

TABLE 3.7B–8 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE CRACKED MODEL, RESPONSE SPECTRUM ANALYSIS (CONTINUED)

Degree of *Freedom	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
	N-S	E-W		N-S	E-W	
18	0.300xE-2	0.948xE-3	0.686xE-3	0.726xE-4	0.156xE-4	0.056xE-4
19	0.252	0.078	0.065	0.608xE-2	0.114xE-2	0.502xE-3
20	0.039	0.021	0.112	0.325xE-3	0.151xE-3	0.408xE-3
21	0.083	0.214	0.032	0.134xE-2	0.695xE-2	0.204xE-3
22	0.130xE-2	0.311xE-2	0.546xE-3	0.161xE-4	0.807xE-4	0.029xE-4
23	0.176xE-2	0.418xE-2	0.405xE-3	0.405xE-4	0.127xE-3	0.034xE-4
24	0.327xE-2	0.105xE-2	0.715xE-3	0.785xE-4	0.168xE-4	0.061xE-4
25	0.278	0.078	0.065	0.687xE-2	0.129xE-2	0.548xE-3
26	0.057	0.028	0.116	0.892xE-3	0.206xE-3	0.43xE-3
27	0.082	0.257	0.032	0.131xE-2	0.867xE-2	0.213xE-3
28	0.139xE-2	0.321xE-2	0.587xE-3	0.178xE-4	0.854xE-2	0.032xE-4
29	0.182xE-2	0.429xE-2	0.414xE-3	0.423xE-4	0.131xE-3	0.035xE-4
30	0.341xE-2	0.111xE-2	0.750xE-3	0.817xE-4	0.175xE-4	0.064xE-4
31	0.307	0.080	0.065	0.773xE-2	0.161xE-2	0.598xE-3
32	0.051	0.031	0.122	0.721xE-3	0.415xE-3	0.447xE-3
33	0.084	0.271	0.030	0.153xE-2	0.914xE-2	0.209xE-3
34	0.148xE-2	0.328xE-2	0.636xE-3	0.187xE-4	0.890xE-4	0.035xE-4
35	0.186xE-2	0.438xE-2	0.425xE-3	0.435xE-4	0.134xE-3	0.036xE-4

TABLE 3.7B–8 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE CRACKED MODEL, RESPONSE SPECTRUM ANALYSIS (CONTINUED)

Degree of *Freedom	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
	N-S	E-W		N-S	E-W	
36	0.353xE-2	0.116xE-2	0.790xE-3	0.842xE-4	0.181xE-4	0.066xE-4
37	0.413	0.084	0.043	0.109xE-1	0.220xE-2	0.794xE-3
38	0.070	0.038	0.142	0.115xE-2	0.374xE-3	0.515xE-3
39	0.077	0.379	0.011	0.208xE-2	0.134xE-1	0.236xE-3
40	0.181xE-2	0.346xE-2	0.853xE-3	0.214xE-4	0.982xE-4	0.043xE-4
41	0.212xE-2	0.481xE-2	0.302xE-3	0.507xE-4	0.150xE-3	0.037xE-4
42	0.396xE-2	0.137xE-2	0.106xE-2	0.922xE-4	0.199xE-4	0.076xE-4
43	0.605	0.132	0.118	0.159xE-1	0.322xE-2	0.121xE-2
44	0.101	0.50	0.173	0.175xE-2	0.435xE-3	0.608xE-3
45	0.137	0.551	0.037	0.323xE-2	0.191xE-1	0.366xE-3
46	0.220xE-2	0.429xE-2	0.989xE-3	0.237xE-4	0.105xE-3	0.051xE-4
47	0.366xE-2	0.656xE-2	0.944xE-3	0.777xE-4	0.167xE-3	0.065xE-4
48	0.474xE-2	0.176xE-2	0.175xE-2	0.102xE-3	0.220xE-4	0.096xE-4
49	0.718	0.175	0.181	0.185xE-1	0.372xE-2	0.147xE-2
50	0.083	0.038	0.196	0.995xE-3	0.282xE-3	0.649xE-3
51	0.199	0.620	0.060	0.427xE-2	0.209xE-1	0.464xE-3
52	0.228xE-2	0.481xE-2	0.105xE-2	0.240xE-4	0.105xE-3	0.052xE-4
53	0.372xE-2	0.667xE-2	0.987xE-3	0.784xE-4	0.168xE-3	0.067xE-4

TABLE 3.7B–8 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE CRACKED MODEL, RESPONSE SPECTRUM ANALYSIS (CONTINUED)

Degree of *Freedom	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
	N-S	E-W		N-S	E-W	
54	0.491xE-2	0.185xE-2	0.187xE-2	0.103xE-3	0.222xE-4	0.098xE-3
55	0.145	0.052	0.026	0.439xE-2	0.341xE-3	0.097xE-3
56	0.029	0.006	0.089	0.029xE-3	0.010xE-3	0.608xE-3
57	0.052	0.127	0.015	0.340xE-3	0.443xE-2	0.059xE-3
58	0.162xE-3	0.733xE-3	0.702xE-4	0.030xE-4	0.447xE-4	0.003xE-4
59	0.366xE-3	0.863xE-4	0.335xE-3	0.010xE-4	0.012xE-4	0.001xE-4
60	0.740xE-3	0.157xE-3	0.169xE-3	0.444xE-4	0.030xE-4	0.005xE-4
61	0.221	0.065	0.034	0.972xE-2	0.707xE-3	0.15xE-3
62	0.022	0.007	0.141	0.033xE-3	0.009xE-3	0.114xE-2
63	0.065	0.207	0.020	0.706xE-3	0.980xE-2	0.090xE-3
64	0.329xE-3	0.132xE-2	0.130xE-3	0.055xE-4	0.808xE-4	0.005xE-4
65	0.220xE-3	0.106xE-3	0.197xE-3	0.018xE-4	0.021xE-4	0.001xE-4
66	0.132xE-2	0.321xE-3	0.277xE-3	0.802xE-4	0.055xE-4	0.010xE-4
67	0.289	0.065	0.032	0.155xE-1	0.108xE-2	0.140xE-3
68	0.022	0.005	0.183	0.04xE-3	0.012xE-3	0.162xE-2
69	0.064	0.286	0.018	0.108xE-2	0.156xE-1	0.082xE-3
70	0.458xE-3	0.175xE-2	0.193xE-3	0.075xE-4	0.109xE-3	0.008xE-4
71	0.283xE-3	0.136xE-3	0.235xE-3	0.026xE-4	0.030xE-4	0.001xE-4

TABLE 3.7B–8 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE CRACKED MODEL, RESPONSE SPECTRUM ANALYSIS (CONTINUED)

Degree of *Freedom	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
	N-S	E-W		N-S	E-W	
72	0.176xE-2	0.450xE-3	0.378xE-3	0.108xE-3	0.075xE-4	0.015xE-4
73	0.351	0.058	0.021	0.215xE-1	0.147xE-2	0.073xE-3
74	0.028	0.005	0.227	0.047xE-3	0.017xE-3	0.204xE-2
75	0.058	0.355	0.011	0.146xE-2	0.216xE-1	0.040xE-3
76	0.538xE-3	0.211xE-2	0.240xE-3	0.090xE-4	0.130xE-3	0.011xE-4
77	0.366xE-3	0.172xE-3	0.318xE-3	0.033xE-4	0.038xE-4	0.002xE-4
78	0.213xE-2	0.530xE-3	0.450xE-3	0.129xE-3	0.090xE-4	0.019xE-4
79	0.416	0.039	0.010	0.273xE-1	0.185xE-2	0.049xE-3
80	0.017	0.007	0.263	0.053xE-3	0.220xE-3	0.238xE-2
81	0.039	0.423	0.006	0.185xE-2	0.275xE-1	0.028xE-3
82	0.519xE-3	0.243xE-2	0.255xE-3	0.101xE-4	0.145xE-3	0.011xE-4
83	0.180xE-3	0.198xE-3	0.621xE-4	0.039xE-4	0.045xE-4	0.002xE-4
84	0.244xE-2	0.583xE-3	0.481xE-3	0.144xE-3	0.101xE-4	0.020xE-4
85	0.476	0.061	0.026	0.317xE-1	0.214xE-2	0.114xE-3
86	0.020	0.008	0.282	0.058xE-3	0.026xE-3	0.257xE-2
87	0.061	0.484	0.015	0.214xE-2	0.320xE-1	0.068xE-3
88	0.584xE-3	0.262xE-2	0.248xE-3	0.106xE-4	0.153xE-3	0.011xE-4
89	0.347xE-3	0.222xE-3	0.276xE-3	0.043xE-4	0.049xE-4	0.002xE-4

TABLE 3.7B–8 CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS AND DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE CRACKED MODEL, RESPONSE SPECTRUM ANALYSIS (CONTINUED)

Degree of *Freedom	Acceleration **			Displacement ***		
	Horizontal		Vertical	Horizontal		Vertical
	N-S	E-W		N-S	E-W	
90	0.262xE-2	0.576xE-3	0.478xE-3	0.151xE-3	0.106xE-4	0.019xE-4
91	0.580	0.058	0.029	0.399xE-1	0.269xE-2	0.135xE-3
92	0.042	0.009	0.318	0.067xE-3	0.03xE-3	0.283xE-2
93	0.058	0.584	0.016	0.268xE-2	0.403xE-1	0.077xE-3
94	0.119xE-2	0.597xE-2	0.427xE-3	0.188xE-4	0.249xE-3	0.019xE-4
95	0.444xE-3	0.533xE-3	0.200xE-4	0.105xE-4	0.123xE-4	0.004xE-4
96	0.597xE-2	0.119xE-2	0.842xE-3	0.247xE-3	0.188xE-4	0.036xE-4
97	0.770	0.067	0.031	0.506xE-1	0.343xE-2	0.141xE-3
98	0.044	0.012	0.356	0.084xE-3	0.037xE-3	0.314xE-2
99	0.067	0.776	0.013	0.342xE-2	0.510xE-1	0.061xE-3
100	0.191xE-2	0.761xE-2	0.732xE-3	0.225xE-4	0.280xE-3	0.034xE-4
101	0.531xE-3	0.645xE-3	0.221xE-4	0.125xE-4	0.148xE-4	0.005xE-4
102	0.773xE-2	0.190xE-2	0.132xE-2	0.278xE-3	0.225xE-4	0.059xE-4

NOTES:

* Degrees of freedom corresponding to x- and y- direction translation and rotation about the z axis are identified in Table 3.7B–9.

** Tabulated accelerations are in units of g for translational degrees of freedom, and g/feet for rotational degrees of freedom.

*** Tabulated displacements are in units of feet for translational degrees of freedom, and rads for rotational degrees of freedom.

**** 10E-4 = 10^{-4} = 0.0010

TABLE 3.7B–9 CONTAINMENT AND INTERNAL STRUCTURES DEGREES OF FREEDOM

Mass Point	Degree of Freedom	Direction of Motion (Global Coordinates) *	
1	1	Translation	X
1	2	Translation	Y
1	3	Translation	Z
1	4	Rotation	X
1	5	Rotation	Y
1	6	Rotation	Z
2	7	Translation	X
2	8	Translation	Y
2	9	Translation	Z
2	10	Rotation	X
2	11	Rotation	Y
2	12	Rotation	Z
3	13	Translation	X
3	14	Translation	Y
3	15	Translation	Z
3	16	Rotation	X
3	17	Rotation	Y
3	18	Rotation	Z
4	19	Translation	X
4	20	Translation	Y
4	21	Translation	Z
4	22	Rotation	X
4	23	Rotation	Y
4	24	Rotation	Z
5	25	Translation	X
5	26	Translation	Y
5	27	Translation	Z

TABLE 3.7B–9 CONTAINMENT AND INTERNAL STRUCTURES DEGREES OF FREEDOM (CONTINUED)

Mass Point	Degree of Freedom	Direction of Motion (Global Coordinates) *	
5	28	Rotation	X
5	29	Rotation	Y
5	30	Rotation	Z
6	31	Translation	X
6	32	Translation	Y
6	33	Translation	Z
6	34	Rotation	X
6	35	Rotation	Y
6	36	Rotation	Z
7	37	Translation	X
7	38	Translation	Y
7	39	Translation	Z
7	40	Rotation	X
7	41	Rotation	Y
7	42	Rotation	Z
8	43	Translation	X
8	44	Translation	Y
8	45	Translation	Z
8	46	Rotation	X
8	47	Rotation	Y
8	48	Rotation	Z
9	49	Translation	X
9	50	Translation	Y
9	51	Translation	Z
9	52	Rotation	X
9	53	Rotation	Y
9	54	Rotation	Z
10	55	Translation	X

TABLE 3.7B–9 CONTAINMENT AND INTERNAL STRUCTURES DEGREES OF FREEDOM (CONTINUED)

Mass Point	Degree of Freedom	Direction of Motion (Global Coordinates) *	
10	56	Translation	Y
10	57	Translation	Z
10	58	Rotation	X
10	59	Rotation	Y
10	60	Rotation	Z
11	61	Translation	X
11	62	Translation	Y
11	63	Translation	Z
11	64	Rotation	X
11	65	Rotation	Y
11	66	Rotation	Z
12	67	Translation	X
12	68	Translation	Y
12	69	Translation	Z
12	70	Rotation	X
12	71	Rotation	Y
12	72	Rotation	Z
13	73	Translation	X
13	74	Translation	Y
13	75	Translation	Z
13	76	Rotation	X
13	77	Rotation	Y
13	78	Rotation	Z
14	79	Translation	X
14	80	Translation	Y
14	81	Translation	Z
14	82	Rotation	X
14	83	Rotation	Y

TABLE 3.7B–9 CONTAINMENT AND INTERNAL STRUCTURES DEGREES OF FREEDOM (CONTINUED)

Mass Point	Degree of Freedom	Direction of Motion (Global Coordinates) *	
14	84	Rotation	Z
15	85	Translation	X
15	86	Translation	Y
15	87	Translation	Z
15	88	Rotation	X
15	89	Rotation	Y
15	90	Rotation	Z
16	91	Translation	X
16	92	Translation	Y
16	93	Translation	Z
16	94	Rotation	X
16	95	Rotation	Y
16	96	Rotation	Z
17	97	Translation	X
17	98	Translation	Y
17	99	Translation	Z
17	100	Rotation	X
17	101	Rotation	Y
17	102	Rotation	Z

TABLE 3.7B–10 CONTAINMENT AND INTERNAL STRUCTURE ENVELOPED ACCELERATIONS AND DISPLACEMENTS HORIZONTAL-MOTION *

MassNo.	Elevation (ft)	Accelerations **				Displacement ***			
		SSE ****		OBE		SSE		OBE	
		N-S	E-W	N-S	E-W	N-S	E-W	N-S	E-W
Internal Structure									
1	(-) 37.25	0.170	0.170	0.090	0.090	0.004	0.006	0.003	0.004
2	(-) 11.25	0.294	0.298	0.168	0.192	0.039	0.060	0.025	0.041
3	3.0	0.357	0.414	0.230	0.283	0.078	0.118	0.051	0.081
4	11.27	0.395	0.397	0.267	0.272	0.092	0.121	0.060	0.083
5	18.17	0.421	0.447	0.270	0.304	0.105	0.145	0.068	0.100
6	24.5	0.451	0.456	0.290	0.309	0.125	0.154	0.077	0.105
7	50.84	0.541	0.517	0.349	0.352	0.167	0.219	0.108	0.149
8	86.5	0.855	0.821	0.552	0.558	0.243	0.316	0.158	0.215
9	109.1	1.070	1.010	0.694	0.685	0.284	0.356	0.184	0.243
External Structure									
10	(-) 6.75	0.223	0.194	0.128	0.114	0.058	0.058	0.038	0.038
11	16.25	0.321	0.292	0.196	0.180	0.127	0.127	0.083	0.083
12	39.25	0.386	0.368	0.240	0.230	0.201	0.202	0.130	0.131
13	62.25	0.439	0.424	0.294	0.268	0.276	0.278	0.180	0.181
14	85.25	0.530	0.468	0.357	0.303	0.350	0.353	0.228	0.229
15	104.0	0.601	0.560	0.407	0.358	0.408	0.410	0.265	0.267
16	133.5	0.745	0.658	0.502	0.427	0.513	0.516	0.334	0.336
17	163.3	0.942	0.855	0.642	0.556	0.650	0.654	0.423	0.425

NOTES:

* Maximum of cracked, uncracked
** Tabulated accelerations are in units of g
*** Displacements are listed in inches
**** N-S applies to X-direction, E-W applies to Z-direction

TABLE 3.7B–11 CONTAINMENT AND INTERNAL STRUCTURE ENVELOPED ACCELERATIONS AND DISPLACEMENTS VERTICAL-MOTION *

Mass No.	Elevation (ft)	Accelerations **		Displacements ***	
		SSE ****	OBE	SSE	OBE
Internal Structure					
1	(-) 37.25	0.113	0.060	0.002	0.001
2	(-) 11.25	0.108	0.068	0.007	0.005
3	3.0	0.188	0.121	0.020	0.013
4	11.27	0.172	0.111	0.011	0.007
5	18.17	0.202	0.130	0.018	0.012
6	24.5	0.204	0.132	0.019	0.012
7	50.84	0.251	0.162	0.024	0.016
8	86.5	0.324	0.207	0.034	0.022
9	109.1	0.317	0.199	0.023	0.015
External Structure					
10	(-) 6.75	0.124	0.071	0.008	0.005
11	16.25	0.179	0.103	0.014	0.009
12	39.25	0.225	0.132	0.020	0.013
13	62.25	0.268	0.163	0.025	0.016
14	85.25	0.308	0.182	0.029	0.019
15	104.0	0.330	0.196	0.032	0.021
16	133.5	0.369	0.230	0.035	0.023
17	163.3	0.431	0.259	0.039	0.025

NOTES:

* Maximum of cracked, uncracked
** Tabulated accelerations are in units of g
*** Displacements are listed in inches

TABLE 3.7B–12 MAIN STEAM VALVE BUILDING SIGNIFICANT MODAL FREQUENCIES AND PARTICIPATION FACTORS

Mode	Frequency (Hz)	Participation Factors		
		X	Y	Z
1	8.380	126.101	0.963	0.630
2	9.456	0.813	-2.172	-133.882
3	13.093	17.672	0.510	0.919
4	23.991	7.296	-46.766	91.569
5	24.478	108.417	5.424	-6.261
6	25.663	2.749	-159.477	-30.236
7	29.689	23.791	5.297	3.240
8	33.833	36.719	-1.254	-0.800
9	34.306	2.605	-18.092	35.379
10	50.549	2.403	-28.776	-121.336
11	51.756	125.939	0.508	1.685
12	52.469	1.052	-110.173	48.891
13	61.221	20.459	26.390	10.175
14	61.438	7.329	-51.449	-28.932
15	64.407	3.223	-2.901	0.345
16	84.853	1.025	59.045	2.653

TABLE 3.7B–13 MAIN STEAM VALVE BUILDING SIGNIFICANT MODE SHAPES (EIGENVECTORS)

Mode	Joint	Displacement		
		X	Y	Z
1	1	0.013	0.000	0.000
	2	0.043	0.000	0.000
	3	0.232	0.005	0.002
	4	1.0	0.005	0.005
2	1	0.000	0.000	0.017
	2	0.000	0.001	0.053
	3	-0.001	0.011	0.318
	4	-0.007	0.011	1.0
3	1	0.033	0.001	0.002
	2	0.106	0.001	0.010
	3	0.205	0.017	-0.034
	4	1.0	0.020	0.091
4	1	0.007	-0.036	0.093
	2	0.014	-0.050	0.167
	3	0.074	-0.134	1.0
	4	0.008	-0.357	-0.155
5	1	0.101	0.004	-0.006
	2	0.209	0.005	-0.011
	3	1.0	0.014	-0.056
	4	-0.166	0.036	0.007
6	1	-0.003	0.125	0.032
	2	-0.001	-0.173	0.064
	3	-0.013	-0.458	0.321
	4	-0.013	1.0	-0.094
7	1	0.210	0.036	0.029
	2	0.167	0.047	0.066
	3	0.256	0.066	0.124

TABLE 3.7B–13 MAIN STEAM VALVE BUILDING SIGNIFICANT MODE SHAPES (EIGENVECTORS) (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	4	1.0	0.217	-0.007
8	1	0.160	-0.004	-0.003
	2	0.403	-0.006	-0.009
	3	1.0	0.010	-0.075
	4	-0.696	-0.033	0.069
9	1	0.009	-0.048	0.124
	2	0.021	-0.064	0.305
	3	0.053	-0.105	1.0
	4	-0.035	-0.146	-0.776
10	1	-0.012	0.107	0.599
	2	-0.018	0.135	1.0
	3	0.004	0.256	-0.350
	4	0.0	-0.159	0.026
11	1	0.629	0.002	0.008
	2	1.0	0.001	0.014
	3	-0.393	0.009	-0.005
	4	0.033	-0.006	0.0
12	1	-0.007	0.533	-0.315
	2	-0.004	0.678	-0.369
	3	-0.003	1.0	0.090
	4	-0.001	-0.662	-0.053
13	1	0.857	0.829	0.426
	2	-0.193	1.0	0.092
	3	0.619	0.287	0.070
	4	0.186	-0.467	0.094
14	1	-0.157	0.829	0.621
	2	0.040	1.0	0.148
	3	-0.117	0.508	0.079

TABLE 3.7B–13 MAIN STEAM VALVE BUILDING SIGNIFICANT MODE SHAPES (EIGENVECTORS) (CONTINUED)

Mode	Joint	Displacement		
		X	Y	Z
	4	-0.034	-0.704	0.133
15	1	-0.769	0.519	-0.082
	2	-0.055	0.617	-0.858
	3	-0.026	0.155	1.0
	4	-0.264	-0.387	0.042
16	1	-0.015	-0.634	-0.038
	2	0.002	-0.646	0.003
	3	0.0	1.0	-0.002
	4	0.002	-0.224	0.008

TABLE 3.7B–14 MAIN STEAM VALVE BUILDING CSM* ACCELERATIONS FOR SAFE SHUTDOWN EARTHQUAKE FROM RESPONSE SPECTRUM ANALYSIS

Degree of Freedom **	X ***	Y***	Z ***
1	0.103	0.006	0.015
2	0.007	0.059	0.045
3	0.015	0.040	0.107
4	0.0	0.001	0.002
5	0.001	0.0	0.001
6	0.002	0.0	0.0
7	0.156	0.003	0.012
8	0.008	0.072	0.057
9	0.012	0.050	0.160
10	0.001	0.002	0.003
11	0.001	0.0	0.0
12	0.003	0.001	0.001
13	0.193	0.010	0.018
14	0.008	0.115	0.089
15	0.019	0.083	0.187
16	0.001	0.003	0.004
17	0.003	0.0	0.0
18	0.003	0.002	0.001
19	0.414	0.004	0.005
20	0.014	0.168	0.095
21	0.006	0.022	0.395
22	0.001	0.002	0.006
23	0.003	0.0	0.0
24	0.005	0.001	0.001

NOTES:

* Closely spaced modes.

** The relationship between the degrees of freedom and the motion of the lumped masses is given in Table 3.7B–17.

*** Tabulated accelerations are in units of g for translational degrees of freedom, and g/feet for rotational degrees.

TABLE 3.7B–15 MAIN STEAM VALVE BUILDING CSM* DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE FROM RESPONSE SPECTRUM ANALYSIS

Degree of Freedom **	X ***	Y ***	Z ***
1	$0.703x10^{-4}$	$0.210x10^{-5}$	$0.310x10^{-5}$
2	$0.230x10^{-5}$	$0.279x10^{-4}$	$0.173x10^{-4}$
3	$0.310x10^{-5}$	$0.154x10^{-4}$	$0.716x10^{-4}$
4	$0.10x10^{-6}$	$0.20x10^{-6}$	$0.330x10^{-5}$
5	$0.40x10^{-6}$	0.0	$0.10x10^{-6}$
6	$0.42x10^{-5}$	$0.10x10^{-6}$	$0.10x10^{-6}$
7	$0.217x10^{-3}$	$0.270x10^{-5}$	$0.560x10^{-5}$
8	$0.310x10^{-5}$	$0.379x10^{-4}$	$0.230x10^{-4}$
9	$0.540x10^{-5}$	$0.246x10^{-4}$	$0.198x10^{-3}$
10	$0.10x10^{-6}$	$0.40x10^{-6}$	$0.660x10^{-5}$
11	$0.150x10^{-5}$	$0.10x10^{-6}$	$0.30x10^{-6}$
12	$0.840x10^{-5}$	$0.20x10^{-6}$	$0.20x10^{-6}$
13	$0.113x10^{-2}$	$0.143x10^{-4}$	$0.259x10^{-4}$
14	$0.237x10^{-4}$	$0.927x10^{-4}$	$0.630x10^{-4}$
15	$0.287x10^{-4}$	$0.107x10^{-3}$	$0.116x10^{-2}$
16	$0.30x10^{-6}$	$0.110x10^{-5}$	$0.160x10^{-4}$
17	$0.55x10^{-5}$	$0.30x10^{-6}$	$0.10x10^{-5}$
18	$0.242x10^{-4}$	$0.50x10^{-6}$	$0.40x10^{-6}$

TABLE 3.7B–15 MAIN STEAM VALVE BUILDING CSM* DISPLACEMENTS FOR SAFE SHUTDOWN EARTHQUAKE FROM RESPONSE SPECTRUM ANALYSIS

Degree of Freedom **	X ***	Y ***	Z ***
19	$0.480x10^{-2}$	$0.249x10^{-4}$	$0.406x10^{-4}$
20	$0.289x10^{-4}$	$0.199x10^{-3}$	$0.111x10^{-3}$
21	$0.379x10^{-4}$	$0.463x10^{-3}$	$0.340x10^{-2}$
22	$0.90x10^{-6}$	$0.240x10^{-5}$	$0.336x10^{-4}$
23	$0.274x10^{-4}$	$0.20x10^{-6}$	$0.140x10^{-5}$
24	$0.417x10^{-4}$	$0.60x10^{-6}$	$0.60x10^{-6}$

NOTES:

* Closely spaced modes.
** The relationship between the degrees of freedom and the motion of the lumped masses is given in Table 3.7B–17.
*** Tabulated displacements are in units of feet for translational degrees of freedom, and g⁄radians for rotational degrees.

TABLE 3.7B–16 MAIN STEAM VALVE BUILDING ABS * OF SAFE SHUTDOWN EARTHQUAKE RESPONSES

Degree of Freedom **	Acceleration ***	Displacement ****
1	0.170	0.750×10^{-4}
2	0.113	0.50×10^{-4}
3	0.170	0.917×10^{-4}
4	0.003	0.360×10^{-5}
5	0.002	0.50×10^{-6}
6	0.002	0.440×10^{-5}
7	0.170	0.225×10^{-3}
8	0.137	0.667×10^{-4}
9	0.222	0.225×10^{-3}
10	0.006	0.710×10^{-5}
11	0.001	0.190×10^{-5}
12	0.005	0.880×10^{-5}
13	0.221	0.117×10^{-2}
14	0.212	0.183×10^{-3}
15	0.290	0.129×10^{-2}
16	0.008	0.174×10^{-4}
17	0.003	0.680×10^{-5}
18	0.006	0.251×10^{-4}
19	0.424	0.487×10^{-2}
20	0.277	0.342×10^{-3}
21	0.423	0.368×10^{-2}
22	0.009	0.369×10^{-4}

TABLE 3.7B–16 MAIN STEAM VALVE BUILDING ABS * OF SAFE SHUTDOWN EARTHQUAKE RESPONSES (CONTINUED)

Degree of Freedom **	Acceleration ***	Displacement ****
23	0.003	0.290×10^{-4}
24	0.007	0.429×10^{-4}

NOTES:

* Absolute sum of each direction of excitation.
** Table 3.7B–18 defines degrees of freedom.
*** Tabulated accelerations are in units of g for translational degrees of freedom and g/feet for rotational degrees of freedom.
**** Tabulated displacements are in units of feet for translational degrees of freedom and rads for rotational degrees.

TABLE 3.7B–17 MAIN STEAM VALVE BUILDING DEGREES OF FREEDOM

Mass Point*	Degree of Freedom	Direction of Motion
1	1	Translation X
1	2	Translation Y
1	3	Translation Z
1	4	Rotation X
1	5	Rotation Y
1	6	Rotation Z
2	7	Translation X
2	8	Translation Y
2	9	Translation Z
2	10	Rotation X
2	11	Rotation Y
2	12	Rotation Z
3	13	Translation X
3	14	Translation Y
3	15	Translation Z
3	16	Rotation X
3	17	Rotation Y
3	18	Rotation Z
4	19	Translation X
4	20	Translation Y
4	21	Translation Z
4	22	Rotation X
4	23	Rotation Y
4	24	Rotation Z

NOTE:

* For locations of lumped masses, refer to Figure 3.7B–10.

TABLE 3.7B–18 EMERGENCY GENERATOR ENCLOSURE ACCELERATIONS AND DISPLACEMENTS HORIZONTAL-MOTION*

Mass No.	Elevation (ft-in)	Accelerations ** SSE		OBE	
		N-S	E-W	N-S	E-W
3	66-0	0.660	0.547	0.426	0.357
2	51-0	0.514	0.438	0.341	0.288
1	24-6	0.317	0.285	0.173	0.163

* N-S applies to X-direction, E-W applies to Z-direction.

** Tabulated accelerations are in units of g.

Mass No.	Elevation (ft-in)	Displacements *** SSE		OBE	
		N-S	E-W	N-S	E-W
3	66-0	0.1070	0.0425	0.0692	0.0278
2	51-0	0.0830	0.0340	0.0551	0.0224
1	24-6	0.0379	0.0170	0.0234	0.0102

*** Displacements are listed in inches.

TABLE 3.7B–19 EMERGENCY GENERATOR ENCLOSURE ACCELERATIONS AND DISPLACEMENTS VERTICAL - MOTION

Mass No.	Elevation ft-in	Accelerations*	
		SSE	OBE
3	66-0	0.250	0.148
2	41-0	0.220	0.128
1	24-6	0.176	0.100

Mass No.	Elevation ft-in	Displacements **	
		SSE	OBE
3	66-0	0.0055	0.0033
2	51-0	0.0048	0.0028
1	24-6	0.0040	0.0024

NOTES:

* Tabulated accelerations are in units of g.
** Displacements are listed in inches

TABLE 3.7B–20 COMPARISON OF RESPONSE SPECTRA AND TIME HISTORY ANALYSIS RESULTS CONTAINMENT AND INTERNAL STRUCTURES (UNCRACKED PROPERTIES)

	SSE Accelerations **					
	Response Spectrum			Time History		
Mass No.*	N-S	E-W	Vertical	N-S	E-W	Vertical
1	0.170	0.170	0.113	0.180	0.179	0.140
2	0.271	0.298	0.109	0.222	0.246	0.170
3	0.336	0.414	0.174	0.276	0.342	0.205
4	0.370	0.397	0.161	0.302	0.335	0.208
5	0.392	0.447	0.183	0.320	0.370	0.218
6	0.415	0.456	0.186	0.341	0.380	0.223
7	0.486	0.517	0.224	0.429	0.483	0.249
8	0.748	0.821	0.303	0.598	0.675	0.282
9	0.946	1.010	0.312	0.701	0.760	0.290
10	0.188	0.171	0.122	0.226	0.213	0.181
11	0.260	0.235	0.179	0.302	0.290	0.224
12	0.359	0.316	0.225	0.383	0.354	0.263
13	0.439	0.376	0.268	0.451	0.406	0.302
14	0.530	0.449	0.308	0.502	0.462	0.336
15	0.601	0.501	0.330	0.541	0.507	0.355
16	0.745	0.621	0.369	0.671	0.620	0.380
17	0.942	0.792	0.431	0.839	0.809	0.411

NOTES:

* Figure 3.7B–9 shows the location of the lumped masses.

** Safe shutdown earthquake accelerations are in units of g.

TABLE 3.7B–21 COMPARISON OF RESPONSE SPECTRA AND TIME HISTORY ANALYSIS RESULTS CONTAINMENT AND INTERNAL STRUCTURES (CRACKED PROPERTIES)

	SSE Accelerations **					
	Response Spectrum			Time History		
Mass No. *	N-S	E-W	Vertical	N-S	E-W	Vertical
1	0.170	0.170	0.113	0.178	0.178	0.127
2	0.294	0.258	0.108	0.225	0.242	0.161
3	0.357	0.340	0.188	0.302	0.338	0.197
4	0.395	0.329	0.172	0.332	0.339	0.204
5	0.421	0.372	0.202	0.355	0.373	0.211
6	0.451	0.385	0.204	0.380	0.382	0.215
7	0.541	0.467	0.251	0.486	0.464	0.239
8	0.855	0.724	0.324	0.679	0.638	0.270
9	1.070	0.878	0.317	0.782	0.714	0.281
10	0.223	0.194	0.125	0.215	0.215	0.159
11	0.321	0.292	0.169	0.272	0.276	0.236
12	0.386	0.368	0.210	0.358	0.361	0.304
13	0.430	0.424	0.260	0.431	0.435	0.382
14	0.465	0.468	0.287	0.490	0.498	0.437
15	0.562	0.560	0.310	0.535	0.538	0.464
16	0.667	0.658	0.369	0.717	0.711	0.497
17	0.867	0.855	0.412	0.962	0.991	0.538

NOTES:

* Figure 3.7B–9 shows the location of the lumped masses.

** Safe shutdown earthquake accelerations are in units of g.

TABLE 3.7B–22 COMPARISON OF RESPONSE SPECTRA AND TIME HISTORY ANALYSIS RESULTS MAIN STEAM VALVE BUILDING SSE ACCELERATIONS*

Mass No. **	Response Spectrum			Time History		
	N-S	E-W	Vertical	N-S	E-W	Vertical
1	0.170	0.170	0.113	0.174	0.174	0.118
2	0.170	0.222	0.137	0.180	0.181	0.120
3	0.221	0.290	0.212	0.250	0.254	0.135
4	0.424	0.423	0.277	0.488	0.441	0.159

NOTES:

* Safe shutdown earthquake accelerations are in units of g.

** Figure 3.7B–10 shows the location of the lumped masses.

TABLE 3.7B–23 1 G FLAT RESPONSE

Model Beam	Fundamental Frequency	Max Dynamic		Max Static		Location	S/C A/U	K S/U	
		Sum	Moment (in-lb	Load Type	Moment (in-lb)				
Cantilever	1	SRSS	620,000	Conc *	700,000	Fixed End	0.89	0.89	0.99
		ABS	649,000	Unif **	700,000				
Simple-Simple	1	SRSS	179,000	Conc	348,000	Midspan	0.51	1.03	1.07
		ABS	186,000	Unif	174,000				
Fixed-Fixed	1	SRSS	103,000	Conc	174,000	Fixed End	0.59	0.89	0.97
		ABS	112,000	Unif	116,000				
Simple-Fixed - No Overhang	1	SRSS	152,000	Conc	261,000	Fixed End	0.58	0.87	0.97
		ABS	169,000	Unif	174,000				
Simple-Fixed - 16% Overhang	1.34	SRSS	83,200	Conc	162,000	Fixed End	0.51	0.75	1.03
		ABS	114,000	Unif	111,000				
Simple-Fixed - 33% Overhang	1.04	SRSS	57,000	Conc	77,400	Fixed End	0.74	0.74	1.15
		ABS	89,000	Unif	77,200				
Simple-Fixed - 50% Overhang	0.62	SRSS	152,000	Conc	174,000	Simple Support	0.87	0.87	1.01
		ABS	176,000	Unif	174,000				

NOTES:

* Conc = Concentrated load

** Unif = Uniform load

TABLE 3.7B–24 MODAL DENSITY, n*

Mode No	Cantilever Frequency (Hz)	Fixed-Fixed Frequency (Hz)	Simple-Fixed Frequency (Hz)	Simple-Simple Frequency (Hz)	Simple-Fixed 33% Overhang Frequency (Hz)
1	1.0	1.0	1.0	1.0	1.0
2	5.8	2.7	3.2	3.8	2.9
3	15.3	4.9	6.3	8.2	6.5
4	22.8	7.5	10.2	13.6	8.4
5	28.0	10.2	14.0	19.5	13.3
6	43.2	12.9	17.9	25.4	17.6

NOTE:

* Modal density is based on a ±10 percent criterion.

TABLE 3.7B–25 AMPLIFIED RESPONSE DYNAMIC FACTOR STUDY

(For Simple-Fixed Beam with 33% Overhang)

Model Beam	First Mode	Dynamic Load		Maximum Dynamic Moment Sum Moment (S) (in-lb)		Location	Maximum Static Moment Uniform Load (U) (in-lb)*	K S/U A/U
		High	Low					
Simple-Fixed 33%	g_o	g_{max}	g_c					
Overhang	f_o	Δf	f_c					
Model 6A	0.10	2.87	0.33	SRSS	20,000	Fixed	222,000 s	0.09
	0.70	3-4	20	ABS	30,000	Fixed		0.13
Model 6B	0.10	2.87	0.33	SRSS	148,00	Fixed	222,000 s	0.67
	1.0	3-4	20	ABS	157,00	Fixed		0.71
Model 6C	2.87	2.87	0.33	SRSS	102,00	Simple Support	222,000 s	0.45
	3.3	3-4	20	ABS	118,00	Simple Support		0.53
Model 6D	0.40	2.87	0.33	SRSS	22,000	Fixed	31,000 s	0.71
	10.0	3-4	20	ABS	32,000	Fixed		1.03
Model 6E	0.33	2.87	0.33	SRSS	20,000	Fixed	25,700 s	0.78
	20.0	3-4	20	ABS	27,000	Fixed		1.05
Model 6F	0.30	2.87	0.33	SRSS	18,000	Fixed	23,400 s	0.77
	33.0	3-4	20	ABS	25,000	Fixed		1.07

NOTE:

* g = gmax, if $f_o < f_p$; g at $f_o \geq f_p$ (where f_p is the frequency at peak)

TABLE 3.7B–26 PIPING SYSTEM SEISMIC DESIGN AND ANALYSIS CRITERIA

ASME Section III Code Class	Type of Earthquake	Type of Seismic Analyses	Combined Stress Calculations and Stress Criteria
Class 1 (Sizes 1.25 inch NPS and larger)	SSE	Dynamic response spectra	ASME Code, Section III, Subarticle NB-3600
	OBE	Dynamic response spectra	ASME Code, Section III, Subarticle NB-3600
Class 1 (Sizes 1 inch NPS and below)	SSE	Simplified analyses	ASME Code, Section III, Subarticle NB-3600
	OBE	Simplified analyses	ASME Code, Section III, Subarticle NB-3600
Classes 2 and 3 (Sizes 2.5 inch NPS and larger)	SSE	Dynamic response spectra	ASME Code, Section III, Subarticles NC-3600 and ND-3600
	OBE	Dynamic response spectra	ASME Code, Section III, Subarticles NC-3600 and ND-3600
Classes 2 and 3 (Sizes 2 inch NPS and below)	SSE	Dynamic response spectra or simplified analyses	ASME Code, Section III, Subarticles NC-3600 and ND-3600
	OBE	Dynamic response spectra or simplified analyses	ASME Code, Section III, Subarticles NC-3600 and ND-3600

TABLE 3.7N–1 DAMPING VALUES USED FOR SEISMIC ANALYSIS FOR WESTINGHOUSE SUPPLIED EQUIPMENT

Item	Damping (Percent of Critical)	
	Upset Conditions (OBE)	**Faulted Conditions (SSE, DBA)**
Primary coolant loop system components	2	4
Welded steel structures	2	4
Bolted and/or riveted steel structures	4	7
Fuel assemblies	7	10
CRDMs/CRDM supports	5	5

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 CONCRETE CONTAINMENT

This section provides the following information on concrete containment and on concrete portions of steel/concrete containments:

1. The physical description
2. The applicable design codes, standards, and specifications
3. The loading criteria, including loads and load combinations
4. The design and analysis procedures
5. The structural acceptance criteria
6. The materials, quality control programs, and special construction techniques
7. The testing and inservice inspection programs

3.8.1.1 Description of the Containment

The containment structure is a steel-lined conventionally reinforced concrete pressurized water reactor containment structure designed to operate under sub-atmospheric conditions. The structure has a vertical cylindrical wall and hemispherical dome supported on a flat base mat which is founded on bedrock (Figures 3.8–1 and 3.8–2).

The design, analyses, and construction of the containment structure is similar to the designs for the following plants:

Connecticut Yankee Atomic Power Company, Nuclear Power Plant (Docket No. 50-213)

Virginia Electric and Power Company, Surry Power Station, Units 1 and 2 (Docket Nos. 50-280 and 50-281)

Maine Yankee Atomic Power Company, Maine Yankee Atomic Power Station (Docket No. 50-309)

Duquesne Light Company, Beaver Valley Power Station, Unit No. 1 (Docket No. 50-334)

Virginia Electric and Power Company, North Anna Power Station, Units 1 and 2 (Docket Nos. 50-338 and 50-339)

3.8.1.1.1 Base Foundation

The base foundation slab is 10 feet thick with a 158 foot diameter. The bottom reinforcement is a rectangular grid pattern and the top reinforcement consists of concentric circular bars combined with radial bars arranged to permit uniform spacing of the vertical wall reinforcing bars (rebars) which extend into the mat. Splices in adjacent parallel bars are staggered.

The floor liner plate of 0.25 inch thickness is anchored to the mat by means of rebars welded to 7 inch by 0.5 inch continuous vertical plates which are in turn welded to the liner seams (Figure 3.8–3). Where interior wall and column rebars are anchored to the mat, a 3 inch x 6 inch rectangular bar (bridging bar) or 1.25 inch thick plate was welded through the liner with test channels all around to ensure the continuity of the steel membrane. The vertical rebar was Cadwelded to the top of the bar and butt-welded to the bottom of these members providing continuity of rebar without creating multiple penetrations of the liner.

A reinforced concrete slab approximately 2 feet thick, was placed over and anchored through the mat liner to stiffen it against negative pressures and to protect it from heat associated with a DBA similar to other pressurized water reactor containment structures (e.g., North Anna 1 and 2, Docket No. 50-338 and 50-339). his slab also serves as anchorage and support for equipment located on the lowest level of the containment.

3.8.1.1.2 Cylindrical Wall

The cylindrical wall is 4 feet 6 inches thick with an inside diameter of 140 feet and a height from mat to spring line of 131 feet 3 inches. Hoop tension in the cylindrical wall of the containment structure is resisted by horizontal bars located near both the outer and inner faces of the wall. Splices in these bars are staggered where possible. Meridional tension is resisted by rows of vertical bars placed near each face of the wall. The vertical bars are placed in groups of not more than 20 bars of equal length. These groups are arranged so that no adjacent group in the same face of the wall has splices closer than 3 feet apart vertically.

Radial shear in the containment structure wall resulting from the design basis accident (DBA) varies from a maximum at the base of the wall and approaches zero at some level above the top of the mat. To resist the large radial shear near the base of the wall, flat steel bars inclined at approximately 45 degrees with the horizontal are welded to the vertical reinforcement as shown in Figure 3.8–4. The welded flat bars terminate at approximately elevation (-) 18 feet where the radial shear load is reduced. Above this level, radial shear is resisted by Z type reinforcing steel. This Z reinforcing continues to elevation (-) 4 feet (-) 8 inches where the shear becomes insignificant. In the lower portion of the containment structure wall, supplementary reinforcing steel, normal to the face of the wall is provided to resist potential splitting of the concrete in the plane of the vertical bars. Tangential shear (Vu) resulting from the earthquake loading is resisted by the concrete and diagonal reinforcing bars. A maximum allowable tangential shear stress is assigned to the concrete. Stresses in excess of Vc are resisted by inclined reinforcing bars anchored in the mat. No diagonal rebars are required above the spring line.

A nominal minimum clearance of 2 inches is provided as an earthquake rattlespace at all levels between the interior steel work, and platforms and the containment structure liner. Actual clearances of less than 2 inches were reviewed on a case-by-case basis and were determined to provide adequate clearance to preclude interaction with the containment liner for the worst case design bases movements.

Penetrations in the liner are provided for piping, electrical conductors, fuel transfer tube, and purge air ducts as indicated in Section 3.8.1.1.4. Access openings in the structure are the 7 foot inside diameter personnel access lock and the 15 foot inside diameter equipment hatch (Figure 3.8–7). The reinforcement around these penetrations is shown in Figures 3.8–5 through 3.8–9.

Brackets for support of internal miscellaneous small piping, and electrical conductors are attached to insert plates in the liner and/or to overlay plates on the liner. Major equipment and pipe loads, except at the penetrations, are not carried by the exterior wall. Steel anchors were used to attach the insert plates and the liner to the concrete structure. External supports on the containment wall for the enclosure structure are shown in Figures 3.8–10 and 3.8–61.

3.8.1.1.3 Dome

The inside radius of the 2 foot 6 inch thick dome is 70 feet. The internal height from base mat to the center of the dome is 201 feet 3 inches. The dome reinforcement consists of a layer of reinforcement placed meridionally extending from the vertical bars of the cylindrical wall near each face and horizontal hoop bars in similar layers. Mechanical anchorages are provided on the last set of meridional bars extending up toward the crown of the dome. All other meridional bars are terminated at a point where they are adequately developed beyond the point at which they were required for design purposes. Near the crown meridional bars are welded to a ring cast in the concrete concentric with the centerline of the containment. Figure 3.8–11 shows the rebar at the dome cylinder junction.

The dome liner plate is 0.5 inch thick. Attachments to the dome are basically for the quench and recirculation spray nozzle lines. One major connection to the dome structure is the top pivot support for the spray header maintenance truss (Figure 3.8–12). This structure is supported at the top of the dome and on the polar crane rail.

3.8.1.1.4 Steel Liner and Penetrations

A. Steel Liner

The steel liner consists of a vertical cylindrical portion, closed at the top by a hemispherical dome and attached at the bottom by a mat liner portion. As shown in Figure 3.8–13, the liner is welded to a skirt ring (knuckle plate assembly) that is welded to a plate which in turn is embedded and anchored into the concrete mat. The knuckle plate-to-liner junction and the knuckle plate-to-mat anchorage is proportioned to develop the full strength of the liner.

The liner plate is a continuously welded steel membrane supported by and anchored to the inside of the containment at sufficiently close intervals with anchor studs and deformed bars so that the overall deformation of the liner under the parameters derived from the design basis accident (DBA) and normal operation is essentially the same as that of the concrete containment structure.

The function of the liner is to act as a gas-tight membrane under conditions that can be encountered throughout the operating life of the plant. The liner is designed to resist all direct loads and accommodate deformation of the concrete containment structure without jeopardizing leak-tight integrity. Under DBA conditions, the liner is under a state of biaxial compressive strain due to thermal effects and during the test condition, the liner plate is under a state of biaxial tensile strain. The anchor studs prevent buckling of the liner and act as nodal points. Tests conducted at Northeastern University, Boston, Massachusetts, using 5/8-inch diameter studs and 3/8-inch thick plate, show that shear failure occurs in the stud adjacent to the weld connecting the stud to the plate; in no instance was the plate damaged. Tests conducted for the stud manufacturer under the direction of Dr. I.M. Viest (TRW, Inc. 1975) indicate that, with the manufacturer's recommended depth of embedment of the stud in concrete, the ultimate strength of the stud material can be developed in direct tension. The reinforcement ring and liner adjacent to the hatches are anchored to the concrete containment with a denser stud pattern.

The liner pressure boundary includes embedments, insert plates, and penetrations. Liner dimensions are given in Sections 3.8.1.1.1 to 3.8.1.1.3 and shown in Figure 3.8–14. Leak chase channels are installed over penetration to liner seams and over knuckle plate to liner seams.

B. Embedments

Three types of embedments are used to maintain the leaktightness of the steel membrane while transferring loads across the mat liner plate to the concrete mat. One is a 3 x 6 rectangular forged bar also called a bridging bar, another is a 1.25 inch thick plate, and the other is a 5 inch thick forged plate to which the neutron shield tank is mounted. Leak test channels are welded all around the embedments to ensure the leaktightness of the steel membrane. Vertical reinforcing steel is Cadwelded to the top and bottom of the embedments providing reinforcing bar continuity without creating multiple penetrations.

C. Insert Plates

Loads from supports for piping such as the spray headers and other miscellaneous equipment are transferred to the containment concrete wall through insert plates and their anchors. Each insert plate is designed to provide a rigid base for the attached supports and anchor studs. Sufficient insert plate anchorage is provided so that the adjacent liner plate sees negligible stress from the applied support loads and the leak tight integrity is well maintained.

D. Penetrations

Penetrations are used for personnel and equipment access, process piping, electrical service, and a mechanical fuel transfer system through the containment wall (refer to Figures 3.8–15 through 3.8–22). Containment penetrations are anchored and transfer loads to the reinforced concrete containment wall. Leak test channels for repaired penetrations are partially cut and not replaced. These channels are welded over all seams between the penetration sleeve, and reinforcement plate and reinforcement plate to liner. These penetrations are classified as follows:

1. Sleeved Piping Penetration

   These penetrations have a sleeve around the outside of forged piping with integral flued head. Sleeved penetrations are used for multiple small pipes passing through one penetration and for thermally hot piping systems. Thermally hot piping is insulated to prevent the operating temperature of the concrete adjacent to the sleeve, during normal operation or any other long-term period, from exceeding 150°F except at local areas around the penetrations which are allowed to have increased temperatures not exceeding 200°F; for accident or other short-term periods, the temperatures are not to exceed 350°F for the interior surface. However, local areas are allowed to reach 650°F from steam or water jets in the event of pipe failure. Penetrations in which the insulation would be insufficient to maintain the concrete within the allowable temperature limit are equipped with a cooling jacket located inside the sleeve. The cooling water for the cooling jacket is supplied by the component cooling water subsystem. Each penetration sleeve carrying thermally hot piping is designed with adequate space between the sleeve and the piping to allow for the required pipe insulation and for the cooling jacket. Piping located outside the containment carrying the cooling water does not require any secondary penetration of the containment structure. The thermally hot piping is connected to the sleeve by a forged flued headed pipe providing a transition from the pipe to the sleeve and designed so that stress concentrations are minimized. The penetration sleeve is welded to a liner reinforcement plate (Figure 3.8–16). Loads are transferred from piping to the containment structure through the forged flued head to the sleeve and liner reinforcement plate. The forging is designed so that the heat flow from the pipe to the sleeve is at a minimum and structurally adequate to take the pipe load. Shear bars are provided on the outside of the sleeve to carry torsional pipe loads. Multiple small piping penetrations pass through a forged attached plate which is welded to a sleeve (Figure 3.8–17).

2. Unsleeved Piping Penetrations

   These penetrations consist of piping installed through the containment wall that are thermally cold piping systems and only one pipe is passing through the penetration. The process pipe is welded directly to the reinforcement plate (Figure 3.8–18).

3. Electrical Penetrations

Electrical penetrations (Figure 3.8–19) are used to carry electrical cables and instrumentation leads through the containment wall. hey range in size from small thermocouple leads to solid rods for power circuits. Each penetration required either a 12 inch or an 18 inch diameter steel sleeve. The sleeves are welded to liner reinforcement plates. The electrical leads are installed in the penetration assemblies which are mounted to the pipe sleeve by a welded flange. Individual conductors can be replaced without cutting the containment liner or sleeve. The penetrations are constructed and tested in accordance with IEEE Standard 317 dated 9-20-72, "Electrical Penetration Assemblies in Containment Structures for Nuclear Generating Stations." Each installed penetration is periodically tested for leak tightness.

4. Fuel Transfer Tube (Mechanical Transfer System)

This penetration is provided for fuel transfer between the containment structure and the fuel building. The penetration consists of a stainless steel pipe installed inside a stainless steel enclosure with bellows expansion joints to compensate for differential movements of the buildings. The inner pipe acts as the transfer tube and connects the containment refueling canal with the spent fuel pool in the fuel building via the fuel transfer canal. The enclosure is welded to the containment liner. The bellows were selected to withstand thermal expansion differentials, seismic motions, and radial and axial differential movements of the fuel building and the containment structure. The enclosure has a tap which provides means for leak testing the system. A blind flange is provided on the inner tube of the containment side and a valve on the fuel building side (Figure 3.8–20).

5. Personnel Air Lock

The personnel air lock is a double closure penetration (Figure 3.8–21). Each closure head is hinged and double gasketed with a leakage test tap between the "O" rings. The enclosed space between the "O" rings is pressurized to containment design pressure to test for leakage through the access door when it is locked in place. The personnel access lock can be independently pressurized up to containment design pressure for testing. Both doors are hydraulically latched and hydraulically swung. Both doors are interlocked so that in the event one door is opening the other cannot be actuated. Both doors are furnished with a pressure equalizing connection. The equalizing valves are manually and automatically operated by the person entering or leaving the personnel access lock. The personnel access lock is externally protected from tornado missiles by concrete shield walls and a roof.

6. Equipment Hatch

The equipment hatch is a single closure penetration (Figure 3.8–22). The equipment hatch cover is mounted inside the containment structure and is double gasketed with a leakage test tap between the "O" rings. The enclosed space between the "O" rings is pressurized to containment design pressure to test for leakage through the access door when it is bolted in place. The equipment hatch cover is provided with a hoist with two point suspension and a sliding rail for storage. A positive locking device is furnished to prevent circular swing. A removable concrete tornado missile shield protects the equipment hatch. These concrete shield blocks are removed during Modes 5 and 6. Based on a probabilistic analysis, the mean value of a tornado missile impacting the equipment hatch during Modes 5 and 6 is on the order of 10^{-7} per year.

3.8.1.1.5 Ring Girder

A reinforced concrete ring girder is provided which encircles the containment structure and is isolated from the containment wall by a compressible material (Figure 3.8–23). The ring girder is provided to prevent postulated sliding of rock wedges toward the containment wall during a seismic event. Sliding would occur when seismic forces exceeded the frictional forces on the rock joint and/or foliation planes. The ring is analyzed as a laterally loaded and supported ring, in which only inward unrestrained deflection is possible. This is because the rock itself resists lateral outward movement. The ring girder does not interact with the containment structure except that the containment mat gives vertical support for the ring gravitational forces. In the area of the ESF building, radial walls in the lower chambers act as struts between the ring and the ESF building east wall (Figure 3.8–24). Through these walls, relatively continuous support for the ring is provided around this area of the containment. Lateral displacements of the ring are calculated to be a maximum of 0.2 inch, which is negligible with respect to the 4 inches of compressible material which is provided to isolate the containment shell from any lateral loads above the mat. The ring girder is a seismic Category I structure and is designed for all applicable loadings including: dead, seismic, and rock pressure loads. Loads and load combinations are in accordance with Section 3.8.1.3.

The allowable shear stress for the design of the ring girder is in accordance with ACI 318-71 because no circumferential tension exists in the ring. Materials and quality control are in accordance with Section 3.8.1.6. There are no testing or inservice inspection requirements.

3.8.1.2 Applicable Codes, Standards, and Specifications

Structural design, materials, the tests of material and the methods of testing, where applicable, conform to the following codes, standards, and specifications unless otherwise stated.

3.8.1.2.1 General

The letters ASTM, AISC, AWS, ACI, and BOCA refer to the American Society for Testing and Materials, American Institute of Steel Construction, American Welding Society, American Concrete Institute, and Building Officials and Code Administrators, respectively.

Where multiple dates are shown for structural specifications in Section 3.8.1.2.2, these reflect documents whose applicable issue has changed during construction. The engineering specifications reflect the specific issue date in effect at any given time.

3.8.1.2.2 Structural Specifications

1.	ACI 211.1-70	Recommended Practice for Selecting Proportions for Normal Weight Concrete
2.	ACI 214-65	Recommended Practice for Evaluation of Compression Test Results of Field Concrete
3.	ACI 301-72	Specification for Structural Concrete for Buildings
4a.	ACI 614-59	Recommended Practice for Measuring, Mixing, and Placing Concrete
4b.	ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
5.	ACI 305-72	Recommended Practice for Hot Weather Concreting
6.	ACI 306-66	Recommended Practice for Cold Weather Concreting
7.	ACI 318-71	Building Code Requirements for Reinforced Concrete
8.	ACI 347-68	Recommended Practice for Concrete Formwork
9.	AISC	Specification for the Design, Fabrication and Erection of Structural Steel for Buildings Seventh Edition (February 12, 1969); Supplement No. 1 (November 1, 1970), Supplement No. 2 (December 8, 1971), and Supplement No. 3 (June 12, 1974)

NOTE: AISC Eighth Edition (1980) used in design of containment building enclosure structure

10.	AISC	Specification for Structural Joints Using ASTM A325 or A490 Bolts (April 18, 1972)
11.	ASTM A 36-74 (1977)	Specification for Structural Steel

NOTE: ASTM A36-77 Used for fabrication of shield doors

12.	ASTM A 193-73	Standard Specification for Alloy Steel and Stainless Steel Bolting Materials for High-Temperature Services
13.	ASTM A 307-74	Specification for Low Carbon Steel Externally and Internally Threaded Standard Fasteners
14.	ASTM A 325-66, 74	Specification for Low Carbon Steel Externally and Internally Threaded Standard Fasteners
15.	ASTM A 440-74	Specification for High Strength Structural Steel
16.	ASTM A 441-74	Specification for High Strength Low-Alloy Structural Manganese Vanadium Steel

17.	ASTM A 490-66, 74	Specification for Quenched and Tempered Alloy Steel Bolts for Structural Steel Joints
18.	ASTM A 588-79	Specification for High-Strength Low-Alloy Structural Steel with 50,000 psi Minimum Yield Point to 4 Inches Thick
19.	ASTM A 615-72	Standard Specification for Deformed Billet Steel Bars for Concrete Reinforcement including Supplement S-1
20.	ASTM C 31-69	Making and Curing Concrete Compressive and Flexural Strength Test Specimens in the Field
21.	ASTM C 33-71a	Standard Specification for Concrete Aggregates (and 1978 Revision)
22.	ASTM C 94-71 (1974)	Specification for Ready-Mixed Concrete
23.	ASTM C 109-1973	Method of Test for Compressive Strength of Hydraulic Mortars (using 2-inch (50 mm) Cube Specimens)
24.	ASTM C 143-7	Method of Test for Slump of Portland Cement Concrete
25.	ASTM C 150-73	Specification for Portland Cement
26.	ASTM C227-71	Test for Potential Reactivity of Cement Aggregate Combinations (Mortar Bar Method)
27.	ASTM C 233-69 (1973)	Standard Method of Testing Air-entraining Admixtures for Concrete
28.	ASTM C 235-68	Test for Scratch Hardness of Coarse Aggregate Particles
29.	ASTM C 260-69 (1973, 1974)	Air-entraining Admixtures for Concrete
30.	ASTM C 289-71	Test for Potential Reactivity of Aggregates (Chemical Method)
31.	ASTM C 295-1965 (1973)	Recommended Practice for Petrographic Examination of Aggregates for Concrete
32.	ASTM C 586-69	Test for Potential Alkali Reactivity of Carbonate Rocks for Concrete Aggregates
33.	AWS D1.1-72 Rev. 1-73 (1979)	Structural Welding Code

NOTE: AWS D1.1-79 used for fabrication of watertight, airtight, pressure resistant, and shield doors.

34.	AWS D12.1-61	Recommended Practices for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction

35. NRC Regulatory Guides as qualified in Section 1.8 on the following topics:

a.	Cadweld Splices	1.8.1.10
b.	Reinforcing Bar Testing	1.8.1.15
c.	Structural Acceptance Testing	1.8.1.18
d.	Placement of Concrete	1.8.1.55
e.	Design Response Spectra	1.8.1.60
f.	Seismic Damping Values	1.8.1.61

36.	BOCA	Basic Building Code of the Building Officials and Code Administrators International, Inc., 1970
37.		State of Connecticut Basic Building Code, 197

3.8.1.2.3 Steel Liner and Penetrations

There was no applicable code for the design of concrete containment structure liners at the beginning of the construction of the Millstone liner. However, ASME Section III, Divisions 1 and 2, and Section VIII were used as a guide.

Design, materials, fabrication, testing, and inspection, where applicable, conform to the following codes, standards, and specifications:

a. ASME Boiler and Pressure Vessel Code Sections II, III, and V, 1971 issue, including addenda up to and including the 1973 Summer addendum, and Section III, Division 2, Subsection CC, 1980 issue, including addenda up to and including Summer 1982.

b. ASME Boiler and Pressure Vessel Welding Qualifications, Section IX, issue in effect at the time of qualification.

c. American National Standards Institute, ANSI N101.4, Quality Assurance for Protective Coatings Applied to Nuclear Facilities, 1972 issue.

d. American Society for Testing and Materials, ASTM-E208, Conducting Drop Weight Test to Determine Nil Ductility Transition Temperature of Ferritic Steels, 1969 issue.

e. American Society for Testing and Materials, ASTM E436, Drop Weight Tear Tests of Ferritic Steels, 1971 issue.

f. American Welding Society Structural Welding Code D1.1, 1972 issue.

g. Institute of Electrical and Electronics Engineers, IEEE 317, Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations, February 4, 1971 issue.

h. Steel Structures Painting Council, Paint Applications Specification, SSPC-PA1, Shop and Field Maintenance Painting 1964 issue.

i. Steel Structures Painting Council, Surface Preparation Specifications SP1, SP2, SP5, SP6, and SP10, 1963 issue.

j. Steel Structures Painting Council, Surface Preparation Specification, SSPC-PS8.01, Rust Preventive System with Thick Film Compounds, 1964 issue.

k. American Society for Nondestructive Testing, Recommended Practice for Nondestructive Testing Personnel, Qualifications, and Certification, ASNT-TC-1A, 1971 issue.

l. American Society for Testing and Materials (ASTM) Specifications A105, A108, and American Society of Mechanical Engineers (ASME) Material Specifications SA131, SA182, SA193, SA194, SA213, SA240, SA234, SA312, SA320, SA333, SA350, SA376, SA403, SA496, SA508, SA516, and SA537.

Earlier or subsequent revisions (or reissues) of the referenced codes, standards, and specifications may be used insofar as their use does not in any way degrade the safety or strength of any structure or portion of a structure.

Section 3.8.1.6 describes materials and quality control procedures and any exceptions to the above listed codes.

Fire Protection is described in Section 9.5.1.

3.8.1.3 Loads and Loading Combinations

The following sections discuss the loads and load combinations used in the design of the

3.8.1.3.1 Containment Mat, Shell, and Dome

The loadings applicable to concrete containment design include the following:

a. Those loads encountered during preoperational testing.

b. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, and thermal loads due to operating temperature.

c. Those loads to be sustained during severe environmental conditions, including those induced by the design wind and the operating basis earthquake.

d. Those loads to be sustained during extreme environmental conditions, including those induced by the design basis tornado and the safe shutdown earthquake.

e. Those loads to be sustained during abnormal plant conditions, which include a loss-of-coolant accident (LOCA). The main abnormal plant condition for containment design is the design basis LOCA. Also considered are other accidents involving various high energy pipe ruptures. Loads induced on the containment by such accidents include elevated temperatures and pressures and localized loads such as jet impingement and associated missile impact.

f. Those loads to be sustained after abnormal plant conditions including flooding of the containment subsequent to a LOCA for fuel recovery.

Loads generated by the design basis accident (DBA) are described in Section 6.2.1. The design basis wind and tornado loadings are discussed in Section 3.3. The loadings for groundwater and flood are given in Section 3.4.

Missile loads are described in Section 3.5.1.4.

The design bases for protection against postulated rupture of piping are described in Section 3.6.

The earthquake loadings which include consideration of simultaneous excitation from two (orthogonal) horizontal and one vertical earthquake motion are described in Section 3.7.

Normal operating temperatures are described in Section 9.4.6.

For the containment structure shell and its foundation mat, the load combinations given in Section 9.3.2 of ACI 318 are replaced by the following:

> For service load categories, the structure is designed using stresses well within elastic limits. Specifically, the allowable compressive stress in concrete is 0.45 f'c, the allowable stress for steel is 0.5 fy, and shear stresses are 50 percent of capacity as given in ACI 318. In effect, this permits stress levels approximately 50 percent of those for the factored load conditions. However, structural members subjected to test pressure, temperature, or wind, when combined with other forces, are designed for the allowable stresses increased by 33 percent.

CONTAINMENT LOAD COMBINATIONS AND FACTORS

Case	**Load Category**	**Load Combination**
	Service:	
1	Test	$1.OD + 1.OL + 1.0P_t + 1.0T_t$
2	Construction	$1.OD + 1.OL + 1.OT_o$

Case	Load Category	Load Combination
3	Normal	$1.OD + 1.OL + 1.OT_o + 1.0\ OBE + 1.OR_o + 1.0P_v$
	Factored:	
4	Extreme environmental	$1.OD + 1.OL + 1.OT_o + 1.OW_t + 1.OP_v + 1.OR_o$
5	Extreme environmental	$1.OD + 1.OL + 1.OT_o + 1.0SSE + 1.OR_o + 1.OP_v$
6	Abnormal	$1.OD + 1.OL + 1.5P_a + 1.OT_a + 1.OR_a$
7	Abnormal/Severe environmental	$1.OD + 1.OL + 1.25P_a + 1.OT_a + 1.25OBE + 1.OR_a$
8	Abnormal/Severe environmental	$1.OD + 1.OL + 1.25P_a + 1.OT_a + 1.25W + 1.OR_a$
9	Abnormal/Extreme environmental	1.OD + 1.OL + 1.OPa + 1.OTa + 1.0SSE + 1.ORa

In Case 3, W is substituted for OBE if its effect is greater.

In Cases 6, 7, 8, and 9, Yr is substituted for Ra if its effect is greater.

Nomenclature is as follows:

D = Dead load of structures and contents, including effects of earth and hydrostatic pressures, buoyancy, ice, and snow loads.

SEE = Safe shutdown earthquake (see note).

OBE = Operating basis earthquake (see note).

L = Live load.

P_a = Pressure load from DBA pressure transient, including the design margin, as defined in Section 6.2.1.

P_t = Test pressure.

P_v = Subatmospheric minimum operational pressure of containment structure (or atmosphere).

R_a = Piping loads due to increased temperature resulting from the DBA.

R_o = Pipe loads during normal operating conditions.

T_a = Load due to temperature gradient through the concrete shell and mat plus load exerted by the liner, based on temperature associated with the DBA. This is that transient temperature which, when combined with the coincident internal pressure, produces the most adverse effects on the containment structure.

T_o = Load due to temperature gradient through the concrete shell and mat, plus load exerted by the liner based on temperature associated with normal operation.

T_t = Load from test temperature.

W = Wind load (see note).

W_t = Tornado load (see note).

Y_r = Reaction from pipe restraint (penetration) due to pipe break (local effects).

NOTE: Wind or tornado loads are not coincident with earthquake loads.

A load factor of 1.0 is assigned to loads caused by the design tornado because the wind velocity given in Section 3.3.2 is a conservative estimate of the actual tornado wind velocity to be expected, and the probability of a tornado striking the unit is low (Section 2.3.1.3). A load factor of 1.0 is also used for loads caused by the SSE (Section 3.7B.1) since the SSE acceleration is two times the largest acceleration expected at the site.

Tornado wind effects are determined as described in Section 3.3.

The effects of hydrostatic loading due to groundwater and the probable maximum flood level (Section 3.4.2) are considered so as to reduce the effect of the dead load (D).

During construction, dewatering was maintained until the weight of the containment exceeded the postulated hydrostatic uplift forces.

Section 3.8.1.4 gives the design stresses to be used with these loads.

3.8.1.3.2 Steel Liner and Penetrations

The containment structure liner plate, penetrations, brackets and attachments, anchors, and access openings were designed for the load combinations presented in Tables 3.8–1 and 3.8–2. Sleeves of hot penetrations were considered part of the containment boundary. The maximum containment structure design pressure is 45 psig. The minimum containment structure design pressure is 8.0 psia. This pressure is equivalent to the minimum operating pressure minus the pressure drop due to the maximum, hypothetical containment cooldown situation. During this condition, the containment atmosphere pressure is assumed to be decreased below normal operational pressure by the inadvertent operation of the quench spray system during normal unit operation. The resulting total pressure is above the minimum containment design pressure of 8.0 psia. For a further discussion of external pressure, refer to Section 6.2.1.1.3.5.

The change in barometric pressure due to tornadoes is not expected to exceed 3 psi and the change due to the maximum probable hurricane is approximately 1.1 psi. These pressure changes result in a decrease in the atmospheric pressure, which, in turn, decreases the differential between atmospheric pressure and the containment atmosphere ambient pressure, thereby decreasing the level of stresses in the containment structure. Penetrations were designed to take pipe rupture, thermal, and seismic loads.

Figures 3.8–25 through 3.8–55 are scaled load plots of moments, shears, and membrane forces for the containment structure. The loading conditions are listed on the plots. Compressive forces are plotted as negative values and bending moments are plotted on the tension side.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 Containment Structure

The containment structure consists of a hemispherical dome, a cylindrical shell, and a circular mat supported by an elastic subgrade. The design, analyses, and construction of the containment structure is similar to and takes full advantage of SWEC experience gained in the designs for plants indicated in Section 3.8.1.1.

Under design internal pressure, discontinuity forces exist at the junction of the mat and the cylindrical shell and also at the junction of the shell and the dome. The cylindrical shell is of such a length that the influence of one discontinuity on the other is negligible.

The analysis of the mat for axisymmetric loading is accomplished using the program MAT 5 which is described in Appendix 3A. This program treats the mat as a symmetrically loaded circular plate on an elastic foundation. The general method is described by Zhemochkin et al., (1962). The program is set up so that the foundation stiffness can be formulated either through Boussinesq's approach or through a Winkler-type assumption. Zhemochkin's method is modified to account for a finite depth of elastic foundation, i.e., the distance between mat and underlying rock. In addition, the cylindrical containment wall, crane wall, and primary shield wall are considered as elastic constraints, which are determined by applying compatibility conditions at the shell mat interface.

For the purpose of calculating the elastic constraint of the containment, the base of the cylinder is assumed to be completely cracked vertically and cracked horizontally to the neutral axis of the transformed section. At this location, the cylinder has a hoop stiffness of the circumferential rebars and the meridional bending stiffness of the transformed section.

A short distance above the mat, the meridional bending moment becomes so small that the entire meridional cross-section is in a state of tension. Above this plane, the cylinder is assumed to be completely cracked horizontally and vertically.

Thus, the elastic constraint is determined from a cylinder which is divided into a short shell having the properties of a cross-section completely cracked vertically and partially cracked horizontally, and a long shell having the properties of a cross-section completely cracked horizontally and vertically.

Seismic analysis of the containment structure described in Section 3.7B.2 provides the dynamic loads imposed on the mat as static loads. Since these loads are asymmetric, the mat is analyzed using the finite difference computer program SHELL I.

The discontinuity forces at the mat-shell junction, calculated as a part of the mat analysis, are used as boundary conditions for the analysis of the cylindrical shell which is performed using the program SHELL I. The seismic analysis of the containment structure provides the accelerations to which the containment structure would be subjected. These accelerations are used for determining static loading on the shell. The tangential shear caused by the seismic loading is resisted by the concrete or by the concrete and a system of diagonal rebars. The maximum allowable tangential shear stress carried by the concrete, (Vc) is assumed to be 40 psi. Stresses in excess of Vc are resisted by diagonal reinforcing bars. The specified design compressive strength of the concrete carrying the tangential shear is not less than 3,000 psi with coarse aggregate not smaller than size No.67 as given in ASTM C33.

The diagonal reinforcement required to supplement the tangential shear capacity of the concrete consists of No. 18 rebars anchored in the mat as shown in Figure 3.8–56. The spacing between these diagonal bars is increased as the design tangential shear decreases at higher wall elevations. No diagonal rebars are required above the spring line.

The requirements for diagonal reinforcement necessary for carrying the (Vu-Vc) shear force are determined by an analysis based on a paper by Prof. M. J. Holley, Jr. (1969). The mesh of vertical, horizontal, and diagonal reinforcement is assumed to be a continuum. Forces can be determined in each type of bar for symmetric loadings such as internal pressure, dead load, vertical, and horizontal earthquake loads. The diagonal reinforcement is designed to resist force due to symmetric loadings and the entire tangential shear not carried by the concrete (Vu-Vc).

Two temperature conditions are considered in the analysis of the containment:

1. Temperature under normal operating condition.

2. Temperature associated with the DBA, when combined with the coincident internal pressure, produces the most adverse effects on the reactor containment structure.

Under normal operating conditions, the temperature gradient to produce the highest stress resultant is used.

The effect of the liner temperature increase associated with the DBA condition on the concrete containment shell is determined by the following procedure. For this condition, it is assumed that the temperature of the liner increases and the concrete remains at its ambient temperature. Liner expansion is limited by the concrete shell and a pressure develops between the concrete shell and the liner. The equivalent pressure exerted by the liner on the concrete shell is given by the expression:

P_e = LH*hL/RL

where:

P_e = Equivalent pressure

LH = Circumferential liner stress

hL = Thickness of liner

RL = Radius of liner

At the junction of the mat and shell, it is assumed that the mat prevents radial movement of the shell; therefore, at this location, the circumferential stress in the liner used to determine equivalent pressure is:

LH = alpha*Es*delta T

where:

alpha = Coefficient of thermal expansion of the liner

Es = Young's modulus of the steel liner

delta T = Change in temperature due to DBA

A short distance above the mat, where the effect of the mat to shell discontinuity is negligible, the liner and concrete shell expand due to the DBA pressure and temperature of the liner. Free radial displacement of the liner due to DBA temperature is larger than the displacement of the reinforced concrete shell due to the DBA pressure. Thus, the liner is constrained by the concrete shell. The resulting pressure exerted by the liner is determined from the expression for equivalent pressure given before. The liner stress is determined by the following procedure:

Expressions for stresses in the shell and liner are written in terms of the meridional and circumferential strains. These are inserted into the force equilibrium equations resulting in an explicit solution for liner strains from which liner stresses are determined.

The effects of creep and shrinkage of concrete are not important considerations in the analysis and design of a nonprestressed concrete containment. Shrinkage results in meridional and radial displacements which are the opposite of the displacements caused by the principal loads, temperature and internal pressure. Consequently, the effects of creep and shrinkage can be safely ignored.

Cracking is an important consideration in the analysis and design of a reinforced concrete containment. For this reason, stiffness of the concrete is adjusted for the extent of cracking present under various design conditions. When the concrete is completely cracked, calculation of the stiffness of the structure uses only the properties of the reinforcing steel. The steel liner is assumed to make no contribution to the structural integrity of the containment shell except during the structural acceptance test.

The penetrations through the containment wall are grouped into the following three categories for the purposes of analysis and design:

1. 12-inch diameter (nominal) or less

   No special or additional reinforcing is provided. The principal wall reinforcement is located to avoid interference with the penetration.

2. All piping penetrations larger than 12-inch diameter (nominal).

   Reinforcing bars terminated at penetrations are replaced by at least twice the number of bars, half of these being placed on each side of the opening. Diagonal reinforcing bars are also provided around openings to take shear and tension. The anchorage length of the additional bars that frame the openings is determined by using a conservative value for bond stress. In addition, shear assemblies as shown in Figure 3.8–5 are added around 48-inch penetrations. This method is consistent with established practice and pressure-tested at the plants previously listed.

3. Personnel Access and Equipment Access Hatches

   Penetrations for the equipment and personnel access hatches are analyzed using the 3-dimensional finite element capability of the computer program STRUDL II which is discussed in Appendix 3A.

   The thickened ring beam and cylinder wall for both hatches are assumed to be cracked, and to have the extensional stiffness of the reinforcing bars only. The analysis shows that sizable tangential (in plane) shears exist in the wall near the ring beam. These shears are resisted by reinforcing bars which are placed parallel to the typical earthquake shear bars.

   The ring beam is designed to resist the axial tension and shears resulting from the loading criteria listed in Section 3.8.1.3. The axial tension is assumed to be resisted by the reinforcing bars only. The shears, including torsional shear, are resisted entirely by stirrups placed radially around the penetrations.

   In effect, any concrete resistance to tension and shear is neglected. The principal circumferential and meridional reinforcing bars, as designed, are extended to the inner face of the ring beam, hooked and Cadwelded to each other, thereby providing shear resistance additional to that provided in the design.

   The normal pattern of membrane forces and moments (meridional and circumferential) in the containment wall is disrupted in the region of the hatch openings. The redistribution of these forces and moments is provided by the finite element computer program and extra reinforcement is added to areas of marked deviation from the normal pattern.

3.8.1.4.2 Steel Liner and Penetrations

Stresses due to strain compatibility of the liner with the reinforced concrete shell due to various combinations of pressure, thermal, self-weight, and seismic load were determined using Stone & Webster's "KALNINS" program. This is a direct integration program for static analysis of multilayered thin shells of revolution. The stress analysis of a shell subjected to mechanical and thermal surface loads and edge loads, is reduced to a boundary value problem governed by a system of nonhomogeneous, linear, partial differential equations. The equations are separable with respect to the meridional and circumferential coordinates of the shell. The solution for each separable component of the loads is obtained by solving a typical two point boundary value problem governed by eight first order linear ordinary differential equations using direct integration. Local stresses due to irregular spacing of liner-headed anchor studs were determined using the ANSYS finite element program and by manual calculations. Analytical evaluation of the penetration discontinuities were modeled on the ASAAS program (Asymmetric Stress Analysis of Axisymmetric Solids). The method of analysis employed is based on a finite element idealization of an axisymmetric solid. Each element is an axisymmetric ring of a constant cross section. Since such a solid may be loaded and may deform in nonaxisymmetric modes and since the properties of the material may vary in all directions (e.g., due to temperature variations), all the dependent variables including the material properties are expressed as truncated Fourier series with the circumferential coordinate being the independent variable.

Temperature profiles for the penetrations were determined using the TAC-2D program. TAC-2D is a computer program for calculating steady-state and transient temperatures in two dimensional problems by the finite difference method. The configuration of the body to be analyzed is described in the rectangular, cylindrical, or circular (polar) coordinate system by orthogonal lines of constant coordinate called grid lines. The grid lines specify an array of nodal elements. Nodal points are defined as lying midway between the bounding grid lines of these elements. A finite difference equation is formulated for each nodal point in terms of its capacitance, heat generation and heat flow paths to neighboring nodal points.

3.8.1.5 Structural Acceptance Criteria

3.8.1.5.1 Containment Structure

The containment structure is designed for the loads and load combinations presented in Section 3.8.1.3.1. Allowable stresses, unless otherwise defined, are in accordance with ACI 318-71. For the factored load combinations, design of the containment structure meets the broad intent of Article CC-3400 of ASME III Division 2. Details of the design conform to ACI 318-71 and the additional requirements discussed in Section 3.8.1.4, rather than the parallel requirements of CC-3000. Major features of the concrete design are similar using either code.

Except for test conditions, the specific limits for service loads given in CC-3430 are not addressed as acceptance criteria for ACI 318-71. However, design of the containment equals or exceeds ACI 318-71 requirements for serviceability. Predicted stresses and strains for structural acceptance tests are well within the limits stated in CC-3430.

The tangential shear stress, Vu, resulting from earthquake loading is resisted by the concrete and by diagonal reinforcing bars. As discussed in Section 3.8.1.4.1, the maximum allowable tangential shear stress, Vc, carried by the concrete is assumed to be not greater than 40 psi.

3.8.1.5.2 Steel Liner and Penetration

The containment structure liner plate, brackets, attachments, anchors, and access openings are designed to meet the design allowables presented in Table 3.8–1. Initial penetration sizing is performed in accordance with Table 3.8–2. The final design verification is in accordance with Tables 3.8–4 through 3.8–6.

To minimize the probabilities of crack propagation as the containment is exposed to the design conditions as indicated in Table 3.8–1, stress levels reached are kept within the limits given by Pellini and Loss in NRL Report 6900, Integration of Metallurgical and Fracture Mechanics Concepts of Transition Temperature Factors Relating to Fracture-Safe Design for Structural Steels.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

Section 3.8.1.2 contains applicable codes, standards, and specifications.

The quality assurance activities required by this section are described in Section 17.1.

3.8.1.6.1 Concrete

Materials and Quality Control

ACI 301, ACI 347, and ACI 318 form the general basis for the concrete specifications.

ACI 301 was supplemented, as necessary, with mandatory requirements relating to types and strengths of concrete, including minimum concrete densities, proportioning of ingredients, reinforcing steel requirements, joint treatments, and testing agency requirements.

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 were specified in detail.

All Portland cement conformed to requirements of ASTM C150 for Portland cement, Type II. Low alkali Portland cement was used where examinations and tests of aggregates, or of structures containing similar aggregates, indicate potential reactivity with the cement. One exception was that calcium aluminate (high-alumina) cement was used for porous concrete enclosed within the waterproof membrane under the containment mat. This substitution is made to reduce the possibility of clogging of voids by continued hydration that occurs with Portland cement. Certified copies of mill tests, showing that the cement meets or exceeds the ASTM requirements for Portland cement, were furnished by the manufacturer. An independent testing laboratory was retained to perform tests on the cement to substantiate the mill test reports and compliance with the specifications for the cement.

An air-entraining admixture was used in the concrete. The total air content, expressed as a percentage by volume, conforms to the requirements in Table 3.4.1 of ACI 301, Total Air Content for Various Sizes of Coarse Aggregates for Normal Weight Concrete. This admixture conforms to the requirements of ASTM C 260 when tested in accordance with ASTM C 233. The air-entraining agent was added separately to the batch in solution. The solution was batched by means of a mechanical dispenser capable of accurate measurement and was subject to periodic checks. Air-entrained cement was not used.

Other admixtures to control the rate of set, reduce the water content, or improve the workability and cohesiveness of concrete were used in specific instances. Such admixtures were used only after tests were made in combination with the cement and aggregates being used and were specifically approved by the structural engineer. Calcium chloride was not used under any circumstance.

Mixing water and ice were clean and free from injurious amounts of oils, acids, alkalies, salts, organic materials, or other substances which may be deleterious to concrete or steel. The mixing water and water used for making ice are periodically checked and tested for suitability by ASTM C 109 Method of Test for Compressive Strength of Hydraulic Cement Mortars (using 2-inch (50-mm) Cube Specimens).

Fine and coarse aggregates conform to the requirements of ASTM C 33. Coarse aggregates contained not more than 5 percent soft fragments in accordance with ASTM C 235. Aggregates were evaluated for potential chemical alkali reactivity prior to use by ASTM C 295, Recommended Practice for Petrographic Examination of Aggregates for Concrete, and ASTM C 289, Test for Potential Reactivity of Aggregates (Chemical Method). Also, prior to use of an aggregate, one of the following tests for alkali reactivity (whichever is appropriate for the aggregate) was started: ASTM C 227, Test for Potential Reactivity of Cement Aggregate Combinations (Mortar Bar Method), or ASTM C 586, Test for Potential Alkali Reactivity of Carbonate Rocks for Concrete Aggregates. Test results were evaluated for aggregate suitability with the cement. Aggregates were free from any materials that would be deleteriously reactive in any amount sufficient to cause excessive expansion of mortar or concrete. All aggregates were tested for compliance with the above requirements by an independent testing laboratory.

Proportioning of structural concrete conforms to ACI 301, Chapter 3. In general, concrete mixes were of a 28 day strength of 3,000 psi unless otherwise specified by the Engineers.

Concrete used for biological shielding has a density of not less than 140 lb/cu ft.

Proportioning of ingredients in concrete mixes were determined and tests conducted in accordance with the methods detailed in ACI 301 and ACI 211.1 for combinations of materials to be established by trial mixes.

Slump of mass concrete is in accordance with ACI 301, Chapter 3.

Concrete protection for reinforcement, preparation and cleaning of construction joints, concrete mixing, delivering, placing, and curing met or exceeded the requirements of Regulatory Guide 1.55 with the exceptions given in Section 1.8.

Batching and mixing conformed to ACI 301, Chapter 7. Placing of concrete was by bottom dump buckets, chuting, concrete pump, or by conveyor belt. Aluminum pipe was not permitted for pumping concrete. The rate of placing concrete was controlled so that concrete was effectively placed and compacted with particular attention given around embedded items and near the forms.

Vertical drops greater than 5 feet for any concrete was not permitted, except where suitable means were provided to prevent segregation. Placing equipment and methods were reviewed for compliance with the specifications.

The ACI and ASTM specifications were supplemented as necessary with mandatory requirements relating to types and strengths of concrete, minimum concrete density, proportioning of ingredients, reinforcing steel requirements, joint treatments, testing requirements, and quality control.

Curing and protection of freshly deposited concrete conform to Chapter 12 of ACI 301, with the following supplementary provisions:

1. Concrete cured with water was kept wet by covering with an approved water saturated material, by a system of perforated pipes or mechanical sprinklers, or by any other approved methods which kept surfaces continuously wet.

2. The surfaces on which curing compounds may be used were specified by the structural engineer. Curing compounds whose base is composed of sodium silicate, magnesium fluosilicate, or zinc fluosilicate were used on surfaces to which additional concrete was to be bonded, except those surfaces specifically requiring water curing. Other curing compounds were not used on surfaces to which additional concrete is bonded.

Construction joints that were to be bonded or transferring shear through shear friction (per ACI-318) were properly prepared as follows.

After the initial concrete set had occurred, but before the concrete had reached its final set, the surfaces of these construction joints are thoroughly cleaned to remove all laitance and to expose clean, sound aggregate. After cutting, the surface was washed and rinsed. All excess water which was not absorbed by the concrete was removed.

Where, in the opinion of the field engineer, the use of an air-water jet was not advisable, then that surface was roughened by bushhammering, sand blasting, or other satisfactory means to produce the requisite clean surface. Horizontal construction joints were covered by a 0.5 inch thick layer of sand/cement grout and new concrete then placed immediately against the fresh grout.

Concrete strength tests were performed in accordance with Chapter 16 of ACI 301, (Section 16.3/ 4.6.1.7). One strength test being made for each 100 cubic yards or fraction thereof for each mix design of concrete placed in any one day.

The test specimens for compressive strength were 6 inch diameter by 12 inch long cylinders conforming to ASTM C 31.

When required, concrete strength tests were evaluated on a statistical basis by the engineers in accordance with ACI 214, Recommended Practice for Evaluation of Compression Test Results of Field Concrete, and Chapter 17 of ACI 301. The strength level of the concrete was considered satisfactory if it conformed to Section 4.3.3 of ACI 318.

The field tests for slump of Portland cement concrete followed ASTM C 143, “Method of Test for Slump of Portland Cement.” Any batch not meeting specified requirements was rejected. Slump tests were made frequently during concrete placement and each time concrete test specimens were made.

Shop detail drawings for the reactor containment mat, shell, and dome reinforcement were checked by the designer.

Special Construction Techniques

No special construction techniques were used in the placing of concrete for the containment structures.

3.8.1.6.2 Reinforcing Steel

Materials

N14 and N18 reinforcing bars are controlled chemistry steel of 50,000 psi minimum yield point. They conform to Grade 40 of the “Standard Specification for Deformed Billet-Steel Bars for Concrete Reinforcement” ASTM A 615, as modified, to meet the following chemical and physical requirements:

Carbon	0.35 percent max
Manganese	1.25 percent max
Silicon	0.15 to 0.25 percent
Phosphorous	0.05 percent max
Sulfur	0.05 percent max
Yield strength	50,000 psi min
Elongation	13 percent min in an 8-in test sample
Tensile strength	70,000 psi min

Reinforcing bars smaller than N14 are grade 40 or grade 60 conforming to ASTM A 615.

Cadweld T-Series reinforcing steel splices are full tension splices manufactured by Erico Products, Inc., Cleveland, Ohio, and are used to splice N14 and N18 reinforcing bars. In restricted areas, reinforcing bars were butt welded in a manner conforming to the requirements of AWS D12.1. Cadweld splices were made in accordance with the instructions for their use issued by the manufacturer, Erico Products, Inc.

Reinforcing bars smaller than N14 were generally lap spliced. Where lap splicing was impractical, splicing is accomplished by:

1. Use of mechanical (Cadweld) splices as manufactured by Erico Products, Inc., Cleveland, Ohio, or equivalent, using the T-series sleeves that develop the full tensile strength of the reinforcing bars, or

2. Butt welding in accordance with the requirements of AWS D12.1.

Quality Control

1. Reinforcing Bars

   For the special chemistry N14 and N18 bars used in the containment structure, ingots and billets were traced with identifying heat numbers. Bundles of bars were tagged with a heat number as they come off the rolling mill. special mark was rolled into bars conforming to special chemistry N14 and N18 bar to identify them as possessing the chemical and mechanical qualities specified.

   Testing of reinforcing bars for Seismic Category I concrete structures met the requirements of Regulatory Guide 1.15, as described in Section 1.8.

   SWEC inspectors witnessed, on a random basis, the pouring of the heats and the physical and chemical tests performed by the manufacturer of the special chemistry reinforcing bars.

   Bars containing unacceptable inclusions or failing to conform to the required chemistry and physical requirements were rejected.

   Mill tests reports showing actual chemical ladle analysis, tensile properties, bend properties, variations in weight, and conformance of deformations were obtained from the manufacturer for each heat. In addition, confirmatory tests for each 50 tons or fraction thereof of each heat of steel for each bar size were made to determine tensile properties. Further, for the special chemistry bars an actual chemical check analysis of each heat was made in addition to confirm the chemical content.

Full size test specimens of all reinforcing bars were tested on a testing machine using an 8-inch gage length.

2. Cadweld Splices

Splicing complied with the requirements of Regulatory Guide 1.10, with the exceptions given in Section 1.8.

3. Butt-Welded Splices

The ends of the bars to be joined by butt-welding were prepared by sawing or flame cutting, and dressing by grinding, where necessary. Welders were qualified in accordance with AWS D12.1.

All welds were visually inspected. Any cracks, porosity, or other defects were removed by chipping or grinding until sound metal was reached, and then repaired by welding. Peening was not permitted.

Completed reinforcing steel butt-welded splices were selected on a random basis from the containment structure and tensile tested in accordance with the following frequency for each welder:

a. One out of first 10 splices.

b. One out of next 90 splices.

c. Two out of the next and subsequent units of 100 splices.

In addition, completed reinforcing steel butt-welded splices were selected on a random basis from the containment structure and radiographically inspected to meet the following frequency for each welder:

a. One out of the first 10 splices.

b. One out of the next and subsequent units of 25 splices.

Reinforcing steel bars butt welded to steel plate were tested by sister splice, in accordance with the following schedules:

a. One sister splice out of the first 10 production splices.

b. Four sister splices for the next 90 production splices.

c. Three sister splices for the next and each subsequent units of 100 production splices.

CONSTRUCTION TECHNIQUES

1. General

Placing of reinforcing steel, in general, conformed to the requirements of Chapter 5 of ACI 301 and Chapter 7 of ACI 318.

Section 3.8.1.1 describes the placing of the reinforcing steel for the containment structure.

Tack welding of designed reinforcing steel was not permitted.

2. Special

No special construction techniques were used in the installation of reinforcing steel for the containment structure.

3.8.1.6.3 Structural Steel

Material

Steel specifications invoked for structural framing, brackets, and attachments are discussed in Section 3.8.1.2.2.

Quality Control

All main members, columns, baseplates, bracing, trusses, girts, and bolts larger than 1 inch in diameter are traceable to a specific heat number. Traceability to a specific heat number for all clip angles, seats, stiffeners, gusset plates, bolts of 1 inch diameter and smaller, and weld filler metal is confirmed in the suppliers' shop or upon receipt at the site. The storage and issuance of these materials for construction is controlled in a manner which assures only those items procured as QA Category I are installed in QA Category I applications.

Construction Techniques

Structural steel material, erection, and fabrication tolerances are in accordance with the AISC Specification for the Design Fabrication, and Erection of Structural Steel for Buildings.

Welding of structural steel is in accordance with AWS D 1.1-72, Revision 1-73.

3.8.1.6.4 Waterproofing Membrane

A waterproofing membrane (Figure 3.8–57) was placed below the containment structure mat and Engineered Safety Features Building and carried up the containment wall and Engineered Safety Features Building walls to above groundwater level. Attached to and entirely enveloping the part of the containment structure and Engineered Safety Features Building below ground level, the membrane protects the structures from the effects of groundwater and the steel liner from external hydrostatic pressure. When water penetrates or otherwise circumvents the membrane, the water drains to a layer of porous concrete directly below the mats and above the membrane. This layer of porous concrete serves as a horizontal drain under the entire containment structure and Engineered Safety Features Building. The porous layer is drained into the Engineered Safety Features Building. A non safety-related pump is used as necessary to remove the water during normal plant operation, post-LOCA and LNP conditions (see Section 9.3.3.2.4.1 for additional information).

A standpipe assembly has been installed through the waterproofing membrane extending to the floor at elevation (-) 34 feet 9 inches in the Engineered Safety Features Building. This assembly consists of a one inch diameter pipe with a ball valve, a pressure indicator and a globe valve. The purpose of this assembly is to measure the hydrostatic pressure and sample the water below the membrane. Essentially, the standpipe and valve assembly replace a small piece of the membrane. The standpipe has been securely grouted and sealed into place to preclude membrane leakage.

Core samples have been removed from the high alumina cement porous concrete layer from under the Engineered Safety Features Building basemat. The coring process has disturbed the membrane at these locations. The waterproofing membrane has been replaced with grout at these locations.

The surface of the containment structure steel liner in contact with concrete is not subject to corrosion because of the alkaline nature of the concrete.

3.8.1.6.5 Steel Liner and Penetrations

Materials

Liner plate up to 1.25 inches inclusive and bridging plate are made from SA 537 Class 2 Quenched and Tempered, nil-ductility transition temperature (NDTT) test not higher than -10°F, with the exception of dome liner plate which is made from SA 537, Class 2 normalized to Class 1 practice, NDTT not higher than -10°F. All liner insert plates and embedment material greater than one inch thick was ultrasonically tested prior to installation for the purpose of detecting possible laminations.

Toughness tests (Charpy V-notch) were performed on all materials which form part of the containment structure boundary. Nil-ductility Transition Temperature Tests were also performed on all ferritic steel that formed part of the pressure boundary but were not required of backing plates, test channels, hatch bolts, and hatch nuts.

Penetration sleeves are made of SA537 Grade B Q&T, SA516 Grade 60 fine grain, normalized and SA333 Grade 6 fine grain normalized, all with a NDTT of -10°F.

Neutron shield tank embedment base and the carbon steel penetration forgings are SA508 Class 1 with a NDTT of +10°F.

Penetration coolers, equipment hatch, personnel airlock, shear lugs, and backing plates are SA516 Grades 60 and 70 fine grain normalized with NDTT of -10°F.

Bridging bars are made of SA350 Grade LF1 and SA516 Grade 70 normalized with NDTT of 0°F. Sump liners and bellows are made of Type 304 stainless steel SA240. The stainless steel penetration forgings are made of types 304 and 316, SA182.

Special Construction Techniques

Erection of the cylindrical portion of the liner plate followed completion of the concrete mat. The liner plates served as the internal form for the concrete containment during construction. All liner seams are double butt welded, except for the lower 31 feet of the cylindrical shell liner, the liner fire damage repair areas, and the mat, where the plates are welded using backing plates. The liner plate is continuously anchored to the concrete shell with steel anchor studs and deformed bars.

The maximum difference in cross-sectional diameters of the liner is in accordance with the rules shown in paragraph NB-4221.1 of Section III, ASME Boiler and Pressure Vessel Code, Nuclear Power Plant Components, 1971 Edition. The maximum misalignment between liner plates is in accordance with paragraph NB-4232 of the ASME Boiler and Pressure Vessel Code, Nuclear Power Plant Components, 1971 Edition. All measurements were taken on parent metal and not at welds. Flat spots or sharp angles were not allowed.

The allowable deviation from true circular form does not affect the elastic stability of the containment liner because of the restraint provided by the anchor studs and deformed bars tying it to the reinforced concrete shell.

3.8.1.6.6 Backfill Around Containment Structure

Concrete was used to backfill around the containment structure. A compressible material was used between the backfill and the containment structure wall to provide a rattle space.

3.8.1.7 Testing and Inservice Surveillance Requirements

3.8.1.7.1 Concrete Containment

A structural acceptance test of the containment was performed after the liner was completed, the last concrete poured, and all penetrations, sleeves, and hatches installed. The test is conducted to confirm that the design and construction of the containment are adequate to withstand the loads caused by the loss-of-coolant accident as described in Section 6.2. The test conforms to the requirements of Regulatory Guide 1.18, Structural Acceptance Test for Concrete Primary Reactor Containments, dated December 28, 1972. The measuring points may be varied or relocated in accordance with paragraph C.3 of the guide.

The structure is surveyed, measured, and inspected for cracks prior to the test. The containment is subject to an internal pressure equal to 115 percent of the design pressure. The pressure test commences at atmospheric pressure and is raised to 115 percent of design pressure in a minimum of four increments. The containment is depressurized in a minimum of four increments. Measurements are made at each pressurization and depressurization level. The pressure is held constant for at least 1 hour at each level before deflections are recorded. Crack patterns are measured and recorded at atmospheric pressure both before and after the test and at the maximum pressure level.

Radial deflection is measured along 6 meridians at 13 feet 6 inches above the top of mat, at mid-height between mat and springline, and at the springline of the dome. The exact locations of the measurements are indicated in the design specification. Vertical deflections are measured at the springline of the dome and the apex. Radial measurements are made using differential transducers supported from the crane wall. Radial and tangential deflections are recorded around the periphery of the equipment hatch. Vertical measurement is made using invar tapes.

Mapping of cracks is performed on exterior surfaces of the containment at locations selected prior to start of the pressure application. Mapping is on one meridian line at three locations, and one location around the equipment hatch.

Testing is not conducted under extreme weather conditions. The environmental conditions are measured and monitored to permit the evaluation of their contribution to the response of the containment. The testing sequence is repeated if the test pressure drops for unexpected reasons to or below the next lower pressure level, or if significant modification or repairs are made to the containment following the test.

The anticipated deflections of the containment and equipment hatch are given on Figure 3.8–58. The anticipated deflections are calculated by taking into account the interaction of the liner, reinforcing and concrete including the effects of concrete cracking.

A limit of 1.3 times the anticipated deflections were used for comparison to the measured deflections. This is based on a comparison of predicted and allowable stresses in the membrane hoop reinforcing at the 52 psi test pressure level. The stress in the hoop reinforcing is predicted to be 25,600 psi at the 52 psi test pressure. When increased by 30 percent, the stress is 33,000 psi or 2/3 of yield, which is the allowable reinforcing stress for this loading condition.

Cracking was expected in the portion of the containment structure shell away from the mat. (Vertical cracks at approximately 18 inches on center, and horizontal cracks at the construction joints 6 foot on center and at 2 foot to 3 foot intervals between the construction joints.) This was based on observations of previous tests.

The final test report, “Report on Structural Acceptance Test of the Concrete Primary Containment Millstone Nuclear Power Station Unit 3,” was prepared covering the test performed on July 10-12, 1985. This report contains the information outlined in Regulatory Guide 1.18, Regulatory Position 13. The report was compiled by Stone and Webster and presented to NUSCO in September 1985.

For liner inservice surveillance requirements and testing, including leak rate testing, see Section 6.2.6.

3.8.2 STEEL CONTAINMENT

This section, as outlined in the NRC Regulatory Guide 1.70, Rev 3, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, regarding steel containment, is not applicable to the Millstone 3 Containment Structure. A steel lined reinforced concrete containment is being used as described in Section 3.8.1. The equipment hatch, personnel hatch, and portions of the penetration sleeves that are not backed by concrete, and normally considered to be steel containment, are also described in Section 3.8.1.

3.8.3 CONCRETE AND STRUCTURAL STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

3.8.3.1 Description of Internal Structures

The containment structure interior arrangement (Figures 3.8–59 and 3.8–60) consists of heavily reinforced concrete walls and slabs which are designed to support the principal nuclear steam supply equipment. The interior concrete also provides interior biological shielding and protection from missiles resulting from postulated component failure (Section 3.5).

The reactor vessel is supported on the reactor vessel support/shield tank located within, and laterally supported by, the concrete primary shield wall.

The refueling cavity, located above the reactor vessel, and the fuel transfer canal are stainless steel lined concrete structures.

The pressurizer and each reactor coolant loop, including their steam generator, reactor coolant pumps, and associated valves and piping are enclosed within separate concrete cubicles.

A 540-ton capacity overhead polar crane is supported by the polar crane wall. The crane is provided with earthquake restraints to preclude its dislodgment from the rails during the safe shutdown earthquake (SSE).

Main steam and steam generator feedwater lines, electrical cable trays and conduits, HVAC ducts, and other miscellaneous pipes are supported by structural steel hangers and supports located on the interior structure walls and slabs.

A minimum clearance, in accordance with the following, is provided at all levels between the interior structural steel work, platforms, pipe supports, conduit supports, and the containment liner and its attachments.

Elevation	Minimum Clearance
Above EL 57 feet 0 inches	1.05 inches.
Between EL 13 feet 0 inches to 57 feet 0 inches (inclusive)	1 inch

Elevation	Minimum Clearance
Below EL 13 feet 0 inches	0.75 inches

No new or unique structural features are used in the design of the internal structure of the containment.

NSSS component supports are described in Section 3.9B.3.

3.8.3.2 Applicable Codes, Standards, and Specifications

Codes, standards, specifications, and NRC regulations and Regulatory Guides used in establishing design methods and material properties for concrete and steel internal structures are given in the following sections:

Codes, Specifications, Design Methods, and Material Properties	Section 3.8.1.2 Section 3.8.1.6
NRC Regulatory Guides 1.10, 1.15, 1.55, 1.60, and 1.61	NRC Regulatory Guide, as qualified in Section 1.8

Section 3.8.3.6 describes materials and quality control procedures used for containment structure interior.

3.8.3.3 Loads and Loading Combinations

Interior concrete structures and structural steel within the containment are designed to withstand the pressure buildup resulting from the loss of coolant accident (LOCA) discussed in Section 6.2.1. The blowdown of a postulated rupture of a main coolant pipe is assumed to be in any one of the steam generator cubicles or adjacent to the reactor vessel. Because the volume of each of these cubicles is less than the entire containment structure, initial differential pressures exist between the interior and exterior of the cubicle until full pressurization of the containment is attained. Structural components, walls, floors, and beams enclosing these cubicles are designed to withstand these differential pressures.

Pipe rupture may also cause blowdown jet and reactive forces. The magnitude of a blowdown jet forcing function resulting from a pipe rupture is dependent upon the geometry and distance of the target from its source. Critical structures and components are protected from, or designed to resist, the effects of these forces. Section 3.6 describes the protection provided against the dynamic effects associated with a postulated rupture of piping.

The interior concrete structures protect the containment shell from internal missiles generated by an accident. Safety features are physically protected from potential sources of missiles either by physical separation, barriers, or by providing restraints on the potential missiles. Section 3.5 describes internal missile generation and the design of barriers to resist these hazards.

Containment internal structures (other than the containment structure mat, shell, and dome) are designed for the applicable loads and loading combinations in Table 3.8–3. Loads are applied to each structure as applicable.

3.8.3.4 Design and Analysis Procedures

The interior structures generally comprise a series of frames, box type structures, and assemblies of slabs. Structural analyses are based on elastic behavior using commonly accepted principles of engineering mechanics appropriate to the geometry of the structure.

Material quality control procedures, as described in Section 3.8.3.6, ensured that material strength requirements were achieved. Over strength of materials is not a factor because the strength method of design, as described in Section 3.8.3.5, is used in proportioning and reinforcing concrete sections.

The amount of reinforcing steel required is determined in accordance with the procedures outlined in ACI 318 and the principal reinforcement patterns are located in the direction of tensile stresses. Bond and anchorage requirements of ACI 318 are complied with, and where biaxial tensile fields exist, the development lengths required by Section 12.5 of ACI 318 are increased by a minimum of 25 percent.

Structural steel is designed in accordance with the procedures outlined in the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, the Structural Welding Code, AWS D1.1, and the loading combinations given in Section 3.8.3.3.

The heat generated in the primary concrete shield wall due to gamma rays and neutrons emanating from the reactor core and due to gamma rays arising from neutron interactions in the primary shield wall is negligible and has no effect on the strength of the structure.

Computer programs that have been used in the design and analysis of structural elements are identified in Appendix 3A.

3.8.3.5 Structural Acceptance Criteria

Design of interior concrete structures follows ACI 318, using the required strength section based on the strength design method. The basic criterion for concrete strength design is expressed as:

Required Strength ≤ 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 which are multiplied by their appropriate load factors. Calculated strength is computed according to the provisions of ACI 318, including the appropriate capacity reduction factors. Capacity reduction factors are the same as those given in Section 9.2 of ACI 318.

Design of interior steel structures is based either on elastic working stress design methods using normal working stress levels given in Part 1 of AISC Specification or on the plastic design methods of Part 2 of AISC.

Section 3.7B.2.8 describes the variations incorporated into the seismic analysis structural model to account for a cracked and an uncracked containment structure shell, for variations and uncertainties in subgrade shear modulus and spring constants, for virtual mass embedment, and for contact pressure distribution. Design of the internal structures is based upon the most conservative values resulting from these variations in assumptions and design parameters.
Section 3.7B.3.6 discusses differential seismic movement relating to interconnected components, systems, and equipment.

Design for horizontal shear forces in the plane of internal structure walls is in accordance with the requirements of Section 11.6 of ACI 318. Section 11.16 incorporates the combined effects of shear and tensile stresses into the nominal permissible shear stress, Vc, allowed to be carried by the concrete.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

Material and quality controls used for the internal structures are as described in Sections 3.8.1.2 and 3.8.1.6. The 60 day compressive strength of concrete is specified as 5,000 psi, except for the concrete ballast slab covering the floor liner, which is 3,000 psi concrete.

Structural steel material, erection, and fabrication tolerances are in accordance with the "AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings" (February 12, 1969), Supplement No. 1 (November 1, 1970), Supplement No. 2 (December 8, 1971), and Supplement No. 3 (June 12, 1974). In general, steel used for structural framing conforms to ASTM A 36, "Specification of Structural Steel." In areas where the design indicates that a higher strength steel is required, the steel conforms to the requirements of ASTM A 440, ASTM A 441, or ASTM A 588 as required.

Certified copies of mill test reports showing actual chemical and physical properties are furnished for each heat of steel used in making Seismic Category I structural steel.

Welding of structural steel is in accordance with AWS D 1.1-72 and AWS D1.1-Rev 1-73 except that undercut on structural welds is limited to 0.01 inch deep when its direction is transverse to the primary tensile stress in the part that is undercut only for the following cases:

1. Where cyclic fatigue is a design parameter and/or

2. Where thin material is used (i.e., 0.5 inch thick and less).

This requirement was implemented through notation of specific inspection criteria on the engineers drawings. In all other situations, undercut is limited to a maximum depth of 1/32 inch.

Connection bolts conform to ASTM A 325 or ASTM A 490. Anchor bolts conform to ASTM A 36 or ASTM A 193.

The material, installation, and inspection of high strength bolts conform to the requirements of the Research Council on Riveted and Bolted Structural Joint Specification using ASTM A 325 or A 490 bolts.

Radiation damage to concrete is insignificant for conditions of neutron fluence at least to 2×10^{19} nvt and temperatures at least to 120 C (Clark 1958). The neutron fluence and the temperature levels in the primary concrete shield wall are both lower than these levels; therefore, no structural damage due to radiation or temperature effects occurs.

3.8.3.7 Testing and Inservice Surveillance Requirements

No testing or inservice surveillance of the reactor containment interior structure is planned.

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES (AND MAJOR NONSAFETY RELATED STRUCTURES)

3.8.4.1 Description of the Structures

The Seismic Category I structures are indicated on the plot plan (Figure 1.2–2). The arrangement drawings for major Seismic Category I structures other than the reactor containment are:

Title	**Figure**	
Containment Enclosure Building	3.8–61	(EA-42)
Auxiliary Building	3.8–62	(EM-6)
Fuel Building	3.8–63	(EM-7)
Control Building	3.8–64	(EE-27E)
Cable Tunnel	3.8–65	(EC-7A)
Emergency Generator Enclosure and Diesel Fuel Oil Tank Vault	3.8–66	(EM-13)
Engineered Safety Features Building	3.8–67	(EM-2)
Main Steam Valve Building	3.8–68	(EM-2)
Circulating and Service Water Pumphouse	3.8–69	(EM-8)
Hydrogen Recombiner Building	3.8–70	(EM-2)
Circulating Water Discharge Tunnel and	3.8–71	(EC-16C)
Discharge Structure		(EC-17A)
Railroad Canopy		(EC-38W,S,Y,Z)
Millstone Stack Elevations and Sections	3.8–83	
Millstone Stack - Mathematical Model	3.8–84	

The refueling water storage tank and the demineralized water storage tank are shown on the plot plan (Figure 1.2–2).

Arrangement drawings for major Non-Category I structures are:

Title	Figure	
Service Building	3.8–72	(EM-5)
Turbine Building	3.8–73	(EM-3)
Waste Disposal Building	3.8–74	(EM-9)
Warehouse 5 and Millstone 2 Condensate Polishing Facility	3.8–75	(EM-50)
Auxiliary Boiler and Condensate Polishing Building	3.8–76	(EM-38)
Miscellaneous Yard Tankage	1.2–2	Plot Plan

Plant grade is approximately elevation 24 feet 0 inch.

Containment Enclosure Building (Seismic Category I)

The containment enclosure building, a cylindrical steel framed structure with uninsulated metal siding and builtup roofing over insulated metal roof deck, envelops the containment structure above grade. It has a diameter of 156 feet, a height above grade of approximately 166 feet, and is supported on the containment structure.

The containment structure below grade is surrounded by ribbed fiberglass sheets, which extend from the top of the containment mat to above grade. This arrangement provides vertical ventilation channels which vent directly into the containment enclosure building or into the abutting buildings. The corrugated fiberglass sheet is covered by a 2 inch thick layer of compressible material, a 40 mil thick waterproofing sheet membrane, and then one more layer of 2 inch thick compressible material. The waterproofing membrane encloses the containment substructure and extends beneath the mat and above plant grade. The enclosure building design incorporates horizontal and vertical sliding joints to ensure that the integrity of the containment enclosure building is maintained during maximum possible pressure transients of the containment structure under DBA conditions. It assures the proper performance of the supplementary leak collection and release system, as described in Section 6.5.1.

To provide the required degree of air tightness and to maintain a partial vacuum within the containment enclosure building, the metal siding, metal deck side joints, and end laps have two continuous lines of caulking at all joints. Neoprene gaskets and sheets are used to provide a flexible seal between the containment enclosure and the other buildings. The base of the containment enclosure building wall is sealed in a similar manner to the subgrade waterproofing membrane.

The structural framing, but not the siding or roofing, is designed to remain intact under tornado loading. In addition, the design of the building is based on the structural loads being transferred through the structural steel bracing and into the containment structure. The building is considered to be round and moderately smooth.

Auxiliary Building (Seismic Category I)

The auxiliary building is located west of the fuel building, east of the service building, and north of the containment structure. The auxiliary building, approximately 102 feet by 177 feet in plan, is supported on a reinforced concrete mat founded on rock. Rock dowels are installed along the exterior walls of the structure to resist uplifting during seismic loading. Section 2.5.4 describes these rock dowels. The basement floor is approximately 20 feet below grade. There is one floor at grade and two above. The concrete roof is located approximately 69 feet above grade. The substructure and superstructure are reinforced concrete construction. Interior shield walls are reinforced concrete.

The motor control center (MCC) and rod control area is located in the southern end of the auxiliary building. This area is open on the side adjacent to the containment structure for direct access to electrical penetrations.

Fuel Building (Partially Seismic Category I)

The fuel building is located east of the auxiliary building, south of the waste disposal building, and north of the containment and engineered safety features structures. The building, approximately 112 feet by 92.5 feet, is supported on compacted fill and/or rock.

The Seismic Category I portion of the building is as follows:

1. Spent Fuel Pool
2. Spent Fuel Shipping Cask Area
3. Auxiliary-Emergency Safety Features Building Pipe Tunnel
4. Canopy over Fuel Cask Shipping Area
5. Demineralizer Area
6. Spent Fuel Shipping Cask Washdown Area
7. Spent Fuel Pool Cooler Cubicle

The remainder of the building is non-seismic and includes:

1. New Fuel Handling Area

2. New Fuel Storage Vault Area

3. Equipment Decontamination Area

4. Equipment Storage Area

5. Heat Tracing Room

These non-seismic areas are designed to withstand seismic loading so as to prevent their collapse onto adjacent Category I areas. The fuel building has a ground floor at grade elevation and a basement 13 feet below grade. The spent fuel pool portion of the building is reinforced concrete construction from approximately 24 feet below to 28 feet above grade. The spent fuel areas are protected from tornado missiles by a reinforced concrete superstructure. The new fuel handling and equipment decontamination areas of the building have reinforced concrete walls with a steel framed roof and metal deck covered with 4-ply asphalt and gravel roofing.

The spent fuel pool is "L" shaped, with the bottom of the pool approximately 13 feet below grade. The floor is 8 foot thick reinforced concrete. The spent fuel shipping cask storage area is 14 feet wide by 30 feet long. The bottom of this area is stepped from approximately 20 feet below grade to 1 foot above grade to 28 feet above grade to limit the lifting height of the spent fuel shipping cask. The walls of the spent fuel pool and the spent fuel shipping cask storage area are 6 foot thick reinforced concrete with the exception of the east wall of the spent fuel shipping cask storage area which is 5 feet thick and extends to approximately 28 feet above grade. Concrete dividing walls permit dewatering the spent fuel shipping cask storage area and the fuel transfer canal without dewatering the entire pool. The interior walls and floor of both the new fuel storage vault and spent fuel pool, the spent fuel shipping cask storage area, and the fuel transfer canal are lined with 0.25 inch thick stainless steel plate.

The new fuel storage vault is 24 feet 0 inch long by 15 feet 9 inches wide by 18 feet 4 inches deep with the bottom approximately 9 feet 6 inches above grade.

<u>Control Building</u> (Seismic Category I)

The control building, approximately 101 feet by 116 feet, is located north of the turbine building, south of the emergency diesel generator building, east of the technical support center and office building, and west of the service building; it is supported by a reinforced concrete mat foundation. Except for the east portion of the top level described below, all exterior walls, floor slabs, and roof slabs are reinforced concrete with interior framing and columns of structural steel.

The basement level approximately 20 feet below grade houses emergency (essential) switchgear, battery chargers, battery rooms, inverters, and the emergency shutdown panel. The second floor is the cable spreading room. The third floor is the control room level which contains the control room, instrument rack room, computer room, and facilities for personnel. The fourth floor contains the ventilation and air-conditioning equipment rooms. The roof is approximately 60 feet above grade.

The east portion of the top level is nonsafety related; however, the steel superstructure is designed to withstand tornado and seismic loads.

Cable Tunnel (Auxiliary Building to Control Building) (Seismic Category I)

The electrical cable tunnel measures approximately 24 feet by 29 feet in section and runs through the basement level of the service building for its full width. The tunnel is contiguous with and provides support for portions of the service building. The tunnel is reinforced concrete construction (Figure 3.8–65).

Emergency Generator Enclosure and Fuel Oil Tank Vault (Seismic Category I)

The emergency generator enclosure, located north of the control building, is a reinforced concrete enclosure approximately 65 feet by 72 feet. The roof is located approximately 41 ft above plant grade. Its foundation consists of reinforced concrete footings placed upon glacial till. The structure is founded approximately 15 feet below grade. Each combination of emergency generator and diesel engine has its own foundation founded on compacted fill. The roof and walls are reinforced concrete, a minimum of 2 feet thick. A reinforced concrete wall separates the two emergency generator units. Removable sections of the east and west walls in front of the diesel units provide for replacement of equipment.

The fuel oil tank vault, approximately 32 feet by 65 feet by 23 feet high, is located east of the main structure. The vault is below grade to provide protection from tornado missiles. A 1 foot-6 inch thick fire wall separates the two tanks.

Engineered Safety Features Building (Seismic Category I)

The engineered safety features building wraps around the east side of the containment structure. The structure is founded on rock at elevation 0 feet 6 inch and is approximately 140 feet long by 40 feet wide. Four pump shafts extend down to the containment mat for the containment recirculation pumps. The structure has three floor levels and extends 32 feet above grade. The entire building, including the pump shafts, is reinforced concrete construction.

Main Steam Valve Building (Seismic Category I)

The main steam valve building, located west of and directly adjacent to the containment structure protects the main steam valves and piping from tornado missiles. The building consists of a reinforced concrete structure with a 2 foot thick wall and roof supported on the rock. The structure extends from approximately 16 feet below grade to approximately 63 feet above grade.

Circulating and Service Water Pumphouse (Service Water Portion, Seismic Category I)

The circulating and service water pumphouse is located on the shoreline of Niantic Bay west of Millstone 3. Approximately 128 feet by 86 feet, it houses the circulating water pumps and the service water pumps. It is constructed of reinforced concrete founded on bedrock approximately 30 feet below mean low water. The service water pump room and its supporting elements are

Seismic Category I, i.e., designed to withstand tornado and earthquake loads, and are protected against flooding to 25.5 feet above mean low water.

A retaining wall is located on the west side of the circulating and service water pumphouse and is a Category I counterfort type reinforced, concrete retaining wall. The wall is founded on bedrock and is an extension of the west wall of the circulating and service water pumphouse. The function of the west retaining wall is to protect the Category I service water and electrical lines located behind the wall and to be part of the shoreline protection. For a discussion of the shoreline protection, refer to FSAR Section 2.5.5. The seawall located to the east of the circulating and service water pumphouse is not a Category I structure and therefore, is not discussed in this section.

Hydrogen Recombiner Building (Seismic Category I)

The hydrogen recombiner building is located adjacent to the containment structure, on the southeast side, directly below the equipment hatch. The building, approximately 56 feet by 50 feet by 27 feet high, houses the hydrogen recombiner equipment and provides access to the equipment hatch and support for the removable hatch missile shield. It is constructed of reinforced concrete, for protection of safety related equipment, and founded on fill concrete.

Circulating Water Discharge Tunnel (Seismic Category I)

The circulating water discharge tunnel is a reinforced concrete structure founded entirely below grade on rock, concrete fill, or till. The portion of the circulating water discharge tunnel downstream of the service water discharge point is designed as a Seismic Category I structure.

Railroad Canopy (Seismic Category I)

The railroad canopy is located to the east of the fuel building.

The canopy structure is approximately 75 feet long by 26 feet wide with buttresses extending out an additional 20 feet. It has a mat which is about 77 feet long by 54 feet wide and 8 feet thick and is founded on concrete fill. The 50.6 and 51.2 line walls and roof are reinforced concrete 2 feet thick. The buttresses, L-line wall and west wall are all reinforced concrete, 3 feet thick.

The top of the mat is at grade. A railroad spur enters the east side of the building at ground level.

The entire canopy structure is designed for seismic and tornado loads.

The building protects the spent fuel pool from tornado missiles.

Millstone Stack

Description

The unlined, free standing, tapered, reinforced concrete stack has the following dimensions:

Overall height above foundation	385’6”
Height above adjacent grade	375’
Inside diameter at top	7’
Outside diameter at base	27’6”
Thickness at top	7”

The stack configuration is shown on Figure 3.1-7. The Millstone stack was originally designed as seismic Class I.

Maximum stack stresses occur in the first 208 feet above grade, due to load combinations which include maximum wind.

The stack is designed for a maximum exhaust air temperature of 150°F and a minimum ambient temperature of 0°F. Design and construction of the stack is in accordance with applicable requirements of ACI “Standard Specifications for the Design and Construction of Reinforced Concrete Chimneys” (ACI-505) as follows:

Wind Pressures

For normal design conditions, it was designed at specific ACI allowable unit stresses. Wind pressures on projected areas for various height zones above ground conformed to the following requirements for a basic wind pressure area of 30 psf.

Height Pressure	**Design Wind Pressure**
0-49 ft	32 psf
50-99 ft	39 psf
100-199 ft	45 psf
200-299 ft	48 psf
300-375 ft	51 psf

Tabulated wind pressures include the reduction for the circular shape of the stack. The tabular value for pressure of ACI-505 was increased for a gust velocity of 140 mph in accordance with Paragraph 400 of ACI-505.

The stack has a 2.94 factor of safety with regard to overturning or rocking. The maximum overturning moment due to wind on the stack is 52,500 kip feet and the stability moment for the stack is 154,000 kip feet.

Earthquake Loading

In developing a mathematical model, for dynamic analysis, the stack was treated as a flexible cantilever system with the base fixed at the top of the foundation. (The stack base is neither rigid nor can it rotate, but is considered rigid as this assumption results in a shorter period for the stack. This analytical representation gives higher, more conservative response acceleration.) Forty mass points were considered to be supported my weightless elastic columns. The model is depicted by Figure 3.1-8. Subsequent to the formation of mass and stiffness matrices for the cantilever system, the periods and mode shapes were calculated, displacement and inertia force time histories were established and a time history of shears, moments, displacements and accelerations determined.

The top of the stack is at elevation 389 feet or 238 feet above the top of the Reactor Building. The stack is located 416 feet east of the Reactor Building east wall. Thus, the top of the stack could not strike the operating floor of the Reactor Building, even if toppled intact about its base, because the stack height is 385 feet from the base. The ventilation exhaust is brought through breeching which gathers the various exhaust ducts together. The stack is provided with a one foot by four foot access opening at the base, three galvanized steel balconies, an outside ladder for the full stack height, aviation obstruction lighting, lightning protection and handling of the isokinetic sampler.

Service Building (Partially Seismic Category I)

The service building is located between the control and the auxiliary buildings. Approximately 100 feet by 80 feet, it is founded on bedrock and is of steel frame construction with metal siding and builtup roofing on insulated metal deck. This building consists of one level below grade, one level at grade, and two levels above grade. The roof is located approximately 43 feet above plant grade. The below-grade level houses nonsafety related switchgear and the Seismic Category I electrical tunnel. The grade level houses offices, change rooms, radiation protection facilities, first-aid room, and other service facilities. The first floor above grade houses the lunch room, instrument repair room, and locker area, while the second above-grade level houses mechanical equipment.

The structural framing, but not the siding and roofing, is designed to withstand tornado winds and seismic forces.

Turbine Building (Nonsafety Related)

The turbine building, approximately 325 feet by 115 feet, is located west of the containment structure and is supported on spread footings on basal till and compacted select granular fill. The turbine building has a basement level 10 feet below grade and a roof 107 feet above grade. The foundation walls are reinforced concrete to grade. The superstructure is steel framed with metal siding and builtup roofing on an insulated metal deck. There is an auxiliary bay of the same construction, approximately 50 feet by 300 feet by 75 feet high above grade, on the east side of the turbine building.

The structural framing, but not the siding and roofing, is designed to withstand tornado winds and seismic forces.

Waste Disposal Building (Nonsafety Related)

The liquid waste disposal building is located directly north of the fuel building and east of the auxiliary building. It is a reinforced concrete structure approximately 48 feet by 114 feet with a steel framed HVAC penthouse. The building is founded on bedrock and basil till with a basement level 20 feet below grade. The penthouse roof is approximately 73 feet above grade.

The solid waste disposal building is located directly north of the liquid waste building. The building is approximately 38 feet by 114 feet and is founded at grade on soil backfill. The superstructure consists of a 24 foot high reinforced concrete shell and a steel framed enclosure with builtup roofing on insulated metal deck approximately 42 feet above grade.

The design equals or exceeds the requirements of Regulatory Guide 1.143.

Warehouse Number 5 and Millstone 2 Condensate Polishing Facility (Nonsafety Related)

The warehouse structure, approximately 98 feet by 211 feet, is located north of the Millstone 2 turbine building and south of the Millstone 3 condensate polishing facility and auxiliary boiler building. The structure consists of three main levels and a penthouse. The condensate polishing facility is located approximately 20 feet below grade with portions of the waste handling equipment extending to 4 feet and 26 feet above grade. Warehouse storage and Millstone 3 ultrafiltration equipment are located 4 feet above grade. Records storage is located 26 feet above grade. The penthouse, 50 feet by 46 feet, is located 40 feet above grade in the middle of the structure near the west wall. This enclosure houses the elevator machine room and building ventilation equipment.

The superstructure is steel framed with the main roof approximately 40 feet above grade and the penthouse roof approximately 58 feet above grade.

The warehouse structural framing, but not the siding and roofing, is designed to withstand tornado winds and seismic loadings.

Auxiliary Boiler and Condensate Polishing Facility (Nonsafety Related)

The auxiliary boiler room is located south of the turbine building, east of the condensate polishing facility, and north of the warehouse. The area is approximately 66 feet by 58 feet in plan and is supported on reinforced concrete footings with a floor 0 feet 6 inches above grade. A roof with insulation and 4-ply asphalt and gravel is provided at approximately 36 feet above grade.

The condensate polishing facility is located north of Warehouse Number 5 and west of the auxiliary boiler room. The condensate polishing enclosure is a reinforced concrete two story structure, approximately 58 feet by 64 feet, designed for radiation protection. The structure is supported by spread footings. The basement floor is approximately 10 feet below grade with the second level approximately 14 feet above grade.

1. Yard Structures

<u>Vacuum Priming Pumphouse</u> (Nonsafety Related)

The vacuum priming pumphouse is a reinforced concrete structure located on top of the outfall structure. The area is approximately 40 feet by 35 feet with a floor 0 feet 6 inches above grade. A roof with insulation and 4-ply asphalt and gravel is provided at approximately 17 feet above grade.

2. Miscellaneous Yard Tankage

Boron recovery tanks, primary grade water tanks, demineralized water storage tank, refueling water storage tank, boron and waste test tanks, condensate storage tank, condensate surge tank and water treatment storage tanks are located on concrete pads with oil sand cushion 0 feet 6 inches above grade. The demineralized water storage tank is protected by 2 feet 0 inch thick reinforced concrete walls and roof. The boron recovery tanks are enclosed in a concrete and steel structure.

3. Electrical/Conduit Manholes

Electrical manholes are reinforced concrete structures constructed below grade with access through manhole covers at grade.

All other nonsafety related structures are located such that their failure does not damage safety related systems, structures, or components.

3.8.4.2 Applicable Codes, Standards, and Specifications

Codes, standards, specifications, and NRC regulatory guides used in establishing design methods and material properties for Seismic Category I concrete and steel structures other than the containment are given in Section 3.8.1.2.

3.8.4.3 Loads and Loading Combinations

All Seismic Category I structures other than the containment structure mat, shell, and dome are designed for the loads and load combinations in Table 3.8–3. Section 3.8.4.5 describes allowable stress levels.

For the spent fuel pool, the effects of loads imparted to the structure by the spent fuel racks as well as the effects of hydrostatic and seismically induced hydrodynamic loads are considered in the design. The historical design of the spent fuel pool walls and mat considered the thermal effects based on the temperatures indicated in Figures 3.8–79 and 3.8–80. The analysis for classifying a full core off load as a normal evolution evaluated the thermal effects based on temperatures indicated in Figure 3.8–82. The spent fuel pool walls and mat were also investigated for the revised thermal transient effects due to the storage of higher enrichment fuels as shown in Figure 3.8–81. Utilizing the loads and load combinations in Table 3.8–3, the allowable stress levels described in Section 3.8.4.5 were satisfied.

3.8.4.4 Design and Analysis Procedures

In general, design and analysis procedures conform to the requirements of the ACI 318-71 Code and the AISC specification, 7th Edition, for the Design, Fabrication and Erection of Structural Steel for Buildings except as noted in Section 3.8.4.3. Structural analyses are based on elastic behavior using commonly accepted principles of engineering mechanics appropriate to the geometry of the structure.

The boundary conditions assumed for structural elements under design are based on the stiffness of the elements into which these elements are framed.

Material quality control procedures, as referred to in Section 3.8.4.6, ensure that minimum strength material requirements are achieved.

Seismic accelerations on the structures are determined by dynamic analysis as described in Section 3.7B.3.4. Forces are determined and then applied statically in the design of the structures. The analytical techniques used to determine the forces are given in Section 3.7B.2.

A shake space, consistent with building displacements, is provided, above grade, between all independent Seismic Category I and nonseismic structures so as to prevent their interaction during a seismic event.

Tornado loads (described in Section 3.3.2) include wind force loads and the loads from tornado generated missiles. Section 3.5.3 gives tornado missile impact effects.

Computer programs which have been used in the design and analysis of structural elements are identified in Appendix 3A.

3.8.4.5 Structural Acceptance Criteria

Seismic Category I structures, as identified in Table 3.2–1, are designed to withstand the loading combinations given in Table 3.8–3.

Concrete structures are designed by the strength design method of ACI 318-71.

The basic criterion for strength design is expressed as:

Required Strength ≤ Calculated Strength

All concrete members and all sections of concrete 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 have been multiplied by their appropriate load factors. Calculated strength is that computed by the provisions of ACI 318, including the appropriate capacity reduction factors. Capacity reduction factors are taken as given in Section 9.2 of ACI 318. Allowable stresses and strains for concrete structures are within the limits specified in Section 9.2 of ACI 318.

Steel structures, except as noted in this section, are designed in the elastic range to maintain actual stresses less than allowable stress given in Part 1 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings" and also the Structural Welding Code, AWS D1.1. For members requiring tornado or seismic (SSE) design, allowable stresses are as follows:

Tension

$F_t = 0.90F_y$

Shear

$F_v = 0.60F_y$

Compression Members

Allowable stresses are 1.6 times the values given by Section 1.5.1.3 (p 5-64) of AISC Specification, 7th Edition, for the Design, Fabrication, and Erection of Structural Steel for Buildings.

Bending

Tension and compression for compact, adequately braced members symmetrical about, and loaded in, the plane of their minor axis.

$F_b = 0.90F_y$

Allowable stresses for other members are 1.6 times those values given in AISC Section 1.5.1.4, but in no case greater than 0.90 F_y.

Connections

Welds - 1.6 times values given in AISC Section 1.5.3, Table 1.5.3

High Strength Bolts - 1.6 times values given in AISC Section 1.5.2, Table 1.5.2.1

The limits of deflections are within those specified in Section 9.5 of ACI 318 for reinforced concrete construction. Stress levels, rather than deflections, are normally the criteria for structural steel design.

The majority of the Millstone 3 structures have a relatively small height-to-width ratio; therefore, column drift due to wind is not a problem. The only building with significant height, the turbine building, has been checked for wind deflections. These deflections are limited such that overall frame stability is maintained and deflections under service loads are within common practice for the industry.

3.8.4.6 Materials, Operating Control, and Special Construction Techniques

Sections 3.8.1.2 and 3.8.1.6 describe material and quality control. There are no special techniques used in constructing the structures (Section 3.8.4.1).

The 60 day compressive strength of concrete for the spent fuel pool and the fuel building is specified as 5,000 psi.

3.8.4.7 Testing and Inservice Surveillance Requirements

There are no special testing or inservice surveillance requirements for Category I structures outside the containment.

3.8.4.8 Masonry Walls

Masonry walls in safety related areas in the plant comply with the requirements in Appendix A to SRP Section 3.8.4. Locations of walls are given in Figure 3.8–64, Sheet 1 and 3.

3.8.5 FOUNDATIONS

3.8.5.1 Description of the Foundations

Foundations for all of the major structures consist of soil or rock supported reinforced concrete mats or spread footings as described in Table 2.5.4–14. Figures 2.5.4–1 through 2.5.4–17 show plan and section views of the major foundations.

To provide for independent movement of structures during a seismic event, a minimum of 1 inch thick compressible material is provided below grade between all structures. The containment is separated, below grade between all structures, by a minimum 4 inch shake space filled with a compressible material.

Horizontal shear keys are provided for the control, fuel, and auxiliary buildings and for the circulating and service water pumphouse foundations. Figure 3.8–77 shows the arrangement of these horizontal shear keys and Figure 3.8–78 shows a typical detail. Rock dowels are used in the auxiliary building foundation (Section 3.8.5.5).

Horizontal shear resulting from seismic acceleration of the containment structure is transferred through interface friction to the surrounding rock.

Rock dowels are used in the exterior walls of the auxiliary building to resist uplift during seismic loading. Section 2.5.4 describes the rock dowels and the installation program.

The containment enclosure building is supported entirely on the containment structure and has no foundations. Figure 3.8–61 shows typical details of the interface between the enclosure and ground/adjacent structures.

Significant amounts of groundwater are not expected. Figure 2.5.4–37 shows the design levels for groundwater. No Seismic Category I dewatering system is required. However, the following features have been incorporated to prevent seepage of groundwater into portions of structures below the piezometric surface:

1. All structures, except the containment, have waterstops installed at construction joints below grade.

2. The containment substructure is encased with a waterproof membrane to elevation 25 feet 0 inches or to the bottom or approximate midpoint at slabs abutting the containment structure below elevation 25 feet 0 inches. Such slabs are provided with waterstops or the membrane is continued as an encasement for the abutting structure to preclude seepage at the interface of the slab and the containment wall. A drainage system is provided on the containment/Engineered Safety Features Building side of the membrane and connects to a sump located in the Engineered Safety Features Building to remove groundwater which leaks through the membrane. A non safety-related pump is used as necessary to remove the water during normal plant operation, post-LOCA and LNP conditions (see Section 9.3.3.2.4.1 for additional information).

3. The service, control, auxiliary, and Engineered Safety Features Building substructures are encased with a waterproof membrane to elevation 23 feet 6 inches and have drainage systems located under the mat of each building. These run into sumps for collection and then discharge. The coefficient of friction between the membrane and the concrete is equal to or greater than that between the concrete below the membrane and the soil or rock. Sliding stability is therefore not affected by the presence of the membrane.

4. The Technical Support Center, fuel, and waste disposal buildings are provided with perimeter and substructure drains.

3.8.5.2 Applicable Codes, Standards, and Specifications

Section 3.8.3.2 contains the codes, standards, specifications, and NRC regulatory guides used in establishing design methods and material properties for foundations and concrete supports.

3.8.5.3 Loads and Loading Combinations

Foundation design is based upon appropriate loading combinations. The loads and loading combinations given in Section 3.8.1.3 are used for the containment foundation design. The loads and loading combinations given in Table 3.8–3 are used for the design of all other Seismic Category I foundations.

In addition to the above loads and load combinations, the following were used to check against sliding and overturning due to earthquakes, winds, tornadoes, and the design basis flood:

1. D + H + OBE
2. D + H + W
3. D + H + SSE
4. D + H + W_t
5. D + F

where:

D, OBE, W, SSE, W_t are as defined in Table 3.8–3

H = The lateral earth pressure

F = The buoyant and lateral force effects of the design basis flood

Section 3.8.5.5 gives stability factors for these conditions.

Where Seismic Category I structures extend below the surface of the finished ground grade, their external walls are designed for seismic lateral earth pressure and groundwater effects in addition to static lateral earth pressures due to soil loads, surcharge loads applied at ground surface, and lateral and buoyant force effects from groundwater or flood and loads of adjacent footings or mats.

3.8.5.4 Design and Analysis Procedures

Section 3.8.1.4 describes the design and analysis of the reactor containment foundation mat.

Section 3.8.4.4 describes design and analysis of Seismic Category I foundations other than the reactor containment.

Section 3.7B.2.14 describes determination of seismic overturning moments.

3.8.5.5 Structural Acceptance Criteria

Structural design of the reactor containment foundation is in accordance with ACI 318, using the load criteria given in Section 3.8.1.3. Structural design of all Seismic Category I foundations other than the reactor containment foundation is in accordance with ACI 318, using strength design and the load criteria given in Section 3.8.4.3.

The basic criterion for strength design is expressed as:

Required Strength ≤ 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 factors. The calculated strength for all Seismic Category I foundations is that computed by the provisions of ACI 318, including the appropriate capacity reduction factors. Capacity reduction factors are as given in Section 9.2 of ACI 318.

Sliding and overturning factors of stability are:

LOADING CONDITION (Section 3.8.5.3)	MINIMUM FACTORS OF SAFETY		
	OVERTURNING	SLIDING	FLOTATION
1. Operating Basis Earthquake	1.5	1.5	-
2. Normal Wind	1.5	1.5	-
3. Safe Shutdown Earthquake	1.1	1.1	-
4. Tornado	1.1	1.1	-
5. Design Basis Flood	-	-	1.1

No differential settlement of the reactor containment is anticipated.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

Porous concrete is used to provide subsurface drainage under and around the containment structure. This type of concrete is formed by the omission of the fine aggregate from a standard concrete mix. The mix is designed to have a minimum 28 day compressive strength of 1,000 psi.

Porosity tests of porous concrete performed at Northeastern University in September 1962, used 6 inch diameter by 12 inch long cylinders. These cylinders were prepared in the laboratory by compacting the material in three layers with a standard tamping rod. Results indicated water porosities of from 28 to 47 gpm/sq ft, depending upon the amount of compaction and resulting density of the cylinders.

The waterproofing membrane is a butyl rubber membrane. Adhesives and tapes used for joints and seals in the membrane are the membrane manufacturer's recommended material for the applicable conditions.

Section 3.8.4.6 describes the materials and quality control used for Seismic Category I foundations and supports. No special construction techniques are used for Seismic Category I foundations.

3.8.5.7 Testing and Inservice Surveillance Requirements

The entire reactor containment undergoes structural acceptance testing, as described in Section 3.8.1.7. Except for this test, no other testing or inservice surveillance of foundation systems is planned.

There are no special testing or inservice surveillance requirements for Seismic Category I structures outside the containment.

3.8.6 REFERENCES FOR SECTION 3.8

3.8-1 Clark, R.G. 1958. Radiation Damage to Concrete, US AEC Report H.W.-56195, Hanford Atomic Products Operation, March 31, 1945.

3.8-2 Holley, M.J., Jr. 1969. Provision of Required Seismic Resistance. MIT, Cambridge, Mass.

3.8-3 TRW, Inc. 1975. Embedment Properties Headed Studs. TRW Nelson Division, Lorain, Ohio.

3.8-4 Zhemochkin, B.N. and Sinitzin, A.P. 1962. Practical Methods for Analysis of Beams and Plates on Elastic Foundations. In Russian, Gosstroiizdar, Moscow.

TABLE 3.8–1 LOADING CONDITIONS - LINER PLATE AND ACCESS OPENINGS

Category	Load Conditions	Design Allowables (per ASME III Nomenclature)	
Emergency	$D+P_D+T_D+SSE$	$P_m+P_b+Q < 3S_m$	
Test	$D+1.15P$	$P_m < 0.9S_y$	
		$P_m+P_b < 1.35S_y$	
		+ “CAT” curve considerations	
Normal	100 cycles of Δ P	NB-3222.4 (d) or (e)	
	400 cycles of Δ T		
	100 cycles of 1/2-SSE		
Severe Operational	$D+P_{min}+T_{min}$ + 1/2-SSE	$P_m < S_m$	Without temperature
		$P_m+P_b < 1.5S_m$	
		$P_m+P_b+Q < 3S_m$	
Emergency	$D+P_{min}+T_D$ + SSE	$\varepsilon_{SC} = 0.014$	
	ANCHORS		
Emergency	$D+P_D+T_D+SSE$	Maximum shear < 0.425 S_u	
Severe Operational	$D+P_{min}+T_{min}$+ 1/2-SSE	tensile < 0.45 S_u	
Emergency	$D+P_{min}+T_D+SSE$	Shear $\delta_a = 0.5\ \delta_u$	
		Tension $\delta_a = 0.5\ \delta_u$	

NOTE:

The normal and test load combinations are producing negligible effects.

Where:

D = Dead load effect of reinforced concrete structure acting on the liner plus dead load of the liner

P_D = Design pressure (pressure resulting from design basis accident and safety margin)

Where:

T_D	=	Load due to thermal expansion, resulting when the liner is exposed to the design temperature
SSE	=	Stresses in the liner derived from applying the effect of the safe shutdown earthquake
ΔP	=	Differential pressure between operating pressure and atmospheric pressure (100 cycles are assumed on the basis of 2.5 hour refueling cycles per year on a 40 year span)
ΔT	=	Load due to thermal expansion, resulting when the liner is exposed to the differential temperature between operating and seasonal refueling temperatures (400 cycles are assumed on the basis of 10 such variations per year, on a 40 year span (100 cycles of 1/2-SSE is an assumed number of cycles for this type of earthquake)
P_{min}	=	Minimum pressure resulting during operation of the containment
T_{min}	=	Load due to thermal expansion resulting when the liner is exposed to the minimum pressure
S_y	=	Yield strength of the material
S_m	=	The smaller of 1/3-ultimate strength or 2/3-yield strength
S_u	=	Ultimate strength of the stud material
δ_a	=	Allowable displacement for liner anchors, inches
δ_u	=	Ultimate displacement capacity for liner anchors, inches
ε_{SC}	=	Allowable liner plate compressive strain, inches/inches

TABLE 3.8–2 LOADING CONDITIONS, PENETRATIONS

Areas of Analysis (See Figure 3.8–15)	Category	Load Combinations	Stress Allowables per ASME III Nomenclature
1	Design	M_p or T_p or J_{ax} or J_{sh}	$P_m < 0.9\ S_y$
			$P_m + P_b < 0.9\ S_y$
			P concrete bearing < 2,400 psi
	Emergency	$P_d + T_d + R_o$	$P_m + P_b + Q < 3\ S_m$
2	Design(1)	M_p or T_p or J_{ax} or J_{sh}	$P_m < 0.9\ S_y$ *
			Pm + $P_b < 0.9\ S_y$ *
	Design (2)	$P_g + T_g$ + Design (1)	$(P_1 + P_b) < 1.5\ S_m$ $(P_m + P_b + Q) < 3\ S_m$
	Normal	$P_g + T_g + R_e$	ASME III Table NC-3611.1(b) (3)-1 or Para. NB-3222.4(d) or (e)

* For the pipe portion, refer to Section 3.7B.3.1

Where:

M_p = Yielding moment = Required bending moment to produce stresses equal to the yield strength of the pipe material

T_p = Yielding torque = Required torsional moment to produce stresses equal to the yield strength of the material

J_{ax} = Axial jet force = Load equal to the piping design pressure times the inside area of the pipe, acting in the axial direction of the piping

J_{sh} = Shear jet force = Load equal to the piping design pressure times the inside area of the pipe acting transversely to the pipe

P_d = Containment design pressure

T_d = Containment design temperature

P_g = Piping design pressure

T_g = Piping design temperature

R_o = Piping reactions due to normal operation (including SSE effects)

R_e = Piping reactions due to normal operation (including 1/2 - SSE)

Design (1) - Applies to the sizing of the sleeve and attachment plate

Design (2) - Applies to the evaluation of stresses in the area of Analysis 2 due to the given load combinations

TABLE 3.8–3 LOADS AND LOADING COMBINATIONS

1. Concrete Structures (Containment Internal Structures and Category I Structures, other than the containment mat, shell, and dome).

Loads and loading combinations are based on ACI 318, and AEC Enclosure 3 - Structural Design Criteria for Evaluating Effects of High Energy Pipe Breaks on Category I Structures Outside the Containment, Structural Engineering Branch, Directorate of Licensing.

1. $U = 1.4D + 1.7L$
2. $U = 1.4D + 1.7L + 1.7H$

2a. $U = 0.9D + 1.7H$

3. $U = 1.4D + 1.7L + 1.4F$

3a. $U = 0.9D + 1.4F$

4. $U = 0.75 (1.4D + 1.7L + 1.7W)$

4a. $U = 0.90D + 1.3W$

5. $U = 0.75 (1.4D + 1.7L + 1.7 x 1.1 (1/2 SSE)$

5a. $U = 0.90D + 1.3 x 1.1 (1/2 SSE)$

6. $U = 1.1 D + 1.1 L + 1.1 SSE$

6a. $U = 0.9D + 1.1 SSE$

7. $U = 1.1 D + 1.1 L + 1.0 Wt$

7a. $U = 0.9D + 1.0 Wt$

8. $U = D + L + T_a + R_a + 1.5P_a$
9. $U = D + L + T_a + R_a + 1.25P_a + 1.25 OBE + 1.0 (Y_r + Y_j + Y_m)$
10. $U = D + L + T_a + R_a + P_a + SSE + 1.0 (Y_r + Y_j + Y_m)$

Notes - Concrete Structures

(1) U is the required section strength based on strength design methods described in ACI 318-71.

(2) In combinations 8, 9, and 10, the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, shall be used unless a time-history analysis is performed to justify otherwise.

(3) For load combinations 9 and 10, local section strengths and stresses may be exceeded under the concentrated loads Y_r, Y_j, and Y_m, provided there will be no loss of function of any safety related system.

TABLE 3.8–3 LOADS AND LOADING COMBINATIONS (CONTINUED)

(4) For load combinations 7 and 7a, local section strengths and stresses may be exceeded under the tornado missile load provided there will be no loss of function of any safety related system.

2. Steel Structures

A. Elastic Working Stress Design Service Load Conditions

1. $1.0\ S = D + L$

1a. $1.5\ S = D + L + T_o + R_o$

(If T_o and R_o = 0 use Equation 1)

2. $1.0\ S = D + L + OBE$

2a. $1.5\ S = D + L + T_o + R_o + OBE$

(If T_o and R_o = 0 use Equation 2)

3. $1.33\ S = D + L + W$

3a. $1.5\ S = D + L + W + T_o + R_o$

(If To and Ro = 0 use Equation 3)

Factored Load Conditions

4. $1.6\ S = D + L + T_o + R_o + SSE$

5. $1.6\ S = D + L + T_o + R_o + Wt$

6. $1.6\ S = D + L + T_a + R_{\mathbf{a}} + P_a$

7. $1.6\ S = D + L + T_a + R_a + P_a + 1.0$
$(Y_r+Y_j+Y_m) + OBE$

8. $1.7\ S = D + L + T_a + R_a + P_a + 1.0$
$(Y_r+Y_j+Y_m) + SSE$

B. Plastic Design

Factored Load Conditions

9. $0.9Y = D + L + T_a + R_a + 1.5\ P_a$

9a. $1.0Y = D + L + T_a + R_a + 1.5\ P_a$

10. $0.9Y = D + L + T_a + R_a + 1.25\ P_a + 1.25\ OBE + 1.0\ (Y_r+Y_j+Y_m)$

10a. $1.0Y = D + L + T_a + R_a + 1.25\ P_a + 1.25\ OBE + 1.0\ (Y_r+Y_j+Y_m)$

TABLE 3.8–3 LOADS AND LOADING COMBINATIONS (CONTINUED)

11. $0.9Y = D + L + T_a + R_a + P_a + SSE + 1.0\ (Y_r+Y_j+Y_m)$

11a. $1.0Y = D + L + T_a + R_a + P_a + SSE + 1.0\ (Y_r+Y_j+Y_m)$

Notes - Steel Structures

(1) S is the required section strength based on the elastic design methods and allowable stresses defined in Part 1 of the AISC, Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings.

(2) Y is the section strength required to resist design loads based on plastic design methods described in Part 2 of the AISC, Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings.

(3) Both cases of L having its full value or being completely absent are checked for load combinations 1, 1a, 2, 2a, 3, 3a, 4, and 5.

(4) In combinations 4 to 8 and 9 to 11, thermal loads are neglected when it can be shown that they are secondary and self-limiting in nature, or where the material is ductile.

(5) In combinations 6, 7, 8, 9, 10, and 11, the maximum values of P_a, T_a, R_a, Y_r, Y_j, and Ym, including an appropriate dynamic factor, are used unless a time-history analysis is performed to justify otherwise.

(6) Combination 5 shall be satisfied without the tornado missile load. Combinations 7, 8, 10 and 11 shall be first satisfied without Y_r, Y_j, and Y_m. When considering these loads, however, local section strengths may be exceeded under the effect of these concentrated loads, provided there will be no loss of function of any safety related system. Furthermore, in computing the required section strength, S, the plastic section modulus of steel shapes may be used for combinations 7 and 8.

(7) Combinations 1a, 2a, 3a, 9a, 10a, and 11a were added for use in design evaluations since September 1985. These combinations are consistent with NUREG-0800, Section 3.8.4, (Revision 1, July 1981).

Loads, Definition of Terms, and Nomenclature

1. Normal Loads - Those loads encountered during normal plant operations and shutdown. They include the following:

   D - Dead loads or their related internal moments and forces including any permanent equipment loads

   L - Live loads or their related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence

   F - Lateral and vertical pressure of liquids, or their related internal moments and forces; F is included in D for steel structures.

TABLE 3.8–3 LOADS AND LOADING COMBINATIONS (CONTINUED)

H - Lateral earth pressure, or its related internal moments and forces; H is included in L for steel structures

T_o - Thermal loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition; To is included in D for concrete structures equations 1, 2, 2a, 3, 3a, 4, 4a, 5, 5a, 6, 6a, 7, and 7a.

R_o - Pipe loads during operating or shutdown conditions, based on the most critical transient or steady-state condition; Ro is included in D for concrete structures equations 1, 2, 2a, 3, 3a, 4, 4a, 5, 5a, 6, 6a, 7, and 7a.

2. Severe Environmental Loads - Those loads that could infrequently be encountered during the plant life. They include:

OBE- Loads generated by the operating basis earthquake

W - Loads generated by the design wind specified for the plant site (Section 3.3.1)

3. Extreme Environmental Loads - Those loads which are credible but highly improbable. They include:

SSE- Loads generated by the safe shutdown earthquake

W_t - Loads generated by the design tornado specified for the plant site (Section 3.3.2)

4. Abnormal Loads. Those loads generated by a postulated high-energy pipe break accident within a building and/or compartment thereof. Included in this category are the following:

P_a = Maximum differential pressure load generated by a postulated break

T_a = Thermal loads under accident conditions generated by a postulated break

R_a = Pipe and equipment reactions under accident conditions generated by a postulated break

Y_r = Loads on the structure generated by the reaction on the broken high-energy pipe during a postulated break

Y_j = Jet impingement load on a structure generated by a postulated break

Y_m = Missile impact load on a structure generated by or during a postulated break, such as pipe whipping

TABLE 3.8–4 LOAD COMBINATIONS FOR ASME III CLASS 2 PENETRATIONS EXCEPT QUENCH, RECIRCULATION, AND SAFETY INJECTION PIPING

Plant Operating Condition	NC 3600 Equations	Load Combinations[1]	Allowable Stress
Design	8	$P_d + D$	S_h
Normal/Upset	9	$P_p + D + E + H$	$1.2\ S_h$
	10 (2)	$T + R + A$	S_A
	10a	S	$3\ S_c$
	11 (2)	$P_d + D + T + R + A$	$S_A + S_h$
Emergency	9	$P_p + D + H + E' + Y$	$1.8\ S_h$
Faulted	9	$P_p + D + H + E' + Y' + A1$	$2.4\ S_h$ (3)
		$P_p + D + B'$	$2.4\ S_h$ (3)
Test	8	$P_t + D$	S_c

NOTES:

1. Refer to Table 3.8–7 for definition of loadings.
2. Either the requirements of Equation (10) or (11) must be satisfied.
3. In pipe break exclusion zones, the allowable stress is 1.8 Sh.
4. In break exclusion area, sum of stress given by Equations (9) and (10) should not exceed 0.8 ($S_A + 1.2\ S_h$).

TABLE 3.8–5 LOAD COMBINATIONS FOR ASME III CLASS 2 PENETRATIONS FOR THE QUENCH SPRAY, RECIRCULATION SPRAY, AND SAFETY INJECTION SYSTEMS

Plant Operating Condition	NC 3600 Equations	Load Combinations (1)	Allowable Stress
Design	8	$P_d + D$	S_h
Normal/Upset	9	$P_p + D + E + H$	$1.2\ S_h$
	10 (2)	$T + R + A$	S_A
	10a	S	$3\ S_c$
	11 (2)	$P_d + D + T + R + A$	$S_A + S_h$
Emergency	9	$P_p + D + H + E' + Y$	$1.8\ S_h$
Faulted	9	$P_p + D + H + E' + Y' + A1$	$1.8\ S_h$
		$P_p + D + B'$	$1.8\ S_h$
	10 (2)	$T + R' + A' + X$	S_A
	11 (2)	$P_d + D + T + R' + A' + X$	$S_A + S_h$
Test	8	$P_t + D$	S_c

NOTES:
1. Refer to Table 3.8–7 for definition of loadings.
2. Either the requirements of Equation (10) or (11) must be satisfied.
3. In break exclusion area, sum of stress given by Equations (9) and (10) should not exceed 0.8 (S_A + 1.2 S_h).

TABLE 3.8–6 LOAD COMBINATIONS FOR ASME III CLASS MC SLEEVED PENETRATIONS

Plant Design or Operating Condition	ASME Code Reference	Load Combinations(1)	Allowable Stress(2)
Design			
Primary Stress Intensity	Fig. NE-3221-1	$P_d + D + E + H$	$P_m < S_m$. $P_L < 1.5S_m$ and $P_L + P_b < 1.5S_m$
Normal/Upset			
Primary and Secondary Stress Range	NB-3222.2	$P_O + T + R + A + E + H + L$	$P_L + P_b + P_e + Q < 3S_m$
Peak Stress Range	NB-3222.4e	$P_O + T + R + A + E + H + L$	$P_L + P_b + P_e + Q + F$ (3)
Simplified Plastic-Elastic Analysis	NB-3228.3(a)	$P_O + D + E + H + L$	$P_L + {}_{Pb} + Q < 3S_m$ (4)
Expansion Stress Intensity	NB-3222.3	$T + R$	$P_e < 3S_m$
Emergency	Fig. NB-3224-1	$P_L + D + E' + H + Y$	(5)
Faulted	Appendix F	$P_L + D + E' + H + Y' + A1$	(6)
	Appendix F	$P_L + D + E' + H + B'$	(6)
	Appendix F	$P_L + D + E' + X + T + R' + A'$	(6) (7))

NOTES:

1. Refer to Table 3.8–7 for definition of loadings.
2. Class MC allowables from Table I-1.1 of ASME Code per NE 3131(d).
3. The allowable cumulative usage factor is 1.0.
4. Q excludes the thermal bending stress.
5. P_m <1.2S_m or S_y, PL <1.8S_m or 1.5S_y and $P_L + P_b$ <1.8S_m or 1.55$_y$ (use the largest value).
6. P_m <S_m, P_L <1.5S_m and $P_L + P_b$ <1.5S_m

   Elastic - Elastic Analysis - Use an S_m value equal to the lesser of 2.4S_m or 0.7OS_u(8).

   Inelastic - Elastic Analysis - Use an S_m value equal to the greater of 0.7S_u or [S_y + 1/3 (S_u - S_y)](8).
7. Includes the steady-state load from B' and the stress induced in the sleeve from the liner.
8. Use 85 percent of these values if Y' or B' exists in the load combination.

TABLE 3.8–7 NOMENCLATURE FOR TABLES 3.8-4 THROUGH 3.8-6

D	-	Sustained mechanical loads, including deadweight of piping, components, contents, and insulation.
T	-	Loads due to thermal expansion of the system in response to average fluid temperature.
R	-	Loads induced in the piping due to the thermal growth of equipment and/or structures to which the piping is connected as a result of plant normal or upset plant conditions.
R'	-	Loads induced in the piping due to thermal growth of equipment and/or structures to which the piping is connected as a result of plant faulted plant conditions. Note that R' includes R.
E	-	Inertia effects of the OBE.
E'	-	Inertia effects of the SSE.
A1	-	Loads induced in the piping due to inertia effects of LOCA or displacements due to LOCA or displacements due to pipe rupture.
A	-	Loads induced in the piping due to response of the connected equipment and/or civil structures to the OBE (commonly referred to as OBE anchor movements).
A'	-	Loads induced in the piping due to response of the connected equipment and/or civil structures to the SSE (commonly referred to as SSE movements).
S	-	Loads induced due to building settlement effects.
H	-	Loads resulting from occasional loads other than seismic. Examples of these loads would be: water hammer, steam hammer, opening and closing of safety relief valves, etc, as defined for the emergency plant condition.
Y	-	Effects of pipe striking pipe (pipe whip) or effects of blowdown of an adjacent system (jet impingement loads), as defined for the emergency plant condition.
Y'	-	Effects of pipe striking pipe (pipe whip) or effects of blowdown of an adjacent system (jet impingement loads), as defined for the faulted plant condition.
X	-	Loads induced in the piping due to pressure response (growth) of the containment during a faulted plant condition.
L	-	Local stress effects in piping and/or piping components due to sudden changes in fluid temperature. These loads are commonly referred to as thermal transient effects.
B'	-	Loads on restraints induced by blowdown and subsequent pipe response of a ruptured system for faulted plant conditions.
P_d	-	Internal pressure loads due to design pressure.
P_p	-	Internal pressure loads due to peak pressure.
P_t	-	Internal pressure loads due to test pressure.
P_1	-	Internal pressure loads for emergency and faulted conditions as applicable.
P_o	-	Internal pressure loads due to range of the operating pressure.

FIGURE 3.8–1 CONTAINMENT STRUCTURE FOUNDATION DETAIL

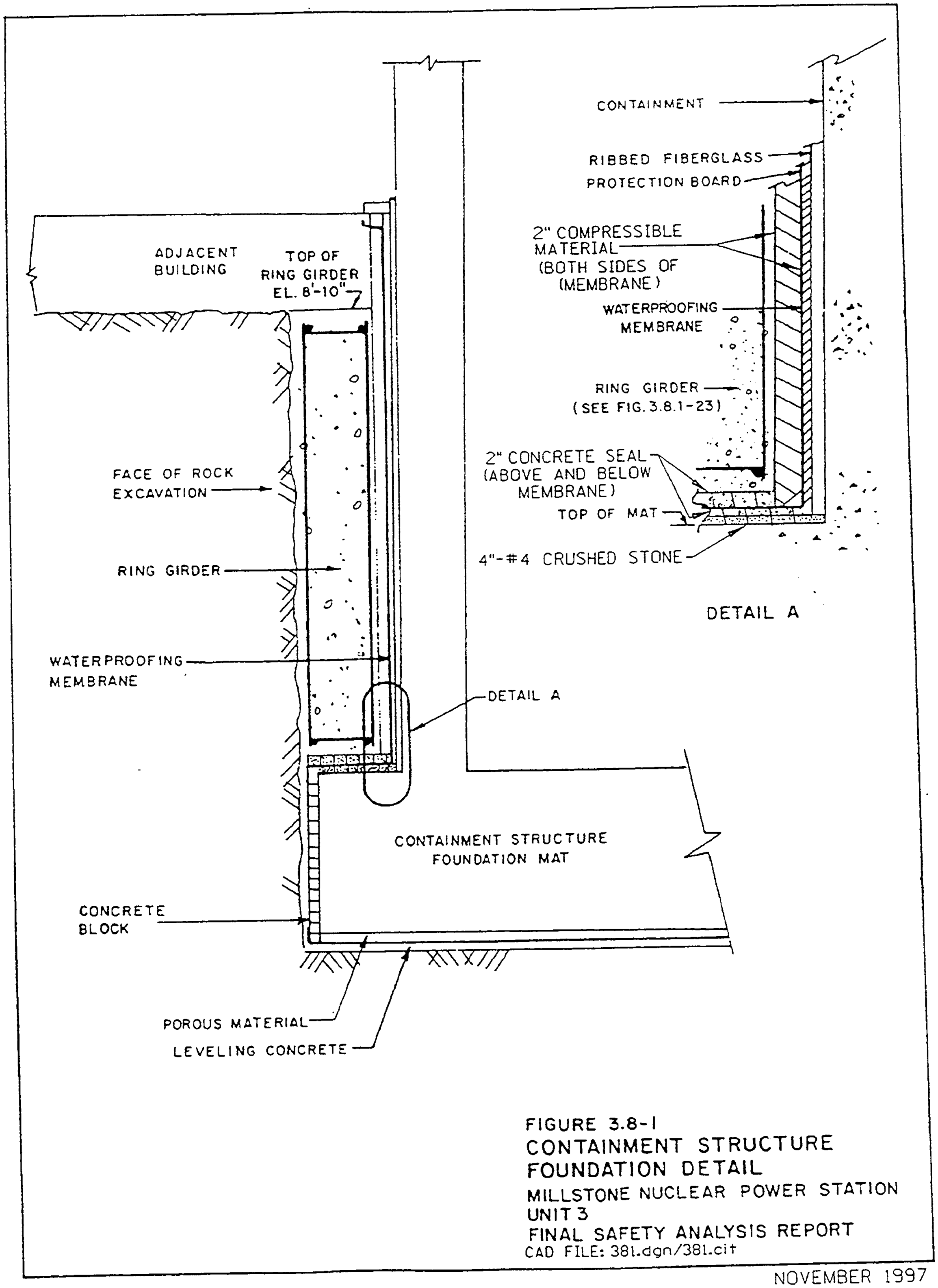


FIGURE 3.8–2 TYPICAL DETAIL OF DOME CYLINDER JUNCTION

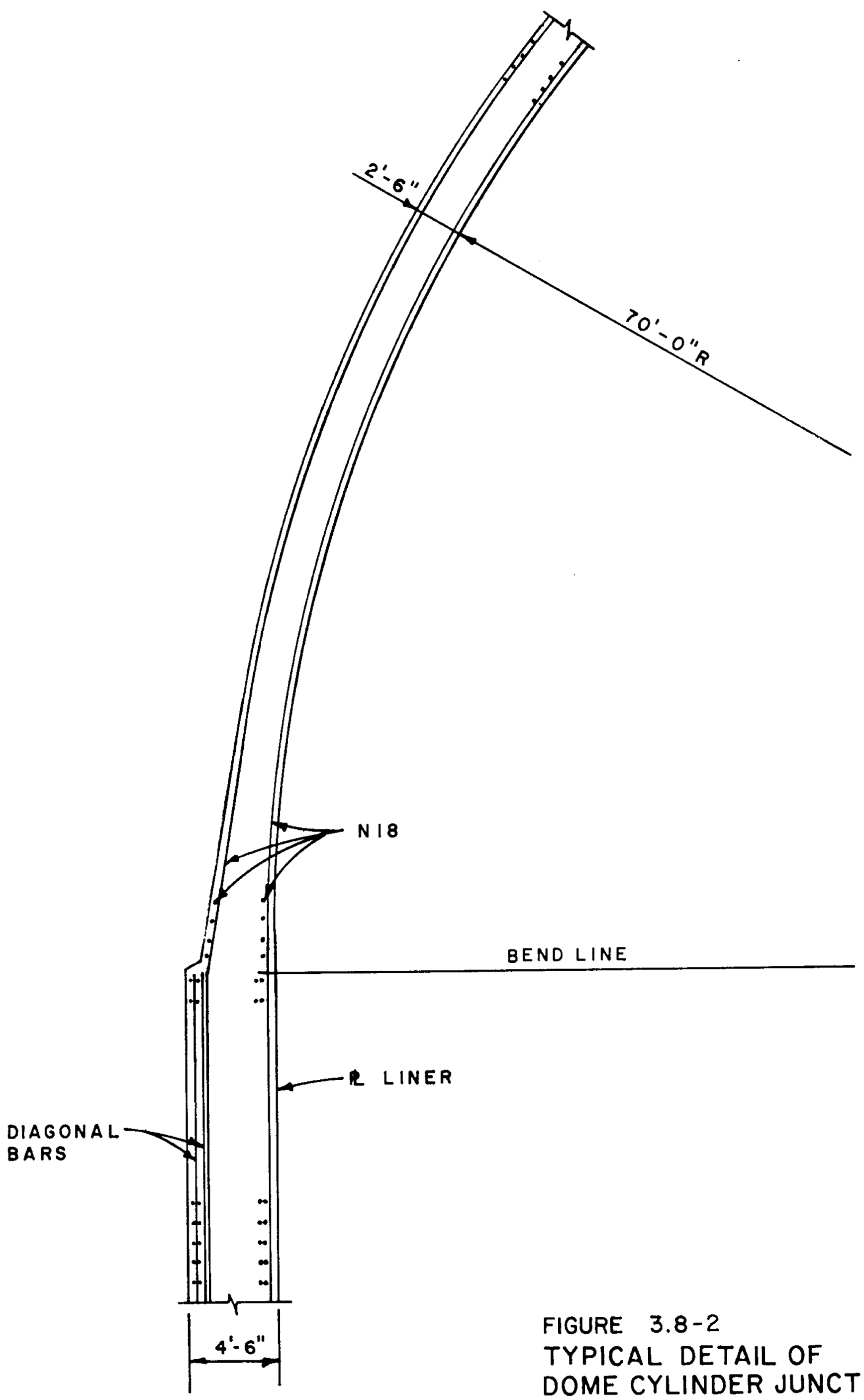


FIGURE 3.8-2
TYPICAL DETAIL OF
DOME CYLINDER JUNCTION
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–3 KNUCKLE PLATE

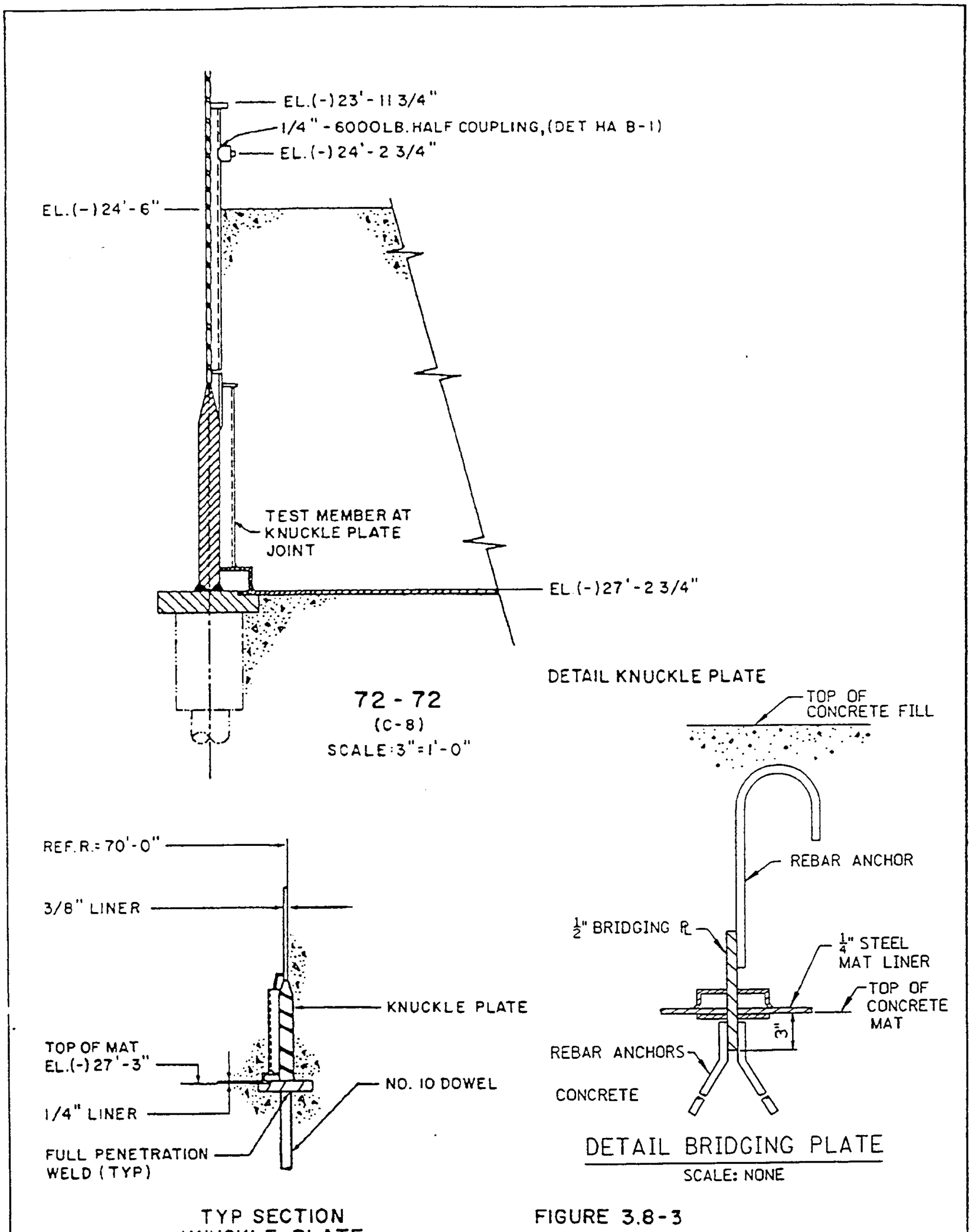


FIGURE 3.8-3
KNUCKLE PLATE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
CAD FILE: 383.dgn/383.cit

FIGURE 3.8–4 MAT REINFORCEMENT WITH RADIAL SHEAR BAR ASSEMBLY

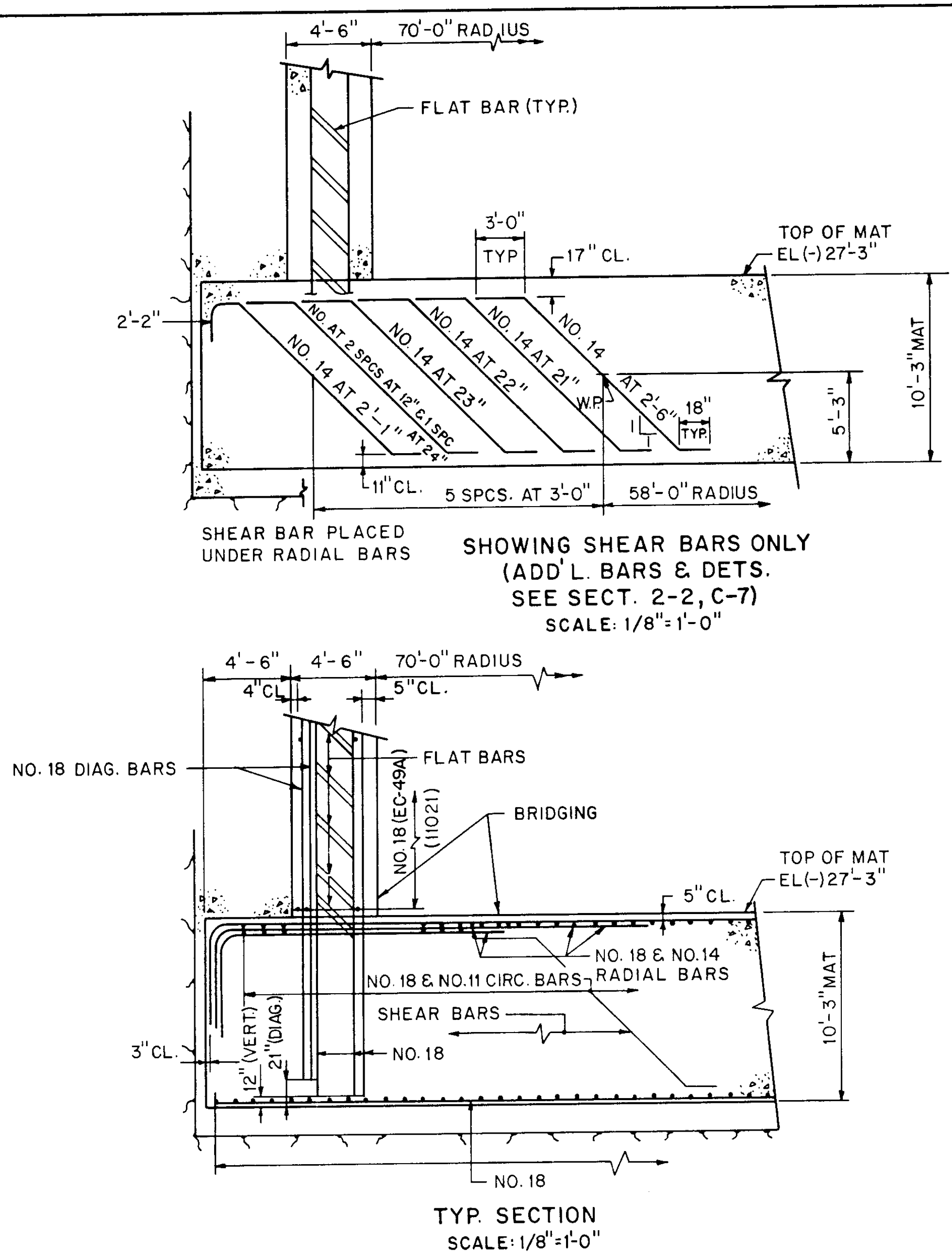


FIGURE 3.8-4
MAT REINFORCEMENT WITH
RADIAL SHEAR BAR ASSEMBLY
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–5 REINFORCEMENT AROUND MAIN STEAM PENETRATIONS

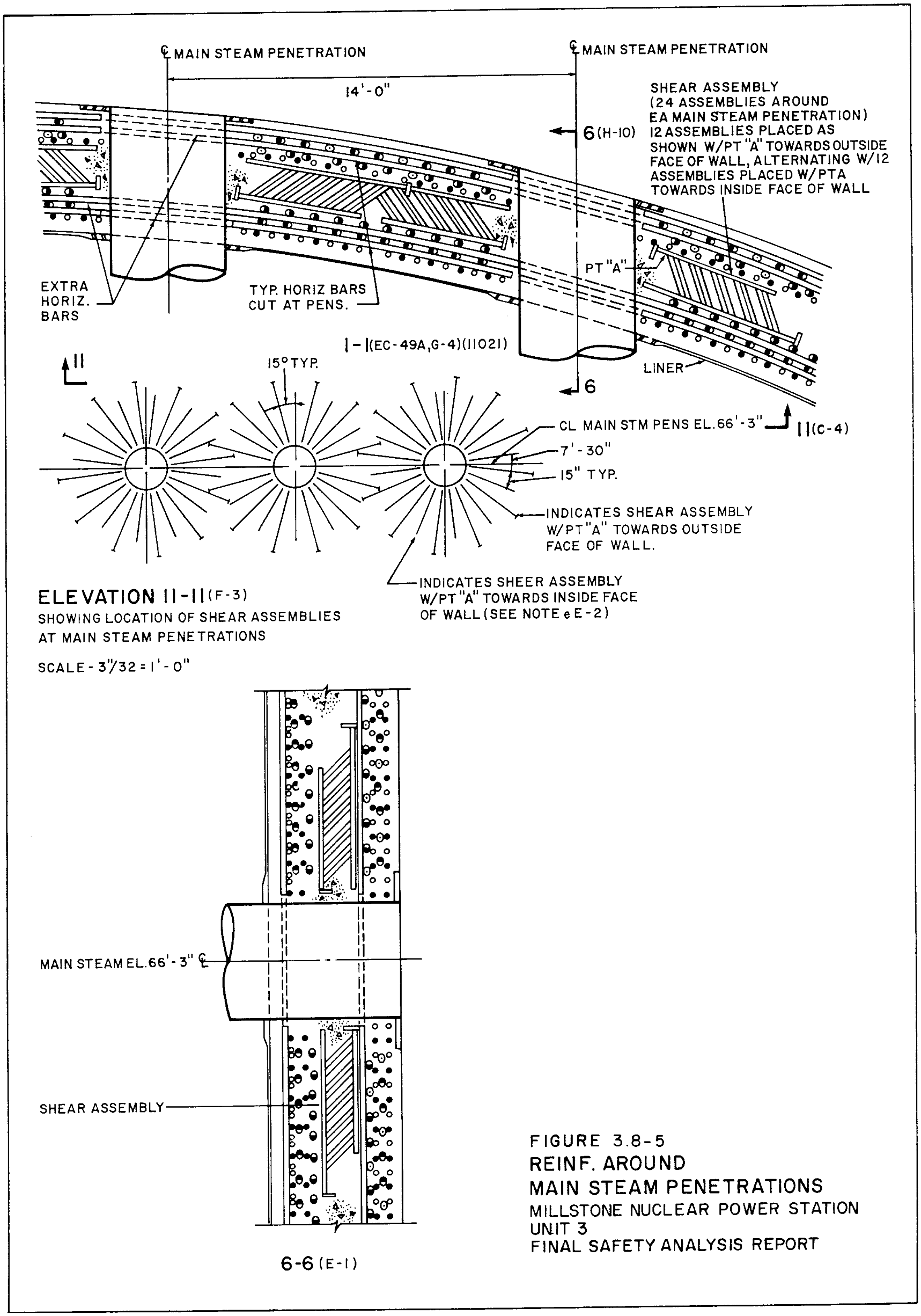


FIGURE 3.8–6 REINFORCING DETAILS SECTIONS THROUGH RING BEAM OF PERSONNEL ACCESS LOCK

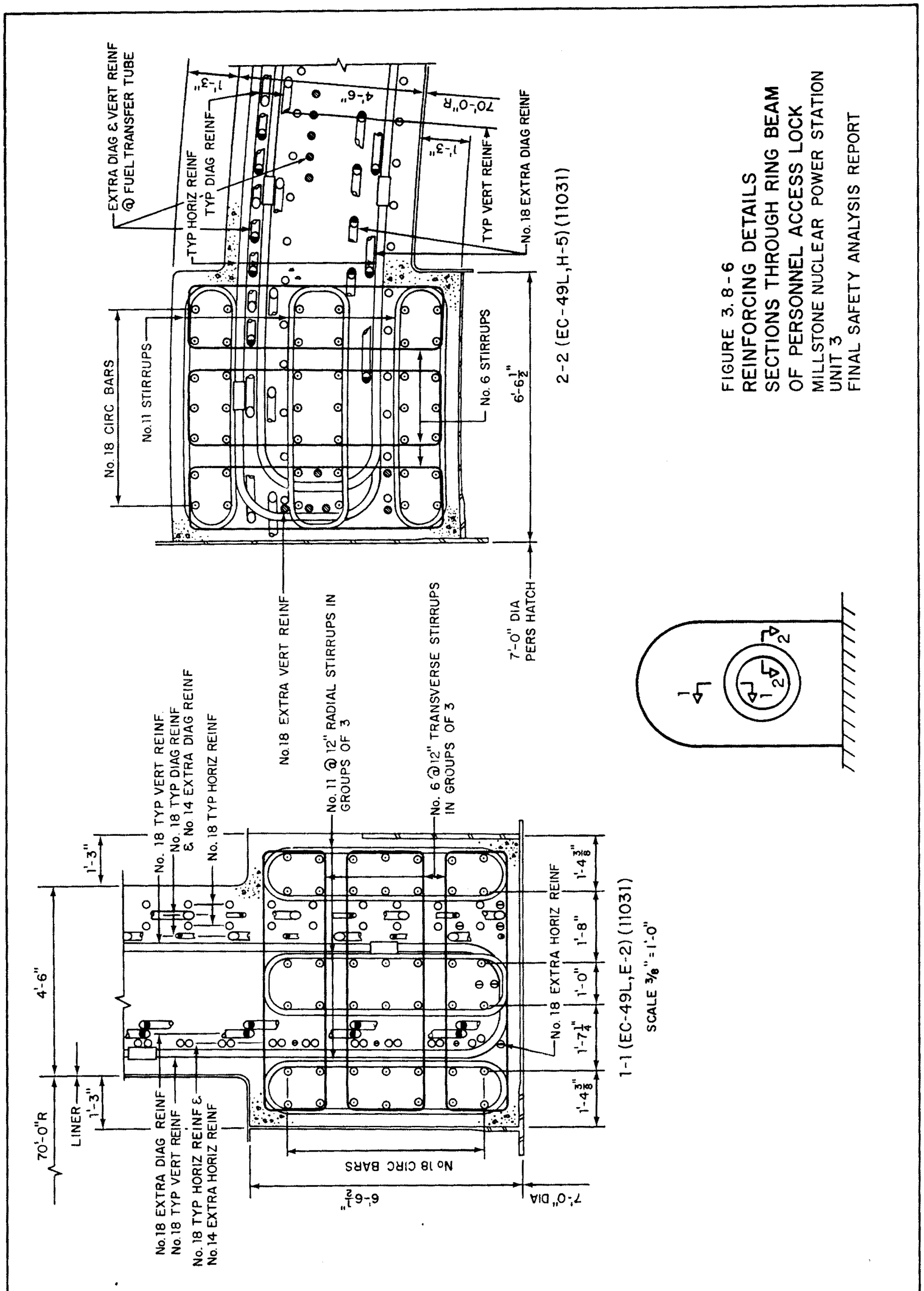
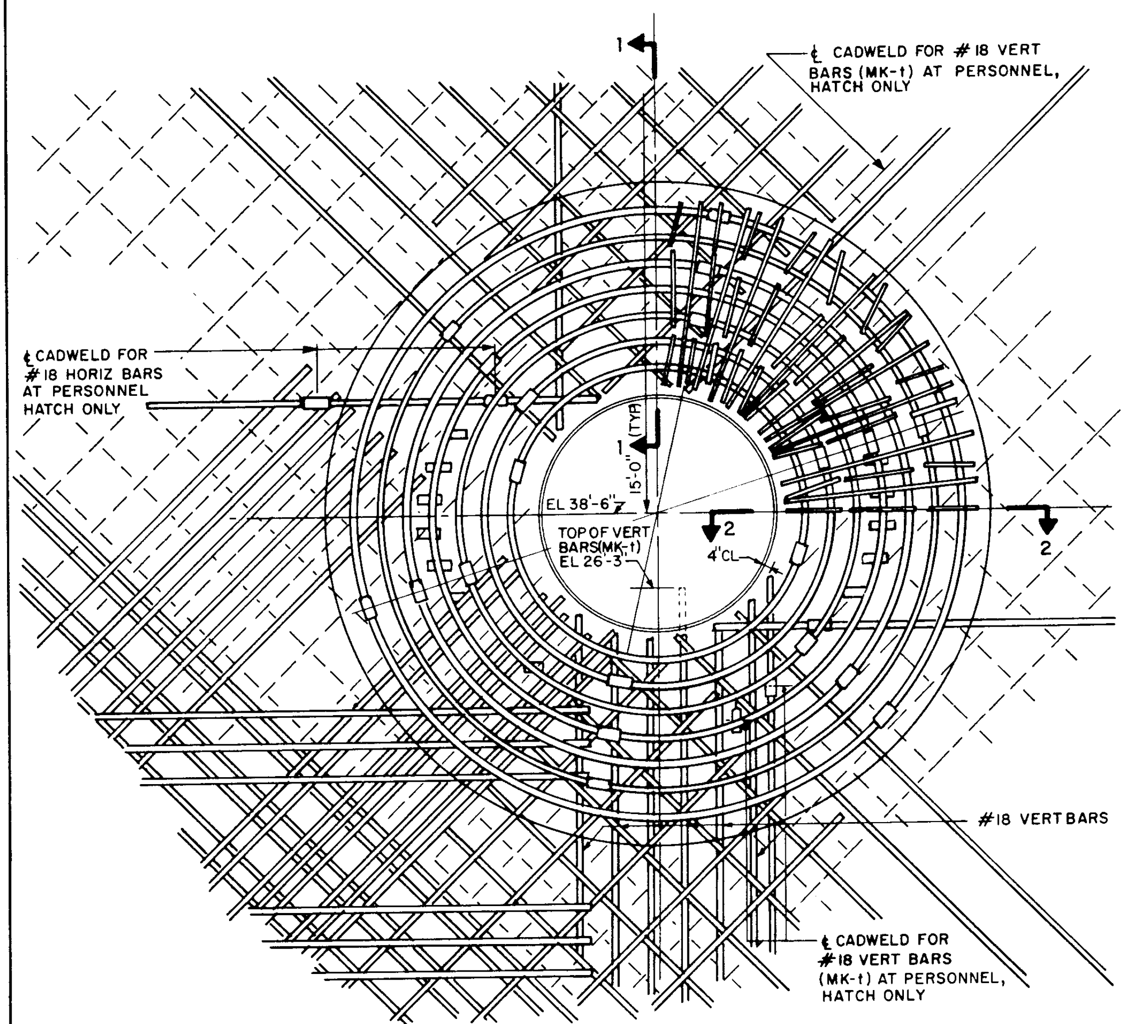


FIGURE 3.8–7 REINFORCING DETAILS PERSONNEL ACCESS LOCK OPENING



REF: EC 49M-3

FIGURE 3.8-7
REINFORCING DETAILS
PERSONNEL ACCESS LOCK OPENING
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–8 TYPICAL REINFORCING DETAILS SECTIONS THROUGH RING BEAM OF EQUIPMENT ACCESS HATCH

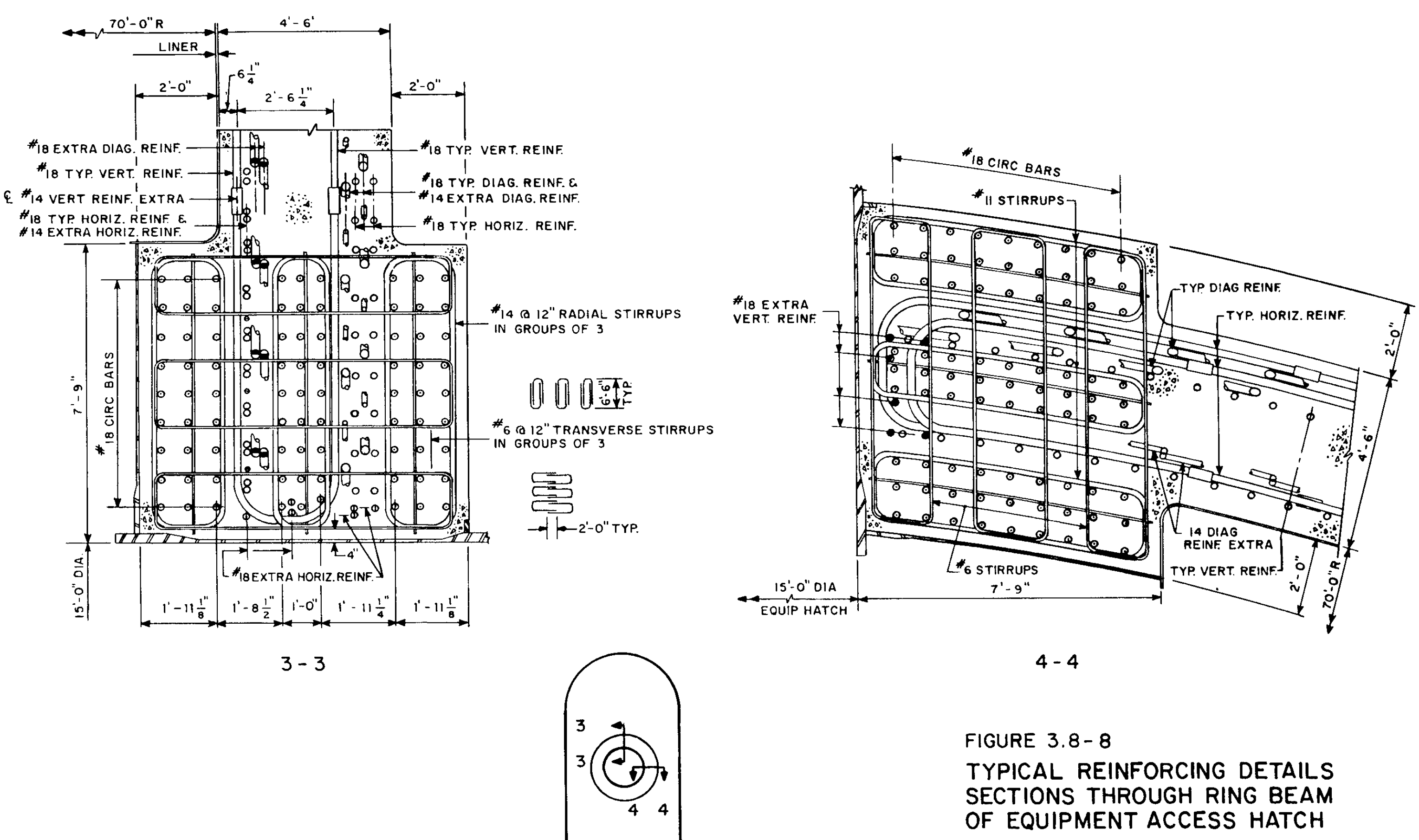


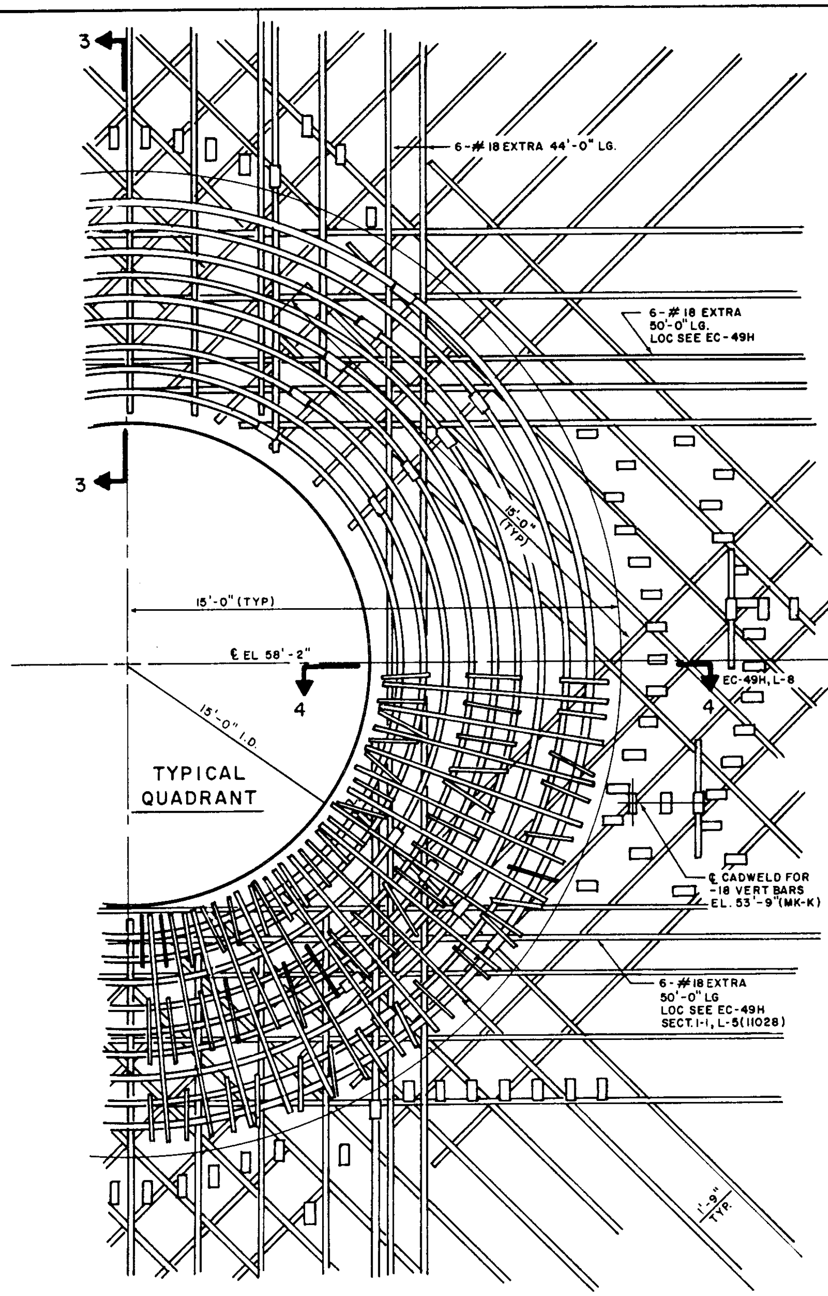
FIGURE 3.8-8

TYPICAL REINFORCING DETAILS SECTIONS THROUGH RING BEAM OF EQUIPMENT ACCESS HATCH

MILLSTONE NUCLEAR POWER STATION UNIT 3

FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–9 REINFORCING DETAILS EQUIPMENT ACCESS HATCH OPENING



REF.: EC 496

FIGURE 3.8-9
REINFORCING DETAILS
EQUIPMENT ACCESS HATCH OPENING
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–10 BASE PLATE DETAILS CONTAINMENT ENCLOSURE STRUCTURE

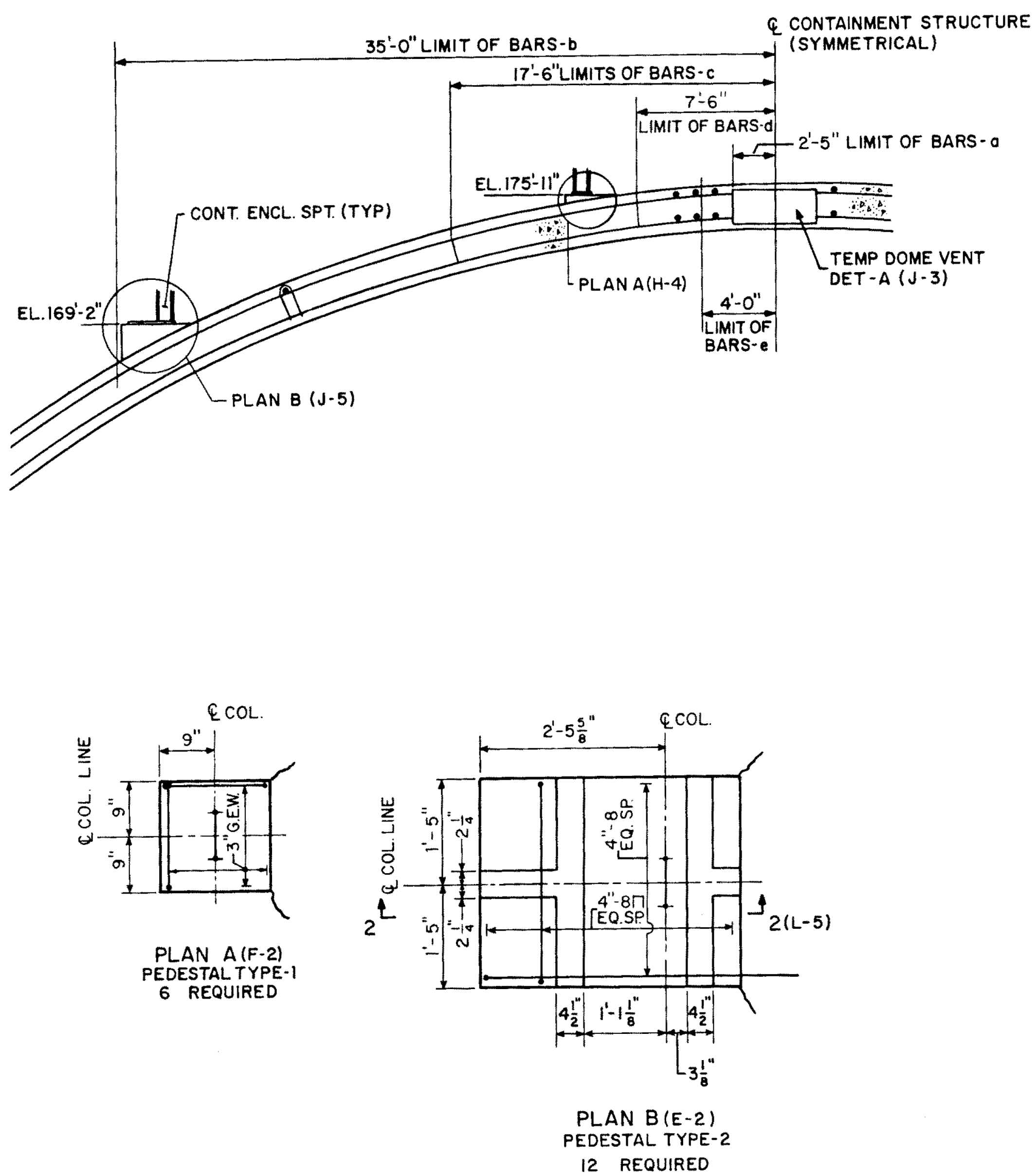


FIGURE 3.8-10
BASE PLATE DETAILS
CONTAINMENT ENCLOSURE STRUCTURE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–11 RING AT APEX OF CONTAINMENT STRUCTURE DOME

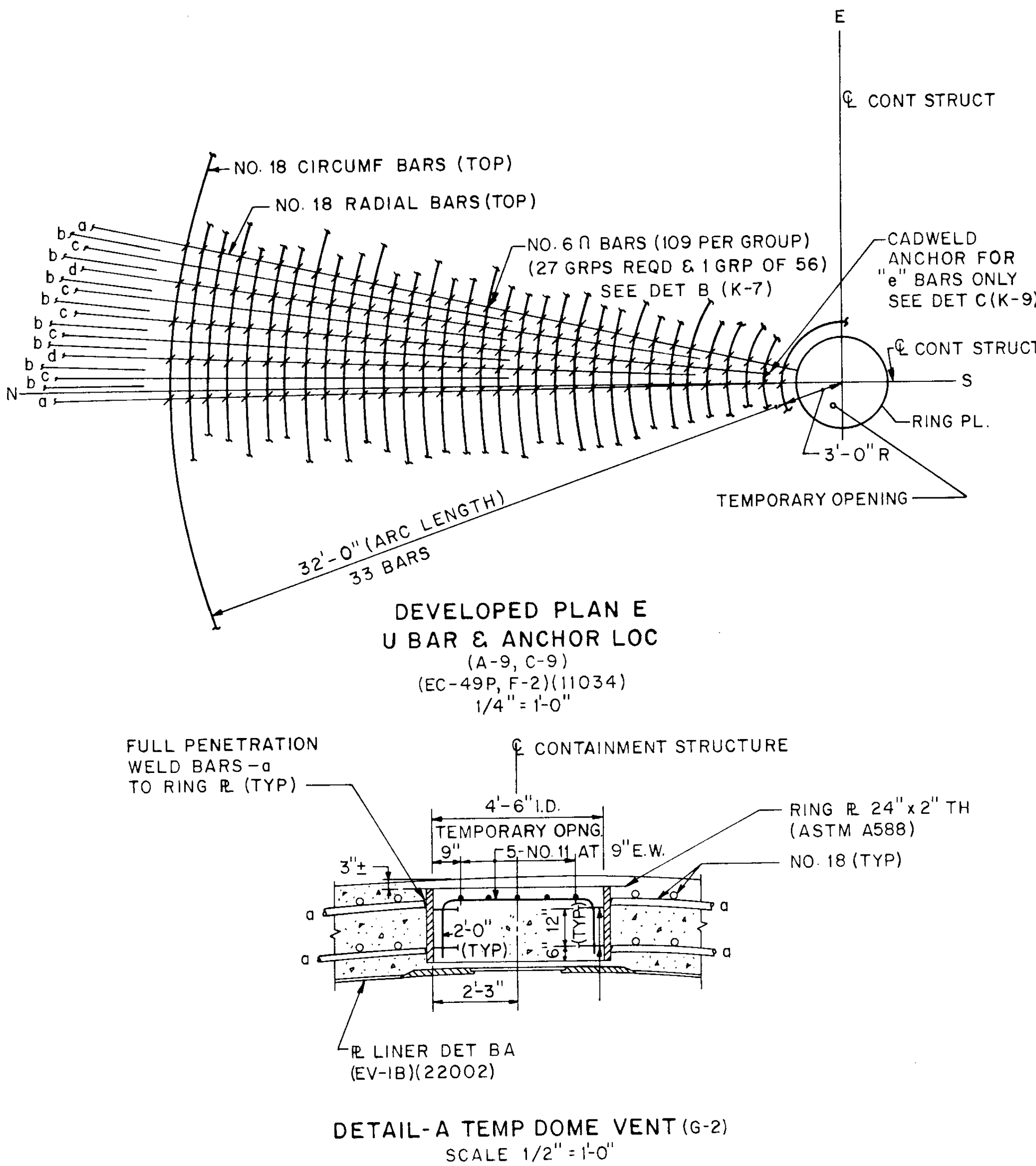


FIGURE 3.8-11
RING AT APEX OF
CONTAINMENT STRUCTURE DOME
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–12 ELEVATION DOME MAINTENANCE TRUSS

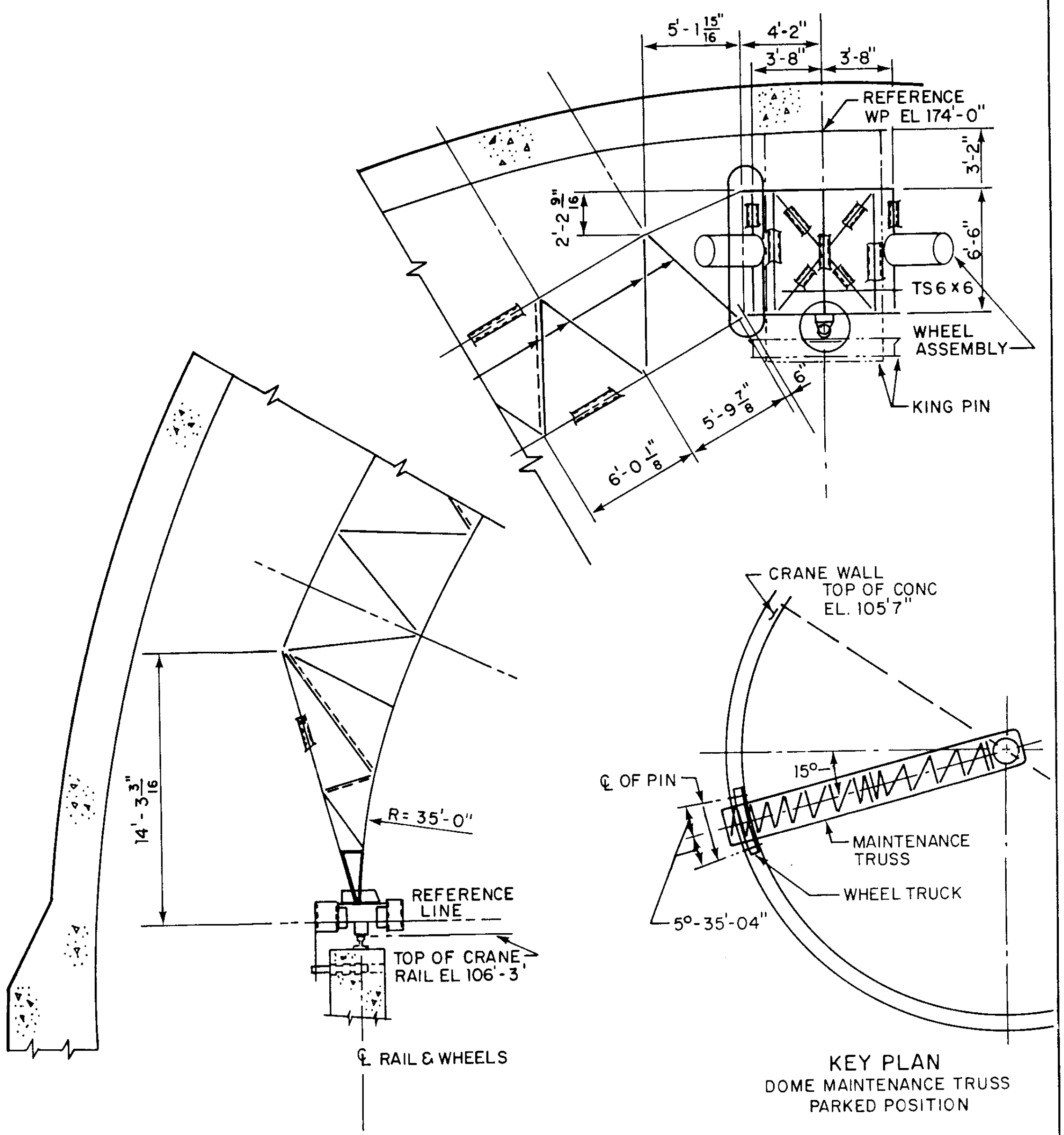


FIGURE 3.8-12
ELEVATION DOME MAINTENANCE TRUSS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–13 LINER KNUCKLE PLATE

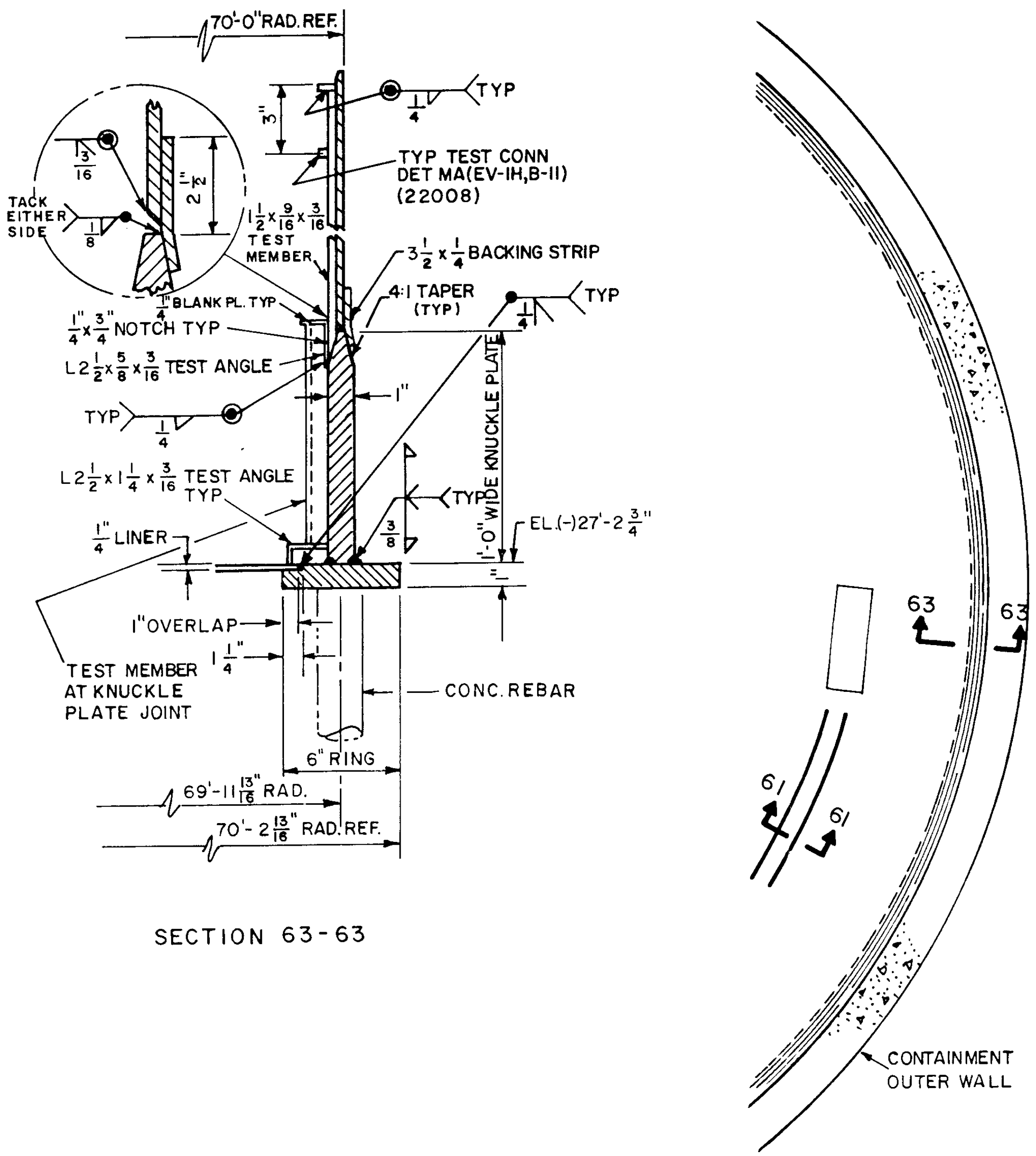


FIGURE 3.8-13
LINER KNUCKLE PLATE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–14 BASIC LINER DIMENSIONS

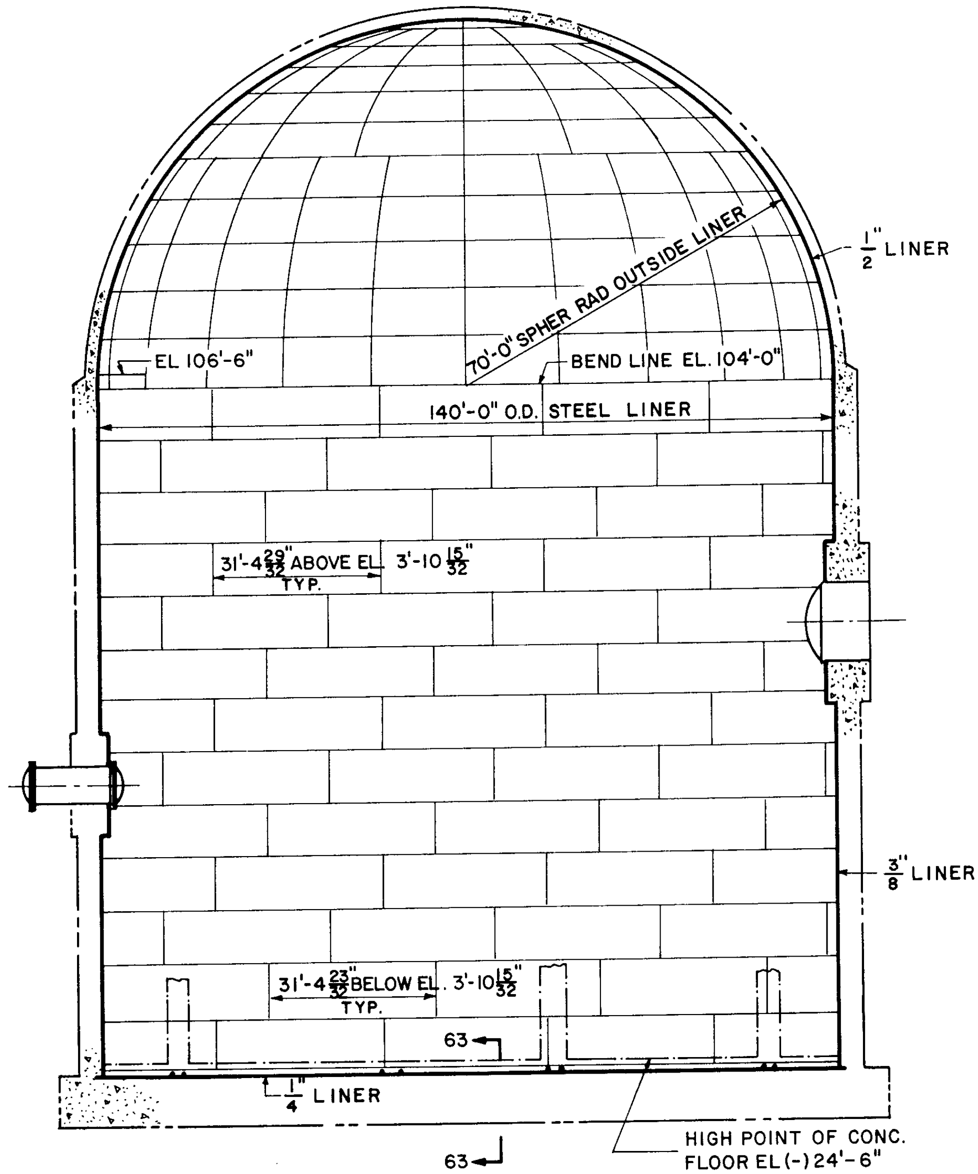


FIGURE 3.8-14
BASIC LINER DIMENSIONS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–15 TYPICAL PIPING PENETRATIONS

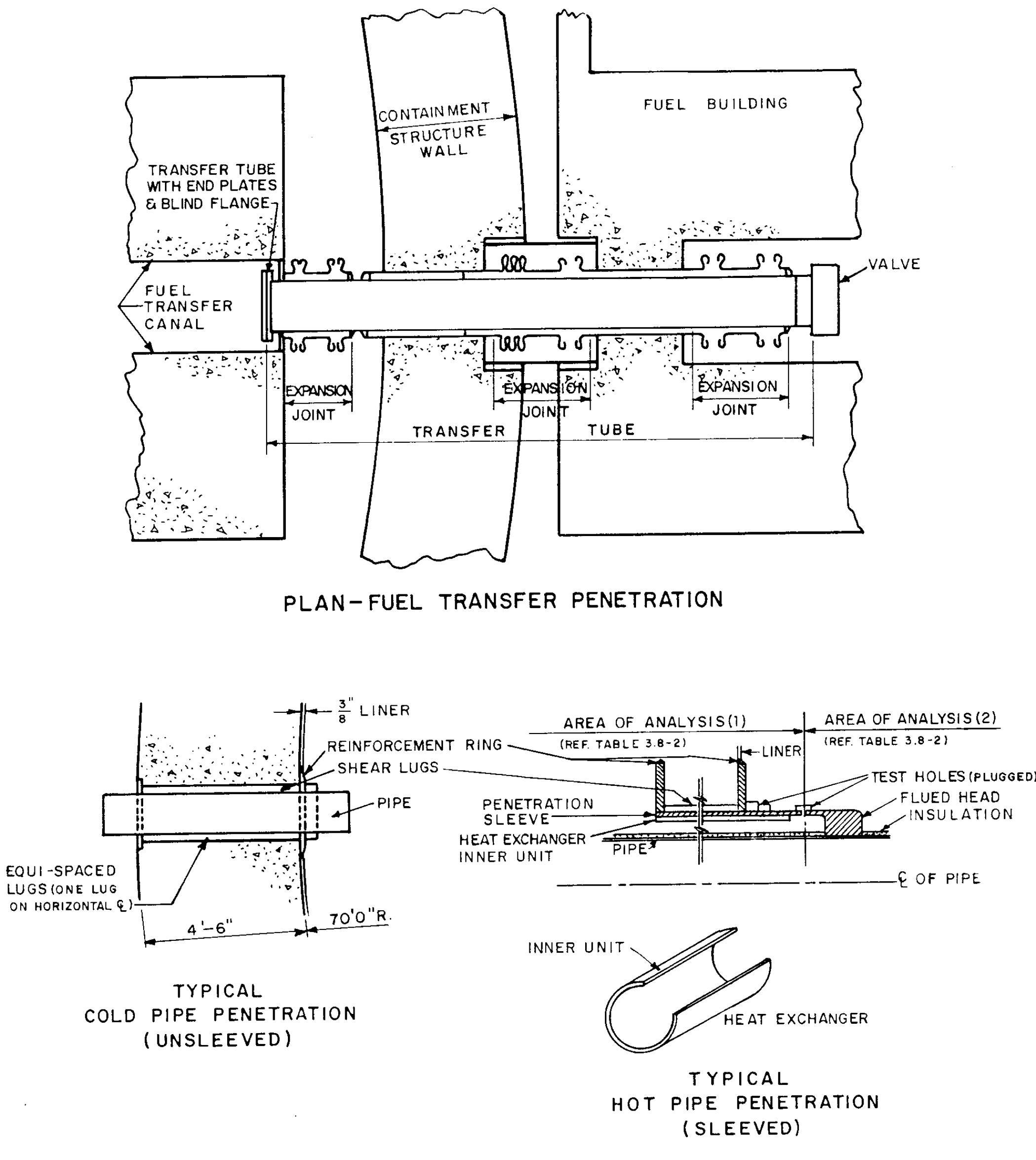


FIGURE 3.8-15
TYPICAL PIPING PENETRATIONS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

AMENDMENT 15 SEPTEMBER 1985

FIGURE 3.8–16 TYPICAL SLEEVED PIPING PENETRATION

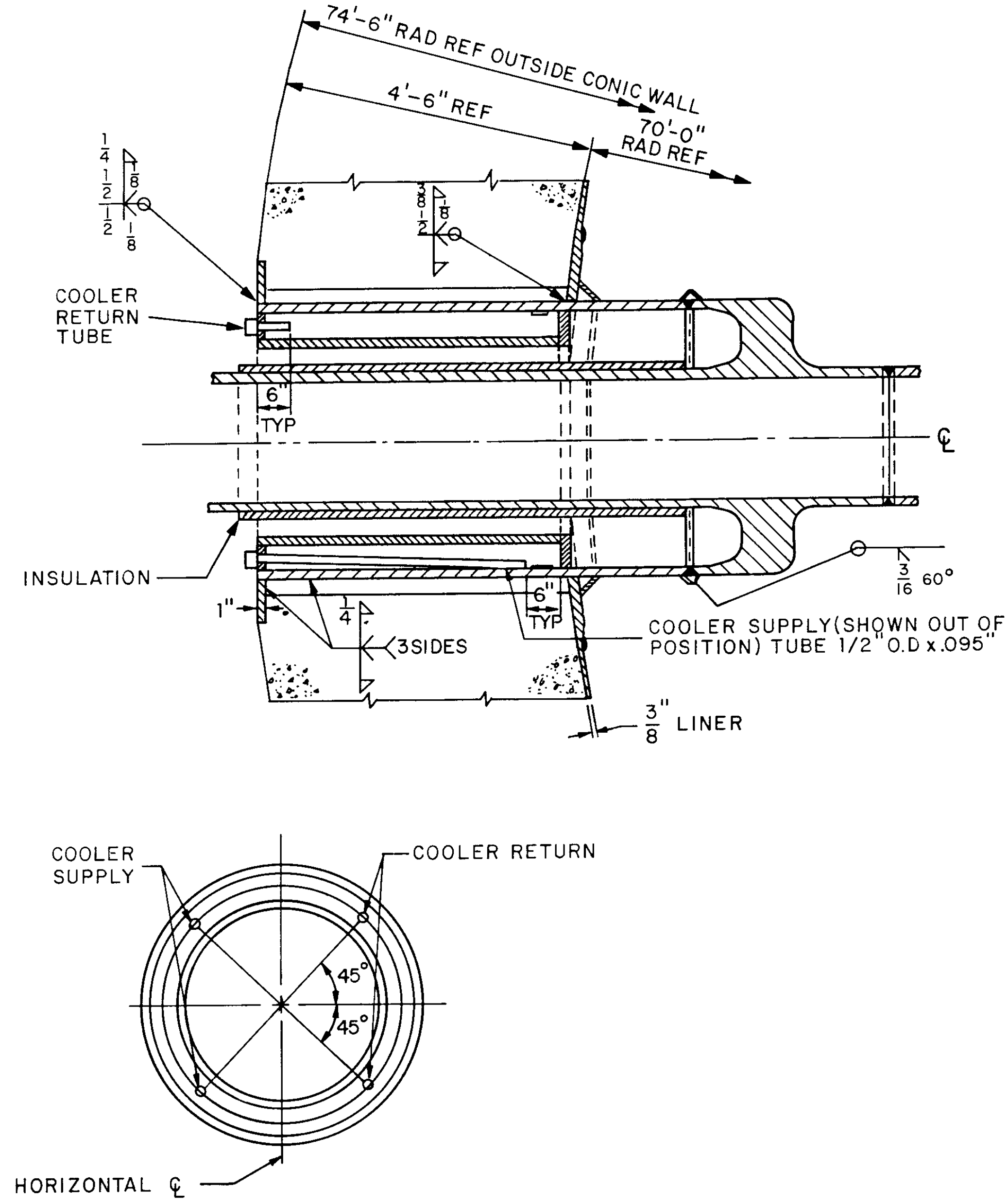
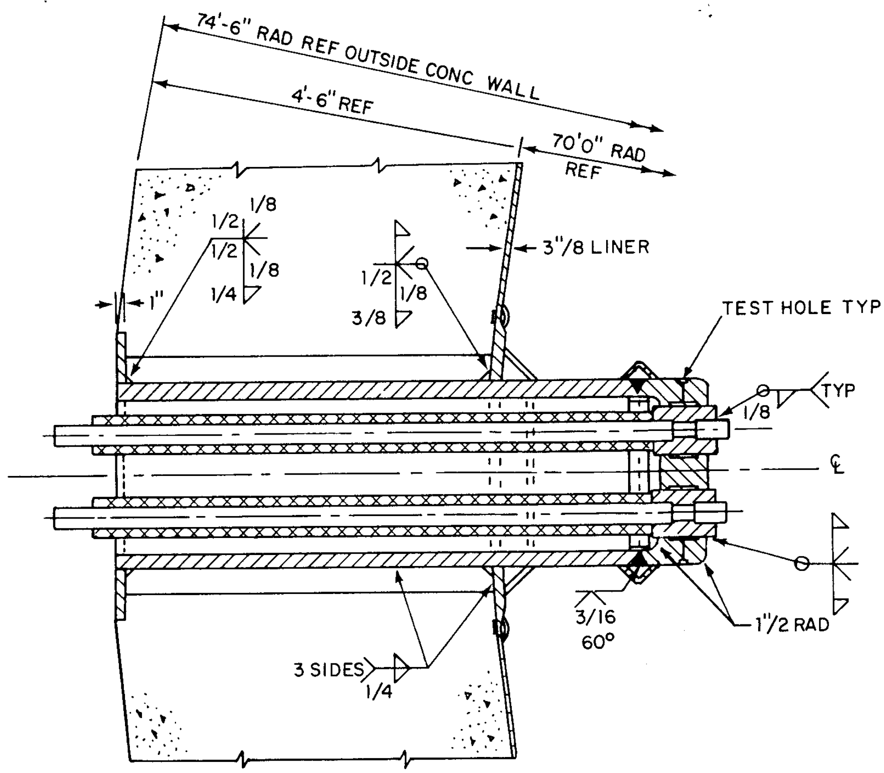


FIGURE 3.8-16
TYPICAL SLEEVED PIPING PENETRATION
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–17 MULTIPLE PIPE PENETRATION ASSEMBLY



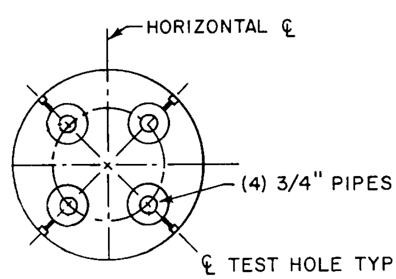
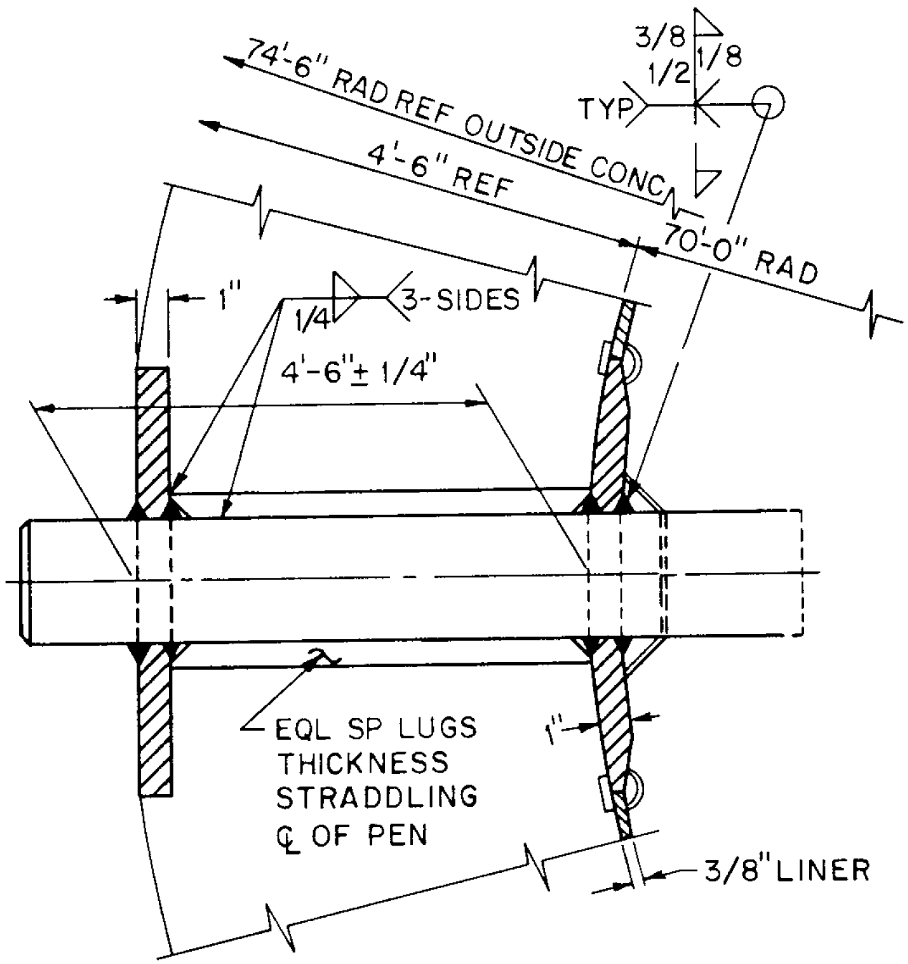


FIGURE 3.8–18 UNSLEEVED PIPING PENETRATION



PLAN

SINGLE RADIAL PENETRATION THRU LINER

FIGURE 3.8–19 ELECTRICAL PENETRATION

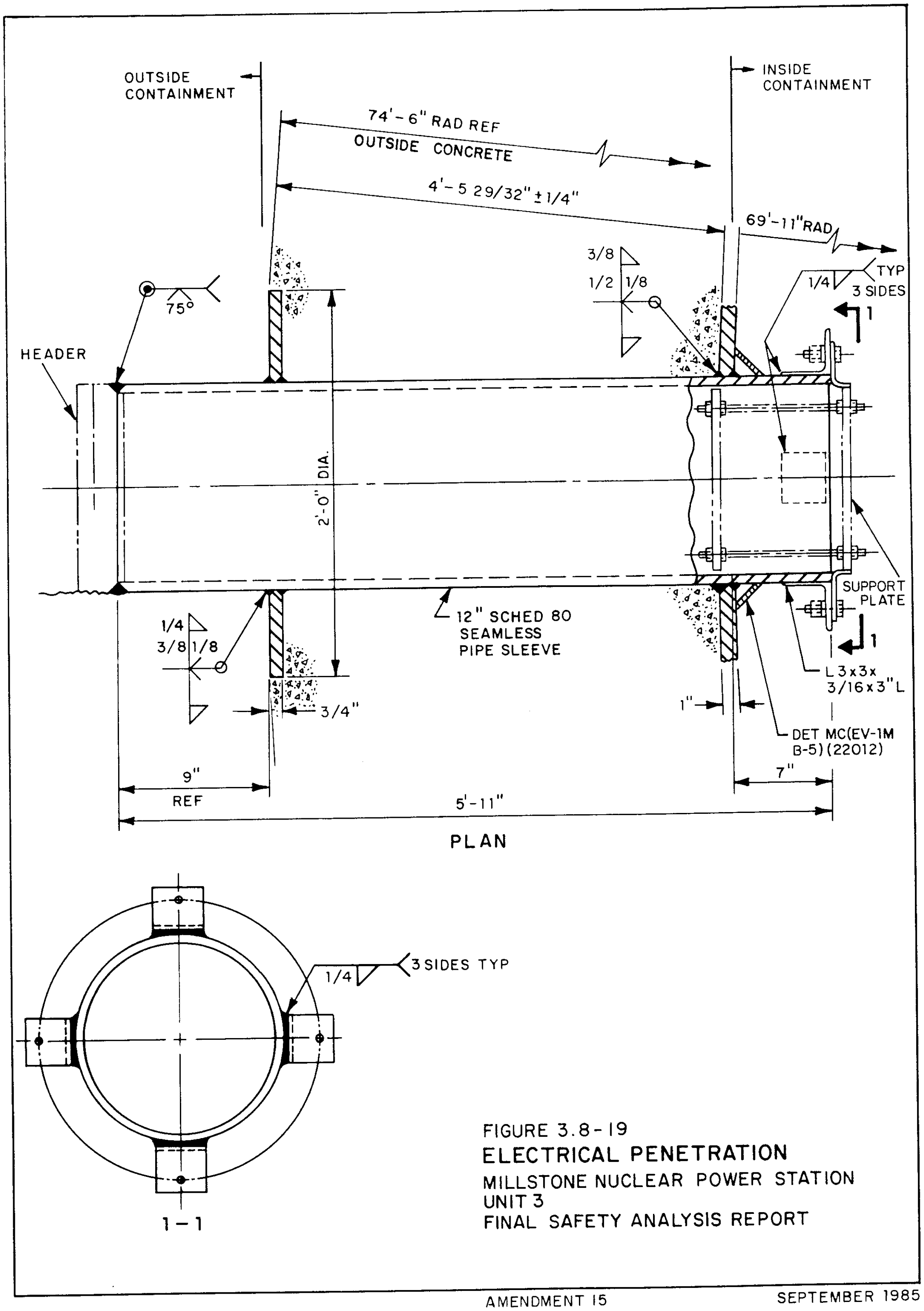


FIGURE 3.8–20 FUEL TRANSFER TUBE

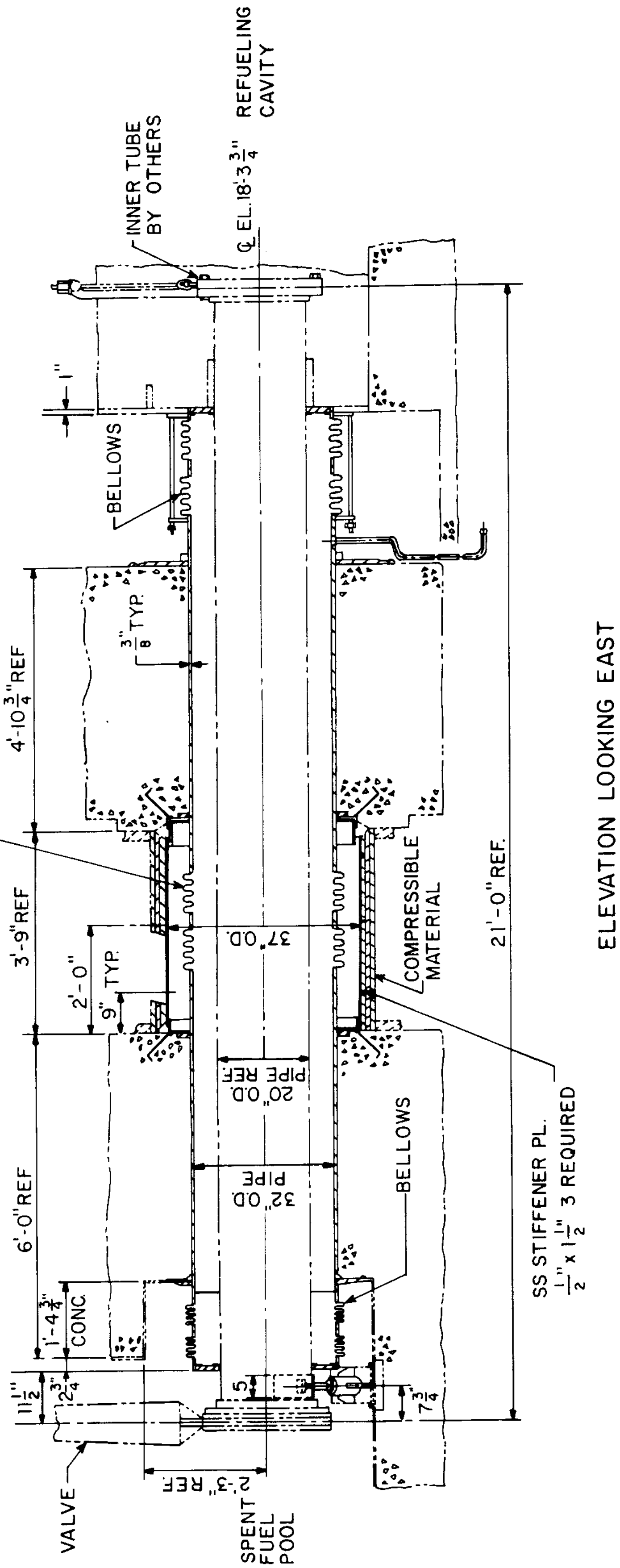


FIGURE 3.8–21 PERSONNEL HATCH

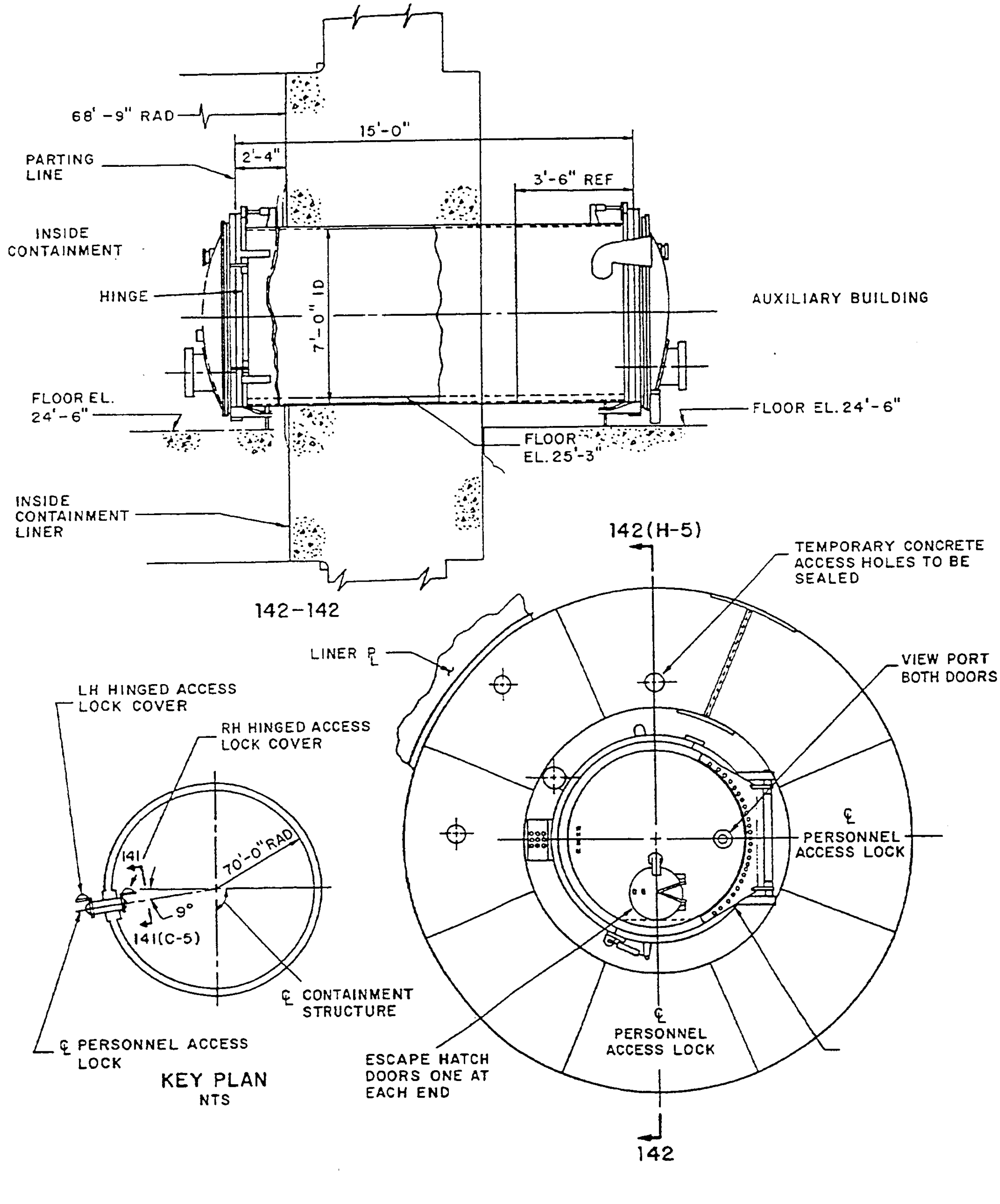


FIGURE 3.8–22 EQUIPMENT HATCH

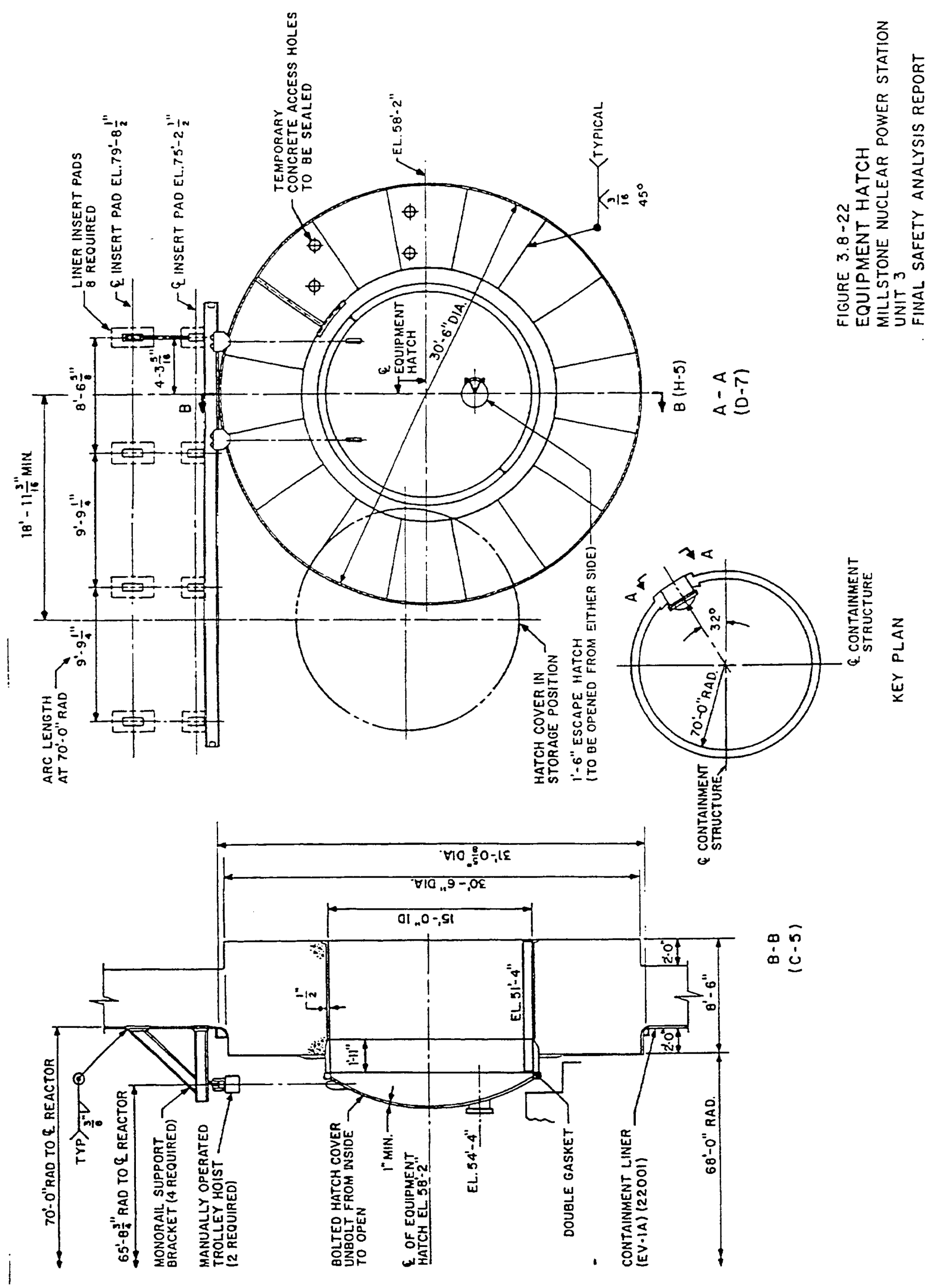


FIGURE 3.8-22
EQUIPMENT HATCH
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–23 CONTAINMENT STRUCTURE FOUNDATION DETAIL WITH RING GIRDER (SHEET 1 OF 2)

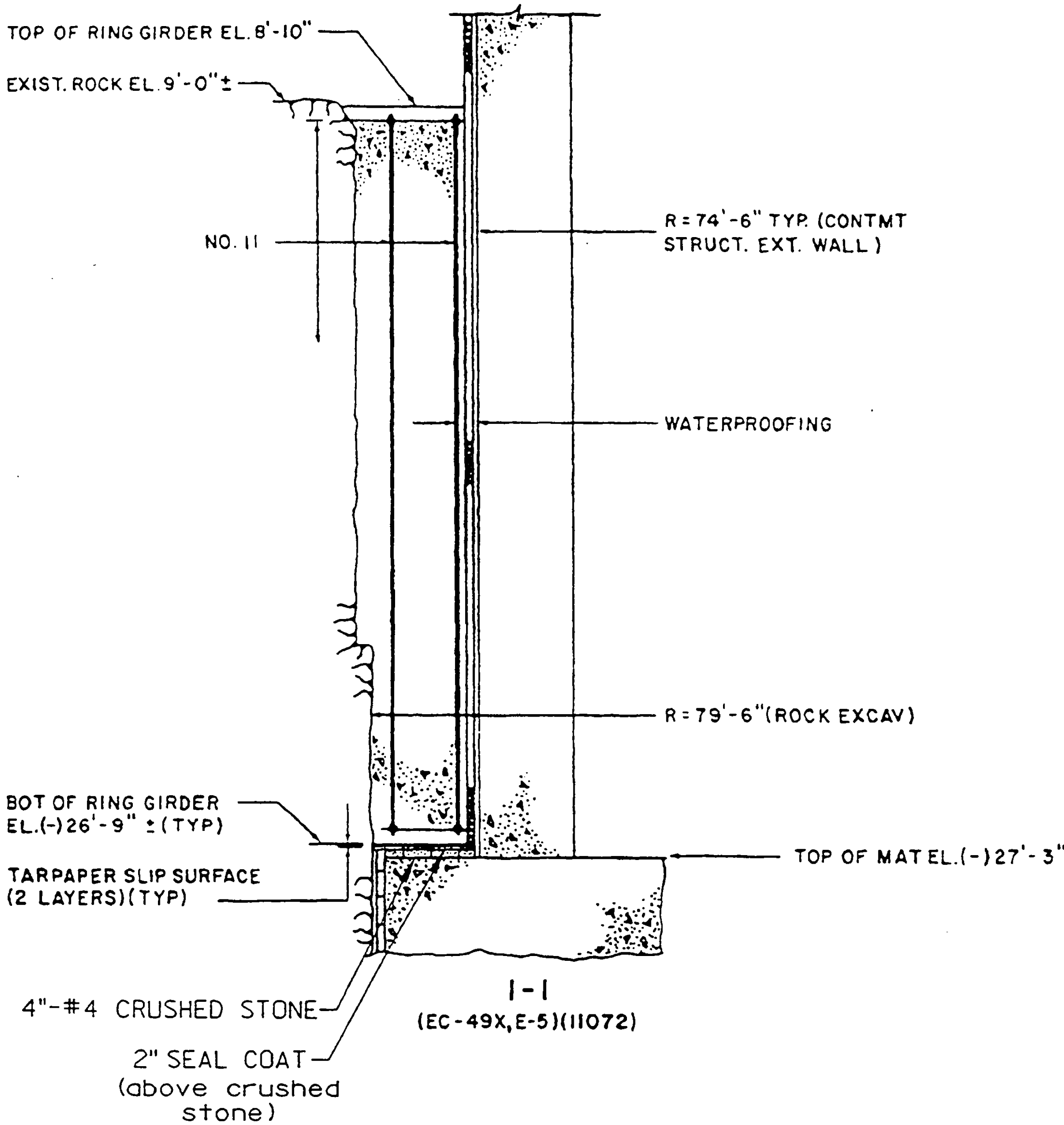


FIGURE 3.8–23 CONTAINMENT STRUCTURE FOUNDATION DETAIL WITH RING GIRDER (SHEET 2 OF 2)

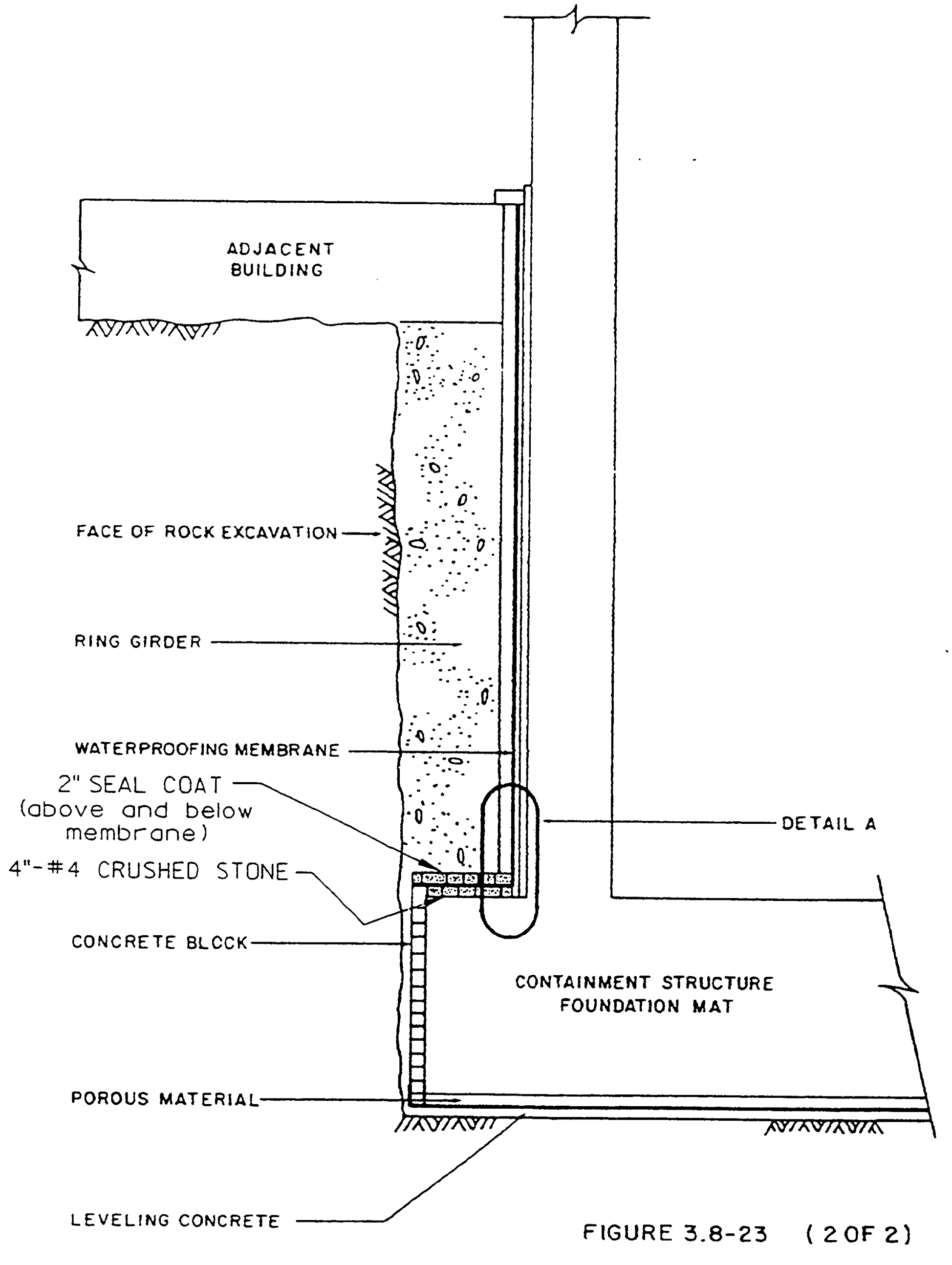
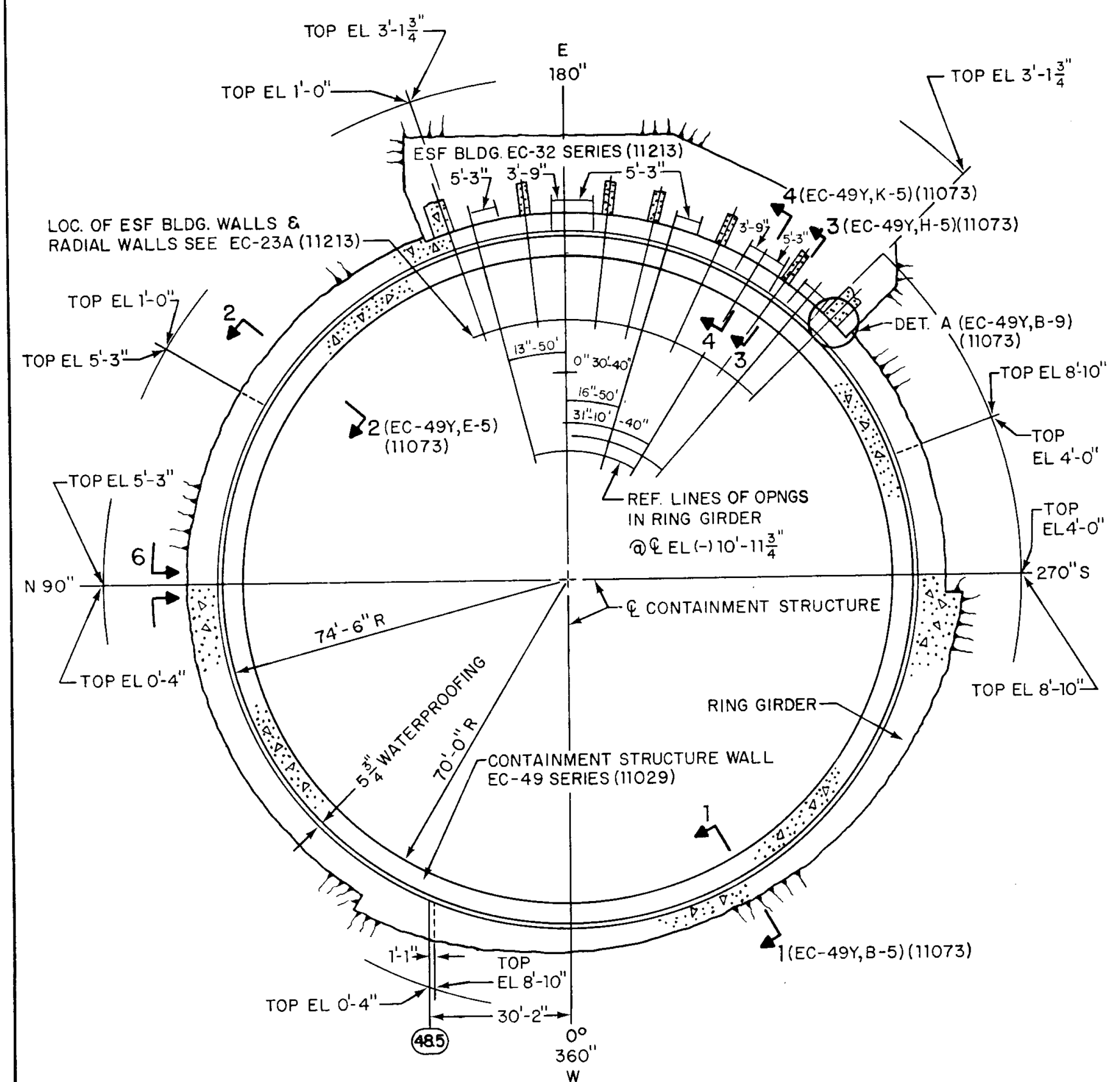


FIGURE 3.8–24 DETAIL OF RING GIRDER AT ESF BUILDING (SHEET 1 OF 3)



PLAN @ EL (-) 26'-0"

SCALE: 1/32 = 1'-0"

FIGURE 3.8-24 (1 OF 3)
DETAIL OF RING GIRDER AT ESF BUILDING
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–24 DETAIL OF RING GIRDER AT ESF BUILDING (SHEET 2 OF 3)

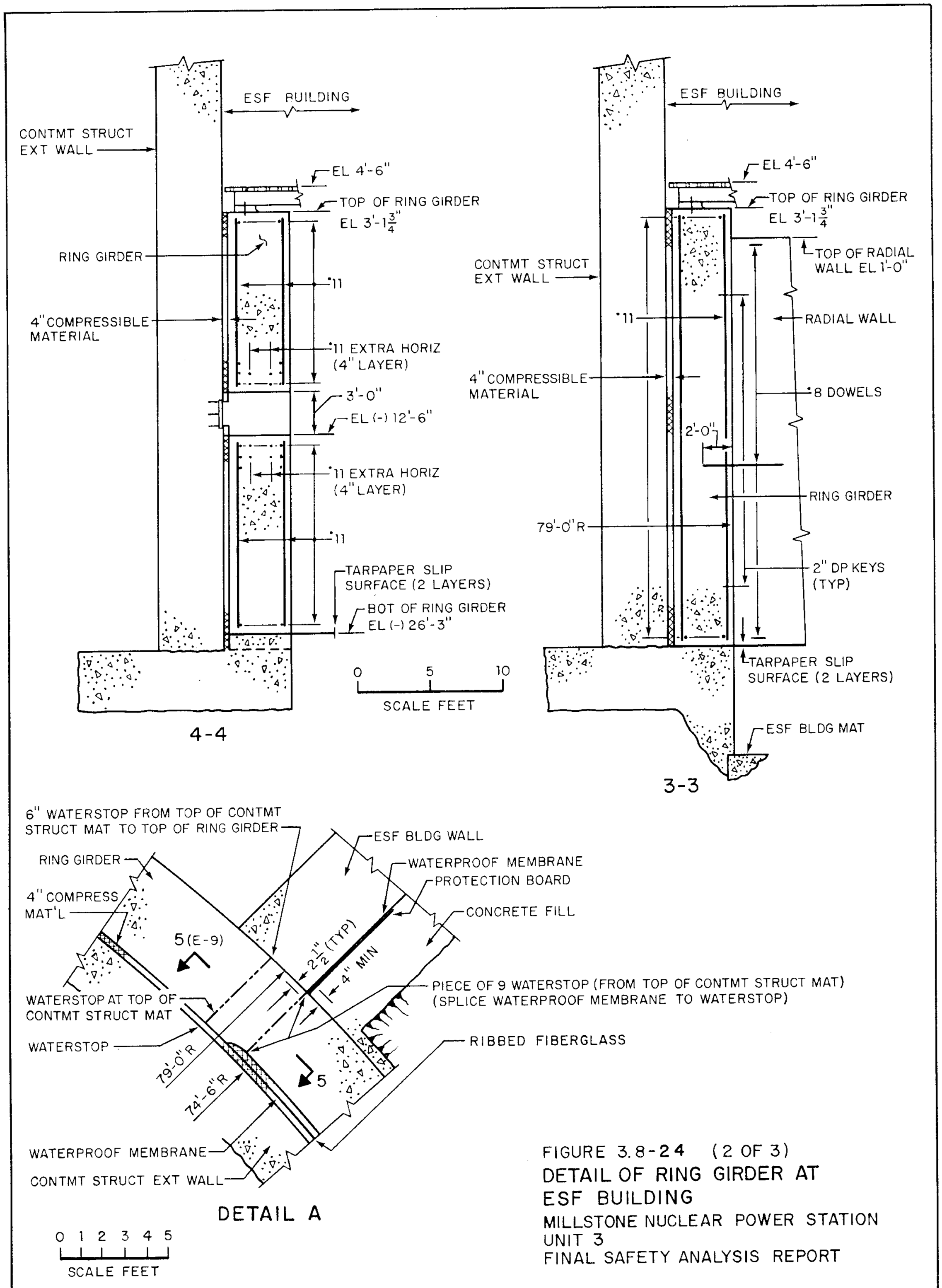


FIGURE 3.8-24 (2 OF 3)
DETAIL OF RING GIRDER AT ESF BUILDING
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

AMENDMENT 15 SEPTEMBER 1985

FIGURE 3.8–24 DETAIL OF RING GIRDER AT ESF BUILDING (SHEET 3 OF 3)

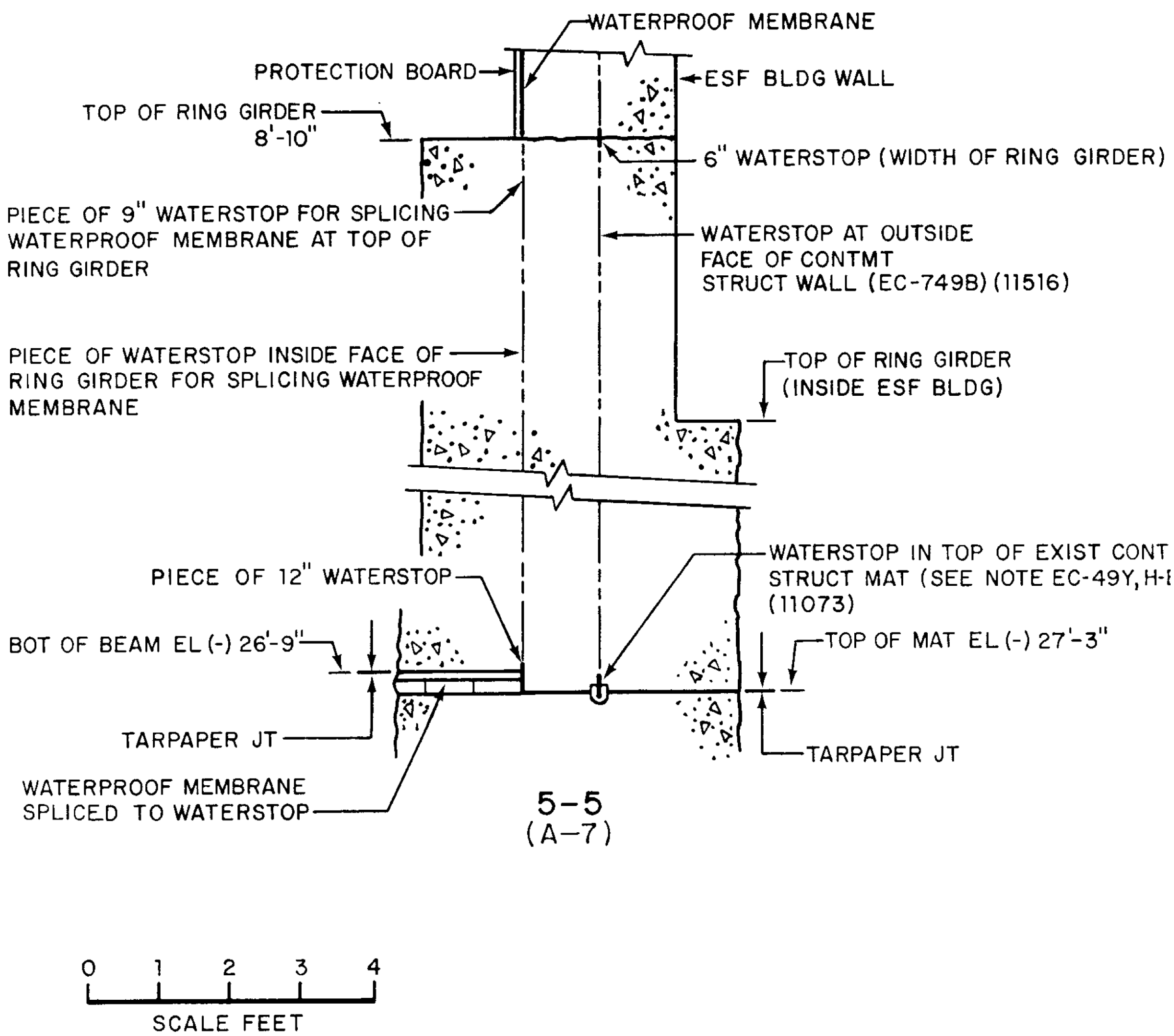


FIGURE 3.8–25 CONTAINMENT STRUCTURE MAT MOMENT AND SHEAR DIAGRAMS

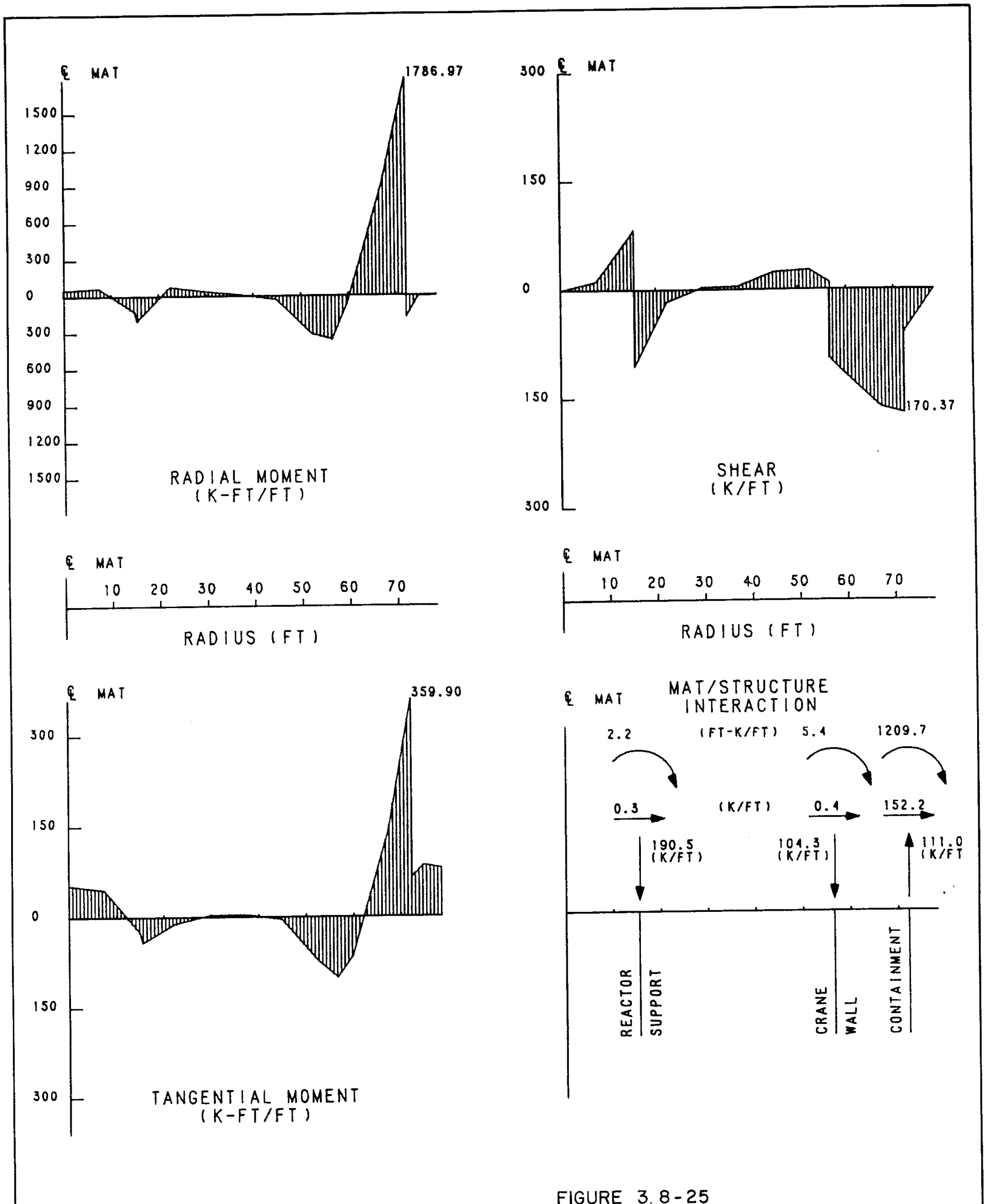


FIGURE 3.8-25
CONTAINMENT STRUCTURE MAT
MOMENT AND SHEAR DIAGRAMS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–26 CONTAINMENT STRUCTURE MAT MOMENT AND SHEAR DIAGRAMS

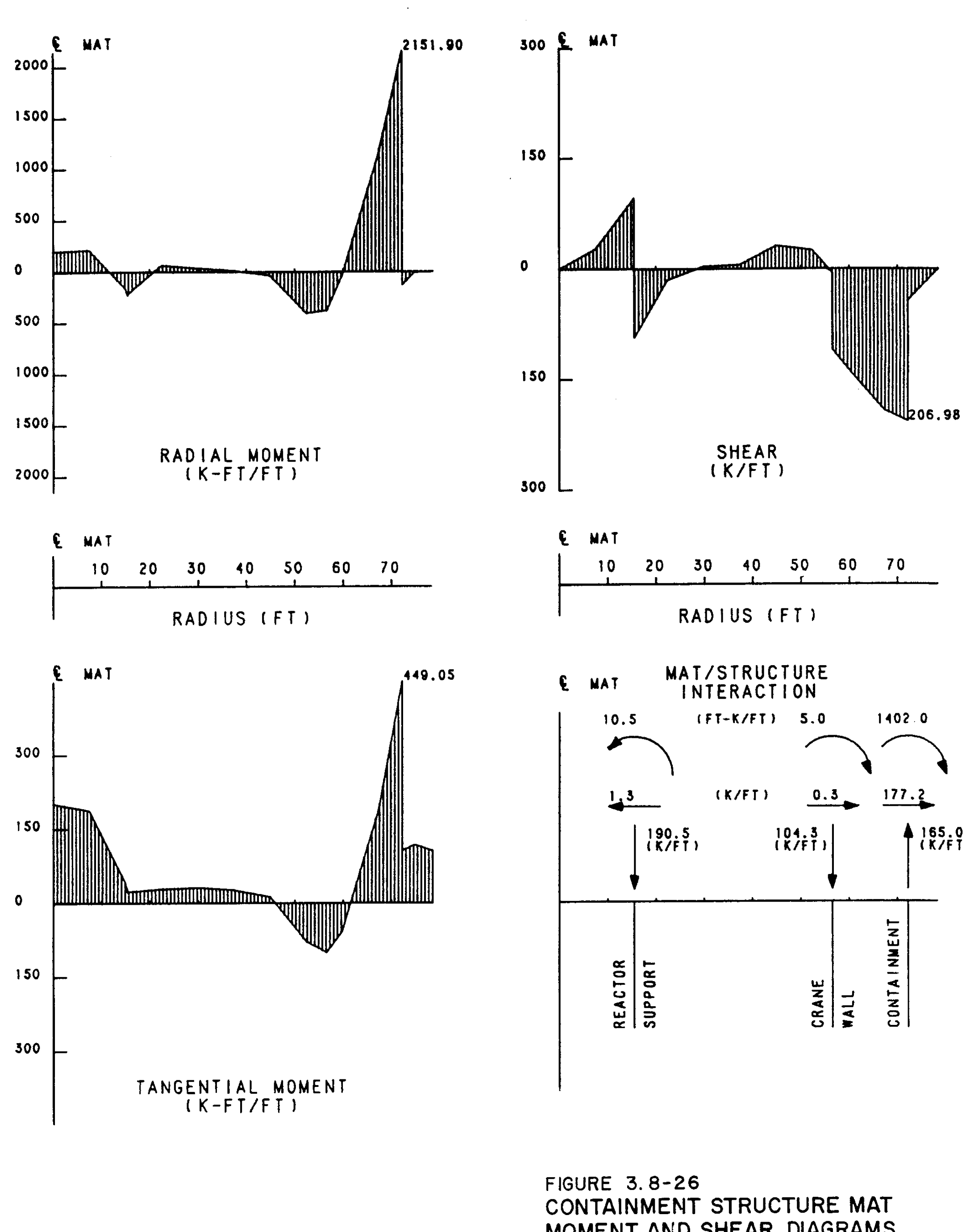
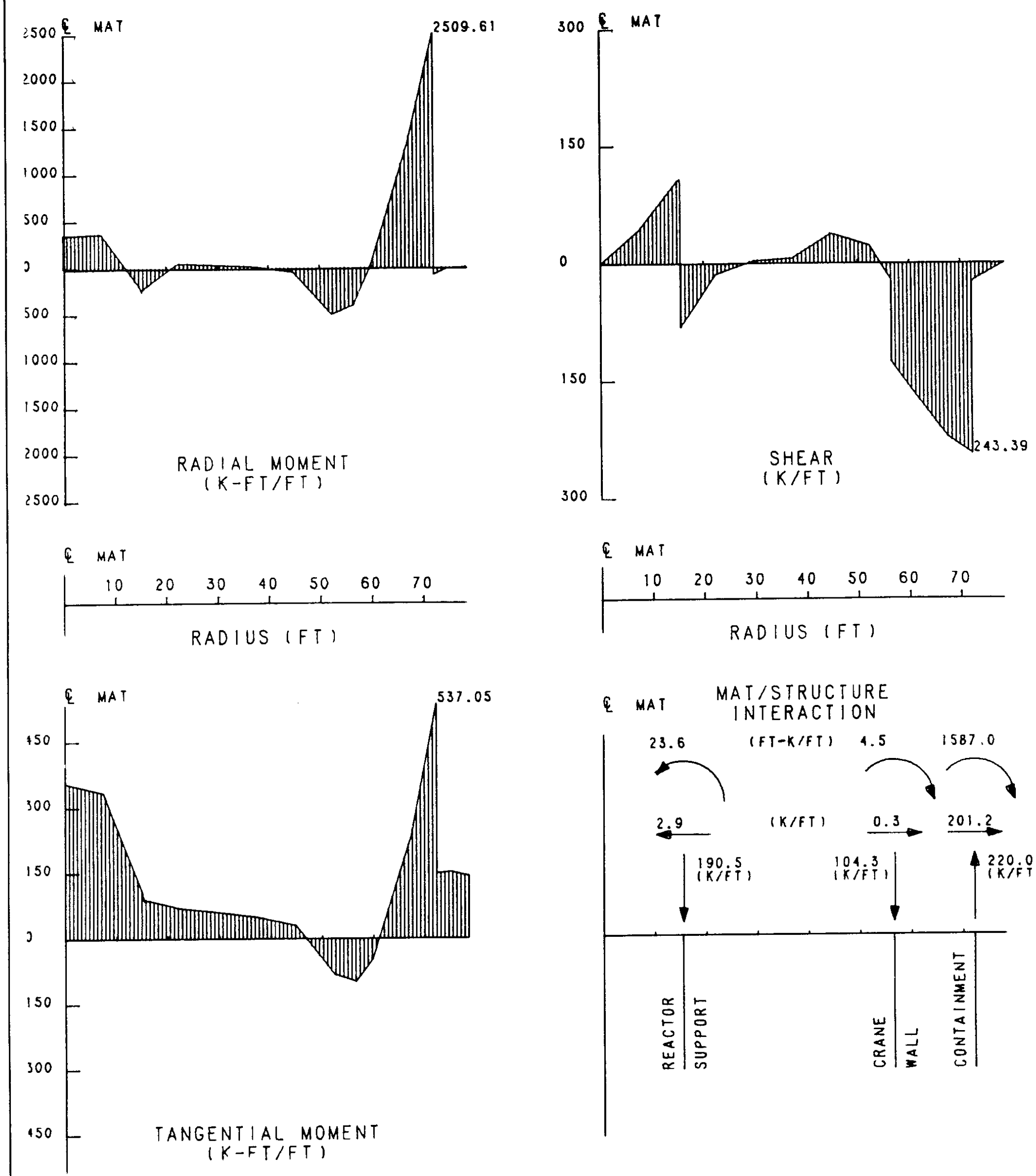


FIGURE 3.8-26
CONTAINMENT STRUCTURE MAT
MOMENT AND SHEAR DIAGRAMS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–27 CONTAINMENT STRUCTURE MAT MOMENT AND SHEAR DIAGRAMS



LOADING

DL+1.5P+1.0(T+TL)

FIGURE 3.8-27
CONTAINMENT STRUCTURE MAT
MOMENT AND SHEAR DIAGRAMS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–28 CONTAINMENT STRUCTURE MAT MOMENT AND SHEAR DIAGRAMS

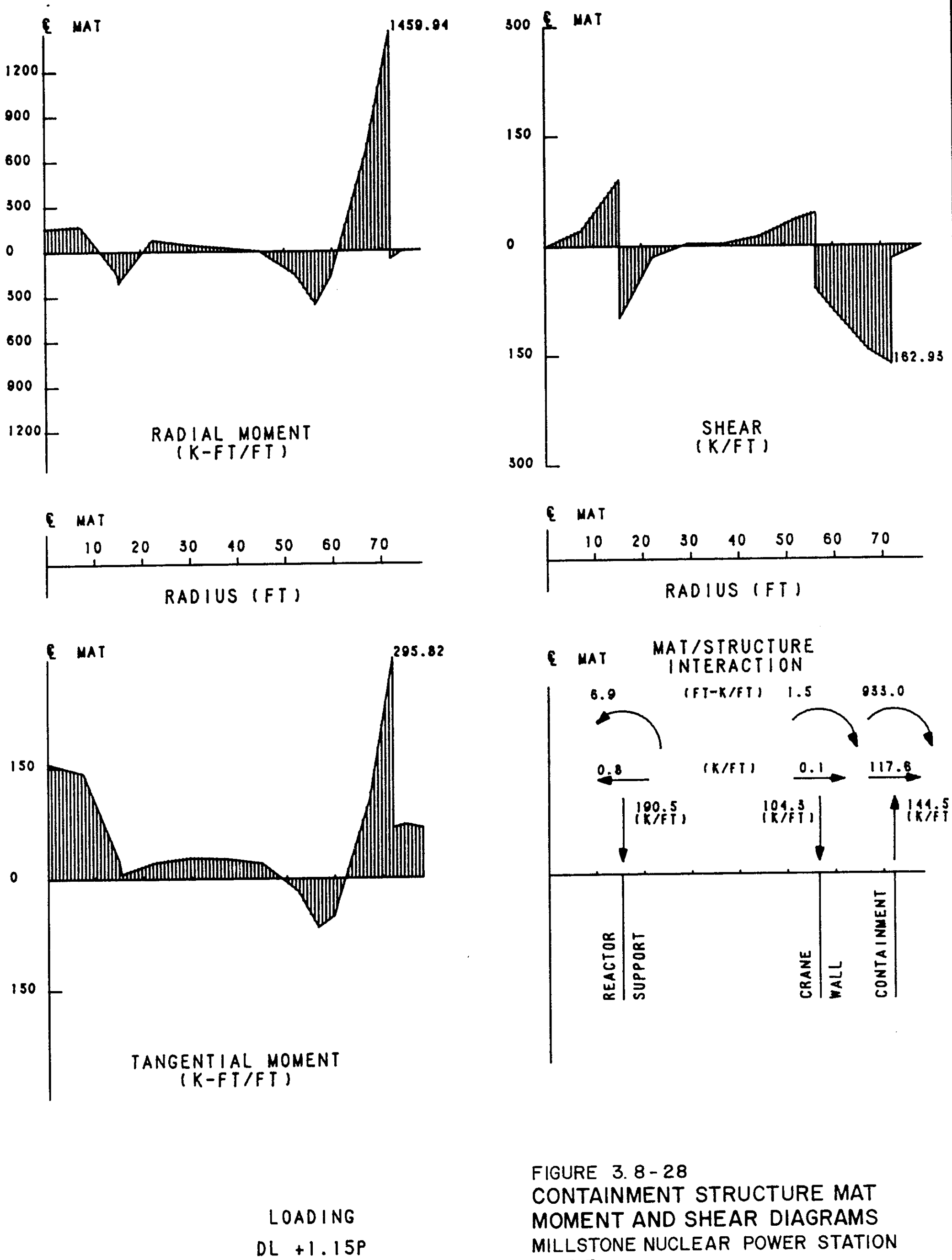


FIGURE 3.8-28
CONTAINMENT STRUCTURE MAT
MOMENT AND SHEAR DIAGRAMS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–29 CONTAINMENT STRUCTURE MAT MOMENT AND SHEAR DIAGRAMS

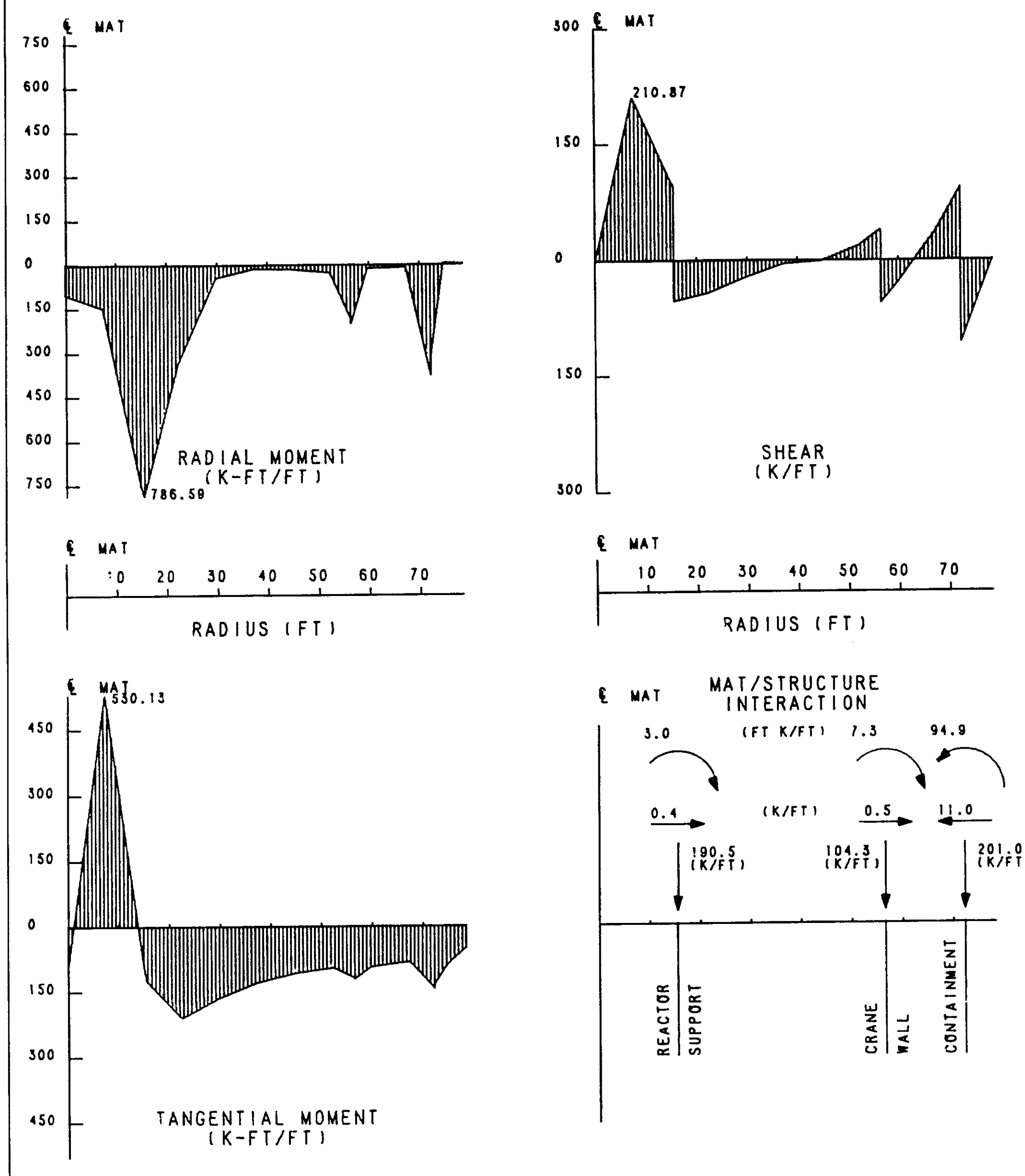


FIGURE 3.8-29
CONTAINMENT STRUCTURE MAT MOMENT AND SHEAR DIAGRAMS
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–30 CONTAINMENT STRUCTURE MAT MOMENT AND SHEAR DIAGRAMS

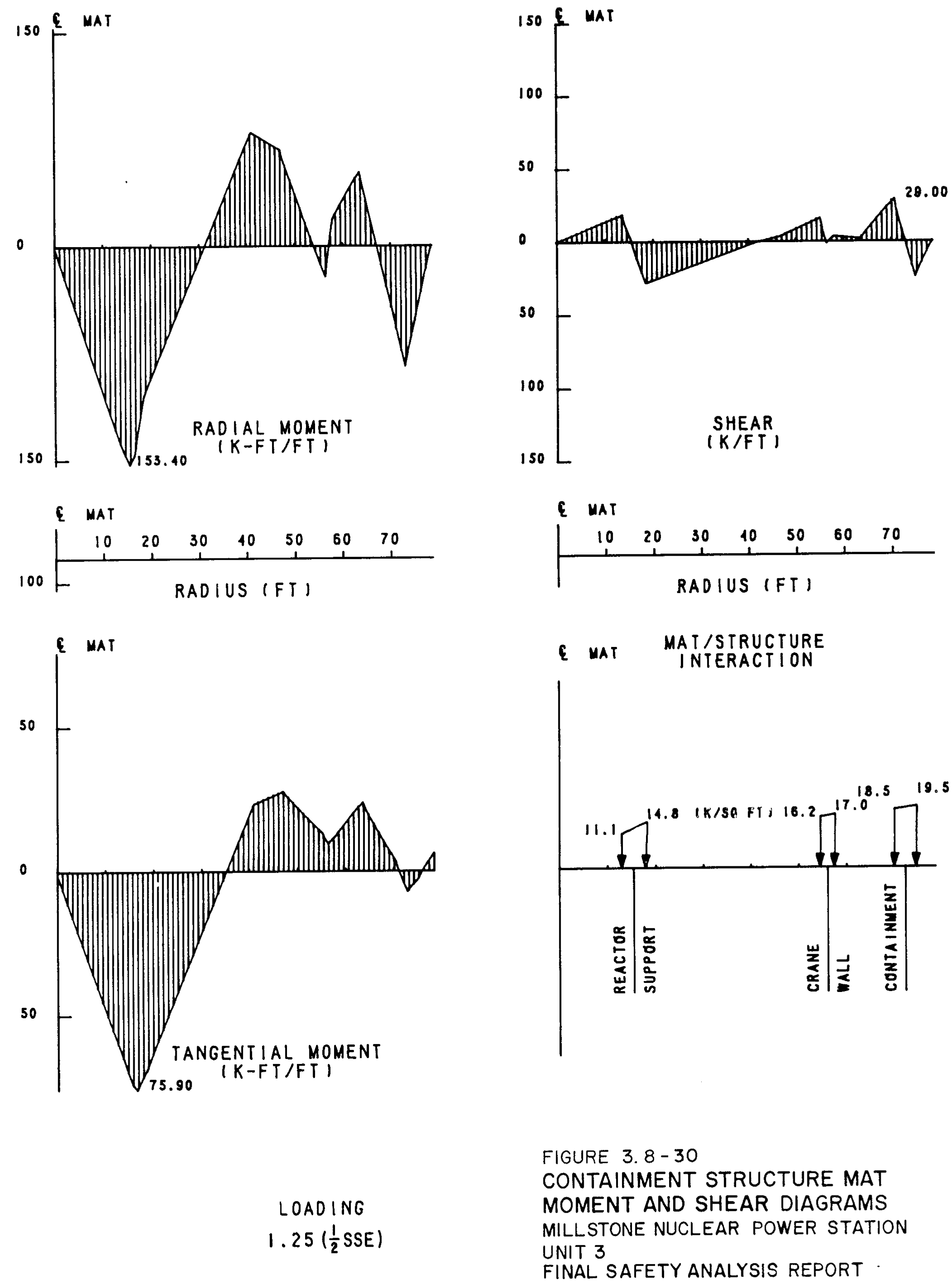


FIGURE 3.8-30
CONTAINMENT STRUCTURE MAT
MOMENT AND SHEAR DIAGRAMS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–31 CONTAINMENT STRUCTURE MAT MOMENT AND SHEAR DIAGRAMS

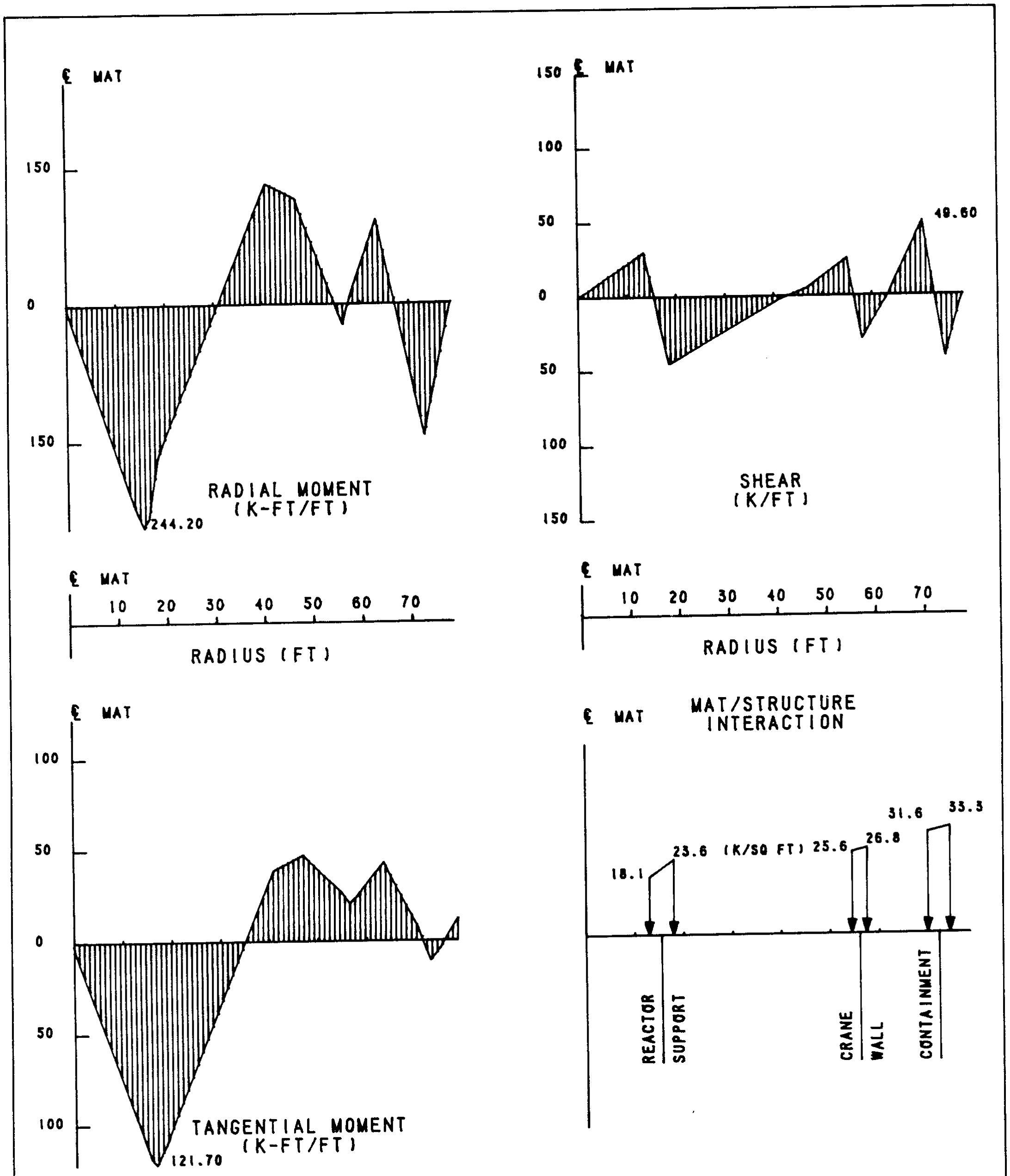


FIGURE 3.8-31
CONTAINMENT STRUCTURE MAT
MOMENT AND SHEAR DIAGRAMS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–32 LOAD PLOT NOMENCLATURE

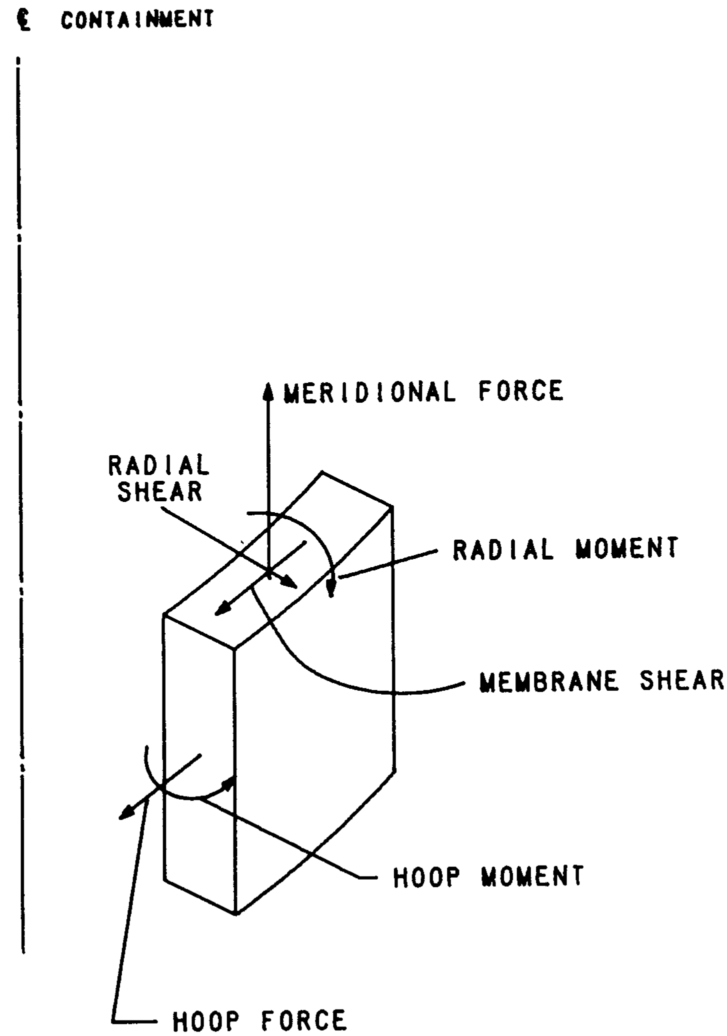


FIGURE 3.8–33 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

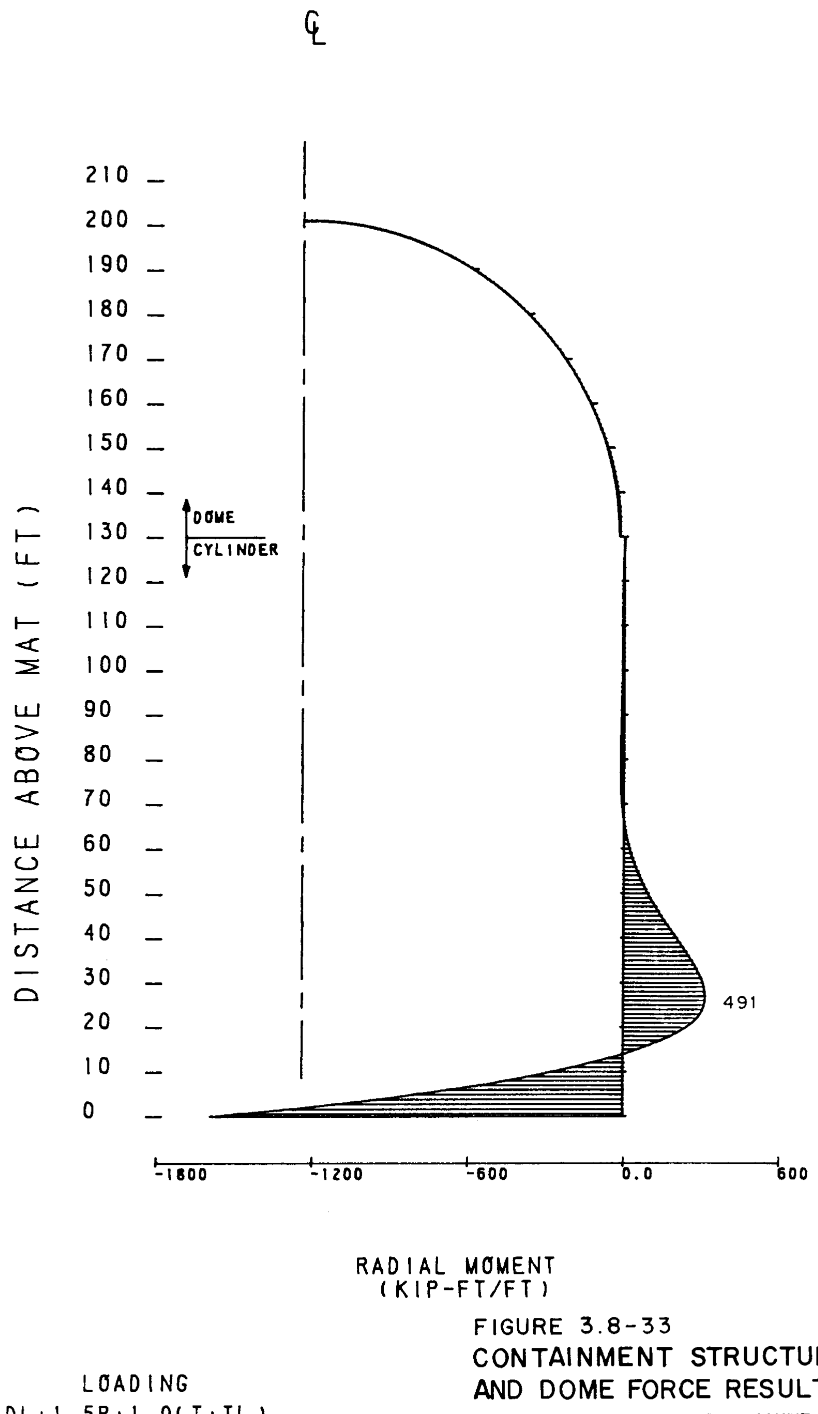
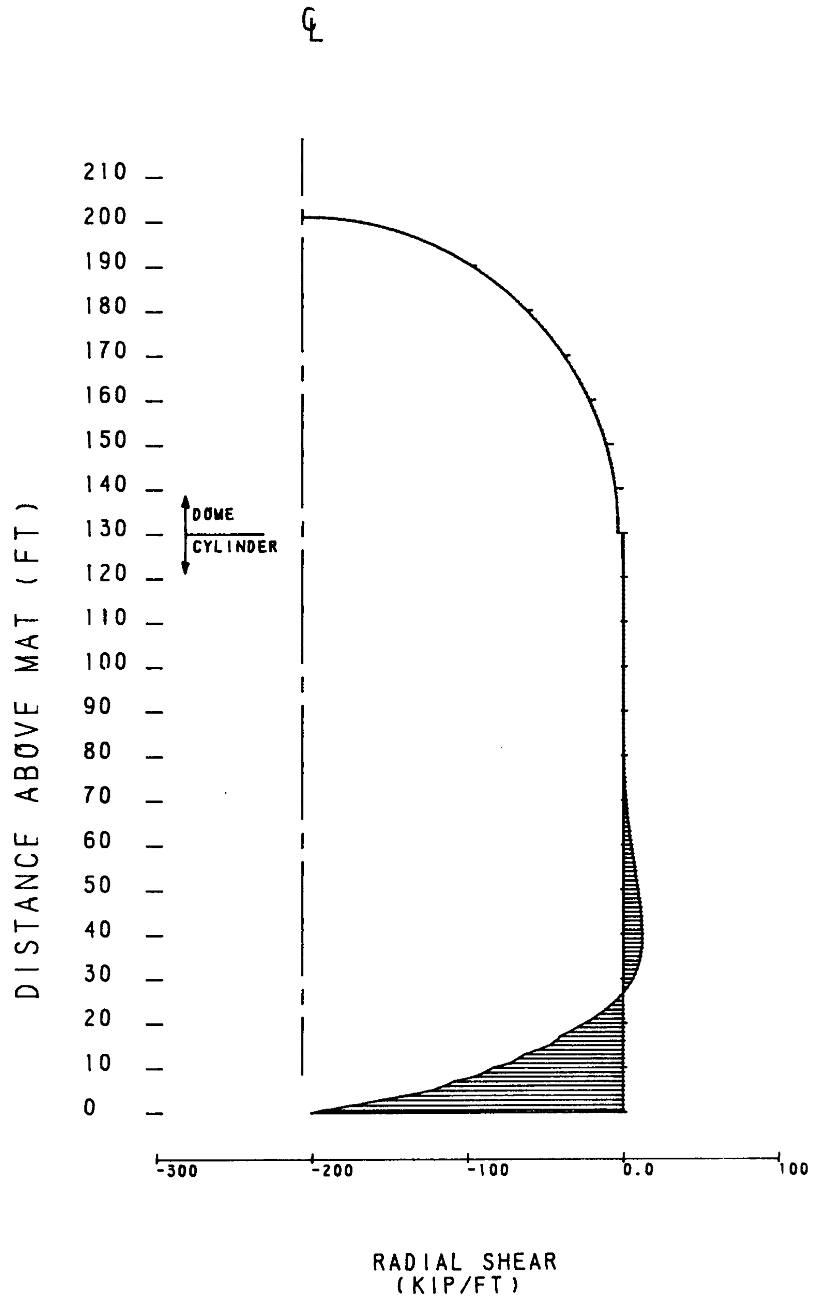


FIGURE 3.8-33
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–34 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS



LOADING
DL+1.5P+1.0(T+TL)

FIGURE 3.8-34
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–35 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

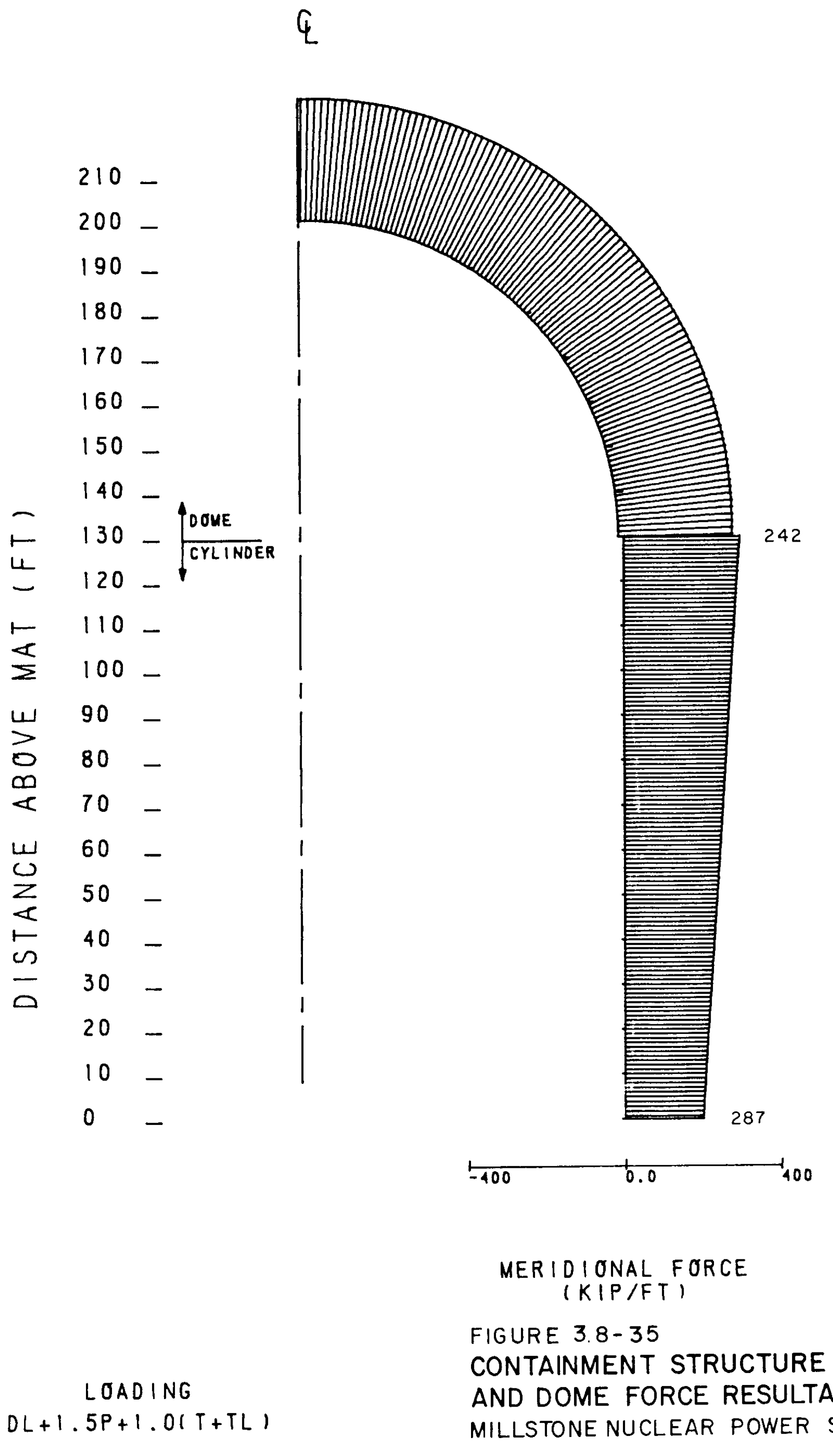


FIGURE 3.8-35
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–36 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

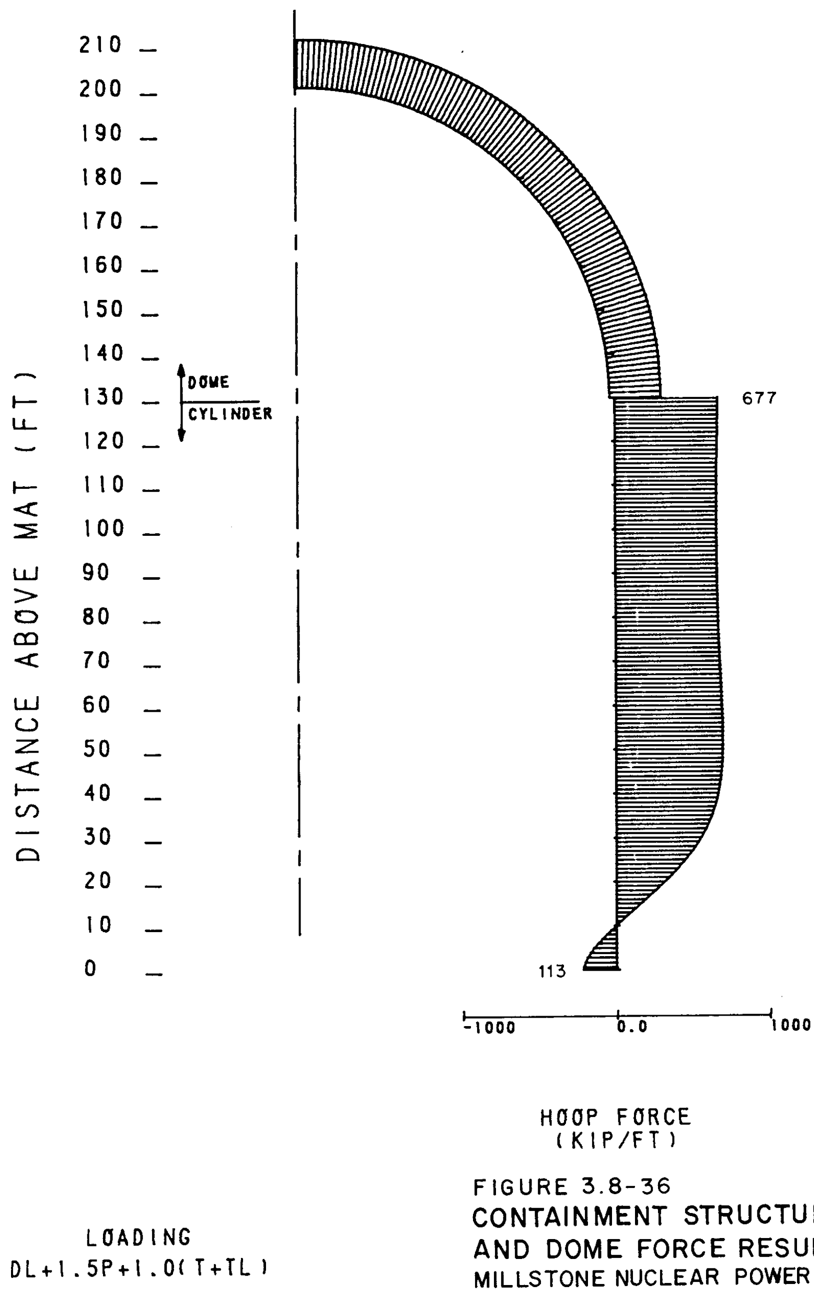
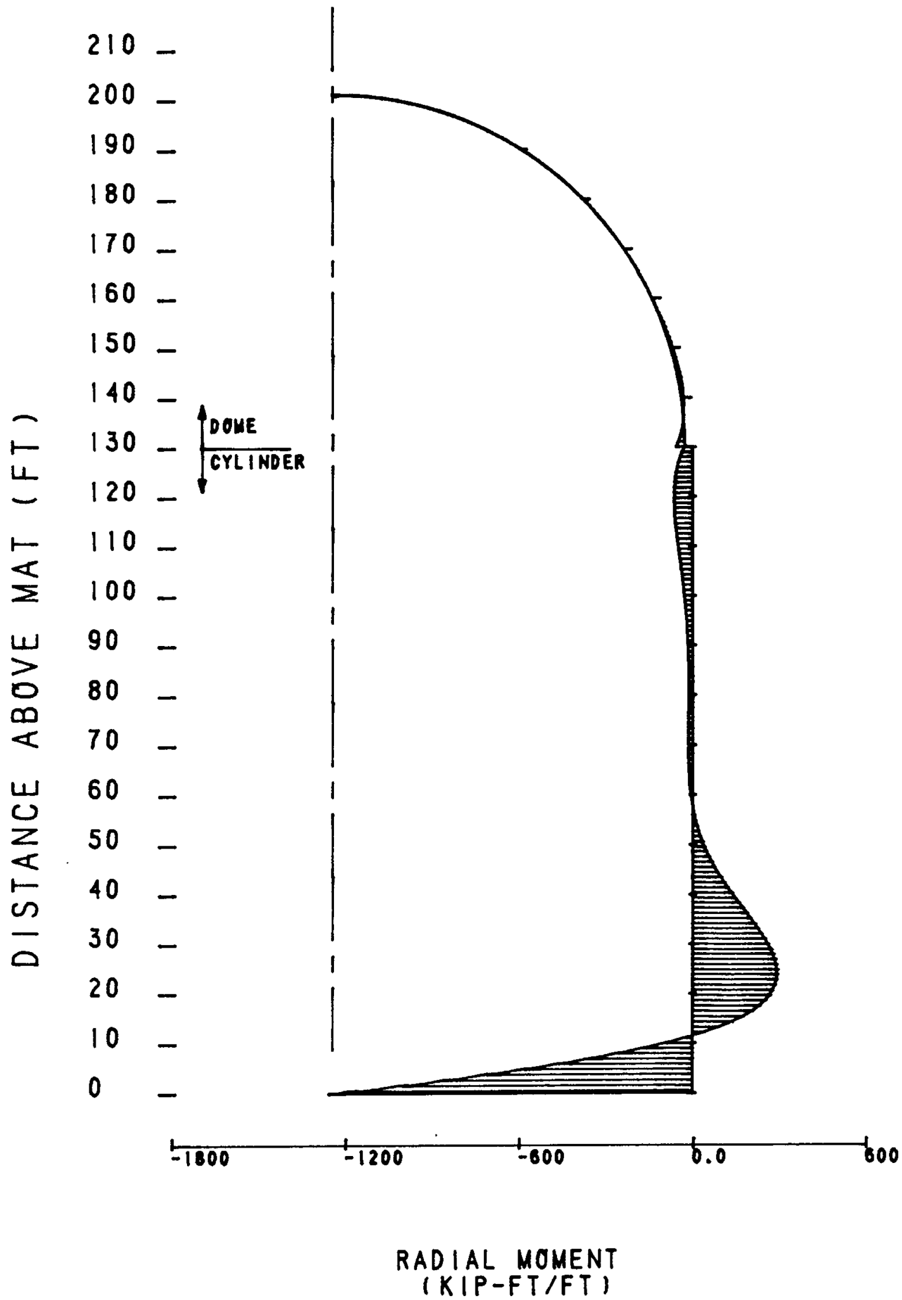


FIGURE 3.8-36
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

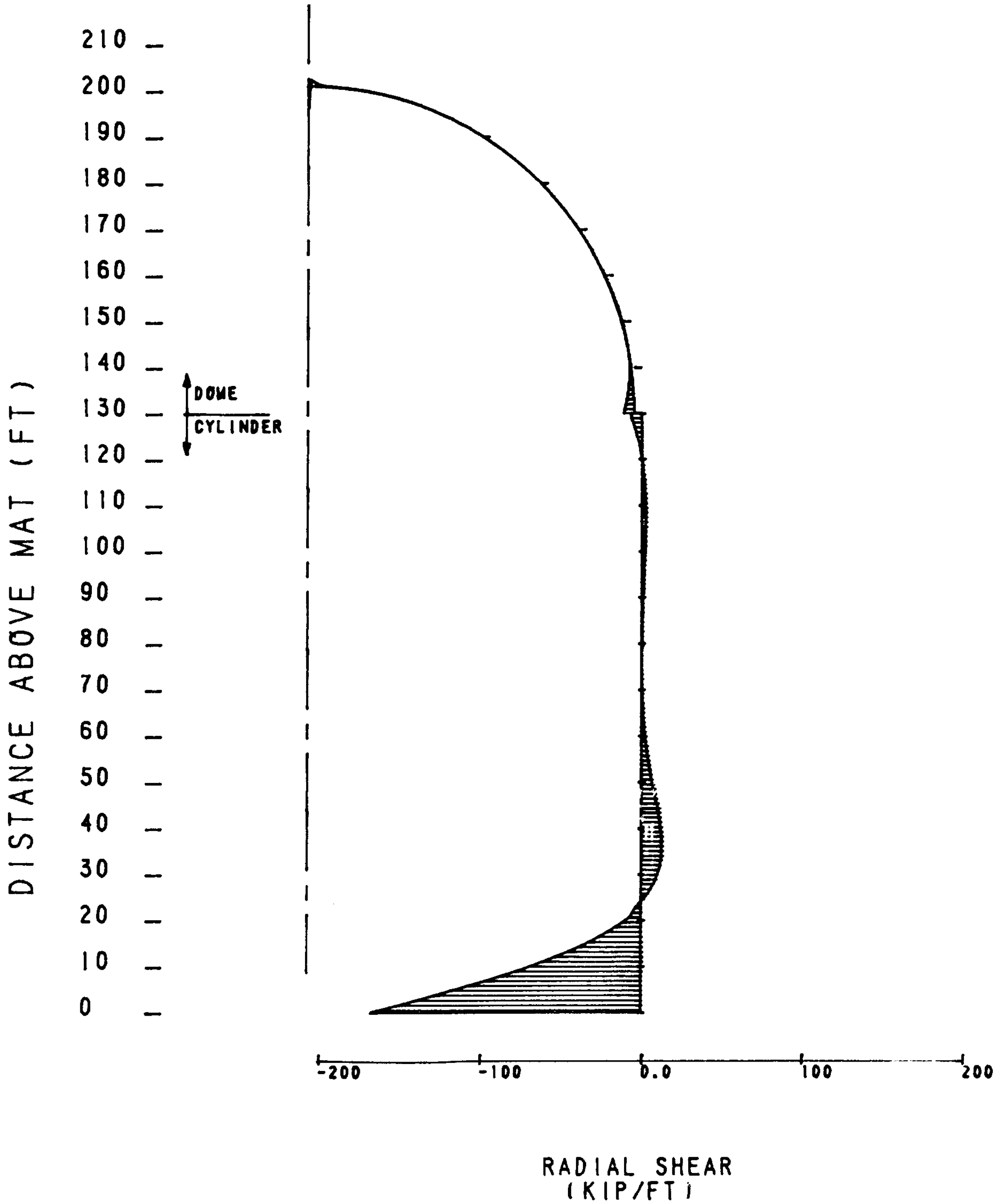
FIGURE 3.8–37 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS



LOADING
DL+1.25P+1.0(T+TL)+1.25($\frac{1}{2}$SSE)

FIGURE 3.8-37
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–38 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS



LOADING
DL+1.25P+1.0(T+TL)+1.25($\frac{1}{2}$SSE)

FIGURE 3.8-38
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–39 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

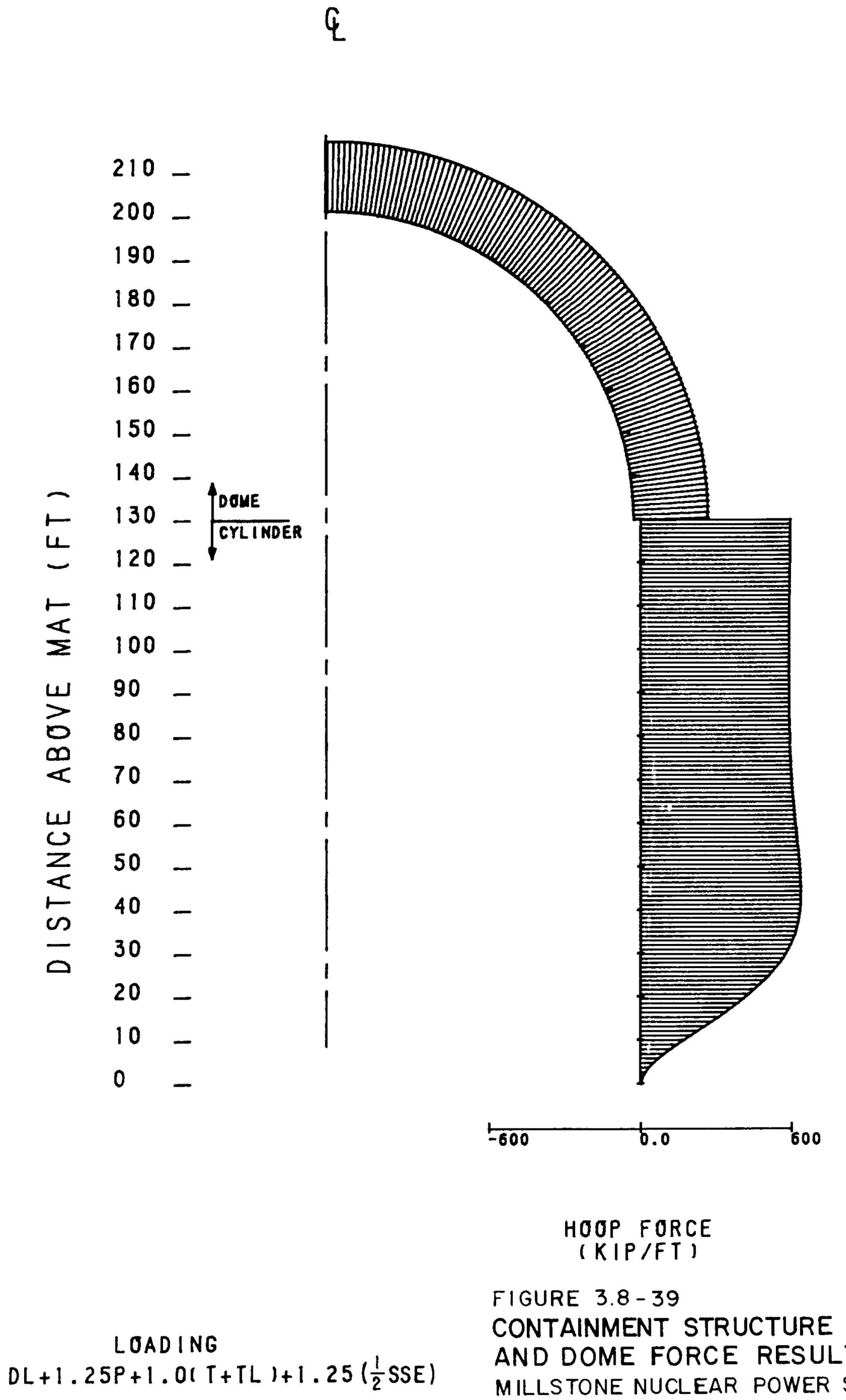


FIGURE 3.8-39
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–40 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

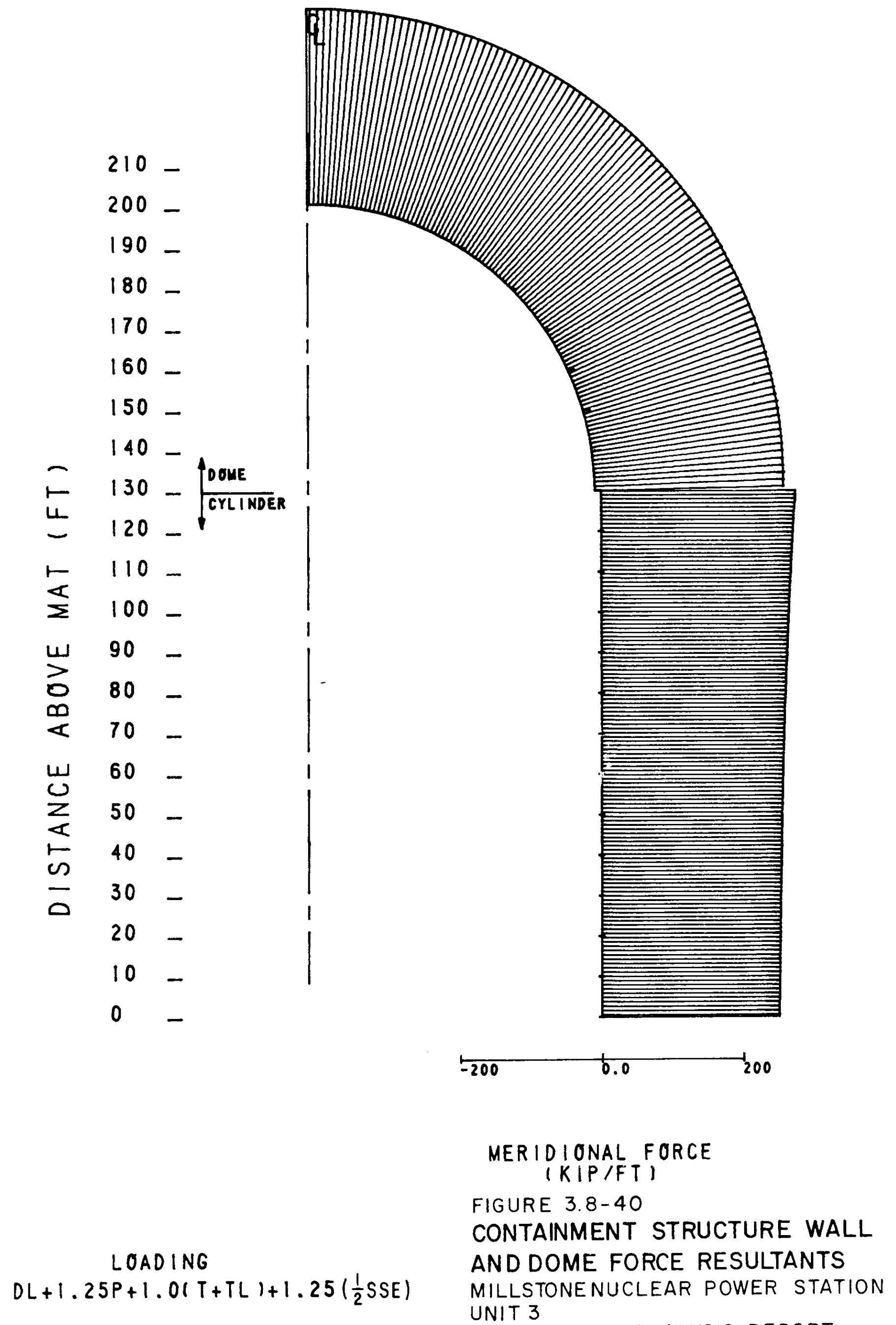


FIGURE 3.8-40
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–41 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

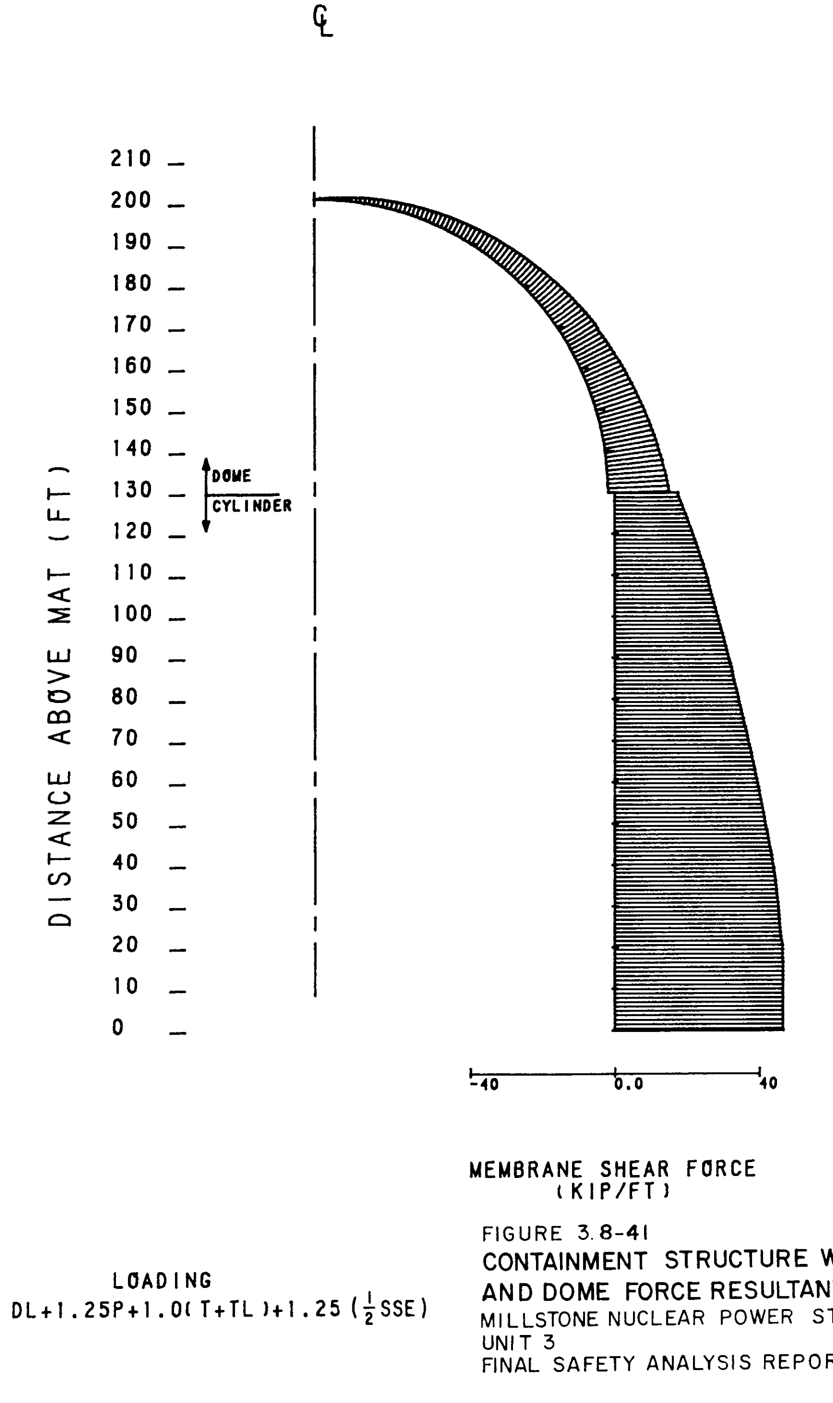


FIGURE 3.8-41
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–42 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

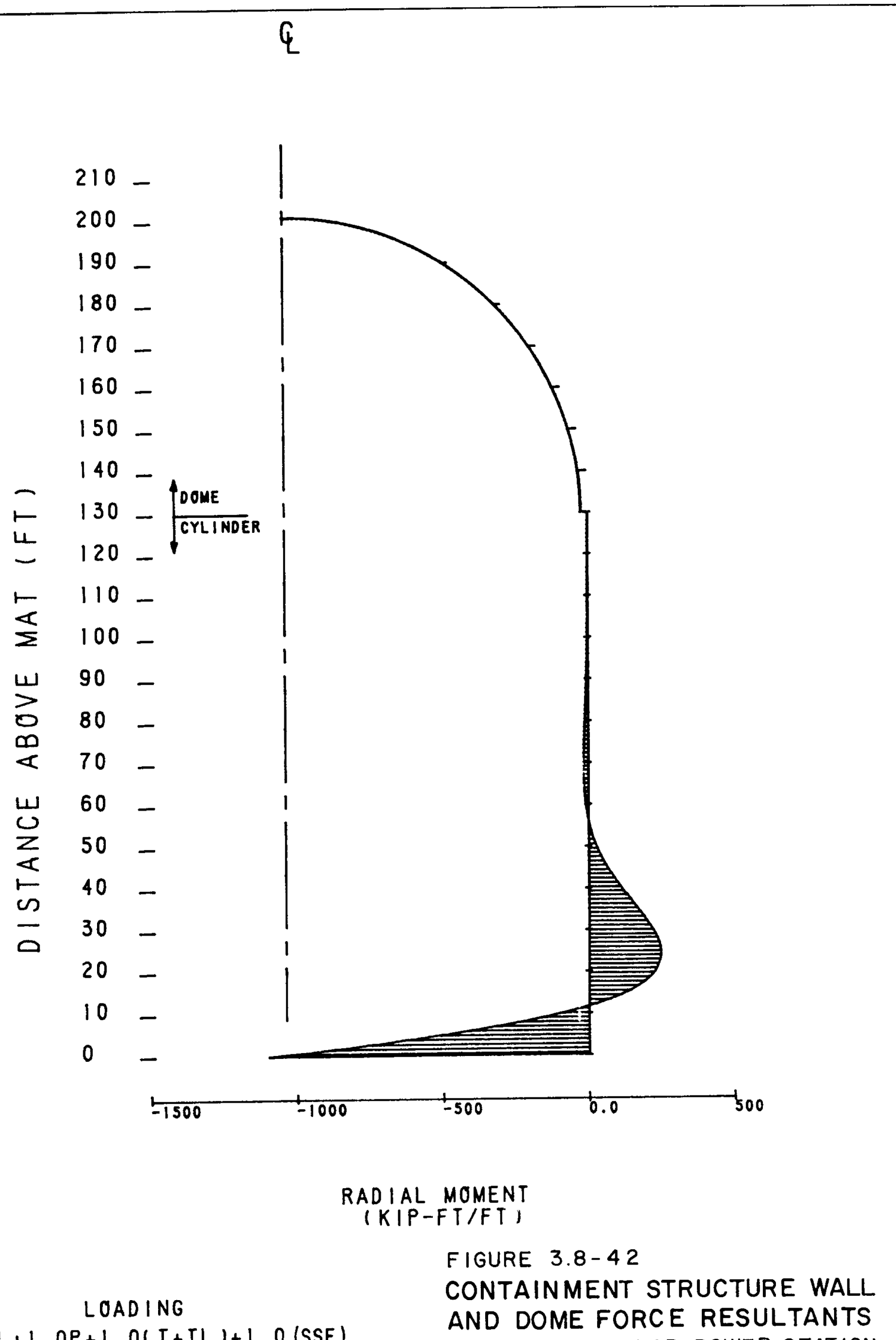


FIGURE 3.8-42
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–43 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

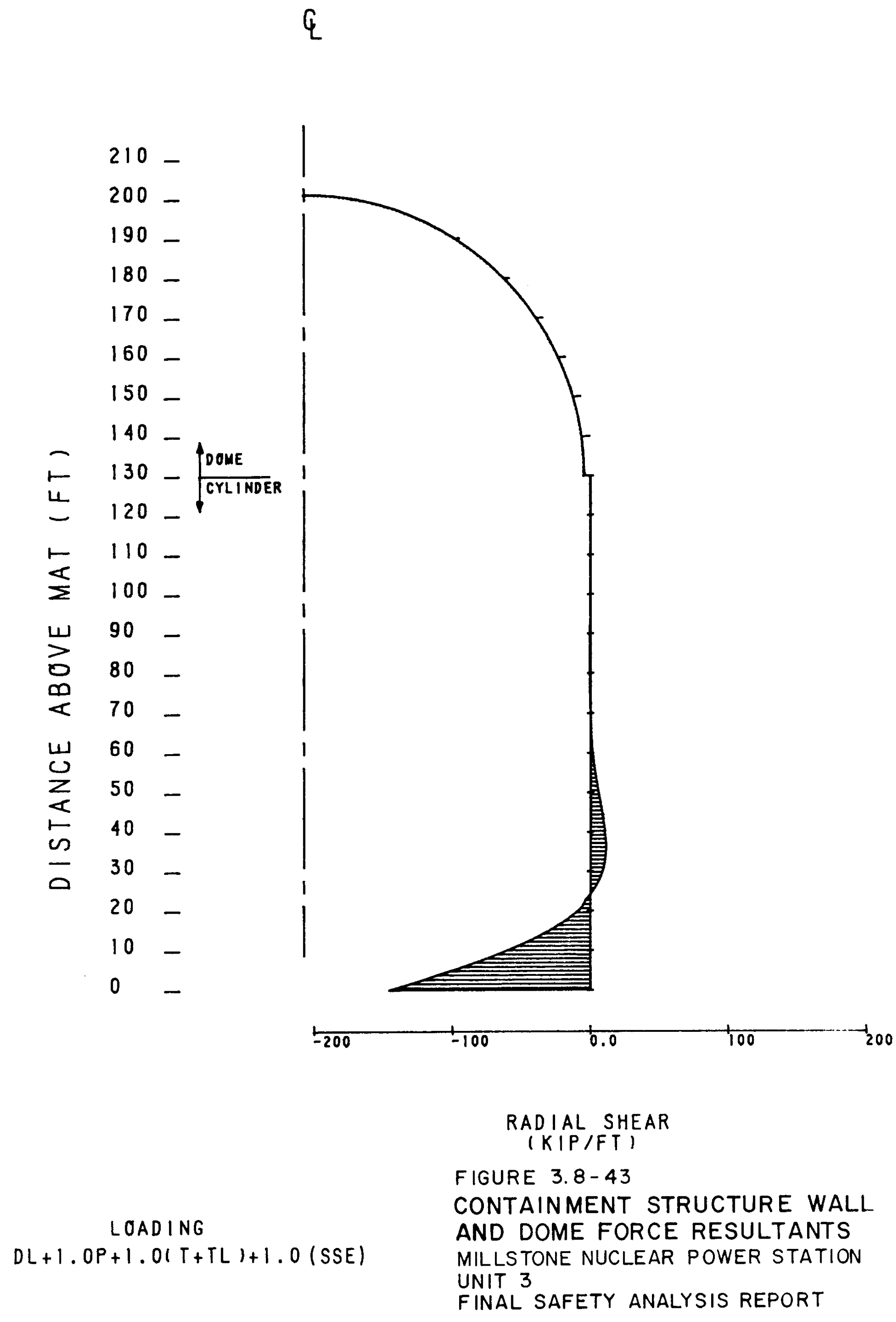


FIGURE 3.8-43
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–44 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

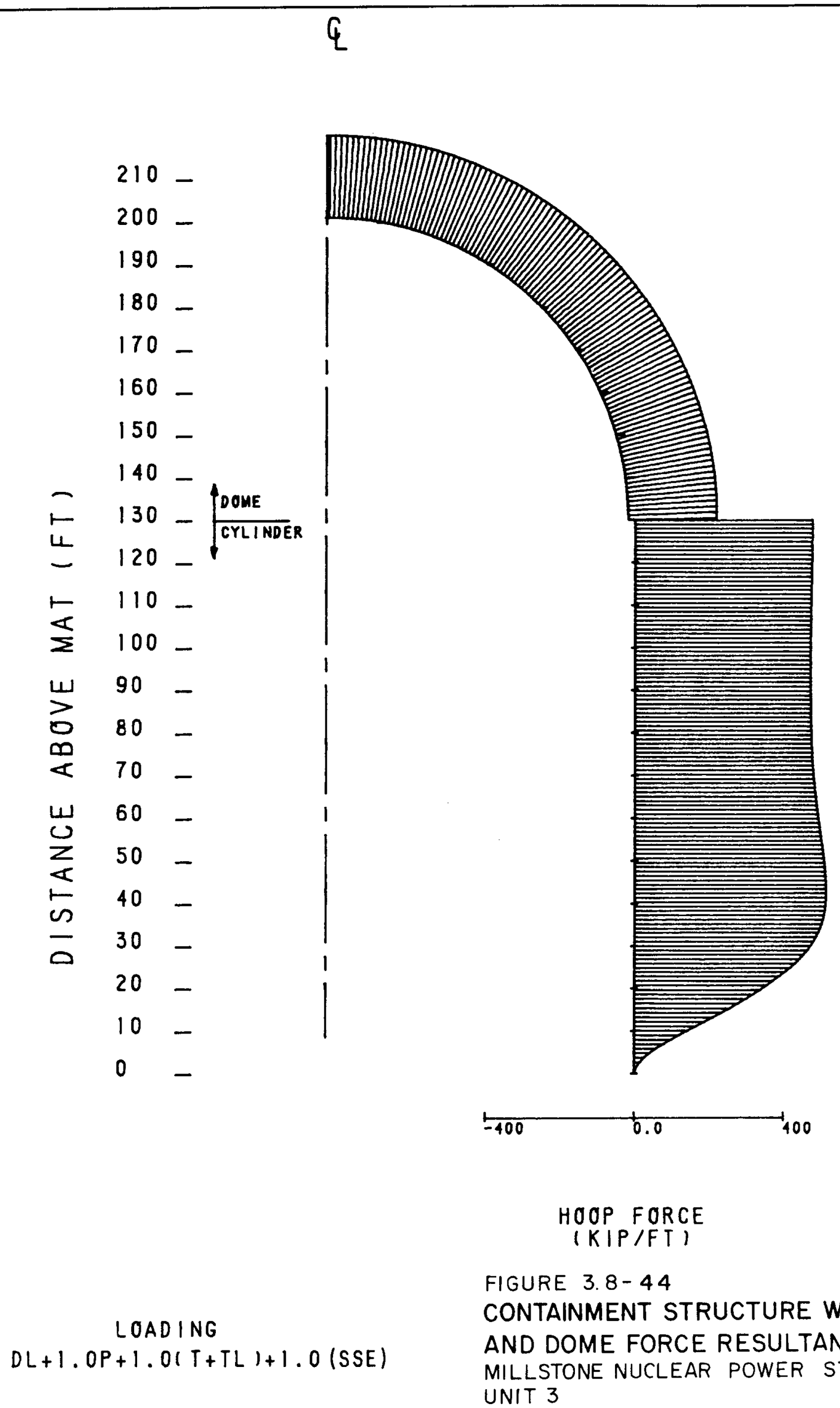


FIGURE 3.8-44
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–45 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

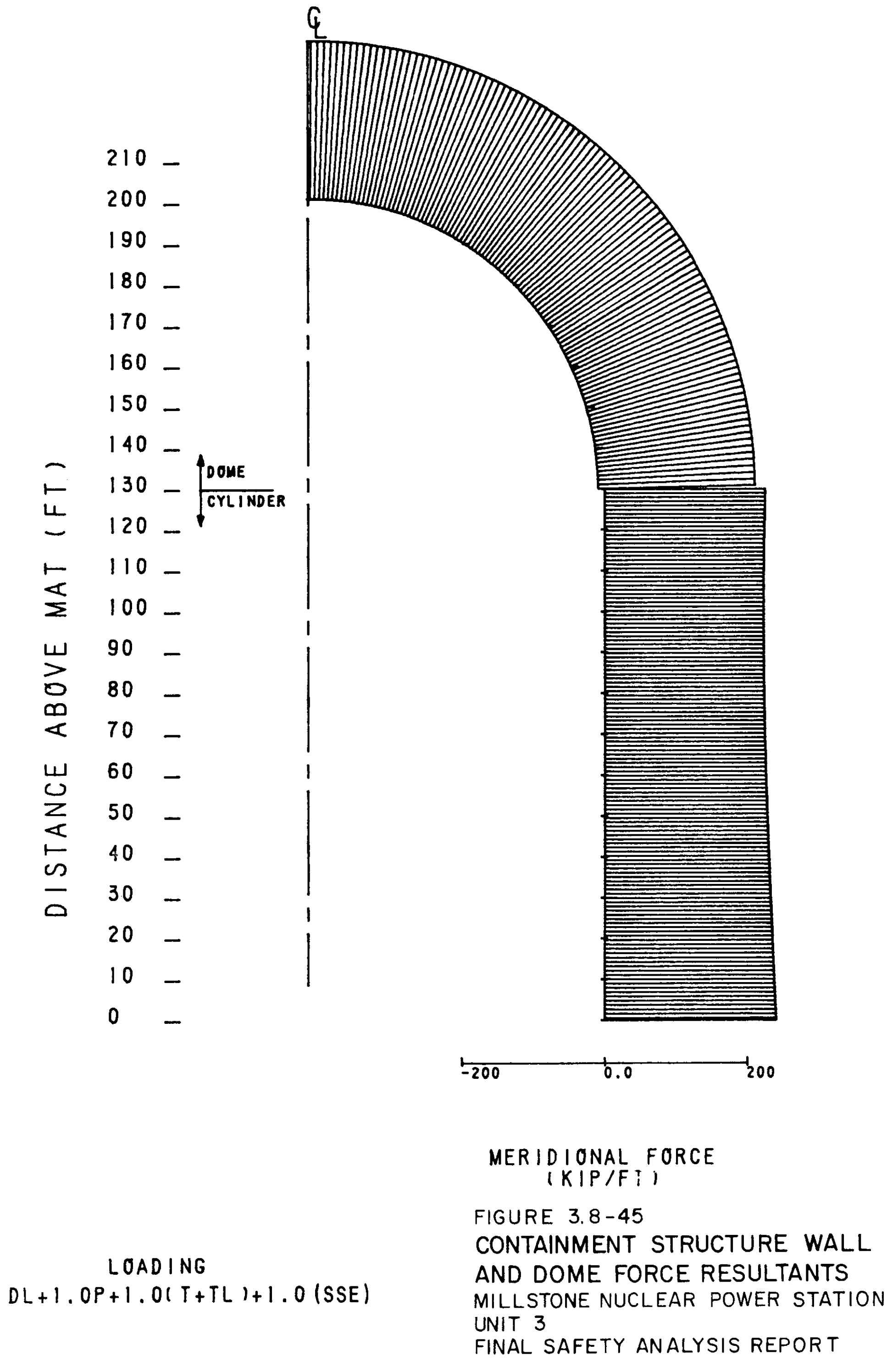


FIGURE 3.8-45
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–46 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

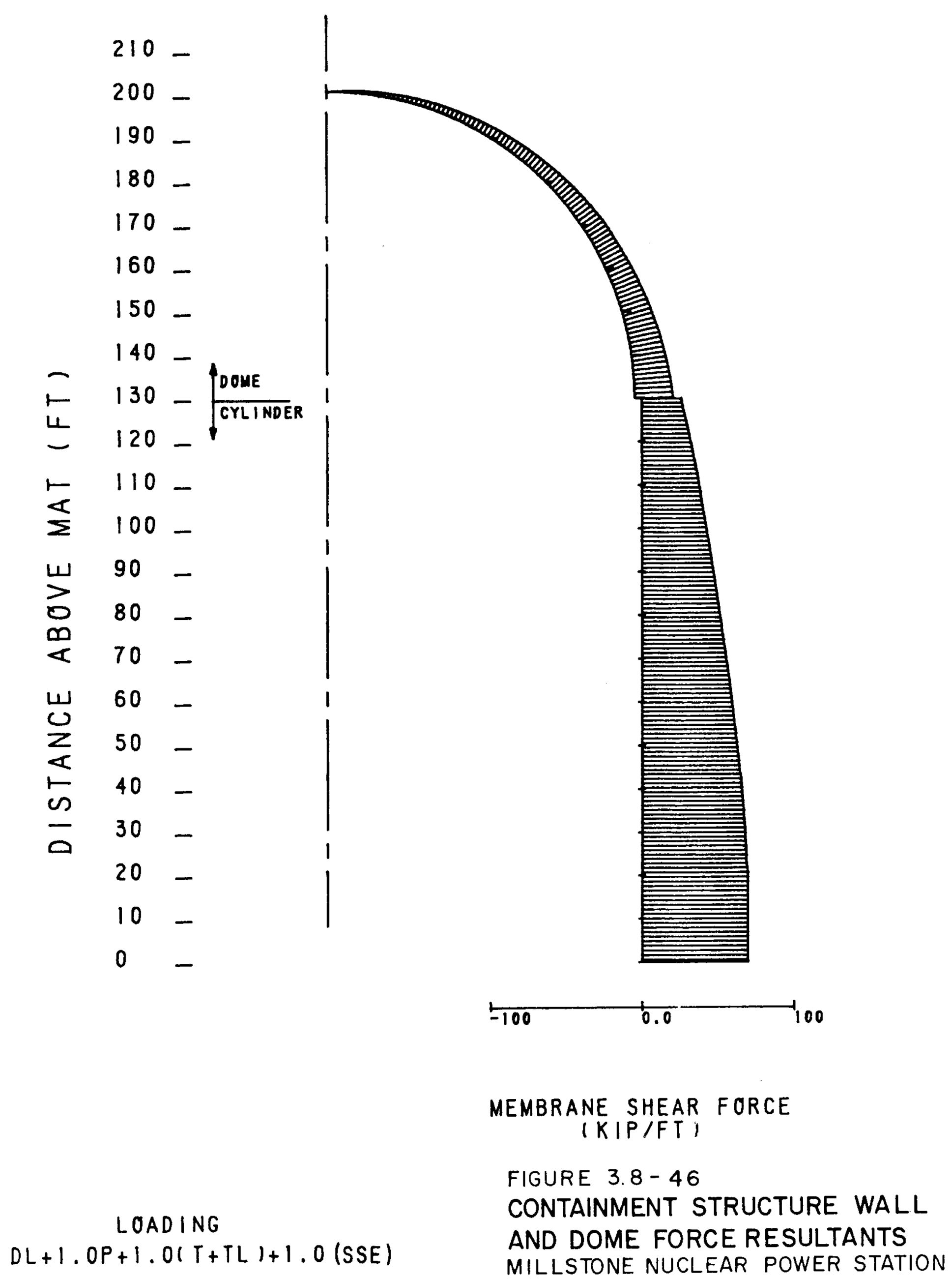


FIGURE 3.8-46
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–47 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

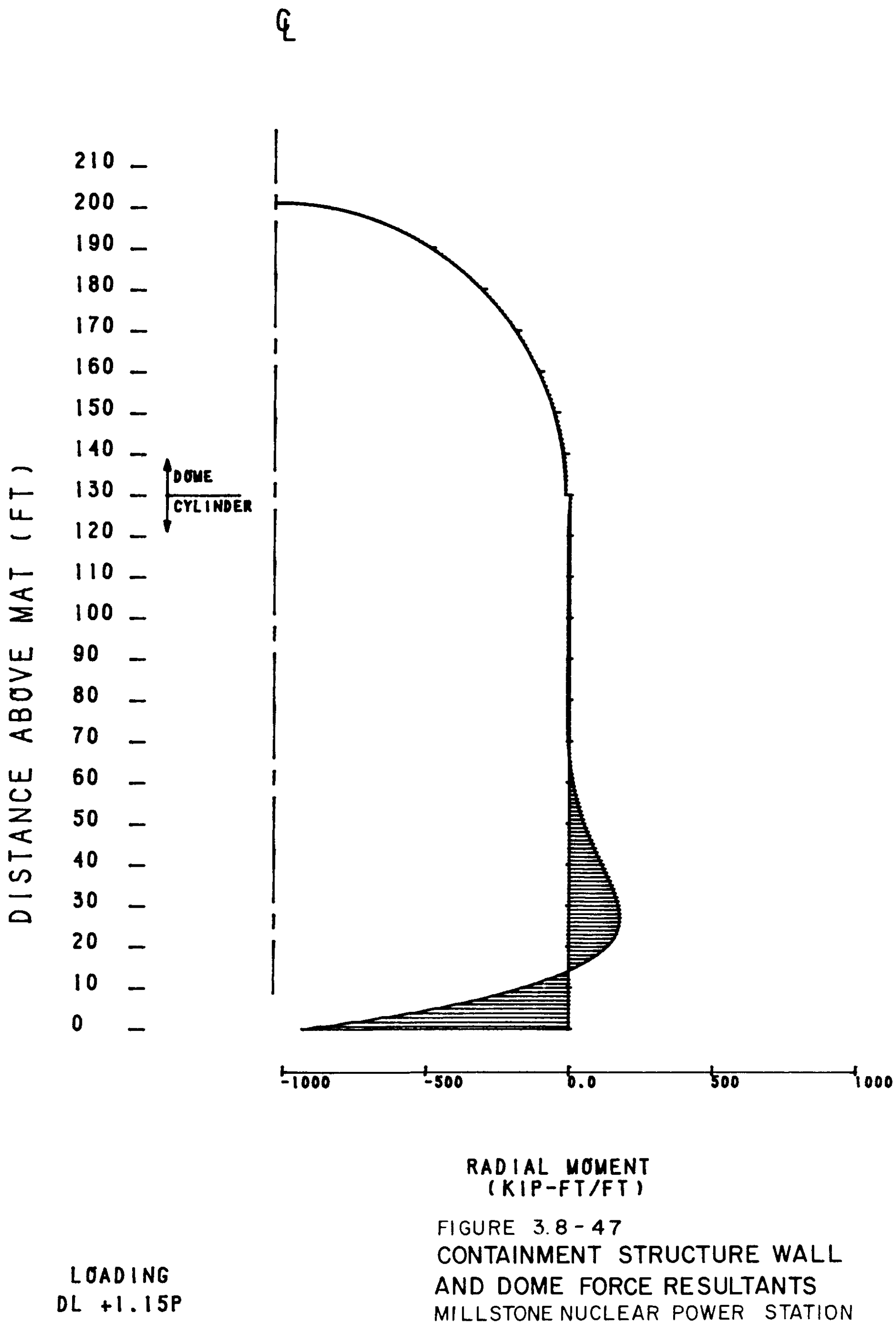


FIGURE 3.8-47
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–48 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

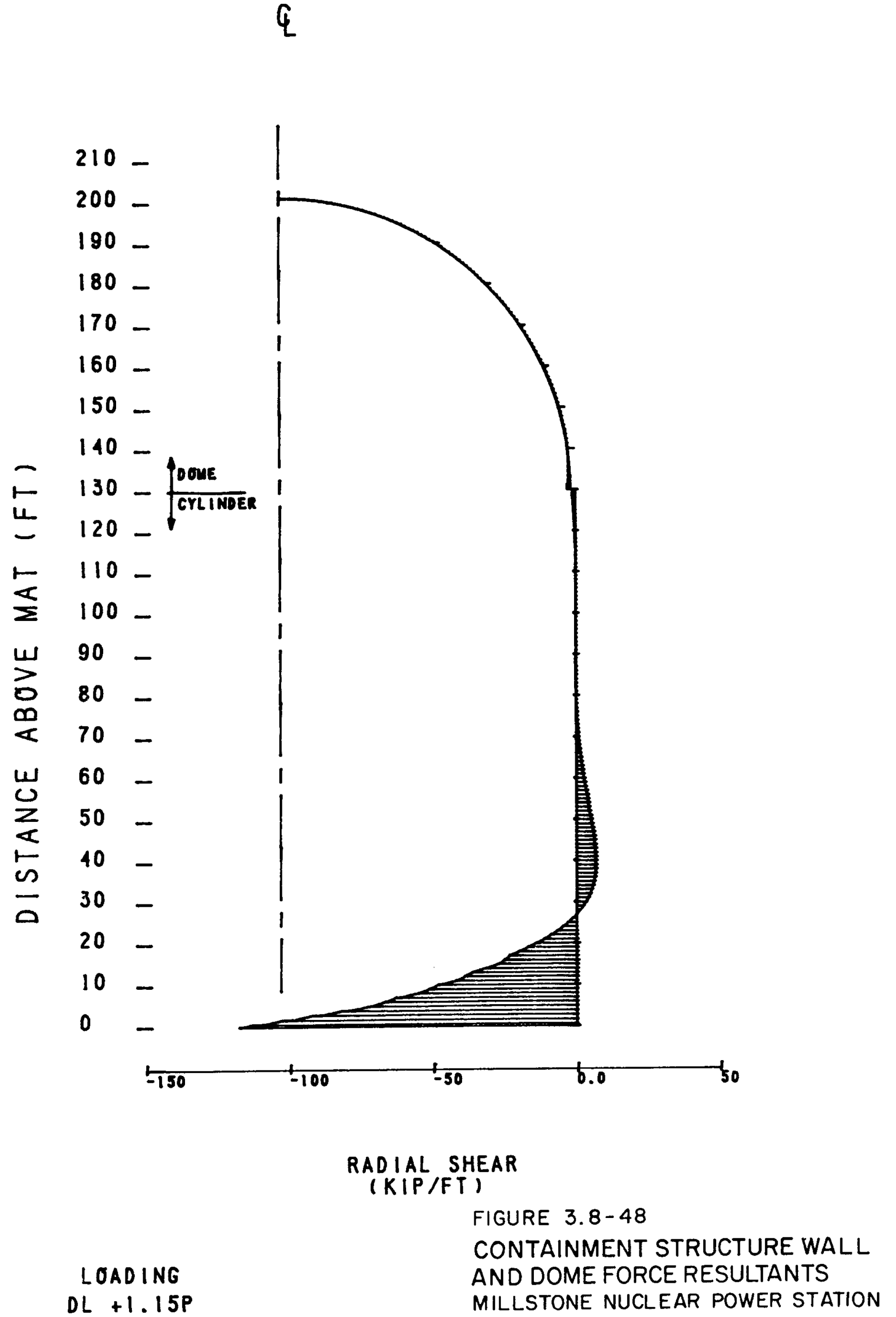


FIGURE 3.8-48
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–49 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

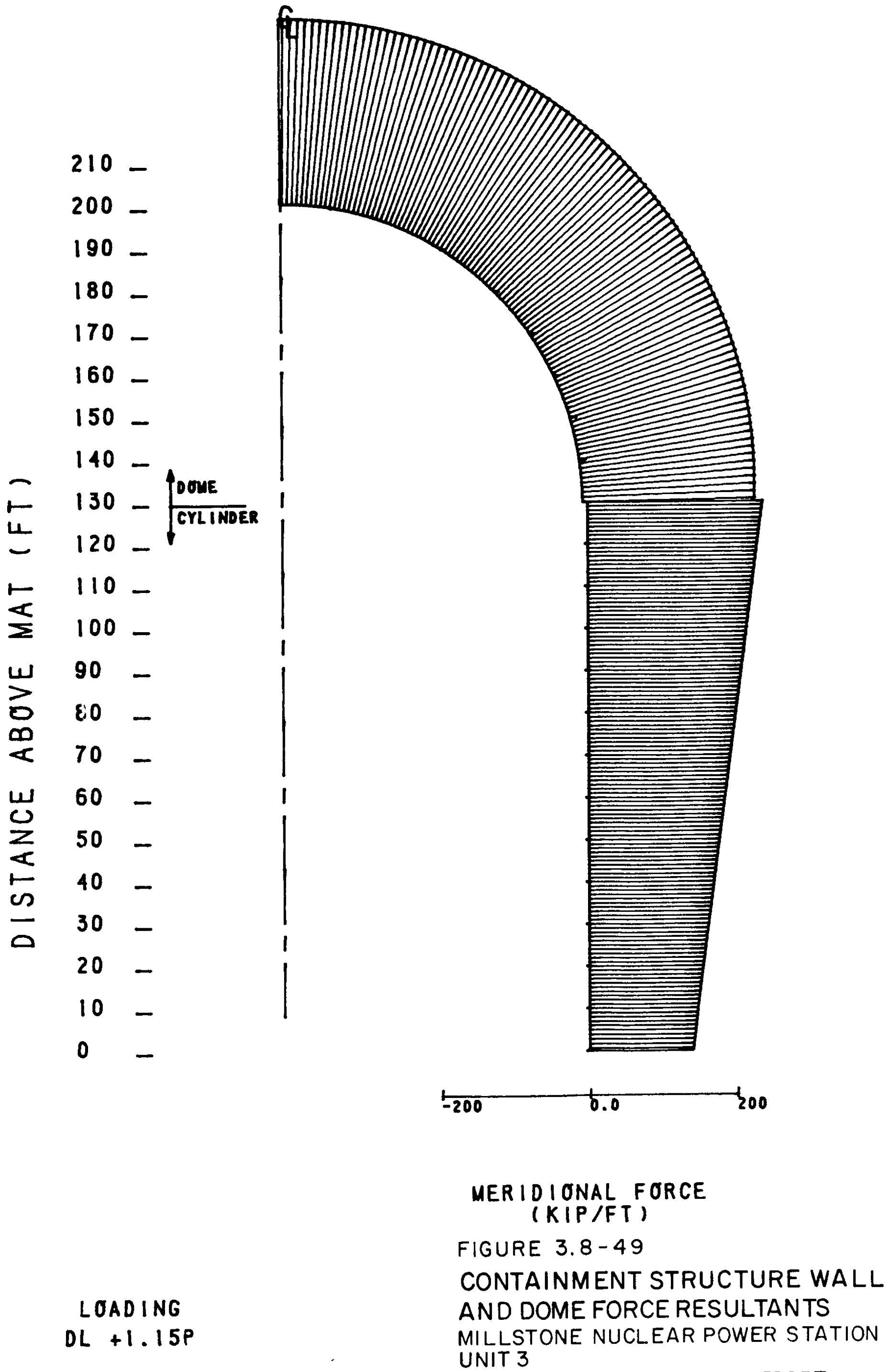


FIGURE 3.8-49
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–50 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

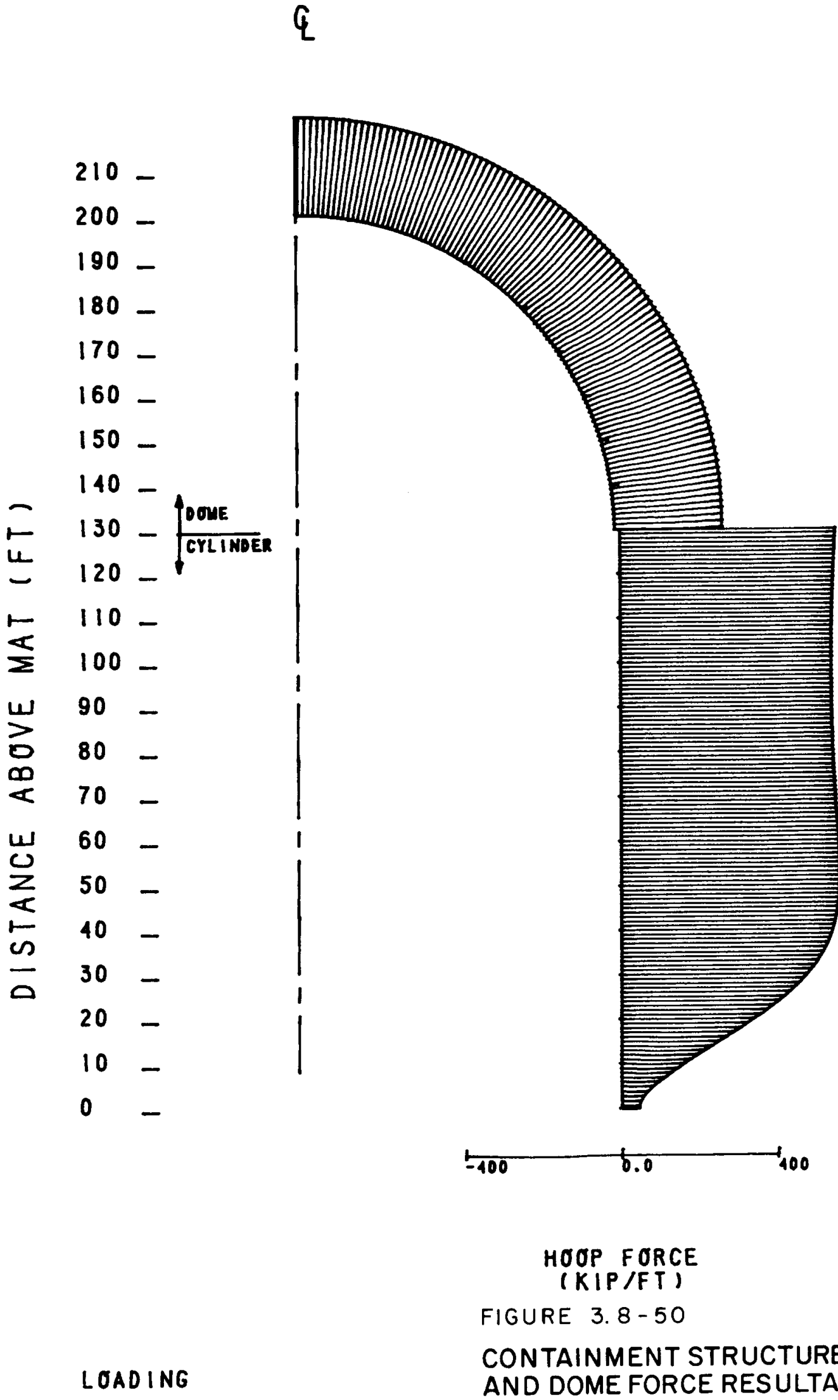


FIGURE 3.8-50

CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–51 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

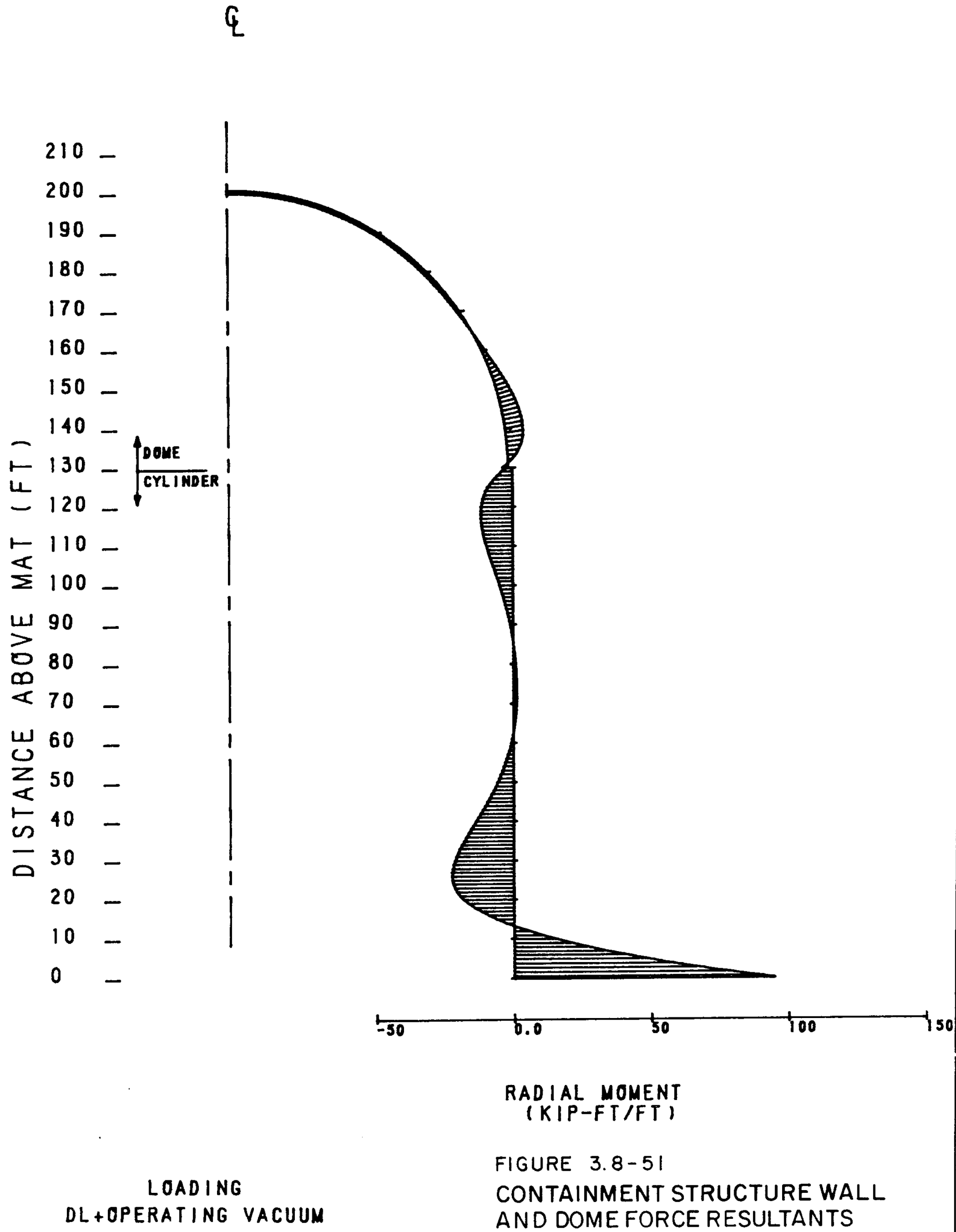


FIGURE 3.8-51
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–52 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

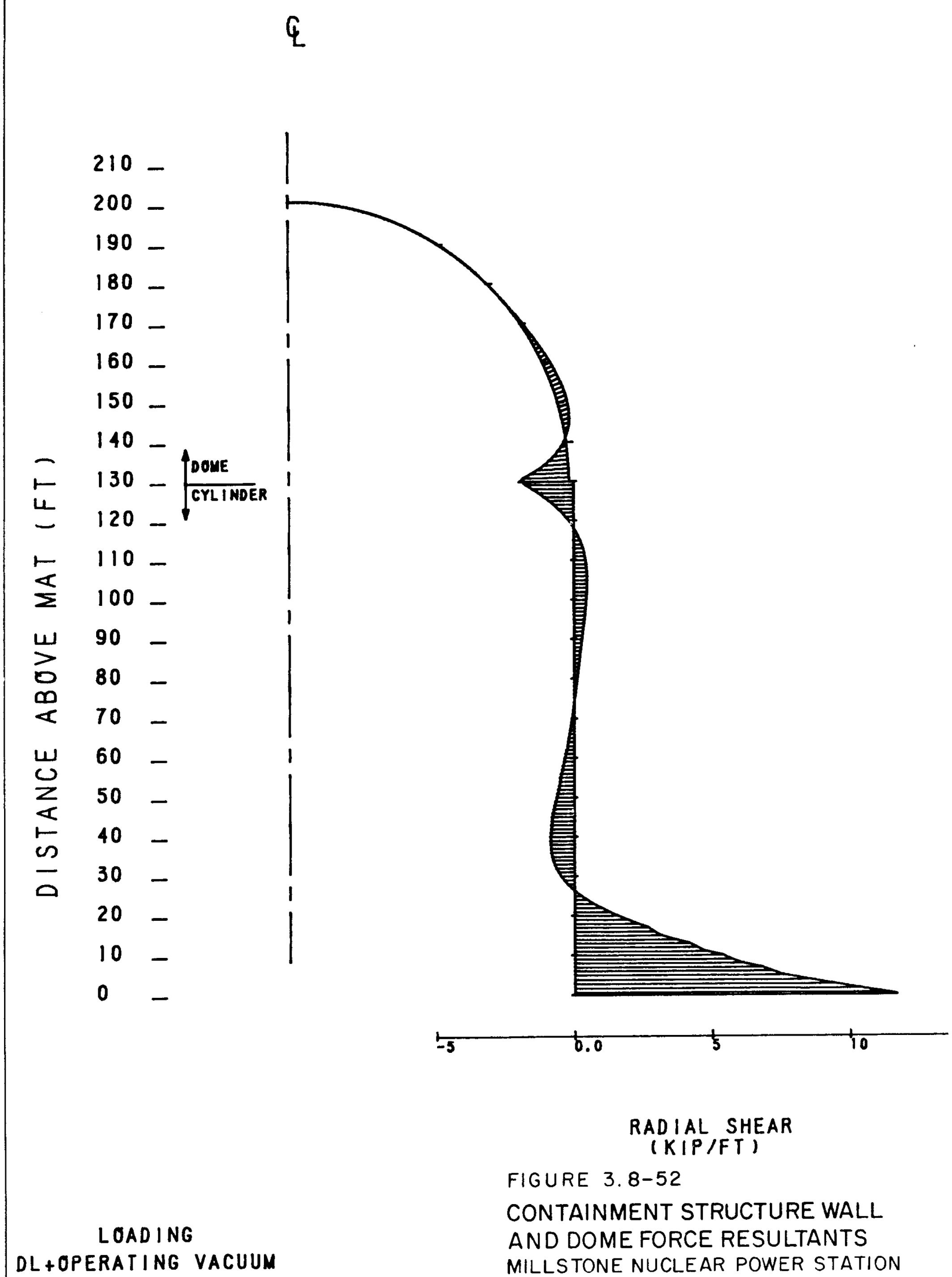


FIGURE 3.8-52
CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–53 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

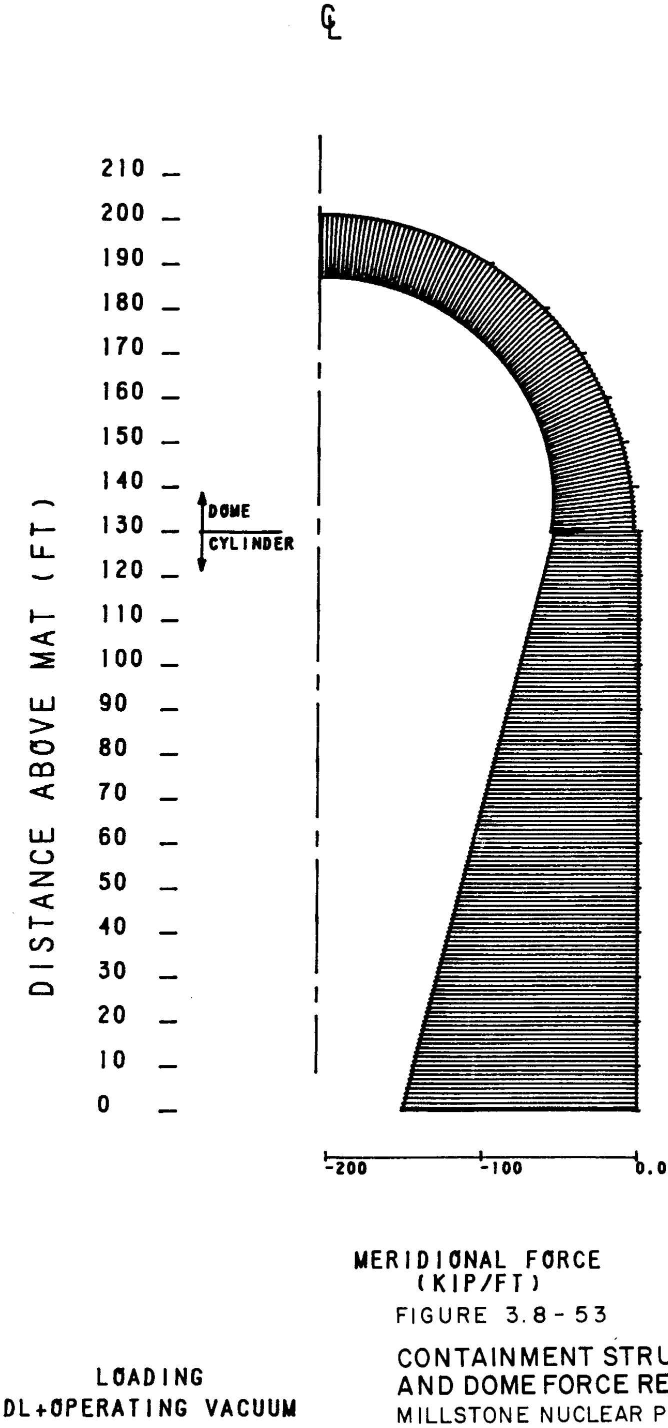


FIGURE 3.8-53

CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

MILLSTONE NUCLEAR POWER STATION UNIT 3

FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–54 CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

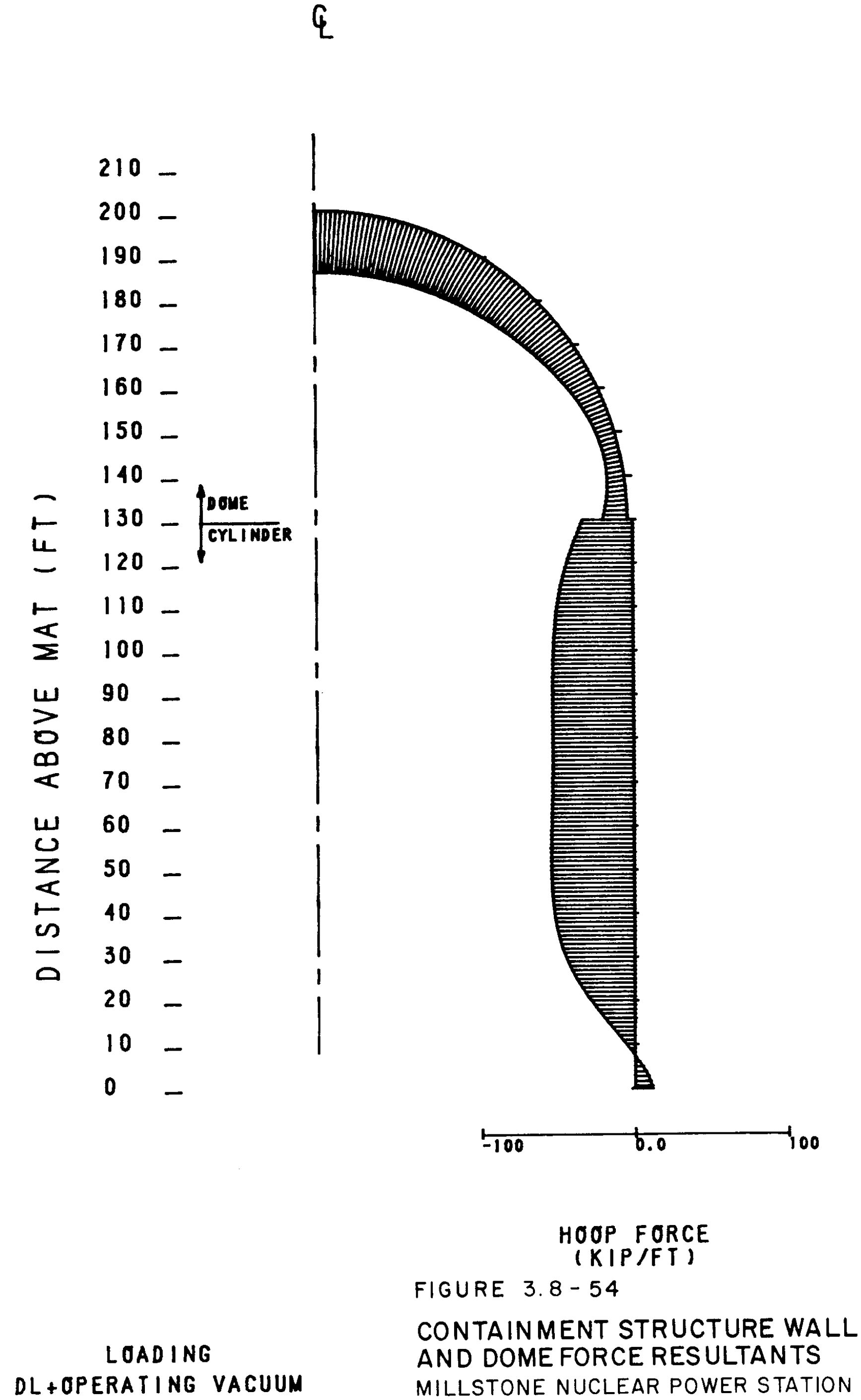


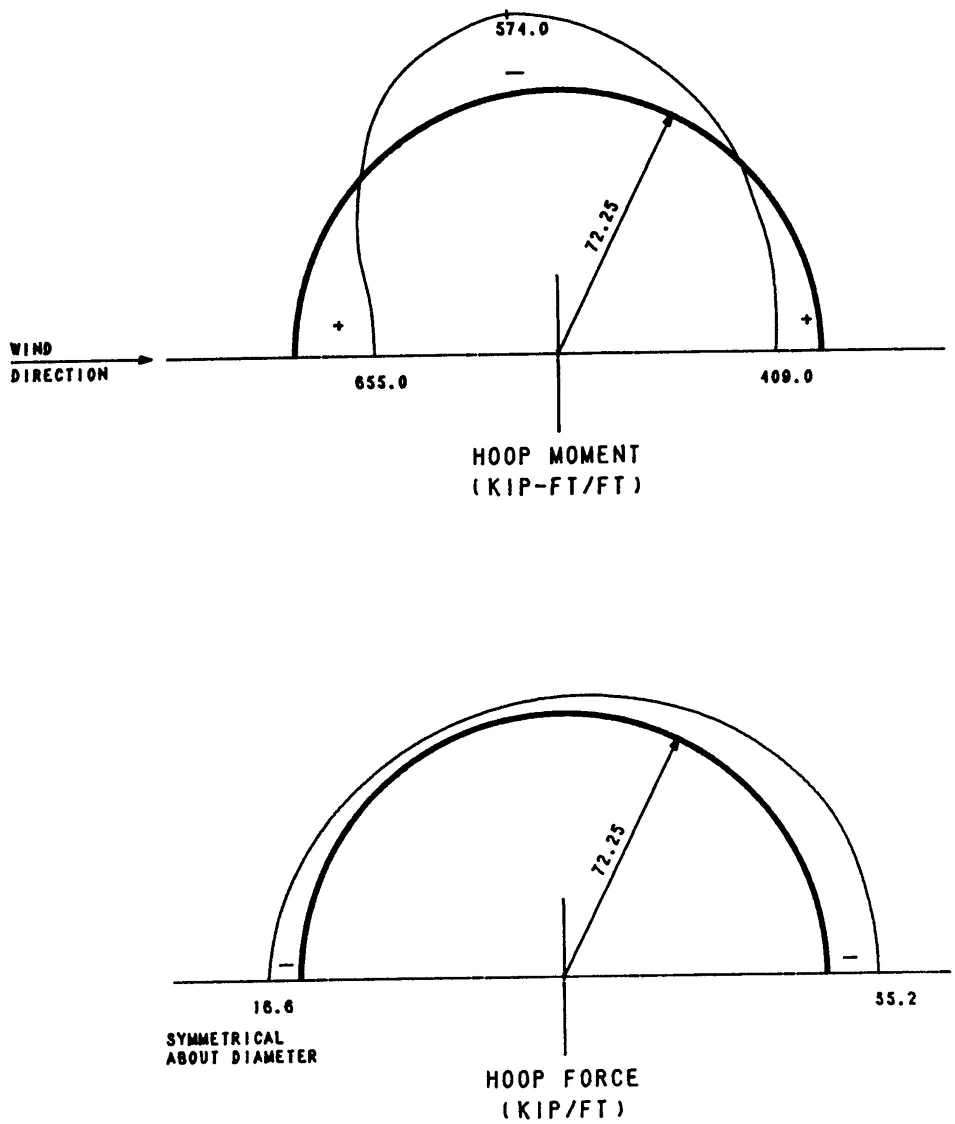
FIGURE 3.8 - 54

CONTAINMENT STRUCTURE WALL AND DOME FORCE RESULTANTS

MILLSTONE NUCLEAR POWER STATION UNIT 3

FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–55 CONTAINMENT STRUCTURE WALL AND DOME



PLAN OF CONTAINMENT CYLINDER

FIGURE 3.8-55

CONTAINMENT STRUCTURE
WALL AND DOME
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

LOADING: TORNADO WINDS

VELOCITY = 360 MPH,
EQUIVALENT TO LOAD OF 0.35 K/SQ. FT.

LOAD DISTRIBUTION BASED ON
ASCE PAPER NO. 3269

FIGURE 3.8–56 DETAILS OF WALL DIAGONALS AT CONTAINMENT MAT

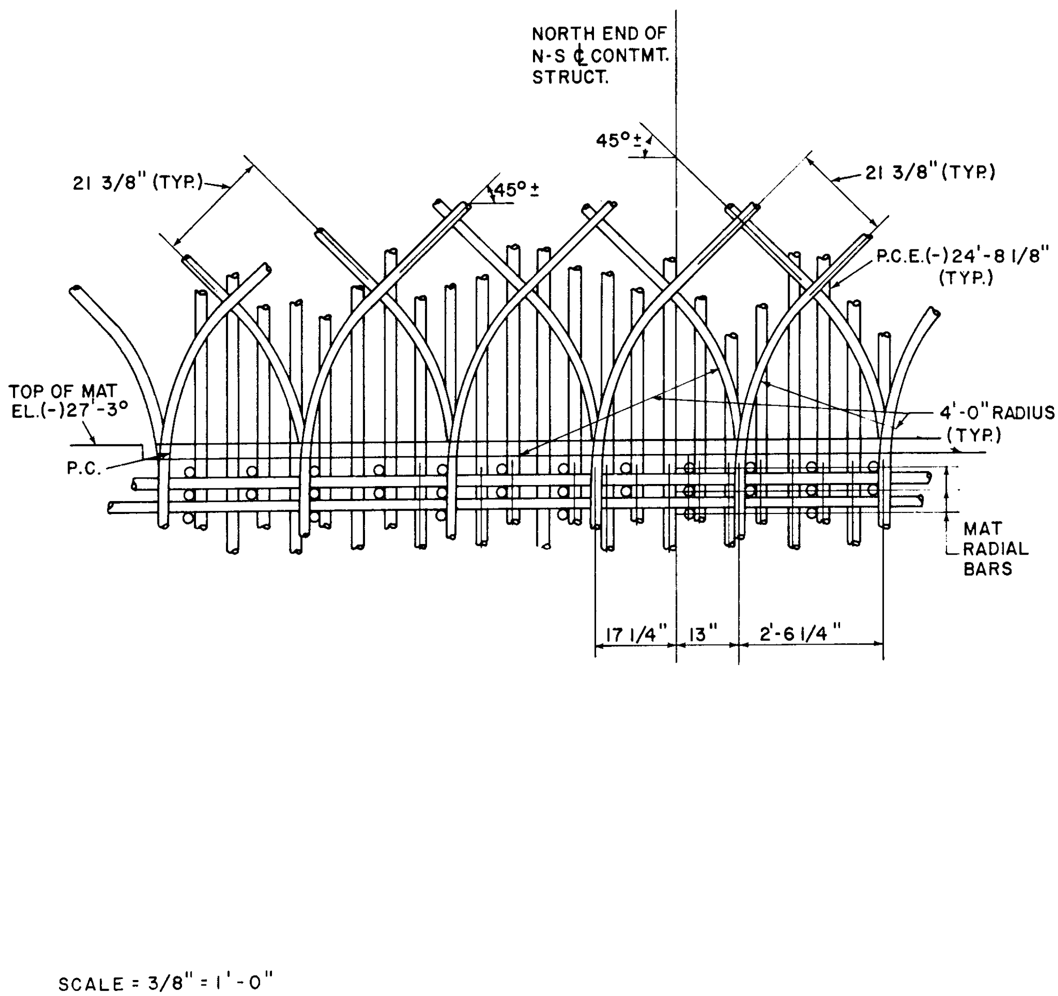


FIGURE 3.8-56
DETAILS OF WALL DIAGONALS
AT CONTAINMENT MAT
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–57 DETAILS OF WATERPROOF MEMBRANE (SHEET 1 OF 2)(PART 1 OF 2)

MNPS-3 FSAR

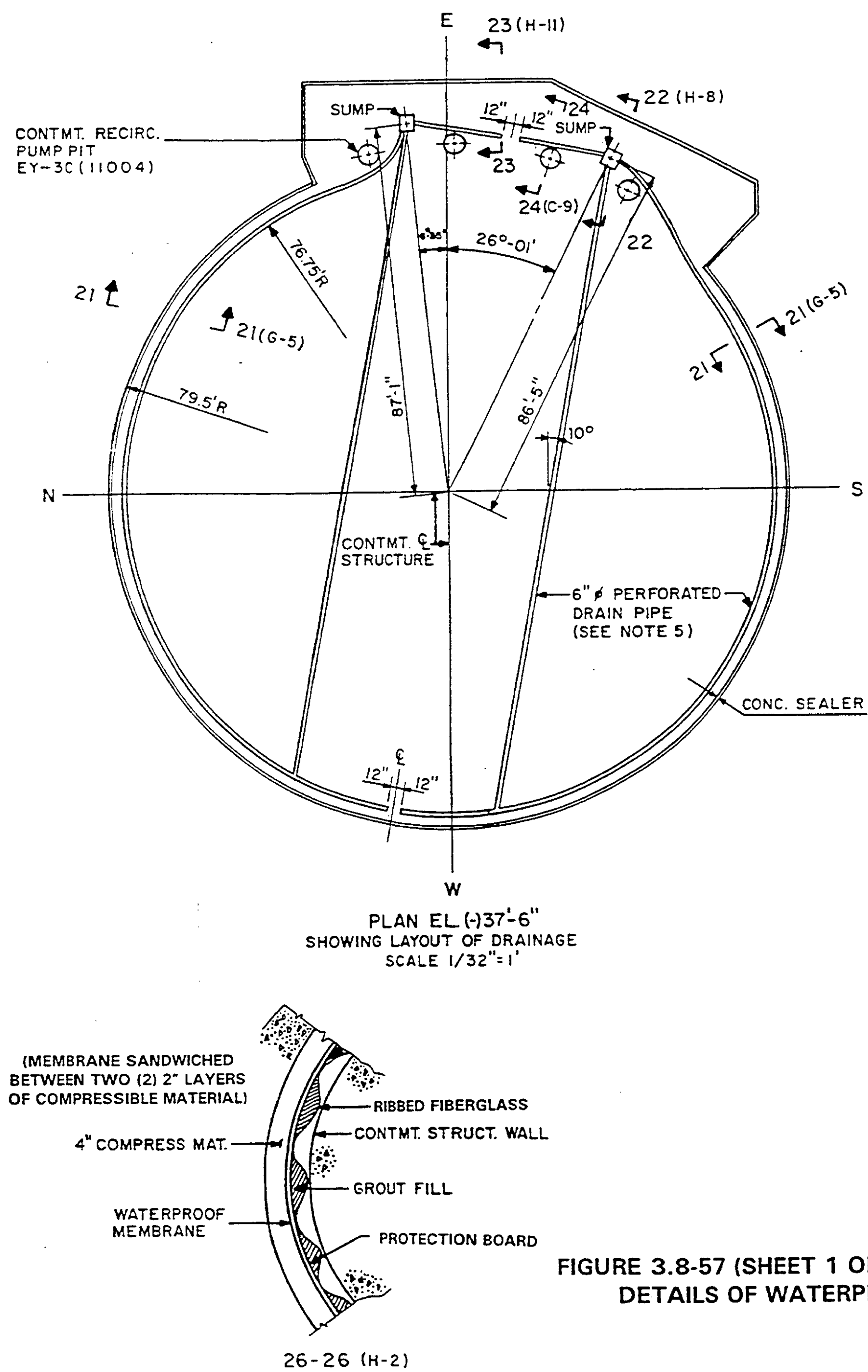
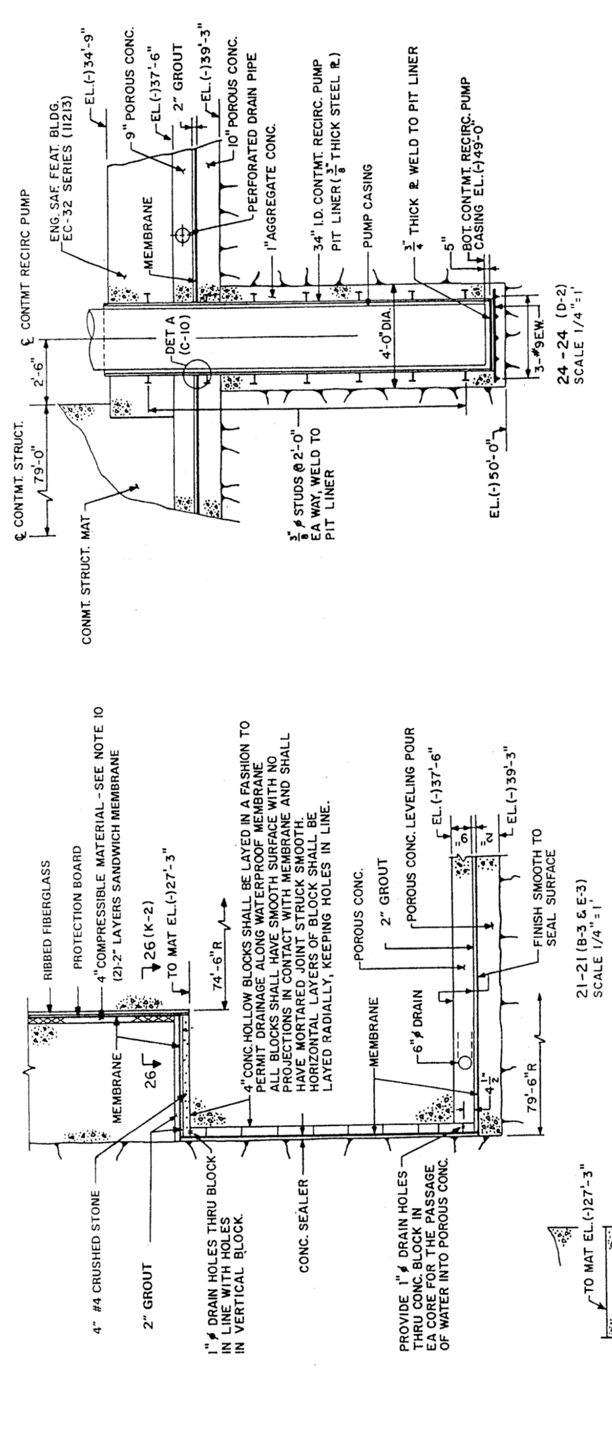


FIGURE 3.8-57 (SHEET 1 OF 2)(PART 1 OF 2)
DETAILS OF WATERPROOF MEMBRANE

NOVEMBER 1997

FIGURE 3.8–57 DETAILS OF WATERPROOF MEMBRANE (SHEET 1 OF 2)(PART 2 OF 2)



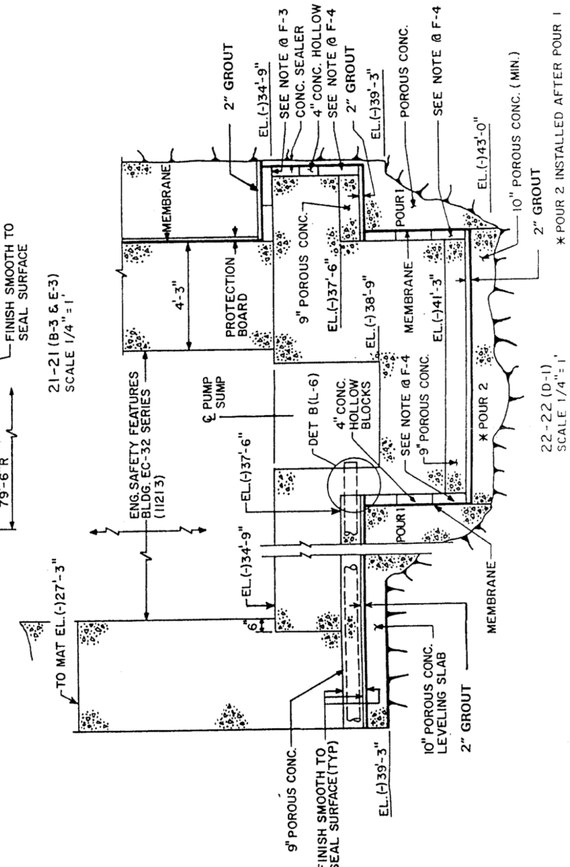


FIGURE 3.8-57 (SHEET 1 OF 2)(PART 2 OF 2)
DETAILS OF WATERPROOF MEMBRANE

NOVEMBER 1997

FIGURE 3.8–57 DETAILS OF WATERPROOF MEMBRANE (SHEET 2 OF 2)

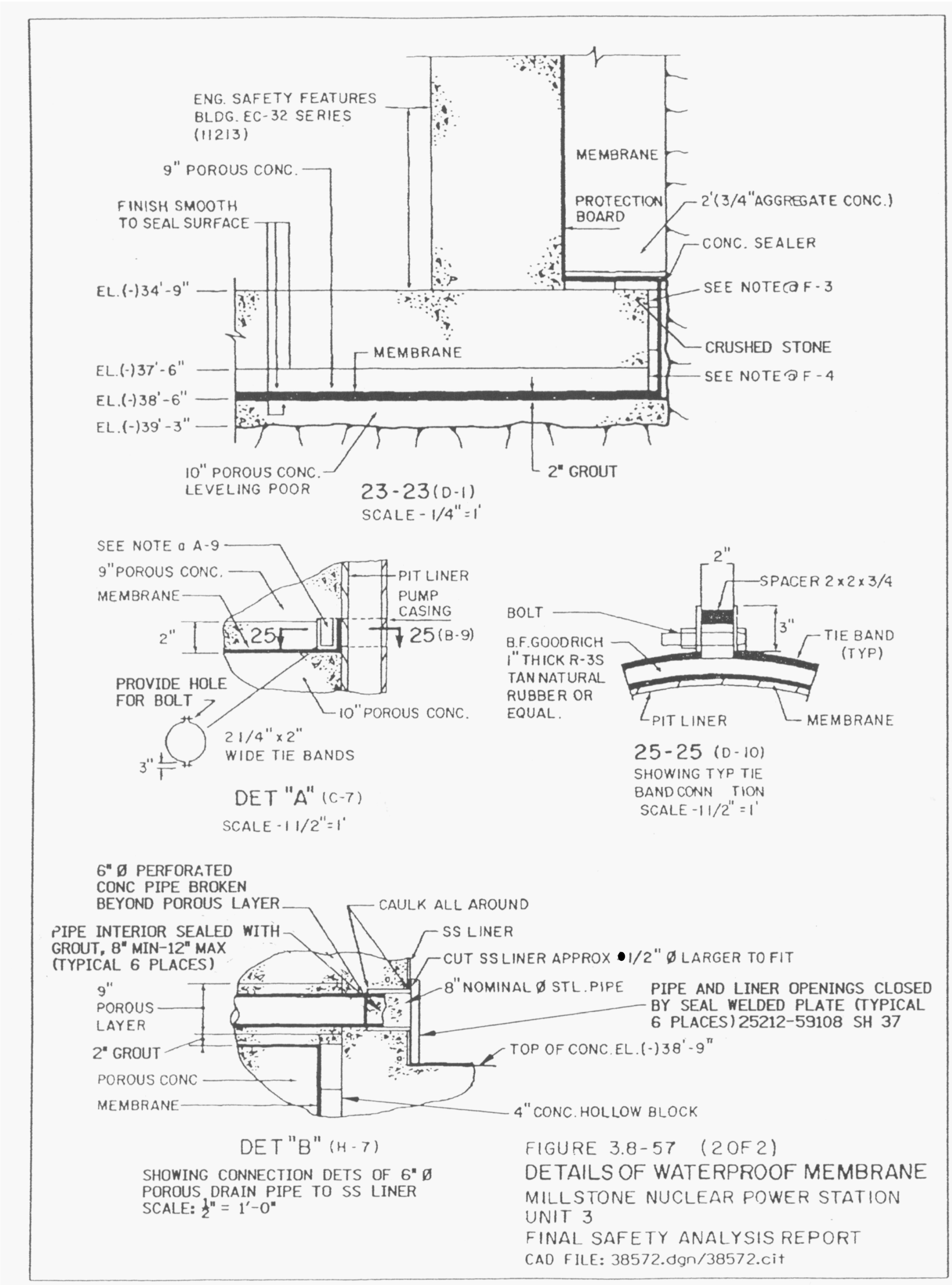


FIGURE 3.8–58 DISPLACEMENT PROFILE FOR CONTAINMENT SHELL UNDER PRESSURE TEST (PREDICTED) (SHEET 1 OF 2)

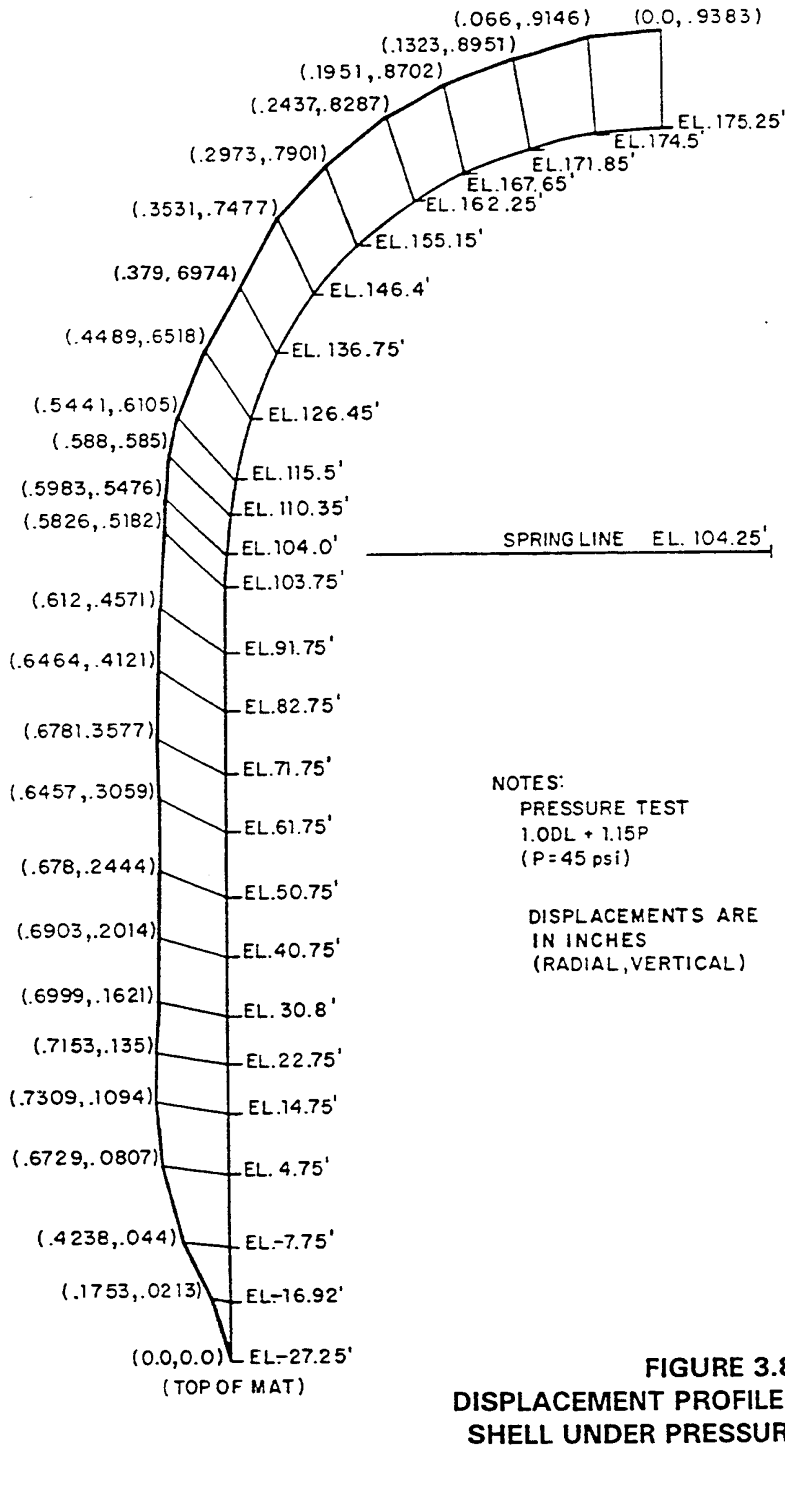


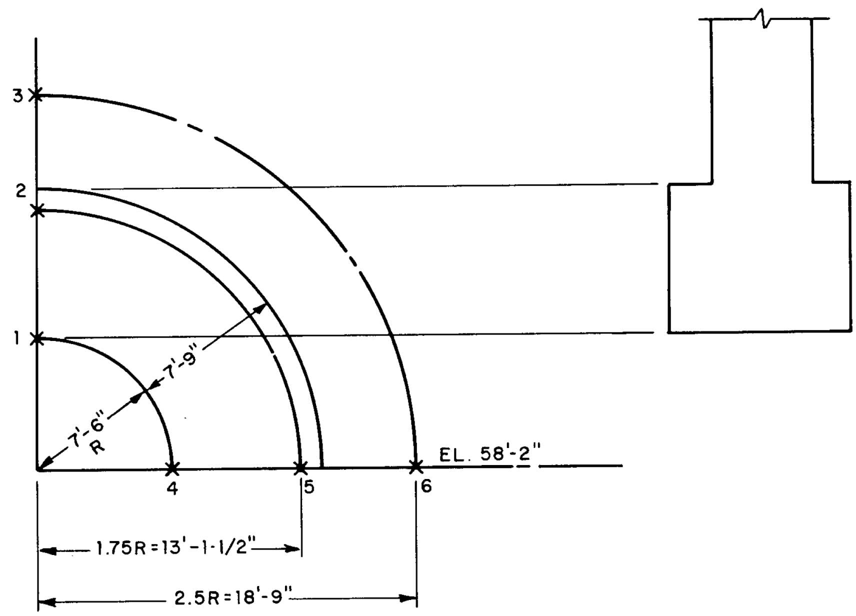
FIGURE 3.8-58 (1 OF 2)
DISPLACEMENT PROFILE FOR CONTAINMENT SHELL UNDER PRESSURE TEST (PREDICTED)

FIGURE 3.8–58 DISPLACEMENT PROFILE FOR CONTAINMENT SHELL UNDER PRESSURE TEST (PREDICTED) (SHEET 2 OF 2)

LOCATION	RADIAL DISPL. -IN.	TANG. DISPL. -IN.
1	0.638	-
2	0.633	-
3	0.624	-
4	0.659	0.044
5	0.644	-0.0013
6	0.660	0.011

NOTES:

1. TANGENTIAL DISPLACEMENTS ARE HORIZONTAL, IN THE PLANE OF THE PAPER, POSITIVE TO THE RIGHT.
2. RADIAL DISPLACEMENTS ARE HORIZONTAL, NORMAL TO THE PLANE OF THE PAPER, POSITIVE INTO THE PAPER.
3. DISPLACEMENTS AT 6 AND 9 O'CLOCK ARE SYMMETRICAL TO THOSE GIVEN AT 12 AND 3 O'CLOCK RESPECTIVELY.
4. PRESSURE TEST 1.0DL + 1.15P (P = 45psi)



ELEVATION EQUIPMENT HATCH

FIGURE 3.8-58 (2 OF 2)
DISPLACEMENT PROFILE FOR CONTAINMENT SHELL UNDER PRESSURE TEST (PREDICTED)
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–59 CONTAINMENT STRUCTURE, PLAN VIEWS

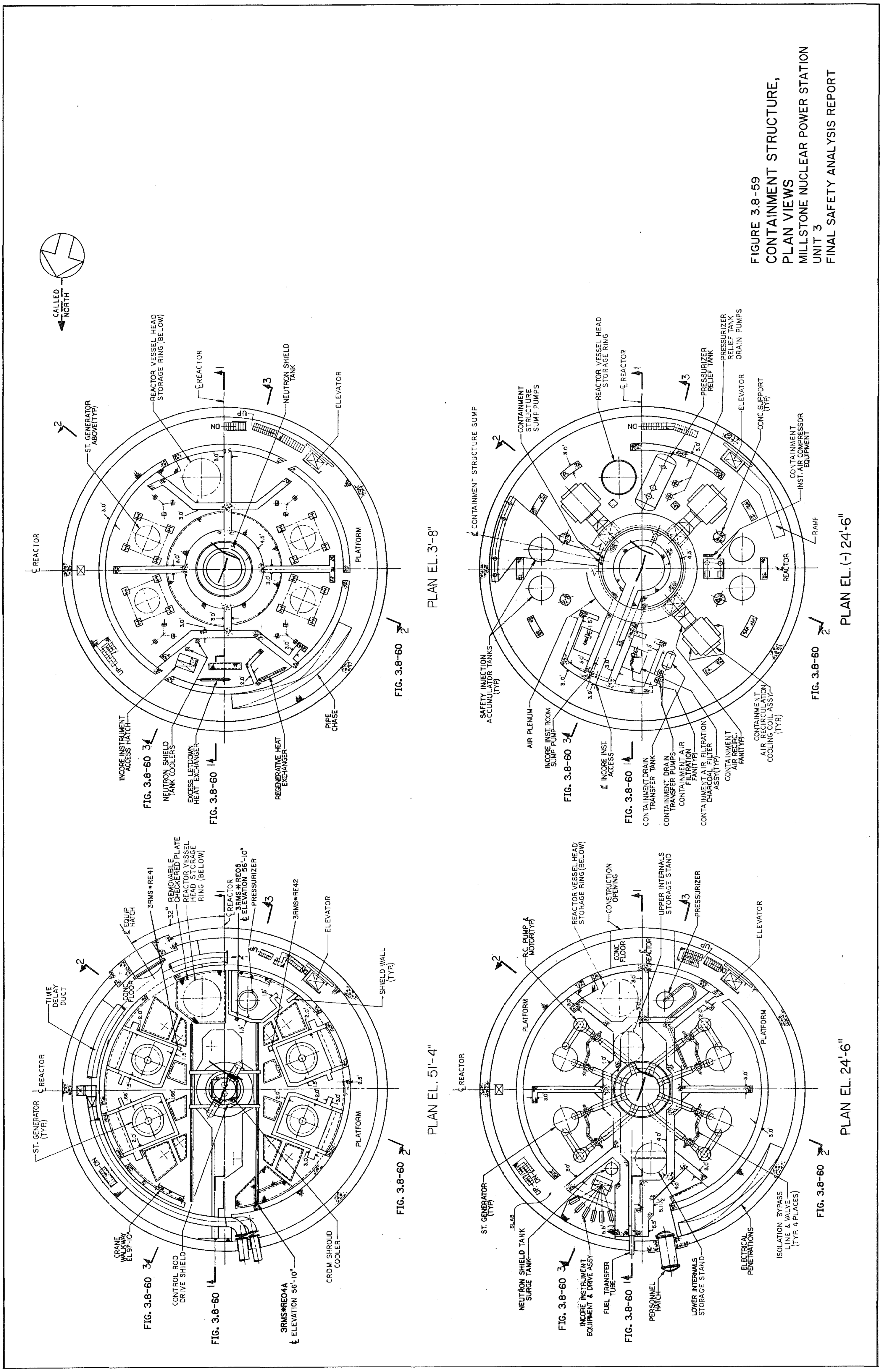


FIGURE 3.8–60 CONTAINMENT STRUCTURE, ELEVATION VIEWS

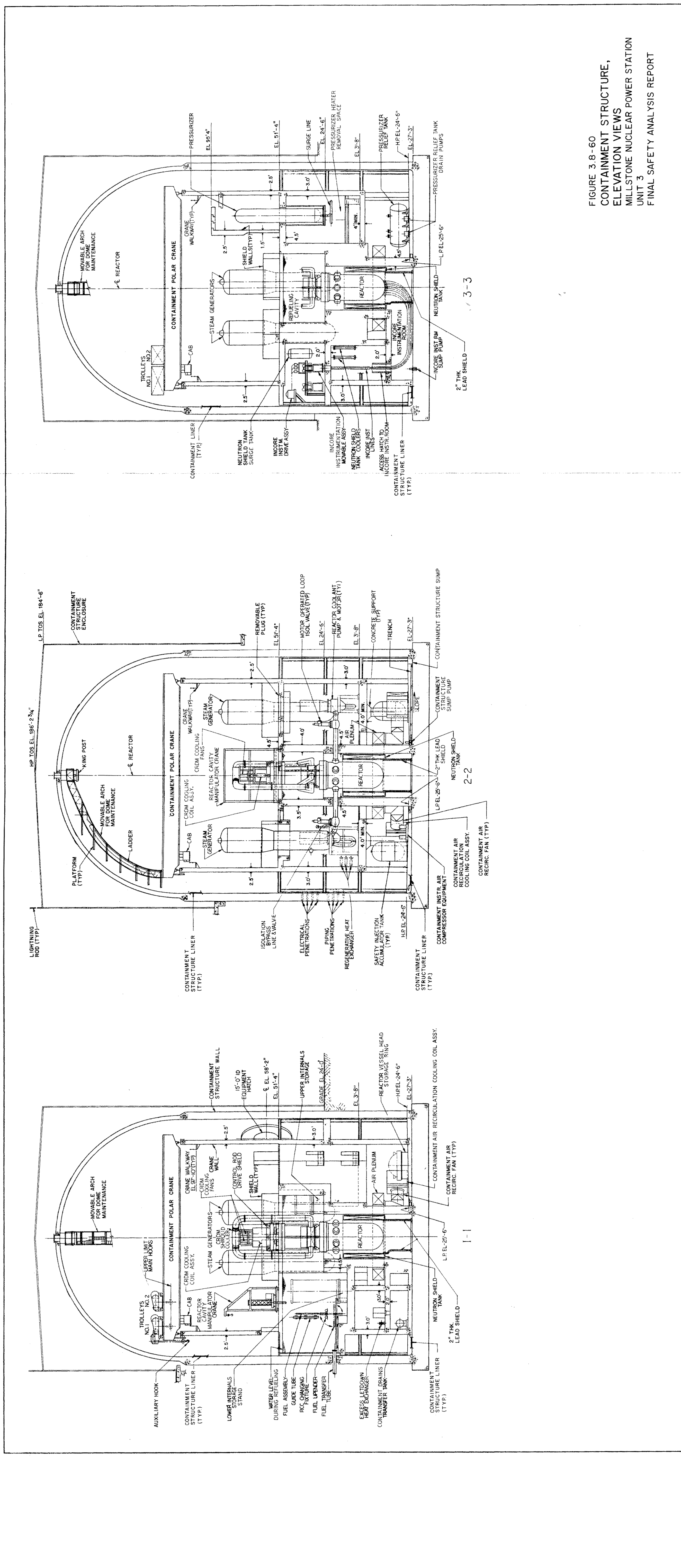
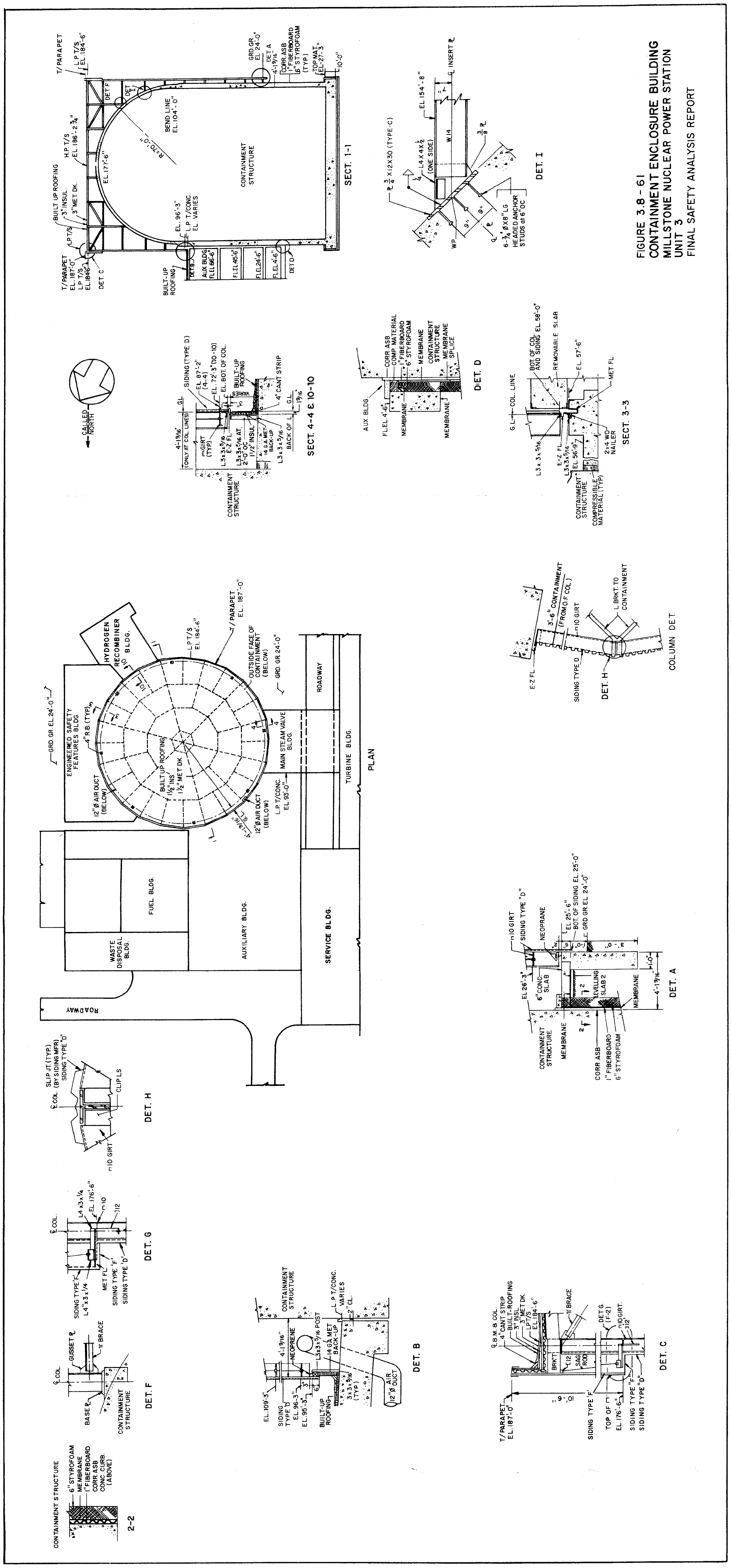


FIGURE 3.8–61 CONTAINMENT ENCLOSURE BUILDING



Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–62 AUXILIARY BUILDING (SHEET 1 OF 10)

FIGURE 3.8–62 AUXILIARY BUILDING (SHEET 2 OF 10)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–62 AUXILIARY BUILDING (SHEET 3 OF 10)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–62 AUXILIARY BUILDING (SHEET 4 OF 10)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–62 AUXILIARY BUILDING (SHEET 5 OF 10)

[redacted]

[redacted]

FIGURE 3.8–62 AUXILIARY BUILDING (SHEET 6 OF 10)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–62 AUXILIARY BUILDING (SHEET 7 OF 10)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–62 AUXILIARY BUILDING (SHEET 8 OF 10)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–62 AUXILIARY BUILDING (SHEET 9 OF 10)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–62 AUXILIARY BUILDING (SHEET 10 OF 10)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–63 FUEL BUILDING (SHEET 1 OF 6)

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–63 FUEL BUILDING (SHEET 2 OF 6)

[redacted]

[redacted]

FIGURE 3.8–63 FUEL BUILDING (SHEET 3 OF 6)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–63 FUEL BUILDING (SHEET 4 OF 6)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–63 FUEL BUILDING (SHEET 5 OF 6)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–63 FUEL BUILDING (SHEET 6 OF 6)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–64 CONTROL ROOM AREA (SHEET 1 OF 4)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–64 CONTROL ROOM AREA (SHEET 2 OF 4)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–64 CONTROL ROOM AREA (SHEET 3 OF 4)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–64 CONTROL ROOM AREA (SHEET 4 OF 4)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–65 CABLE TUNNEL

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–66 EMERGENCY GENERATOR ENCLOSURE (SHEET 1 OF 5)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–66 EMERGENCY GENERATOR ENCLOSURE (SHEET 2 OF 5)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–66 EMERGENCY GENERATOR ENCLOSURE (SHEET 3 OF 5)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–66 EMERGENCY GENERATOR ENCLOSURE (SHEET 4 OF 5)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–66 EMERGENCY GENERATOR ENCLOSURE (SHEET 5 OF 5)

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

NOTES TO FIGURE 3.8–66 DATA FOR PIPE RUPTURE RESTRAINTS STEAM GENERATOR BLOWDOWN SYSTEM

[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
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[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–67 ENGINEERED SAFETY FEATURES BUILDING (SHEET 1 OF 9)

[redacted]

FIGURE 3.8–67 ENGINEERED SAFETY FEATURES BUILDING (SHEET 2 OF 9)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–67 ENGINEERED SAFETY FEATURES BUILDING (SHEET 3 OF 9)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–67 ENGINEERED SAFETY FEATURES BUILDING (SHEET 4 OF 9)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–67 ENGINEERED SAFETY FEATURES BUILDING (SHEET 5 OF 9)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–67 ENGINEERED SAFETY FEATURES BUILDING (SHEET 6 OF 9)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–67 ENGINEERED SAFETY FEATURES BUILDING (SHEET 7 OF 9)

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–67 ENGINEERED SAFETY FEATURES BUILDING (SHEET 8 OF 9)

FIGURE 3.8–67 ENGINEERED SAFETY FEATURES BUILDING (SHEET 9 OF 9)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–68 MAIN STEAM VALVE BUILDING (SHEET 1 OF 4)

FIGURE 3.8–68 MAIN STEAM VALVE BUILDING (SHEET 2 OF 4)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–68 MAIN STEAM VALVE BUILDING (SHEET 3 OF 4)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–68 MAIN STEAM VALVE BUILDING (SHEET 4 OF 4)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–69 CIRCULATING AND SERVICE WATER PUMPHOUSE (SHEET 1 OF 4)

[redacted]

FIGURE 3.8–69 CIRCULATING AND SERVICE WATER PUMPHOUSE (SHEET 2 OF 4)

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–69 CIRCULATING AND SERVICE WATER PUMPHOUSE (SHEET 3 OF 4)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–69 CIRCULATING AND SERVICE WATER PUMPHOUSE (SHEET 4 OF 4)

[redacted]

[redacted]

FIGURE 3.8–70 RECOMBINER BUILDING (SHEET 1 OF 4)

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–70 RECOMBINER BUILDING (SHEET 2 OF 4)

[redacted]

[redacted]

FIGURE 3.8–70 RECOMBINER BUILDING (SHEET 3 OF 4)

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–70 RECOMBINER BUILDING (SHEET 4 OF 4)

[redacted]

[redacted]

FIGURE 3.8–71 CIRCULATING DISCHARGE STRUCTURE AND TUNNEL

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–72 SERVICE BUILDING (SHEET 1 OF 4)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–72 SERVICE BUILDING (SHEET 2 OF 4)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–72 SERVICE BUILDING (SHEET 3 OF 4)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–72 SERVICE BUILDING (SHEET 4 OF 4)

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–73 TURBINE BUILDING (SHEET 1 OF 5)

[redacted]

FIGURE 3.8–73 TURBINE BUILDING (SHEET 2 OF 5)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–73 TURBINE BUILDING (SHEET 3 OF 5)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–73 TURBINE BUILDING (SHEET 4 OF 5)

FIGURE 3.8–73 TURBINE BUILDING (SHEET 5 OF 5)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–74 SOLID AND LIQUID WASTE DISPOSAL BUILDING (SHEET 1 OF 7)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–74 SOLID AND LIQUID WASTE DISPOSAL BUILDING (SHEET 2 OF 7)

[redacted]

FIGURE 3.8–74 SOLID AND LIQUID WASTE DISPOSAL BUILDING (SHEET 3 OF 7)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–74 SOLID AND LIQUID WASTE DISPOSAL BUILDING (SHEET 4 OF 7)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–74 SOLID AND LIQUID WASTE DISPOSAL BUILDING (SHEET 5 OF 7)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–74 SOLID AND LIQUID WASTE DISPOSAL BUILDING (SHEET 6 OF 7)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–74 SOLID AND LIQUID WASTE DISPOSAL BUILDING (SHEET 7 OF 7)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 1 OF 12)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 2 OF 12)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 3 OF 12)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 4 OF 12)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 5 OF 12)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 6 OF 12)

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 7 OF 12)

[redacted]

[redacted]

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 8 OF 12)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 9 OF 12)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 10 OF 12)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 11 OF 12)

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–75 WAREHOUSE NO. 5 (SHEET 12 OF 12)

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–76 AUXILIARY BOILER ENCLOSURE (SHEET 1 OF 5)

[redacted]

FIGURE 3.8–76 AUXILIARY BOILER ENCLOSURE (SHEET 2 OF 5)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–76 AUXILIARY BOILER ENCLOSURE (SHEET 3 OF 5)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

FIGURE 3.8–76 AUXILIARY BOILER ENCLOSURE (SHEET 4 OF 5)

[redacted]

FIGURE 3.8–76 AUXILIARY BOILER ENCLOSURE (SHEET 5 OF 5)

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 3.8–77 ARRANGEMENT OF HORIZONTAL SHEAR KEYS

[redacted]

FIGURE 3.8–78 TYPICAL SHEAR KEY DETAIL

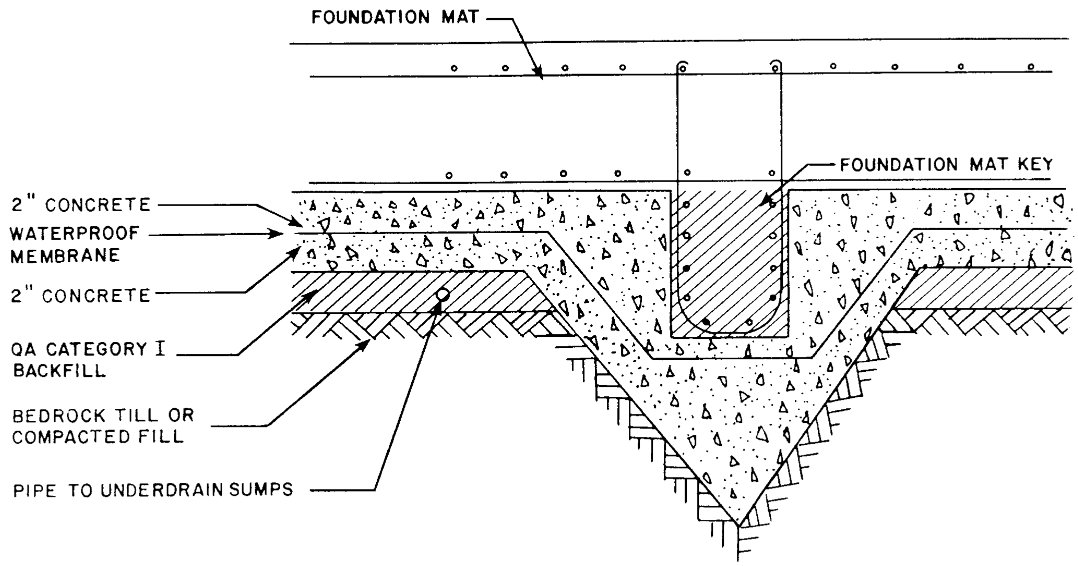


FIGURE 3.8-78
TYPICAL SHEAR KEY DETAIL
MILLSTONE NUCLEAR POWER PLANT
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–79 FUEL POOL TEMPERATURE TRANSIENTS

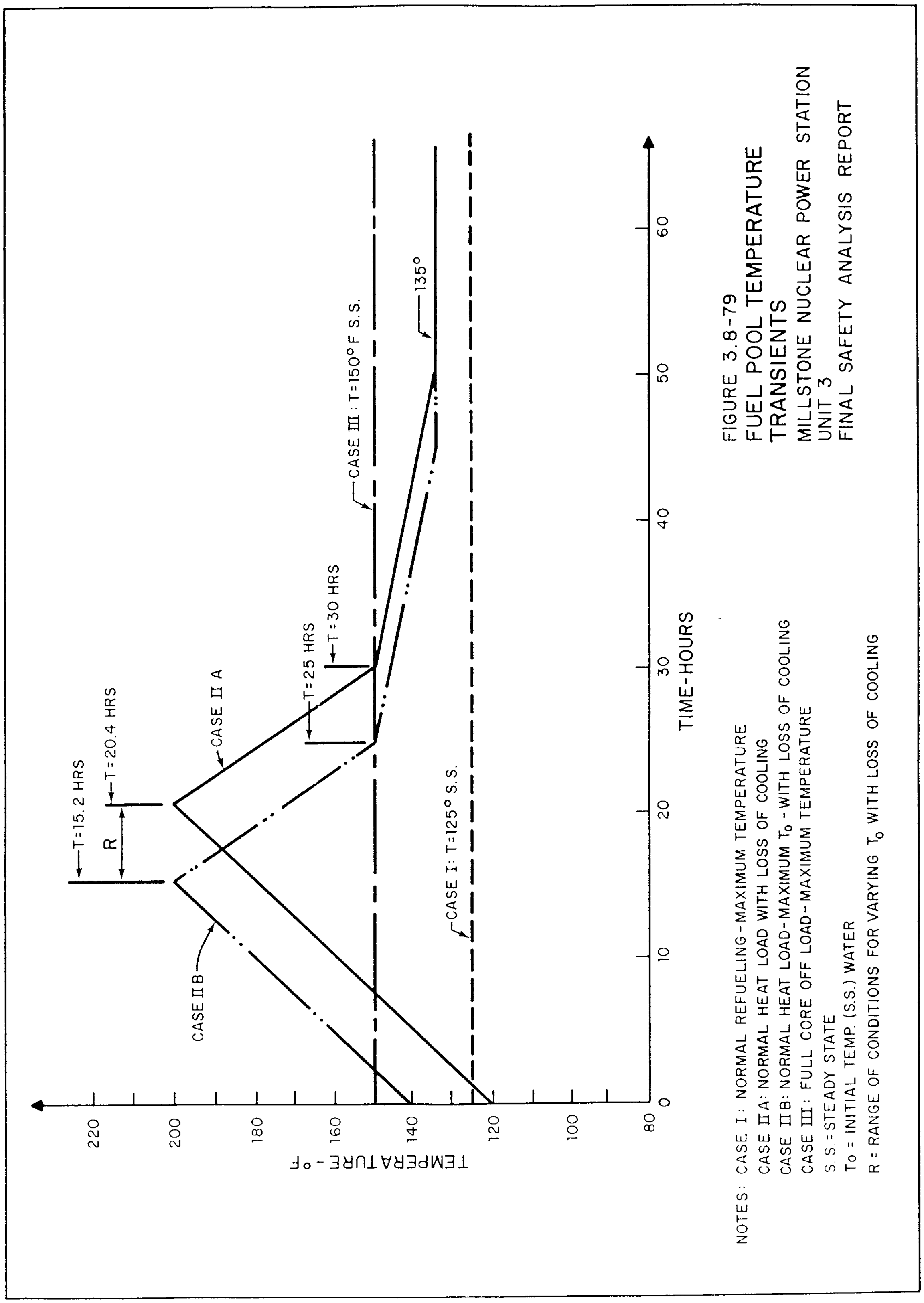
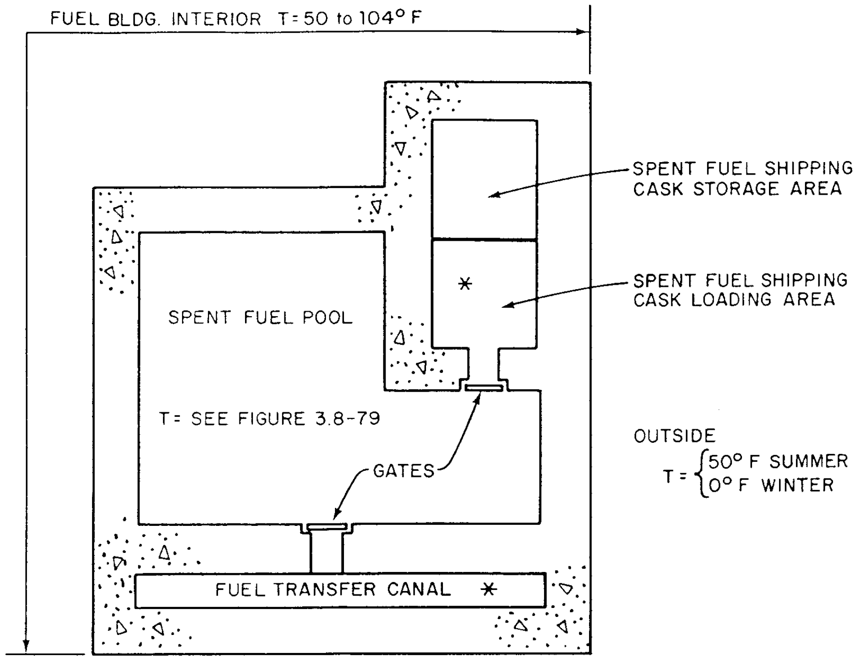


FIGURE 3.8–80 FUEL POOL--FUEL BUILDING TEMPERATURE CONDITIONS



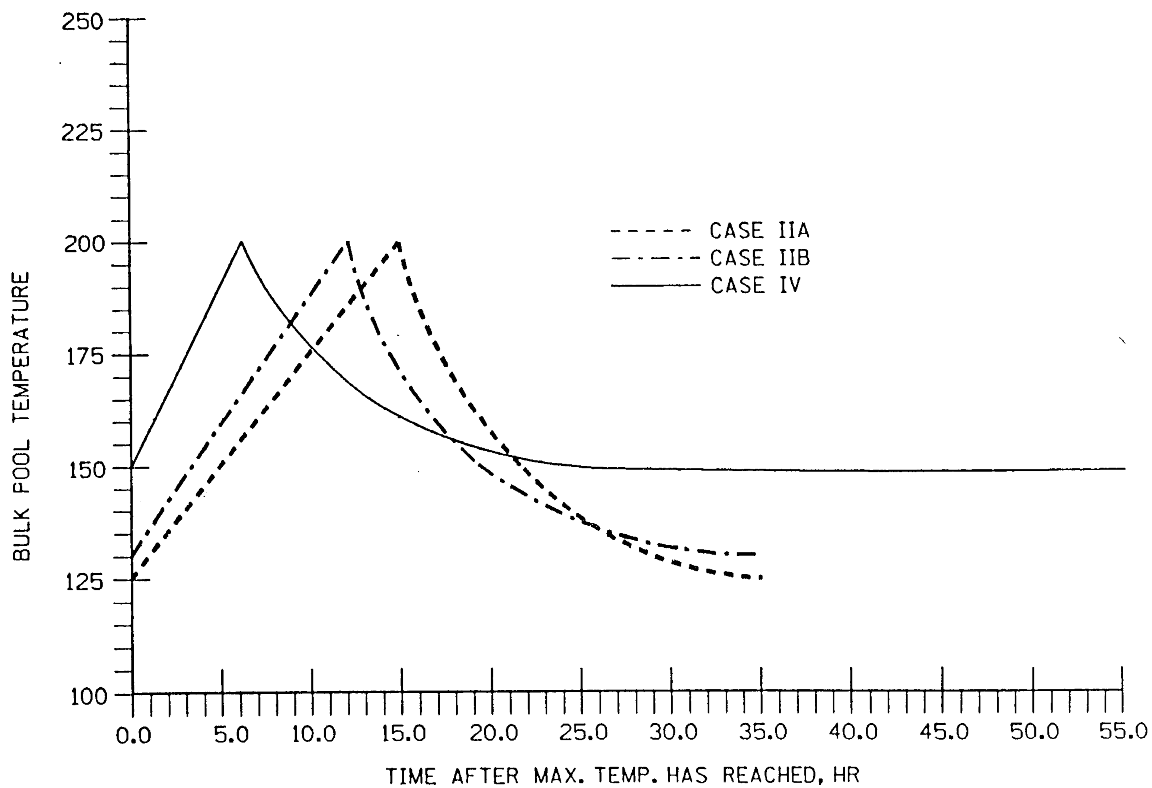
NOTES: T_{OC} = INITIAL TEMPERATURE OF CONCRETE (70 to 90°F)

$*\begin{cases}\text{T=50°F WHEN EMPTY (GATE IN PLACE)}\\ \text{T=CASE II A \& B, FOR FUEL POOL TEMP. WHEN FULL (GATE NOT IN PLACE)}\end{cases}$

T (6 FEET BELOW GRADE) = 43° F MINIMUM

FIGURE 3.8-80
FUEL POOL-FUEL BUILDING
TEMPERATURE CONDITIONS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.8–81 FUEL POOL TEMPERATURE TRANSIENTS--FOR HIGHER ENRICHMENT FUEL



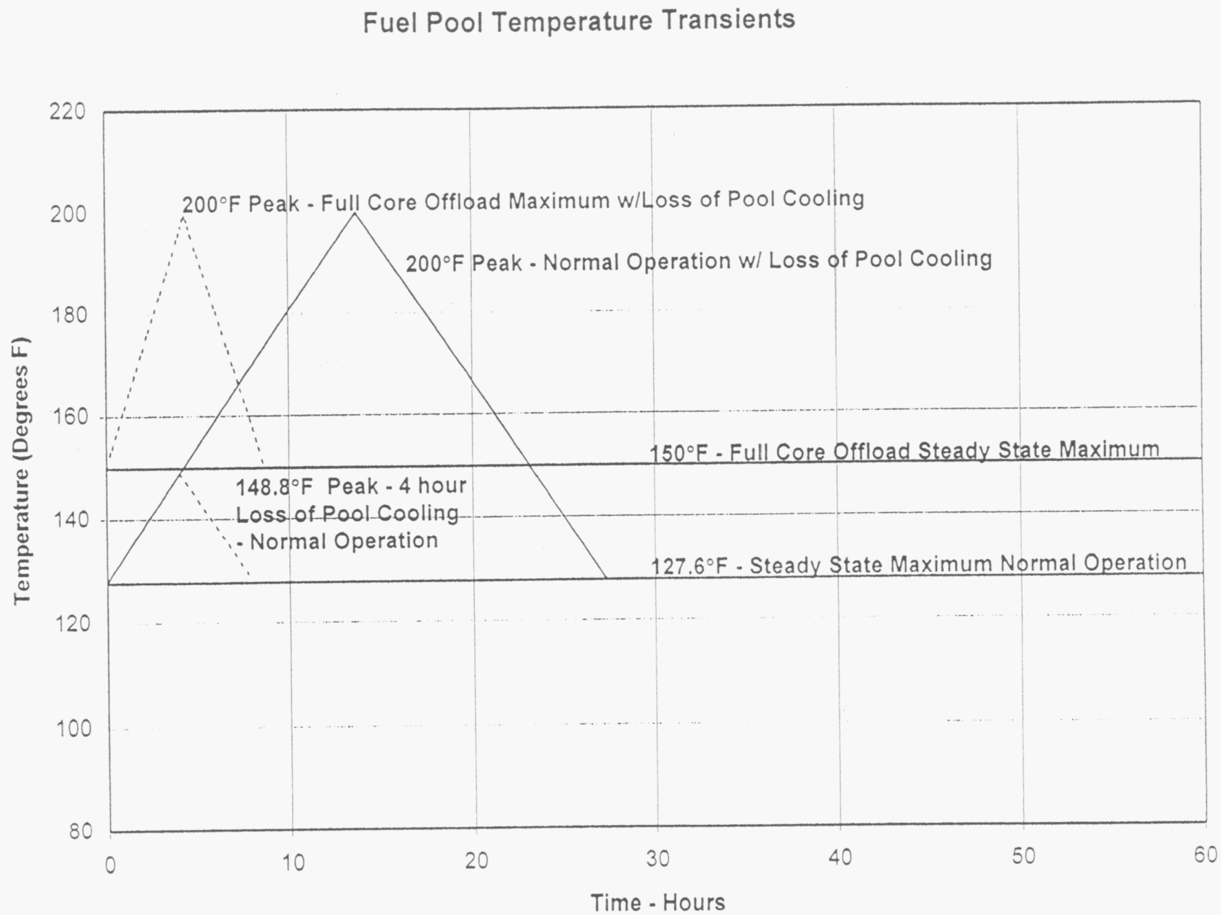
NOTES:

CASE IIA: NORMAL HEAT LOAD WITH LOSS OF COOLING
CASE IIB: NORMAL HEAT LOAD-MAXIMUM TO-WITH OF COOLING
CASE IV: FULL CORE OFF LOAD WITH LOSS OF COOLING

FIGURE 3.8-81
FUEL POOL TEMPERATURE TRANSIENTS -
FOR HIGHER ENRICHMENT FUEL
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

JUNE 1992

FIGURE 3.8–82 FUEL POOL TEMPERATURE TRANSIENTS--FULL CORE OFFLOAD



FSAR Figure 3.8.82
Fuel Pool Temperature Transients - Full Core Offload

FIGURE 3.8–83 MILLSTONE STACK ELEVATIONS AND SECTIONS

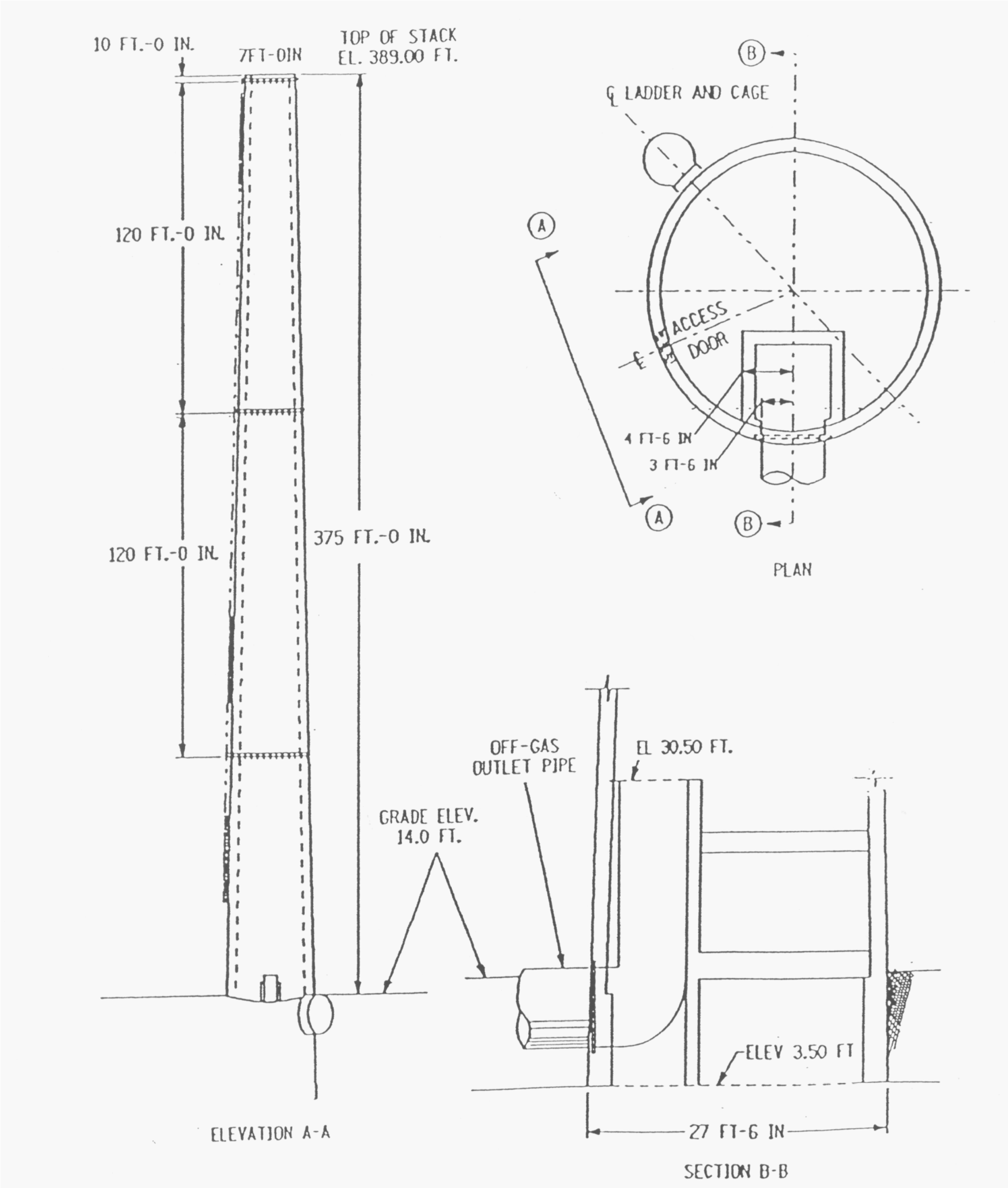
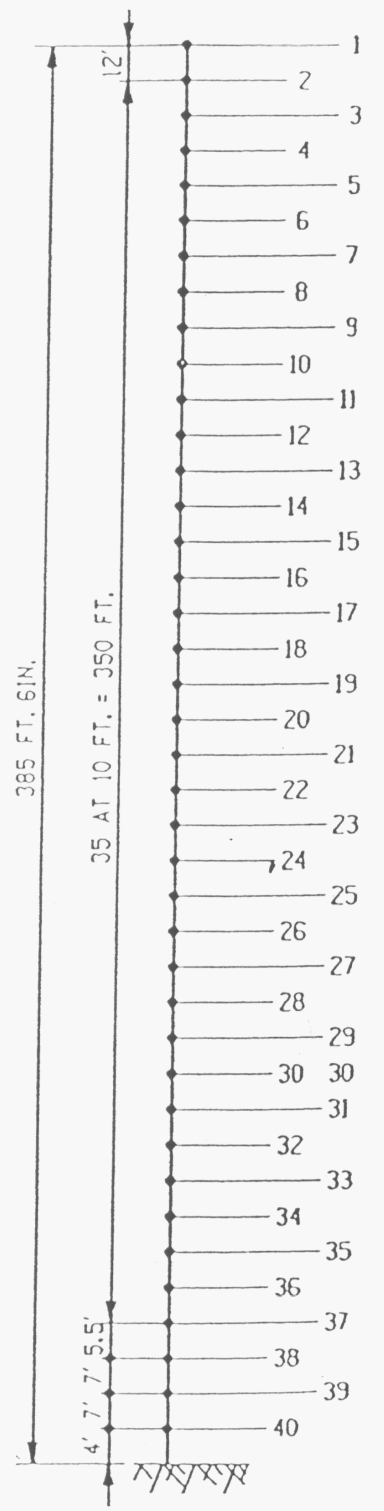


FIGURE 3.8–84 MILLSTONE STACK - MATHEMATICAL MODEL



MASS	WEIGHT(K)	MEMBER	A (FT^2)	I(FT^4)
1	14.68	1-2	16.56	129
2	28.07	2-3	17.47	152
3	26.82	3-4	18.31	175
4	28.08	4-5	19.14	200
5	29.32	5-2	19.97	227
6	30.57	6-6	20.80	257
7	31.83	7-7	21.64	289
8	33.07	8-8	22.47	324
9	34.27	9-10	23.30	361
10	35.58	10-11	24.14	401
11	36.82	11-12	24.97	444
12	38.70	12-13	26.65	508
13	41.28	13-14	28.39	578
14	43.29	14-15	30.17	656
15	46.64	15-16	32.03	741
16	49.47	16-17	33.94	834
17	52.38	17-18	35.90	935
18	55.35	18-19	37.91	1046
19	58.41	19-20	39.98	1166
20	61.56	20-21	42.10	1296
21	64.79	21-22	44.28	1437
22	68.08	22-23	46.51	1589
23	71.48	23-24	48.79	1752
24	74.96	24-25	51.53	1928
25	78.35	25-26	53.52	2117
26	82.12	26-27	55.98	2319
27	85.83	27-28	58.46	2532
28	89.61	28-29	61.02	2762
29	93.49	29-30	63.63	3009
30	97.46	30-31	66.30	3272
31	101.49	31-32	69.03	3552
32	107.76	32-33	74.66	4017
33	116.33	33-34	80.43	4518
34	125.12	34-35	86.38	5060
35	133.90	35-36	92.15	5648
36	147.70	36-37	104.43	6711
37	169.96	37-38	152.52	10602
38	179.35	38-39	177.86	13350
39	335.20	39-40	195.16	12114
40	185.09	40-BASE	213.82	16423

3.9 MECHANICAL SYSTEMS AND COMPONENTS

Sections whose identification numbers include the letter B contain material within the balance-of-plant (BOP) scope, while sections whose identification numbers include the letter N contain material within the nuclear steam supply system (NSSS) scope.

3.9B.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9B.1.1 Design Transients

The design of the reactor coolant system (RCS), RCS component supports, and reactor internals considers the following five operating conditions, defined in Section III of the ASME Boiler and Pressure Vessel Code.

1. Normal Conditions - Any condition in the course of startup, operation in the design power range, hot standby and system shutdown, other than upset, emergency, faulted or testing conditions.

2. Upset Conditions (Incidents of Moderate Frequency) - Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include:

    a. Transients which result from any single operator error or control malfunction.

    b. Transients caused by a fault in a system component requiring its isolation from the system.

    c. Transients due to loss of load or power.

    d. Abnormal incidents not resulting in a forced outage.

    e. Forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition is included in the design specifications for each component.

3. Emergency Conditions (Infrequent Incidents) - Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity results as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events does not cause more than 25 stress cycles having an S value greater than that for 10^6 cycles from the applicable fatigue design curves of the ASME Code Section III.

4. Faulted Conditions (Limiting Faults) - Those combinations of conditions associated with extremely low probability postulated events whose consequences are such that the integrity and operability of the plant may be impaired to the extent that consideration of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

5. Testing Conditions - Testing conditions are those pressure overload tests including hydrostatic tests, pneumatic tests, and leak tests specified. Other types of tests are classified under normal, upset, emergency or faulted conditions.

To provide the necessary high degree of integrity for the components in the RCS the transient conditions selected for component fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. The transients provided are representative of operating conditions which prudently should be considered to occur during plant operation and are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients analyzed may be regarded as a conservative representation of transients which, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application in accordance with the requirements of ASME Section III.

The MNPS-3 documents that provide detailed descriptions of design transients used for fatigue evaluation of RCP and associated Class 1 branch piping are listed in Table 3.9B–1. The applicable design transients for RPV internals are described in Section 3.9N.1.

3.9B.1.2 Computer Programs Used in Analysis

Lists of computer programs that are used in the design of Seismic Category I components and piping systems within the BOP scope are provided in Appendix 3A. Also included in Appendix 3A are brief descriptions of each program, the extent of its application, and program verifications which demonstrate the applicability and validity of each program.

3.9B.1.3 Experimental Stress Analysis

No experimental stress analysis methods are employed. Analytical methods for the design of BOP equipment, components, and piping systems are used exclusively.

3.9B.1.4 Consideration for the Evaluation of the Faulted Conditions

3.9B.1.4.1 Loading Conditions

The structural stress analysis performed on the reactor coolant loop piping and other Seismic Category I ASME Code Class 1, 2, and 3 piping consider the loadings and load combinations specified in Tables 3.9B–10, 3.9B–11, and 3.9B–12. The stress limits utilized for the faulted plant condition are also outlined in Tables 3.9B–10 and 3.9B–11.

The loading conditions and associated stress limits defined in the above tables are applicable for elastic analysis only. Inelastic analysis is not generally used in the qualification of piping and piping components. Where inelastic analysis is utilized in qualifying the piping and/or components, the MNPS-3 documents containing details of such analysis are identified in Table 3.9B–2. The procedure used for the modeling and analytical methods for evaluating the reactor coolant loop and supports for faulted loading conditions are described in Section 3.9B.1.4.2. The procedure used for modeling of remaining Seismic Category I ASME

Code Class 1, 2, and 3 piping systems and analytical methods employed for pipe stress analysis are provided in Section 3.7B.3.2. The analysis, thus performed, complies with ASME Section III Subsections NB, NC, ND (1971 through Summer 1973 addenda) and Appendix F.

3.9B.1.4.2 Evaluation of Reactor Coolant Loop and Supports for Faulted Loading Condition

The faulted loading condition of reactor coolant loop and supports considers loading due to:

1. Internal pressure
2. Weight
3. Safe shutdown earthquake
4. Loss-of-coolant accident (pipe break)
5. Transients

Internal Pressure

Internal pressure is identified as faulted condition pressure in accordance with ASME Section III requirements.

Weight

Weight consists of the weight of the piping system, insulation (if any), and contained fluid during normal operating conditions.

Seismic

The earthquake loads are part of the mechanical loading conditions specified in the design specification for piping systems. Mechanical stresses are evaluated for SSE (safe shutdown earthquake) condition consistent with ASME Section III requirements for faulted condition loading evaluation.

Loss of Coolant Accident

Mechanical loads are developed in the broken and unbroken reactor coolant loops and in the reactor vessel as a result of transient flow and pressure fluctuations following a postulated pipe break in one of the reactor coolant loops. Structural consideration of dynamic effects of postulated pipe breaks requires postulation of a finite number of break locations. Postulated pipe break locations are given in Section 3.6.

Time history dynamic analysis is performed for these postulated break cases. Hydraulic models are used to generate time-dependent hydraulic forcing functions used in the analysis of the reactor coolant system for each break case. For a further description of the hydraulic forcing functions, refer to Section 3.6.

Transients

Asymmetric pressurization, resulting from the postulated pipe breaks, of the reactor cavity, steam generator, or pressurizer cubicles also imposes unbalanced mechanical loads on the RCS piping and equipment. Pressure time histories are developed based on release rates of energy and fluid mass from the break, which are compatible with the hydraulic analysis cited above. The mechanical response to asymmetric pressure loading is calculated separately and then combined with the response to hydraulic forces.

Additional localized loadings due to direct impingement of a fluid jet from the postulated pipe break are also combined. Where the jet target is a support, the effect is assumed to be limited to the support and its immediate connection joints. Where the target is a major RCS component, the jet force in excess of asymmetric pressure loadings is included in the RCS dynamic analysis (Section 6.2.1).

3.9B.1.4.3 Reactor Coolant Loop Models and Methods

The analytical methods used in obtaining the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the response spectra method or time history method for seismic dynamic analysis, and time history direct integration method for the loss-of-coolant accident dynamic analysis.

The integrated reactor coolant loop/supports system model is the basic system model used to compute loadings on components, component supports, and piping. The system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, the stiffness of supports, the stiffness of auxiliary line piping which affects the system and the stiffness of piping restraints. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

Static

The reactor coolant loop/supports system model, constructed for the NUPIPE-SW computer program, is represented by an ordered set of data which numerically describes the physical system. Figure 3.9B–1 shows an isometric line schematic of this mathematical model for one of four reactor coolant loops. The steam generator and reactor coolant pump vertical and lateral support members are described in Section 5.4.14.

The spatial geometric description of the reactor coolant loop model is based upon the reactor coolant loop piping layout and equipment drawings. The node point coordinates and incremental lengths of the members are determined from these drawings. Geometrical properties of the piping and elbows along with the modulus of elasticity E, the coefficient of thermal expansion, the average temperature change from ambient temperature T, and the weight per unit length are specified for each element. The primary equipment supports are represented by stiffness matrices which define restraint characteristics of the supports. Due to the symmetry of the static loadings, the reactor pressure vessel nozzles are represented by a fixed boundary in the system mathematical model. The thermal growth of the reactor vessel is considered in the construction of the model.

The model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained. These loads are incorporated into the overall load vector.

After all the sections have been defined in this manner, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system.

The static solutions for deadweight and thermal loading conditions are obtained by using the NUPIPE-SW computer program.

Seismic

The model used in the static analysis is modified for the dynamic analysis by including the mass characteristics of the piping and equipment. The effect of the equipment motion on the reactor coolant loop/supports system is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model.

The steam generator is typically represented by five discrete masses. The lower mass is located near the intersection of the centerlines of the inlet and outlet nozzles of the steam generator. A middle mass is located near the center of the steam generator and a top mass is located at the outlet of the steam generator. The other two intermediate masses are located at the preheater section and steam separator section, respectively.

The reactor coolant pump is typically represented by a two-discrete-mass model. The lower mass is located at the intersection of the centerlines of the pump suction and discharge nozzles. The upper mass is located near the center of gravity of the motor.

The component upper and lower lateral supports are passive during plant heatup, cooldown, and normal plant operating conditions. However, these restraints become active under the rapid motions of the reactor coolant loop components that occur from the dynamic loadings. Component supports are represented by stiffness matrices and/or individual tension or compression spring members in the dynamic model.

The solution for the seismic disturbance employs the response spectra or the time history method. Both methods employ the lumped mass technique, linear elastic properties, and the principle of modal superposition. The floor response spectra or the base-mat motion histories are applied along both horizontal axes and the vertical axis simultaneously.

From the mathematical description of the system, the overall stiffness matrix [K] is developed from the individual element stiffness matrices using the transfer matrix method. After deleting the rows and columns representing rigid restraints, the stiffness matrix is revised to obtain a reduced stiffness matrix [K_R] associated with mass degrees of freedom only. From the mass matrix and the reduced stiffness matrix, the natural frequencies and the normal modes are determined.

In the response spectra method, the modal participation factor matrix is computed and combined with the appropriate response spectra value to give the modal amplitude for each mode. The total modal amplitude is obtained by taking the square root of the sum of the squares of the contributions for each direction. The modal amplitudes are then converted to displacements in the global coordinate system and applied to the corresponding mass point. From these data the forces, moments, deflections, rotations, support reactions and piping stresses are calculated for all significant modes. The total seismic response is computed by combining the contributions of the significant modes by using the methods described in Section 3.7.

For the time history method, the seismic model described above is expanded in scope to include all four reactor coolant loops, the reactor vessel and its support system, and the containment-building- model described in Section 3.8.1 with the exception of the base-mat and soil springs. The resulting model is coded for the STARDYNE (Appendix 3A2.5) program system.

Figure 3.9B–2 shows an isometric sketch of a typical loop of the STARDYNE model. The center section model is shown schematically in Figure 3.9B–3. The plan view of the latter figure indicates the points of attachment of the four individual loop submodels to the reactor vessel. In addition, all embedment points of steam generator and pump supports are connected to the building model at the mass point of nearest elevation. This connectivity is shown in the computer plots of the complete STARDYNE model (Figure 3.9B–4).

Prior to extraction of the natural frequencies and mode shapes, a mass condensation is performed to reduce the number of dynamic degrees-of-freedom (D-O-F) to 350. The other approximately 3,000 D-O-F are recovered by eigenvector expansion.

Loss-of-Coolant Accident

The mathematical model used in the time history seismic analyses is modified for the loss-of-coolant accident analyses. To represent the severance of the reactor coolant loop piping at a postulated guillotine break location, two distinct nodes, each containing six dynamic degrees of freedom and located on each side of the break, are included in the mathematical model.

When no nonlinear elements are active, i.e., for a longitudinal split or severance of the SI, RHR, PS, MS, or FW lines, the dynamic structural solution for the loss-of-coolant accident is obtained by using the STARDYNE subprogram DYNRE, as in the time-history seismic analysis. The natural frequencies and eigenvectors are determined from each broken model with the STARDYNE subprogram STAR/HQR.

When elements of the system can be represented as single acting members (tension or compression members), they are considered as nonlinear elements, which are represented mathematically by the combination of a gap, a spring, and a viscous damper. The force in each nonlinear element is treated as an externally applied force in an iterative solution technique which converges when the force and displacement are compatible. Multiple nonlinear elements can be applied at the same node, if necessary, or at several nodes. The time-history solution is performed in the STARDYNE subprogram DYNRE. The input to this subprogram consists of the system masses and stiffnesses, applied forces, and nonlinear elements.

The transient applied forces are described in Section 3.6B.2.

To simulate the release of the strain energy in the pipe for a postulated guillotine break, the internal forces in the system at the break location due to the initial steady state hydraulic forces, thermal forces, and weight forces, are determined by a STAR static analysis. The release of the strain energy is accounted for by applying the negative of these internal forces as a step function loading concurrent with the hydraulic forces. The initial conditions are equal to zero because the solution is for the transient problem (the dynamic response of the system from the static equilibrium position). To be consistent with this analysis scheme, the hydraulic forces are adjusted so as to eliminate the initial value which corresponds to the initial static loading condition.

The loss of coolant accident displacements of the reactor vessel are applied in time history form as another input to the dynamic analysis of the reactor coolant loop. The loss of coolant accident analysis of the reactor vessel includes all the forces acting on the vessel including internals reactions, cavity pressure loads, and loop mechanical loads. The reactor vessel analysis is described in Section 3.9N.1.4.3.

The time-history displacement response is used in computing support loads and in performing stress evaluation of the reactor coolant loop piping. The support loads [F] are computed by multiplying the support stiffness matrix [K] and the displacement vector [X] at the support point. The support loads are used in the evaluation of the supports.

The time-history displacements of the primary loop piping are used to determine the internal forces, deflections, and stresses at each end of the piping elements. For this calculation, the displacements are treated as imposed deflections on the reactor coolant piping. The results of this solution are used in the piping stress evaluation.

As part of the Stretch Power Uprate (SPU) evaluation, the LOCA analysis was revised taking into account the updated pipe break hydraulic forcing functions and LOCA RPV motions. This revised LOCA analysis was performed using the NUPIPE-SWPC computer program and the leak-before-break (LBB) criteria for primary reactor coolant loop pipe breaks.

3.9B.1.4.4 Primary Component Supports Models and Methods

The static and dynamic structural analyses employ the matrix method and normal mode theory for the solution of lumped-parameter, multimass structural models, or time-history integration method. The equipment support structure models are dual-purpose since they are required to quantitatively represent the elastic restraints which the supports impose upon the loop, and to evaluate the individual support member stresses due to the forces imposed upon the supports by the loop.

Models for the STARDYNE computer program (Appendix 3A) are constructed for the steam generator lower, steam generator upper lateral, reactor coolant pump lower and pressurizer supports. The reactor vessel support is modeled using the STRUDL computer program. Structure geometry, topology and member properties are used in the modeling.

A description of the supports is found in Section 5.4.14. Detailed models are developed using beam elements and plate elements, where applicable.

The respective computer programs are used with these models to obtain support stiffness matrices and member influence coefficients for the steam generator, reactor coolant pump, pressurizer, and reactor vessel supports. Unit force along and unit moment about each coordinate axis are applied to the models at the equipment vertical centerline joint. Stiffness analyses are performed for each unit load for each model.

Joint displacements for applied unit loads are formulated into flexibility matrices. These are inverted to obtain support stiffness matrices which are included in the reactor coolant loop model.

Loads acting on the supports are obtained from the reactor coolant loop analysis. For each support analyzed, the following is performed:

a. Combine the various types of support plane loads to obtain operating condition loads (Normal, Upset, Emergency or Faulted).

b. Multiply member influence coefficients by operating condition loads to obtain all member internal forces and moments.

c. Solve appropriate stress or interaction equations for the specified operating condition. Maximum normal stress, shear stress, and combined load interaction equation values are printed. ASME Boiler and Pressure Vessel Code Section III, Subsection NF, stress and interaction equations are used and results are compared with limits for the operating condition specified.

The reactor vessel support structure is analyzed for all loading conditions using a finite element model. Vertical and horizontal forces delivered to the support structures from the reactor vessel shoe are applied to the structure, and element stresses and concrete forces are obtained.

3.9B.1.4.5 Equipment and Components

The elastic analysis techniques described in Section 3.7B.3.1.1 are utilized in the qualification of Seismic Category I ASME code and noncode equipment within BOP scope. Stress limits utilized for the faulted plant condition are as outlined in Section 3.9B.3.1. The design conditions and stress limits defined are applicable for an elastic system (and equipment) analysis. Inelastic analysis is not employed. If, and when, inelastic analysis is contemplated, detailed design bases, demonstrating maintenance of function and/or structural integrity, will be established prior to implementation.

3.9B.2 DYNAMIC TESTING AND ANALYSIS

3.9B.2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

A preoperational vibration, thermal expansion (in discrete temperature step increments), and dynamic effects testing program is conducted on:

1. ASME Code Class 1, 2, and 3 piping systems.

2. High energy piping systems inside Seismic Category I structures.

3. High energy portions of systems whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable level.

4. Seismic Category I portions of moderate-energy piping systems located outside containment. The purpose of the tests is to confirm that these piping systems, restraints, components, and supports have been designed adequately to withstand the flow-induced dynamic loadings under operational transient and steady state conditions anticipated during service and to confirm that normal thermal motion is not restrained.

A list of the systems and the types of tests being conducted is contained in Table 3.9B–3. The different flow modes of operation and transients to which each system was subjected during the tests are contained in Chapter 14. The test titles, test prerequisites, test objectives, and summary of testing are also described in Chapter 14. For each system defined in items 1 through 4, all flow modes of operation that the systems are subjected to during the tests are visually observed where accessible. In addition, systems that were stress analyzed for fluid flow instabilities have instrumented measurements at selected locations for the specific flow modes analyzed. The measured results were compared to the analytically predicted values. Instrumented measurements are conducted (as needed) for all other systems and conditions. For ASME Code Class 1, 2, and 3 piping systems, design and supervision of the tests, definition of acceptance criteria, evaluations of test results, and the making of any changes in the piping system necessary to ensure that the piping is adequately designed and supported, were performed as required by Section III of the ASME code.

Observed vibrations which from visual examination appeared to be excessive in the opinion of experienced engineers who supervised, conducted, and witnessed the various tests, resulted in either:

1. An instrumented test program is conducted and the system re-analyzed (or compared to existing analysis) to demonstrate that the observed levels do not cause ASME code stress and fatigue limits to be exceeded.

2. The cause of vibration is eliminated.

3. A corrective support system is designed and installed and the effect of the modification incorporated in the Pipe Stress Analysis.

In addition to the above, thermal expansion effects of piping are observed during the testing. Locations for monitoring thermal expansion are chosen based on:

1. Expected large movements

2. Areas with tight clearances

3. Snubber locations to verify correct and unrestricted motion

3.9B.2.2 Seismic Qualification Testing of Safety Related Mechanical Equipment

The methods and procedures used in the design and qualification of Seismic Category I mechanical equipment within balance-of-plant (BOP) scope are outlined in Sections 3.9B.3.1.1, 3.9B.3, and 3.10B.1. Loading combinations include operating as well as earthquake loading for qualification by testing and/or analytical methods.

Safety related mechanical equipment (Seismic Category I), not covered by the ASME Boiler and Pressure Vessel Code, are seismically qualified in accordance with the procedures of Section 3.7B.3.1.1. Non-ASME Code equipment typically include diesel generators, fans, coolers, and emergency ventilation equipment. Cranes are seismically qualified in accordance with criteria that preclude the possibility of the crane being dislodged by a seismic disturbance.

Except as noted elsewhere in the FSAR, if codes are used in the design of a component, the guidelines generally require the addition of operating loads to the operating base earthquake (OBE) (1/2 Safe Shutdown Earthquake (SSE)) load with no increase in code allowable stress. If no codes are used, the stress level under the combined loading is limited to 75 percent of the minimum yield strength of the material per the ASTM specification. The general criteria for analysis of the SSE, pipe rupture (if applicable), and operating loads require that deformation of components be allowed only with no loss of safety function. Stresses under combined loadings are generally limited to the smaller of 100 percent of the minimum yield strength, or 70 percent of the minimum ultimate tensile strength, of the material (at temperature) per the ASTM, or equivalent specification for the material.

3.9B.3 ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

3.9B.3.1 Loading Combinations, Design Transients, and Stress Limits

3.9B.3.1.1 ASME III Class 1 Components

Components and systems which are Seismic Category I and which are within the jurisdiction of ASME Section III, Subsection NB (Class 1), have the following combinations and categorizations of load with relevant stress limits in accordance with NB-3000:

1. Design condition - Design pressure and temperature; design mechanical loads.

2. Normal condition - Normal operating loads (deadweight, pressure, thermal, etc.).

3. Upset condition - Upset plant condition loads plus the operating basis earthquake (OBE).

4. Emergency condition - Emergency plant condition loads.

5. Faulted condition - Dynamic system loads associated with the faulted plant condition plus the safe shutdown earthquake (SSE).

6. Testing condition - Hydrostatic tests, pneumatic tests, leak tests, as applicable.

For the conditions specified, the allowable stress limits defined in Tables 3.9B–5 and 3.9B–10 are applicable to stress results obtained by elastic analysis techniques and are compared with Regulatory Guide 1.48 in Table 3.9B–6. The analysis techniques described in Section 3.7B.3.1 are utilized in implementing these criteria. Design transients included in the above category are defined in Section 3.9B.1.

The general extent of compliance with Regulatory Guide 1.48 is discussed in Section 1.8.

3.9B.3.1.2 ASME III Class 2 and 3 Components

Tables 3.9B–7 and 3.9B–11 provide loading conditions and stress limits for ASME Code Class 2 and 3 components of Seismic Category I fluid systems which are constructed in accordance with ASME III Subsections NC and ND. The conditions generally relate to ASME Section III, Code Class 1 requirements, and include combinations as follows:

1. Design Condition I - includes the specified design loads (temperature, pressure, etc.) plus the OBE loads.

2. Design Condition II - includes the specified design loads (as above) plus SSE loads, or pipe rupture loads (if applicable).

The design loading combinations are analogous to either the Code Class 1 normal or upset conditions for Design Condition I, or to the faulted (or emergency, if applicable) condition for Design Condition II. The stress limits for these design conditions are presented in Tables 3.9B–7 and 3.9B–11. Since design temperature and pressure exceed those associated with upset, emergency, and faulted conditions, satisfaction of primary stress limits is assured.

These requirements are intended to be consistent with the present code format and philosophy. When implemented, they were a supplement to the requirements of ASME III, Subsection NC and ND.

The stress limits and design conditions presented in Table 3.9B–7 are intended to ensure that no gross deformation of the component occurs. Table 3.9B–8 compares Table 3.9B–7 with Regulatory Guide 1.48. These limits are applicable for an elastic system (and component) analysis. Inelastic deformation is not performed on any ASME Code Class 2 and 3 Components. If, and when, inelastic analysis is contemplated, detailed design bases demonstrating maintenance of either function and/or structural integrity will be proposed prior to implementation. The analysis techniques of Section 3.7B.3.1 are utilized in implementing these criteria.

The general extent of compliance with Regulatory Guide 1.48 is discussed in Section 1.8.

3.9B.3.2 Pump and Valve Operability Assurance

Pumps and valves installed in Seismic Category I piping systems are designed in accordance with the requirements of ASME III, NB, NC, and ND. Inactive pumps and valves are designed for the loading combinations of Sections 3.9B.3.1 and 3.9B.3.2, and for the stress limits indicated in Table 3.9B–7.

Active components are those that must perform a mechanical motion during the course of accomplishing its safety function.

Inactive components are those for which mechanical movement does not occur in order for the component to accomplish its intended safety function.

Operability of active pumps and valves is assured by satisfying the requirements of various programs. Safety related valves are qualified by prototype testing and analysis, and safety related active pumps by analysis with suitable stress limits and nozzle loads. The content of these programs is detailed below.

3.9B.3.2.1 Pump Operating Program

Active pumps are qualified for operability by being subjected to rigid tests both prior to and after installation in the plant. The in-shop tests include:

1. Hydrostatic tests to ASME Section III requirements.

2. Seal leakage tests at the same pressure used in the hydrostatic tests.

3. 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.

Also monitored during these operation tests are bearing temperatures and vibration levels, which are shown to be below appropriate limits specified to the manufacturer for design of each active pump.

After the pump is installed in the plant, it undergoes the cold hydro tests, hot functional tests, and the required periodic inservice inspection and operation as applicable. These tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, the safety related active pumps are qualified for operability during an SSE condition by assuring that:

1. The pump is not damaged during the seismic event.

2. The pump continues operating when subjected to the SSE loads.

The pump manufacturer is required to show that the pump operates normally when subjected to the maximum applicable amplified seismic (floor) accelerations, attached piping nozzle loads, and dynamic system loads associated with the faulted operating condition. Analysis and/or testing procedures are utilized in accordance with those outlined in Section 3.7B.3.1.1. Natural frequency calculations are performed in order to determine maximum seismic accelerations based on applicable amplified (floor) response spectra.

To avoid damage during the faulted condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited as indicated in Table 3.9B–7. The average membrane stress (P^m) for the faulted condition loads is maintained at 1.2 S, or approximately 0.75 S_y (S_y = yield stress) and the maximum stress in local fibers (P_m + bending stress P_b) is limited to 1.8 S, or approximately 1.1 S_y. In addition, the pump stresses caused by the maximum seismic nozzle loads are limited to the stresses outlined in Table 3.9B–7. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur. A static shaft deflection analysis of the rotor is performed with horizontal and vertical accelerations based on floor response levels. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The nature of seismic disturbances dictates that the maximum contact (if it occurs) is of short duration.

Class 1 pumps are designed/analyzed according to the rules of ASME Section III, Subsection NB 3400.

Performance of these analyses with the conservative loads stated and, with the restrictive stress limits of Table 3.9B–5 as allowables, assures that critical parts of the pump are not damaged during the short duration of the faulted condition and that the reliability of the pump for post-faulted condition operation is not impaired by the seismic event.

In addition to the post-faulted condition operation, it is necessary to assure that the pump functions throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized (i.e., no rotation). Typically, the rotor can be seized 5 full seconds before a circuit breaker trips to prevent damage to the motor. However, the high rotary inertia in the operating pump rotor and the nature of the random, short duration loading characteristics of the seismic event prevent the rotor from becoming seized. In actuality, the seismic loadings cause only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore, the pump does not shut down during the SSE and operates at the design despite the SSE loads.

To complete the seismic qualification procedures, the pump motor is independently qualified for operation during the maximum seismic event. Any auxiliary equipment which is vital to the operation of the pump or pump motor, and which is not qualified for operation during the pump analysis or motor qualifications, is also separately qualified for operation at the accelerations occurring at its mounting. The pump motor and vital auxiliary equipment are qualified by meeting the requirements of IEEE Std 344-1975. If the testing option is chosen, sinusoidal or sine-beat testing is justified by satisfying one or more of the following requirements to demonstrate that the multi-frequency response is negligible or the input is of sufficient magnitude to conservatively account for this effect:

1. The equipment response is basically due to one mode.

2. The sinusoidal or sine-beat response spectra envelop the floor response spectra in the region of significant response.

3. The floor response spectra consist of one dominant mode and have a peak at this frequency.

In general, the degree of coupling in the equipment determines if a single or multi-axis test is required. Multi-axis testing is required if there is considerable cross coupling. If coupling is very light, then single axis testing is justified. If the degree of coupling can be determined, then single axis testing can be used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

From previous arguments, the safety related pump/motor assemblies are not damaged and continue operating under SSE loading and, therefore, perform their intended functions. These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps. The functional ability of active pumps after a faulted condition is assured, since only normal operating loads and steady state nozzle loads exist. Since it is demonstrated that the pumps are not damaged during the faulted condition, the post-faulted condition operating loads are identical to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9B.3.2.2 Valve Operability Program

Safety related active valves must perform their mechanical motion during the course of performing their safety function. Assurance that these valves will operate during a seismic event must be supplied. Qualification tests accompanied by analyses are conducted for all active valve assemblies.

Valves without significant extended structure are proven seismically adequate by analysis of piping seismic adequacy. For valves with operators having significant extended structures, and if these structures are essential to maintaining pressure integrity, analysis is performed based upon static forces resulting from equivalent earthquake accelerations acting at the centers of gravity of the extended masses. For active valves, this requirement for analysis is extended to the mechanical (nonpressure boundary) components of valve top-works to ensure operability.

The safety related valves are subjected to a series of stringent tests prior to service and during plant life. Prior to installation, the following tests are performed:

1. Shell hydrostatic test to ASME III requirements.

2. Backseat and main seat leakage tests.

3. Disc hydrostatic test.

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

5. Operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident environmental simulation, etc.) according to IEEE Std 382-1972.

Cold hydro qualification tests, hot functional qualification tests, periodic inservice inspections, and periodic inservice operation are performed in situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant. The valves are designed using either stress analyses or the pressure containing minimum wall thickness requirements. On all active valves, an analysis of the extended structure is also performed for static equivalent seismic SSE loads supplied at the centers of gravity of the extended structure. The maximum stress limits allowed in these analyses show structural integrity. The limits that are used for Class 2 and 3 active valves are shown in Table 3.9B–7. Class 1 valves are designed/ analyzed according to the rules of ASME Section III, Section NB-3500.

In addition to these tests and analyses, representative valves of each design type are tested for verification of operability during a simulated seismic event by demonstrating operational capabilities within the specified limits. The testing procedures are described below.

The valve is mounted in a manner which conservatively represents a typical valve installation. The valve includes the operator and all appurtenances normally attached to the valve in service.

The operability of the valve during an SSE is demonstrated by satisfying the following criteria:

1. Active valves are designed to have a first natural frequency greater than 33 Hz. This may be shown by suitable test or analysis.

2. The actuator and yoke of the valve system are statically loaded an amount greater than that determined by an analysis, as representing SSE accelerations applied at the center of gravity of the operator alone in the direction of the weakest axis of the yoke. The design pressure of the valve is simultaneously applied to the valve during the static deflection tests.

3. The valve is then operated while in the deflected position (i.e., from the normal operating mode to the faulted mode). The valve must perform its safety related function within the specified operating time limits.

4. Motor operators and other electrical appurtenances necessary for operation are qualified as operable during an SSE by appropriate IEEE Seismic Qualification Standards, such as IEEE Std 382-1972 and IEEE Std 344-1975, prior to their installation on the valve.

The accelerations used for static valve qualification are 3.0 g horizontal and 2.0 g vertical applied with the valve in its proper orientation. Where it is necessary to allow for random orientation of a valve, a 3.0 g horizontal and a 3.0 g vertical are applied. The piping designer maintains the motor operator accelerations to these levels with an adequate margin of safety.

If the frequency of the valve, by test or analysis, is less than 33 Hz, the valve system is analyzed to determine the equivalent acceleration applied during the static test. The analysis provides the amplification of the input acceleration considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted acceleration is then used in the static analysis and valve operability is assured by the methods outlined in steps (2) to (4) above, using the modified acceleration input.

The above testing program applies only to valves with overhanging structures (i.e., the motor operator). The testing is conducted on a representative number of valves. Valves from each of the primary safety related design types (e.g., motor-operated gate valve) are tested. Valve sizes which cover the range of sizes in service are qualified by the tests and the results are used to qualify all valves within the intermediate range of sizes. Stress and deformation analysis is used to support the extrapolation.

Valves which are safety related, but can be classified as not having an overhanging structure, such as check valves and safety-relief valves, are considered separately.

Check valves are characteristically simple in design and their operation is not affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact and there are no extended structures or masses whose motion could cause distortions which could restrict operation of the valve. The nozzle loads due to maximum seismic excitation do not affect the functional ability of the valve, since the valve disc is designed to be isolated from the casing wall. The clearance supplied by the design around the disc prevents the disc from becoming bound or restricted due to any casing distortions caused by nozzle loads. Therefore, the design of these valves is such that once the structural integrity of the valve is assured using standard design or analysis methods, the ability of the valve to operate is assured by the design features. In addition to these design considerations, the valve also undergoes the following tests and analysis:

1. Stress analysis including the SSE loads.

2. In-shop hydrostatic test.

3. In-shop seat leakage test.

4. Periodic in situ valve exercising and inspection to assure the functional ability of the valve.

Safety and relief valves are subjected to tests and analyses similar to check valves; stress and deformation analyses for SSE loads, in-shop hydrostatic and seat leakage tests, and periodic in situ valve inspection. In addition, a static load equivalent to the SSE is applied to the top of bonnet and the pressure is increased until the valve mechanism is activated. Successful actuation within design requirements assures its overpressurization safety capabilities during a seismic event.

Using the methods described, all the safety related valves in the system are qualified for operability during a seismic event. These methods conservatively simulate the seismic event and ensure that the active valves perform their safety related function when necessary.

Alternative valve operability testing, such as dynamic vibration testing, is allowed if it is shown to adequately assure the faulted condition functional ability of the valve system.

3.9B.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

The design criteria for all safety and relief valves are in accordance with the rules in Subarticles NB-3677 and NC-3677 of the ASME Boiler and Pressure Vessel Code, Section III (see Component Description for relief valves Letdown Line Downstream of Low Pressure Letdown Valves, Sealwater Return Line, and Letdown Reheat Heat Exchanger Section 9.3.4.2.5), and the rules of Code Case 1569, applicable to the classification of the piping component under investigation. For open relief systems, the design criteria and the analyses used to calculate maximum stresses and stress intensities are in accordance with Subarticles NB-3600 and NC-3600 of Section III. The maximum stresses are calculated based upon the full discharge loads, including the effects of the system dynamic response, and the system design internal pressure. Stresses are determined for all significant points in the piping system including the safety valve inlet pipe nozzle and the nozzle to shell juncture.

3.9B.3.3.1 Open Relief System

The total stead-state discharge thrust load for an open system discharge is expressed as the sum of the pressure and momentum forces as follows:

$$F = 144PA + [PAV^2]/g \qquad (3.9B\text{-}1)$$

where:

F = Total reaction force (lb)

A = Exit flow area (sq ft)

P = Exit pressure (psig)

V = Exit fluid velocity (fps)

P = 32.2 fps^2

To ensure consideration of the effects of the suddenly applied load at the junction of the valve nozzle and pipe run, a dynamic load factor is computed. The calculation of dynamic load factor is based on modeling the valve and nozzle as a single degree of freedom dynamic system. The lumped mass of this system corresponds to the weight of the valve and nozzle and is assumed to be at the valve center of gravity. The rotational degree of freedom of this system is considered to be in the direction that causes maximum bending stress in the nozzle at the junction of the nozzle and run-pipe. Rotational flexibility of the system is computed by a series combination of nozzle flexibility and local run-pipe flexibility (at the junction of the nozzle and run-pipe).

The rise time of the discharge force at the outlet of the safety valve elbow is assumed to be the minimum valve opening time, and the discharge force is assumed to rise linearly with time and remain thereafter constant value. The ratio of maximum dynamic rotations predicted by this single degree of freedom system to the corresponding static rotation caused by the steady state discharge force represents the dynamic load factor.

To ensure the consideration of the effects of the suddenly applied loads on the pipe system, a dynamic time-history analysis is performed on the piping system. The forcing function applied at the point of discharge is a linear force change from zero to the value of F that is determined in the above equation over a time period, t, that corresponds to the valve opening time which is provided by the valve manufacturer. After time t has been reached, the force remains at the value of F until the conclusion of the time-history integration. The lumped mass model that represents the piping system includes the safety/relief valves.

Where more than one valve is mounted on a common header, two cases are computed. In the first, full discharge of all valves is assumed to occur simultaneously. In the second, the forcing functions are applied to a combination of valves that yields the worst-load case. This worst-load case is first verified by trial through a series of static load cases.

3.9B.3.3.2 Closed Relief System

For relief valves discharging into a closed system, an analytical model of one-dimensional transient flow characteristics following the blow-off of the upstream safety/relief valve into the discharging piping system is established. The time-dependent pressure, temperature, density, velocity, and hence the momentum of the downstream pipe flow are then computed from this conservative hydrodynamic/thermodynamic flow model. The effects such as flow restrictions and frictional resistance are considered.

The unbalanced transient hydraulic forcing function acting on the piping system computed from the flow model is then used to determine the transient dynamic responses of the piping structural model. Adapting the lumped-mass method incorporated with the modal analysis of piping system, the time-history modal responses are computed. Computations of maximum stress intensities for ASME Code Class 1 piping or maximum stress levels for ASME Code Class 2 and 3 piping are based on the dynamic analysis of the system.

3.9B.3.4 Component Supports

Component supports which are Seismic Category I and which are within the jurisdiction of ASME Section III, Subsection NF, utilize applicable loading conditions outlined in Sections 3.9B.3.1.1 and 3.9B.3.1.2 for ASME Class 1, 2, and 3 components. Stress limits and loading combinations to be utilized for component support evaluation are as outlined in Tables 3.9B–9 and 3.9B–9A, respectively. The stress limits are used in conjunction with the applicable analytical procedures outlined in Section 3.7B.3 to assure integrity of supports and mounted components.

Component supports which do not come under the jurisdiction of ASME Section III, Subsection NF are summarized in Section 5.4.14.

The loads considered in the analysis of the supports are normal operating, seismic (OBE and SSE), and pipe rupture loads. The latter includes the effects of the following:

1. The blowdown forces in the primary loop.

2. Asymmetric pressurization of the reactor cavity and the SG and RCP cubicles.

3. Fluid jet impinging on the supports or components, as applicable.

Dynamic analyses were performed to determine the stresses in the component supports as well as the pipe rupture restraints. The results of the analysis are shown in the following tables.

1. Table 3.9B–15 gives the location of the postulated breaks in the primary loop system.

2. Tables 3.9B–16 and 3.9B–17 give the embedment loads of the SG and RCP.

3. Table 3.9B–18 gives the stresses in the component supports of the SG and RCP.

4. Table 3.9B–19 gives the stresses in the component support of the pressurizer.

5. Table 3.9B–20 gives the stresses in the component support of the pressurizer safety valve support.

6. Table 3.9B–21 gives the stresses in the primary members of the RPV support.

7. Table 3.9B–22 gives the stresses in the primary loop bumper sections.

FIGURE 3.9B–1 REACTOR COOLANT PPG

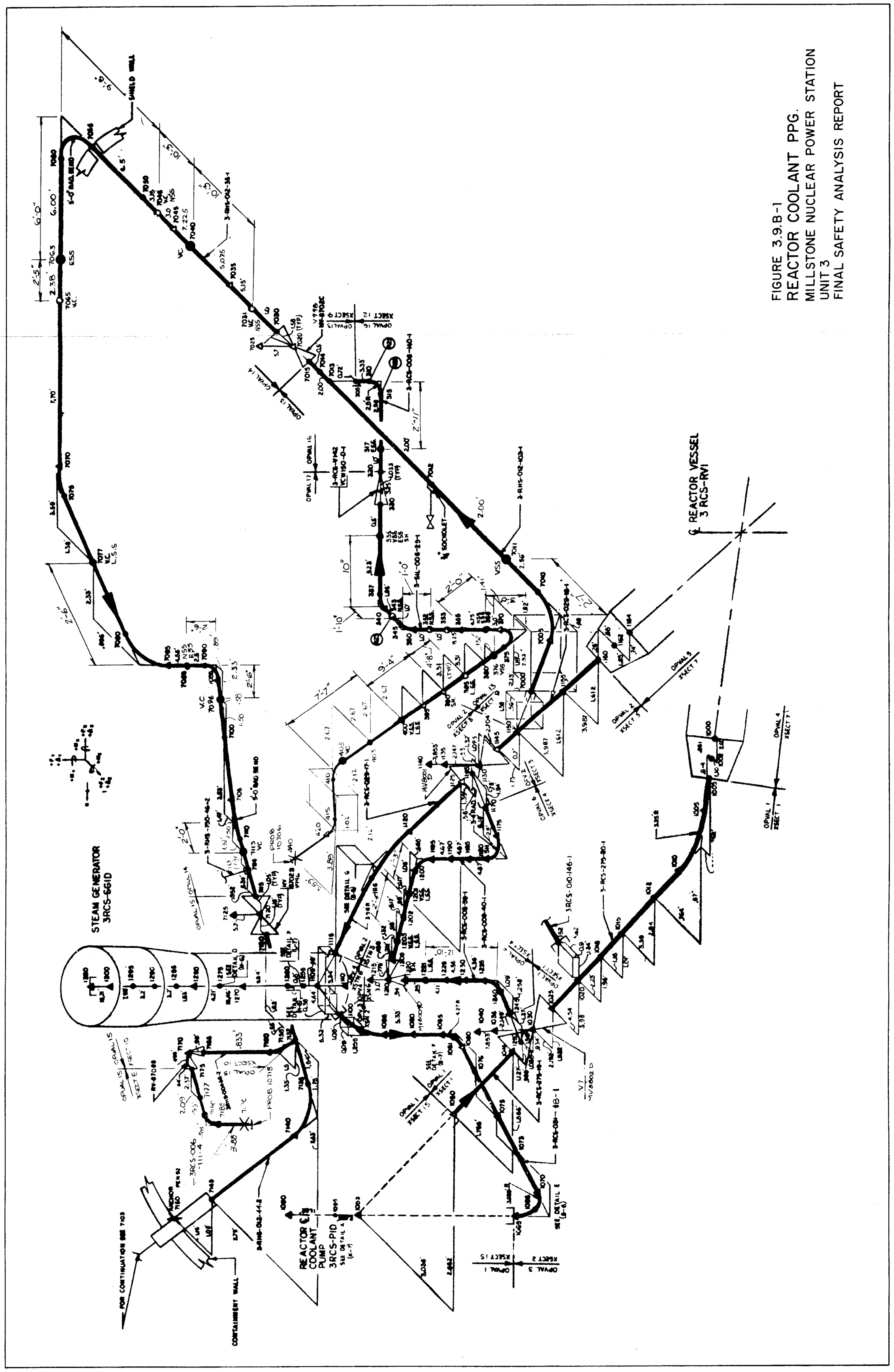


FIGURE 3.9B–2 DYNAMIC MODEL - TYPICAL RCS LOOP

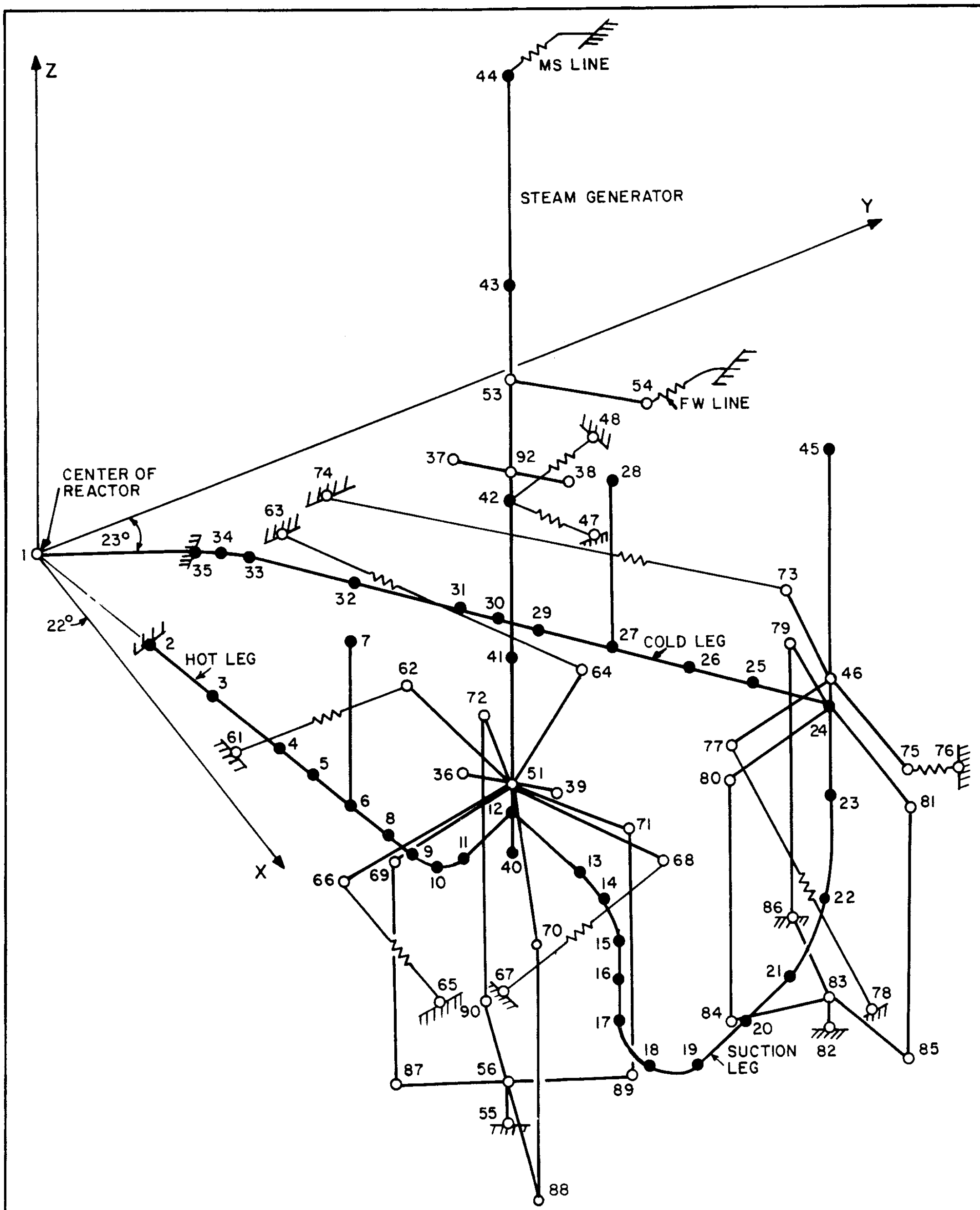


FIGURE 3.9B-2
DYNAMIC MODEL-TYPICAL RCS LOOP
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.9B–3 DYNAMIC MODEL - CENTER SECTION

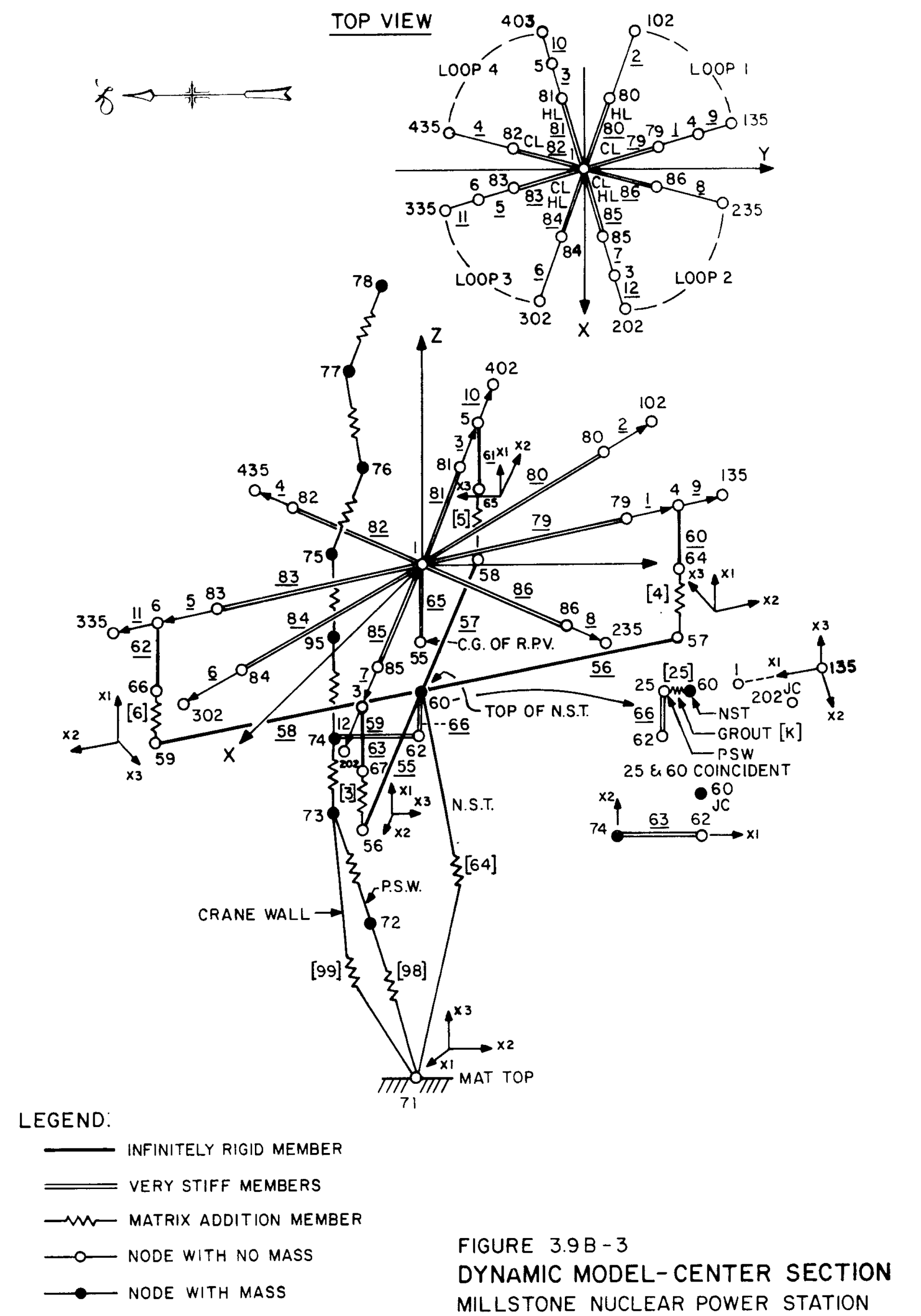


FIGURE 3.9B-3
DYNAMIC MODEL-CENTER SECTION
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.9B–4 X2 VIEW - ROTATED AXES

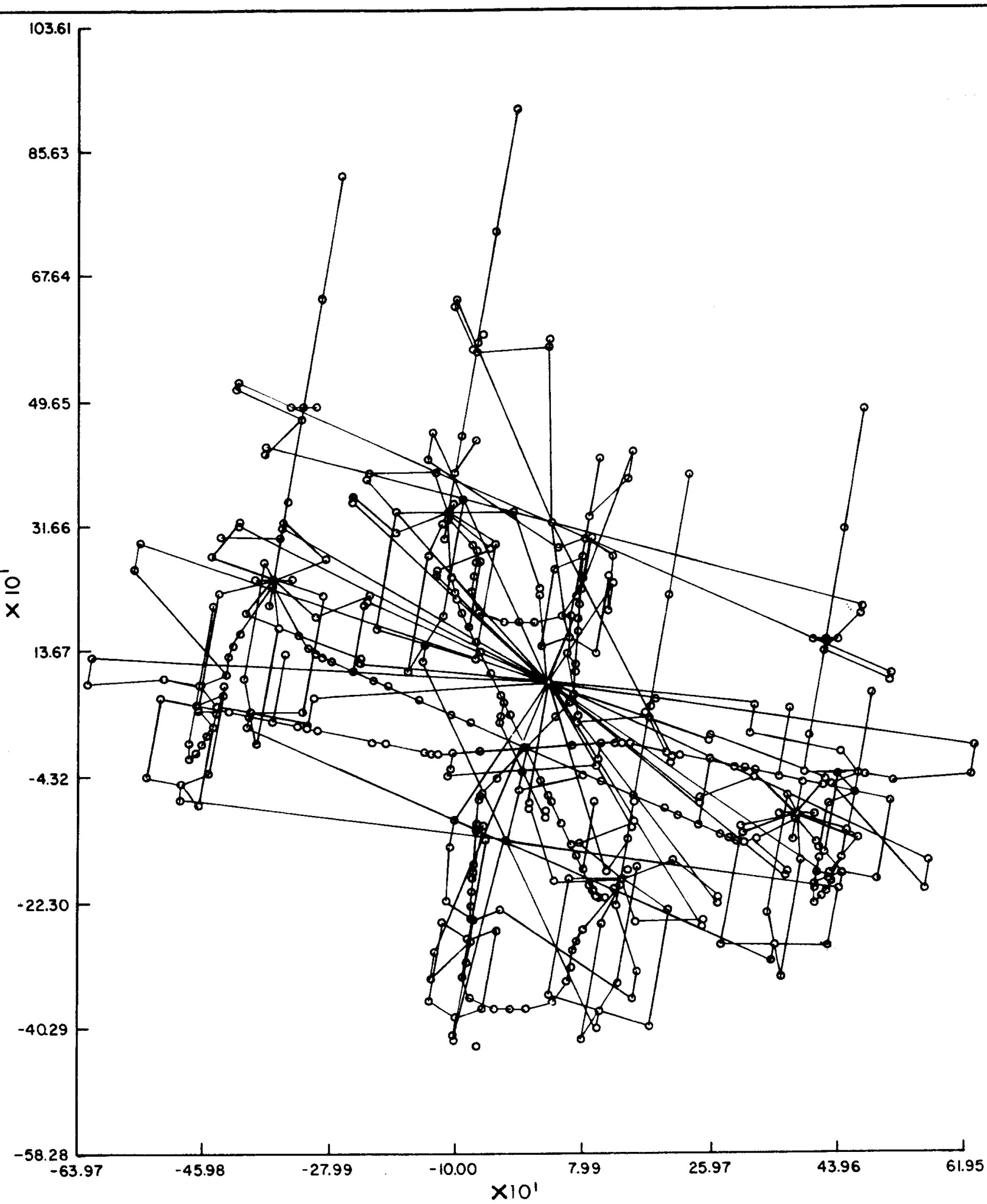


FIGURE 3.9B-4
X2 VIEW-ROTATED AXES
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.9B–5 SNUBBER ORIENTATION

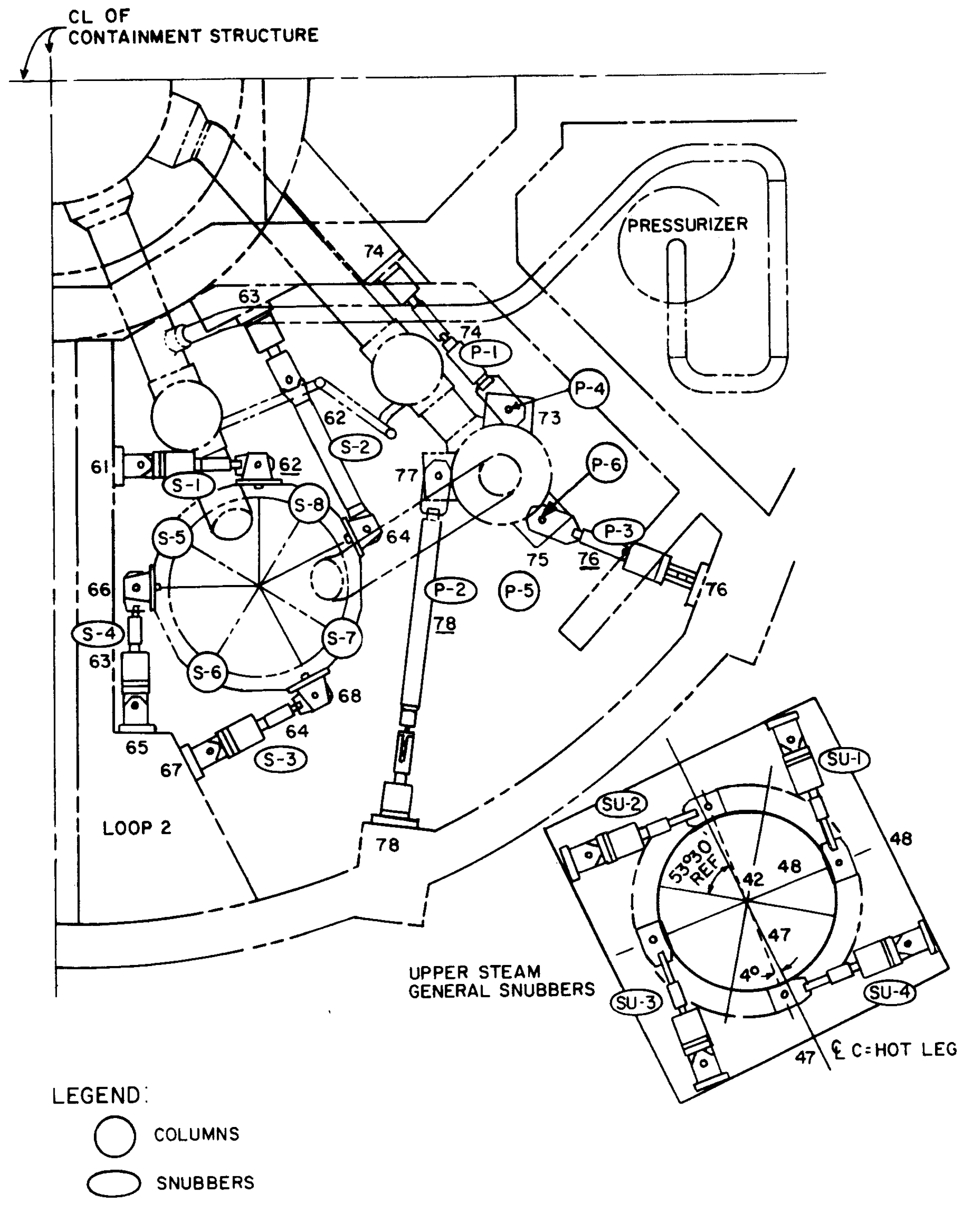


FIGURE 3.9B-5
SNUBBER ORIENTATION
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

3.9N MECHANICAL SYSTEMS AND COMPONENTS

3.9N.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9N.1.1 Design Transients

The following five operating conditions, as defined in Section III of the ASME B&PV Code, are considered in the design of the reactor coolant system (RCS) components and reactor internals.

1. Normal Conditions

   Any condition in the course of startup, operation in the design power range, hot standby and system shutdown, other than upset, emergency, faulted or testing conditions.

2. Upset Conditions (Incidents of Moderate Frequency)

   Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage.

3. Emergency Conditions (Infrequent Incidents)

   Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity results as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events does not cause more than 25 stress cycles having an S value greater than that for 10^6 cycles from the applicable fatigue design curves of the ASME Code Section III.

4. Faulted Conditions (Limiting Faults)

   Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that consideration of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

5. Testing Conditions

Testing conditions are those pressure overload tests including hydrostatic tests and pneumatic tests. Other types of tests are classified under normal, upset, emergency or faulted conditions as appropriate.

To provide the necessary high degree of integrity for the equipment in the RCS the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. To a large extent, these transients are based upon engineering judgment and experience, and are considered to be of such magnitude and/or frequency as to be significant in the component design and fatigue evaluation. The transients selected may be regarded as a conservative representation of transients which, used as a basis for component design and evaluation, provide confidence that the component is appropriate for its application over the design life of the plant.

The design transients and the number of cycles of each that are used for fatigue evaluations are shown in Table 3.9N–1. In accordance with ASME III, emergency and faulted conditions are not included in fatigue evaluations.

Normal Conditions

The following primary system transients are considered normal conditions:

1. Heatup and cooldown at 100°F per hour
2. Unit loading and unloading at 5 percent of full power/per minute
3. Step load increase and decrease of 10 percent of full power
4. Large step load decrease with steam dump
5. Steady state fluctuations
    a. Initial
    b. Random
6. Feedwater cycling at hot shutdown
7. Not Used.
8. Unit loading and unloading between 0 and 15 percent of full power
9. Boron concentration equalization
10. Refueling

11. Reduced temperature return to power

12. Reactor coolant pumps startup and shutdown

13. Turbine roll test

14. Primary side leak test

15. Secondary side leak test

16. Tube leakage test

17. Heaters out of service

1. Heatup and Cooldown at 100°F per Hour

The design heatup and cooldown cases are conservatively represented by continuous operations performed at a uniform temperature rate of 100°F per hour. (These operations can take place at lower rates approaching the minimum of 0°F per hour. The expected normal rates are 50°F per hour).

For these cases, the heatup occurs from ambient (assumed to be 120°F) to the no-load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will not be attained because of other limitations such as:

a. Material ductility considerations which limit temperature rates of change, as functions of plant pressure and temperature.

b. Slower initial heatup rates when using pump energy only.

c. Interruptions in the heatup and cooldown cycles due to such factors as pressurizer steam bubble formation, rod withdrawal, sampling, water chemistry and gas adjustments.

The number of such complete heatup and cooldown operations is specified as 200 each, for the 60 year plant design life.

2. Unit Loading and Unloading at 5 Percent of Full Power per Minute

The unit loading and unloading operations are conservatively represented by continuous and uniform ramp power change of 5 percent per minute between 15 percent load and full load. This load swing is the maximum possible consistent with operation under automatic reactor control. The reactor temperature will vary with load as prescribed by the reactor control system. The number of loading and unloading operations is defined as 13,200 during the 60 year design life of the plant.

3. Step Load Increase and Decrease of 10 Percent of Full Power

The ±10 percent step change in load demand results from disturbances in the electrical network into which the plant output is tied. The reactor control system is designed to restore plant equilibrium without reactor trip following a 10 percent step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15 percent and 100 percent of full load, the power range for automatic reactor control. In effect, during load change conditions, the reactor control system attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed setpoint at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the RCS average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With the load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature setpoint change is made as a function of turbine-generator load as determined by first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash which reduces the rate of pressure decrease. Subsequently the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient.

The number of each operation is specified at 2,000 times for the 60 year plant design life.

4. Large Step Load Decrease with Steam Dump

This transient applies to a step decrease in turbine load from full power, of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump that will prevent both reactor trip and lifting of steam generator and pressurizer safety valves. This plant is designed to accept a step decrease of 50 percent from full power, and the steam dump system provides the heat sink to accept the difference in allowable unloading rates between the turbine and the reactor coolant system.

The number of occurrences of this transient is specified at 200 times for the 60 year plant design life.

5. Steady State Fluctuations

It is assumed that reactor coolant pressure and temperature and pressure at any point in the system vary around the nominal (steady state) values. For design purposes two cases are considered:

a. Initial Fluctuations - These are due to control rod cycling during the first 20 full-power months of reactor operation. Temperature is assumed to vary ±3°F and pressure by ±25 psi, once during each 2 minute period. The total number of occurrences is limited to 1.5×10^5. The fluctuations are assumed to occur consecutively, and not simultaneously with the random fluctuations.

b. Random Fluctuations - Temperature is assumed to vary by ±0.5°F and pressure by ±6 psi, once every 6 minutes. With a 6 minute period, the total number of occurrences during the plant design life does not exceed 3.0×10^6.

6. Feedwater Cycling at Hot Shutdown

These transients can occur when the plant is at "no load" conditions, during which intermittent feeding of 32°F feedwater into the steam generators is assumed. Due to fluctuations arising from this mode of operation, the reactor coolant average temperature decreases to a lower value and then immediately begins to return to normal no-load temperature. This transient is assumed to occur 2,000 times over the life of the plant.

7. Not Used

8. Unit Loading and Unloading Between 0 and 15 Percent of Full Power

The unit loading and unloading cases between the zero and 15 percent power are represented by continuous and uniform ramp power changes, requiring 30 minutes for loading and 5 minutes for unloading. During loading, reactor coolant temperatures are increased from the no load value to the normal load program temperatures at the 15 percent power level. The reverse temperature change occurs during unloading.

Prior to loading, it is assumed that the plant is at hot shutdown conditions, with 32°F feedwater cycling. During the 2-hour period following the beginning of loading, the feedwater temperature increases from 32°F to 300°F due to steam dump and turbine startup heat input to the feedwater. Subsequent to unloading, feedwater heating is terminated, steam dump is reduced to residual heat removal requirements, and feedwater temperature decays from 300°F to 32°F.

The number of these loading and unloading transients is assumed to be 500 each during the 60 year plant design life.

9. Boron Concentration Equalization

Following any large change in boron concentration in the RCS, spray is initiated in order to equalize concentration between the loops and the pressurizer. This can be done by manually operating the pressurizer backup heaters, thus causing a pressure increase, which will initiate spray at a compensated pressurizer pressure of approximately 2,275 psia. The proportional sprays return the pressure to 2,250 psia and maintain this pressure by matching the heat input from the backup heater until the concentration is equalized. For design purposes, it is assumed that this operation is performed once after each load change in the design load follow cycle. The total number of load change occurrences is 26,400 over the 60 year design life.

10. Refueling

At the end of the plant cooldown, the temperature of the fluid in the RCS is less than 125°F. At this time, the vessel head is removed and the refueling canal is filled. This is done by pumping water from the refueling water storage tank, which is outside and conservatively assumed to be at 32°F, into the loops by means of the low head safety injection pumps. The refueling water flows directly into the reactor vessel via the accumulator connections and cold legs.

This operation is assumed to occur 80 times over the life of the plant.

11. Reduced Temperature Return to Power

The reduced temperature return to power operation is designed to improve the spinning reserve capabilities of the plant during load following operations without part length rods. The transient will normally begin at the ebb (50 percent) of a load follow cycle and will proceed at a rapid positive rate (typically 5 percent per minute) until the abilities of the control rods and the coolant temperature reduction (negative moderator coefficient) to supply reactivity are exhausted. At that point, further power increases are limited to approximately one percent per minute by the ability of the boron system to dilute the reactor coolant. The reduction in primary coolant temperature is limited by the protection system to about 20°F below the programmed value.

The reduced temperature return to power operation is not intended for daily use. It is designed to supply additional plant capabilities when required because of network fault or upset condition. Hence, this mode of operation is expected to occur 2,000 times in 60 years.

12. Reactor Coolant Pump (RCP) Startup and Shutdown

The reactor coolant pumps are started and stopped during routine operations such as RCS venting, plant heatup and cooldown, and in connection with recovery from certain transients such as loss of power. Other (undefined) circumstances may also require pump starting and stopping.

Of the spectrum of RCS pressure and temperature conditions under which these operations may occur, three conditions have been selected for defining transients:

a. Cold condition - 70°F and 400 psig

b. Pump restart condition - 100°F and 400 psig

c. Hot condition - 557°F and 2235 psig

For RCP starting and stopping operations, it is assumed that variations in RCS primary side temperature and in pressurizer pressure and temperature are negligible and that the steam generator secondary side is completely unaffected. The only significant variables are the primary system flow and the pressure changes resulting from the pump operations. Occurrences for the pump starting/stopping conditions are given in Table 3.9N–1.

13. Turbine Roll Test

This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power is used to heat the reactor coolant to operating temperature (no-load conditions) and the steam generated is used to perform a turbine roll test. However, the plant cooldown during this test exceeds the 100°F per hour design rate.

The number of such test cycles is specified at 20 times, to be performed at the beginning of plant operating life prior to reactor operation.

14. Primary Side Leakage Test

Subsequent to each time the primary system has been opened, a leakage test will be performed.

During this test the primary side pressure may be raised to a maximum of 2500 psia, in accordance with the system temperature limitations imposed y the reactor vessel material ductility requirements, while the system is checked for leaks. Normally, to prevent the pressurizer safety valves from lifting during the leak test, the primary system will be pressurized to approximately 2,235 psig, as measured at the pressurizer.

During this leakage test, the secondary side of the steam generator must be pressurized so that the pressure differential across the tube sheet does not exceed 1,600 psi. This is accomplished with the steam, feedwater, and blowdown lines closed off. For design purposes it is assumed that 200 cycles of this test will occur during the 60 year life of the plant.

15. Secondary Side Leakage Test

During the life of the plant, it may be necessary to check the secondary side of the steam generator (particularly, the manway closure) for leakage. For the design purposes, it is assumed that the steam generator secondary side is pressurized to just below its design pressure, to prevent the safety valves from lifting. In order not to exceed a secondary side to primary side pressure differential of 670 psi, the primary side must also be pressurized. The primary system must be above the minimum temperature imposed by reactor vessel material ductility requirements. It is assumed that this test is performed 80 times during the 60 year life of the plant.

16. Tube Leakage Test

During the life of the plant it may be necessary to check the steam generator for tube leakage and the tube to tube sheet leakage. This is done by visual inspection of the underside (channel head side) of the tube sheet for water leakage, with the secondary side pressurized. Tube leakage tests are performed during plant cold shutdowns.

For these tests the secondary side of the steam generator is pressurized with water, initially at a relatively low pressure, and the primary system remains depressurized. The underside of the tube sheet is examined visually for leaks. If any are observed, the secondary side is then depressurized and repairs made by tube plugging. The secondary side is then repressurized (to a higher pressure) and the underside of the tube sheet is again checked for leaks. This process is repeated until all the leaks are repaired. The maximum (final) secondary side test pressure reached is 840 psig.

The total number of tube leakage test cycles is defined as 800 during the 60 year life of the plant. Following is a breakdown of the anticipated number of occurrences at each secondary side test pressure:

Test Pressure (psig)	**Number of Occurrences**
200	400
400	200
600	120
840	80

Both the primary and secondary sides of the steam generators will be at ambient temperatures during these tests.

17. Heaters Out of Service

These transients occur when one or more feedwater heaters are taken out of service. During the period of time that the heaters are out of service, it is desirable to maintain the plant at full rated thermal load. To accomplish this, first the steam flow is reduced to the amount that will maintain the plant at full rated thermal load when the heater(s) is taken out of service. It takes approximately 10 minutes for plant conditions to reach a new steady state. Then the heater(s) is taken out of service.

The two cases considered here are one heater out of service and one bank of heaters out of service. For design purposes, it is assumed that each of these transients occurs 120 times over the life of the plant.

Upset Conditions

The following primary system transients are considered upset conditions:

1. Loss of load (without immediate reactor trip)

2. Loss of power

3. Partial loss of flow

4. Reactor trip from full power

5. Inadvertent reactor coolant system depressurization

6. Control rod drop

7. Inadvertent safety injection actuation

8. Operating basis earthquake

9. Excessive feedwater flow

10. RCS Cold Overpressurization

1. Loss of Load (without immediate reactor trip)

This transient applies to a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip and represents the most severe pressure transient on the RCS under upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the reactor protection system (RPS). Since redundant means of tripping the reactor are provided as part of the RPS, transients of this nature are not expected, but is included to ensure a conservative design.

The number of occurrences of this transient is specified at 80 times for the 60 year plant design life.

2. Loss of Power

This transient applies to a blackout situation involving the loss of offsite electrical power to the station, assumed to be operating initially at 100-percent power, followed by reactor and turbine trips. Under these circumstances, the reactor coolant pumps are deenergized and, following coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater, assumed to be at 32°F, from the auxiliary feedwater system operating from diesel generator power. Steam is removed for reactor cooldown through atmospheric relief valves provided for this purpose.

The number of occurrences of this transient is specified at 40 times for the 60 year plant design life.

3. Partial Loss of Flow

This transient applies to a partial loss of flow from full power, in which a reactor coolant pump is tripped out of service as the result of a loss of power to that pump. The consequences of such an event are a reactor trip, on low reactor coolant flow, followed by turbine trip and automatic opening of the steam dump system. Flow reversal occurs in the affected loop which causes reactor coolant at cold leg temperature to pass through the steam generator and be cooled still further. This cooler water then flows through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizable reduction in the hot leg coolant temperature of the affected loop.

The number of occurrences of this transient is specified at 80 times for the 60 year plant design life.

4. Reactor Trip From Full Power

A reactor trip from full power may occur from a variety of causes resulting in temperature and pressure transients in the RCS and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the RPS causes the control rods to move into the core. For design purposes, reactor trip is assumed to occur a total of 400 times over the life of the plant.

The severity of the cooldown transients associated with reactor trips depends on the extent of secondary side cooling. Three cooldown cases are considered:

a. Reactor trip with no inadvertent cooldown - 230 occurrences

b. Reactor trip with cooldown but no safety injection - 160 occurrences

c. Reactor trip with cooldown actuating safety injection - 10 occurrences

5. Inadvertent Reactor Coolant System Depressurization

Several events can be postulated as occurring during normal plant operation which will cause rapid depressurization of the RCS. These include:

a. Actuation of a single pressurizer safety valve.

b. Inadvertent opening of one or both pressurizer power-operated relief valve due either to equipment malfunction or operator error.

c. Malfunction of a single pressurizer pressure controller causing two pressurizer spray valves to open.

d. Inadvertent opening of one pressurizer spray valve, due either to equipment malfunction or operator error.

e. Inadvertent auxiliary spray. A “lockout” feature on auxiliary spray valve prevents inadvertent spray of unheated water in the pressurizer.

Of these events, the pressurizer safety valve actuation causes the most severe transients, and is used as an “umbrella” case to conservatively represent the reactor coolant pressure and temperature variations arising from any of them.

When a pressurizer safety valve opens, and remains open, the system rapidly depressurizes, the reactor trips, and the safety injection system is actuated. Also, the passive accumulators of the SIS are actuated when pressure decreases by approximately 1,600 psi. The depressurization and cooldown are eventually terminated by operator action. All of these effects are completed within approximately 18 minutes. It is conservatively assumed that none of the pressurizer heaters are energized.

With pressure constant and safety injection in operation, boil off of hot leg liquid through the pressurizer and open safety valve will continue.

For design purposes this transient is assumed to occur 20 times during the 60 year design life of the plant.

6. Control Rod Drop

This transient occurs if a bank of control rods drops into the fully inserted position due to a single component failure. The reactor is assumed to be tripped on OTΔT, OPΔT or low pressurizer pressure. It is assumed that this transient occurs 80 times over the life of the plant.

7. Inadvertent Safety Injection Actuation

For design purposes, no credit is taken for the P–19 cold leg injection permissive for the inadvertent safety injection actuation. A spurious safety injection signal results in an immediate reactor trip followed by actuation of the high head centrifugal charging pumps. Without crediting P–19, these pumps deliver borated water to the RCS cold legs. The initial portion of this transient is similar to the reactor trip from full power with no cooldown. Controlled steam dump and auxiliary feedwater flow after trip removes core residual heat. Reactor coolant temperature and pressure decrease as the control rods move into the core.

Later in the transient, the injected water causes the RCS pressure to increase to the pressurizer power operated relief valve (PORV) setpoint and the primary and secondary temperatures to decrease gradually. Operator action is credited to verify that the PORV block valves are open within 10 minutes of initiation of the event. Water relief through the safety valves will not occur provided at least one PORV actuates on demand. The transient continues until the operator stops the charging pumps. It is assumed that the plant is then returned to no load conditions, with pressure and temperature changes controlled within normal limits.

For design purposes this transient is assumed to occur 60 times during the 60 year design life of the plant.

8. Operating Basis Earthquake

The operating basis earthquake is that earthquake which can reasonably be expected to occur during the plant life.

9. Excessive Feedwater Flow

An excessive feedwater flow transient is conservatively defined as an umbrella case to cover occurrence of several events of the same general nature. The postulated transient results from inadvertent opening of a feedwater control valve while the plant is at the hot standby or no load condition, with the feedwater, condensate, and heater drain systems in operation.

It is assumed that the stem of a feedwater control valve fails and the valve immediately reaches the full open position. In the steam generator directly affected by the malfunctioning valve ("failed loop"), the feedwater flow step increases from essentially zero flow to the value determined by the system resistance and the developed head of all operating feedwater pumps. Steam flow is assumed to remain at zero and the temperature of the feedwater entering the steam generator is conservatively assumed to be 32°F. Feedwater flow is isolated on a reactor coolant low T_{avg} signal; a low pressurizer pressure signal actuates the safety injection system. Auxiliary feedwater flow, initiated by the safety injection signal, is assumed, to continue with all pumps discharging into the affected steam generator. It is also assumed, for conservatism in the secondary side analysis, that auxiliary feedwater flows to the steam generators not affected by the malfunctioned valve, in the "unfailed loops". Plant conditions stabilize at the values reached in 600 seconds, at which time auxiliary feedwater flow is terminated. The plant is then either taken to cold shutdown, or returned to the no load condition at a normal heatup rate with the auxiliary feedwater system under manual control.

For design purposes, this transient is assumed to occur 30 times during the life of the plant.

10. RCS Cold Overpressurization

RCS cold overpressurization can occur during startup and shutdown conditions at low temperature, with or without the existence of a steam bubble in the pressurizer, and is especially severe when the reactor coolant system is in a water-solid configuration. The event is inadvertent, and usually generated by any one of a variety of malfunctions or operator errors. All events which have occurred to date may be categorized as belonging to either events resulting in the addition of mass (mass input transient), or events related to the addition of heat (heat input transient). All these possible transients are represented by composite "umbrella" design transients, referred to as RCS cold overpressurization.

For design purposes, this transient is assumed to occur 10 times during the 60 year design life of the plant.

Emergency Conditions

The following primary system transients are considered emergency conditions:

1. Small loss-of-coolant accident
2. Small steam line break
3. Complete loss of flow

1. Small loss of coolant accident

For design transient purposes the small loss of coolant accident is defined as a break equivalent to the severance of a 1 inch inside diameter branch connection. (Liquid breaks are limited to less than 0.375 inch inside diameter and pressurizer steam space breaks are limited to 0.25 inch by the installation of an orifice. This ensures that the break can be handled by the normal makeup system and produce no significant fluid systems transients). Breaks which are much larger than one inch will cause accumulator injection soon after the accident and are regarded as faulted conditions. For design purposes it is assumed that this transient occurs five times during the life of the plant. It should be assumed that the safety injection system is actuated immediately after the break occurs and subsequently delivers water at a minimum temperature of 32°F to the RCS.

2. Small Steam Line Break

For design transient purposes, a small steam break is defined as a break equivalent in effect to a steam safety valve opening and remaining open. This transient is assumed to occur 5 times during the life of the plant. The following conservative assumptions are used in defining the transients:

a. The reactor is initially in a hot, zero-power condition.

b. The small steam break results in immediate reactor trip and safety injection actuation.

c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

d. The safety injection system operates at a design capacity and repressurizes the RCS within a relatively short time.

3. Complete Loss of Flow

This accident involves a complete loss of flow from full power resulting from simultaneous loss of power to all reactor coolant pumps. The consequences of this incident are a reactor trip and turbine trip on undervoltage followed by automatic opening of the steam dump system. For design purposes this transient is assumed to occur five times during the plant lifetime.

Faulted Conditions

The following primary system transients are considered faulted conditions. Each of the following accidents should be evaluated for one occurrence:

1. Reactor coolant pipe break (Large loss-of-coolant accident)
2. Large steam line break
3. Feedwater line break
4. Reactor coolant pump locked rotor
5. Control rod ejection
6. Steam generator tube rupture
7. Safe shutdown earthquake

1. Reactor Coolant Pipe Break (Large Loss-of-Coolant Accident)

Following postulated rupture of a reactor coolant pipe resulting in a large loss of coolant, the primary system pressure decreases causing the primary system temperature to decrease. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal will remain at or near the operating temperature by the end of the blowdown. It is conservatively assumed that the safety injection system is actuated to introduce water at a minimum temperature of 32°F into the RCS. The safety injection signal will also result in reactor and turbine trips.

2. Large Steam Line Break

The transient is based on the postulated complete severance of the largest steam line. The following conservative assumptions are made:

a. The reactor is initially in a hot, zero-power condition.

b. The large steam line break results in immediate reactor trip and in actuation of the safety injection system.

c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transients.

d. The safety injection system operates at design capacity and repressurizes the reactor coolant system within a relatively short time.

3. Feedwater Line Break

This accident involves the double-ended rupture of a main feedwater piping from full power, resulting in rapid blowdown of one steam generator and the termination of main feedwater flow to the others. The blowdown is completed in approximately 27 seconds. Conditions were conservatively chosen to give the most severe primary side and secondary side transients. All auxiliary feedwater flow exits at the break. The incident is terminated when the operator manually realigns the auxiliary feedwater system to isolate the break and to deliver auxiliary feedwater to the intact steam generators.

4. Reactor Coolant Pump Locked Rotor

This accident is based on the instantaneous seizure of a reactor coolant pump with the plant operating at full power. The locked rotor can occur in any loop. Reactor trip occurs almost immediately, as the result of low coolant flow in the affected loop.

5. Control Rod Ejection

This accident is based on the single most reactive control rod being instantaneously ejected from the core. This reactivity insertion in a particular region of the core causes a severe pressure increase in the RCS such that the pressurizer safety valves will lift and also causes a more severe temperature transient in the loop associated with the affected region than in the other loops. For conservatism the analysis is based on the reactivity insertion and does not include the mitigating effects (on the pressure transient) of coolant blowdown through the hole in the vessel head vacated by the ejected rod.

6. Steam Generator Tube Rupture

This accident postulates the double-ended rupture of a steam generator tube resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to the resulting safety injection signal. In addition, safety injection actuation automatically isolates the feedwater lines, by tripping all feedwater pumps and closing the feedwater isolation valves. When this accident occurs, some of the reactor coolant blows down into the affected steam generator causing the shell side level to rise. The primary system pressure is reduced below the secondary safety valve setting. Subsequent recovery procedures call for isolation of the steam line leading from the affected steam generator. The recovery actions may involve the use of unheated auxiliary spray into the pressurizer. Such an action will require override of the auxiliary spray valve "lockout."

7. Safe Shutdown Earthquake

The safe shutdown earthquake is defined as the maximum vibratory ground motion which can reasonably be predicted from geologic and seismic evidence.

Test Conditions

The following primary system transients under test conditions are discussed:

1. Primary Side Hydrostatic test

2. Secondary Side Hydrostatic test

1. Primary Side Hydrostatic Test

The pressure tests include both shop and field hydrostatic tests which occur as a result of component or system testing. This hydro test is performed at a water temperature which is compatible with reactor vessel material ductility requirements and a test pressure of 3,107 psig (1.25 times design pressure). In this test, the RCS is pressurized to 3,107 psig coincident with steam generator secondary side pressure of 0 psig. The RCS is designed for 10 cycles of these hydrostatic tests, which are performed prior to plant startup. The number of cycles is independent of other operating transients.

Additional hydrostatic tests may be performed to meet the inservice inspection requirements of ASME Section XI subarticle IS5-20. A total of four such tests is expected. The increase in the fatigue usage factor caused by these tests is easily covered by the conservative number (200) of primary side leakage tests that are considered for design.

2. Secondary Side Hydrostatic Test

The secondary side of the steam generator is pressurized to 1.25 design pressure with a minimum water temperature of 120°F coincident with the primary side at 0 psig.

For design purposes it is assumed that the steam generator will experience 10 cycles of this test.

These tests may be performed either prior to plant startup, or subsequently following shutdown for major repairs, or both.

3.9N.1.2 Computer Programs Used in Analyses

The following computer programs have been used in the analysis of Seismic Category I components and equipment within the NSSS scope.

1. ANSYS - General purpose finite element code used or problems of structural dynamic response in addition to static elastic and inelastic analysis. The code is widely used in the public domain.

3.9N.1.3 Experimental Stress Analysis

No experimental stress analysis methods are used for Category I systems or components. However, for reactor internals, Westinghouse makes extensive use of measured results from prototype plants and various scale model tests as discussed in Section 3.9N.2.

3.9N.1.4 Considerations for the Evaluation of the Faulted Condition

3.9N.1.4.1 Loading Conditions

The structural evaluations performed on the reactor coolant system consider the loadings specified in Table 3.9N–2. These loads result from thermal expansion, pressure, weight, operating basis earthquake (OBE), safe shutdown earthquake (SSE), design basis loss-of-coolant accident, and plant operational thermal and pressure transients.

3.9N.1.4.2 Analysis of Primary Components

Equipment which serves as part of the pressure boundary in the reactor coolant loop includes the steam generators, the reactor coolant pumps, the pressurizer, and the reactor vessel. This equipment is ANS Safety Class 1 and the pressure boundary meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

Loads are applied to the RCS components for all loading conditions on an “umbrella” load basis. That is, on the basis of previous plant analyses, a set of loads are determined which should be larger than those seen in any single plant analysis. The umbrella loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. The results of the reactor coolant loop analysis are used to determine the actual loads acting on the equipment nozzles and the support/component interface locations. Upon completion of the system analysis, conformance is demonstrated between the actual plant loads and the loads used in the analyses of the components. Any deviations where the actual load is larger than the umbrella load are handled by individualized analysis.

Seismic analyses are performed individually for the reactor coolant pump, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation are used for the component analysis. The reactor pressure vessel is qualified by static analysis based on loads derived from dynamic system analysis.

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of ASME III.

Valves in sample lines connected to the RCS are not considered to be ANS Safety Class 1 nor ASME Class 1. This is because the nozzles, where the lines connect to the primary system piping, are orificed to a 3/8-inch hole. This hole restricts the flow such that loss through a severance of one of these lines can be made up by normal charging flow.

3.9N.1.4.3 Dynamic Analysis of Reactor Pressure Vessel for Postulated Loss-of-Coolant Accident

3.9N.1.4.3.1 Introduction

This section presents the method of computing the reactor pressure vessel response to a postulated loss-of-coolant accident (LOCA). The resulting structural analysis considers loads on the reactor vessel resulting from the reactor coolant loop mechanical loads, internal hydraulic pressure transients, and reactor cavity pressurization (for postulated breaks in the reactor coolant pipe at the vessel nozzles). The vessel is restrained by reactor vessel support pads and shoes beneath 4 of the reactor vessel nozzles, and the reactor coolant loops with the primary supports of the steam generators and the reactor coolant pumps.

For the limiting break in the main loop piping, pipe displacement restraints installed in the primary shield wall limit the break opening area of the vessel nozzle pipe breaks to less than 144 square inches. This break area was determined to be an upper bound by using worst case vessel and pipe relative motions based on similar plant analyses. Detailed studies have shown that pipe breaks at the hot or cold leg reactor vessel nozzles, even with a limited break area, would give the highest reactor vessel support loads and the highest vessel displacements, primarily due to the influence of reactor cavity pressurization. By considering these breaks, the most severe reactor vessel support loads are determined. In addition, two breaks outside the shield wall, for which there is no cavity pressurization, were also analyzed; specifically, the pump outlet nozzle pipe break and the steam generator inlet nozzle pipe break were considered.

Since the plant has been approved for the leak-before-break (LBB) analysis methodology, the LOCA analyses of the reactor vessel system are not required to include the dynamic effects of the main RCS pipe breaks, in accordance with GDC-4. The most limiting breaks to be considered are the branch line breaks. These breaks consist of breaks in the accumulator line in cold leg; the pressurizer surge line in the hot leg; and the residual heat removal (RHR) line in the hot leg. These three breaks were considered for the dynamic effects of the reactor vessel system. For the analyses of the branch line breaks, the reactor cavity pressurization loads described in the next section are not utilized in the analysis.

The results of the LOCA analysis of the reactor pressure vessel were determined using separate analyses. Westinghouse performed analyses to determine the reactor vessel displacement due to vessel internal hydraulic loads. RPV support and reactor coolant loop stiffnesses, supplied by SWEC, were included in the RPV structural model. The results of the Westinghouse analyses were dynamic displacement data for the RPV.

SWEC determined the reactor vessel displacements for the remaining LOCA loads. By combining these displacements with the displacements from the Westinghouse analyses, the total combined RPV displacements were determined for the postulated pipe ruptures.

3.9N.1.4.3.2 Loading Conditions

Following a postulated pipe rupture at the reactor vessel nozzle, the reactor vessel is excited by time-history forces. As previously mentioned, these forces are the combined effect of three phenomena:

1. Reactor Coolant Loop Mechanical Loads

   The reactor coolant loop mechanical forces are derived from the elastic analysis of the loop piping for the postulated break.

2. Reactor Cavity Pressurization Forces

   Reactor cavity pressurization forces arise for the pipe breaks at the vessel nozzles from the steam and water which is released into the reactor cavity through the annulus around the broken pipe. The reactor cavity is pressurized asymmetrically with higher pressure on the side of the broken pipe resulting in horizontal forces applied to the reactor vessel. Smaller vertical forces arising from pressure on the bottom of the vessel and the vessel flanges are also applied to the reactor vessel.

   The internals reaction forces develop from asymmetric pressure distributions inside the reactor vessel. For a vessel inlet nozzle break, the depressurization wave path is through the broken loop inlet nozzle and into the downcomer annulus between the core barrel and reactor vessel. The initial waves propagate up, down, and around the downcomer annulus and up through the fuel. In the case of an RPV outlet nozzle break and steam generator inlet nozzle break, the wave passes through the RPV outlet nozzle and directly into the upper internals region, depressurizes the core, and enters the downcomer annulus from the bottom of the vessel. Thus, for an outlet nozzle break, the downcomer annulus is depressurized with much smaller differences in pressure horizontally across the core barrel than for the inlet break. For both the inlet and outlet nozzle breaks, the depressurization waves continue their propagation by reflection and translation through the reactor vessel fluid but the initial depressurization wave has the greatest effect on the loads.

3. Reactor Internal Hydraulic Forces

   The reactor internals hydraulic pressure transients were calculated including the assumption that the structural motion of the core barrel is coupled with the pressure transients. This phenomenon is known as hydro-elastic coupling or fluid-structure interaction. The hydraulic analysis considers the fluid-structure interaction of the core barrel by accounting for the deflections of constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized regions. The analytical methods used to develop the reactor internals hydraulics are described in WCAP-8708.

3.9N.1.4.3.3 Reactor Vessel and Internals Modeling

The mathematical model of the RPV is a three-dimensional nonliner finite element model which represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. The RPV three-dimensional nonlinear finite element model is shown in Figure 3.9N–1. The model consists of three concentric structural submodels connected by nonlinear impact elements and stiffness matrices, which is connected to a submodel of the CRDMs and CRDM seismic platform, tie rods, and lifting legs. The first submodel, represents the reactor vessel shell and associated components. The reactor vessel is restrained by four reactor vessel supports (situated beneath alternate nozzles) and by the attached primary coolant piping. Each reactor vessel support is modeled by a linear horizontal stiffness and a vertical impact element. The attached piping is represented by a stiffness matrix.

The second submodel, represents the reactor core barrel (RCB), neutron panels, lower support plate, tie plates, and secondary core support components. This submodel is physically located inside the first, and is connected to it by a stiffness matrix at the internals support ledge. Core barrel to vessel shell impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radial support locations.

The third and innermost submodel, represents the upper support plate, guide tubes, support columns, upper and lower core plates, and fuel. The third submodel is connected to the first and second by stiffness matrices and nonlinear elements.

Fluid-structure or hydro-elastic interaction is included in the reactor pressure vessel model for seismic evaluation. The horizontal hydro-elastic interaction is significant in the cylindrical fluid flow region between the core barrel and reactor vessel (the downcomer). Mass matrices with off-diagonal terms (horizontal degrees-of-freedom only) attach between nodes on the core barrel and reactor vessel shell.

The diagonal terms of the mass matrix are similar to the lumping of water mass to the vessel shell and core barrel. The off-diagonal terms reflect the fact that all the water does not participate when there is no relative motion of the vessel and core barrel. It should be pointed out that the hydrodynamic mass matrix has no artificial virtual mass effect and is derived in a straight-forward, quantitative manner.

The matrices are a function of the properties of two cylinders with a fluid in the cylindrical annulus, specifically; inside and outside radius of the annulus, density of the fluid and length of the cylinders. Vertical segmentation of the RCB allows inclusion of radii variations along the RCB height and approximates the effects of RCB beam deformation. These mass matrices were inserted between selected nodes on the core barrel and reactor vessel shell.

In the finite element approach, the structure is divided into a finite number of members or elements. Nodal displacements and impact forces are stored for post-processing.

The reactor vessel is restrained by the four attached reactor coolant loops with the steam generator and reactor coolant pump primary supports and the four reactor vessel supports, situated beneath alternate reactor vessel nozzles. The RPV support model (supplied by SWEC) is shown in Figure 3.9N–3. All support spring elements are linear, double-acting elements and represent the restraint provided by the attached piping and the vessel supports, including the primary shield wall.

3.9N.1.4.3.4 Analytical Methods

The time history effects of the internals hydraulic loads are applied to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis is the time history displacements of the reactor vessel system. The output from the analysis is used for detailed component evaluation.

3.9N.1.4.4 Stress Criteria for Reactor Coolant System Components

All RCS components are designed and analyzed for the design, normal, upset, and emergency conditions to the rules and requirements of the ASME Code Section III. The analysis methods and associated stress or load allowable limits used in evaluation of faulted conditions are those that are defined in Appendix F of the ASME Code.

Loading combinations and allowable stresses for RCS components are given in Tables 3.9N–2 and 3.9N–3, respectively. For faulted condition evaluations, the effects of the safe shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) are combined using the square root of the sum of the squares (SRSS) method.

3.9N.2 DYNAMIC TESTING AND ANALYSIS

3.9N.2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

See Section 3.9B.2.1.

3.9N.2.2 Seismic Qualification Testing of Safety Related Mechanical Equipment

See Section 3.10.

3.9N.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady State Conditions

The vibration characteristics and behavior due to flow induced excitation are very complex and not readily ascertained by analytical means alone. Reactor components are excited by the flowing coolant which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area should provide the forcing functions to be used in the dynamic analysis of the structures. In view of the complexity of the geometries and the random character of the pressure oscillations, a closed form solution of the vibratory problem by integration of the differential equation of motion is not always practical and realistic. The determination of the forcing functions as a direct correlation of pressure oscillations cannot be practically performed independent of the dynamic characteristics of the structure. The main objective is to establish the characteristics of the forcing functions that essentially determine the response of the structures. By studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing function can be deduced. These studies indicate that the most important forcing functions are flow turbulence, and pump-related excitation. The relevance of such excitations depends on many factors such as type and location of component and flow conditions. The effects of these forcing functions have been studied from test runs on models, prototype plants and in component tests, (WCAP-8303-P-A, WCAP-8517; Trojan Final Safety Analysis Report, Appendix A-12; WCAP 8780, WCAP-9945).

The Indian Point No. 2 plant (Docket No. 50.247) has been established as the prototype for a four-loop plant internals verification program and was fully instrumented and tested during hot functional testing. In addition, the Trojan plant (Docket No. 50-344) has provided, and the Sequoyah No. 1 plant (Docket No. 50-327) has provided, prototype data applicable to Millstone 3, (WCAP-8517; Trojan Final Safety Analysis Report, Appendix A-12; WCAP-9945).

Millstone 3 is similar to Indian Point No. 2; the only significant differences are the modifications resulting from the use of 17 x 17 fuel, replacement of the annular thermal shield with neutron shielding panels, and the change to the UHI-style inverted top hat support structure configuration. These differences are addressed below.

1. 17 x 17 fuel

   The only structural changes in the internals resulting from the design change from the 15 x 15 to the 17 x 17 fuel assembly is the guide tube. The new 17 x 17 guide tubes are stronger and more rigid, hence they are less susceptible to flow induced vibration. The fuel assembly itself is relatively unchanged in mass and spring rate, and thus no significant deviation of internal vibration is expected from the vibration with the 15 x 15 fuel assemblies vibration characteristics.

2. Neutron shield panel lower internals

   The primary cause of core barrel excitation is flow turbulence, which is not affected by the upper internals (WCAP-8517). The vibration levels due to core barrel excitation for Trojan and Millstone 3 both having neutron shield panels are expected to be similar. The coolant inlet density of Millstone 3 is slightly lower than Trojan 1 and the flowrate is slightly higher. Scale model tests show that the core barrel vibration varies as velocity, raised to a small power (WCAP-8317). The difference in fluid density and flowrate result in approximately 3 percent higher core barrel vibration for Millstone 3, than for Trojan 1. However, scale model test results (WCAP-8317-A) and results from Trojan (WCAP-8780), show that core barrel vibration of plants with neutron shielding pads is significantly less than that of plants with thermal shields. This information and the fact that low core barrel stresses with large safety margins were measured at Indian Point No. 2 (thermal shield configuration) lead to the conclusion that stresses approximately equal to those of Indian Point No. 2 result on Millstone 3 internals with the attendant large safety margins.

3. UHI-style inverted top hat upper support configuration

   The components of the upper internals are excited by turbulent forces due to axial and crossflows in the upper plenum (WCAP-8517) and by pump-speed related excitations (WCAP-8517; WCAP-8780). Sequoyah and Millstone 3 have the same basic upper internals configuration, therefore, the general vibration behavior is not changed.

   Results from Sequoyah Unit 1 plant testing (Altman et al., 1981) show high factors of safety for upper internals components. These results, which are supported by scale model test results and analytical work, can be used to determine the adequacy of the Millstone 3 new upper internals.

The original test and analysis of the four-loop configuration is augmented by WCAP-8317-A; WCAP-8517; Trojan Final Safety Analysis Report, Appendix A-12, WCAP-8780; and WCAP-9945, supported by scale model and analytical work to cover the effects of successive hardware modifications.

3.9N.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Because the Millstone 3 reactor internals design configuration is well-characterized, as was discussed in Section 3.9N.2.3, it is not considered necessary to conduct instrumented tests of the Millstone Unit 3 hardware. The recommendations of Regulatory Guide 1.20 are met by conducting the confirmatory preoperational testing examination for integrity per Section D, of Regulatory Guide 1.20, Regulations for Reactor Internals Similar to the Prototype Design. This examination will include some 30 points with special emphasis on the following areas:

1. All major load-bearing elements of the reactor internals relied upon to retain the core structure in place

2. The lateral, vertical and torsional restraints provided within the vessel

3. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals

4. Those other locations on the reactor internal components which are similar to those examined on the prototype designs

5. The inside of the vessel will be inspected before and after the hot functional test, with all the internals removed, to verify that no loose parts or foreign materials are in evidence.

A particularly close inspection is made of the following items or areas using a 5X or 10X magnifying glass or other appropriate inspection.

1. Lower internals

   a. Upper barrel to flange girth weld

   b. Upper barrel to lower barrel girth weld

   c. Upper core plate aligning pin. Examine bearing surfaces for any shadow marks, burnishing, buffing or scoring. Inspect welds for integrity.

   d. Irradiation specimen guide screw locking devices and dowel pins. Check for lockweld integrity.

   e. Baffle assembly locking devices. Check for lockweld integrity.

   f. Lower barrel to core support girth weld

   g. Neutron shield panel screw locking devices and dowel pin lockwelds. Examine the interface surfaces for evidence of tightness. Check for lockweld integrity

   h. Radial support key welds

   i. Insert screw locking devices. Examine soundness of lockwelds.

   j. Core support columns and instrumentation guide tubes. Check the joints for tightness and soundness of the locking devices.

   k. Secondary core support assembly weld integrity

   l. Lower radial support keys and inserts. Examine bearing surfaces for shadow marks, burnishing, buffing or scoring. Check the integrity of the lockwelds. These members supply the radial and torsional constraint of the internals at the bottom relative to the reactor vessel while permitting axial and radial growth between the two. Subsequent to the hot functional testing, the bearing surfaces of the key and keyway will show burnishing, buffing or shadow marks which indicate pressure loading and relative motion between these parts. Minor scoring of engaging surfaces is also possible and acceptable.

   m. Gaps and baffle joints. Check gaps between baffle-to-baffle joints.

2. Upper internals

a. Thermocouple conduits, clamps, and couplings

b. Guide tube, support column, orifice plate, and thermocouple assembly locking devices

c. Support column and thermocouple conduit assembly clamp welds

d. Upper core plate alignment inserts. Examine bearing surfaces for shadow marks, burnishing, buffing or scoring. Check the locking devices for integrity of lockwelds.

e. Thermocouple conduit fitting locktab and clamp welds

f. Guide tube enclosure and card welds

Acceptance standards are the same as required in the shop by the original design drawings and specifications.

During the hot functional test, the internals will be subjected to a total operating time greater than normal full-flow conditions (four pumps operating) of at least 240 hours. This provides a cyclic loading of approximately 10^7 cycles on the main structural elements of the internals. In addition there will be some operating time with only one, two and three pumps operating.

Pre- and post-hot functional inspection results serve to confirm predictions that the internals are adequately designed for flow induced vibrations. When no signs of abnormal wear, no harmful vibrations are detected and no apparent structural changes take place, the four-loop core support structures are considered to be structurally adequate and sound for operations.

3.9N.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

The reactor internals analysis under faulted events considers the following conditions:

- LOCA
- SSE (Safe Shutdown Earthquake)

The criteria for acceptability in regard to mechanical integrity analyses are that adequate core cooling and core shutdown must be assured. This implies that the deformation of the reactor internals must be sufficiently small so that geometry remains substantially intact. Consequently, the limitations established for the internals are concerned with the deflections and stability of the parts in addition to stress criteria to assure integrity of the components.

The MULTIFLEX digital computer program (WCAP-8708 and WCAP-8709) which was developed for the purpose of calculating local fluid pressure, flow, and density transients that occur in pressurized water reactor coolant systems during a loss-of-coolant accident is applied to the subcooled, transition, and saturated two-phase blowdown regimes. This is in contrast to programs such as WHAM (Fabic 1967) which are applicable only to the subcooled region and which, due to their method of solution, could not be extended into the region in which large changes in the sonic velocities and fluid densities take place. This blowdown code is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy, are solved numerically using a fixed mesh in both space and time.

Although spatially one-dimensional conservation laws are employed, the code can be applied to describe three-dimensional system geometries by use of the equivalent piping networks. Such piping networks may contain any number of pipes of channels of various diameters, dead ends, branches (with up to 6 pipes connected to each branch), contractions, expansions, orifices, pumps and free surfaces (such as in the pressurizer). System losses such as friction, contraction, expansion, etc are considered.

The MULTIFLEX code evaluates the pressure and velocity transients at various locations throughout the system. These pressure and velocity transients are stored as a permanent tape file and are made available to the programs LATFORC and FORCE2 which utilize a detailed geometric description in evaluating the loadings on the reactor internals. The LATFORC computer code calculates the lateral hydraulic loads on the reactor vessel wall and core barrel, while the FORCE2 code calculates the vertical hydraulic loads on the reactor vessel internals.

Each reactor component for which LATFORC and FORCE2 calculations are performed is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

1. The pressure differential across the element

2. Flow stagnation on, and unrecovered orifice losses across the element

3. Friction losses along the element

Input to the codes, in addition to the blowdown pressure and velocity transients, includes the effective area of each element on which the force acts due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

The reactor internals analysis has been performed using conservative assumptions. Some of the more significant assumptions are:

1. The mechanical and hydraulic analyses have considered the effect of hydroelasticity.

2. The reactor internals are represented by concentric pipes, 3-D beams and a multi-mass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components.

The model described is considered to have a sufficient number of degrees of freedom to represent the most important modes of vibration in the horizontal and vertical directions.

The pressure waves generated within the reactor are highly dependent on the location and nature of the postulated pipe failure. In general, the more rapid the severance of the pipe, the more severe the imposed loadings on the components. A one millisecond severance time is taken as the limiting case.

In the case of the hot leg break, the vertical hydraulic forces produce an initial upward lift of the core. A rarefaction wave propagates through the reactor hot leg nozzle into the interior of the upper core barrel. Since the wave has not reached the flow annulus on the outside of the barrel, the upper barrel is subjected to an impulsive compressive wave. Thus, dynamic instability (buckling) or large deflections of the upper core barrel, or both, are possible responses of the barrel during hot leg break results in transverse loading on the upper core components as the fluid exits the hot leg nozzle.

In the case of the cold leg break, a rarefaction wave propagates along a reactor inlet pipe, arriving first at the core barrel at the inlet nozzle of the broken loop. The upper barrel is then subjected to a non-axisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel. After the cold leg break, the initial steady state hydraulic lift forces (upward) decrease rapidly (within a few milliseconds) and then increase in the downward direction. These cause the reactor core and lower support structure to move initially downward.

If a simultaneous seismic event with the intensity of the SSE is postulated with the loss-of-coolant accident, the imposed loading on the internals component may be additive in certain cases and therefore the combined loading must be considered. In general, however, the loading imposed by the earthquake is small compared to the blowdown loading.

The summary of the mechanical analysis follows:

Transverse Excitation Model for Blowdown

Various reactor internal components are subjected to transverse excitation during blowdown. Specifically, the barrel, guide tubes, and upper support columns are analyzed to determine their response to this excitation.

Core Barrel - For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse.

The barrel is then analyzed for dynamic buckling using the following conservative assumptions:

1. The effect of the fluid environment is neglected.

2. The shell is treated as simply supported.

During cold leg blowdown, the barrel is subjected to a non-axisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel.

The analysis of transverse barrel response to cold leg blowdown is performed as follows:

1. The core barrel is analyzed as a shell with two variable sections to model the support flange and core barrel.

2. The barrel with the core and neutron shield panels, is analyzed as a beam supported at the top and supported at bottom the lower radial support and the dynamic response is obtained.

Guide Tubes - The guide tubes in closest proximity to the outlet nozzle of the ruptured loop are the most severely loaded during a blowdown. The transverse guide tube forces decrease with increasing distance from the ruptured nozzle location.

All of the guide tubes are designed to maintain the function of the control rods for a break size of 144 in^2 and smaller. No credit for the function of the control rods is assumed for break size areas above 144 in^2. However, the design of the guide tube will permit control rod operation in all but four control rod positions, which is sufficient to maintain the core in a subcritical configuration, for break sizes up to a double-ended hot leg break. This double-ended hot leg break imposes the limiting lateral guide tube loading.

Upper Support Columns - Upper support columns located close to the broken nozzle during hot leg break will be subjected to transverse loads due to cross flow. The loads applied to the columns are computed with a method similar to the one used for the guide tubes, i.e., by taking into consideration the increase in flow across the column during the accident. The columns are studied as beams with variable section and the resulting stresses are obtained using the reduced section modulus and appropriate stress risers for the various sections.

The effects of the gaps that could exist between vessel and barrel, between fuel assemblies, and between fuel assemblies and baffle plates are considered in the analysis. The stresses due to the safe shutdown earthquake (vertical and horizontal components) were combined in the most unfavorable manner with the blowdown stresses in order to obtain the largest principal stress and deflection.

All reactor internals components were found to be within acceptable stress and deflection limits for both hot leg and cold leg loss of coolant accidents occurring simultaneously with the safe shutdown earthquake.

These results indicate that the maximum deflections and stress in the critical structures are below the established allowable limits. For the transverse excitation, it is shown that the upper barrel does not buckle during a hot leg break and that it has an allowable stress distribution during a cold leg break.

Even though control rod insertion is not required for plant shutdown, this analysis shows that most of the guide tubes will deform within the limits established experimentally to assure control rod insertion. For the guide tubes deflected above the no loss of function limit, it must be assumed that the rods will not drop. However, the core will still shut down due to the negative reactivity insertion in the form of core voiding. Shutdown will be aided by the great majority of rods that do drop. Seismic deflections of the guide tubes are generally negligible by comparison with the no loss of function limit.

3.9N.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

As stated in Section 3.9N.2.3, it is not considered necessary to conduct instrumented tests of the reactor vessel internals. WCAP-8516-P, WCAP-8517 (1975) and the Trojan Final Safety Analysis Report, Appendix A12 describe predicted vibration behavior based on studies performed prior to the plant tests. These studies, which utilize analytical models, scale model test results, component tests, and results of previous plant tests, are used to characterize the forcing functions and establish component structural characteristics so that the flow induced vibratory behavior and response levels for reactor internals are estimated. These estimates are supported by values deduced from plant test data obtained from the Sequoyah and Trojan internals vibration measurement programs. Adequacy of the Millstone 3 internals will be verified by the use of Sequoyah and Trojan results supported by scale model tests and analytical work.

3.9N.3 ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

The ASME Code Class components are constructed in accordance with the ASME Code, Section III.

A detailed discussion of ASME Code Class 1 components is provided in Section 3.9N.1. For core support structures, design loading conditions are given inSection 3.9N.2.3. Loading conditions are discussed in Section 4.2.

Method of analysis and testing for core support structures are discussed in Sections 3.9N.2.3, 3.9N.2.5, and 3.9N.2.6.

3.9N.3.1 Loading Combinations, Design Transients, and Stress Limits for Class 2 Components

3.9N.3.1.1 Design Loading Combinations

The design loading combinations for ASME Code Class 2 and 3 components and supports are given in Table 3.9N–4.

3.9N.3.1.2 Design Stress Limits

The design stress limits established for the components are sufficiently low to assure that violation of the pressure retaining boundary will not occur. These limits, for each of the loading combinations, are component oriented and are presented in Tables 3.9N-5 through 3.9N-10 for vessels, pumps and valves. Active* pumps and valves are further discussed in Section 3.9N.3.2. The design of component supports is discussed in Section 3.9N.3.4.

3.9N.3.2 Pumps and Valve Operability Assurance

Equipment for Millstone 3 is designed to comply with the intent of Regulatory Guide 1.48; i.e., it is designed/analyzed to ensure structural integrity and operability. However, the load combinations and stress limits that were used reflect NRC requirements when the components were purchased and subsequently designed. The codes and procedures which were available when the components were purchased are based on conservative design requirements. These codes and procedures have been used by the nuclear industry for the design of components that are installed in plants that are presently operating. A list of active pumps and valves is presented in Tables 3.9N–11 and 3.9N–12, respectively.

3.9N.3.2.1 ASME Code Class Pumps

Active pumps were designed in accordance with Section III of the ASME Code. The stress levels in the pumps do not exceed those provided in Tables 3.9N–7 and 3.9N–10. Forces resulting from seismic accelerations in the horizontal and vertical directions are included in the analysis of the pumps and their supports. To eliminate any amplification of the seismic floor accelerations in the pump support structure, the supports were designed to have natural frequencies in excess of 33 Hz.

The pumps are subjected to a series of tests prior to installation and after installation in the plant. In-shop tests include hydrostatic tests to 150 percent of the design pressure, seal leakage tests, net positive suction head (NPSH) tests to qualify the pumps for the minimum available NPSH, and functional performance tests. For the NPSH and functional performance tests, the pumps are placed in a test loop and subjected to operating conditions. After installation, the pumps undergo cold hydrostatic tests, hot functional tests to verify operation, and periodic inservice inspection and operation.

3.9N.3.2.2 Valve Operability

Safety related valves are subjected to a series of tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test to ASME Code, Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, and operational tests to verify that the valve will open and close. Qualification of motor operators for environmental conditions is discussed in Sections 3.11 and 1.8 (in the discussion of Regulatory Guide 1.73, Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants). Cold hydrostatic tests, hot functional tests, periodic inservice inspections, and periodic inservice operations are performed in situ to verify the functional ability of the valve.

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* Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered in the respective operating condition categories.

These tests guarantee reliability of the valve for the design life of the plant. The valves are constructed in accordance with the ASME Code, Section III. The stress limits used for active Class 2 and 3 valves are shown in Table 3.9N–8. Active valves are designed to have a first natural frequency which is equal to or greater than 33 hertz and an analysis of the extended structure is performed for static equivalent safe shutdown earthquake loads applied at the center of gravity of the extended structure.

In addition to these tests and analyses, representative valves of each design type are tested during a simulated plant faulted condition event by demonstrating operational capabilities within the specified limits. These tests verify the analysis methods described above. The testing procedures are described below.

The valve is mounted in a manner which conservatively represents typical valve installations. The valve includes the operator, solenoid valves, and limit switches when such are normally attached to the valve in service. The operability of the valve during a faulted condition is demonstrated by satisfying the following criteria:

1. The actuator and yoke of the valve are statically deflected an amount equal to the deflection caused by the faulted condition accelerations applied at the center of gravity of the extended structure in the direction of the weakest axis of the yoke. The design pressure of the valve is applied to the valve.

2. The valve is cycled while in the deflected position. The time required to open or close the valve in the deflected position will be compared to similar data taken in the undeflected condition to evaluate the significance of any change.

3. Motor operators, external limit switches, and solenoid valves necessary for operation are qualified in accordance with IEEE Standard 344-1975.

The piping designer must limit the inputs to the valve to those levels for which the valve is qualified.

The above operability program description applies to valves with extended structures. Valves which are safety related but can be classified as not having an extended structure, such as swing check valves and safety valves, are considered separately. Valves are qualified by analysis and testing verifies the method of analysis.

Pressurizer safety valves are qualified for operability in the same manner as valves with extended structures as described above. The qualification methods include analysis of the bonnet for static equivalent safe shutdown earthquake loads, in shop hydrostatic and seat leakage tests, and periodic in situ valve inspection. Additionally, representative pressurizer safety valves are tested to verify analysis methods. This test is described as follows:

1. The safety valve is mounted to represent the worst case installation.

2. The valve body is pressurized to its normal system pressure.

3. A static load representing the faulted condition seismic load is applied to the top of the valve bonnet in the direction of the weakest axis of the extended structure.

4. The pressure is increased until the valve actuates.

5. Actuation of the valve at its setpoint ensures its operability during the faulted condition.

Using these methods, all the safety related valves in the systems are qualified for operability during a faulted event. These methods outlined above conservatively simulate the seismic event and ensure that the active valves will perform their safety related function when necessary.

3.9N.3.2.3 Pump Motor and Valve Operator Qualification

Active pump motors and active valve motor operators (and limit switches and solenoid valves) are seismically qualified in accordance with IEEE Standard 344-1975.

Qualification is accomplished by analysis, testing, or by a combination of analysis and testing. When analysis is used, such methods can be justified by:

1. Demonstrating that equipment being qualified is amenable to analysis, and

2. That the analysis can be correlated with test or be performed using standard analysis techniques.

3.9N.3.3 Design and Installation Details for Mounting of Pressure-Relieving Devices

Refer to Section 3.9B.3.3.

3.9N.3.4 Component Supports

The criteria for Westinghouse supplied supports for ASME Code Class 1 mechanical equipment is presented in Section 3.9N.1.

The criteria for Westinghouse supplied supports for ASME Code Class 2 and 3 mechanical equipment is presented in Section 3.9N.3.4.1.

3.9N.3.4.1 Component Supports for Tanks and Heat Exchangers

Component supports for Class 2 and 3 tanks and heat exchangers are of two types: linear and, for the most part, plate and shell type supports. The supports for these tanks and heat exchangers are designed and analyzed to the rules and requirements of Subsection NF of Section III of the ASME Code. The design analyses and associated stress or load allowable limits for faulted conditions are those defined in Appendix F of the ASME Code. The only exception to the above is the supports for the volume control tank which, because the procurement date of the tank predates Subsection NF, are designed to the requirements of the AISC Code.

3.9N.3.4.2 Component Supports for Pumps

Component supports for Class 2 and 3 auxiliary pumps are of two types: plate and shell and, for the most part, linear type supports. These supports, with the exception of the supports for the charging and safety injection pumps, are designed by the pump manufacturer to pressure boundary stress limits, but in no case is yield stress of the material exceeded. The supports for the charging and safety injection pumps meet the requirements of Subsection NF of Section III of the ASME Code.

3.9N.4 CONTROL ROD DRIVE SYSTEM (CRDS)

3.9N.4.1 Descriptive Information of CRDS

Control Rod Drive Mechanism

Control rod drive mechanisms are located on the dome of the reactor vessel head. They are coupled to rod cluster control assemblies (RCCAs) which have neutron absorber material over the entire length of the control rods and derive their name from this feature. The control rod drive mechanism is shown in Figure 3.9N–4 and schematically in Figure 3.9N–5.

The primary function of the control rod drive mechanism is to insert, withdraw or hold stationary, RCCAs within the core to control average core temperature and to shutdown the reactor.

The control rod drive mechanism is a magnetically-operated jack. A magnetic jack is an arrangement of three electromagnets which are energized in a controlled sequence by a power cycler to insert or withdraw rod cluster control assemblies in the reactor core in discrete steps. Rapid insertion of the rod cluster control assemblies occurs when electrical power is interrupted.

The control rod drive mechanism consists of four separate subassemblies. They are the pressure vessel, coil stack assembly, latch assembly, and the drive rod assembly.

1. The pressure vessel assembly includes a latch housing and a rod travel housing which are connected by a threaded, seal welded, maintenance joint which facilitates replacement of the latch assembly. The closure at the top of the rod travel housing is a threaded cap with a canopy seal weld for pressure integrity. Seismic support of the control rod drive mechanism is attained by the spacer plates of the rod position indicator coil stack assembly and the seismic support ring.

   The latch housing is the lower portion of the pressure vessel and encloses the latch assembly. The rod travel housing is the upper portion of the pressure vessel and provides space for the drive rod assembly during its upward movement as the control rods are withdrawn from the core.

2. The coil stack assembly includes the coil housings, an electrical conduit and connector, and three operating coils:

1. The stationary gripper coil

2. The movable gripper coil

3. The lift coil

The coil stack assembly is a separate unit which is installed on the control rod drive mechanism by sliding it over the outside of the latch housing. It rests on the base of the latch housing without mechanical attachment.

Energizing the operating coils causes movement of the pole pieces and latches in the latch assembly.

3. The latch assembly includes the guide tube, stationary pole pieces, movable pole pieces, and two sets of latches:

    1. The movable gripper latches

    2. The stationary gripper latches

    The latches engage grooves in the drive rod assembly. The movable gripper latches are moved up or down in 5/8-inch steps by the lift pole to raise or lower the drive rod assembly. The stationary gripper latches hold the drive rod assembly while the movable gripper latches are repositioned for the next 5/8-inch step.

4. The drive rod assembly includes a coupling, a drive rod, a disconnect button, a disconnect rod, and a locking button.

    The drive rod is machined with external grooves on a 5/8-inch pitch which receives the latches during holding or moving of the drive rod assembly. The coupling is attached to the drive rod and provides the means for coupling to the RCCA.

    The disconnect button, disconnect rod assembly, and locking button provide positive locking of the coupling to the RCCA and permits remote disconnection of the drive rod assembly.

The control rod drive mechanism can be tripped during any part of the power cycler sequencing if electrical power to the coils is interrupted, thereby releasing the drive rod assembly and inserting the RCCA.

The control rod drive mechanism is threaded and seal welded on a head adaptor on top of the reactor vessel head. The drive rod assembly is coupled to the rod cluster control assembly directly below.

The mechanism is capable of raising or lowering a 360 pound load, (which includes the drive rod assembly weight) at a rate of 45 inches/minute. Withdrawal of the RCCA is accomplished by energizing the magnetic coils, and insertion is by gravity.

The mechanism internals are designed to operate in 650°F reactor coolant. The pressure vessel assembly is designed to retain reactor coolant at 650°F and 2,500 psia. The three operating coils are designed to operate at an internal coil temperature of 392°F with forced air cooling required to maintain that temperature.

The control rod drive mechanism shown schematically in Figure 3.9N–5 withdraws and inserts an RCCA as shaped electrical pulses are received by the operating coils. An ON or OFF sequence, repeated by silicon controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 42 discrete coils mounted on the rod position indicator coil stack assembly surrounding the rod travel housing. Each coil magnetically senses the entry and presence of the top of the ferromagnetic drive rod assembly as it moves through the coil center line.

During plant operation the stationary gripper coil of the control rod drive mechanism holds the rod cluster control assembly in a static position until a stepping sequence is initiated at which time the movable gripper coil and lift coil are energized sequentially.

Drive Rod Assembly - RCCA Withdrawal

The drive rod assembly along with a coupled RCCA is withdrawn by repetition of the following sequence of events (Figure 3.9N–5).

1. The drive rod assembly is held in a stationary position by the stationary gripper (SG) latches with the SG coil energized.

2. Movable Gripper Coil - ON

   The latch locking plunger raises the swings the movable gripper latches into the drive rod assembly groove. A nominal 0.059-inch axial clearance exists between the latch tips and the drive rod.

3. Stationary Gripper Coil - OFF

   The force of gravity acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move downward 0.059 inch until the load of the drive rod assembly and attached control rod is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

4. Lift Coil - ON

The 5/8-inch gap between the movable gripper pole and the lift pole closes and the drive rod assembly with attached RCCA raises one step length (5/8 inch).

5. Stationary Gripper Coil - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 0.059-inch. The nominal 0.059 inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

6. Movable Gripper Coil - OFF

   The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

7. Lift Coil - OFF

   The gap between the movable gripper pole and lift pole opens. The movable gripper latches drop 5/8 inch to a position adjacent to a drive rod assembly groove.

8. Repeat Step 1

   The sequence described above (Items 1 through 7) is defined as one step or one cycle. The rod cluster control assembly moves 5/8 inch for each step or cycle. The sequence is repeated at a rate of up to 72 steps per minute and the drive rod assembly (which has a 5/8-inch groove pitch) is raised 72 grooves per minute. The rod cluster control assembly is thus withdrawn at a rate up to 45 inches per minute.

<u>Drive Rod Assembly/RCCA Insertion</u>

The sequence for RCCA insertion is similar to that for control rod withdrawal, except the timing of lift coil ON and OFF is changed to permit lowering the RCCA. The sequence begins with the SG energized in hold mode.

1. Lift Coil - ON

   The 5/8-inch gap between the movable gripper and lift pole closes. The movable gripper latches are raised to a position adjacent to a drive rod assembly groove.

2. Movable Gripper Coil - ON

The latch locking plunger raises and swings the movable gripper latches into a drive rod assembly groove. A nominal 0.059-inch axial clearance exists between the latch tips and the drive rod assembly.

3. Stationary Gripper Coil - OFF

   The force of gravity, acting upon the drive rod assembly and attached RCCA, causes the stationary gripper latches and plunger to move downward 0.059 inch until the load of the drive rod assembly and attached RCCA is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

4. Lift Coil - OFF

   The force of gravity and spring force separates the movable gripper pole from the lift pole and the drive rod assembly and attached RCCA drop down 5/8 inch.

5. Stationary Gripper Coil - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the three stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 0.059 inch. The nominal 0.059-inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

6. Movable Gripper Coil - OFF

   The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

7. Repeat Step 1

   The sequence is repeated, as for rod cluster control assembly withdrawal, up to 72 times per minute which gives an insertion rate of 45 inches per minute.

<u>Holding and Tripping of the Control Rods</u>

During most of the plant operating time, the control rod drive mechanisms hold the RCCA’s withdrawn from the core in a static position. In the holding mode, only one coil, the stationary gripper coil, is energized on each mechanism. The drive rod assembly and attached RCCA’s hang suspended from the three latches.

If power to the stationary gripper coil is cut off, the stationary gripper return spring combined with the weight of the drive rod assembly and the RCCA is sufficient to move latches out of the drive rod assembly groove. Following a power interruption to the coil causing the magnetic field to collapse, the control rod falls by gravity into the core.

3.9N.4.2 Applicable CRDS Design Specifications

For those components in the control rod drive system comprising portions of the reactor coolant pressure boundary, conformance with the General Design Criteria and 10CFR50, Section 50.55a is discussed in Sections 3.1 and 5.2 conformance with regulatory guides pertaining in Sections 4.5 and 5.2.3 and Section 1.8.

Design Bases

Bases for temperature, stress and structural members, and material compatibility are imposed on the design of the reactivity control components.

Design Transient and Loading Combinations

The CRDS is designed to withstand stresses originating from various operating design transients summarized in Table 3.9N–1. Structural evaluation performed on CRDM pressure retaining components consider the loading combinations specified in Table 3.9N–2.

Allowable Stresses

Allowable stresses for CRDM pressure retaining components are given in Table 3.9N–3.

Dynamic Analysis

The cyclic stresses due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the CRDS.

Control Rod Drive Mechanisms

The control rod drive mechanism (CRDM) pressure housings are Class 1 components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Boiler and Pressure Vessel Code. Both static and alternating stress intensities are considered. The stresses originating from the required design transients are included in the analysis.

A dynamic seismic analysis is required on the CRDM’s when a seismic disturbance has been postulated to confirm the ability of the pressure housing to meet ASME Code, Section III allowable stresses and to confirm its ability to trip when subjected to the seismic disturbance.

Control Rod Drive Mechanism Operational Requirements

The basic operational requirements for the CRDM's are:

1. 5/8-inch step
2. 144-inch travel (nominal)
3. 360 pound maximum load
4. Step in or out of 45 inches/minute (72 steps/minute)
5. Electrical power interruption shall initiate release of drive rod assembly/RCCA
6. Trip delay time of less than 150 milliseconds or less - Free fall of drive rod assembly shall begin less than 150 milliseconds after power interruption no matter what holding or stepping action is being executed with any load and coolant temperature of 100°F to 550°F
7. 60 year design life with normal refurbishment

3.9N.4.3 Design Loads, Stress Limits, and Allowable Deformations

3.9N.4.3.1 Pressure Vessel Assembly

The pressure retaining components are analyzed for loads corresponding to normal, upset, emergency and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

1. Control rod trip (equivalent static load)
2. Differential Pressure
3. Spring preloads
4. Coolant flow forces (static)
5. Temperature gradients
6. Differences in thermal expansion

a. Due to temperature differences

b. Due to expansion of different materials

7. Interference between components

8. Vibration (mechanically or hydraulically induced)

9. All operational transients listed in Table 3.9N–1

10. Pump overspeed

11. Seismic loads (operational basis earthquake and safe shutdown earthquake)

12. Blowdown forces (due to cold and hot leg break)

The main objective of the analysis is to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also to limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard method of strength of materials are used to establish the stresses and deflections of these components. The dynamic behavior of the reactivity control components has been studied using experimental test data and experience from operating reactors.

3.9N.4.3.2 Drive Rod Assembly

All postulated failures of the drive rod assemblies either by fracture or uncoupling lead to a reduction in reactivity. If the drive rod assembly fractures at any elevation, that portion remaining coupled falls with, and is guided by, the rod cluster control assembly. This always results in reactivity decrease for the control rods.

3.9N.4.3.3 Latch Assembly and Coil Stack Assembly

Results of Dimensional and Tolerance Analysis

With respect to the control rod drive mechanism system as a whole, critical clearances are present in the following areas:

1. Latch assembly (diametral clearances)

2. Latch arm-drive rod clearances

3. Coil stack assembly-thermal clearances

4. Coil fit in coil housing

The following defines clearances that are designed to provide reliable operation in the CRDM in these four critical areas. These clearances have been proven by life tests and actual field performance at operating plants.

Latch Assembly - Thermal Clearances

The magnetic jack has several clearances where parts made of Type 410 stainless steel fit over parts made from Type 304 stainless steel. Differential thermal expansion is therefore important. Minimum clearances of these parts at 68°F is 0.011 inches. At a maximum design temperature of 650°F minimum clearance is 0.0045 inches and at the maximum expected operating temperatures of 550°F is 0.0057 inches.

Latch Arm - Drive Rod Clearances

The CRDM incorporates a load transfer action. The movable or stationary gripper latch are not under load during engagement, as previously explained, due to load transfer action.

Figure 3.9N–6 shows latch clearance variation with the drive rod as a result of minimum and maximum temperatures. Figure 3.9N–7 shows clearance variations over the design temperature range.

Coil Stack Assembly - Thermal Clearances

The assembly clearances of the coil stack assembly over the latch housing was selected so that the assembly could be removed under all anticipated conditions of thermal expansion.

At 70°F, the inside diameter of the coil stack is 7.428/7.438 inches. The outside diameter of the latch housing is 7.390/7.380 inches.

Thermal expansion of the mechanism due to operating temperature of the control rod drive mechanism results in minimum inside diameter of the coil stack being 7.440 inches at 222°F and the maximum latch housing diameter being 7.426 inches at 650°F.

Under the extreme tolerance conditions listed above it is necessary to allow time for a 70°F coil stack assembly to heat during a replacement operation.

Four similar style coil stack assemblies were removed from four hot control rod drive mechanisms mounted on 11.035-inch centers on a 550°F test loop, allowed to cool, and then replaced without incident as a test to prove the preceding.

Coil Fit in Core Housing

CRDM coils and coil housing clearances are selected so that coil heat up results in a close to tight fit. This is done to facilitate thermal transfer and coil cooling in a hot CRDM.

3.9N.4.4 CRDS Performance Assurance Program

Evaluation of Material’s Adequacy

The ability of the pressure housing components to perform throughout the design lifetime as defined in the design specification is confirmed by the stress analysis report required by the ASME Code, Section III.

Internal components subjected to wear have withstood a minimum of 2,500,000 steps without refurbishment as confirmed by life tests (Cooper 1974).

To confirm the mechanical adequacy of the fuel assembly, the CRDM, and rod cluster control assembly (RCCA), functional test programs have been conducted on a full scale 12-foot control rod. The 12-foot prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1,000 hours. The prototype mechanism accumulated about 2,500,000 steps and 600 trips. At the end of the test the CRDM was still operating satisfactorily. A correlation was developed to predict the amplitude of flow-excited vibration of individual fuel rods and fuel assemblies. Inspection of the drive line components did not reveal significant fretting.

These tests include verification that the trip time achieved by the CRDM's meet the design requirement of 2.2 seconds from start of RCCA motion to dashpot entry. This trip time requirement will be confirmed for each CRDM prior to initial reactor operation and at periodic intervals after initial reactor operation as required by the Technical Specifications.

There are no significant differences between the prototype CRDMs and the production units. Design materials, tolerances and fabrication techniques (Section 4.2.3.3.2) are the same.

These tests have been reported in WCAP-8446 and WCAP-8449.

It is expected that all CRDM's will meet specified operating requirements for the duration of plant life with normal refurbishment.

If an RCCA cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one RCCA can be tolerated. More than one inoperable RCCA could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable RCCA's has been limited to one as discussed in the Technical Specifications (Chapter 16).

In order to demonstrate proper operation of the CRDM and to ensure acceptable core power distributions during operation, RCCA partial-movement checks are performed (Technical Specifications). In addition, periodic drop tests of the RCCA are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements, to ensure core subcriticality after reactor trip, and to limit potential reactivity insertions from a hypothetical RCCA ejection. During these tests the acceptable drop time of each assembly is not greater than

2.2 seconds, at full flow and operating temperature, from the beginning of motion to dashpot entry.

Actual experience in operating Westinghouse plants indicates excellent performance of CRDM's.

All units are production tested prior to shipment to confirm ability of the CRDM to meet design specification-operation requirements.

Each production CRDM undergoes a production test as listed below:

Test	**Acceptance Criteria**
Cold (ambient) hydrostatic pressure test	ASME Section III
Confirm step length and load transfer (stationary gripper to movable gripper or movable gripper to stationary gripper)	Step Length: 5/8 ± 0.015 inches axial movement Load Transfer: 0.059 inches nominal axial movement
	Operating Speed: 45 inches/minute
Cold (ambient) performance Test at design load -5 full travel excursions	Trip Delay: Free fall of drive rod assembly to begin within 150 milliseconds as verified by normal gripper latch opening times recorded during CRDM performance tests

3.9N.5 REACTOR VESSEL INTERNALS

3.9N.5.1 Design Arrangements

The reactor vessel internals are described as follows:

> The components of the reactor internals are divided into three parts consisting of the lower core support assembly (including the entire core barrel and neutron shield pad assembly), the upper core support assembly and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and CRDM’s, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the incore instrumentation. The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the vessel wall and then into a plenum at the bottom of the vessel. It then reverses and flows up through the core support and through the lower core plate. The lower core plate is sized to provide the

desired inlet flow distribution to the core. After passing through the core, the coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles. A small portion of the coolant flows between the baffle plates and the core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel outlet nozzles.

Lower Core Support Assembly

The major containment and support member of the reactor internals is the lower core support assembly, shown in Figure 3.9N–8. This assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support which is welded to the core barrel. All the major material for this assembly is Type 304 stainless steel. The lower core support assembly is supported at its upper flange from a ledge in the reactor vessel flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are an axial baffle and a lower core plate, both of which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core support assembly and principally the core barrel serve to provide passageways and control for the coolant flow. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a member through which the necessary flow distribution holes for each fuel assembly are located. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the core support of the core barrel in order to provide stiffness and to transmit the core load to the core support. Adequate coolant distribution is obtained through the use of the lower core plate and core support.

The neutron shield pad assembly consists of four pads that are bolted and pinned to the outside of the core barrel. These pads are constructed of Type 304 stainless steel and are approximately 48 inches wide by 148 inches long by 2.8-inches thick. The pads are located azimuthally to provide the required degree of vessel protection. Specimen guides in which material surveillance samples can be inserted and irradiated during reactor operation are attached to the pads. The samples are held in the guides by a preloaded spring device at the top and bottom to prevent sample movement. Additional details of the neutron shield pads and irradiation specimen holders are given in WCAP-7870 (1972).

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, hydraulic loads, and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the core support and thence through the core barrel shell to the core barrel flange supported by the vessel flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell and distributed between the lower radial support to the vessel wall and to the vessel flange. Transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by upper core plate alignment pins which are welded into the core barrel.

The main radial support system of the lower end of the core barrel is accomplished by “key” and “keyway” joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel clevis block is welded to the vessel inner diameter. Another Inconel insert block is bolted to each of these blocks and has a “keyway” geometry. Opposite each of these is a “key” which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam supported at the top and bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. With this system, cyclic stresses in the internal structures are within the ASME Section III limits. In the event of an abnormal downward vertical displacement of the internals following a hypothetical failure, energy absorbing devices limit the displacement after contacting the vessel bottom head. The load is then transferred through the energy absorbing devices of the internals to the vessel.

The energy absorbers base plate is contoured on its bottom surface to the reactor vessel bottom geometry. Assuming a downward vertical displacement the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.

Neutron Shield Panel Design for Millstone 3

The neutron shielding panel design of Millstone 3 consists of four sets of stainless steel plates strategically placed on the core barrel in areas of peak fast neutron flux on the reactor pressure vessel. See Figures 3.9N–11 and 3.9N–12. Attachment of each of the pad sections to the core barrel is accomplished through a series of sixteen 7/8-inch stainless steel bolts and three 2-3/8-inch stainless steel pins. The bolts are designed to resist most of the primary flow-induced and seismic normal loads; however, the pins carry all of the weight and are designed to resist the accident loads. In addition, since the pins are press fit into position, they retain a high compression-induced friction force in the pressure vessel radial direction. This enables the pins to also function as a redundant support in the pressure vessel radial direction.

The pads are divided into two sections to reduce the effects of vertical relative thermal expansion between the barrel and the pads. Six specimen baskets are utilized in this design and are positioned on the neutron pads in both tandem and single configurations. The baskets are attached to the neutron pads by sets of eight 3/4-inch stainless steel bolts and two 7/8-inch pins.

Substitution of the neutron pads for the thermal shield results in some significant design advantages, specifically:

1. The ΔP in the downcomer region (annulus between core barrel and pressure vessel) is reduced by approximately 80 percent.

2. The velocity in the downcomer region is reduced by approximately 15 percent.

3. There is a net reduction in weight of 50,000 pounds. The thermal shield weighs approximately 75,000 pounds, while the weight of the neutron pad assembly is approximately 25,000 pounds.

4. The peak fast neutron flux is slightly reduced since the steel is closer to the core.

The smaller pressure drop and lower velocity leads to a reduction in the magnitude of the exciting forces on the internals.

The lower weight increases the natural frequency of the lower internals system. This is also an aid in reducing the tendency for induced vibrations. The adequacy of the design was confirmed by both analysis and test. Since the design satisfies Section III of the ASME Code, there is assurance that sufficient margin exists in the design.

Fatigue tests were performed on the bolts to simulate the effect of loading placed on the bolts resulting from the relative thermal deflection between the neutron shielding pads and the core support barrel. The tests performed on the bolts indicate that the bolts satisfy Appendix 1-10 of ASME Section III failure criteria of 20 times expected number of cycles (18,300) or 2 times expected stress. Actually, the bolts were subjected to 370,000 cycles at double the expected operating amplitude.

To confirm that there were no deleterious effects from flow-induced vibration, flow tests were performed on a 1/24 scale flow model of the internals with neutron pads, and the results compared to similar tests with a thermal shield. The results indicated extremely low levels of vibration.

In summary, the neutron shield panel design of Millstone 3 has been shown to be structurally adequate by the following:

1. An in-depth analysis, considering loadings for all plant operating conditions, indicates that the design satisfies all the criteria of Section NG 3000 of Section III of the ASME Code.

2. Tests performed on bolts indicate that the bolts satisfy the failure criteria of 20 times expected number of fatigue cycles (18,300) or double the expected stress levels. Actually, the test data indicated that the safety factors are approximately 3 on stress and 2×10^5 on cycles.

3. Flow tests were performed on a 1/24 scale flow model of the internals with neutron pads and the results compared to similar tests with a thermal shield. The levels of vibration for the neutron shield pads were negligible.

Upper Core Support Assembly

The upper core support assembly, shown in Figures 3.9N–9 and 3.9N–10 consists of the upper support, the upper core plate, the support columns, and the guide tube assemblies. The support columns establish the spacing between the upper support and the upper core plate. They are fastened at top and bottom to these plates. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouples. The guide tube assemblies, sheath and guide the control rod drive shafts and control rods. They are fastened to the upper support and are restrained by pins in the upper core plate for proper orientation and support.

The upper core support assembly is positioned in its proper orientation with respect to the lower core support assembly by flat-sided pins in the core barrel flange. At an elevation in the core barrel where the upper core plate is positioned, four equally spaced flat-sided pins are located. Four mating sets of inserts are located in the upper core plate at the same positions. As the upper support assembly is lowered into the lower support assembly, the inserts engage the flat-sided pins in the axial direction. Lateral displacement of the plate and the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support assembly, the upper core support assembly, the fuel assemblies and control rods are thereby ensured by this system of locating pins and guidance arrangement. The upper and lower core support assemblies are restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by installation of the reactor vessel head.

Vertical loads from weight, earthquake acceleration, hydraulic loads and fuel assembly preload are transmitted through the upper core plate via the support columns to the upper support and then into the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the upper support and upper core plate. The upper support is particularly stiff to minimize deflection.

Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom (Figure 7.7–9 shows the basic flux-mapping system).

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to inline columns that are in turn fastened to the upper support. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are stainless steel tubes.

In addition to the upper incore instrumentation, there are reactor vessel bottom instrumentation columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduit tubes extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal table. The minimum bend radii are approximately 144 inches and the trailing ends of the thimbles (at the seal table) are extracted approximately 15 feet during refueling the reactor. The Conduit Tubes are classified as ASME Section III Class 1 and designed in accordance with Subsection NC. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and conduits are provided at the seal table. During normal operation, the retractable thimbles are stationary. They can move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal table is cleared for the retraction operation.

The incore instrumentation support structure is designed for support of instrumentation during reactor operation and is rugged enough to resist damage under the conditions imposed during the refueling sequence.

3.9N.5.2 Design Loading Conditions

<u>Normal and Upset Conditions</u>

The normal and upset loading conditions that provide the basis for the design of the reactor internals are:

1. Fuel and reactor internals weight
2. Fuel and core component spring forces, including spring preloading forces
3. Differential pressure and coolant flow forces
4. Temperature gradients
5. Vibratory loads including OBE seismic loads
6. Normal and upset operational thermal transients listed in Table 3.9N–1
7. Control rod trip (equivalent static load)
8. Loads due to loop(s) out of service
9. Loss of load/pump overspeed

<u>Emergency Conditions</u>

The emergency loading conditions that provide the basis for design of the reactor internals are:

1. Small LOCA
2. Small steam break
3. Complete loss of flow

Faulted Conditions

The faulted loading conditions that provide the basis for the design of the reactor internals are:

1. Branch line breaks, as determined from leak-before-break (LBB) analysis
2. SSE

3.9N.5.3 Design Loading Categories

The combination of design loadings fit into the normal, upset, emergency or faulted conditions as defined in the ASME Code, Section III.

Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from components weights, hydraulic forces and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions as summarized in Table 3.9N–1.

The scope of the stress analysis problem is very large requiring many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation under consideration.

Allowable Deflections

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For the loss of coolant accident plus the safe shutdown earthquake condition, the deflection criteria of critical internal structures are the limiting values given in Table 3.9N–13. The corresponding no loss of function limits are included in Table 3.9N–13 for comparison purposes with the allowed criteria.

The criteria for the core drop accident are based upon analyses which have to determine the total downward displacement of the internal structures following a hypothesized core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately one half inch. An additional displacement of approximately 3/4 inch would occur due to strain of the energy absorbing devices of the secondary core support; thus the total drop distance is about 1-1⁄4 inches which is insufficient to permit the tips of the rod cluster control assembly to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident of the core support (core barrel, barrel flange, etc.). There are 4 supports in each reactor. This device limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated assuming a complete and instantaneous failure of the primary core support and is absorbed during the plastic deformation of the controlled volume of stainless steel, loaded in tension.

The maximum deformation of this austenitic stainless steel piece is limited to approximately 15 percent, after which a positive stop is provided to ensure support.

3.9N.5.4 Design Bases

The design bases for the mechanical design of the reactor vessel internals components are as follows:

1. The reactor internals in conjunction with the fuel assemblies shall direct reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements are met for all modes of operation. In addition, required cooling for the pressure vessel head shall be provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

2. In addition to neutron shielding provided by the reactor coolant, a separate neutron pad assembly is provided to limit the exposure of the pressure vessel in order to maintain the required ductility of the material for all modes of operation.

3. Provisions shall be made for installing incore instrumentation useful for the plant operation and vessel material test specimens required for a pressure vessel irradiation surveillance program.

4. The core internals are designed to withstand mechanical loads arising from operating basis earthquake, safe shutdown earthquake and pipe ruptures and meet the requirement of Item 5 below.

5. The reactor shall have mechanical provisions which are sufficient to adequately support the core and internals and to assure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.

6. Following the design basis accident, the plant shall be capable of being shut down and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

The functional limitations for the core structures during the design basis accident are shown in Table 3.9N–13. To ensure no column loading or rod cluster control guide tubes, the upper core plate deflection is limited to not exceed the value shown in Table 3.9N–13.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Section 3.9N.2.

The basis for the design stress and deflection criteria is identified below:

Allowable Stresses

For normal operating conditions the intent of Section III of the ASME Code is used as a basis for evaluating acceptability of calculated stresses. Both static and alternating stress intensities are considered.

It should be noted that the allowable stresses in Section III of the ASME Code are based on unirradiated material properties. In view of the fact that irradiation increases the strength of the Type 304 stainless steel used for the internals, although decreasing its elongation, it is considered that use of the allowable stresses in Section III is appropriate and conservative for irradiated internal structures.

The allowable stress limits during the design basis accident used for the core support structures are based on the 1974 Edition of the ASME Code for Core Support Structures, Subsection NG, and the Criteria for faulted conditions.

3.9N.6 INSERVICE TESTING OF PUMPS AND VALVES

Refer to Section 3.9.7.

FIGURE 3.9N–1 REACTOR PRESSURE VESSEL AND INTERNALS SYSTEM MODEL

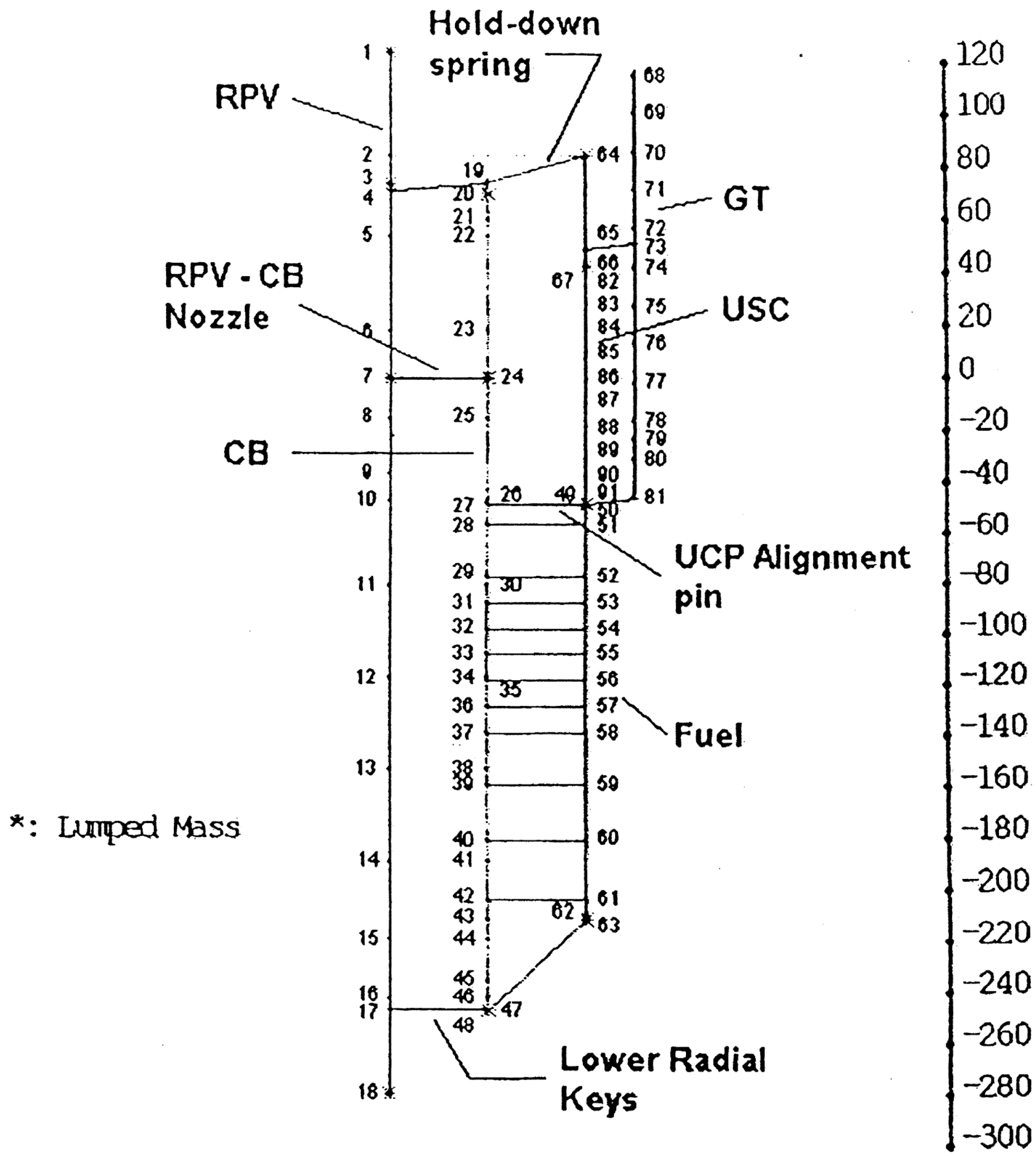


FIGURE 3.9N–2 DELETED BY PKG FSC 07-MP3-039

FIGURE 3.9N–3 RPV SUPPORT MODEL FOR WESTINGHOUSE INTERNALS RESPONSE ANALYSIS

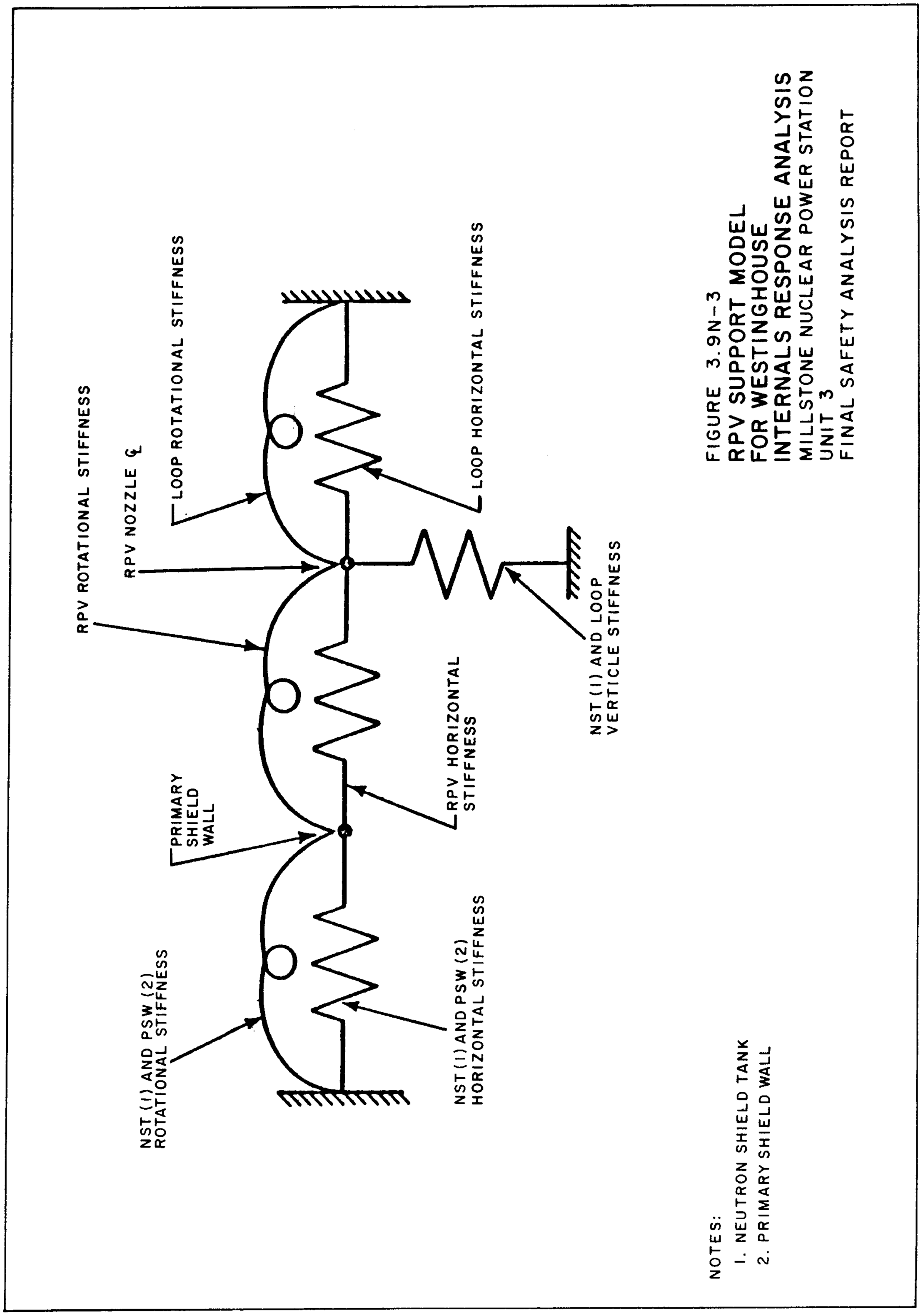


FIGURE 3.9N–4 CONTROL ROD DRIVE MECHANISM

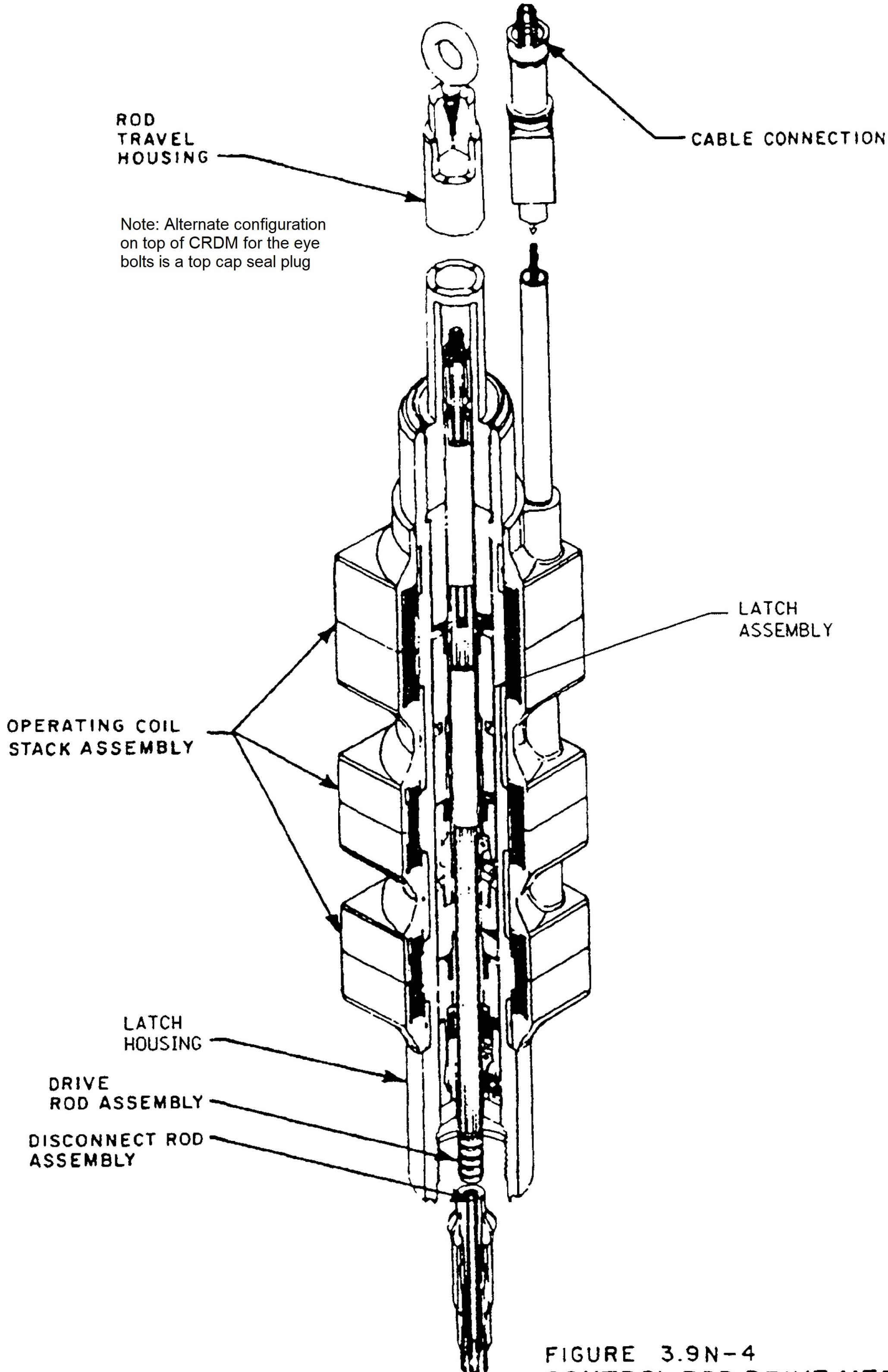


FIGURE 3.9N-4
CONTROL ROD DRIVE MECHANISM
MILLSTONE NUCLEAR POWER STATION
UNIT 3

FIGURE 3.9N–5 CONTROL ROD DRIVE MECHANISM SCHEMATIC

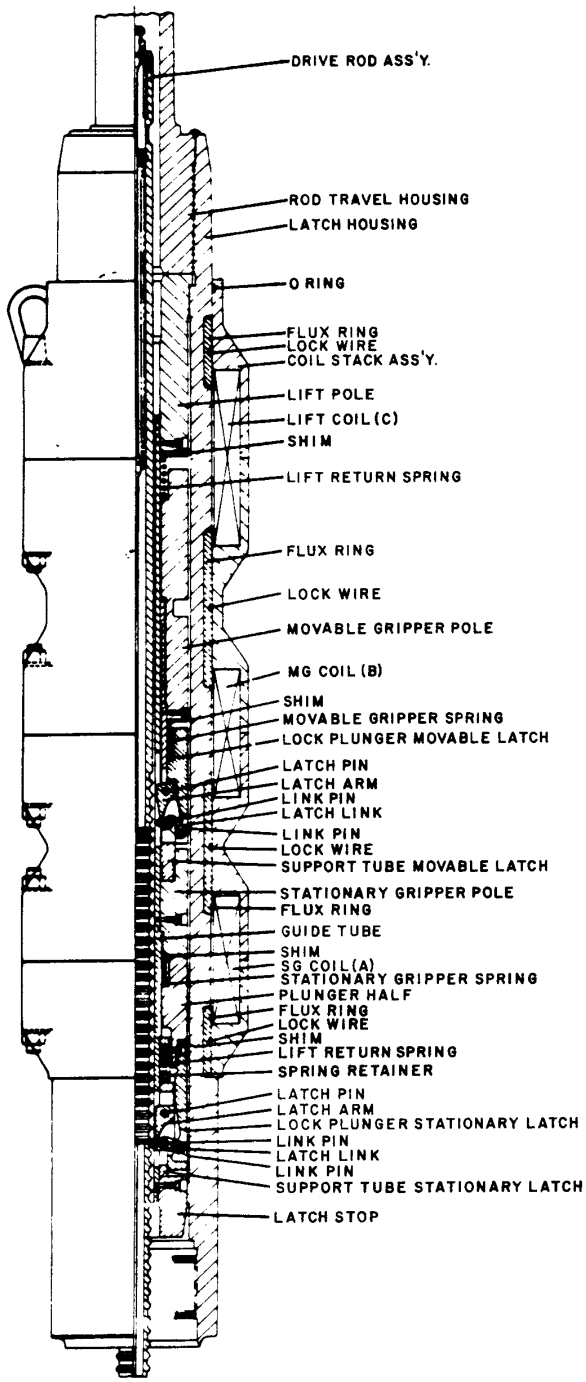
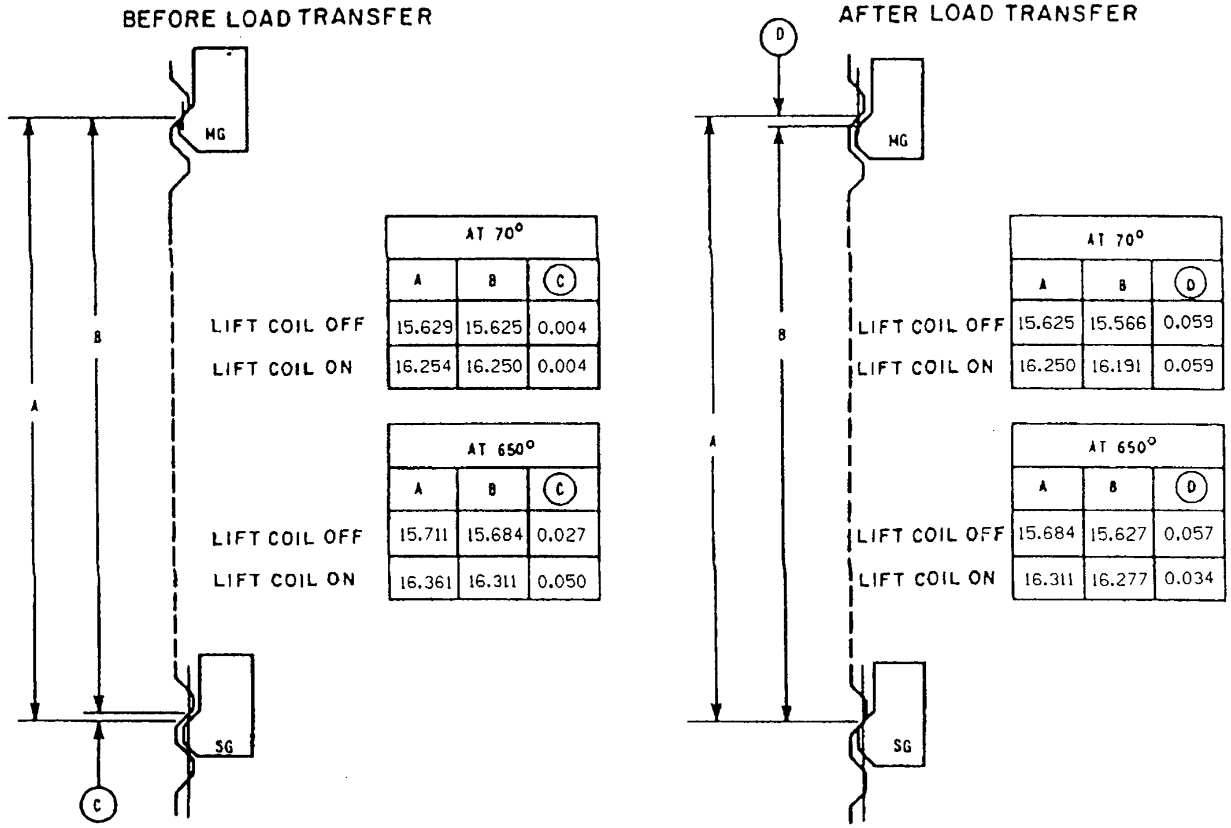


FIGURE 3.9N-5
CONTROL ROD DRIVE
MECHANISM SCHEMATIC
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.9N–6 NOMINAL LATCH CLEARANCE AT MINIMUM AND MAXIMUM CLEARANCE



BEFORE LOAD TRANSFER

	AT 70°		
	A	B	C
LIFT COIL OFF	15.629	15.625	0.004
LIFT COIL ON	16.254	16.250	0.004

	AT 650°		
	A	B	C
LIFT COIL OFF	15.711	15.684	0.027
LIFT COIL ON	16.361	16.311	0.050

AFTER LOAD TRANSFER

	AT 70°		
	A	B	D
LIFT COIL OFF	15.625	15.566	0.059
LIFT COIL ON	16.250	16.191	0.059

	AT 650°		
	A	B	D
LIFT COIL OFF	15.684	15.627	0.057
LIFT COIL ON	16.311	16.277	0.034

FIGURE 3.9N-6
NOMINAL LATCH CLEARANCE
AT MINIMUM
AND MAXIMUM CLEARANCE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
CAD FILE:39n6dgn/39n6.cit

FIGURE 3.9N–7 CONTROL ROD DRIVE MECHANISM LATCH CLEARANCE THERMAL EFFECT

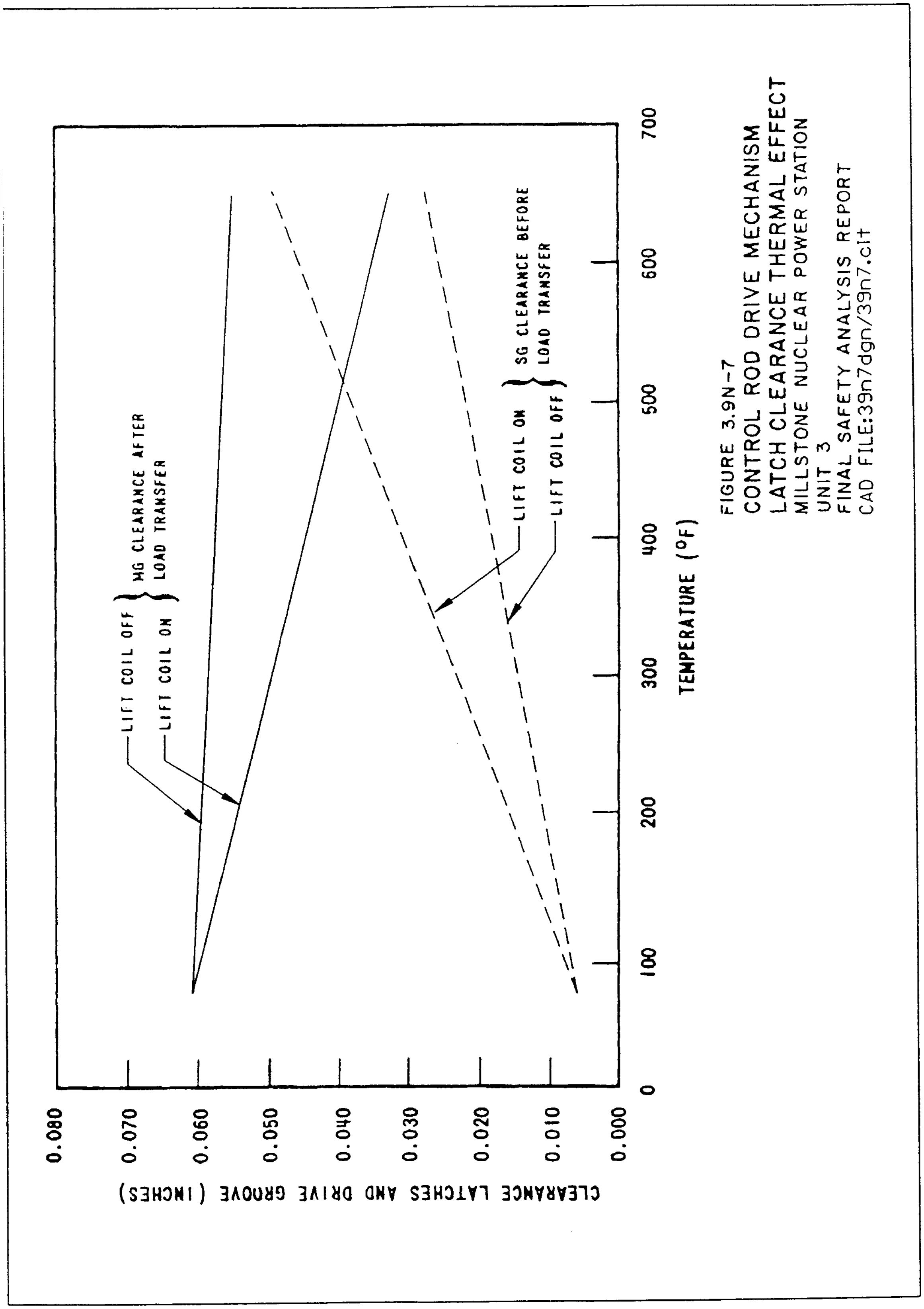


FIGURE 3.9N–8 LOWER CORE SUPPORT ASSEMBLY (CORE BARREL ASSEMBLY)

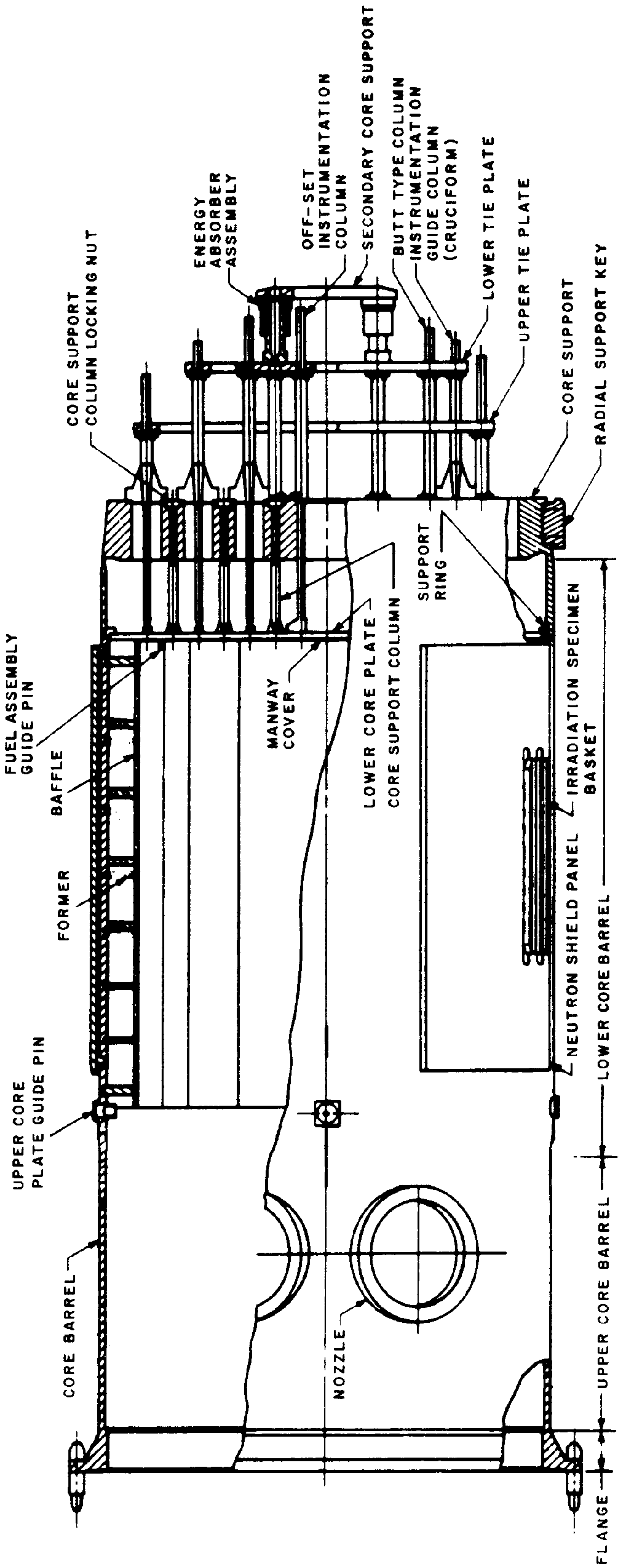


FIGURE 3.9N–9 UPPER CORE SUPPORT ASSEMBLY

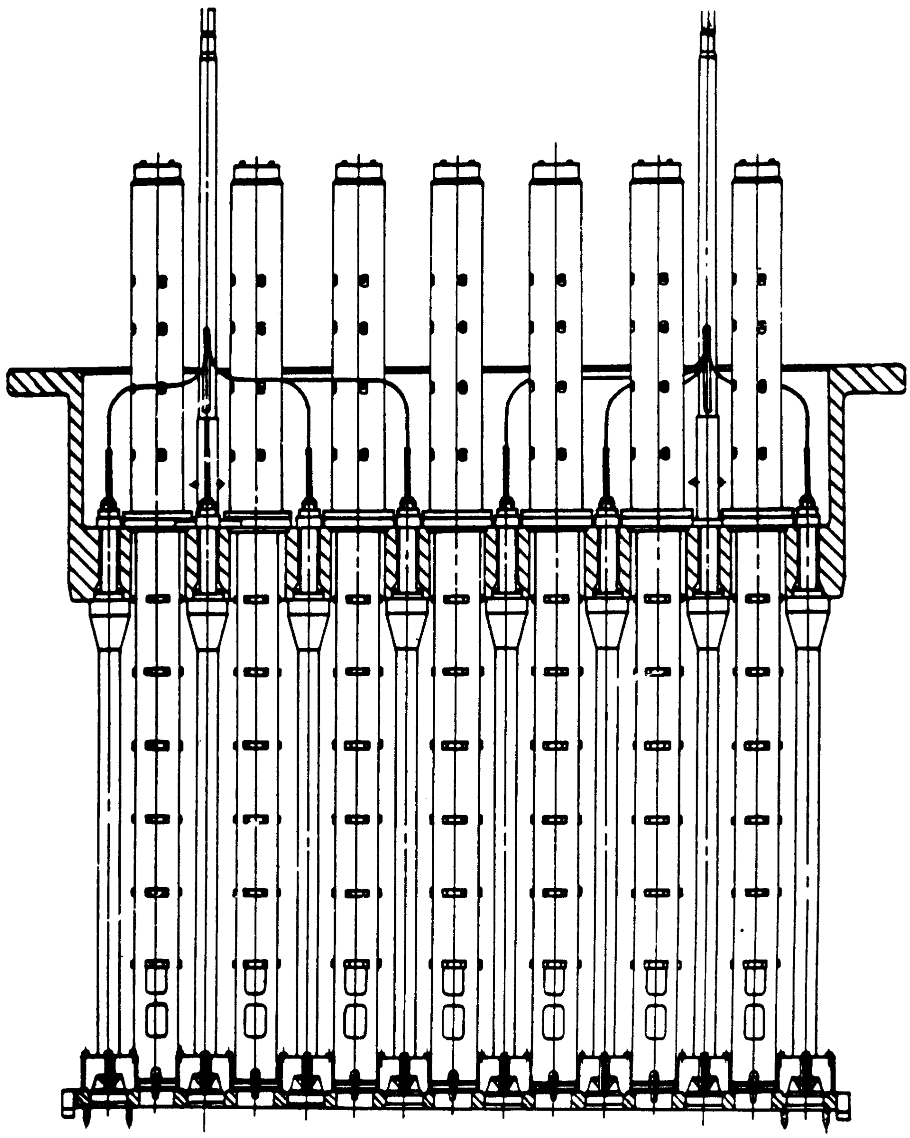


FIGURE 3.9N-9
UPPER CORE SUPPORT ASSEMBLY
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.9N–10 PLAN VIEW OF UPPER CORE SUPPORT ASSEMBLY

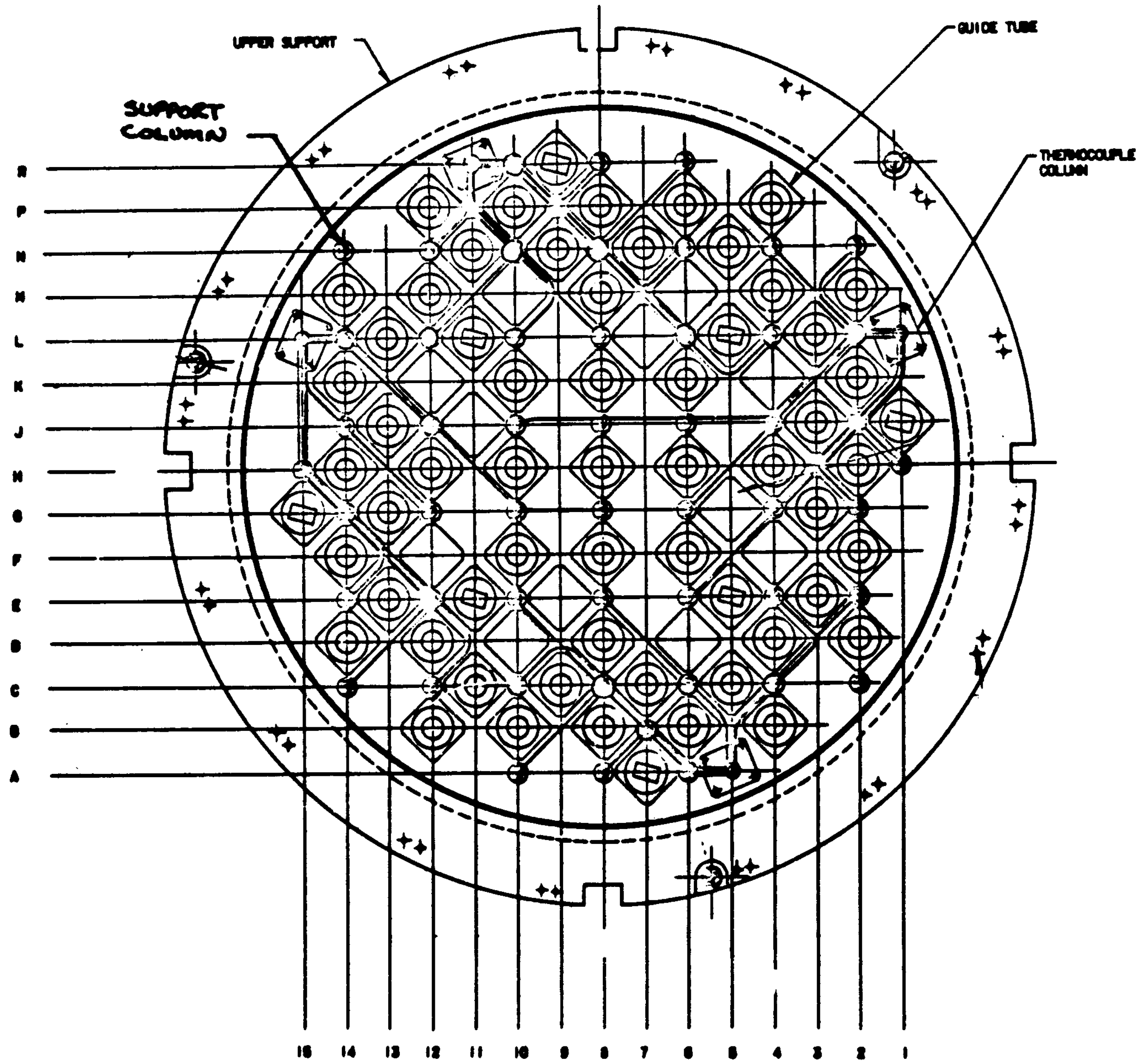


FIGURE 3.9N–11 NEUTRON SHIELD PANEL DESIGN

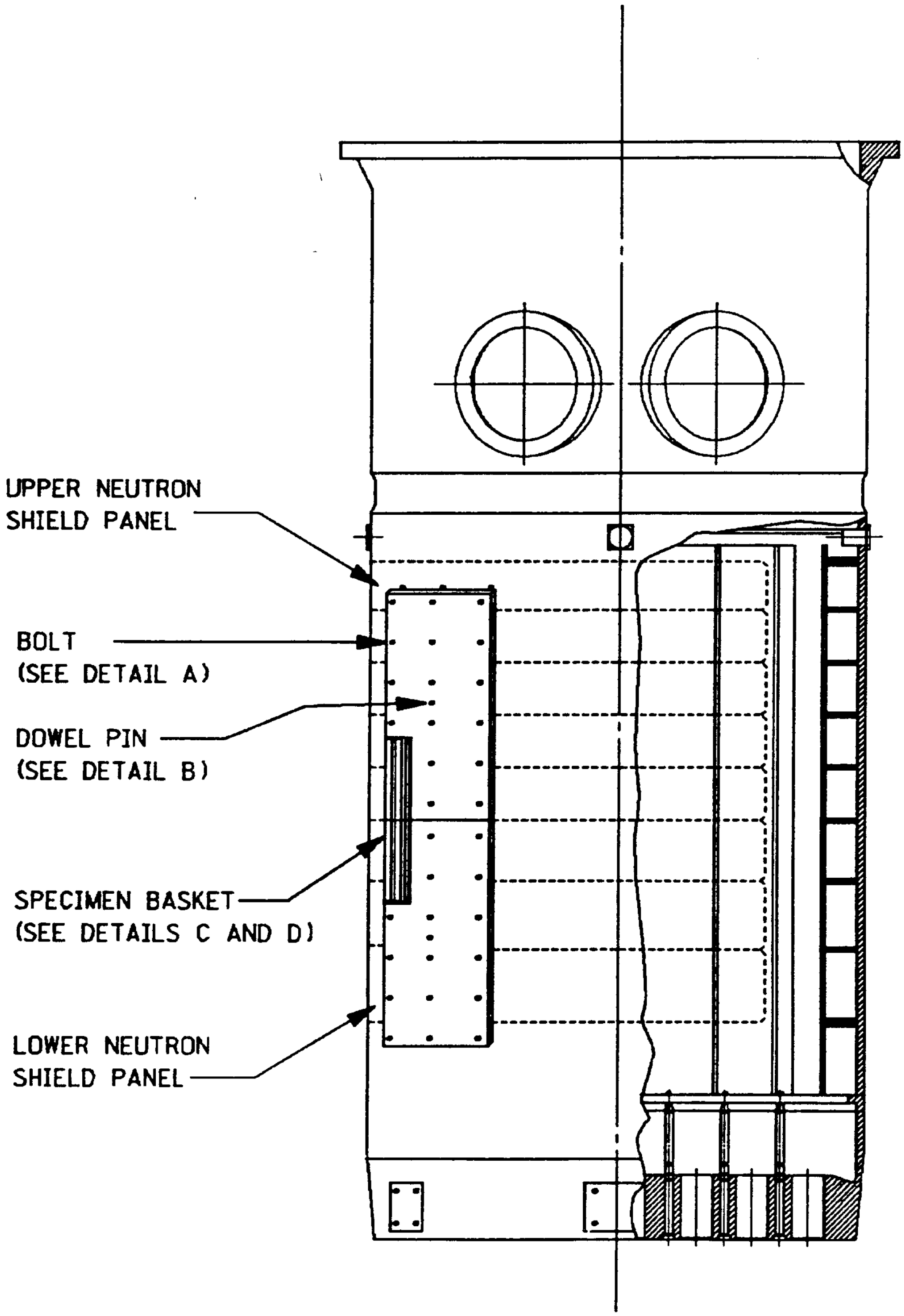
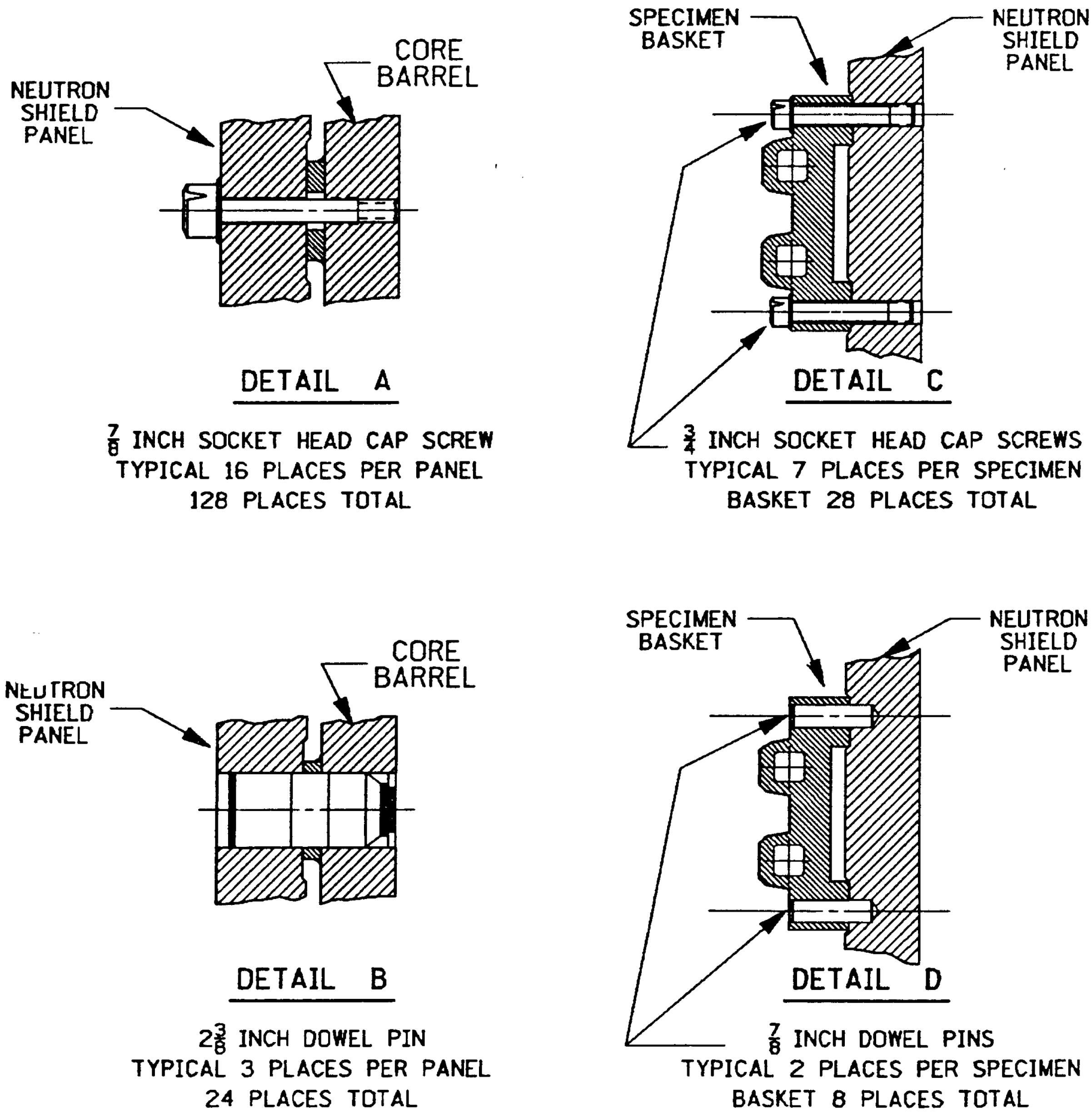


FIGURE 3.9N–12 NEUTRON SHIELD PANEL DESIGN DETAILS



3.9.7 INSERVICE TESTING OF PUMPS AND VALVES

A test program has been developed to ensure that all safety related pumps and valves will be in a state of operational readiness throughout plant life.

The ASME Code, Section XI, 1980 Edition through Winter 1980 Addendum, provided the basic rules used to identify applicable pumps and valves and to develop test requirements.

3.9.7.1 Inservice Testing of Pumps

Inservice testing is required for all Class 1, 2, and 3 pumps (both centrifugal and displacement types) that are provided with an EMERGENCY POWER SOURCE. Drivers are excluded except when the pump and driver form an integral unit and the pump bearings are in the driver.

All tests and examination procedures required by the ASME code for Operation and Maintenance of Nuclear Power Plants including schedules, reference values, the location, and type of measurement for each of the required test quantities, records of the results, and all corrective action taken are defined in the Inservice Test Pump and Valve Program and performed by the owner for Millstone 3.

3.9.7.2 Inservice Testing of Valves

This section describes the Class 1, 2, and 3 valves required to be exercised and tested to verify operational readiness. These valves (with their actuating and position indicating devices) are required to perform a special function in bringing a reactor to cold shutdown condition or in mitigating the consequences of an accident.

Valves used for operating convenience only such as manual vent, drain, instrument and test valves, and valves used for maintenance only do not require inservice testing. The following categories of valves are subject to inservice testing:

1. Category A - Valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their function.

2. Category B - Valves for which seat leakage in the closed position is inconsequential for fulfillment of their function.

3. Category C - Valves which are self-actuating in response to some system characteristic, such as pressure (relief valves) or flow direction (check valves).

4. Category D - Valves which are actuated by an energy source capable of only one operation, such as rupture disks or explosive actuated valves.

Test and examination procedures required by Subsection IWV, including schedules and the limiting values of observed parameters are defined in the Inservice Test Pump and Valve Program and performed by the licensee for Millstone 3.

3.9.8 REFERENCES FOR SECTION 3.9

3.9-1 Bohm, G.J. and LaFaille, J.P. 1971. Reactor Internals Response Under a Blowdown Accident, Procedures, First Intl. Conf. on Structural Mech. in Reactor Technology. Berlin, September 20-24, 1971.

3.9-2 DeSalvo, G.J. and Swanson, J.A. 1972. ANSYS User's Manual. Engineering Analysis Systems Report, October 1, 1972.

3.9-3 Fabic, S. 1967. Computer Program WHAM for Calculation of Pressure, Velocity, and Force Transients in Liquid Filled Piping Networks. Kaiser Engineers Report No. 67-49-R.

3.9-4 Trojan Final Safety Analysis Report, Appendix A-12 (Docket No. 50-344).

3.9-5 WCAP-7870, 1972, Kraus, S. "Neutron Shielding Pads."

3.9-6 WCAP-8252, Revision 1, 1977, "Documentation of Selected Westinghouse Structural Analysis Computer Codes."

3.9-7 WCAP-8303-P-A (Proprietary) and WCAP-8317-A (Non-Proprietary), 1975, "Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests."

3.9-8 WCAP-8446 (Proprietary) and WCAP-8449 (Non-Proprietary), 1974, Cooper, F.W. Jr. 1974. 17x17 Drive Line Components Tests - Phase IB, II, III, D-Loop Drop and Deflection.

3.9-9 WCAP-8516-P (Proprietary) and WCAP-8517 (Non-Proprietary), 1975. Bloyd, C.N. and Singleton, N.R. UHI Plant Internals Vibration Measurement Program and Pre and Post Hot Functional Examinations.

3.9-10 WCAP-8708-P-A (Proprietary) and WCAP-8709-A (Non-Proprietary), 1977. Takeuchi, K. "MULTIFLEX-A Fortran-IV Computer Program for Analyzing Thermal - Hydraulic - Structure System Dynamics."

3.9-11 WCAP-8780, 1976, Bloyd, C.N.; Ciaramitaro, W.; and Singleton, N.R., "Verification of Neutron Pad and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant."

3.9-12 WCAP-8929, 1977, "Benchmark Problem Solutions Employed for Verification WECAN Computer Program."

3.9-13 WCAP-9945, 1981. Altman, D.A. et al., "Verification of Upper Head Injection Reactor Vessel Internals by Preoperational Tests on the Sequoyah Unit 1 Power Plant."

TABLE 3.9B–1 LIST OF INPUT DOCUMENTS DESCRIBING DESIGN TRANSIENT FOR FATIGUE ANALYSIS OF RCP AND ASSOCIATED CLASS 1 PIPING

Piping System Description	MPS-3 Piping Drawing Series	SWEC (MPS-3) Design Transient Report	SWEC (MPS-3) Piping Stress Analysis Problem Number
Reactor coolant loops (including surge attachments to coolant loops)	EP 70	TR 2658-2	7000 7001 7002 7003
Residual heat removal piping	EP 71	TR 2658-3	7000 7001
Chemical volume and control system	EP 74 and part of EP107 & 108	TR 2658-4	7419 7420 7422 7423 7425 7427
Low pressure safety injection system	EP 82	TR 2658-5	7000 7001 7002 70013
Annulus piping (contains portions of high pressure safety injection, auxiliary pressurizer spray and chemical volume and control)	EP 107	TR 2658-6	10700 10701 10702 10703 10704 10706 10707 10717 10729
Cubicle piping (includes letdown)	EP 108	TR 2658-7	10800 10802 10803
Pressurizer spray, safety and relief	EP 109	TR 2658-8	10900 10901

TABLE 3.9B–2 LIST OF MPS-3 DOCUMENTS DESCRIBING COMPONENTS REQUIRING INELASTIC ANALYSIS

Component Description	Associated MPS-3 PPG System & Component Location	Document Providing Details of Analysis
3 inch Charging Nozzle on RC Loop	RC Loop 1 & 3RCS-003-149-1 EP 74B (H-8)	Teledyne Engineering Services Report TR 2658-21
3 inch Charging Nozzle on RC Loop	RC Loop 4 & 3RCS-003-145-1 EP 74B (E-8)	Teledyne Engineering Services Report TR 2658-21
3 inch Safety Injection Nozzle on RC Loop	RC Loop 1 & 3RCS-003-121-1 EP 108A (I-4)	Teledyne Engineering Services Report TR 2658-22
3 inch Safety Injection Nozzle on RC Loop	RC Loop 2 & 3RCS-003-133-1 EP 108B (G-7)	Teledyne Engineering Services Report TR 2658-22
3 inch Safety Injection Nozzle on RC Loop	RC Loop 3 & 3RCS-003-139-1 EP 108C (C-7)	Teledyne Engineering Services Report TR 2658-22
3 inch Safety Injection Nozzle on RC Loop	RC Loop 4 & 3RCS-003-147-1 EP 108D (C-5)	Teledyne Engineering Services Report TR 2658-22
12 sets of Circumferential Butt Welds (as welded)		
Sets 1 through 3	Low Pressure Safety Injection Piping to RC Loop 10 inch Schedule 140 Butt Welds	SWEC Calculation 12179-NP(B)-315-XI
Sets 4 & 5	Boron Injection System 1.5 inch Schedule 160 Butt Welds	SWEC Calculation 12179-NP(B)-317-XI
Sets 6 through 10	RCS Drains Loops 1, 2, 3 & 4 2 inch Schedule 160 Butt Welds	SWEC Calculation 12179-NP(B)-299-XI
Set 11	Pressurizer Spray Nozzle C-1 Weld 3RCS*TK1 3RCS-004-224-1	SWEC Calculation 12179-NP (F)-316-XI
Set 12	Bimetallic Welds between Steam Generator and RCS Loop	SWEC Calculation 12179-NP(B)-4019-XI

TABLE 3.9B–3 PREOPERATIONAL TESTS [1]

System Code	System Title	Reg. Guide 1.68, Rev. 2 Classification [3]	Types of Tests [2]		
			Thermal Expansion	Transient Vibrations	Steady State Vibrations
PGS	Primary Grade Water	A, D	NR	NR [4]	V [5]
IAS	Instrument Air	A, D	NR	NR	NR
GSN	Nitrogen System	A, B	NR	NR	V
CCE	Charging Pump Cooling	A	NR	NR	V
SWP	Service Water	A, D	NR	NR [6]	V
WTC	Water Treating - Chlorination	A	NR	NR	NR
CHS	Chemical and Volume Control	A, B, A1	V & I [7]	V	V
CCP	Reactor Plant Component Cooling	A, D	NR	NR	V
EGF	Emergency Diesel Fuel	A, A1	NR	NR	V
CDS	Chilled Water	A, D	NR	NR	V
EGA	Air Startup Emergency Diesel	A	V	NR	V
EGD	Emergency Generator Exhaust and Combustion Air	A	V	NR	V
EGS	Emergency Diesel Jacket and Intercooler Water	A	NR	NR	V
MSS	Main Steam	A, B	V & I	V & I	V
DTM	Turbine Plant Miscellaneous Drains	A	V	NR	V
QSS	Quench Spray	A, D	NR	NR	V [8]
CCI	Safety Injection Pump Cooling	A, A1	NR	NR	V

TABLE 3.9B–3 PREOPERATIONAL TESTS [1] (CONTINUED)

System Code	System Title	Reg. Guide 1.68, Rev. 2 Classification [3]	Types of Tests [2]		
			Thermal Expansion	Transient Vibrations	Steady State Vibrations
HVC	Air Conditioning - Control Building	A	NR	NR	NR
HVK	Chilled Water - Control Building	A	NR	NR	V
SIH	Safety Injection - High Pressure	A, B	NR	V	V
FWS	Feedwater	A, B	V & I	V & I	V
FWA	Auxiliary Feedwater and Recirculation	A, B, A1	NR	NR	V
RHS	Residual Heat Removal	A, B, D	V & I	NR	V
SIL	Safety Injection - Low Pressure	A, B	V & I	NR	V
RCS	Reactor Coolant Main Loops	A	V & I	V & I	V
BDG	Steam Generator Blowdown	A, B	V & I	V	V
DAS	Reactor Plant Aerated Drains	A, D	V	NR	V
SGF	Steam Generator - Chemical Feed	A	NR	NR	V
RSS	Containment Recirculation Spray	A, B, C	NR	NR	V [8]
SSR	Sampling System - Reactor Plant	A	V	NR	NR
HVU	Ventilation - Containment Structure	A	NR	NR	NR
SFC	Fuel Pool Cooling and Purification	A, D	NR	NR	V
HCS	Hydrogen Recombiner	A, A1	NR	NR	V

TABLE 3.9B–3 PREOPERATIONAL TESTS (1) (CONTINUED)

System Code	System Title	Reg. Guide 1.68, Rev. 2 Classification (3)	Types of Tests (2)		
			Thermal Expansion	Transient Vibrations	Steady State Vibrations
SSP	Sampling System - Post Accident	A	V	NR	NR
GWS	Radioactive Gaseous Waste (9)	A	V	NR	NR
VRS	Reactor Plant Gaseous Vents	A, D	V	NR	NR
LMS	Containment Leakage Monitoring	A	NR	NR	NR
DGS	Reactor Plant Hydrogenated Drains	A	V	NR	NR
CMS	Containment Atmosphere Monitoring	A	NR	NR	NR
CVS	Containment Vacuum	A, B, D	V	NR	V
ICI	Incore Instrument Lines	A, B	NR	NR	NR
RCS	Pressurizer Safety and Relief System	A, B	V & I	V & I	V
RCS	Pressurizer Spray System	A, B	V & I	NR	V
FPW	Fire Protection Water	D	NR	NR	NR
DWS/ PBS	Domestic Water/ Sanitary System	D	NR	NR	NR
SAS	Service Air Containment Service Air	D	NR	NR	NR
SVV	Main Steam Safety Valve Steam Vents and Drains	B	V	NR	NR
WSS	Radioactive Solid Waste	D	NR	NR	NR

TABLE 3.9B–3 PREOPERATIONAL TESTS [1] (CONTINUED)

System Code	System Title	Reg. Guide 1.68, Rev. 2 Classification [3]	Types of Tests [2] Thermal Expansion	Transient Vibrations	Steady State Vibrations
VAS	Nuclear Aerated Vents	D	NR	NR	NR
LWS	Radioactive Liquid Waste	B	V	NR	NR
BRS	Boron Recovery	B, D	V	NR	V
NSS	Neutron Shield Tank Cooling System	A	NR	NR	NR
ASS	Auxiliary Steam	B	V	NR	V

NOTES:

1. The detailed test plans shall identify those portions of each system to be tested.
2. Type of tests reflect the graded approach.
3. NRC Regulatory Guide 1.68, Revision 2 Classifications:

   A= ASME III, High Energy Piping, Classes 1, 2, and 3.

   A1= ASME III, Moderate Energy Piping, Classes 1, 2, and 3.

   B = Other ASME III, High Energy Piping Inside Seismic Category I Structure.

   C = Other Non-ASME III, High Energy Outside Seismic Category I Structures whose failure could reduce the functioning of any seismic Category I plant feature to any unacceptable level. Since there are no seismic category plant features outside Seismic Category I Structures, this classification does not exist for Millstone 3.

   D = Seismic Category I portions of Moderate Energy Piping located outside containment structures.
4. NR = Testing not specifically required. Temperature < 200°F and/or significant vibrations are not expected; therefore testing testing will not be practical. Systems in this category will be observed by testing personnel for any evidence of concern for steady-state vibration.
5. V = Visual. When V appears in the table, hand held instruments may be required to perform observations.
6. I = Instrumented measurements. When V & I appears in the table, only a portion of the system requires instrument observations.
7. Fluid transient source was eliminated during initial system checkout (Phase II testing) by addition of an open-to-air vent. No retesting required during preoperational or initial startup testing program.
8. Only that portion of system isolatable from the spray rings.
9. Radioactive Gaseous Waste (GWS) System is non-seismic for design and analysis purposes.

TABLE 3.9B–4 OMITTED

TABLE 3.9B–5 STRESS LIMITS FOR ASME SECTION III CLASS 1 (NB) SEISMIC CATEGORY I COMPONENTS (ELASTIC ANALYSIS)

Pressure Vessels, Pumps and Valve Bodies - Pressure Boundary - Designed by Analysis (1)

Condition of Design	Reference Paragraph ASME Section III	Primary Stress Limits			Secondary Stress Limits		Peak Stress Limits
		P_m	P_L	$P_L + P_b$	P_e	$P_L + P_b + P_e + Q$	$P_L + P_b + P_e + Q + E$
Design (2)	NB-3221	S_m	1.5 S_m	1.5 S_m		Not Required	Not Required
Normal (3)	NB-3222	(4)	(4)	(4)	3 S_m	3 S_m	S_a
Upset (3)	NB-3223	(4)	(4)	(4)	3 S_m	3 S_m	S_a
Emergency (3) (5)	NB-3224	Greater of 1.2S_m or 1.0S_y	Greater of 1.8S_m or 1.5S_y	Greater 1.8S_m or 1.5S_y		Not Required	Not Required
Faulted (3) (4)	NB-3225, NB-3221, App. F F 1323.1	Lesser of 2.4S_m or 0.7S_u	Lesser of 3.6S_m or 1.05S_u	Lesser of 3.6S_m or 1.05S_u		Not Required	Not Required
Testing (6)	NB-3226	0.90 S_y		1.35S_y		Not Required	Not Required

NOTES:

1. The nomenclature, conditions, and applications of the above allowables are in accordance with ASME Section III. Stress limits above apply to design by elastic analysis. Limit and plastic analysis is allowed in accordance with ASME Section III criteria. Special stress limits of NB 3227 as applicable. Stress limits of Subsection NF are used for the design of supports as applicable.
2. Use design loads.
3. Use operating loads.
4. Primary stresses often evaluated and combined with secondary effects.
5. Use above limits for materials of Table 1-1.2 (ASME Section III). Use 0.7S_u for materials of Table 1-1.1 (ASME Section III).
6. Use test loads (pressure, temperature).

TABLE 3.9B–6 COMPARISON OF CLASS 1 REQUIREMENTS REGULATORY GUIDE 1.48 VS. TABLES 3.9B–5 AND 3.9B–10

Component	Normal or Upset + OBE	Emergency	Normal + Faulted + SSE	Regulatory Position	Comparison With Regulatory Position
Vessels[1]	NB-3223	NB-3224	NB-3225	C.1	Agree
Pipe [1]	NB-3654	NB-3655	NB-3656	C.1	Agree
Nonactive Pumps, Valves (Design by Analysis) [2]	NB-3223	NB-3224	NB-3225	C.2	Agree
Nonactive Valves (Design by Standard or Alternate Design Rules)	$\leq 1.1\ P_r$	$\leq 1.2\ P_r$	$\leq 1.5\ P_r$	C.3	Agree
Active Pumps, Valves (Design by Analysis) [2][3]	NB-3222	NB-3222	NB-3222	C.4	Alternate acceptable basis, exception to footnotes [3] and [4]
Active Valves (Design by Standard or Alternate Design Rules) [3]	$\leq 1.0\ P_r$	$\leq 1.0\ P_r$	$\leq 1.0\ P_r$	C.5	Agree with pressure rating factors, exception to footnote [3]

NOTES: (from Regulatory Guide 1.48, pages 1.48-7 and -8)

(1) Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code including 1972 Winter Addenda thereto.

(2) The requirements of Case 1552 (Interpretations of ASME Boiler and Pressure Vessel Code) should be met for all sizes of Code Class 1 valves designed by analysis.

(3) In addition to compliance with design limits specified, assurance of operability under all design loading combinations should be provided by an appropriate combination of the following suggested measures:

In situ testing (e.g., preoperational testing after the component is installed in the plant)

(a) Full-scale prototype testing

(b) Reduced-scale prototype testing

(c) Detailed stress and deformation analyses (includes experimental stress and deformation analyses)

In the performance of tests or analyses to demonstrate operability, the structural interaction of the entire assembly (e.g., the valve-operator assembly and pump-motor assembly) should be considered. If superposition of test results for other than the combined loading condition is proposed, the applicability of such a procedure should be demonstrated. The design limits for nonactive pumps and valves designed by analysis for applicable loading combinations if assurance is provided by detailed stress and deformation analyses that operability is not impaired when designed to these limits. Similarly, the primary-pressure ratings P for nonactive valves designed by standard or alternative design rules may be used for the applicable loading combinations if appropriate testing demonstrates that operability is not impaired when the valve is so rated.

(4) Secondary affects (stresses and deformations) should be evaluated for the loading combinations designated by Regulatory Positions 4.a.(2) and 4.a.(3). Local affects (peak stresses) need not be considered for these loading combinations.

(5) Applies to all components (vessels, piping, pumps, and valves) that are relied upon to cope with the effects of specified plant conditions.

(6) Identification of the specific transients or events to be considered under each plant condition will be addressed in a future regulatory guide.

(7) The provisions of NB-3411 and NB-3413 may be applied for all sizes of Code Class 1 pumps designed by analysis.

(8) Table 1.3-0, “Permanent Strain Limiting Factors” of Appendix I of the ASME Boiler and Pressure Vessel Code, Section III, may be used as an aid in determining the relationship between design stress and deformation (see Note 2 to Table I-1.2 of Section III of the ASME Code).

TABLE 3.9B–7 STRESS LIMITS FOR ASME SECTION III CLASS 2 AND 3 COMPONENTS (ELASTIC ANALYSIS)

Design Condition	ASME III Class Code	Primary Stress Limits (1)	
		Membrane (P_m)	Membrane Plus Bending ($P_m + P_b$)
Pressure Vessels			
I	2(NC3300) or	1.1 S	1.65 S
II	3(ND3300)	2.0 S	2.40 S
I (2)	2(NC3200)	1.1 S_m	1.65 S_m
II (3)		2.0 S_m	2.40 S_m
Pumps (4) (5), Inactive			
I	2(NC3400) or	1.1 S	1.65 S
II	3(ND3400)	2.0 S	2.40 S
Pumps (4)(5), Active			
I	2(NC3400) or	1.0 S	1.50 S
II	3(ND3400)	1.2 S	1.80 S
Valves (5)(6), Active and Inactive			
I	2(NC3500) or	1.1 S	1.65 S
II	3(ND3500)	2.0 S	2.40 S
Tanks (5) (Steel)			
I	2(NC38-3900) or	1.1 S	1.65 S
II	3(ND38-3900)	2.0 S	2.40 S

NOTES:

(1) S - Allowable stress values at design temperature from ASME Section III, Appendix I, as allowed by class

Sm - Design stress intensity values at design temperature from ASME Section III, Appendix I, as allowed by class

(2) Fatigue analysis may be required with operating conditions, reference paragraph NC-3219 and Appendix XIV of ASME Section III, Subsection NC.

(3) When a complete analysis is performed in accordance with NC 3211.1(c), the faulted stress limits of Appendix F shall apply.

(4) In accordance with NC-3400 and ND-3400, any design method which has been demonstrated to be satisfactory for the specified design conditions may be used.

(5) Stress limits of ASME Section III, Subsection NF, are used for the design of supports as applicable.

(6) The standard or alternative design rules of NC-3500 and ND-3500 may be used in conjunction with the stress limits specified.

Valve nozzle (piping load) stress analysis is not required when both the following conditions are satisfied by calculation:

(a) Section modulus and area at the plane normal to the flow passage through the region at the valve body crotch is at least 110 percent of that for the piping connected (or joined) to the valve body inlet and outlet nozzles; and,

(b) Code allowable stress, S_{vlv}, for valve body material, is equal to or greater than the code allowable stress, S_{pip}, of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in (a) above shall be multiplied by the ratio of the allowable stress for the pipe divided by the allowable stress of the valve.

If unable to comply with these requirements, the design by analysis procedure of NB-3545.2 is an acceptable alternative method.

Casting quality factor of 1.0 shall be used.

Design requirements listed in this table are not applicable to valve discs, stems, cast rings, or other parts of valves which are contained within the confines of the body and bonnet.

TABLE 3.9B–8 COMPARISON OF CLASSES 2 AND 3 REQUIREMENTS REGULATORY GUIDE 1.48 VS TABLE 3.9B–7

	Regulatory Guide 1.48				Table 3.9B–7			
		Design Limits		Regulatory Position	Loading Combinations	Design Limits		
Components	Loading Combinations	P_m	P_m(or P^L)+P_b			P_m	P_m(or P_1)+P_b	Comparison
Pressure Vessels Classes 2 and 3	Normal	1.1 S	1.65 S	C.6a	Design+OBE	1.1 S	1.65 S	Acceptable alternate, Code Case 1607, Regulatory Guide 1.84
	Upset+OBE	1.1 S	1.65 S	C.6a				
	Emergency	1.1 S	1.65 S	C.6a	Design+SSE	2.0 S	2.40 S	
	Faulted + SSE	1.5 S	2.25 S	C.6b				
Piping Classes 2 and 3	Normal				Normal			Acceptable alternate, Code Case 1606, Regulatory Guide 1.84
	Upset+OBE	NC-3611.1 (b)(4)(c) (b)(1)	1.2 S_h	C.8.a	Upset +OBE	NC-3611.1 (b)(4)(c) (b)(1)	1.2 S_h	
	Emergency				Emergency	NC-3611.1 (b)(4)(c) (b)(2)	1.8 S_h	
	Faulted+SSE	NC-3611.1 (b)(4)(c) (b)(2)	1.8 S_h	C.8.b	Faulted+SSE		2.4 S_h	
Pumps Classes 2 and 3								Acceptable alternate, Code Case 1636, Regulatory Guide 1.84
Inactive	Normal	1.1 S	1.65 S		Design+OBE	1.1 S	1.65 S	
	Upset+OBE	1.1 S	1.65 S	C.9.a				
	Emergency	1.1 S	1.65 S		Design+SSE	2.0 S	2.40 S	
	Faulted+SSE	1.2 S	1.8 S	C.9.b				

TABLE 3.9B–8 COMPARISON OF CLASSES 2 AND 3 REQUIREMENTS REGULATORY GUIDE 1.48 VS TABLE 3.9B–7

	Regulatory Guide 1.48				Table 3.9B–7			
		Design Limits		Regulatory Position	Loading Combinations	Design Limits		
Components	Loading Combinations	P_m	$P_m(\text{or } P^L)+P_b$			P_m	$P_m(\text{or } P_1)+P_b$	Comparison
Active	Normal							Acceptable alternate program
	Upset+OBE	S	1.5 S	C.10.a	Design+OBE	1.0 S	1.5 S	
	Emergency							
	Faulted+SSE				Design+SSE	1.2 S	1.8 S	
Valves Classes 2 and 3								Acceptable alternate, Code Case 1635, Regulatory Guide 1.84
Inactive	Normal				Design+OBE	1.1 S	1.65 S	
	Upset+OBE	1.1 P_r		C.11.a				
	Emergency				Design+SSE	2.0 S	2.4 S	
	Faulted+SSE	1.2 P_r		C.11.b		including P_r		
Active	All Conditions	1.0 P_r		C.12 a		Same as for inactive		Acceptable alternate program

NOTES from Regulatory Guide 1.48, pages 1.48-7 and -8 Rev. 0, May 1973):

(1) For loadings designated in Regulatory Position 8.a(2), only Equation 9 of NC-3651 need be met.

(2) In addition to compliance with the design limits specified, assurance of operability under all design loading combinations should be provided by any appropriate combination of the following suggested measures:

(a) In situ testing (e.g., preoperational testing after the component is installed in the plant)

(b) Full-scale prototype testing

(c) Reduced-scale prototype testing

(d) Detailed stress and deformation analyses (includes experimental stress and deformation analyses)

In the performance of tests or analyses to demonstrate operability, the structural interaction of the entire assembly (e.g., valve-operator and pump-motor assembly) should be considered. If superposition of test results for other than the combined loading condition is proposed, the applicability of such a procedure should be demonstrated. The design limits for nonactive pumps ad valves may be used for the applicable loading combinations if appropriate analyses and/or testing confirms that operability is not impaired when designed to these limits.

TABLE 3.9B–9 STRESS LIMITS FOR ASME SECTION III CLASS 1, 2, AND 3 COMPONENT SUPPORTS*

Condition of Design	Plate and Shell			Linear Type			Component Standard Type		
	Class 1 (NF3220)	**Class 2 (NF3320)**	**Class 3 (NF3400)**	**Class 1 (NF3230)**	**Class 2 (NF3330)**	**Class 3 (NF3400)**	**Class 1 (NF3240)**	**Class 2 (NF3340)**	**Class 3 (NF3400)**
Design	NF3221	NF3321.1	NF3321.1	NF3231.1a	NF3231.1a	NF3231.1a	NF3221	NF3321.1	NF3321.1
							or	or	or
	NF3226			App. XVII	App. XVII	App. XVI	NF3231.1a	NF3231.1a	NF3231.1a
							App. XVII	App. XVII	App. XVII
Normal	NF3222	NF3221.2	NF3321.2	NF3231.1a	NE3231.1a	NF3231.1a	NF 3222	NF3321.2	NF3321.2
							or	or	or
	NF3226	NF3321.1	NF3321.1	App. XVII	App. XVII	App. XVII	NF3231.1a	NF3231.1a	NF3231.1a
							App. XVII	App. XVII	App. XVII
Upset	NF3223	NF3321.2	NF3321.2	NF3231.1a	NF3231.1a	NF3231.1a	NF3223	NF3321.2	NF3321.2
							or	or	or
	NF3222	NF3321.1	NF3321.1	App. XVII	App. XVII	App. XVII	NF3231.1a	NF3231.1a	NF3231.1a
	NF3226						App. XVII	App. XVII	App. XVII
Emergency	NF3224	NF3321.3	NF3321.3	NF3231.1b	NF3231.1b	NF3231.1b	NF3224	NF3321.3	NF3321.3
							or	or	or
							NF3231.1b	NF3231.1b	NF3231.1b
Faulted	NF3225	NF3321.4	NF3321.4	NF3231.1c	NF3231.1c	NF3231.1c	NF3231.1c	NF3321.4	NF3321.4
	App. F1320			App. F1370	App. F1370	App. F1370	App. F1370	or	or

TABLE 3.9B–9 STRESS LIMITS FOR ASME SECTION III CLASS 1, 2, AND 3 COMPONENT SUPPORTS* (CONTINUED)

Condition of Design	Plate and Shell			Linear Type			Component Standard Type		
	Class 1 (NF3220)	**Class 2 (NF3320)**	**Class 3 (NF3400)**	**Class 1 (NF3230)**	**Class 2 (NF3330)**	**Class 3 (NF3400)**	**Class 1 (NF3240)**	**Class 2 (NF3340)**	**Class 3 (NF3400)**
	App. F1370						or NF3225	NF3231.1c	NF3231.1c
								App. F1370	App. F1370

NOTE:

* The nomenclature, conditions, and applications of the above referenced allowables are in accordance with ASME III requirements.
Bolts are qualified in accordance with Appendix XVII as directed by NF3280.
Welded joints are designed in accordance with NF3290.

TABLE 3.9B–9A LOAD COMBINATIONS FOR ASME SECTION III CLASS 1, 2, AND 3 COMPONENT SUPPORTS

Plant Operating Condition	Load Combination
Normal	D + T + P (d + r)
Upset	D + T + E + P (d + r + e + a + h)
Emergency	D + T + E' + P (d + r + e' + a' + h)
Faulted	D + T + E' + Al + P (d + r + e' + a' + h + al)

Loadings Applicable to Component Supports

D - Sustained mechanical loads, including deadweight of equipment and contents
T - Loads on supports due to thermal expansion (constraint of free-end displacement) of components
E - Inertia effects of the OBE
E'- Inertia effects of the SSE
Al- Loads resulting from primary loop pipe rupture asymmetric pressure effects (see FSAR Section 3.9B.1.4)
P- Piping associated loads as follows:
- d - sustained deadweight of piping contents and insulation
- r - loads induced on component supports due to thermal and pressure growth of piping for appropriate plant condition
- e - inertia effects of OBE
- e'- inertia effects of SSE
- a - loads induced in component supports due to response of civil structure for OBE (OBE anchor movement)
- a'- loads induced in component supports due to response of civil structure of SSE (SSE anchor movement)
- h - loads resulting from occasional loads other than seismic (water hammer, steam hammer, safety relief valve opening or closing, etc.) as appropriate for plant condition
- al - loads induced in component supports due to the effects of LOCA

TABLE 3.9B–10 ASME III CLASS 1 STRESS AND FATIGUE ANALYSIS REQUIREMENTS PER NB3650

	Normal and Upset Conditions	Emergency Conditions	Faulted Conditions
Primary Stress Intensity (Equation 9)	$_1\left(\frac{PD_o}{2t}\right) + B_2\left(\frac{D_o}{2I}\right)M_1 \le 1.5S_m$	$\le 2.25S_m$ but not greater than $1.8S_y$	$\le 3.0S_m$
Primary and Secondary Stress Range (Equation 10)	$S_n = C_1\left(\frac{P_oD_o}{2t}\right) + C_2\left(\frac{D_o}{2I}\right)M_1 + \frac{1}{2(1-v)}E\alpha\lvert\Delta T_1\rvert + C_3E_{ab}\lvert\alpha_aT_a - \alpha_bT_b\rvert \le 3S_m$	N/R	N/R
Peak Stress Range (Equation 11)	$S_p = K_1C_1\left(\frac{P_oD_o}{2t}\right) + K_2C_2\left(\frac{D_o}{2I}\right)M_1 + \frac{1}{2(1-v)}K_3E_\alpha\lvert\Delta T_1\rvert + K_3C_3E_{ab}\lvert\alpha_aT_a - \alpha_bT_b\rvert + \frac{1}{1-v}E\alpha\lvert\Delta T_2\rvert$	N/R	N/R
Thermal Expansions Range (Equation 12)	$S_e = C_2\left(\frac{D_o}{2I}\right)M_1 \le 3S_m$	N/R	N/R
Primary and Secondary Membrane, and Bending Stress (Equation 13)	$C_1\left(\frac{P_oD_o}{2t}\right) + C_2\left(\frac{D_o}{2I}\right)M_1 + C_2^1\, E_{ab}\lvert\alpha_aT_a - \alpha_bT_b\rvert \le 3S_m$	N/R	N/R
Alternating Stress (Equation 14)	$S_{alt} = 1/2\ K_eS_p$	N/R	N/R
Usage Factor	$U = \frac{Actual\ No.\ Cycles}{Allowable\ No.\ Cycles}\ \Sigma U \le 1.0$	N/R	N/R

NOTES:

Nomenclature is as described in ASME Section III, NB-3600.

$B_1 = 0.5$ may be used in lieu of 1.0 for branch connections, curved pipe/elbows, and tees.

C_1 and K_1 indices may be derived from NUREG CR-0778, June 1979.

LOAD COMBINATIONS FOR ASME III CLASS 1 PIPING[1]

Plant Operating Condition	Equations	Load (Moment Combination) [2]
Normal/Upset	9	P_d + D + E+ H + V
	10	P_o + T + R + H + E + A + V + L
	11	P_o + T + R + H + E + A + V + L
	12	T + R
	13	P_o + D + E+ H + V + L
	14	P_o + T + R+ H + E + A + V + L
	12a	S
	9	P_1 + D + H + Y
Emergency	9	P_1 + D + Y' + E' + H + A1
Faulted		P_t
Test		P_t + PD

LOAD COMBINATIONS FOR ASME III CLASS 1 PIPING (1)

Plant Operating Condition	Equations	Allowable Stress
Normal/Upset	9	1.5 S_m
	10	-3.0 S_m
	11	-
	12	3.0 S_m
	13	3.0 S_m
	14	-
	12a	3.0S_m
Emergency	9	2.25 S_m or 1.8S_y
Faulted	9	3.0 S_m
Test		0.9S_y
		1.35S_y

NOTES:

1. This includes the following piping for Millstone 3:
   Reactor Coolant Piping (including RC bypass, surge lines).
   Portions of High and Low Pressure Safety Injection System Piping.
   Portions of Residual Heat Removal System Piping.
   Portion of Chemical and Volume Control Piping
   Portions of Pressurizer Safety, Relief and Spray Piping.
2 See Table 3.9B-12 for Definition of Loadings.
3 Class 1 piping does not experience loading designated by “W.”
4. Occasional dynamic loads such as waterhammer, steamhammer, safety valve discharge, etc. are combined with seismic inertia by SRSS method.

TABLE 3.9B–11 ASME III CLASS 2 AND 3* STRESS ANALYSIS REQUIREMENTS PER NC3650 AND ND3650

	Normal and Upset Conditions	Emergency Conditions	Faulted Conditions
Sustained Loads (Equation 8)	$\frac{PD_o}{4t} + \frac{(0.75\mathrm{i})M_A}{Z} \leq S_h$	N/R	N/R
Occasional Loads (Equation 9)	$\frac{PD_o}{4t} + \frac{(0.75\mathrm{i})M_A}{Z} + \frac{(0.75\mathrm{i})M_B}{Z} \leq 1.2S_h$	$1.8S_h$	$2.4S_h$
Thermal Expansion Loads (Equation 10)	$i\frac{M_c}{Z} \leq S_A$ $\leq F(1.25S_c + 0.25S_h)$	N/R	N/R
Sustained and Thermal Expansion Loads (Expansion 11)	$\frac{PD_o}{4t} + \frac{(0.75\mathrm{i})M_A}{Z} + \frac{(\mathrm{i})M_C}{Z} \leq S_a + S_h$	N/R	N/R

NOTE:

* All nomenclature are as defined in ASME Section III NC, ND 3600.

TABLE 3.9B–11 ASME III CLASS 2 AND 3* STRESS ANALYSIS REQUIREMENTS PER NC3650 AND ND3650
LOADING COMBINATIONS FOR QUENCH SPRAY, RECIRCULATION, AND SAFETY INJECTION

Plant Operating Condition	NC 3600 Equations	Load (Moment) Combinations **	Allowable Stress
Normal/Upset	8	P_d + D	S_h
	9	P_p + D + E + H + V + W	1.2 S_h
	10*	T + R + A	S_A
	10a	S	3 S_C
	11*	P_d + D + T + R + A	$S_h + S_A$
Emergency	9	P_p + D + H + Y + W	1.8 S_h
Faulted	9	P_p + D + E' + H + Y' + W + A1	1.8 S_h
	10*	T + R' + A' + X	S_A

TABLE 3.9B–11 ASME III CLASS 2 AND 3* STRESS ANALYSIS REQUIREMENTS PER NC3650 AND ND3650
LOADING COMBINATIONS FOR QUENCH SPRAY, RECIRCULATION, AND SAFETY INJECTION (CONTINUED)

Plant Operating Condition	NC 3600 Equations	Load (Moment) Combinations **	Allowable Stress
	11*	$P_d + D + T + R' + A' + X$	$S_h + S_A$
Test	8	P_t	S_c
		$P_t + D$	$0.9\ S_y$

NOTES:

* Either the requirements of Eq. (10) or Eq. (11) must be satisfied.

** See Table 3.9B–12 for definition of loading.

(1) Class 1 portions of the Safety Injection are analyzed in accordance with Table 3.9B–10.

(2) In break exclusion area, sum of stresses given by Equations 9 and 10 should not exceed 0.8 ($S_A + 1.2\ S_h$). In crack exclusion area, sum of stresses given by Equations 9 and 10 should not exceed 0.4 ($S_A + 1.2S_h$)

(3) Occasional dynamic loads such as waterhammer, steamhammer, safety valve discharge, etc. are combined with seismic inertia by SRSS method.

TABLE 3.9B–11 ASME III CLASS 2 AND 3* STRESS ANALYSIS REQUIREMENTS PER NC3650 AND ND3650
LOAD COMBINATIONS FOR ASME III CLASS 2 AND 3 PIPING EXCEPT QUENCH, RECIRCULATION, AND SI

Plant Operating Condition	NC 3600 Equations	Load (Moment) Combinations*	Allowable Stress
Normal/Upset	8	$P_d + D$	S_h
	9	$P_p + D + E + H + V + W$	$1.2\ S_h$
	10(1)	$T + R + A$	S_A
	10a	S	$3\ S_c$
	11(1)	$P_d + D + T + R + A$	$S_h + S_A$
Emergency	9	$P_p + D + H + Y + W$	$1.8\ S_h$
Faulted	9	$P_p + D + E' + H + W + A1 + Y'$	$2.4\ S_h$
Test	8	P_t	S_c

TABLE 3.9B–11 ASME III CLASS 2 AND 3* STRESS ANALYSIS REQUIREMENTS PER NC3650 AND ND3650 LOAD COMBINATIONS FOR ASME III CLASS 2 AND 3 PIPING EXCEPT QUENCH, RECIRCULATION, AND SI (CONTINUED)

Plant Operating Condition	NC 3600 Equations	Load (Moment) Combinations*	Allowable Stress
		$P_t + D$	$0.9\ S_y$

NOTE:

* See Table 3.9B–12 for definition of loading.

(1) Either the requirements of Equation 10 or 11 must be satisfied.

(2) In break exclusion area, sum of stresses given by Equations 9 and 10 should not exceed 0.8 ($S_A + 1.2\ S_h$). In crack exclusion area, sum of stresses given by Equations 9 and 10 should not exceed 0.4 ($S_A + 1.2S_h$).

(3) Occasional dynamic loads such as waterhammer, steamhammer, safety valve discharge, etc. are combined with seismic inertia by SRSS method.

TABLE 3.9B–11 ASME III CLASS 2 AND 3* STRESS ANALYSIS REQUIREMENTS PER NC3650 AND ND3650 LOAD COMBINATIONS FOR ANSI B31.1 PIPING

Plant Operating Condition	104.8 Equations*	Load (Moment) Combinations**	Allowable Stress
Design	11	$P_d + D$	S_h
Normal/Upset	12	$P_p + D + H + W$	$1.2\ S_h$
	13	$T + R$	S_A
	14	$P_d + D + T + R$	$S_h + S_A$
	10a	S	$3\ S_c$
Seismic	*		
Test	11	$P_t + D$	S_c
		P_t	S_c

NOTES:

* Equation numbers are from 1973 Edition ANSI B31.1, dated June 30, 1973.

** Refer to Table 3.9B–12 for identification of loadings.

*** See NETM-42 for evaluation of ANSI B31.1 for seismic loading.

TABLE 3.9B–12 LOADINGS APPLICABLE TO PIPING SYSTEMS

D - Sustained mechanical loads, including deadweight of piping, components, contents, and insulation.

P - Internal pressure loads.

T - Loads due to thermal expansion of the system in response to average fluid temperature.

R - Loads induced in the piping due to the thermal growth of equipment and/or structures to which the piping is connected as a result of plant normal or upset plant conditions.

R' - Loads induced in the piping due to thermal growth of equipment and/or structures to which the piping is connected as a result of plant faulted plant conditions.

E - Inertia effects of the OBE.

E' - Inertia effects of the SSE.

A - Loads induced in the piping due to response of the connected equipment and/or civil structures to the OBE (commonly referred to as OBE anchor movements).

A' - Loads induced in the piping due to response of the connected equipment and/or civil structures to the SSE (commonly referred to as SSE movements).

S - Loads induced due to building settlement effects.

H - Loads resulting from occasional loads other than seismic. Examples of these loads would be: water hammer, steam hammer, opening and closing of safety relief valves, etc.

L - Local stress effects in piping and/or piping components due to sudden changes in fluid temperature. These loads are commonly referred to as thermal transient effects.

Y - Effects of piping striking pipe (pipe whip) or whip or effects of blowdown of an adjacent system (jet impingement loads), as defined for the emergency plant condition

Y' - Effects of pipe striking pipe (pipe whip) or effects of blowdown of an adjacent system (jet impingement loads), as defined for the faulted plant condition.

V - Loads induced by operation of the system other than those loads described above (operational vibration).

NOTE: These loads are evaluated during preoperational and testing phase

B - Loads on restraints induced by blowdown and subsequent pipe response of a ruptured system for emergency plant conditions.

B' - Loads on restraints induced by blowdown and subsequent pipe response of a ruptured system for faulted plant conditions.

X - Loads induced in the piping due to pressure response (growth) of the containment during a faulted plant condition.

P1 - Internal pressure loads for emergency and faulted condition as applicable.

P_d - Internal pressure due to design pressure.

P_o - Internal pressure due to range of operating pressure.

P_p - Internal pressure due to peak pressure.

P_t - Internal pressure due to test pressure.

A1 - Loads induced in piping due to inertia effects of LOCA or displacements due to LOCA or displacements due to pipe rupture.

W - Loads imposed by wind, snow, or ice.

TABLE 3.9B–13 ACTIVE PUMPS AND VALVES

This table has been deleted. Refer to the Plant Design Data System (PDDS), Seismic Qualification Tracking System (SQT) List for details.

TABLE 3.9B–14 OMITTED

TABLE 3.9B–15 POSTULATED PRIMARY LOOP BREAKS

Break No.	Location of Break	Type of Break
1	Reactor Vessel Outlet Nozzle	1/4 Double-Ended Guillotine
2	Reactor Vessel Inlet Nozzle	1/2 Double-Ended Guillotine
3	Steam Generator Inlet Nozzle	Double-Ended Guillotine
4	Steam Generator Outlet Nozzle	Double-Ended Guillotine
5	Reactor Coolant Pump Inlet Nozzle	Double-Ended Guillotine
6	Reactor Coolant Pump Outlet Nozzle	Double-Ended Guillotine
7	50-Degree Elbow on the Intrados	Split
8	Flow Entrance to the 90-degree Elbow (Steam Generator Side) on the Crossover Leg	Double-Ended Guillotine
9	Residual Heat Removal/Primary Loop Connection	Guillotine (viewed from the residual heat removal line)
10	Safety Injection/Primary Coolant Loop Connection	Guillotine (viewed from the Safety Injection line)
11	Pressurizer Surge/Primary Coolant Loop Connection	Guillotine (viewed from the pressurizer surge line)
12	Loop Closure Weld in Crossover Leg	Double-Ended Guillotine

TABLE 3.9B–16 STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORT SNUBBER EMBEDMENT LOADS[1] (KIPS)

		Normal Operation [3] [4]			Faulted Condition [4] [5]		
Component	**No.** [2]	F_x	V	M	F_x	V	M
Steam Generator Upper Snubbers	1 or 3	668	3	59	+1153	+59	+1144
					-1443	-56	-1088
	2 or 4	825	7	137	+1553	+59	+1144
					-1443	-56	-1088
Steam Generator Lower Snubbers	1	106	6	117	+1375	+63	+1226
					-988	-73	-1423
	2	74	9	413	+969	+92	+4523
					-409	-78	-3607
	3	118	6	117	+988	+63	+1226
					-1375	-73	-1423
	4	105	6	117	+988	+63	+1226
					-1375	-73	-1423
Reactor Coolant Pump Snubbers	1	83	6	297	+1168	73	3333
					-1307		
	2	113	9	413	+841	+89	+4362
					-930	-91	-4510
	3	126	6	36	+970	+75	+3695
					-1391	-86	-4395

NOTES:

(1) Maximum for all loops. All loads are “±” except where indicated. In those cases “+” means outward, “-” means inward on embedment.

(2) Refer to Figure 3.9B–5

(3) Includes seismic (OBE) and deadweight.

(4) F_x = axial, V = resultant shear, M = resultant moment

(5) Includes seismic (SSE), deadweight, and pipe rupture

TABLE 3.9B–17 STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORT COLUMN EMBEDMENT LOADS(1)(KIPS)

		Normal Operation (1) (2)			Faulted Condition (1) (2)		
Component	**No. (1)**	**F_x**	**V**	**M**	**F_x**	**V**	**M**
Steam Generator Support Columns	5	+74 -501	Negligible	Negligible	+1,197 -1383	71	1200
	6	+74 -501	Negligible	Negligible	+1479 -1411	72	1225
	7	+74 -501	Negligible	Negligible	+1354 -1229	67	1146
	8	+74 -446	Negligible	Negligible	+1455 -1282	71	1208
Reactor Coolant Pump Columns	4	+96 -363	Negligible	Negligible	+1996 -1578	132	3039
	5	+96 -363	Negligible	Negligible	+1145 -2228	187	4293
	6	+96 -363	Negligible	Negligible	+1579 -1625	120	2767

NOTES:

(1) Refer to Table 3.9-2

(2) Fx = axial

V = resultant shear (horizontal)

M = resultant moment

TABLE 3.9B–18 FACTORS OF SAFETY FOR PRIMARY MEMBERS OF STEAM GENERATORS AND REACTOR COOLANT PUMP SUPPORTS

			Faulted Condition		
Support Section Members (1)	**Material Designation**	**Material Stress (2) (ksi)**	**Actual Stress (ksi)**	**Allowable Stress(3) (ksi)**	**Factor of Safety**
Steam Generator					
Upper Restraint					
Ring	A543-72 - C1.2	86.9	Stress Ratio = 0.74		1.36
Splice Plate	A543-72 - C1.2	86.9	Stress Ratio = 0.72		1.39
Connecting Rod Assy.	A487 - Gr. 10Q	88.0	Stress Ratio = 0.77		1.3
Coupling	A668-72 C1. L	76.8	Stress Ratio = 0.89		1.12
Cylinder Lug. Assy.	A487 - Gr. 10Q	97.0	Stress Ratio = 0.2		5.0
	A668-72 Gr. M	101.0	Stress Ratio = 0.89		1.12
Wall Clevis	A487 - Gr. 10Q	97.0	28.9	109.0	3.8
Connecting Rod Bolt	A668-72 - C1. M	95.8	94.3	122.0	1.36
Studs	SA-193 Gr. B-16	94.0	22.7	55.9	2.5
Yoke	A-543-72 C1. 2	86.9	35	57	1.6
Lower Restraint					
Ring Clevis	A487 - Gr. 10Q	83.5	Stress Ratio = 0.53		1.88
Snubber Lug Assy.	A487 - Gr. 10Q	97.0	Stress Ratio = 0.258		3.88
Extension Tube	A668-72 - C1. L	76.8	Stress Ratio = 0.76		1.32
Snubber Clevis	A471-70 - C1.5	103.4	Stress Ratio = 0.24		4.1

TABLE 3.9B–18 FACTORS OF SAFETY FOR PRIMARY MEMBERS OF STEAM GENERATORS AND REACTOR COOLANT PUMP SUPPORTS (CONTINUED)

			Faulted Condition		
Support Section Members (1)	**Material Designation**	**Material Stress (2) (ksi)**	**Actual Stress (ksi)**	**Allowable Stress(3) (ksi)**	**Factor of Safety**
Wall Mount	A487 - Gr. 10Q	97.1	53.3	110.0	1.98
Vertical Column					
Upper Column Clevis	A487 - Gr. 10Q	83.5	Stress Ratio = 0.52		1.9
Upper Column Lug	A487 - Gr. 10Q	88.0	Stress Ratio = 0.78		2.28
Column Tube	A668-72 C1. L	76.8	Stress Ratio = 0.47		2.1
Floor Clevis	A487 - Gr. 10Q	97.0	Stress Ratio = 0.4		2.5
Lower Column Lug	A487 - Gr. 10Q	97.0	Stress Ratio = 0.42		2.38
S/G Cap Screws	SA-540 - B23 C1. 2	87.0	Stress Ratio = 0.3		1.8
Reactor Coolant Pump					
Lateral Restraint					
Wall Mount	A487 - Gr. 10Q	97.1	55.3	109.3	1.98
Snubber Clevis	A487 - Gr. 10Q	97.0	Stress Ratio = 0.6		1.7
Extension Tube	A668-72 C1. L	76.8	Stress Ratio = 0.68		1.47
Pump Link	A668-72 C1. M	105.2	Stress Ratio = 0.817		1.22
Pump Lug	A487 - Gr. 10Q	88.0	Stress Ratio = 0.6		1.7
Snubber Clevis Pin	A668-72 C1. M	101.0	58.1	110.0	1.89

TABLE 3.9B–18 FACTORS OF SAFETY FOR PRIMARY MEMBERS OF STEAM GENERATORS AND REACTOR COOLANT PUMP SUPPORTS (CONTINUED)

Support Section Members (1)	Material Designation	Material Stress (2) (ksi)	Faulted Condition: Actual Stress (ksi)	Faulted Condition: Allowable Stress(3) (ksi)	Factor of Safety
Vertical Column					
Upper Clevis	A487 - Gr. 10Q	88.0	Stress Ratio = 0.41		2.42
Upper Lug	A487 - Gr. 10Q	93.2	Stress Ratio = 0.72		1.39
Column Tube	A-668-72 Gr. L	76.8	Stress Ratio = 0.73		1.37
Lower Lug	A487 - Gr. 10Q	93.2	Stress Ratio = 0.72		1.39
Floor Clevis	A487 - Gr. 10Q	97.0	Stress Ratio = 0.37		2.7
Tie Down Pin	A471-70 C1.5	102.6	Stress Ratio = 0.753		1.32
Pin	A668-72 C1.M	115.0	84.2	98.0	1.16

NOTES:

(1) Refer toFigure 5.4–12.

(2) Minimum specified yield or ultimate at temperature.

(3) Even though the allowable stresses are lower for the normal operating conditions, the corresponding design loads were significantly lower, thus making the faulted condition the critical design condition.

TABLE 3.9B–19 MINIMUM DESIGN MARGINS FOR PRESSURIZER SUPPORT

Support Section Members (1)	Material Designation	Material Stress (2) (ksi)	Faulted		Factor of Safety
			Actual Stress (ksi)	Allowable Stress (ksi)	
Ring Girder	SA-516 - Gr. 70	36.3	Stress Ratio = 0.48		20.5
Column Pipe	SA-106-C	39.2	Stress Ratio = 0.84		1.2
Upper Vertical Clevis	A487 - Gr. 10Q	100.	Stress Ratio = 0.87		1.15
Lower Vertical Clevis	A487 - Gr. 10Q	100.	20.8	41.08	1.97
Upper Vertical Lug	A487 - Gr. 10Q	100.	19.8	39.5	2.02
Horizontal Ring and Wall Clevises	A487 - Gr. 10Q	100.	11.6	27.5	2.36
Lower Horizontal Wall Lugs	SA-533 - Gr. A C1. 1	49.	65.9	69.8	1.06
Spherical Plate	SA-516 - Gr. 70	36.8	8	44.2	5.51
Adjusting Rod	SA-540-B23 - C1.5	103.6	25.3	49.7	1.98
Upper Lug Rest.	A-543-72 C1. 2	92.	55.2	77.3	1.4
Pin	SA-540-B23 C1.2	137.	116.4	143.7	1.23
Embedment Plate	SA-516 - Gr. 70	36.8	16.5	55.2	3.34
Skirt Anchor Bolts	SA-540-B23 C1.2	137	84.8	106.1	1.15

NOTES:

1 Refer to Figure 5.4–13.

2 Minimum specified yield or ultimate at temperature.

TABLE 3.9B–20 DESIGN MARGINS FOR THE PRIMARY MEMBERS OF THE PRESSURIZER SAFETY VALVE SUPPORT

Support Section[2]	Material Designation	Material Stress [3] (ksi)	Faulted [1] Actual Stress (ksi)	Faulted [1] Allowable Stress (ksi)	Factor of Safety
Valve Support Flange Weld	SA-537-C1.2	52.2	25.	48.	1.9
Radial Arm	SA-537-C1.2	52.2	9.1	55.7	6.1
Ring Girder	SA-537-C1.2	52.2	40.	55.7	1.39
Column	SA-537-C1.2	47.6	Stress Ratio = 0.73		1.37
Pin	SA-540 B24 C1.2	116.1	124	135	1.1
Support Bracket	SA-537 C1.2	46.	44.3	55.2	1.2

NOTES:

(1) A comparison of normal, upset, and faulted loads show that the faulted loads exceed the other loading conditions by a ratio greater than 2.

(2) Refer to Figure 5.4–15.

(3) Minimum yield or ultimate at temperature

TABLE 3.9B–21 DESIGN MARGINS FOR PRIMARY MEMBERS OF REACTOR PRESSURE VESSEL SUPPORT

		Normal Operating				Faulted		
Support Section Member(1)	**Material Designation**	**Actual Stress (ksi)**	**Allowable Stress (ksi)**	**Factor of Safety**	**Material Stress(2) (ksi)**	**Actual Stress (ksi)**	**Allowable Stress(3) (ksi)**	**Factor of Safety**
R.V. Vertical Restraint Pad	A668-72 Gr. N	small	44.3	large	110.7	28.7	95.3	3.3
R.V. Support Gibkeys	A668-72 Gr. N	19.0	71.1	3.7	118.5	66.3	101.3	1.5
R.V. Support Gibgussets	SA-533-C1.2	6.0	39.96	6.6	66.6	39.8	59.9	1.56
Shell	SA-537-C1.2	10.9	26.7	2.45	57.5	29.1	82.25	2.8
Base Flange	SA-516-70	11.14	69.9	6.3	38.	27.0	68.4	2.53
Anchor Studs	SA-193-B7	14.9	62.5	4.2	105	36.6	87.5	2.39
R.V. Bolts	SA-540-C1.1	small	70.2	large	127.7	49.9	98.3	1.97
Leveling Device	A668-72-N	2.2	73.7	33.5	123	87.5	105	1.2

NOTES:

(1) Refer to Figures 5.4–9 and 5.4–10.

(2) Minimum yield or ultimate at temperatures.

(3) Refer to Table 5.4–18.

TABLE 3.9B–22 DESIGN MARGINS FOR PRIMARY LOOP BUMPER SUPPORT STRUCTURE

Primary Loop Bumper Section	Material Designation	Material Stress (1) (ksi)	Max. Actual Stress (ksi)	Allowable Stress (ksi)	Factor of Safety
Cross Over Leg Bumper					
Intermediate Structure	ASTM 514				
Steam Generator Side		86.0	69.5	70.0	1.01
Pump Side		86.0	50.5	70.0	1.38
Saddle	ASTM 514				
Steam Generator Side		77.0	49.3	70.0	1.42
Pump Side		77.0	56.9	70.0	1.23
Hot Leg Bumper					
Vertical I-Beam	ASTM 588	43.8	Stress ratio = 0.91		1.1
Saddle	ASTM 514	70.4	65.2	70.4	1.08
Cold Leg Bumper					
Key Ring	ASTM 514	77.0	15.0	33.3	2.2
Key Support	ASTM 514	77.0	45.8	70.0	1.53
Key Support Ring	ASTM 514	77.0	55.0	70.0	1.27
Attachment Bolts	ASTM 193 GR B7	125 (ult. stress)	33.6	36.17	1.08

NOTE:

(1) Minimum specified yield at temperatures.

TABLE 3.9N–1 SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Normal Conditions		Occurrences
1.	Heatup and cooldown at 100°F/hr (pressurizer cooldown 200°/hr)	200 (each)
2.	Unit loading and unloading at 5% of full power/min	13,200 (each)
3.	Step load increase and decrease of 10% of full power	2,000 (each)
4.	Large step load decrease with steam dump	200
5.	Steady state fluctuations	
	a. Initial fluctuations	1.5×10^5
	b. Random fluctuations	3.0×10^6
6.	Feedwater cycling at hot shutdown	2,000
7.	Not Used	
8.	Unit loading and unloading between 0 and 15% of full power	500 (each)
9.	Boron concentration equalization	26,400
10.	Refueling	80
11.	Reduce temperature return to power	2,000
12.	Reactor coolant pumps startup/shutdown	3,800
13.	Turbine roll test	20
14.	Primary side leak test	200
15.	Secondary side leak test	80
16.	Tube leakage test	800
17.	Heaters out of service	
	a. One heater	120
	b. One bank of heaters	120
Upset Conditions		**Occurrences**
1.	Loss of load, without immediate reactor trip	80
2.	Loss of power (blackout with natural circulation in the reactor coolant system)	40
3.	Partial loss of flow (loss of one pump)	80
4.	Reactor trip from full power	

TABLE 3.9N–1 SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS (CONTINUED)

	a. Without cooldown	230
	b. With cooldown, without safety injection	160
	c. With cooldown and safety injection	10
5.	Inadvertent reactor coolant depressurization	20
6.	Not Used	
7.	Control rod drop	80
8.	Inadvertent emergency core cooling system actuation	60
9.	Operating basis earthquake (20 earthquakes of 20 cycles each)	400
10.	Excessive feedwater flow	30
11.	Reactor Coolant System (RCS) Cold Overpressurization	10
Emergency Conditions (1)		
1.	Small loss of coolant accident	5
2.	Small steam break	5
3.	Complete loss of flow	5
Faulted Conditions (1)		
1.	Main reactor coolant pipe break (large loss of coolant accident)	1
2.	Large steam break	1
3.	Feedwater line break	1
4.	Reactor coolant pump locked rotor	1
5.	Control rod ejection	1
6.	Steam generator tube rupture	1
7.	Safe shutdown earthquake	1
Test Conditions		Occurrences
1.	Primary side hydrostatic test	10
2.	Secondary side hydrostatic test	10

NOTE:

(1) In accordance with ASME III, emergency and faulted conditions are not included in fatigue evaluation.

TABLE 3.9N–2 LOADING COMBINATIONS FOR REACTOR COOLANT SYSTEM COMPONENTS

Condition Classification	Loading Combination
Design	Design pressure, Design temperature, Deadweight, Operating basis earthquake
Normal	Normal conditions transients, Deadweight
Upset	Upset condition transients, Deadweight, Operating basis earthquake
Emergency	Emergency condition transients, Deadweight
Faulted	Faulted condition transients, Deadweight, Safe shutdown earthquake, Pipe rupture loads

TABLE 3.9N–3 ALLOWABLE STRESSES FOR REACTOR COOLANT SYSTEM COMPONENTS

Operating Condition Classification	Vessels / Tanks	Piping	Pumps	Valves
Normal	ASME Section III NB-3000	ASME Section III NB-3600	ASME Section III NB-3400	ASME Section III NB-3500
Upset	ASME Section III NB-3000	ASME Section III NB-3600	ASME Section III NB-3400	ASME Section III NB-3500
Emergency	ASME Section III NB-3000	ASME Section III NB-3600	ASME Section III NB-3400	ASME Section III NB-3500
Faulted	ASME Section III Appendix F	ASME Section III Appendix F	ASME Section III Appendix F	Note 1

NOTE 1 TO TABLE 3.9N-3:

CLASS 1 VALVE FAULTED CONDITION CRITERIA	
Active	**Inactive**
a) Calculate Pm from para. NB3545.1 with Internal Pressure P_s = 1.25Ps $P_m \leq 1.5S_m$	a) Calculate Pm from para. NB3545.1 with Internal Pressure P_s = 1.50Ps $P_m \leq 2.4S_m$ or $0.7S_u$
b) Calculate S_n from para. NB3545.2 with $C_p = 1.5$ P_s = 1.25Ps $Q_t2 = 0$ P_{ed} = 1.3X value of Ped from equations of 3545.2(b) (1) $S_n \leq 3S_m$	b) Calculate S_n from para. NB3545.2 with Cp = 1.5 P_s = 1.50Ps $Q_t2 = 0$ P_{ed} = 1.3X value of Ped from equations of NB3545.2(b) (1) $S_n \leq 3S_m$

P_e, P_m, P_b, Q_t, C_p, S_n, and S_m as defined by Section III ASME Code.

TABLE 3.9N–4 DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 2 AND 3 COMPONENTS AND SUPPORTS

	Loading Combinations (1) (2)	Design/Service Level Requirements
1.	Design pressure, Design temperature, Deadweight, Nozzle loads	Design and Normal
2.	Normal condition pressure, Normal condition metal temperature, Deadweight, Nozzle loads	Normal
3.	Upset condition pressure, Upset condition metal temperature, Deadweight, Nozzle loads, Operating basis earthquake	Upset
4.	Emergency condition pressure, Emergency condition metal temperature, Deadweight, Nozzle loads	Emergency
5.	Faulted condition pressure, Faulted condition metal temperature, Deadweight, Nozzle loads, Safe shutdown earthquake	Faulted

NOTES:

(1) Temperature is used to determine allowable stress only.

(2) Nozzle loads, pressure, and temperatures are those associated with the respective plant operating conditions (i.e., normal, upset, emergency, and faulted) as noted, for the component under consideration.

TABLE 3.9N–5 STRESS CRITERIA FOR SAFETY CODE CLASS 2 (1) AND CLASS 3 TANKS

Design/Service Level	Stress Limits
Design and Normal	$\sigma_m \leq 1.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.5$ S
Upset	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S
Emergency	$\sigma_m \leq 1.5$ S $(\sigma_m$ or $\sigma_b) + \sigma_b \leq 1.80$ S
Faulted	$\sigma_m \leq 2.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 2.4$ S

NOTE:

(1) Applies for tanks designed in accordance with ASME III, NC-3300.

TABLE 3.9N–6 STRESS CRITERIA FOR ASME CODE CLASS 2 TANKS[1]

Design/Service Level	Stress Limits
Design and Normal	$P_m \leq 1.0\ S_m$
	$P_L \leq 1.5\ S_m$
	$(P_m \text{ or } P_L) + P_b \leq 1.5\ S_m$
Upset	$P_m \leq 1.1\ S_m$
	$P_L \leq 1.65\ S_m$
	$(P_m \text{ or } P_L) + P_b \leq 1.65\ S_m$
Emergency	$P_m \leq 1.2\ S_m$
	$P_L \leq 1.8\ S_m$
	$(P_m \text{ or } P_L) + P_b \leq 1.8\ S_m$
Faulted	$P_m \leq 2.0\ S_m$
	$P_L \leq 3.0\ S_m$
	$(P_m \text{ or } P_L) + P_b \leq 3.0\ S_m$

NOTE:

(1) Applies for tanks designed in accordance with ASME III, NC-3200.

TABLE 3.9N–7 STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3 INACTIVE PUMPS

Design/Service Level	Stress Limits
Design/Normal	The pump shall conform to the requirements of ASME Section III, NC-3400 (or ND-3400) $\sigma_m \leq 1.0S$ and $(\sigma_m$ or $\sigma_L) + \sigma_b < 1.5S$
Upset	$\sigma_m \leq 1.1S$
	$(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65S$
Emergency	$\sigma \leq 1.5S$
	$(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.80S$
Faulted	$\sigma_m \leq 2.0S$
	$(\sigma_m$ or $\sigma_m) + \sigma_b \leq 2.4S$

TABLE 3.9N–8 STRESS CRITERIA FOR SAFETY RELATED ASME CODE CLASS 2 AND CLASS 3 ACTIVE AND INACTIVE VALVES

Condition	Stress Limits*	Pmax**
Design and Normal	Valve bodies shall conform to ASME Section III	
Upset	$\sigma_m \leq 1.1S$	1.1
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65S$	
Emergency	$\sigma_m \leq 1.5S$	1.2
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80S$	
Faulted	$\sigma_m \leq 2.0S$	1.5
	$(\sigma_m \text{ or } \sigma_L) \text{ or } \sigma_b \leq 2.4S$	

NOTES:

* Valve nozzle (piping load) stress analysis is not required when both the following conditions are satisfied: (1) the section modulus and area of every plane, normal to the flow, through the region defined as the valve body crotch are at least 110 percent of those for the piping connected (or joined) to the valve body inlet and outlet nozzles and, (2) code allowable stress for valve body material is equal to or greater than the code allowable stress of connected piping material. If the valve body material allowable stress is than that of the connected piping, the required acceptance criteria ratio shall be 110 percent multiplied by the ratio of the pipe allowable stress to the valve allowable stress. If unable to comply with this requirement, an analysis in accordance with the design procedure for Class I valves is an acceptable alternate method.

** The maximum pressure resulting from upset, emergency, or faulted conditions shall not exceed the tabulated factors listed under P_{max} times the design pressure. If these pressure limits are met, the stress limits in this table are considered to be satisfied.

TABLE 3.9N–9 STRESS CRITERIA FOR ASME CODE CLASS 2 AND 3 PIPING

See Section 3.9B.1, Table 3.9B–11.

TABLE 3.9N–10 DESIGN CRITERIA FOR ACTIVE PUMPS

Loading Combination	Design Criteria
1. Design	ASME Section III Subsection NC-3400 and ND-3400.
Normal	$\sigma_m \leq 1.0S$ $\sigma_m + \sigma_b \leq 1.5S$
2. Upset	$\sigma_m \leq 1.0S$ $\sigma_m + \sigma_b \leq 1.5S$
3. Emergency	$\sigma_m \leq 1.2S$ $\sigma_m + \sigma_b \leq 1.65S$
4. Faulted	$\sigma_m \leq 1.2S$ $\sigma_m + \sigma_b \leq 1.8S$

TABLE 3.9N–11 ACTIVE PUMPS

Pump	Item No.	System	ANS Safety Class	Normal Mode	Post LOCA Mode	Basis
Centrifugal charging Pump Number 1, 2 and 3	APCH	CVCS	2	ON/OFF	ON	Required for high head safety injection [*]; safe shutdown.
Residual removal Pump Number 1 and 2	APRH	RHRS	2	OFF	ON	Required for safety injection.
Safety injection Pump Number 1 and 2	APSI	SIS	2	OFF	ON	Required for safety injection.
Boric acid transfer Pumps 1 and 2	APBA	CVCS	3	ON/OFF	OFF	Required for reactor shutdown.

NOTES:

(*) Only two charging pumps are required for proper safety injection operation, while the third pump is considered an installed spare.

TABLE 3.9N–12 ACTIVE VALVES

This table has been deleted. Refer to the Plant Design Data System (PDDS) Seismic Qualification Tracking System (SQT).

TABLE 3.9N–13 MAXIMUM DEFLECTIONS ALLOWED FOR REACTOR INTERNAL SUPPORT STRUCTURES

Component	Allowable Deflections (inches)	No-Loss-of Function Deflections (inches)
Upper barrel, Radial inward	4.1	8.2
Upper barrel, Radial outward	1.0	1.0
Upper package	0.10	0.15
Rod cluster guide tubes	1.00	1.75

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

This section presents information to demonstrate that instrumentation and electrical equipment classified as Seismic Category I is capable of performing designated safety related functions in the event of an earthquake. The information presented includes identification of the Category IE instrumentation and electrical equipment that are within the scope of Westinghouse nuclear steam supply system (NSSS), and balance of plant (BOP) scope. The qualification criteria employed for each item of equipment, the designated safety related requirements, definition of the applicable seismic environment, and documentation of the qualification process employed to demonstrate the required seismic capability are described in this section.

Sections whose identification numbers include the letter B contain material within the balance-of-plant (BOP) scope, while sections whose identification numbers include the letter N contain material within the nuclear steam supply system (NSSS) scope.

3.10B.1 SEISMIC QUALIFICATION CRITERIA

The methods of meeting the general requirements for seismic qualification of Category I instrumentation and electrical equipment as described by General Design Criteria (GDC) 1, 2, and 23 are described in Section 3.1. The general methods of implementing the requirements of Appendix B to 10 CFR Part 50 are described in Chapter 17.

Seismic Category I instrumentation and electrical equipment are designed to maintain the capability to:

1. Initiate a protective action during a safe shutdown earthquake (SSE)

2. Withstand seismic disturbances during post-accident operation without loss of safety function

Safety related instrumentation and electrical equipment is seismically qualified in accordance with general instructions for earthquake requirements (Section 3.7B.3.1). In order to prevent any threat of impacting damage to Class IE equipment during seismic events, nonsafety related instrumentation and electrical equipment located adjacent to Class IE equipment is also seismically qualified. The earthquake requirements and qualification methods conform to those outlined in IEEE Standard 344-1975, “IEEE Recommended Practices for Seismic Qualifications of Class IE Equipment for Nuclear Power Generating Stations,” (Section 1.8, R.G. 1.100) and are in agreement with the recommendations of Branch Technical Position EICSB 10. Instrumentation and electrical equipment are tested as individual components, either as part of a simulated structural section or as part of a completely assembled module or unit.

3.10B.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Methods and procedures for qualifying electrical equipment and instrumentation are contained in Section 3.7B.3.1 and 3.10B.4.

The response of racks, panels, cabinets, and consoles is considered in assessing the seismic capability of instrumentation and electrical equipment. Mounted equipment is qualified, as a minimum, to acceleration levels consistent with those transmitted by supporting structures. A design objective is to minimize amplification of floor acceleration by supporting members on mounted equipment.

Qualification of seismically related equipment is accomplished by the following methods:

1. Test the equipment under simulated operating conditions

2. Analyze the equipment for verification of proper function

3. Combinations of test and analysis

The decision to qualify a component using one or a combination of the above methods is a judgement of the capability to perform the qualification based on the complexity of the equipment.

3.10B.3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Supports for Category I electrical equipment, instrumentation, and control systems are seismically qualified by the analysis and testing procedures outlined in Section 3.7B.3.1. Supports are designed to withstand the combined effects of normal operating loads acting simultaneously with two orthogonal horizontal and the vertical components of earthquake loading without loss of functional capability or structural integrity as applicable. The stress levels due to the combined loading conditions do not exceed the maximum stress levels permitted under applicable codes. If there are no applicable codes, the stress level under combined loading for 1/2 SSE does not exceed 75 percent of the minimum yield strength of the material at service temperature per the ASTM specification. Under safe shutdown earthquake, the stress level does not exceed the smaller of 100 percent of the specified minimum yield strength or 70 percent of the specified ultimate strength of the material at temperature per ASTM specification.

A design objective is to provide supports for electrical equipment, instrumentation, and control systems that are seismically rigid, i.e., with fundamental natural frequencies above the cutoff frequency, which separates the relatively flat rigid range of a response curve from the resonant range of the relevant amplified response spectra curves. This assures that amplification of floor accelerations through supporting members to mounted equipment is minimized.

The dynamic analysis method is typically used to establish support spacing for cable trays and conduit. Additionally, restraints are used as necessary to limit the horizontal loads to allowable design values established on the basis of raceway loading and unsupported span lengths. Design provisions for significant differential motions between buildings are made by breaks in raceways if these relative displacements would result in unacceptable equipment or support loadings.

In lieu of dynamic analysis, a provision is made for the use of static analysis in qualification of cable tray systems and conduit support systems. For use in the static analysis, the peak resonant "G" values from the ARS are amplified in accordance with Section 3.7. The damping values used for qualification of cable tray systems are 4 percent for 1/2 SSE and 8 percent for SSE. The damping values used for the qualification of conduit support systems are 4 percent for 1/2 SSE and 7 percent for SSE.

3.10B.4 REPLACEMENT ITEMS

Methodologies for demonstrating equipment seismic qualification have been established for the implementation of GDC-2. Qualification methods have changed with time, resulting in various seismic FSAR commitments.

IEEE 934-1987 specifically provides seismic guidance for replacement items and states that parts shall be selected and installed as to preserve the original seismic qualification of Class 1E equipment. Methods to be employed for seismic qualification of replacement electrical equipment are documented in IEEE 344-75 and IEEE 344-87 (Endorsed by Reg Guide 1.100, Rev. 2). Either of these documents may be utilized for plant design changes and modifications. Replacement equipment originally qualified by Westinghouse to IEEE 344-71 meet the original criteria or either IEEE 344-75 or IEEE 344-87. The use of either method will ensure the original qualification basis remain unchanged. The use of seismic experience data may be employed for procurement of seismically insensitive and rugged components.

In cases where a replacement item is not provided with documentation of seismic qualification which is identical to the original item, an equivalency evaluation must be performed. The equivalency evaluation must document the methodology utilized for technically evaluating the replacement item and provide justification that the item meets the seismic performance requirements necessary to maintain the seismic design basis.

3.10B.5 OPERATING LICENSE REVIEW

The MP3 Plant Design Data System provides the information to demonstrate proper implementation of the criteria in Section 3.10B.1 and 3.10B.4 and demonstrate adequate seismic qualification of Seismic Category I instrumentation and electrical equipment. Maintenance of this data is in accordance with the Standard Review Plan Section 3.10.

3.10N SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

This section presents information to demonstrate that instrumentation and electrical equipment classified as Seismic Category I is capable of performing designated safety related functions in the event of an earthquake. The information presented includes identification of the Category IE instrumentation and electrical equipment that are within the scope of Westinghouse nuclear steam supply system (NSSS), and balance of plant (BOP) scope. The qualification criteria employed for each item of equipment, the designated safety related requirements, definition of the applicable seismic environment and documentation of the qualification process employed to demonstrate the required seismic capability are described in this section.

Sections whose identification numbers include the letter B contain material within the BOP scope, while sections whose identification numbers include the letter N contain material within the NSSS scope.

3.10N.1 SEISMIC QUALIFICATION CRITERIA

3.10N.1.1 Qualification Standards

The methods of meeting the general requirements for seismic qualification of Category I instrumentation and electrical equipment as described by General Design Criteria (GDC) 1, 2, and 23 are described in Section 3.1. The general methods of implementing the requirements of Appendix B to 10 CFR Part 50 are described in Chapter 17.

The Commission’s recommendations concerning the methods to be employed for seismic qualification of electrical equipment are contained in Regulatory Guide 1.100, which endorses IEEE-344-1975. Westinghouse meets this standard, as modified by Regulatory Guide 1.100, by either type of test, analysis, or an appropriate combination of these methods. Westinghouse meets this commitment employing the methodology described in the final staff approved version of WCAP-8587, Rev. 2 for all Seismic Category I instrumentation and electrical equipment.

3.10N.1.2 Performance Requirements for Seismic Qualification

Equipment Qualification Data Packages (WCAP-8587) contains an equipment qualification data package (EQDP) for every item of instrumentation and electrical equipment classified as Seismic Category I within the Westinghouse NSSS scope of supply. Table 3.10N–1 identifies the Category I equipment supplied by Westinghouse for this application and references the applicable EQDP contained in Supplement 1 of WCAP-8587. Each EQDP in Supplement 1 contains a section entitled “Performance Specification.” This specification establishes the safety related functional requirements of the equipment to be demonstrated during and after a seismic event. The required response spectrum (RSS) employed by Westinghouse for generic seismic qualification is also identified in the specification, as applicable. The spectra employed have been selected to envelope the plant specific spectra defined in Section 3.7.

3.10N.1.3 Acceptance Criteria

Seismic qualification must demonstrate that Category I instrumentation and electrical equipment is capable of performing designated safety related functions during and after an earthquake of magnitude up to and including the safe shutdown earthquake (SSE). Any spurious actuation must not result in consequences adverse to safety. The qualification must also demonstrate the structural integrity of mechanical supports and structures at the operating basis earthquake (OBE) level. Some permanent mechanical deformation of supports and structures is acceptable at the SSE level provided that the ability to perform the designated safety related functions is not impaired.

3.10N.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION

In accordance with IEEE 344-1975, seismic qualification of safety related electrical equipment is demonstrated by either type testing, analysis, or a combination of these methods. The choice of qualification method employed by Westinghouse for a particular item of equipment is based upon many factors including practicability, complexity of equipment, economics, availability of previous seismic qualification to earlier standards, etc. The qualification method employed for a particular item of equipment is identified in the individual equipment qualification data packages (EQDPs) of equipment qualification data packages (WCAP-8587).

3.10N.2.1 Seismic Qualification by Type Test

From 1969 to mid-1974 Westinghouse seismic test procedures employed single axis sine-beat inputs in accordance with IEEE 344-1971 to seismically qualify equipment. The input form selected by Westinghouse was chosen following an investigation of building responses to seismic events as reported in WCAP-7558 (1971). In addition, Westinghouse has conducted seismic retesting of certain items of equipment as part of the Supplemental Qualification Program (NS-CE-692). This retesting was performed at the request of the NRC staff on agreed upon selected items of equipment employing multi-frequency, multi-axis test inputs (WCAP-8695, 1975) to demonstrate the conservatism of the original sine-beat test method with respect to the modified methods of testing for complex equipment recommended by IEEE 344-1975.

The original single axis sine-beat testing (WCAP-7821, 1971) and the additional retesting completed under the Supplemental Test Program has been the subject of generic review by the staff. For equipment which has been previously qualified by the single axis sine-beat method and included in the NRC seismic audit and, where required by the staff, the Supplemental Qualification Program, no additional qualification testing is required to demonstrate acceptability to IEEE 344-1975 since the Westinghouse aging evaluation program was implemented to determine aging effects on complex electronic equipment located outside containment. This equipment is identified in WCAP-8587, Rev. 2 (1979), Table 7.1, and the test results in the applicable EQDPs of WCAP-8587. Subprogram C of the Westinghouse aging evaluation program (Appendix B, WCAP-8587) has incorporated a representative sample of components from these systems which use complex electronic equipment. This program is completed and is reported in WCAP-8687, Supplement 2, Appendix A2. Subprogram C demonstrates that during the qualified life there are no in-service aging mechanisms capable of reducing the capability of these systems to perform during or after a seismic event.

For equipment tests after July 1974 (i.e., new designs, equipment not previously qualified, or previously qualified equipment that does not meet 1, 2, and 3 above), seismic qualification by test is performed in accordance with IEEE 344-1975. Where testing is utilized, multi-frequency, multi-axis inputs are developed by the general procedures outlined in WCAP-8695 (1975). The test results contained in the individual EQDPs of WCAP-8587 demonstrate that the measured test response spectrum envelopes the applicable required response spectrum (RSS) defined for generic testing as specified in Section 1 of the EQDP (WCAP-8587). Qualification for plant specific use is established by verification that the generic RRS specified by Westinghouse envelopes the applicable plant specific response spectrum. Alternative test methods, such as single frequency, single axis inputs, are used in selected cases as permitted by IEEE 344-1975 and Regulatory Guide 1.100.

3.10N.2.2 Seismic Qualification by Analysis

The structural integrity of safety related motors (Table 3.10N–1 EQDP-AE-2 and AE-3) is demonstrated by a static seismic analysis in accordance with IEEE 344-1975, with justification. Should analysis fail to show the resonant frequency to be significantly greater than 33 Hz, a test is performed to establish the motor resonant frequency. Motor operability during a seismic event is demonstrated by calculating critical deflections, loads, and stresses under various combinations of seismic, gravitational, and operational loads. The worst case (maximum) values calculated are tabulated against the allowable values. On combining these stresses, the most unfavorable possibilities are considered in the following areas:

1. Maximum rotor deflection
2. Maximum shaft stresses
3. Maximum bearing load and shaft slope at the bearings
4. Maximum stresses in the stator core welds
5. Maximum stresses in the stator core to frame welds
6. Maximum stresses in the motor mounting bolts
7. Maximum stress in the motor feet

Where minor differences exist between items of equipment, analysis is employed to demonstrate that the test results obtained for one piece of equipment are equally applicable to a similar piece of equipment.

The analytical models employed and the results of the analysis are described in Section 4 of the Equipment Qualification Data Packages (WCAP-8587).

3.10N.3 METHOD AND PROCEDURES FOR QUALIFYING SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Where supports for the electrical equipment and instrumentation are within the Westinghouse NSSS scope of supply, the seismic qualification tests and/or analyses are conducted including the supplied supports. The EQDPs contained in WCAP-8587 identify the equipment mounting employed for qualification purposes and establish interface requirements for the equipment to ensure that subsequent inplant installation does not affect the qualification established by Westinghouse.

3.10N.4 OPERATING LICENSE REVIEW

The results of tests and analyses that ensure that the criteria established in Section 3.10N.1 have been satisfied employing the qualification methods described in Section 3.10N.2 and 3.10N.3 are included in the individual EQDPs contained in WCAP-8587.

3.10.1 REPLACEMENT ITEMS

Replacement items are procured and documented as shown in Section 3.10B.4.

3.10.2 REFERENCES FOR SECTION 3.10

3.10-1 NS-CE-692. Letter of July 10, 1975 from C. Eicheldinger (Westinghouse) to D.B. Vasselo (NRC).

3.10-2 WCAP-7536-L (Proprietary) and WCAP-7821 (Non-proprietary), 1971 plus Supplements 1-6, "Seismic Testing of Electrical and Control Equipment (High Seismic Plants)," Povochnik, L.M. et al.

3.10-3 WCAP-7558, 1971. "Seismic Vibration Testing with Sine Beats." Morrone, A.

3.10-4 WCAP-8587, Revision 2. 1979, "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," Butterworth, G. and Miller, R.B.

3.10-5 WCAP-8587, 1978, "Equipment Qualification Data Packages," Supplement 1 1978.

3.10-6 WCAP-8634 (Proprietary) 1975 and WCAP-8695 (Non-proprietary), "General Method of Developing Multi-Frequency Biaxial Test Inputs for Bistables," Jarecki, S.J.

3.10-7 EPRI NP-7484: "Guideline for Seismic Technical Evaluation of Replacement Items for Nuclear Power Plants."

3.10-8 EPRI TR-104871: "Generic Seismic Technical Evaluations of Replacement Items for Nuclear Power Plants."

TABLE 3.10N–1 SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY

Equipment	EQD (1)
Safety-Related Valve Electric Motor Operators	EQDP-HE-4
Garrett Power-Operated Relief Valves (PORV)	EQDP-HE-9
Solenoid-Operated Isolation Valve	EQDP-HE-10A
Electronic Control Module	EQDP-HE-10B
Modulating Valve	EQDP-HE-10C
Large Pump Motors (Outside Containment)	EQDP-AE-2
Canned Pump Motors (Outside Containment)	EQDP-AE-3
Pressure Transmitters	EQDP-ESE-1B
Differential Pressure Transmitters	EQDP-ESE-3B
Resistance Temperature Detectors	EQDP-ESE-5 and 6
Excore Neutron Detectors	EQDP-ESE-8 and 9
Nuclear Instrumentation System (NIS)	EQDP-ESE-10
NIS Source Range Preamplifier	EQDP-ESE-11
Operator Interface Modules (OIMs)	EQDP-ESE-12A
Process Protection Sets	EQDP-ESE-13
Indicators, Post-Accident Monitoring	EQDP-ESE-14
Recorders, Post-Accident Monitoring	EQDP-ESE-15
Solid-State Protection System and Safeguard Test Cabinet (2 Train)	EQDP-ESE-16
Reactor Trip Switchgear	EQDP-ESE-20
Loop Stop Valve Cabinet	EQDP-ESE-23
Reactor Coolant Pump Speed Sensor	EQDP-ESE-24
Differential Pressure Indicating Switch (Group B)	EQDP-ESE-40
Incore Thermocouple	EQDP-ESE-43A
Thermocouple Connectors and Thermocouple Splice	WCAP-10919
Incore Thermocouple Reference Junction Box Splice	EQDP-ESE-43C
Incore Thermocouple Reference Junction Box	EQDP-ESE-44A

1. Equipment Qualification Data Package

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Electrical equipment qualification is an integral part in the design, construction, and operation of Millstone Unit 3. The U.S. Nuclear Regulatory Commission's regulations in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" require that categories of structures, systems, and components be designed to accommodate the effects of both normal and accident plant environmental conditions, and that control measures be employed to ensure the adequacy of design. Specific requirements pertaining to environmental qualification of certain categories of electrical equipment are embodied in Section 50.49 of 10 CFR Part 50, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants."

The Millstone 3 Electrical Equipment Qualification (EEQ) Program complies with 10 CFR 50.49. The EEQ Program ensures the continued qualification of equipment that must function during and following design conditions postulated for design basis accidents and the post-accident duration.

The constituent parts of the EEQ Program include the program basis, verification of equipment operability during and following exposure to plant environmental conditions, and proper installation and maintenance of equipment in the plant. These elements are controlled through a set of administrative documents consisting of a program description, implementing procedures, and reference documents. The procedures are retrievable through the Station's electronic procedure management system.

Program output is generated by Equipment Qualification Records. These documents provide the auditable bases and evidence, which demonstrate that Millstone 3 is compliant with 10 CFR 50.49. EQRs utilize two main sources of design input - Test Report Assessments and an Environmental Specification.

Test Report Assessments (TRA) are design calculations that evaluate and summarize the environmental qualification test report(s). The TRA documents the process of assessing a qualification test report, or analysis, as acceptable qualification documentation for use in the EEQ Program.

The Environmental Specification provides the environmental conditions that are required to qualify electrical equipment in performing its function while exposed to normal and accident operating conditions; and provides the environmental design conditions for use as input in development of Equipment Qualification Records.

Seismic qualification of safety-related mechanical and electrical equipment is presented in Sections 3.9 and 3.10, respectively.

THE INFORMATION PROVIDED IN SECTIONS 3.11, 3.11B, AND 3.11N BELOW REPRESENTS HISTORICAL EEQ PROGRAM DETAILS, WHICH IS NOT SUBJECT TO FUTURE UPDATING.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

This section presents information to demonstrate that the safety-related electrical equipment is capable of performing designated safety-related functions while exposed to applicable normal, abnormal, test, accident, and post-accident environmental conditions. The information presented includes the definition of the applicable environmental parameters, and description of the qualification process employed to demonstrate the required environmental capability. The seismic qualification of safety-related mechanical and electrical equipment is presented in Sections 3.9 and 3.10, respectively.

Sections whose identification numbers include the letter B contain material within the balance-of-plant (BOP) scope, while sections whose identification numbers include the letter N contain material within the nuclear steam supply system (NSSS) scope.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

3.11B.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Safety-related equipment and components are qualified to meet their performance requirements under normal, abnormal, and accident operating conditions. Evidence of qualification is presented in the form of an equipment qualification record (EQR), located in Specification (SP-M3-EE-0353), that reflects the equipment functional capability during the described conditions and required time frame. The EQR also demonstrates that the equipment remains in a safe mode after its safety functions are performed.

Since environmental conditions vary for different areas of the plant, there are several environmental zones. The environmental conditions to which the equipment is exposed depend on the environmental zone in which the equipment is located. The safety-related equipment located within each environmental zone can be determined from a sort of the information contained in the EEQ Master List (located in SP-M3-EE-0353).

Engineering Specification SP-M3-EE-0333 provides a listing of the worst-case environmental conditions and profile figures for various areas in the plant and a listing of the environmental zones for various areas in the plant. These conditions are determined by the criteria defined in Section 3.11B.1.1.

3.11B.1.1 Environmental Conditions

Environmental design conditions are specified in Engineering Specification SP-M3-EE-0333. These environmental conditions are listed by a system of zones, each defining a specific area or areas in the plant. The environmental design conditions in each zone are given for normal, abnormal, and accident conditions as applicable. A further description of each of these conditions is also included; for example, a one-time accident environment due to LOCA and MSLB. Environmental parameters include temperature, pressure, relative humidity, chemicals, spray potential, submergence potential, accident duration, and gamma/beta (where applicable) radiation dose. Where applicable, these parameters are given in terms of a time-based profile.

The Equipment Qualification Program does not establish a fixed limitation on component service life to account for the aging effects of non-seismic vibration. Rather than establish a fixed limitation, the program relies on two complimentary elements to address this aging mechanism. Initially, the seismic qualification process provides a level of confidence that safety related components possess sufficient ruggedness to withstand a vibration environment. Thereafter, regular in-service surveillance and preventative maintenance diagnostics monitor vibration behavior thereby providing early warning against premature service life expiration caused by non-seismic vibration environments.

Normal operating environmental conditions are defined as conditions expected during routine plant operations.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Abnormal operating conditions are any deviations from normal conditions, but do not include accident conditions. These environmental conditions include transients caused by a fault in a system component requiring its isolation from the system, transients caused by a loss of load or power, or any system upset not resulting in a forced outage.

Accident environmental conditions result from design basis accidents (DBA), as described in Section 6.2 and Chapter 15. Safety-related equipment functions as required during all normal, abnormal, and accident events.

3.11B.1.2 Equipment Identification

All safety-related equipment and components for each Class 1E specification, located throughout the plant, are listed in the EQRs which are in separate reports located in Specification (SP-M3-EE-0353). Each device specified is labeled, in the EQRs, with its plant identification number, location, and environmental zone.

3.11B.2 QUALIFICATION TESTS AND ANALYSES

The EQRs, listed in separate reports, detail the Millstone 3 compliance with 10 CFR 50.49 and NUREG-0588, Revision 1, as endorsed by Regulatory Guide 1.89, Revision 1. The requirements of General Design Criterion 1 (GDC 1) of 10 CFR 50, Appendix A are achieved by incorporating performance, design, construction, and testing requirements into equipment specifications, and by the establishment of a system of reviews to ensure conformance with the specified requirements. Appropriate auditable records are maintained in a permanent file. Chapter 17 provides a further definition of compliance to Criterion III of Appendix B, 10 CFR 50.

Protection against earthquakes, as required by GDC 2, is provided by incorporating a description of seismically induced vibrations in equipment specifications and by requiring qualification in accordance with Section 3.10.

The environmental requirements of GDC 4 are addressed in Section 3.11B.1. The Class 1E equipment meets the requirements of GDC 4. The equipment is designed to operate satisfactorily or to fail in a safe mode. Since components are procured for their ability to withstand the environments resulting from both abnormal events and accidents, the Millstone 3 Class 1E equipment meets the requirements of GDC 23.

GDC 50 requirements are achieved by analysis and testing of pressure boundary components to ensure containment integrity. Inservice inspection is performed to demonstrate leaktight integrity of components, such as seals and seats.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

3.11B.2.1 Regulatory Guides

The recommendations provided in the following list of regulatory guides have been included in appropriate equipment specifications. A detailed discussion on compliance with the following regulatory guides is provided in Section 1.8:

1. Regulatory Guide 1.30 - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment

2. Regulatory Guide 1.40 - Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants

3. Regulatory Guide 1.63 - Electrical Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants

4. Regulatory Guide 1.73 - Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

5. Regulatory Guide 1.89 - Qualification of Class 1E Equipment for Nuclear Power Plants

6. Regulatory Guide 1.131 - Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants

3.11B.2.2 Safety-Related (Class 1E) Equipment and Component Qualifications

Safety-related equipment located in a mild environment is environmentally qualified based on the following:

1. equipment is purchased based on the normal and abnormal environmental conditions in which the equipment is required to function;

2. a periodic maintenance, inspection, and/or replacement program based on sound engineering practice and recommendations of the equipment manufacturer which is updated as required by the results of an equipment surveillance program;

3. a periodic testing program to verify operability of safety-related equipment within its performance specification requirements; and

4. An equipment surveillance program which includes periodic inspections, analysis of equipment and component failures, and a review of the results of preventive maintenance and periodic testing programs.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation or during anticipated operational occurrences. This type of environment is not in the scope of the EEQ program per 10 CFR 50.49(c)(3).

Environmental qualification of all safety-related equipment meets the requirements of IEEE Std. 323-1974, Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations; the intent of NUREG-0588, Interim Staff Position on Environmental Qualification

Electrical equipment within the scope of the EEQ Program, has a designated qualified life derived in accordance with IEEE Std. 323-1974 by type test, or combination of type test/analysis and/or documented operating experience.

Equipment specifications for components within the EQ Program, define at least the worst case envelope service condition anticipated for particular types of Class 1E electrical equipment.

Qualification by Type Testing or Combination of Type Test/Analysis and/or Documented Operating Experience

Testing is the preferred method of qualification. Analysis has been used to verify or supplement test results.

Service conditions simulated during the test were reviewed to ensure that they enveloped the accident environments. The test duration and environmental parameters utilized in the test were reviewed to ensure that they equaled or exceeded specified values. When tests were found to be less severe than specified, a determination of the adequacy of the test is made on a case-by-case basis.

The test specimen model, design, and construction material are reviewed against the equipment being qualified to verify applicability of test results.

The selected test sequence is reviewed and, when determined not to be in accordance with the guidelines of IEEE Std. 323-1974, is reevaluated for adequacy.

Tests which are successful using components that had not been pre-aged are considered acceptable, provided the components do not contain materials known to be susceptible to significant degradation due to aging effects.

Operational modes tested are reviewed to ensure that they are representative of the actual application requirements as defined in the procurement documents. The length of time that each item of equipment is required to operate is reviewed against test data to ensure that the equipment's designated life is greater than the length of time the equipment is required to operate.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Any failures identified during the environmental qualification effort are reviewed and evaluated relative to their effect on the ability of the component to perform its required function. If a component fails at any time during the test, the applicability of the test with regard to demonstrating the ability of the component to function for the entire period prior to the failure is considered on a case-by-case basis.

Where seals are included as part of the component, the test results and conclusions include the seals. Materials used for terminating cables and similar components are addressed separately in the EQR.

Qualification by a combination of methods (test, evaluation, analysis) is identified in the EQR. A determination of the adequacy of the qualification methods used is made on a case-by-case basis.

Replacement Program

For equipment with a designated life less than design plant life, a maintenance surveillance replacement program based upon test data and analysis is utilized to lengthen the qualified life of all equipment to that of the plant design life. When new test data becomes available, it is used to modify the program as necessary.

Aging

Considerations made for materials which are susceptible to thermal, vibration, electrical, mechanical, and/or radiation aging are addressed in the applicable Environmental Qualification documentation. The order of application of the simulated aging conditions is reviewed to ensure that the most severe sequence is applied. Known synergistic effects, or those found during testing, are investigated on a case-by-case basis.

Equipment, within the EQ Program is assessed for operability at specified time intervals by systematic application of the plant periodic test program. When this program detects any equipment degradation, the equipment or component is analyzed. Any equipment or component found to have unsatisfactory aging characteristics is reviewed to determine necessary action.

Equipment or component operability is defined as the capability to perform its specified function.

Margin

Qualification type tests are reviewed to verify that adequate margin exists between the most severe specified service conditions of the plant and the conditions used in type testing. This accounts for normal variations in commercial production of equipment and reasonable errors in defining satisfactory performance. Increased levels of testing, number of test cycles, and test duration are considered as methods of ensuring adequate margin.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Submergence

Equipment has been evaluated for submergence as a result of Design Basis Accidents (e.g. LOCA or HELB). Where submergence is possible, it is given as an environmental parameter in Appendix 3B. Equipment, within the EQ Program, located in areas with submergence potential, is either qualified to operate submerged, or it is electrically isolated, is administratively controlled, or its location has been verified to be above the submergence level.

Documentation

During the review of submitted qualification documentation, when it is determined that actual test data are not submitted (i.e., test summaries and/or certificates of conformance only are submitted), requests for actual test data are initiated. If such data are not submitted because the data are considered proprietary by the manufacturer, an audit of these data are made. A certificate of conformance by itself is not acceptable unless it is accompanied by verification of test data and information on the qualification program.

A final determination on the acceptability of qualification documentation is made based on the criteria that the documentation available is organized in an auditable form and sufficient to justify the conclusions reached.

3.11B.3 QUALIFICATION TEST RESULTS

This section provides a discussion of the test results associated with environmental qualification of equipment.

3.11B.3.1 Nuclear Steam Supply System Equipment Qualification Program

Section 3.11N describes the Nuclear Steam Supply System Equipment (NSSS) Qualification Program.

3.11B.3.2 Qualification of Safety-Related Equipment (BOP) not covered by Section 3.11.

Qualification test results and supporting documentation for safety-related, non-NSSS equipment are summarized in the EQR (Section 1.7).

3.11B.4 LOSS OF VENTILATION

The following design features preclude the possibility of a total system failure for ventilation systems serving areas where equipment required to function during and following a DBA is located.

1. All HVAC systems serving these equipment areas are designed to Seismic Category I requirements (Section 9.4).

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

2. Sufficient redundancy in equipment and power supplied is provided so that no single active component failure can result in loss of HVAC system function.

3. Redundant HVAC systems are connected to separate and independent onsite standby power supplies to ensure system operation upon loss of offsite power (Section 8.3).

4. Failure modes for isolation valves and dampers are described in Section 9.4. Valves or dampers required for system operation after postulated accidents fail in the safe position.

5. Equipment outside the containment building required to operate following a LOCA or a high-energy pipe break is so located that it is not exposed to resultant post-accident ambient conditions or is designed to withstand these conditions.

6. Instrumentation and controls which incorporate audible and visual alarms enable the operator to monitor the HVAC systems' performances. In the event of system malfunction, the operator has the capability to switch manually to the HVAC standby equipment.

Based upon the above features and the detailed HVAC systems' evaluations in Section 9.4, only partial loss of the ventilation or air-conditioning system could occur in areas where equipment required to function during and/or following a DBA is located. This loss would not adversely affect the availability of the safety-related equipment to function during and following a DBA. The effects of any partial loss of HVAC is reflected in the environments of Engineering Specification SP-M3-EE-0333.

3.11B.5 CHEMICAL AND RADIATION ENVIRONMENT

Components of safety-related systems and their associated instrumentation and electrical equipment located inside the containment structure and elsewhere, which are required to function during and subsequent to an accident, are designed to operate under normal, abnormal, accident, and post-accident environmental conditions that may occur at their installation locations.

3.11B.5.1 Radiation Environment

Equipment is qualified for the radiation doses found in Engineering Specification SP-M3-EE-0333. These doses were calculated using source terms given in Regulatory Guides 1.7 and 1.89, and NUREG 0588.

1. For the design basis loss of coolant accident, which completely depressurizes the primary system, 100 percent of the core noble gases, 50 percent of the halogens, and 1 percent of the solid fission products are released into the containment atmosphere. The sump water is assumed to contain 50 percent of the core halogens and 1 percent of solid fission products. The airborne source is assumed to be 100 percent noble gases and 50 percent halogens.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

2. For the gamma dose from airborne activity, no credit is taken for either internal shielding or radioactivity reduction by sprays or other removal mechanisms except radioactive decay.

3. The beta dose from airborne activity inside containment is based on the infinite medium dose. No beta dose has been considered for components that are sealed in shields which would preclude penetration of the beta particles.

4. LOCA dose values are derived from integration for a period of 1 year from the commencement of the accident. For HELB and fuel handling accidents, integration is over 1 month.

5. Integrated doses for normal operation are the product of the dose rate for full power operations over a 40-year period, corrected by a factor of 0.8 for plant availability.

6. Equipment located within the containment building is qualified for radiation doses comprised of post-accident airborne nuclide concentrations, airborne nuclide concentration due to primary coolant leakage inside containment during normal operations, and location dependent doses from sources containing radioactive liquid, where applicable.

7. Equipment outside the containment building is qualified to integrated doses from post-accident radiation emanating from the containment structure, emergency fluid systems containing recirculating cooling water, and radiation sources resulting from normal operations.

8. Doses due to a fuel handling accident, in addition to doses resulting from normal operation sources, have been used for qualification of equipment in the fuel building.

Assumptions include:

3.11B.5.2 Chemical Environment

A discussion of the LOCA and the source terms is given in Section 15.6. The chemical composition and resulting pH of the spray water in the containment atmosphere, the liquids in the reactor core, and ESF sumps are identified in Section 6.1. The pH range (4.15 – approximately 11) of the containment sump water following a LOCA is achieved from the trisodium phosphate dodecahydrate baskets located in the containment sump structure. (See Section 6.2.2.)

The concentration of chemicals in the containment sprays is equivalent to or more severe than that resulting from the most limiting mode of operation, including a single failure in the spray system.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

3.11N ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section presents information to demonstrate that the safety-related electrical equipment of the engineered safety features and the reactor protection systems are capable of performing their designated safety-related functions while exposed to applicable normal, abnormal, test, accident, and post-accident environmental conditions. The information presented includes identification of the safety-related equipment that is within the scope of the Westinghouse Nuclear Steam Supply System (NSSS) and for each item of equipment, definition of the applicable environmental parameters, and description of the qualification process employed to demonstrate the required environmental capability. The seismic qualification of safety-related mechanical and electrical equipment is presented in Sections 3.9 and 3.10, respectively.

3.11N.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

A complete list of safety-related equipment within the Westinghouse NSSS scope of supply that is required to function during and subsequent to an accident is presented in Table 3.11N–1. The plant specific environmental parameters are discussed in Section 3.11B.1 for normal operating and for accident conditions together with the time each item of equipment is required to perform post-accident, when applicable.

3.11N.2 QUALIFICATION TESTS AND ANALYSIS

3.11N.2.1 Environmental Qualification Criteria

The methods of meeting the general requirements for environmental design and qualification of safety-related equipment as described by GDC 1, 2, 4, and 23 are described in Section 3.1. Additional specific information concerning the implementation of GDC 23 is provided in Section 7.2.2.2. The general methods of implementing the requirements of Appendix B to 10 CFR 50 are described in Chapter 17. Regulatory Guides 1.40, 1.73, and 1.89 concerning environmental qualification are addressed in Section 1.8.

Westinghouse meets the Institute of Electrical and Electronic Engineers (IEEE) Standard 323-1974, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations, including IEEE Standard 323a-1975, the Nuclear Power Engineering Committee (NPEC) Position Statement of July 24, 1975, by either type test, operating experience, analysis, or an appropriate combination of these methods. Westinghouse satisfies this commitment by employing the methodology described in the final staff approved version of WCAP-8587.

3.11N.2.2 Performance Requirements for Environmental Qualification

In response to the NRC staff request for additional detailed information on the qualification program, Westinghouse submitted Supplement 1 to WCAP-8587. The latest revision of this supplement, Supplement 1, WCAP-8587, contains an equipment qualification data package (EQDP) for every item of safety-related electrical equipment supplied by Westinghouse within the NSSS scope of supply. Table 3.11N–1 identifies the equipment supplied by Westinghouse for this application and identifies the applicable EQDP contained in Supplement 1 of WCAP-8587.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Each EQDP in Supplement 1 contains a section entitled Performance Specification. This specification establishes the safety-related functional requirements of the equipment to be demonstrated under normal, abnormal, test, accident, and post-accident conditions. The environmental qualification parameters, e.g., temperature, humidity, pressure, radiation, etc, employed by Westinghouse for generic qualification purposes are also identified in the specification, as applicable. The parameters employed have been reviewed to ensure that the plant specific conditions as described in Section 3.11B.1 have been enveloped.

3.11N.2.3 Methods and Procedures for Environmental Qualification

The basic methodology employed by Westinghouse for qualification of safety-related electrical equipment is described in WCAP-8587. Each EQDP (Supplement 1, WCAP-8587) contains a description of the qualification program plan for that piece of equipment. Qualification may be demonstrated by either type test, operating experience, analysis, or a combination of these methods.

3.11N.3 QUALIFICATION TEST RESULTS

Qualification test results and supporting documentation for safety-related NSSS equipment are summarized in the EQDPs and their associated test reports.

3.11N.4 LOSS OF VENTILATION

Loss of ventilation is discussed in Section 3.11B.4.

3.11N.5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

The plant-specific estimates of the radiation dose incurred by equipment during normal operation are discussed in Section 3.11B.1. The estimated doses and chemical conditions following an accident are defined in Section 3.11B.1. The radiation and chemical environments for which the NSSS scope equipment is qualified are defined in the performance specification of the applicable EQDP contained in Supplement 1, WCAP-8587.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

3.11.6 REFERENCES FOR SECTION 3.11

3.11-1 WCAP-8587, Revision 2, 1979, Butterworth, G. and Miller, R.B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment."

3.11-2 WCAP-8587, 1978, "Equipment Qualification Data Packages, Supplement 1."

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

TABLE 3.11N–1 SAFETY-RELATED EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY

Equipment	EQD [(1)]
Externally Mounted Limit Switches	EQDP-H-3 and 6
Safety-Related Valve Electric Motor Operators	EQDP-HE-4
Garrett Power-Operated Relief Valves (PORV)	EQDP-HE-9
Solenoid-Operated Isolation Valve	EQDP-HE-10A
Electronic Control Module	EQDP-HE-10B
Modulating Valve	EQDP-HE-10C
Large Pump Motors (Outside Containment)	EQDP-AE-2
Canned Pump Motors (Outside Containment)	EQDP-AE-3
Charging Pump Motors	EQDP-AE-5A
Pressure Transmitters	EQDP-ESE-1A and 1B
Differential Pressure Transmitters	EQDP-ESE-3B and 4
Resistance Temperature Detectors (well mounted)	EQDP-ESE-6
	WCAP-11587
Excore Neutron Detectors	EQDP-ESE-8A
Nuclear Instrumentation System (NIS)	EQDP-ESE-10
Operator Interface Modules (OIMs)	EQDP-ESE-12A
Process Protection Sets	EQDP-ESE-13
Indicators, Post-Accident Monitoring	EQDP-ESE-14
Recorders, Post-Accident Monitoring	EQDP-ESE-15
Solid-State Protection System and Safeguard Test Cabinet (2 Train)	EQDP-ESE-16
Reactor Trip Switchgear	EQDP-ESE-20
Pressure Sensor	EQDP-ESE-21
Loop Stop Valve Cabinet	EQDP-ESE-23A
Reactor Coolant Pump Speed Sensor	EQDP-ESE-24A
Differential Pressure Indicating Switch (Group B)	EQDP-ESE-40A
Resistance Temperature Detectors (surface mounted)	EQDP-ESE-42A
Incore Thermocouple	EQDP-ESE-43A

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

TABLE 3.11N–1 SAFETY-RELATED EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY (CONTINUED)

Equipment	EQD [1]
WC Series Connectors and Hardline Extension Cable	EQDP-ESE-43F
Potting Adaptor/Cable Splice Assembly	EQDP-ESE-43G
Thermocouple Connectors and Thermocouple Splice	WCAP-10919
Potting Adaptor/Splice Assembly	WCAP-10920
Incore Thermocouple Reference Junction Box	EQDP-ESE-44A

NOTE:

1. Equipment Qualification Data Package

APPENDIX 3A - COMPUTER PROGRAMS FOR DYNAMIC AND STATIC ANALYSIS OF SEISMIC CATEGORY I STRUCTURES, EQUIPMENT, AND COMPONENTS

PART 1 - Structures

PART 2 - Equipment and Components

PART 3 - Piping Systems

3A.1 STRUCTURES

The following are the major computer programs that are used in dynamic and static analysis of Seismic Category 1 structures. (Note: Minor programs such as post processors and simple programs are not listed.)

1. STRUDL II - Structural Analysis Program
2. SHELL 1 - Shell Analysis
3. STRUDLSW - Structural Analysis Program
4. ASAAS - Asymmetric Stress Analysis of Axisymmetric Solids
5. TAC2D - Heat Transfer Program
6. Time History Program - Dynamic Analysis
7. PLAXLY - Finite Element Soil-Structure Analysis for Plain Strain Problems
8. ANSYS
9. MAT5 - Axisymmetric Mat Analysis
10. MEMBRANE
11. SBMMI
12. GHOSH - WILSON

3A.1.1 STRUDL II

The finite element method (Cheung and Ziankiewicz 1967) provides for the solution of a wide range of solid mechanics problems. Its implementation within the context of the STRUDL analysis facilities expands these for the treatment of plane stress, plane strain, plate bending, shallow shell, and three-dimensional stress analysis problems.

STRUDL II (MIT 1968; 1971) has been designed as a modified subsystem of the Integrated Civil Engineering System (ICES), which was designed and formulated at the Massachusetts Institute of Technology, Department of Civil Engineering.

STRUDL II also provides a dynamic analysis capability for linear elastic structures undergoing small displacements. Either free or forced vibrational response may be obtained and, in the later case, the forcing functions may be in the form of time histories or response spectra.

Seismic Category I structures are analyzed for seismic effect using the dynamic analysis capability of STRUDL II. The analysis yields frequencies of vibrations, mode shapes, displacements, velocities, accelerations, and forces.

STRUDL II is a recognized program in the public domain. Version 2- modification 2 (June 1972) of STRUDL is used. The software system is IBM-MVT-RELEASE 20.7. The hardware configuration is IBM-370 - Model 165. The Structural Design Language Engineering Users Manual. Vol 1, Frame Analysis. Department of Civil Engineering.

3A.1.2 SHELL 1

3A.1.2.1 General Description

This program is based upon the general numerical procedure proposed by B. Budiansky and P.P. Radkowski (1963) and Greenbaum (1963) to analyze a shell of revolution subjected to arbitrary loadings.

This is a finite difference stress analysis computer code. It can be used to determine the forces, moments, shears, displacements, rotations, and stress in a thin shell of revolution subject to arbitrary loads expanded in a Fourier series of up to 150 terms. Single layer shells with up to 30 simply connected branches may be analyzed. Poisson's ratio may change at discontinuity points, and Young's modulus and the thermal coefficient of expansion may be different at each point. The allowed types of loading include elastic restraints, pressures in three orthogonal directions, temperature changes which may have a gradient through the shell thickness, and simplified input for weight of the shell or earthquake forces.

The equilibrium equations for a thin shell are based on Sanders linear theory (Sanders 1959). Sanders' equations are expanded and modified slightly to handle a broader range of problems. All pertinent load, stress, and deformation variables are expanded into a Fourier series. The individual Fourier components of stress and deflection are found separately by solution of the finite difference forms of the appropriate differential equations. The algorithm used to solve these equations is a minor modification of the Gaussian elimination method.

3A.1.2.2 Program Verification - Thin-Wall Cylinder

A long thin-walled circular cylinder is subjected to a constant internal pressure distribution. A solution of this problem may be obtained (Roark 1965).

The pertinent parameters of the cylinder are presented in Table 3A.1.2–1.

The following solution can be verified (Roark 1965).

$$\delta_R = \frac{PR^2}{Et} \tag{3A.1.2-1}$$

$$\delta_{\theta} = \frac{PR}{t} \quad (3A.1.2\text{-}2)$$

The cylinder is idealized by 10 elements, as shown on Figure 3A.1.2–1. Computer results are presented in Table 3A.1.2–2 along with the results obtained from Equations 3A.1.2-1 and 3A.1.2-2. As can be seen, the computer results compare very favorably. Therefore, this problem verifies the accuracy of SHELL 1.

where:

s_{θ} = hoop stress

δ_R = radial stress

P = pressure

R = radius

t = thickness

E = Young's modulus

3A.1.3 STRUDL-SW

3A.1.3.1 General Description

The STRUDL-SW computer code uses the stiffness analysis method to analyze a wide range of structural problems. It handles two and three-dimensional trusses and frames, having linear elastic members and statically applied loading.

STRUDL-SW has been documented by bench marking procedures against the GTSTRUDL computer code. GTSTRUDL is a recognized program in the public domain.

3A.1.4 ASAAS (Asymmetric Stress Analysis of Axisymmetric Solids)

3A.1.4.1 General Description

This is a finite element computer code (Crose 1971). It can be used to determine stresses and displacements in arbitrary axisymmetric solids, including problems involving asymmetric mechanical and thermal loads and asymmetric temperature-dependent mechanical properties. All dependent variables, including the mechanical properties, are input by Fourier series expansions of the circumferential coordinate. The mechanical loads can be surface pressures, surface shears, and nodal point forces.

The explicit stiffness relations for the axisymmetric solid ring elements of the triangular cross section are based on the classical theorem of potential energy and the assumption that, within any element, the displacement variation in the R-Z plane is linear. All dependent variables, including the material properties, are expanded into the Fourier series. The harmonics are coupled and all the equilibrium equations are solved simultaneously. The algorithm used to solve the equations is a block modified square root Cholesky method with iterative refinement (Crose 1971).

3A.1.4.2 Sample Program Verification - Harmonic Axisymmetric Plane Strain

An infinitely long, solid, circular cylinder is subjected to cos θ and cos 2θ pressure distributions. A closed-form solution of this problem may be obtained (Love 1944).

The pertinent parameters of the cylinder are presented in Table 3A.1.4–1.

The following solution can be verified (Love 1944):

$$\sigma_r = P_o\left(\frac{r}{a}\cos\theta + \cos 2\theta\right) \tag{3A.1.4-1}$$

$$\sigma_\theta = P_o\left[3\frac{r}{a}\cos\theta + \frac{2r^2 - a^2}{a^2}\cos 2\theta\right] \tag{3A.1.4-2}$$

$$\sigma_{r_\theta} = P_o\left[\frac{r}{a}\sin\theta - \frac{r^2 - a^2}{a^2}\sin 2\theta\right] \tag{3A.1.4-3}$$

$$\begin{gathered} U_r = U_{r1} + U_{r2} \\ U_{r1} = \frac{P_o(1-4\upsilon)(1+\upsilon)r^2}{2Ea}\cos\theta \\ U_{r2} = P_o\left(\frac{1+\upsilon}{E}\right)\left(r - \frac{2\upsilon r^3}{3a^2}\right)\cos 2\theta \end{gathered} \tag{3A.1.4-4}$$

$$\begin{gathered} U_\theta = U_{\theta_1} + U_{\theta_2} \\ U_{\theta_1} = \frac{P_o(5-4\upsilon)(1+\upsilon)r^2\sin\theta}{2Ea} \\ U_{\theta_2} = P_o\left(\frac{1+\upsilon}{E}\right)\left[\left(1 - \frac{2\upsilon}{3}\right)\frac{r^3}{a^2} - r\right]\sin 2\theta \end{gathered} \tag{3A.1.4-5}$$

The cylinder is idealized by 16 elements, as shown on Figure 3A.1.4–1. Computer results are depicted on Figure 3A.1.4–2, along with the exact results obtained from Equations 3A.1.4-4 and 3A.1.4-5. As can be seen, the computer results are very close to the exact results. Therefore, this problem verifies the accuracy of ASAAS for mechanical loading problems where material properties are not variable.

3A.1.5 TAC2D (A General Purpose Two-Dimensional Heat Transfer Computer Code)

3A.1.5.1 General Description

This is a finite difference computer code (Petersen 1969) which can be used to determine steady-state and transient temperatures in two-dimensional problems. The configuration of the body to be analyzed is described in the rectangular, cylindrical, or circular (polar) coordinate system by orthogonal lines of constant coordinate called grid lines. These grid lines specify an array of nodal elements. Nodal points are defined as lying midway between the bounding grid lines of these elements. A finite difference equation is formulated for each nodal point in terms of its capacitance, heat generation, and heat flow paths to neighboring nodal points. The equations for all the nodal points are assembled and solved using an implicit alternating gradient algorithm.

3A.1.5.2 Program Verification

A sample problem is presented to compare the results from TAC2D with an analytical solution. The objective is to show that the TAC2D program yields the correct solution.

The problem is to determine the transient temperature distribution in a right circular cylinder which is initially at temperature T. At time, t = 0, the temperature at the surface is instantaneously changed to T and maintained at that value.

Mathematically, the problem is defined by the following equations:

$$\frac{1}{r}\left(\frac{d}{dr}\left(\frac{rdT}{dr}\right)\right)+\frac{d^2T}{dZ^2} = \frac{1}{\alpha}\left(\frac{dT}{dt}\right) \qquad (3A.1.5\text{-}0)$$

$$0 \le r \le R \qquad (3A.1.5\text{-}1)$$

$$T(r,z,0) = T_1 \qquad (3A.1.5\text{-}2)$$

$$T(R,z,t) = T_2 \qquad (3A.1.5\text{-}3)$$

$$T\left(r, \pm\frac{L}{Z}, t\right) = T_2 \qquad (3A.1.5\text{-}4)$$

where:

t = is the time,

r = the radius,

z = the axial coordinate,

R = the outside radius of the cylinder,

L = the length of the cylinder, and

σ = the diffusivity.

Further,

$$\sigma = \frac{k}{\rho c} \qquad (3A.1.5\text{-}5)$$

where:

k = the thermal conductivity

ρ = the density,

c = the specific heat capacity.

For the specific problem analyzed, the following numerical values were used:

R = 12.0 inches

L = 48.0 inches

K = 20.0 Btu/hr-ft-°F

ρc = 40.0 Btu/cu ft-°F

T_1 = 0.0 °F

3A.1.5.2.1 Analytical Solution

It may be shown (Carlslaw and Jaeger 1959) that the solution is:

$$\frac{T - T}{T_2 - T_1} = 1 - f(z, t)\ g\ (r, t) \qquad (3A.1.5\text{-}6)$$

$$f(z, t) = \frac{4}{\Pi} \sum_{n=0}^{\infty} \frac{(-1)^n}{(2n+1)} e^{-\propto t \frac{\Pi}{2L}(2n+1)^2} \cos \frac{\Pi z}{2L}(2n+1) \qquad (3A.1.5\text{-}7)$$

$$g(r, t) = \frac{2}{R} \sum_{m=1}^{\infty} \frac{J_{o(r\psi m)}}{\psi_m J_i(R\psi_m)} e^{-\propto \psi_m^2 t} \qquad (3A.1.5\text{-}8)$$

where the Ψm are the roots of:

$$J_o(R\psi) = 0 \qquad (3A.1.5\text{-}9)$$

The roots m of Equation 3A.1.5-9 and the Bessell functions J and (J_i) are tabulated (Jahnke et al., 1945) and need not be computed.

From the definition of the problem there is symmetry about the geometric center of the cylinder and the origin of the coordinate system taken at that point, as is reflected in the boundary conditions, Equations 3A.1.5-3 and 3A.1.5-4.

3A.1.5.2.2 Numerical Solution With TAC2D

A cross section of the problem model for TAC2D is shown on Figure 3A.1.5–1 The model extends only to the axial midplane of the cylinder where an adiabatic boundary may be specified by virtue of the symmetry condition described above. The solid material is represented by one material block. The boundary conditions on the four external boundaries are described by Coolants 1 through 4 (specifically, Coolant Blocks 1 through 4). The material and coolant thermal parameters, as specified by the input functions, are given in Table 3A.1.5–1. All coolants have the standard specific heat of 1.0 Btu per pound-°F (Btu/lb-°F). Coolants 1 and 2, which represent the adiabatic external boundaries, have the standard heat transfer coefficient of 10 Btu/hr-sq ft-°F and the standard flow rate of 10 pounds per hour.

3A.1.5.2.3 Comparison of TAC2D Solution with the Analytical Solution

A comparison of the output from the code with the series solution is shown on Figure 3A.1.5–2. The temperature-versus-time function is plotted at three representative points within the cylinder. It can be seen that the results from TAC2D are almost identical to the series solution results. The maximum difference between the two sets of results is about 2°F out of a mean magnitude of 100°F.

3A.1.6 Time History Program

3A.1.6.1 General Description

The Time History Program computes time history response and amplified response spectra (ARS) at any mass location of a lumped mass system due to a synthetic earthquake input. The responses are computed by integration of the modal equations of the system by exact methods (Nigam and Jennings 1968). The program’s main application is the generation of ARS used in the design of Seismic Category I equipment and piping.

3A.1.6.2 Sample Problem

The Time History Program’s solution to a test problem is substantially identical to the solution obtained using STRADYNE. STRADYNE is a recognized program in the public domain. The sample problem used consists of a structure subjected to an earthquake time history record. The structure is idealized by five lumped masses interconnected by five elastic beam elements, as shown on Figure 3A.1.6–1.

Peak acceleration and displacement as well as the horizontal ARS at the top mass point are compared in Tables 3A.1.6–1 and 3A.1.6–2, respectively.

3A.1.7 PLAXLY

3A.1.7.1 General Description

The PLAXLY program provides a numerical solution for the dynamic analysis of plane systems under general dynamic loadings. This program works with a two-dimensional plane-strain finite element idealization of the soil structure interaction problem.

The original version of PLAXLY was developed at the University of California in Berkeley (Waas 1972). It was later modified and extended at Stone & Webster (SWEC) to incorporate transient seismic excitations, nonlinear soil behavior, and lumped mass representations of the structures.

3A.1.7.2 Sample Problem

The PLAXLY program's solution to a test problem is substantially identical to the solution obtained by using the FLUSH program. FLUSH is a recognized program in the public domain.

The sample problem used consists of a structure represented by five lumped masses interconnected by four elastic beam elements. This structural model is connected to a finite element representation of the soil, as shown on Figure 3A.1.7–1.

The horizontal amplified response spectra (ARS) at the top mass point are compared on Figure 3A.1.7–2.

3A.1.8 MAT5 (Circular Mat with Axisymmetric Loading)

3A.1.8.1 General Description

This program is based upon the general numerical procedures proposed by Boris N. Zhemoshkin (1962) to analyze a circular plate on an elastic foundation. It is used to determine moments, shears, vertical deflections, radial displacements, tangential and radial inplane forces, plus rotations of the circular plate subjected to axisymmetric loadings.

The soil subgrade may be modeled as either a Winkler (1867) or a Boussinesqu (1885) type elastic foundation.

3A.1.8.2 Program Verification

The results of this program have been reviewed in accordance with standard review procedures in effect at the time of use and, based on the user's and reviewer's knowledge of plate and shell theory, have been found to be satisfactory.

3A.1.9 ANSYS

The ANSYS computer program is a large-scale general purpose computer program for the solution of several classes of engineering analysis problems. ANSYS is capable of analyzing structures with static and dynamic loadings, elastic and plastic member properties, and small and large deflections.

The matrix displacement method of analysis based upon finite element idealization is employed throughout the program.

ANSYS is a recognized program in the public domain.

3A.1.10 MEMBRANE (Membrane Stress Analysis)

3A.1.10.1 General Description

This program computes membrane stresses and strains in containment structures due to dead loads, internal pressure, and temperature gradients across the wall. It analyzes cylinders, cones, and spherical domes which consist of a fully cracked reinforced concrete section with a steel liner.

Stresses are computed by shell membrane theory (Billington 1965). The program automatically considers the effect of the uplift force acting on the roof of a cone or cylinder.

3A.1.10.2 Program Verification

As an example, a cylindrical shell was analyzed by the program and by a hand calculation. Table 3A.1.10–1 presents the pertinent parameters of the cylinder. The comparison of results is given in Table 3A.1.10–2.

As can be seen, the computer program's results compare very favorably. This problem verifies the accuracy of MEMBRANE.

3A.1.11 SBMMI (Single Barrier Mass Missile Impact)

3A.1.11.1 General Description

This program computes the elasto-plastic structural response of a barrier due to the following type of loads: (a) static loads; (b) suddenly applied constant dynamic loads which remain permanently on the structure; (c) suddenly applied constant dynamic loads representing missile impact with a finite force and specific momentum; and (d) suddenly applied dynamic load of zero time duration and specific momentum representing missile impact. The barrier is modelled as a single barrier mass and a non-linear spring, with the above loads applied. The equation of motion is integrated in time assuming constant acceleration in each time step (Billington 1965).

3A.1.11.2 Program Verification

As an example, the program SBMMI was run for each load case separately. The computer's result were compared to those obtained from a hand calculation. The hand calculation was based on the elasto-plastic response charts found in Billington (1965) Table 3A.1.11–1 presents the pertinent model and load parameters. The hand and computer program results are compared in

Table 3A.1.11–2. It can be seen that the computer program's result compare very favorably. This problem verifies the accuracy of the SBMMI computer program.

3A.1.12 GHOSH-WILSON

3A.1.12.1 General Description

Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loadings, known as the GHOSH-WILSON computer code, is a finite-element based computer program developed by S. Ghosh and E. Wilson and modified by Stone & Webster as Code ST-200.

GHOSH-WILSON is capable of performing static and dynamic analysis of complex axisymmetric structures subjected to any arbitrary static (mechanical and temperature) and dynamic loading.

The method used to represent the three-dimensional continuum is either as an axisymmetric thin shell, a solid of revolution, or a combination of both. The arbitrary loading in the circumferential direction is represented by a fourier series, and the analysis is carried out for each term and summed up for the total response.

Hamilton's variational principle is used to derive the equation of motion. This leads to a diagonal mass matrix and a stiffness matrix and load vector which is consistent with the assumed displacement field. The equations of motion are solved numerically in the time-domain by direct integration using the Wilson method.

The input required by GHOSH-WILSON is a description of geometry, materials, and boundary conditions. Loadings, damping factors, and time intervals for integration should be provided for each fourier term. Additional inertias can be added at joints during a dynamic analysis.

GHOSH-WILSON provides time history responses of the resultant forces, moments, shears, displacements, rotations, accelerations, and stresses at each node for the dynamic analysis. Maximum responses can also be obtained for each Fourier term.

3A.1.12.2 Program Verification

<u>Static Case</u>

A cylinder is subjected to a constant internal pressure. The cylinder is modeled using the shell element, rectangular element, and the triangular element. The solutions for all three types of elements agree very well.

<u>Dynamic Case</u>

A cylinder simply supported at both ends is subjected to a suddenly applied load at midspan. The solution of the equations of motion is obtained by the direct integration method. The cylinder is modeled using rectangular elements. The GHOSH-WILSON solution (displacement under the applied load) is compared to the solution using the ANSYS computer code. The results compare favorably.

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TABLE 3A.1.2–1 THIN-WALL CYLINDER, PERTINENT PARAMETERS

Dimensions and Properties	Loading and Boundary Conditions
R = 25 inches	At z = 0 inch; $F_r = M = \delta_z = 0$
l = 20 inches	At z = l = 20 inches; $F_r = M = F_z = 0$
t = 0.5 inch	P = 75 psi
E = 28 x 10^6 psi	
Poisson's ratio = 0.3	
M = Moment on free edge	
F_r = Radial force	
F = Force in z - direction	
δ_z = Displacment in z - direction	

TABLE 3A.1.2–2 EXACT AND COMPUTER STRESSES FOR THIN-WALL CYLINDER

Variable	Exact	Shell 1
δ_R (inch)	3.348×10^{-3}	3.348×10^{-3}
σ_R (psi)	3,750	3,750

TABLE 3A.1.4–1 INFINITELY LONG SOLID CYLINDER, PERTINENT PARAMETERS

Dimensions and Properties	Loading and Boundary Conditions
$r_o = a$	$P_r = P_o (\cos \Theta + \cos 2 \Theta)$
$l = a$	$\tau_{ro} = P_o \sin \Theta$
$E = 10 \times 10^6$ psi	$U_z = 0$
$Z = 0.25$	At $r = 0$, $U_r = 0$
$a = 1$ inch	$P_o = 10{,}000$ psi

TABLE 3A.1.5–1 INPUT THERMAL PARAMETER FUNCTIONS FOR TAC2D SAMPLE PROBLEM

Material Thermal Parameters	Coolant Thermal Parameters
SPEC1 (X)= 40.0	H3A (X) = 1.0 x 10^8
RCOH1 (X)= 20.0	FL03A (X) = 1.0 x 10^8
ACON1 (X)= 20.0	TIN3A (X) = 1,460
	H4A (X) = 1.0 x 10^8
	FLO4A (X) = 1.0 x 10^8
	TIN4A (X) = 1,460

TABLE 3A.1.6–1 PEAK ACCELERATION AND DISPLACEMENT

	Time History Program	Stardyne Program
Peak Acceleration	0.922 g	0.922 g
Peak Displacement	0.352 in	0.352 in

TABLE 3A.1.6–2 HORIZONTAL AMPLIFIED RESPONSE SPECTRA

(Two Percent Oscillator Damping)

Period (Seconds)	Time History Program (g)	Stardyne Program (g)
0.02	0.929	0.928
0.04	0.999	0.999
0.06	1.040	1.039
0.08	1.301	1.301
0.10	1.253	1.252
0.12	1.694	1.697
0.14	2.419	2.419
0.16	3.158	3.153
0.18	5.962	5.961
0.20	11.104	11.101
0.22	6.777	6.802
0.24	4.694	5.025
0.26	3.441	3.451
0.28	2.576	2.576
0.30	2.417	2.428
0.34	1.613	1.603
0.38	1.720	1.731
0.42	1.493	1.491
0.46	1.487	1.507
0.50	1.201	1.201
0.70	0.653	0.660

FIGURE 3A.1.2–1 ONE HUNDRED-ELEMENT IDEALIZATION OF THIN-WALL CYLINDER

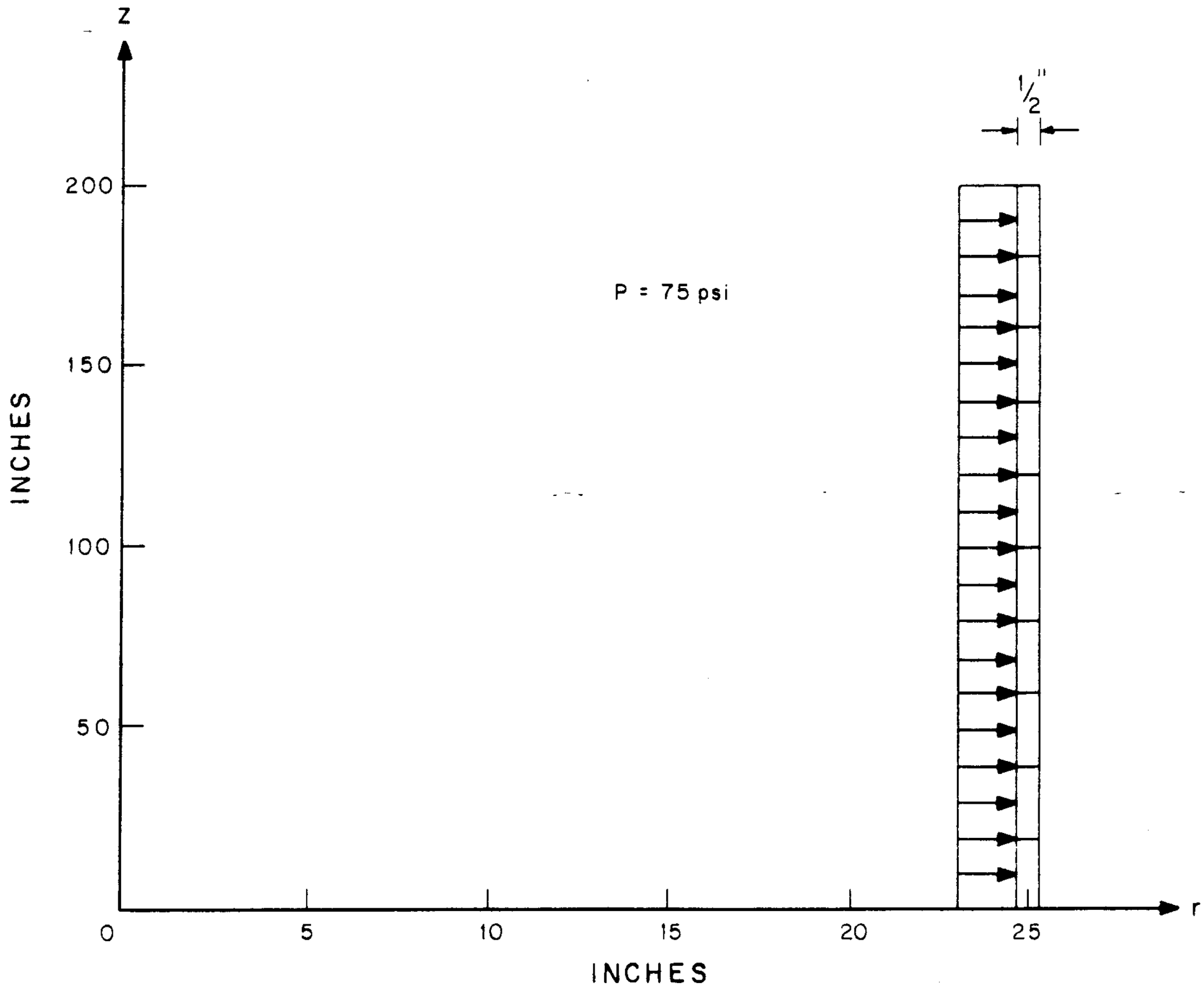

FIGURE 3A.1.4–1 ELEMENT PLOT

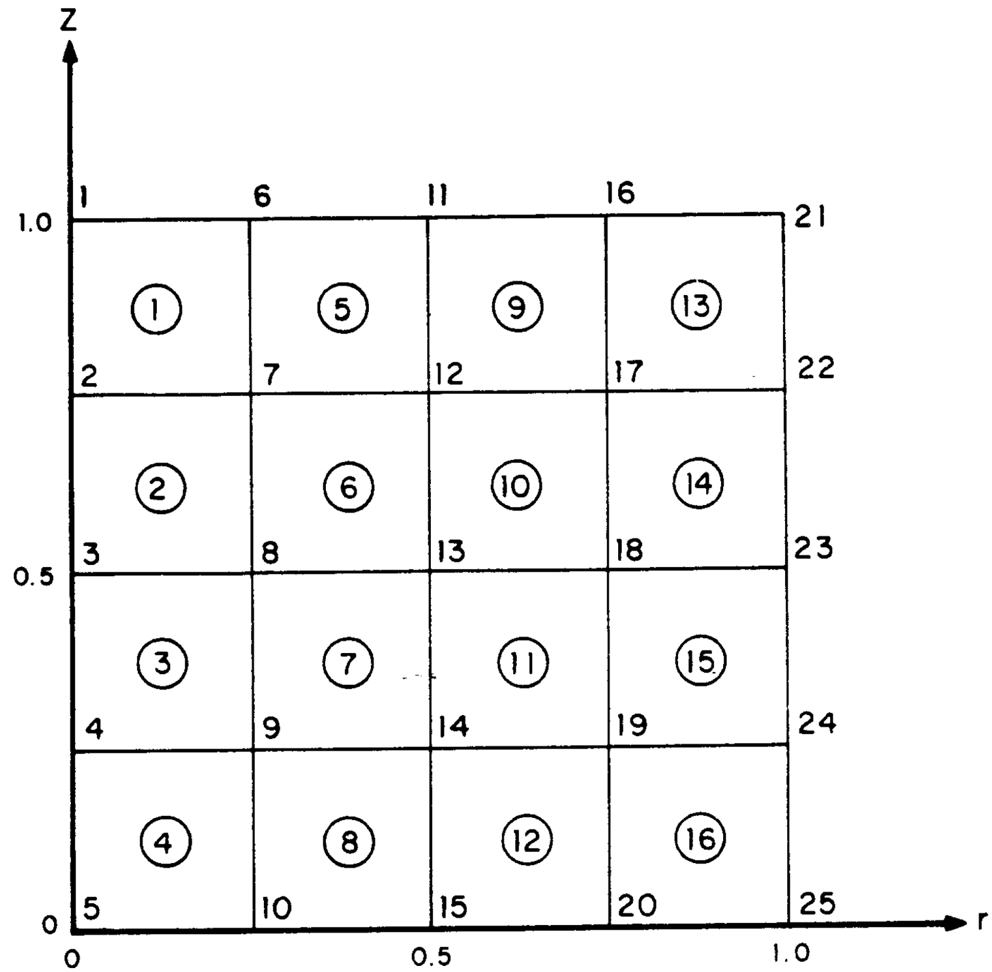

FIGURE 3A.1.4–2 HARMONIC AXISYMMETRIC PLAIN STRAIN

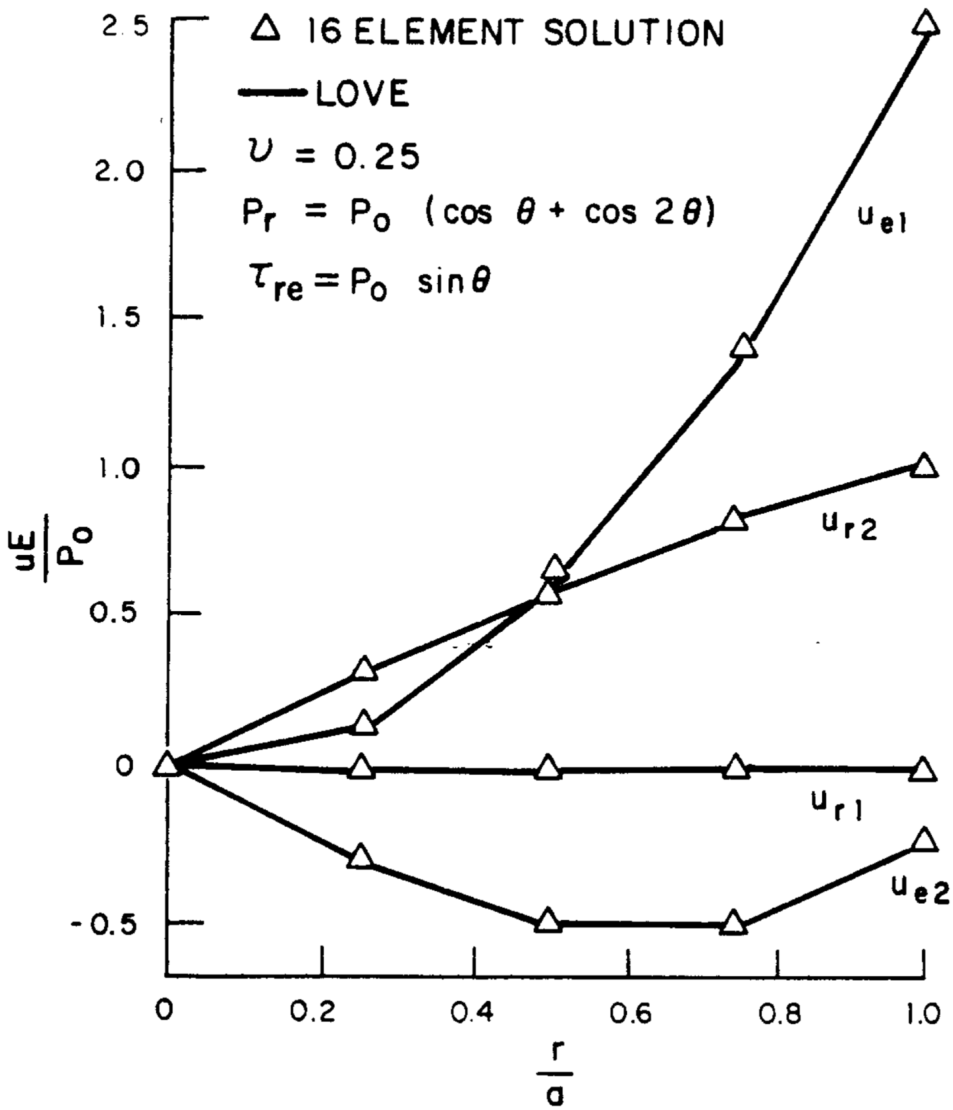

FIGURE 3A.1.5–1 TAC2D SAMPLE PROBLEM THERMAL MODEL

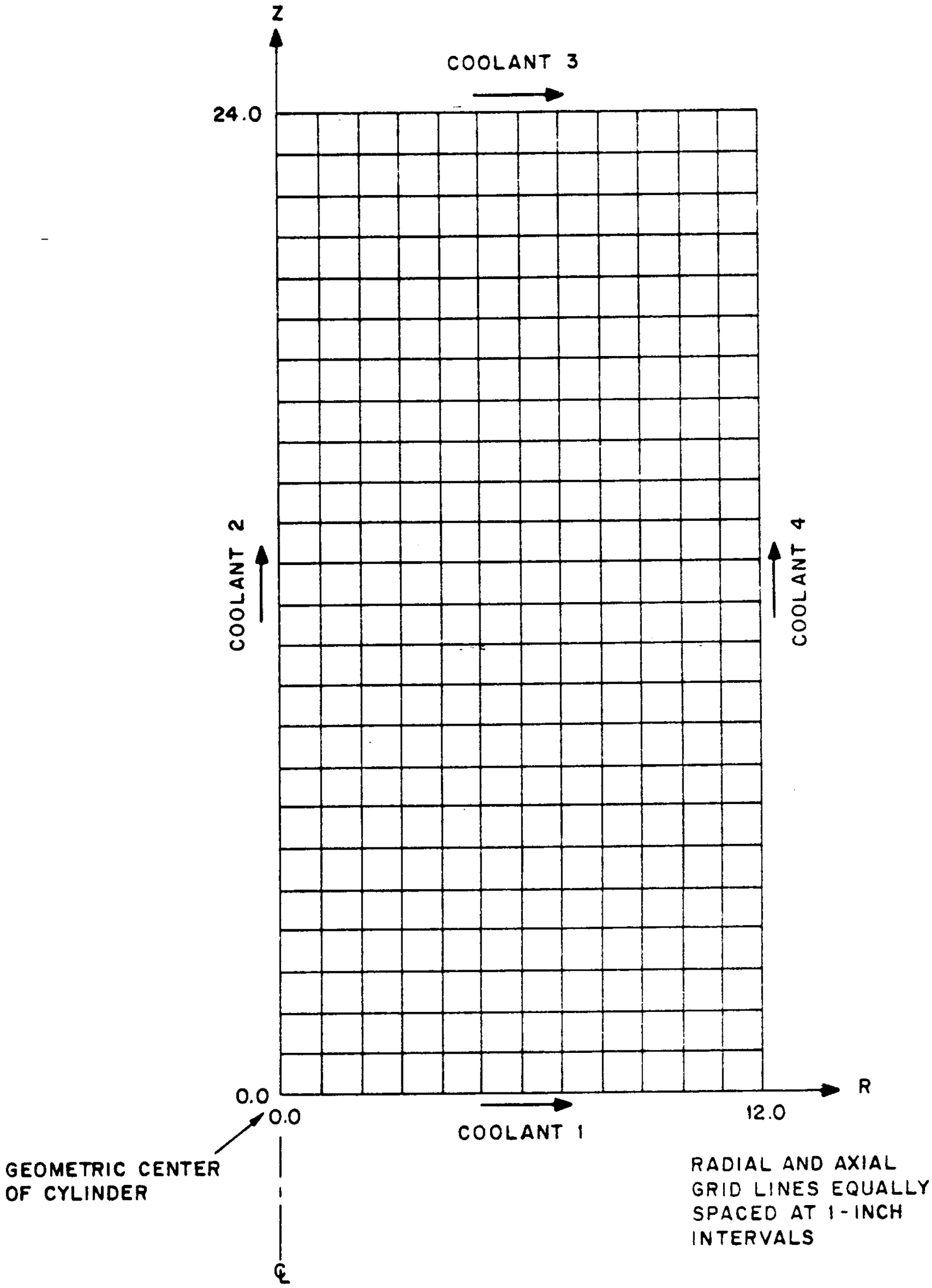

FIGURE 3A.1.5–2 TRANSIENT TEMPERATURES IN A RIGHT CIRCULAR CYLINDER-COMPARISON OF TAC2D RESULTS WITH SERIES SOLUTION

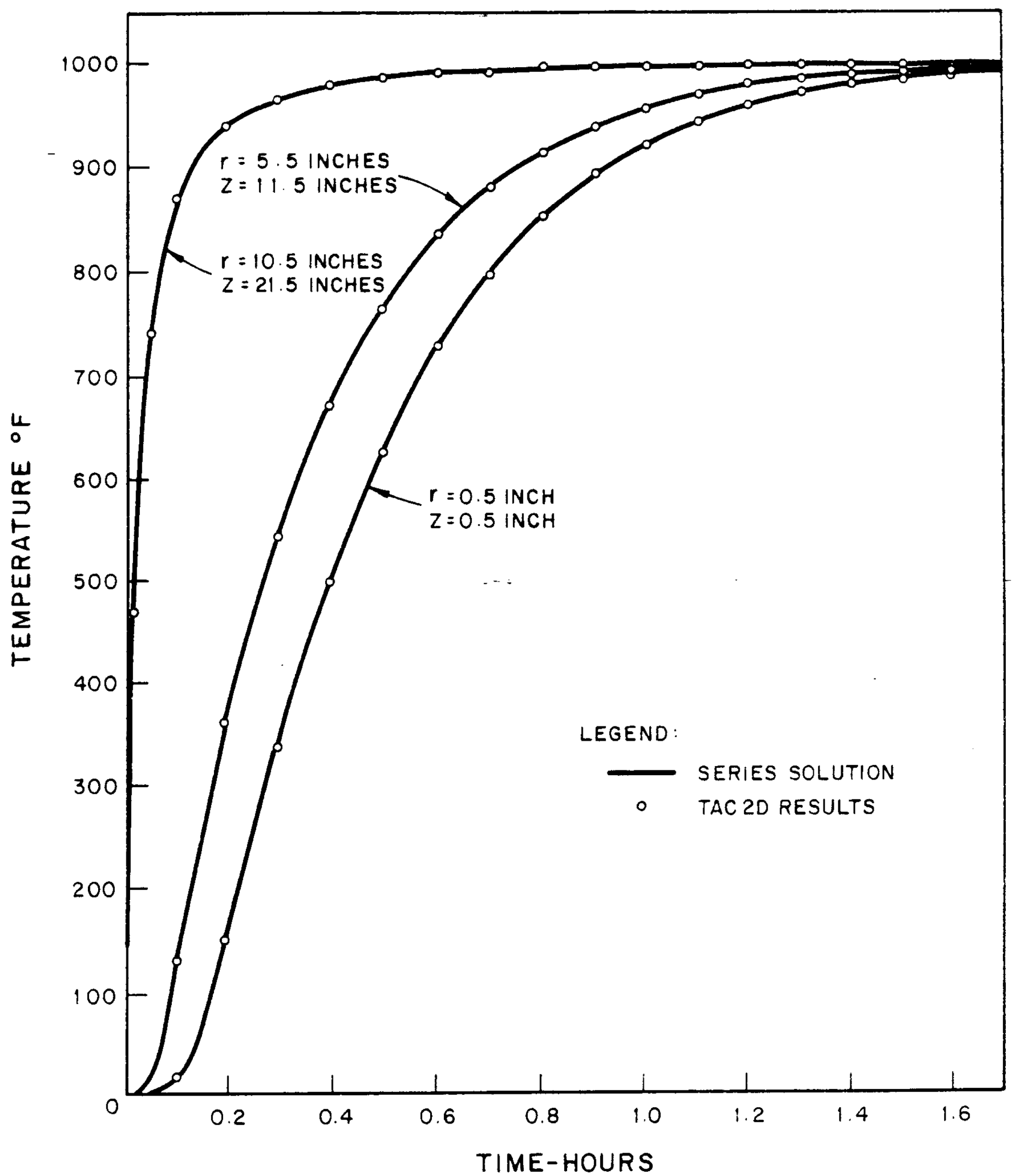

FIGURE 3A.1.6–1 STRUCTURAL MODEL

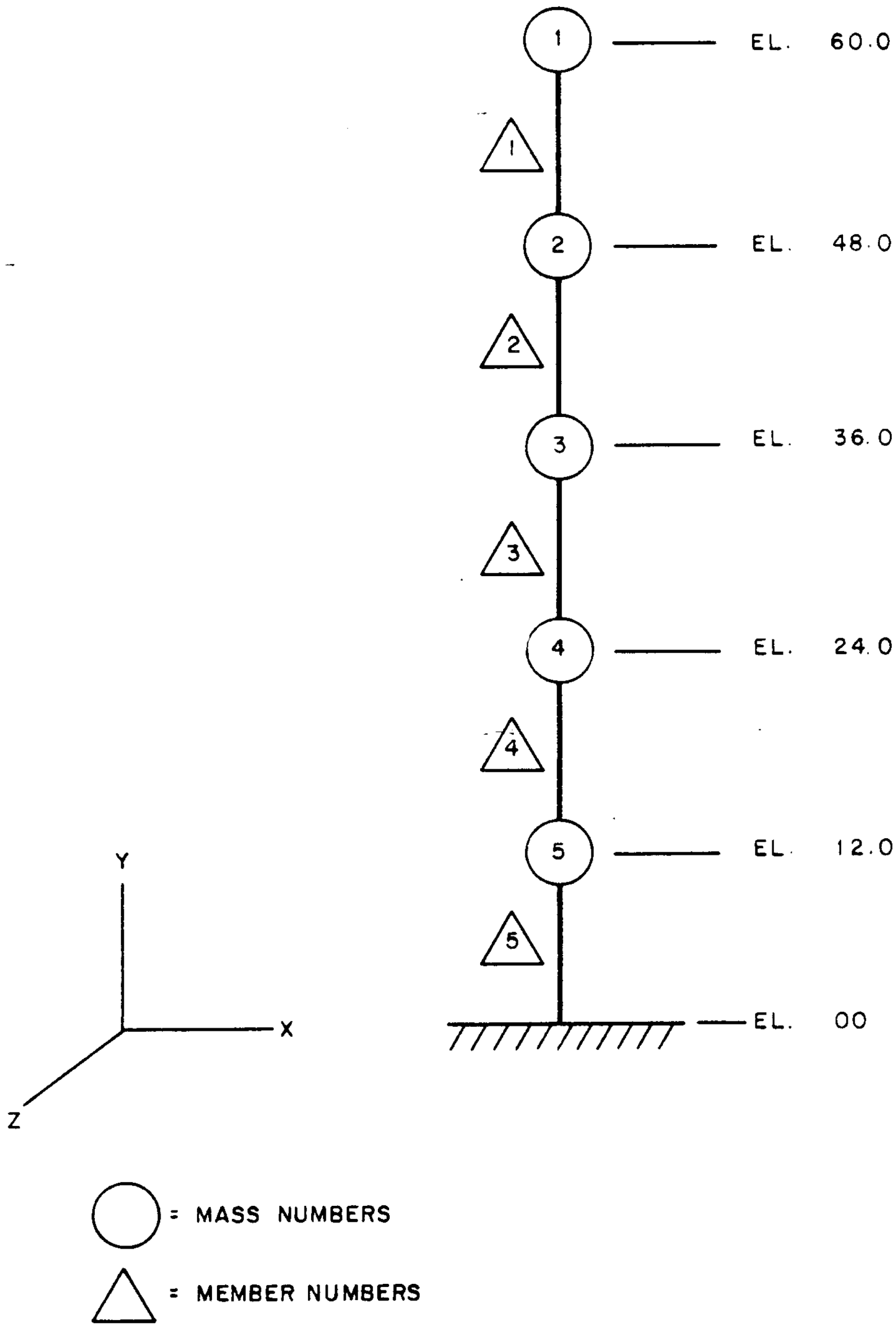

FIGURE 3A.1.7–1 FINITE ELEMENT GRID FOR PLAXLY-FLUSH COMPARISON

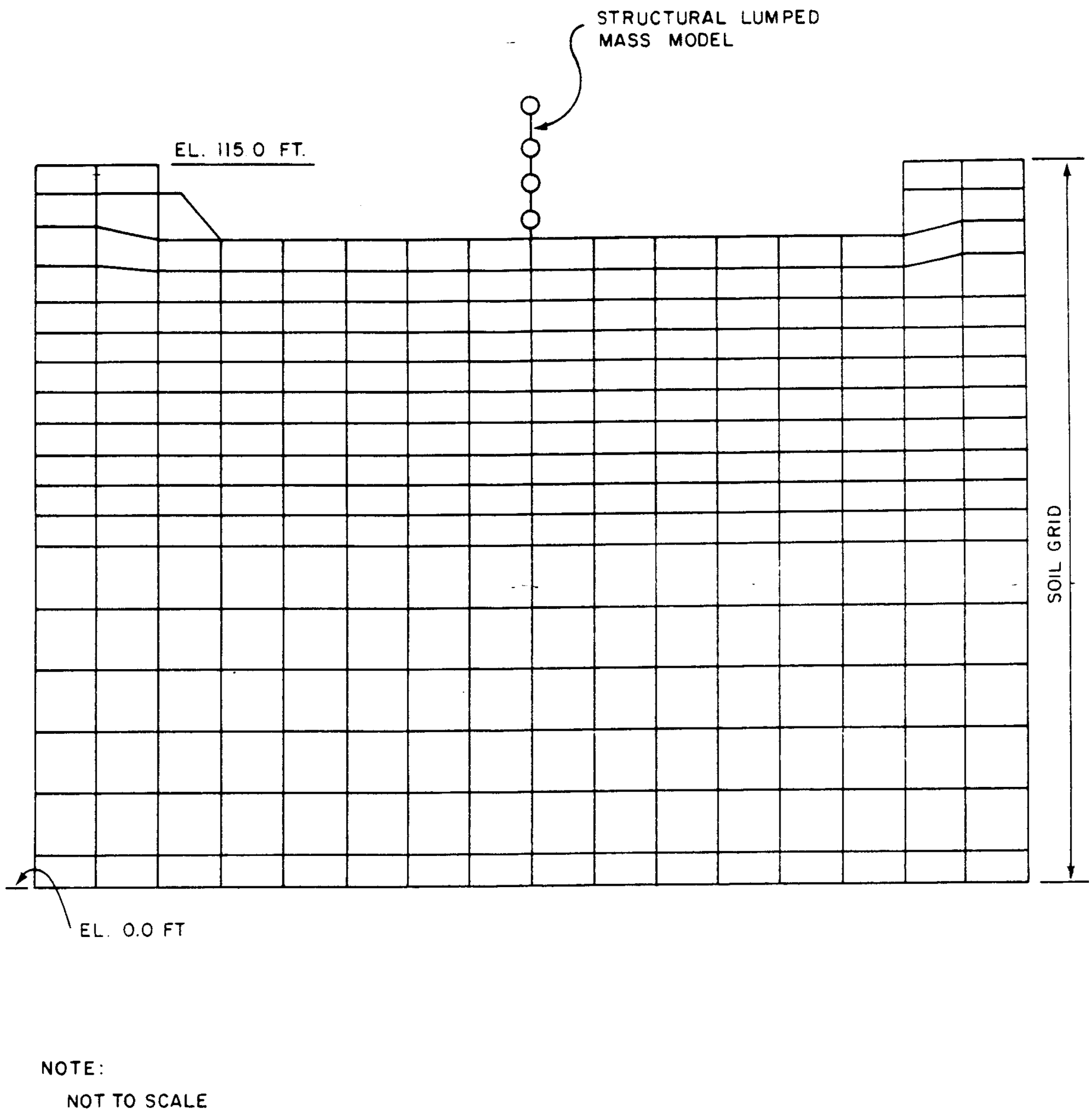

FIGURE 3A.1.7–2 HORIZONTAL AMPLIFIED RESPONSE SPECTRA AT TOP MASS IN BEAM STRUCTURE

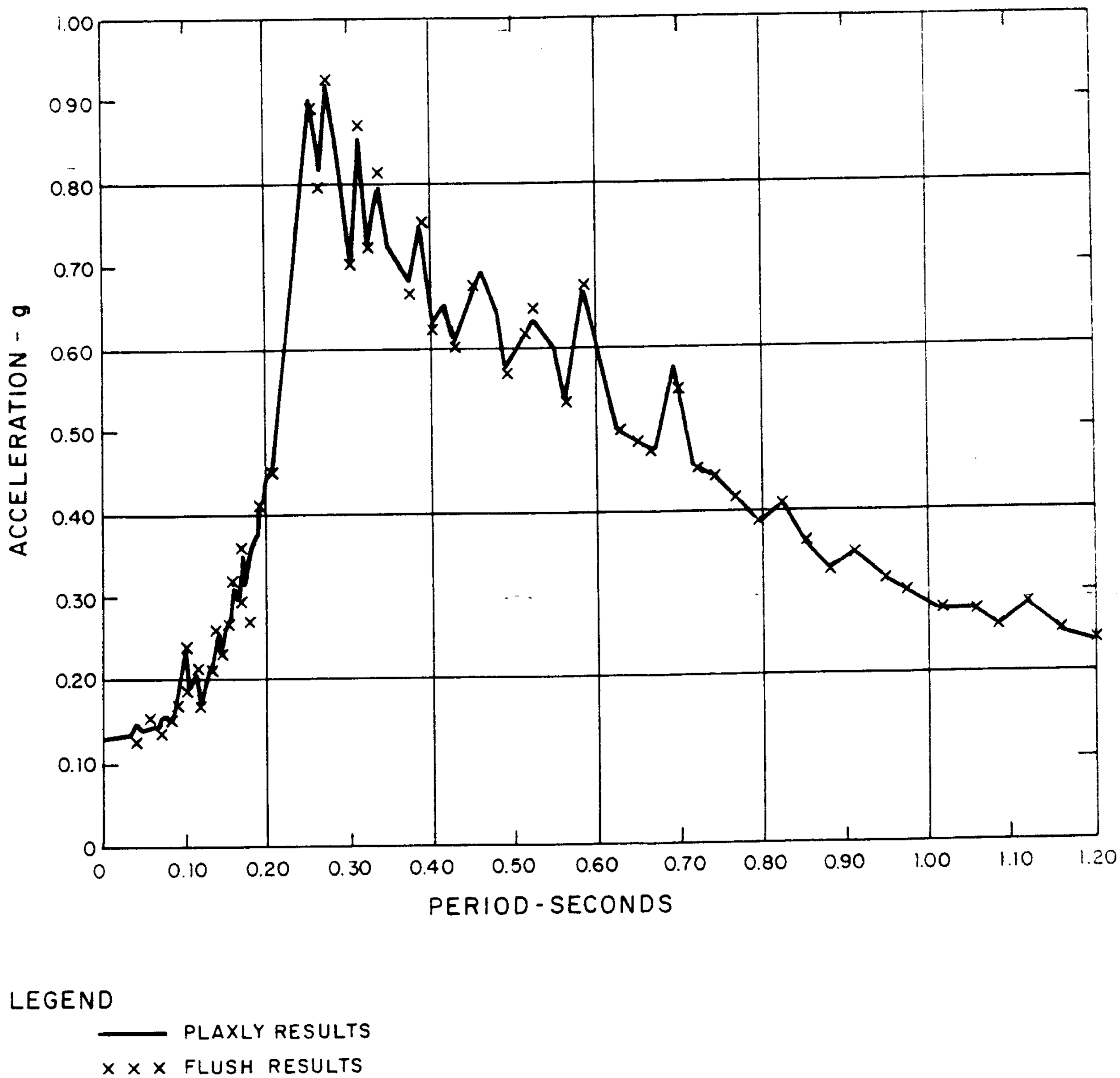

TABLE 3A.1.10–1 PERTINENT PARAMETERS OF A CYLINDRICAL SHELL

Input Data	Parameter
Radius of outer layer of rebars	r_o = 66.5 feet
Radius of inner layer of rebars	r_i = 63.67 feet
Radius of liner	r_1 = 63 feet
Radius of reference surface	r = 65.25 feet
Height of cylinder	h = 122 feet
Meridional steel area/unit length, outer layer	A_o = 4 in^2
Meridional steel area/unit length, inner layer	A_i = 4 in^2
Liner area/unit length	A_l = 4.5 in^2
Circumferential steel area, outer layer	A_o = 8 in^2
Circumferential steel area, inner layer	A_i = 8 in^2
Internal pressure	p = 9.72 ksi
Wall weight per unit surface	q = 0.68 ksi
Total load at top	w = 9726.5 k
Temperature increment, outer rebars	ΔT_o = -12°F
Temperature increment, inner rebars	ΔT_i = 27°F
Temperature increment, liner	ΔT_l = 230°F

TABLE 3A.1.10–2 SUMMARY OF RESULTS

Input Data	Hand Calculation	Computer Run	% Difference
Liner strain:			
Meridional	0.001308	0.001308	0
Circumferential	0.001412	0.001412	0
Membrane stresses: (ksi)			
Meridional rebars:			
Outer layer	41.58	41.59	0.02
Inner layer	33.975	33.99	0.02
Liner	-6.9857	-6.97	0.2
Hoop rebars:			
Outer layer	42.47	42.47	0
Inner layer	36.65	35.65	0
Liner	-4.586	-4.586	0
Membrane Forces: (k/ft)			
Meridional	271.76	271.90	0.05
Circumferential	612.32	612.36	0

TABLE 3A.1.11–1 TEST PROBLEM DATA

Load Type	Load
1. Static load	-15.5 k*
2. Suddenly applied constant load (remains on structure permanently)	62.9 k*
3. Missile Impact - finite force	264 k*
specific momentum	1.4 k*/sec
4. Suddenly applied dynamic load w/zero time duration (applied impulse)	1.2 k*/sec

NOTES:

* Equivalent barrier weight = 16.66 k

Barrier resistance function - From zero displacement to a displacement of 0.003 foot, the barrier resistance increases linearly from 0 to 87.2 k. For displacements greater than 0.0003 foot, the resistance remains at a constant value of 87.2 k.

TABLE 3A.1.11–2 A COMPARISON OF HAND AND COMPUTER PROGRAM RESULTS

Load Number	Load Case	Results from Hand Calculation	Results from Computer Run
1	Barrier deflection	-0.000533 feet	-0.0005 feet
2	Barrier deflection	0.0054 feet	0.0054 feet
	Time of maximum deflection	0.01802 sec	0.018378 sec
3	Barrier deflection	0.0183 feet	0.0187 feet
	Time of maximum deflection	0.0180 sec	0.018391 sec
4	Barrier deflection	0.0171 feet	0.0172 feet
	Time of maximum deflection	0.01481 sec	0.014419 sec

3A.2 EQUIPMENT AND COMPONENTS

The following computer programs were used for the analysis of Seismic Category I equipment and components, as well as for pipe rupture design and analysis.

1. ASAAS - Asymmetric Stress Analysis of Axisymmetric Solids
2. LIMITA 25 - Nonlinear Static Analysis of Plane Frames
3. MISSILE - Turbine Missile Probability Program
4. SLOSH - Simplified Tank Sloshing Analysis
5. LION - Temperature Distribution for Arbitrary Shapes/Complicated Boundary Conditions
6. LIMITA 3 - Nonlinear Dynamic Analysis of Frame Structures
7. TAC2D - Two Dimensional Thermal Analysis
8. SHELL 1 - Thin Shell of Revolution Under Arbitrary Loading
9. NOZZLE (ST-147) - Vessel Penetration Analysis
10. DINASAW - Dynamic Inelastic Nonlinear Analysis by Stone & Webster Engineering Corporation
11. LIMITA 2 - 2-D - Nonlinear Transient Dynamic Analysis
12. STARDYNE - Linear and Nonlinear Elastic Structure Analysis
13. ASYMPR - Asymmetric Pressure Force - Time History
14. LIDOP - Local Inelastic Deformation of Piping
15. DLF - Dynamic Load Factors
16. STRUDL - Structural Analysis Program. STRUDL program descriptions and verification are presented in Sections 3A.3.8 and 3A.3.9 and are not duplicated here.

3A.2.1 ASAAS (Asymmetric Stress Analysis of Axisymmetric Solids)

3A.2.1.1 General Description

This is a finite element computer code (Cross 1971). It can be used to determine stresses and displacements in arbitrary axisymmetric solids, including problems involving asymmetric mechanical and thermal loads and asymmetric temperature-dependent mechanical properties. All dependent variables, including the mechanical properties, are input by Fourier series expansions of the circumferential coordinate. The mechanical loads can be surface pressures, surface shears, and nodal point forces.

The explicit stiffness relations for the axisymmetric solid ring elements of the triangular cross section are based on the classical theorem of potential energy and the assumption that, within any element, the displacement variation in the R-Z plane is linear. All dependent variables, including the material properties, are expanded into the Fourier series. The harmonics are coupled and all the equilibrium equations are solved simultaneously. The algorithm used to solve the equations is a block modified square root Cholesky method with iterative refinement (Cross 1971).

3A.2.1.2 Program Verification - Harmonic Assymetric - Plane Strain

An infinitely long, solid, circular cylinder is subjected to cos and cos 2 pressure distributions. A closed-form solution of this problem may be obtained (Love 1944).

The pertinent parameters of the cylinder are presented in Table 3A.2.1–1.

The following solution can be verified (Love 1944):

$$\sigma_\gamma = P_o(r\cos\theta + \cos 2\theta) \qquad (3A.2.1\text{-}1)$$

$$\sigma_\theta = P_o\left(3r\cos\theta + \frac{2r^2 - \partial^2}{\partial^2}\cos 2\theta\right) \qquad (3A.2.1\text{-}2)$$

$$\sigma_{r\theta} = P_o\left[r\sin\theta = \frac{r^2 - \partial^2}{r^2}\sin 2\theta\right] \qquad (3A.2.1\text{-}3)$$

$$U_r = P_o\left[\frac{(1-4Y)(1+Y)r^2}{2E\partial}\sin\theta + \frac{1+Y}{E}\left(r - \frac{2Yr^3}{3\partial^2}\right)\cos\theta\right] \qquad (3A.2.1\text{-}4)$$

$$U = Po\left[\frac{(5-4Y)(1+Y)r^2}{2E\partial}\sin\theta + \frac{1+Y}{E}\left[\left(1-\frac{2}{3}\right)\frac{r^3}{\partial^2} - Y\right]\sin 2\theta\right] \qquad (3A.2.1\text{-}5)$$

The cylinder is idealized by 16 elements, as shown on Figure 3A.1.4–1. Computer results are depicted on Figure 3A.1.4–2, along with the exact results obtained from Equations 3A.2.1-4 and

3A.2.1-5. As can be seen, the computer results are very close to the exact results. Therefore, this problem verifies the accuracy of ASAAS for mechanical loading problems where material properties are not variable.

3A.2.2 Limita 25 - 2D Nonlinear Transient Dynamic Analysis

3A.2.2.1 General Description

Limita 25 (ST-224) is a computer code written and fully documented by SWEC, which predicts two-dimensional structures. A plane frame is represented mathematically as a discrete system of beam members. Under loadings, the equilibrium at each joint is ensured by the system equilibrium equation (Martin 1966):

$$[K]\ \{q\} = \{F\} \qquad (3A.2.2\text{-}1)$$

where:

[K] = System stiffness matrix

{q} = Global displacement vector

{F} = External force vector

An element of the stiffness matrix, Kij, situated in row i and column j, is the force in the i th degree of freedom required to produce a unit displacement in the j th degree of freedom when all other degrees of freedom are restrained from moving (Martin 1966; Przemieniecki 1968).

To account for nonlinear effects, such as plasticity and large deflections, Equation 3A.2.2-1 is solved by an incremental method at any particular load step, Equation (1) may be written:

$$[K]_i\ \{\Delta q\}_i = \{\Delta F\}_i \qquad (3A.2.2\text{-}2)$$

where:

$\{\Delta q\}_i = \{q\}_i - \{q\}_{i=l},\ \{\Delta F\}_i = \{F\}_i - \{F\}_{i-l}$

$\{\Delta q\}_0 = \{q\}_0\ ,\ \{\Delta F\}_0 = \{F\}_0$

The stiffness matrix-[K], calculated based on the deformed structure at load step i (Martin, AFFDL IR66-80, 1966), is assumed constant through the load step. Displacements and member forces are given at load step i by:

$$\{q\} = \sum_{\alpha = 0}^{i} \Delta q\} \qquad (3A.2.2\text{-}3)$$

$$\{Q\} = \sum_{\alpha = 0}^{i} [K]_{\alpha}\{\Delta\bar{q}\}_{\alpha}$$

where:

$\{Q\}$ = member force vector

[k] = member stiffness matrix

{q} = member displacement vector

The equilibrium equations, Equation (2), are solved by a standard elimination technique.

Since no external loading is applied to a member between joints, the maximum value of the internal force acting on a member occurs at its ends. The transition from the elastic to the fully plastic state is disregarded, and the end sections are assumed to remain linearly elastic until a fully plastic state is reached. The yield surface is defined by scalar function, of the internal member forces, {Q}, having the form (Hodge 1959; Neal 1961; Stockey et al., 1966):

$$\phi (\{Q\}_i) = 1$$

is obtained by integrating stress across the member section with the stress fully developed over the section and satisfying the von Mises (or Tresca) yield criterion:

$$\sigma^2 + \alpha^2 T^2 = \sigma_y{}^2$$

where:

σ = Normal stress

T = Shear stress

σ_y = Yield stress in simple tension

α^2 = 3 (von Mises) or 4 (Tresca)

Thus the function ϕ depends on the shape of the cross section and the force components being considered. For a plane frame, the yielding normally occurs due to either a predominant bending moment or a predominant axial force. Therefore, two plastic models are used.

Bending Yield Model:

Since a section is either elastic or fully plastic, there are four possible states:

1. Both ends A and B are elastic

2. End A is plastic; end B is elastic

3. End A is elastic; end B is plastic

4. Both ends A and B are plastic

A plastic hinge is introduced at any end section which is yielding. The force-displacement relation of the plastic hinge follows an ideal bilinear curve (Clough 1965; Guberson 1967). In situations where force reversal occurs, the elastic stiffness of the hinged member is restored, providing elastic unloading (isotropic strain hardening model).

<u>Axial Yield Model:</u>

There are only two possible states:

1. The entire member is elastic

2. The entire member is plastic

When the member yields, the member elastic Young's modulus is replaced by a plastic tangent modulus and the force-displacement relation follows a bilinear curve. If the member unloads, the elastic modulus is restored.

3A.2.2.2 Program Verification

A center loaded beam, built in at one end and supported at the other end, is analyzed for comparison to data obtained analytically. The analytical solution was obtained using limit analysis as explained in Hodge (1959). The displacement at the point of loading calculated using Limita 25 and calculated analytically preshown in Figure 3A.2.2-1. Additional problems where analyzed to ensure that all program options were exercised and thus demonstrated the function and adequacy of the program.

3A.2.3 MISSILE

3A.2.3.1 General Description

The Missile program calculates the impact probability (P_2) of postulated turbine missiles on specified targets. The solid angle method is used to calculate P_2:

$$P_2 = \frac{1}{\Omega_m} \cdot \int d\Omega$$

where:

Ω = Solid angle subtended by the target

Ω_m = Total solid angle subtended by all possible missile trajectories

The integral is evaluated by numerical integration, with consideration of the missile ejection velocity and the relative positions of the turbine and target (Figure 3A.2.3–1).

3A.2.3.2 High Trajectory Verification

Westinghouse has derived a formula to predict the probability of impact for high trajectory missiles. Some adjustments to the formula are necessary to enable direct comparison with the program results. The formula has been derived on the basis that the initial velocity is random and uniformly distributed between V_1 and V_2. The program uses a deterministic initial velocity. The formula may be specialized to this condition by setting V_1 equal to V_2 after applying L'Hopital's Rule. Also, the formula has been derived assuming a missile fragment occurs in the quadrant of the target, whereas the program assumes a missile fragment can occur in any of the four quadrants. These differing assumptions can be reconciled by using four fragments for program input.

After making the above adjustments, the high trajectory formula becomes:

$$P = G^2/(2\pi\Delta V^4)$$

where:

P = Impact probability per square foot of target

G = Acceleration of gravity (ft/sec^2)

Δ = Deflection angle range (radians)

Missile (MA-057) is a computer code written and fully documented by SWEC for inhouse use.

3A.2.4 SLOSH

3A.2.4.1 General Description

The purpose of this program is to compute the seismically-induced liquid pressures and the maximum vertical displacement of the liquid surface in a container under horizontal acceleration. The mathematical procedures and formulas used in developing the program were taken from AEC Report TID-7024. The program uses data for intensity of ground motion taken from average-acceleration-spectrum curves, as used in the analysis from the report.

The program is used for circular or rectangular, shallow or slender, ground-supported tanks and circular or rectangular, shallow (not slender) tower-supported tanks.

3A.2.4.2 Program Verification

A comparison of results of computer program SLOSH and those given in the AEC Report (US AEC 1963) shows that the program yields correct results (Table 3A.2.4–1).

SLOSH (ME-111) is a computer code written and fully documented by SWEC for inhouse use.

3A.2.5 LION (ME-112)

LION is a digital computer program which is used to solve three-dimensional transient and steady state temperature distribution problems. The program may also consider subcooled nucleate boiling and coolant heat transfer effects. The surface conditions may be forced convection, free convection, or radiation and heat may be externally or internally generated. Input to the program consists of structural geometry, physical properties, boundary conditions, internal heat generation rates, coolant flow properties, and flow rates.

The program solves the transient heat conduction equations for a three-dimensional field using a first forward difference method. To ensure the temperature calculation stability, LION can determine the suitable time increment, if the specified input time increment is too large.

Since the original program (Bray 1954) was developed, subsequent versions have evolved to solve larger and more complex problems (Bray 1954, 1959; Personal Communication; Briggs 1963; Lechliter 1963).

LION is a recognized program in the public domain and has been used extensively.

3A.2.6 LIMITA 3

3A.2.6.1 General Description

LIMITA 3 (ST-225) is a computer code written and fully documented by SWEC for inhouse use. Its formulation is identical to that of Limita 2 (Paragraph 3A.2.11) in that the equations are applicable to a three-dimensional problem. For a space frame, yielding normally occurs due to either a predominant bending moment or a predominant torsion. Therefore, two plastic models are provided.

1. Bending Yield Model

   Since a beam section is either elastic or fully plastic, there are four possible states:

   a. Both ends A and B are elastic

   b. End A is plastic; end B is elastic

   c. End A is elastic; end B is plastic

d. Both ends A and B are plastic.

A plastic hinge is introduced at any end section which is yielding. The force-displacement relation of the plastic hinge follows an ideal bilinear curve (Clough et al., 1965; Giberson 1967). In situations where force reversal occurs, the elastic stiffness of the hinged member is restored, providing elastic unloading (isotropic strain hardening model).

2. Torsional Yield Model

There are only two possible states:

a. The entire member is elastic

b. The entire member is plastic

When the member yields, the member elastic modulus is replaced by a plastic tangent modulus and the force-displacement relation follows a bilinear curve. If the member unloads, the elastic modulus is restored.

3A.2.6.2 Program Verification

3A.2.6.2.1 Elastic Example

As a checkout of LIMITA 3, a space frame is considered. All members are W14x500. A step load of 30 kips is applied vertically at joint 6. This problem was analyzed by LIMITA 3 and ICES STRUDL II elastically. The results of displacements and moment Z at joint 6 were plotted against each other and found to be in excellent agreement.

3A.2.6.2.2 Plastic Example

This example is provided to illustrate this ability of the program to determine the inelastic transient response of a three-dimensional structure. The structures considered consist of cantilevered steel tubes subjected to force transients causing bending and torsion in the structures. The results obtained from an analysis using the LIMITA 3 code are compared with data obtained experimentally.

The experiment was a drop test in which the cantilever tubes were loaded by weights at each tube end. The results tabulated were the peak deflections and their corresponding times and the permanent deflections. These results are compared to those obtained using the LIMITA 3 code in Table 3A.2.6–1.

Additional problems were analyzed (elastic and inelastic) to ensure that all program options were exercised and thus demonstrate the functions and adequacy of the program.

3A.2.7 TAC2D (A General Purpose, Two-Dimensional Heat Transfer Computer Code)

3A.2.7.1 General Description

Refer to Appendix 3A.1, Section 3A.1.5.1.

3A.2.7.2 Program Verification

Refer to Appendix 3A.1, Section 3A.1.5.2.

3A.2.7.2.1 Analytical Solution

Refer to Appendix 3A.1, Section 3A.1.5.2.1.

3A.2.7.2.2 Numerical Solution with TAC2D

Refer to Appendix 3A.1, Section 3A.1.5.2.2.

3A.2.7.2.3 Comparison of TAC2D Solution with the Analytical Solution

Refer to Appendix 3A.1, Section 3A.1.5.2.3.

3A.2.7.2.4 References for TAC2D

Refer to Appendix 3A.1, Section 3A.1.12.

3A.2.8 SHELL 1

3A.2.8.1 General Description

Refer to Appendix 3A.1, Section 3A.1.2.1.

3A.2.8.2 Program Verification - Thin-Wall Cylinder

Refer to Appendix 3A.1, Section 3A.1.2.2.

3A.2.8.3 References for SHELL 1

Refer to Appendix 3A.1, Section 3A.1.12.

3A.2.9 Vessel Penetration Analysis

3A.2.9.1 General Description

The vessel penetration analysis computer code (ST-147) is written and fully documented by SWEC for inhouse use. The code performs various analyses on tanks and pressure vessels. All of the analyses are concerned with local stresses at penetrations. Typical problems which can be handled include the following:

1. Applied load stresses at vessel-nozzle junction for:
   a. Rigid attachment to cylinder
   b. Rigid attachment to sphere
   c. Hollow attachment to sphere
2. Pressure discontinuity analysis for thin shell interaction
3. Allowable load functions on nozzles for each case

Local stresses due to nozzle loads are found by the method prescribed by P.P. Bijlaard (Wichman et al., 1965). The method prescribed by Johns and Orange (Johns and Orange 1961) is used for pressure discontinuity stresses.

3A.2.9.2 Program Verification

A sample problem of a thin-walled cylindrical vessel is subjected to applied loads from a rigid cylindrical attachment. This problem may be solved using Johns and Orange's method (Johns and Orange 1961).

A summary of manual calculations were then compared with the computer summary. Additional problems were considered to ensure that all program options were exercised and thus demonstrate the function and adequacy of the program.

3A.2.10 DINASAW (Dynamic Inelastic Nonlinear Analysis by Stone and Webster)

3A.2.10.1 General Design

DINASAW may be used to predict the nonlinear, dynamic behavior of plane frames (pipes, rings, or beams) including large displacements, plasticity, and impacts. Arbitrary force-time relations may be applied at any station. DINASAW can also be applied to pipe whip and pipe impact problems.

The analysis, as derived (Wu and Witmer 1972; Collins and Witmer 1973), employs the spatial finite-element method in which the tangential and normal displacement fields are represented by cubic interpolations. By applying the principle of virtual work in conjunction with D'Alembert's principle, the equations of motion may be derived in the form:

$$[M] \cdot \{\ddot{q}\} = \{F\} - \{P\} - [H]\{q\}$$

where:

$\{q\}$ and $\{\ddot{q}\}$ = the generalized displacements and generalized accelerations, respectively, for the complete assembled discretized structure defined, with respect to a global coordinate system

[M] = the lumped mass matrix for the complete assembled discretized structure

{F} = the assembled vector of externally-applied loading

{P} = an assembled internal force matrix (replaces conventional stiffness matrix)

[H] {q} = generalized loads arising from both large deflection and plastic behavior

3A.2.10.2 Program Verification

Two examples are discussed here. The first (Wu and Witmer 1972), involves a ring subjected to radial blast wave over a portion of its circumference. The resulting deformation severely distorts the ring, flattening it considerably. Still, the computer code follows very closely not only to the displacement field, but also to the strain time history.

The second case (Collins and Witmer 1973) involves the impact of a rotor segment onto a ring or shroud. Again, the program, in conjunction with the CIVM (Collision Imparted Velocity Method), follows experimental results very closely.

3A.2.11 LIMITA 2

3A.2.11.1 General Description

LIMITA 2 (ST-223) predicts nonlinear, dynamic behavior of plane frames, including large displacements, plasticity, and impact. A plane frame is simulated as a lumped parameter system, consisting of an assembly of discrete lumped masses connected by beam members. Under any loading, the equilibrium at the rth mass point is ensured by the equation of motion:

$$m_r\ddot{q}_r + \sum^{i} C_{ri}q_i + \sum^{i} K_{ri}q_i = \mathrm{f_T} \qquad (3A.2.11\text{-}1)$$

where:

$\sum$ = a series with one term for each of the i displacements

C_{ri} = the damping coefficient, which applies to the ith velocity in the rth equation of motion

K_{ri} = the member stiffness, which is defined as the force necessary to hold the structural member from moving the rth degree of freedom when the ith degree of freedom is given a unit displacement when all other degrees of freedom are restrained from moving (Martin, 1966; Przemieniecki 1968).

f_T = the external load factor

To take account of nonlinear effects, such as plasticity and large deflections, Equation 3A.2.2-1 is solved by an incremental method (Clough and Wilson 1962). At any particular time, t, the displacement increment is obtained from

$$m_r\ddot{q}_r^t + \sum C_{ri}^t q_i^t + \sum^{i} K_{ri}^t \Delta q_i^t = f_r^t - \sum_{t=i}^{S}\left(\sum^{i} K_{ri}^S \Delta q_i^s\right) \qquad (3A.2.11\text{-}2)$$

where:

C^t_{ri} = current damping coefficient

K^t_{ri} = the member stiffness

which are calculated based on the current deformed structure (Martin, 1966) and assumed constant through the time step, Δt.

The displacement and member forces are thus given by:

$$q_r^t = \sum_{t}^{S} \Delta q_r^s \qquad Q_r = \sum_{t}^{S}\left(\sum^{i} K_i^S \Delta q_i^S\right) \qquad (3A.2.11\text{-}3)$$

The second order differential system equations (Equation 3A.1.11-2) are solved by a linear acceleration implicity method (Hildebrand 1956).

Since no external loading is applied to a member between nodes, the maximum value of the internal force acting on a member occurs at its end sections. The transition from the elastic to the fully plastic state is disregarded and the end sections are assumed to remain linearly elastic up to the full plastic yield surface. The yield surface is defined by a scalar function of the internal member forces, Q, of the form (Hodge 1959; Neal 1961; Stokey et al., 1966).

$\Phi\ (Q) = 1$

Here the function is obtained by integrating the stress across the section with the stress fully developed over the section and satisfying the von Mises (or Tresca) yield criterion.

$$\sigma^2 + Y^2T^2 = \sigma_y^2$$

where:

σ = normal stress

T = shear stress

σ_y = yield stress in simple tension

Y^2 = 3 (von Mises) or 4 (Tresca)

Thus, the function depends on the shape of the cross section and the force components being considered.

For a frame structure, the yielding normally occurs due to either a predominant bending moment or to a predominant tension or compression. Thus, two plastic models are provided:

1. Bending predominant members

   Since a section is either elastic or fully plastic, there are four possible states:

   a. Both ends A and B are elastic

   b. End A is yielding and B is elastic

   c. End A is elastic and B is yielding

   d. Both ends A and B are yielding

A plastic hinge is introduced at any end section which is yielding. The force-displacement relation of the plastic hinge follows an ideal bilinear curve (Clough et al., 1965; Giberson 1967). In situations where force reversal occurs, the stiffness of the hinged member is restored, providing unloading along the elastic line (isotropic strain hardening model).

2. Tension or compression of predominant members

   There are only two possible states:

   a. The entire member is elastic

   b. The entire member is plastic

When the member yields, the member is elastic but Young's Modulus is replaced by a plastic tangent modulus and the force-displacement curve follows a bilinear curve. If the member unloads, the elastic modulus is restored.

3A.2.11.2 Program Verification

SWEC sponsored an experimental investigation performed by the Massachusetts Institute of Technology (Wilson 1968). The problem consisted of the cantilevered pipe (Figure 3A.2.11–1) subjected to an impulsive load at its free end. The impulse is imparted by the detonation of a sheet of high explosive, separated from the pipe by a buffer material. A nearly uniform initial velocity is produced in the loaded region and is determined by high speed photography.

This problem was analyzed by LIMITA 2. The results were compared with experimental data and output from another computer program, DINASAW.

Additional problems were considered to ensure that all program options were exercised and thus demonstrate the function and adequacy of the program.

3A.2.12 STARDYNE

The STARDYNE Structural Analysis System, written by Mechanics Research, Inc., of Los Angeles, California, is a fully warranted and documented computer program available at Control Data Corporation. The latest version of this program became available August 1, 1973.

The MRI STARDYNE Analysis System consists of a series of compatible digital computer programs designed to analyze linear and nonlinear elastic structural models. The system encompasses the full range of static and dynamic analyses.

The static capability includes the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal, modal applied loads, and prescribed displacements.

Utilizing the normal mode technique, linear dynamic response analyses can be performed for a wide range of loading conditions, including transient, steady state harmonic, random, and shock spectra excitation types. Dynamic response results can be presented as structural deformations and internal member loads.

The nonlinear dynamic analysis program is integrated in the rest of the STARDYNE system. The equations of motion for the linear portion of the structural model are generated and modified to account for the nonlinear springs. The resulting nonlinear equations of motion are directly integrated using either the Newmark or Wilson implicit integration operators. The user may enter sets of structural loadings which vary with time, and specify time points at which the program is to output the structural response.

3A.2.13 ASYMPR (ME-171)

3A.2.13.1 General Description

This computer program is written to calculate the time history of the resulting forces and moments at assigned nodes in the dynamic model of the RPV support system. These forces are produced due to the external asymmetric pressure in the reactor cavity, resulting from a LOCA near a hot leg or cold leg nozzle. The pressure forces may be acting at the reactor pressure vessel, the primary shield wall, or the neutron shield tank.

The program performs the following calculation:

$$P(t) = \sum Pj(t) = \sum A_i x P_i(t)$$

$$M(t) = \sum Pj\ (t)\ x\ R_j$$

where:

$P_i^{(t)}$ =A pressure time history for pressure area No. i

A_i = An area vector corresponding to the pressure time history p (t)

Pj(t) = Force vector due to pressure acting on No. j area

P(t) = Resultant force vector due to pressures acting on all areas

R_j = Displacement vector from a force application point on an area to the point of rotation

M(t) = Resultant moment vector due to all pressures

To calculate the force and moment time history at a node in a given structural model, the surface area on which the pressure acts is divided into several regions, such that only a constant pressure acts on one region at any time.

The projection of a surface area, multiplied by the pressure acting perpendicular to it, gives the pressure force. This force is broken into three global components by defining the direction cosines of the pressure force vector.

The centroid coordinates of the projected area are then calculated, and when these are subtracted from the coordinates of the node, the displacement components are obtained, which are used in calculating the three moments at the node.

The forces and moments for all projected areas, due to corresponding pressure forces, are thus calculated and summed for each time point to get the force/moment time history for the node.

3A.2.13.2 Program Verification (ME-171)

A sample problem was performed by the use of the computer code ME-171. The results were then compared to results obtained by hand calculations.

3A.2.14 LIDOP (ME-184)

3A.2.14.1 General Description

LIDOP (ME-184) generates crush rigidities and deformation energies for pressurized or unpressurized piping in the following geometries:

1. Ring crush against flat rigid surface
2. Indent of straight pipe against rigid cylinder
3. 1.5D pipe elbow (extrados) against flat rigid surface
4. Pipe bend (extrados) against flat rigid surface
5. Indent of straight pipe against rectangular block

Both dynamic and material properties are considered in generation of the crush characteristics.

Unpressurized force-displacement and energy-displacement characteristics of pipe and elbows are generated from empirical equations which are based on experimental data. Pressurization effects, based on fluid displacement during deformation, are superimposed on the unpressurized characteristics. The overall dimensions of the contact area, where applicable, are generated by empirically corrected geometric relationships. Dynamic effects of elbows are empirically determined from an experimental comparison of static and dynamic impact of spheres. Dynamic effects of all other geometries and elbows in certain cases are based on the results of finite element computer simulations of rings impacting flat, rigid surfaces. The effects of material properties are determined from empirical relationships based on computer predictions (EMTR-2-0, 1976; EMTG-33-0, 1977; EMTR-403-B, 1978; Standard Review Plan 3.6.2, NUREG-75/087).

3A.2.14.2 Program Verification

Several sample problems were performed by the use of the code ME-184 to ensure that all options were exercised and thus demonstrate the function and adequacy of the program.

3A.2.15 Dynamic Load Factors (DLF ME-185)

3A.2.15.1 General Description

The ME-185 program determines the dynamic load factor (DLF) for a single degree-of-freedom harmonic oscillator subject to an arbitrary force history. At time zero, the oscillator is assumed to be in equilibrium and at rest. Its response to the force history, defined by a series of force-time pairs, is then computed. If the force is not specified at time zero, it is automatically set at zero and ramps up linearly to the first specified force-time coordinate. If the force is specified at time zero, it is assumed to be suddenly applied.

Using the initial conditions at time zero, ME-185 solves the equation of motion and searches for maxima during the interval up to the next specified force-time pair. It then determines the boundary conditions (position and velocity) at the end of the interval and uses these as initial conditions for the solution during the next time increment. The process is repeated until the last force-time pair is reached. Assuming this last force is applied as a continuing load, the steady state response is computed and maxima determined. The greatest maxima is then divided by the greatest applied load to determine the maximum load factor. Since the solution method searches for the greatest absolute amplitude of the system response and applied force, the applied force may be positive or negative and may arbitrarily change signs during the specified force history.

The dynamic load factor depends on the natural frequency of the single degree of freedom oscillator. In order to provide the DLF at the frequency of the structural system being analyzed, as well as to show how the DLF changes to an error in the calculated frequency or in the duration of the applied force, DLFs are computed for a range of frequencies. The above calculation method is repeated for each of several discrete frequencies in the range. The frequency range extends approximately one order of magnitude to either side of the frequency corresponding to the period (duration) of the applied force history.

For a force which varies linearly between two specified force-time pairs, the equation of motion is:

$$M\ddot{x} + kx = F_o + a_o t$$

where:

M = mass of the oscillator

k = stiffness of the oscillator

F_0 = applied force at the start of the interval

a_0 = rate of change in the applied force (F/ T)

The solution of this equation is:

$$x = C_1 \sin(wt) + C_2 \cos(wt) + C_3 + C_4 t$$

where:

$w = \sqrt{K/M}$, the natural frequency

$C_1 = (V_o - a_o/k)/w$

$C_2 = X_o - F_o/k$

$C_3 = F_o/k$

$C_4 = a_o/k$

x_0 = initial position

V_0 = initial velocity

The above relations are simplified and the magnitude of the spring force is made identical to displacement of K=1. This relation is used in ME-185.

When transferring from one time interval to the next, the position and velocity must be determined before the new coefficients, $C_1 ... , C_4$ can be calculated. The position at the end of the previous time interval can be computed from the above equation for X and the velocity may be determined from:

$$V = C_1 w \cos(wt) - C_2 w \sin(wt) + C_4$$

In any interval, the maxima or minima may be computed by substituting the times of zero velocity in the equation for X. These times may be computed from the relation:

$$0 = C_1 w \cos(wt) - C_2 w \sin(wt) + C_4$$

If $C_2 = 0$ then,

$$wt = \cos^{-1} (-C_4/(C_1 w))$$

If $C_4 = \tan^{-1} (C_1/C_2)$

$$wt = \tan^{-1} (C_1/C_2)$$

If $C_1 = 0$ then,

$$wt = \sin^{-1} (C_4/C_2 w))$$

otherwise:

$$wt = \cos^{-1}\left[\frac{-C_1 C_4 w \pm C_2 w\sqrt{(C_1^2 + C_2^2)w^2 - C_4^2}}{(C_1^2 + C_2^2)w^2}\right]$$

3A.2.15.2 Program Verification

Several problems were performed by the use of DLF (ME-184) to ensure that all options were used and thus demonstrate the function and adequacy of the program.

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TABLE 3A.2.1–1 INFINITELY LONG SOLID CYLINDER, PERTINENT PARAMETERS

Dimensions and Properties	Loading and Boundary Conditions
r_0 =a	σ_r = P (cos θ + cos 2 θ)
l = a	σ_q = P_0 sin
E = 10 x 10^6 psi	U_z = 0 U_r = 0
Y = 0.25 a = 1 inch	At r = 0, P_o = 10,000 psi

TABLE 3A.2.4–1 COMPARISON OF RESULTS OF SLOSH VS AEC ANALYSIS*

Example 1	Page 188*	SLOSH
W_0, Eq. Impulsive Force (kips)	298.5	299.7
P_0, Impulsive Force (kips)	105.4	105.8
M (EBP), Impulsive Moment (kip-ft)	594	595
M (IBP), Impulsive Moment (kip-ft)	1,120	1,118
W_1, Convective Force (kips)	133	133
M_1 (EBP), Convective Moment (kip-ft)	212	214
M_1 (IBP), Convective Moment (kip-ft)	252	253
M_{max}, Maximum Moment (kip-ft)	1,372	1,371
P_{max}, Maximum Shear (kips)	127.9	128.5

Example 2	Page 192*	SLOSH
W_0, Eq. Impulsive Force (kips)	458	458
P_0, Impulsive Force (kips)	277	277
M (EBP), Impulsive Moment (kip-ft)	3,460	3,470
M (IBP), Impulsive Moment (kip-ft)	4,070	4,074
W, Convective Force (kips)	139	137
M_1 (EBP), Convective Moment (kip-ft)	552	547
M_1 (IBP), Convective Moment (kip-ft)	560	552
M_{max}, Maximum Moment (kip-ft)	4,630	4,626
P_{max}, Maximum Shear (kips)	301	301

Example 3	Page 197*	SLOSH
W_0, Eq. Impulsive Force (kips)	298.5	299.7

TABLE 3A.2.4–1 COMPARISON OF RESULTS OF SLOSH VS AEC ANALYSIS*

W_1, Convective Force (kips)	133	133
Mode 1		
Frequency (cps)	0.333	0.331
FB1, Seismic Force (kips)	23.46	23.80
FA1, Seismic Force (kips)	1.26	1.28
Mode 2		
Frequency (cps)	1.486	1.482
FB2, Seismic Force (kips)	-3.90	-3.91
FA2, Seismic Force (kips)	174.39	174.54
P_{max}, Maximum Shear (kips)	195.21	195.72

Source:

* U.S. Atomic Energy Commission. Nuclear Reactors and Earthquakes, Chapter 6, Dynamic Pressure on Fluid Containers. TID-7024, Division of Reactor Development, Washington, DC, August 1963.

TABLE 3A.2.6–1 COMPARISON OF EXPERIMENTAL DATA WITH ANALYTICAL DATA USING LIMITA 3

	Experimental Value	Limita 3 Computer Results	Percent Difference
Peak Deflection Mode 6	0.297	0.310	4.2
Peak Deflection Mode 9	Not Determined	0.870	-
Time at Peak Deflection in Mode 6	~0.004	~0.0042	4.8
Permanent Deflection in Mode 6	0.144	~0.140	2.8
Permanent Deflection in Mode 9	0.302	~0.310	2.6

FIGURE 3A.2.3–1 TARGET FOR LOW TRAJECTORY MISSILE

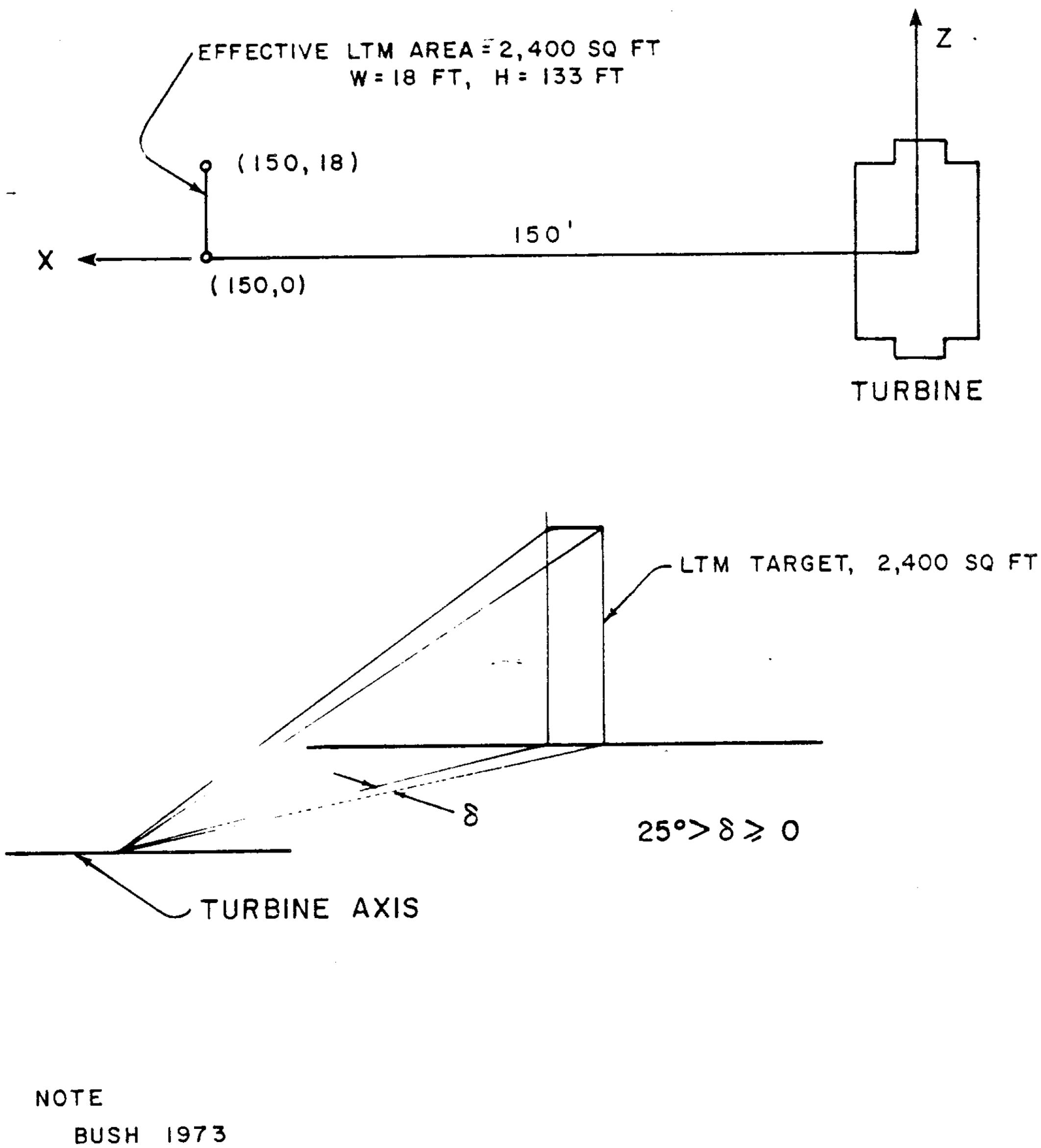
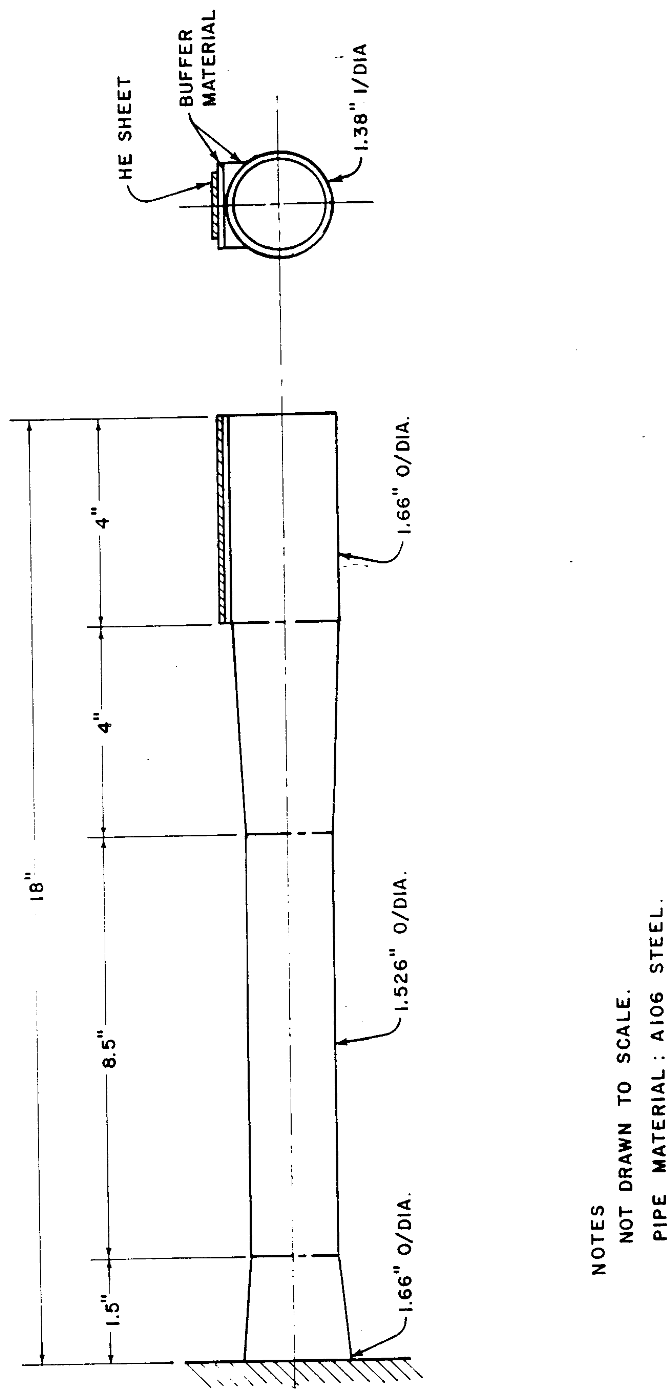


FIGURE 3A.2.11–1 CANTILEVER PIPE USED IN MIT EXPERIMENT



NOTES
NOT DRAWN TO SCALE.
PIPE MATERIAL : A106 STEEL.

FIGURE 3A.2.11-1
CANTILEVER PIPE USED IN MIT EXPERIMENT
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

3A.3 PIPING SYSTEMS

The following computer programs are used for the analysis of Seismic Category I piping systems

1.	NUPIPE II:	Linear Elastic Analysis of 3-D Piping System Subjected to Thermal, Static and Dynamic Loads
2.	PITRUST:	Local Stress Analysis at Junction of 2 Cylindrical Vessels
3.	PILUG:	Local Stress Analysis at Junction of Lug with Pipe or a Cylindrical Vessel
4.	SAVAL:	Stress Analysis at Nozzle/Run-Pipe Junction Due to Atmospheric Safety Valve Discharge
5.	STEHAM:	Steamhammer Transient Analysis
6.	WATHAM:	Waterhammer Transient Analysis
7.	PSPECTRA:	Peak Spreads and Envelopes ARS Curves of Dynamic Events
8.	STRUDL-SW:	Structural Analysis Program
9.	STRUDL-II (ICES):	Structural Analysis Program
10.	APEN:	Anchor Penetration Analysis
11.	PITRIFE:	Finite Element Analysis of Integral Welded Attachment
12.	PITAB:	Piping Support Load Tabulation Program
13.	CHPLOT:	Piping Support Load Tabulation Program
14.	LOADCOMB:	Load Combination Generator for STRUDL
15.	BEARST:	Pipe Bearing Stress Analysis
16.	ANCCOMB:	Anchor Load Combination Program
17.	BENDCORD:	Bend Co-ordinate Program
18.	NUPIPE-SWPC:	Linear Elastic Analysis of 3-D Piping System Subjected to Thermal, Static and Dynamic Loads
19.	PC-PREPS:	Structural Analysis Computer Code
20.	PILUG-PC:	Local Stress Analysis at Junction of Lug with Pipe or a Cylindrical Vessel

3A.3.1 NUPIPE II

1. General Description

The NUPIPE II piping program performs a linear elastic analysis of three dimensional piping systems subjected to thermal, static, and dynamic loads. It utilizes the finite element method of analysis.

NUPIPE II handles all loading conditions required for complete nuclear piping analyses. A given piping configuration may be analyzed successively for a number of static and dynamic load conditions in a single computer run. Separate load cases, such as thermal expansion and anchor displacements, may be combined to form additional analysis cases. The piping deadload analysis considers both distributed weight properties of the piping and any added concentrated weights.

A lumped mass model of the system is used for all dynamic analysis; both translational and rotational degrees of freedom may be considered. Location of lumped masses and degrees of freedom at each mass point are preselected by the analyst. The program automatically computes values of translational lumped masses.

Program input consists basically of program control, piping configuration description, and load specification information. The output for each loading condition analyzed consists of support reactions, internal forces and moments, deflections and rotations, and member stresses. Output from seismic analysis includes system normal mode information. Several reports may be generated based on report specification. These reports include pipe stress summaries, pipe support tabulations, and piping isometric plots.

The NUPIPE II program performs analysis in accordance with ASME Section III, Nuclear Power Plant Components (Code). Features ensuring code conformance include use of accepted analysis methods, incorporation of specified stress indices and flexibility factors, proper combination of moment resultants, and provision to (automatically) generate results of combined loading cases. A program option is available to specify among Class 1 analysis in accordance with NC-3600 of the Code, analysis per ANSI B31.1.0 power piping code and combined Class 1 and Class 2 analysis per Articles NB-3600 and NC-3600 of the Code.

2. Program Verification

The NUPIPE II program has been verified with ADLPIPE (A.D. Little Corp.) for thermal, weight, and response spectrum seismic analysis. The results from both programs are presented in Tables 3A.3–1 through 3A.3–7. The model used for this comparison is presented on Figure 3A.3–1.

The comparison is also made with ASME Benchmark solution (ASME 1972) for force time-history dynamic response. The model used for this comparison is shown on Figure 3A.3–2. The results for comparisons are presented in plots on Figure 3A.3–2. The natural frequencies are given in Table 3A.3–8.

The Class 1 piping stress conforms with the hand calculations. The model used is shown on Figure 3A.3–3. The results are tabulated in Tables 3A.3–9 and 3A.3–10.

3A.3.2 PITRUST

1. General Description

PITRUST is a computer program which calculates the local stress intensity at the junction of two cylindrical vessels. The calculated stresses, including those due to pressure, are determined for the run cylinder. The program has application where a trunnion is welded to a run-pipe or where a branch pipe exits from a vessel or run-pipe.

The method and theory of calculating stresses follows that promulgated by the Welding Research Council Bulletin No. 107 (Wichman et. al, 1965). The program is capable of complying with requirements of ASME Boiler and Pressure Vessel Code - Section III - Nuclear Power Plant Components and ANSI-B31.1 Power Piping. PITRUST input consists basically of program control options, run-pipe dimensions, internal operating pressure, trunnion outside diameter, and loading specification. If the design criteria for the stresses are exceeded, the program can incrementally increase the pad thickness and recalculate the stresses until the lug passes or until the pad reaches 1.5 times the pipe wall thickness.

Program output tabulates the applied loadings and the local stresses at the junction of the trunnion and run-pipe.

2. Program Verification

PITRUST has been verified by comparing its solution of a test problem to the solution obtained by CYLNOZ, an independently written piping local stress program. CYLNOZ, written by Franklin Institute (Philadelphia, Pa.), is a recognized program in the public domain. The test problem is of a 72.375 inches O.D. x 0.375 inches thick run-pipe, reacting under an external loading of 1,000 lb force (normal and shear) and 1,000 in-lb moments transmitted by a 16 in O.D. nozzle. A comparison of results is tabulated in Table 3A.3–11. PITRUST has also been verified by comparing its solution of a test problem to the experimental test results as outlined in Corum and Greenstreet (1971). A comparison of these results is tabulated in Table 3A.3–12.

3A.3.3 PILUG

1. General Description

PILUG is a computer program which calculates local stress intensity at the junction of a lug with a pipe or other cylindrical vessel. The stress intensity calculated is in the run-pipe and includes pressure stresses. The program has specific application where a rectangular attachment is welded to a run-pipe.

The method and theory of calculating stresses follows that promulgated by the Welding Research Council Bulletin No. 107 (Wichman et al., 1965). The program is capable of complying with requirements of ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components and ANSI-B31.1 Power Piping.

PILUG input consists basically of program control, run-pipe dimensions, internal operating pressure, rectangular lug dimensions, and loading specification.

If the design criteria for the stresses are exceeded, the program incrementally increases the pad thickness and recalculates the stresses until the lug passes or until the pad reaches 1.5 times the pipe wall thickness.

Program output tabulates the applied loadings and the local stresses at the junction of the lug and run-pipe.

2. Program Verification

PILUG has been verified by comparing its solution, of a test problem, to results obtained by hand calculations using the formulations specified in Wichman et al. (1965). A comparison of results is tabulated in Table 3A.3–13.

3A.3.4 SAVAL

1. General Description

SAVAL is a computer program which calculates the stresses at the junction of a safety valve nozzle and run-pipe. Calculated moments due to the suddenly applied thrust load are premultiplied by a dynamic load factor (DLF) prior to computing the nozzle and localized piping stresses. The subroutine DLF is incorporated in the program which computes the dynamic load factor.

The stress intensity in the run-pipe is computed and compared to allowable stresses in accordance with Equation 9 of Subsection NC-3652 of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components. Input for SAVAL includes discharge force vector, pipe, nozzle, and pad dimensions, valve weight, valve opening time, pressure, and code allowable stresses.

Computer output consists of resolved forces and moments at center line of run-pipe, dynamic loading factor, and calculated stresses.

2. Program Verification

Program SAVAL has been verified by comparing its solution of a test problem to results obtained by hand calculations. The test problem is illustrated on Figure 3A.3–4. A comparison of results is tabulated in Tables 3A.3–14 and 3A.3–15.

3A.3.5 STEHAM

1. General Description

   STEHAM is a computer program which is used to determine the steamhammer transients of piping systems. This program uses the method of characteristics with finite difference approximations both in space and in time (Jonsson et al.; Luk; Moody). It calculates the one-dimensional transient flow responses and the flow-induced forcing functions in a piping system caused by rapid operational changes of piping components, such as the actuation of a stop valve or safety/relief valve. Flow characteristics of piping components are mathematically formulated as boundary conditions in the program. These components include the flow control valve, the stop valve, the safety/relief valve, the steam manifold, and the steam reservoir. Frictional effects are taken into consideration in this program.

   This program accepts the following as input: (1) the flow network representation of the piping system, (2) the initial flow conditions along the piping system, and (3) time-dependent flow characteristics of piping components. Output consists of time-histories of flow pressures, flow densities, flow velocities, inertia, and momentum functions.

2. Program Verification

   STEHAM is verified by comparing its solutions of a test problem (Figures 3A.3–5 and 3A.3–6) to the results of the same problem obtained by an independent analytical approach, as well as an experimental measurement, as published in Progelhof and Owczarek (1963) and ASME Paper No. 63-WA-10). A comparison of results for time-history pressure responses is plotted on Figures 3A.3–7, 3A.3–8, and 3A.3–9. The forcing functions developed for nodal points of the piping system 3A.3-5 calculated from the relation $F = (p + \rho u^2/g)A$ has also been checked by hand calculations as tabulated in Table 3A.3–16.

3A.3.6 WATHAM

1. Description

   WATHAM is a computer program which is used to determine the flow-induced forcing functions acting on piping systems due to waterhammer. These forcing functions may then be used as input to a structural dynamic analysis such as a NUPIPE program run.

WATHAM is applicable to a waterhammer problem or, more generally, any unsteady, incompressible fluid flow. These events may be caused by normal or abnormal operational changes of piping components, such as the startup and trip of pumps or the rapid opening and closing of values.

The analysis is based upon the method of characteristics with finite-difference approximations both in time and space for the solution of one-dimensional liquid flows. Influences of piping components - including flow valves, pipe connections, reservoirs and pumps - have been considered in the analysis.

WATHAM input requires the geometry of the piping system, pipe properties, water properties, operational characteristics of pump and valve, flow frictional coefficients, and the initial water flow conditions. The output provides the; time history functions of piezometric heads, velocities, and nodal forces for all nodes and the inertial unbalanced force for each segment. It also gives the maximum value of all the above-mentioned functions and their occurring time in the process of flow-transient.

2. Program Verification

For the verification of WATHAM, a problem from Reference SWEC 1977 is employed. Figure 3A.3–10 depicts the flow network with nine pipes, its geometrical properties and steady state flow conditions. The flow-transient mode analyzed is the sudden closure of a valve at the downstream end. Figure 3A.3–11 shows the hydraulic network for WATHAM. Table 3A.3–17 illustrates the input data needed for WATHAM run. Figures 3A.3–12 and 3A.3–13 show the comparison of head-time curves obtained from Streeter and Wylie (1967), Fabic (1967) and WATHAM. Table 3A.3–18 presents the comparison of nodal forces between hand calculation and WATHAM computation.

In general, WATHAM results are in good agreement with Streeter's results (Streeter 1967). The small discrepancy is attributed to the modeling of reservoir boundary condition. In WATHAM, the energy equation between the reservoir is utilized rather than assuming that the head of pipe entrance is the same as that of the reservoir.

3A.3.7 PSPECTRA

1. General Description

The PSPECTRA program peak spreads and envelopes amplified response spectra curves of earthquakes or other dynamic events. The program reads ARS curves from tape, card, or disk files. The created curves are saved on a disk file, and optionally printed, plotted, and punched on a card file. Disk and card format is compatible with the NUPIPE program (ME-110). In fact, one important application of PSPECTRA is the creation of disk or card data sets of OBE and SSE curves for input to NUPIPE.

There are two methods of peak spreading - either the sides of the spread peaks can be vertical or they can be parallel to the sides of the original peaks. There are three methods of enveloping - maximum value, absolute sum, and SRSS. Another program option allows enveloping E-W and N-S direction curves to form one horizontal curve for each curve set on a disk file. ARS curves can also be input from up to four disk files in one run for the purpose of enveloping or plotting curves with different damping values. The last option allows inputting curves with up to four different damping values and superimposing curves with different damping values on the same plot.

The sides of vertical peaks actually span 0.001 second. The sides of parallel peaks are perfectly parallel to the original peaks. That is, the spread up and down the peak is based on the period of the tip of the peak.

All the peaks on a curve are spread, no matter how small they are. The final curve is the maximum value envelope of the individual spread peaks.

The program envelopes as follows: At each 0.001 second interval it either sums, finds the maximum value, or takes the SRSS of the accelerations from the individual digitized curves. The envelopes, therefore, automatically become digitized at the same intervals as the input curves. This means that a maximum value envelope is slightly above the “true” envelope, wherever the input curves intersect at a nonmultiple of 0.001 second.

After peak spreading or enveloping is complete, the curves are processed to eliminate points on lines of constant slope. Many of the interpolated points are eliminated during this process.

PSPECTRA can also be directed to create plots of each curve before and after peak spreading, both superimposed on the same graph.

2. Program Verification

For the verification of PSPECTRA, a problem was run three times, producing maximum, sum, and SRSS envelopes (Luong 1977). The results were verified by comparing its solution of the test problem to results obtained by hand calculations.

3A.3.8 STRUDL-SW

1. General Description

The STIFFNESS ANALYSIS Command of STRUDL-SW initiates the execution of a static, linear analysis of a structure considered as a lumped-parameter system.

The analysis is performed by the stiffness (displacement) method, treating the displacements as unknowns. From the user's structural and loading data, the program constructs the structural stiffness matrix and load vectors.

The computational procedure of the analysis is based on a network interpretation of the governing equations, the principal feature of which is the segmentation in processing of the geometrical, mechanical, and topological relationships of the structure. This allows a concise and systematic computational algorithm that is applicable for different structural types.

The basic steps in the procedure are as follows:

1. The stiffness matrix is determined for each member, considered as cantilevers (primitive stiffness matrix).

2. If there are any member releases specified, the local member stiffness matrix (Step 1) is modified.

3. The applied member loads, if any, are processed.

4. The structural stiffness matrix is assembled in the global coordinate system for free joints and released support joints.

5. The load vector is assembled and the global stiffness matrix and the load vector are modified to account for joint releases.

6. The governing joint-equilibrium equations are solved for the joint displacements.

7. The induced member distortions, member end forces and stresses are computed by back substitution.

Joint displacements are computed using a modified Gauss elimination procedure. The modifications include partitioning of the stiffness matrix and load vectors into blocks that represent the equations of one or more joints, and a “bookkeeping” algorithm that takes advantage of the symmetry, banding, and sparseness of a typical stiffness matrix.

From the joint displacements, other results are calculated for joints and members. For joints, member end forces are summed to yield reactions at support joints and to check the accuracy of analysis at free joints. For members, relative distortions at the end, and forces and moments at the beginning and end are calculated.

The above result types are not the only ones that may be listed following a stiffness analysis; they are, however, the only results that are calculated automatically during analysis. Requests for other types initiate additional calculations when the output request is made.

2. Program Verification

STRUDL-SW has been verified by comparing the results of the STRUDL-SW runs with either hand calculations or results obtained by running the same test problems under a public domain version of STRUDL (Luong 1980). In this particular case, the public domain version of STRUDL used was GTICES STRUDL (February 1980 VIM7).

3A.3.9 STRUDL-II (ICES)

1. General Description

The present capabilities of STRUDL provide the engineer with the ability to specify characteristics of problems, perform analyses, reduce and combine results, and output any information stored within the system. Several different types of analysis procedures are currently available. This version of STRUDL also includes a variety of member design capabilities which provide both a set of useful techniques for aiding the designer in his design studies and a mechanism for adding new design procedures.

Analytic procedures in ICES STRUDL-II apply to both framed structures and continuous mechanics problems. Framed structures are two or three dimensional structures composed of slender, linear members, which can be represented by properties along a centroidal axis. Such a structure is composed of joints, including support joints, and members connecting the joints. A variety of force conditions on member ends and at support joints may be specified implicitly by means of structural type and orientation commands or explicitly for a member or joint. Continuous mechanics problems are treated, in STRUDL, using the finite element method. In this method the domain of the problem is subdivided in one, two, or three dimensional elements, of different shapes, connected at a finite number of nodal points, or joints. This idealization is then entirely analogous to that of framed structures where members are a particular type of element. The present version of ICES STRUDL provides a variety of element types for the solution of plane stress/strain, plate bending, and shell analysis problems.

ICES STRUDL-II also contains the following types of design procedures applicable to the design of framed structures and their components: a frame optimization procedure which employs a discrete variable optimization procedure; a member selection capability with which various design procedures and codes can be used to design members from tables; designs subjected to arbitrary constraints; and reinforced concrete, beam, column, and slab adequacy checking and section proportioning, based on the ACI code.

2. Program Verification

Only partial verification has been performed on STRUDL-II (ICES). No unit testing has been performed on any of its commands (SWEC Computer Library 1977). All the commands, except for the CSM CALCULATIONS command, were qualified by using the comparison method. A system test was run on both STRUDL-II (V10L06) and GTICES (VIM6) using all commands except for the CSM CALCULATIONS command, and the results were compared. The closely-spaced modes feature of STRUDL-II was qualified by the manual method. A system test on STRUDL-II (V10L06) with the CSM CALCULATIONS command included was run and the results were compared with hand calculations.

3A.3.10 APEN

1. General Description

APEN is a computer program that analyzes anchor penetrations. Anchor penetration consists of three phases, as follows:

<u>PHASE 1 Input</u>

In the input phase, the control cards are read and defaults are taken, where necessary. The loads from both sides of the anchor are read, identified, and stored. All earthquake type loads are stored as absolute values.

<u>PHASE 2 Design</u>

The loads from both sides are combined and summed according to the run condition. Anchor dimensions input as zero (or blank) are selected from a table according to pipe size. Each dimension is then checked against the required size, which depends on the set of combined loads. Undersized dimensions are incremented in preset steps until they are equal to or greater than the required size, or until preset maximums are reached. These combined loads are not used in PHASE 3, Stress Analysis.

<u>PHASE 3 Stress Analysis</u>

The load combinations are made on each side of the anchor individually. For each loading combination, the maximum stress intensity at each side is computed. The stresses printed are for the worst side, for that loading combination. When all the loading combinations have been computed, the next problem is attempted. The end of the input signals the program stop. A detailed description of the analysis can be found in calculation 570.470.1.13-NP(B)-012 (Luong).

The restrictions on using the analysis procedure are:

a. Thickness of concrete wall must be 18 inches or greater.

b. Recommended only for piping with an operating temperature at 150°F or lower

c. Local stresses in the concrete wall which are induced by the anchor loads should be reviewed by the Structural group.

2. Program Verification

APEN has been verified by using the manual method, found in the calculation titled, “Verification of ‘APEN’ Computer Program,” Calculation No. 570.470.1.13-NP(B)-010-Z14 (Luong 1977). The results of the test problem from the APEN computer run were compared with that of the hand calculation.

3A.3.11 PITRIFE

1. General Description

PITRIFE is a post processor program that utilizes the results of a finite element model of two intersecting cylinders to determine the local discontinuity stresses at the interface of the two cylinders (pipe and trunnion).

The PITRIFE program provides an efficient and accurate means of going from the stresses generated in the finite element models to the stresses found in an actual pipe-trunnion problem subjected to actual loads. It accomplishes this translation from the finite element model to the actual pipe trunnion problem in stages. In the first stage, the program reads in the pipe and trunnion geometry, the type of weld securing the trunnion to the pipe, all the applied loads, and the User’s choice of one of the four available load combination methods. In the second stage, based upon the problem geometry and the type of weld, the program generates a set of load independent stress coefficients by interpolating between the non-dimensional stress coefficients generated from the finite element models. In the third stage, it selects static and dynamic loads in accordance with the load combination method chosen, forms a total static and a total dynamic load vector, and uses these to generate 64 total load vectors. There are 64 total load vectors because the signs of the dynamic loads are arbitrary this results in 64 possible combinations of the components of the static and dynamic load vectors. In the fourth stage, it computes the principle stresses for each of these loads, determines the maximum stress intensity, and prints the results. The program repeats the third and fourth stages until stresses have been computed for all the desired combinations of static and dynamic loads.

2. Program Verification

The PITRIFE computer program has been verified by demonstrating that the maximum stress intensities as given by PITRIFE equal the values given by the finite element analysis for specific size-on-size and 0.707 size-on-size models. A comparison of these results is tabulated in Table 3A.3–19. The program was verified for other ratios of trunnion to pipe radius by demonstrating that the stress coefficients and maximum stress intensities derived by hand calculation equal the coefficients used in the program to calculate maximum stress intensity. A comparison of these results is given in Table 3A.3–20.

3A.3.12 PITAB

1. General Description

PITAB provides tabulations of vertical and other restraint data consisting of general support data and final design loads. The final design loads for each support are calculated using Millstone procedure. This procedure specifies the method of combining individual support loading conditions for various types of supports to arrive at the final design loads.

The program accepts titles, notes, procedure selection, and support data for each support consisting of general support information, individual loading conditions for the support, and its thermal displacements. The final design load for each support is then calculated using the procedure selected and two copies of the vertical and other support tables are printed. Each table contains a listing of the supports in ascending order of support number with their general support information and final design loads.

2. Program Verification

PITAB has been verified by using the manual method found in the calculation titled, "Verification of 'PITAB' Computer Program" (Quan). The results of the test problem from the PITAB computer run were compared with that of the hand calculation.

3A.3.13 CHPLOT

1. General Description

CHPLOT is a program which plots any number of data values (variables) versus time. Although the plot input data file can be in the form of card data, the more appropriate application of this program is to be used in conjunction with a program that creates a plot data file (on disk or tape) having the format required for input to this program.

Plots are available in two sizes; one with axes of 5 inches (ordinate) by 8 inches (abscissa) that fits the standard 8-1/2 inch by 11 inch page, and the other is 8 inches by 12 inches for fitting an 11 inche by 15 inche page. Plots are normally one data value versus time per graph, although up to 14 data values (plots) can be plotted on one graph.

Each graph's abscissa will be labelled as TIME (SEC) and the ordinate labels for each graph can be input. Ordinate axes can be selectively suppressed. Scaling is performed automatically to fit the size selected. Graphs can be grouped into a maximum of nine groups, within which each group of graphs will be scaled to the same scale factor.

An optional label is available for labelling all graphs at the bottom of each graph.

2. Program Verification

All the output options such as plot size, scale groupings, curve labels, and data values versus time are verified by visual inspection. It is found that CHPLOT performs all the options defined in the users' manual.

Table 3A.3–22 shows a partial listing of the forces plotted on Figure 3A.3–15, which illustrates this visual verification of the plots.

3A.3.14 LOADCOMB

1. General Description

The purpose of LOADCOMB computer program is for the design and selection of pipe supports required to comply with subsection NF of ASME III, load conditions and combinations must be generated to be applied to the support frames. In order for the loads to be input to STRUDL and an NF code check performed, the final output from the program is generated in the form of STRUDL JOINT LOAD commands on a user specified data set.

The LOADCOMB program uses two different coordinate systems, the local run-pipe coordinate system (identical to the NUPIPE, ME-110, local run-pipe coordinate system) and the STRUDL global coordinate system.

The user of LOADCOMB must input the individual load components into LOADCOMB in the local run-pipe coordinate system. The local x axis for this system coincides with the longitudinal axis of the pipe. The y and z axes must coincide with the principal axes of the pipe. (See Attachments 6.1 and 6.2 of EMTG-31-0, 1977). The program internally forms the load combinations in the local run-pipe coordinate system.

The STRUDL JOINT LOAD commands are output in a STRUDL global coordinate system.

2. Program Verification

LOADCOMB has been verified by using the manual method found in the calculation titled, "Qualification Calculation to Verify Computer Program

ME-163, 'LOADCOMB'," (Sexton 1977). The results of the test problem from the LOADCOMB computer run were compared with that of the hand calculation.

3A.3.15 BEARST

1. General Description

BEARST is a computer program that calculates the maximum local stress in a pipe wall due to contact loads between the pipe and a structural element.

The program is capable of complying, as per SATM-17, Rev. 4 with the code requirements of ASME Section III (Class 1, 2, and 3) and ANSI B31.1.

Loads representing the forces that the run-pipe exerts on the frame are input according to a local 1-2-3 coordinate system whose origin is at the center of the run-pipe and where the 3 axis points along the centerline of the run-pipe, the 2 axis points in a general upward direction, and the 1 axis follows the right-hand rule. The program is applicable to all types of 1-way, 2-way, and 3-way restraint problems. The program first assigns a set of signs to the loads in which each component can be + or - (earthquake loads); it then stores them along with the fixed - sign loads in an array of loads at each contact point. Loads at each restraining point are then combined to conform to the various load conditions, and within some load conditions there are a number of load combinations needed to assure for the appropriate multiple thermal loads and occasional loads. Local stresses are computed for each restraining point and stress intensity is listed for each load combination point by point. In general, stress intensities are calculated for each of the two elements taken from the outer and inner surface of run-pipe at the restraining point. Appropriate minimum normal pipe stresses are added to the largest one of the stress intensities for each load combination and the total compared to the appropriate allowables.

Provision is included for considering up to three multiple thermal loads for ASME Section III code compliance and also up to three multiple occasional loads within each operating condition.

2. Program Verification

BEARST computer program has been verified by using the manual method found in the calculation titled, "Verification of BEARST Computer Program," General -NP(B)-011-Z14 (Luong 1976). The results of the test problem from the BEARST computer run were compared with that of the hand calculation.

3A.3.16 ANCCOMB

1. General Description

ANCCOMB is a computer program that combines the loads on both sides of the terminal anchor for each load type, in order to form a total anchor load for that load type and to select the smaller of the two general stresses input for each side of the terminal anchor to obtain minimum normal stress (MNS). Terminal anchors are anchors common to two independently analyzed piping systems, one on each side of the anchor.

The load combination method is based on the EMTG-31-0, Section 2.16 rules.

a. For earthquake load types, OBEI; OBEA; SSEI; and EART, the total anchor load component is the square root of the sum of the squares of the two components from both sides of the anchor.

For example:

$$Mx_{OBEI_{Total}} = \sqrt{Mx_{OBEI_1}^{2} + Mx_{OBEI_2}^{2}}$$

Where:

Mx_{OBEI} = Operational basis inertia load component (Mx) from one side of the terminal anchor

Mx_{OBEI} = Load component from the other side of the anchor.

The ‘MNS’ value associated with these earthquake loads is zero.

b. Operational basis earthquake, combined inertia and anchor movement loads (OBET) are formed by adding together by absolute value summation the individual components of the OBEA and OBEI loads and then taking the ± value of such sum. At terminal anchors, such summation is to occur only after the total OBEI and OBEA anchor loads have been formed.

c. For occasional dynamic loads associated with upset, emergency, and faulted conditions, i.e. OCC (UEF) the two OCCMAX, MIN sets from each side of the anchor are to be combined so as to yield one OCCMAX, MIN set by SRSS'ing individual components from each side (as in accordance with the SRSS operator of Appendix B, Section A) in such a fashion as to obtain the highest possible positive and negative magnitudes for the total OCCMAX, MIN set.

2. Program Verification

ANCCOMB computer program has been verified by using the manual method found in the calculation titled, “Verification of ‘ANCCOMB’ Version 00 Level

01," 570.47.01-NP(B)-017 (Purohit 1979). The results of the test problem from the ANCCOMB computer run were compared with that of the hand calculation.

3A.3.17 BENDCORD

1. General Description

BENDCORD is a Fortran IV Program which supplies, in printed and card form, data for coding segments of a circle for use in the NUPIPE piping program.

BENDCORD has the capacity of operating on an arc which lies in any one of three planes defined by the Cartesian Co-Ordinate System.

The piping system may be divided into equal or unequal segments so as to aide the user in placing supports.

For seismic piping systems, the user is provided the option of locating mass points at alternating nodes or, if desired, at every node.

Various elbow types may be inputted to reflect the requirements of Class 1 analysis.

BENDCORD divides an arc into tangent lines, the lengths of which are then calculated by subtracting the coordinates of the tangent point from the tangent intersection point of the tangent lines.

2. Program Verification

The BENDCORD program is verified by calculating distances (or offsets) between nodal points by hand and comparing these values to those calculated by BENDCORD.

The problem consists of a 180 degree piping arc in the X-Z plane beginning with node 10 at phi = 45 degrees, and ending at node 36. There are 26 included angles in the arc, all equal. Mass points are at every other node. Refer to Figure 3A.3–14 for a graphic representation of the problem.

Partial results are shown in Table 3A.3–21. It can be seen that there are no significant differences between the results from BENDCORD and the hand calculations.

3A.3.18 NUPIPE-SWPC

1. General Description

The NUPIPE-SWPC program is used to perform detailed pipe stress analysis. This program is designed to perform analyses in accordance with the ASME B&PV Code, Section III Nuclear Power Plant Components and the ANSI/ASME B31.1 Power Piping Code.

2. Program Verification

Using an approved Quality Assurance Program, this computer program has been verified and validated and shown to be accurate and acceptable for use in evaluating piping systems in accordance with the ASME B&PV Code, Section III and ANSI/ASME B31.1 Power Piping Codes.

3A.3.19 PC-PREPS

1. General Description

PC-PREPS is a PC based computer program which performs a complete structural analysis, performing an AISC code check, weld qualification and baseplate/anchor bolt qualifications.

2. Program Verification

Using an approved Quality Assurance Program, this computer program has been verified and validated and shown to be accurate and acceptable for use in evaluating ASME B&PV Code, Section III and ANSI/ASME B31.1 Power Piping Code components.

3A.3.20 PILUG-PC

1. General Description

PILUG-PC is a PC based stress analysis program used to calculate stress intensity at the junction of a rectangular attachment perpendicular to round pipe.

2. Program Verification

Using an approved Quality Assurance Program, this computer program has been verified and validated and shown to be accurate and acceptable for use in evaluating ASME B&PV Code, Section III and ANSI/ASME B31.1 Power Piping Code components.

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3A.3-18 Sexton, R.W. 1977. Qualification Call. to Verify Computer Program ME-163, 'LOADCOMB'. SWEC Computer Library, November 23, 1977.

3A.3-19 Streeter, V.L. and Wylie, E.G. 1967. Hydraulic Transients, McGraw-Hill Book Company, New York.

3A.3-20 SWEC (Stone & Webster Engineering Corporation) Computer Library 1977. Part Verification of STRUDL-II (ICES) Computer Program. January 1977.

3A.3-21 Wichman, K.R.; Hopper, A.G.; Mershon, J.L. 1965. Local Stresses in Spherical and Cylindrical Shells due to External Loading, Welding Research Council Bulletin, WRC-107.

TABLE 3A.3–1 COMPARISON OF SUPPORT REACTION DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

		Forces (lb)			Moments (in-lb)		
Node	**Program***	**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

NOTE:

* See Figure 3A.3–1 for General NUPIPE II Model. (All node points listed here are not shown.)

TABLE 3A.3–2 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 3A.3–3 COMPARISON OF STRESS DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

Node	NUPIPE (psi)	ADLPIPE (psi)
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 3A.3–4 COMPARISON OF INTERNAL FORCES DUE TO DEADWEIGHT ANALYSIS

Node	Program	Forces (lb)			Moments (in-lb)		
		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 3A.3–5 COMPARISON OF DEFLECTIONS AND ROTATION DUE TO DEADWEIGHT ANALYSIS

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	-0.015	-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 3A.3–6 COMPARISON OF STRESSES DUE TO DEADWEIGHT ANALYSIS

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 3A.3–7 COMPARISON OF NATURAL FREQUENCIES

Mode	1st	2nd	3rd	4th	5th
NUPIPE	7.109	9.328	12.297	14.681	18.043
ADLPIPE	7.118	9.329	12.492	12.427	17.714

TABLE 3A.3–8 COMPARISON OF NATURAL FREQUENCIES

Mode	1st	2nd
NUPIPE	2.407	13.537
Benchmark Pr.	2.3288	13.0808

TABLE 3A.3–9 COMPARISON OF CLASS 1 PIPE STRESS ANALYSIS

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 (Eq. 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)	99,884 psi	99,781 psi
Eq. 13	18,252 psi	18,246 psi
Alternate Stress (Eq. 14)	218,258 psi	217,811 psi
Usage Factor	Out of Range	

TABLE 3A.3–10 INDIVIDUAL PAIR USAGE FACTOR FOR POINT NO. 30

	Hand Calculation	NUPIPE
P_{air} (1, 5)	0.183	0.1803
P_{air} (1, 8)	1.660	1.7361
P_{air} (1, 9)	0.0001	0.0001
P_{air} (1, 10)	Not in Range	
P_{air} (5, 8)	Not in Range	
P_{air} (5, 9)	0.221	0.2646
P_{air} (5, 10)	0.747	0.8051
P_{air} (8, 9)	0.857	0.8832
P_{air} (8, 10)	5.5518	5.8608
P_{air} (9, 10)	0.0001	0.0001

TABLE 3A.3–11 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) (lb)	395.00	399.00	399.99
p (Bending) (lb)	1875.00	1883.00	1877.30
M (Normal) (in-lb)	35.85	35.57	36.06
M (Bending) (in-lb)	364.70	366.60	354.30
M (Normal) (in-lb)	79.05	79.66	79.54
M (Bending) (in-lb)	90.52	80.57	79.42
Axial			
p (Normal) (lb)	813.00	812.00	814.80
p (Bending) (lb)	812.30	827.00	810.60
M (Normal) (in-lb)	91.79	105.00	95.45
M (Bending) (in-lb)	158.80	160.00	158.80
M (Normal) (in-lb)	37.06	37.00	37.12
M (Bending) (in-lb)	117.90	105.00	103.85
Shear Stress by M (psi)	6.63	6.63	6.63
Shear Stress by V (psi)	106.10	106.10	106.10
Shear Stress by V (psi)	106.10	106.10	106.10

TABLE 3A.3–12 COMPARISON OF PITRUST WITH REFERENCE 8 RESULTS

Location and Cause	PITRUST Results	Exp. Results (Ref. 8)
Element "A"		
Longt. Moment		(Fig. 16, Ref. 8)
Circumf. Stress (psi)	20,438.9	20,000
Axial Stress (psi)	26,292.6	25,000
Element "B"		
Circumf. Moment		(Fig. 15, Ref. 8)
Circumf. Stress (psi)	22,016.2	24,000
Axial Stress (psi)	13,105.8	13,000

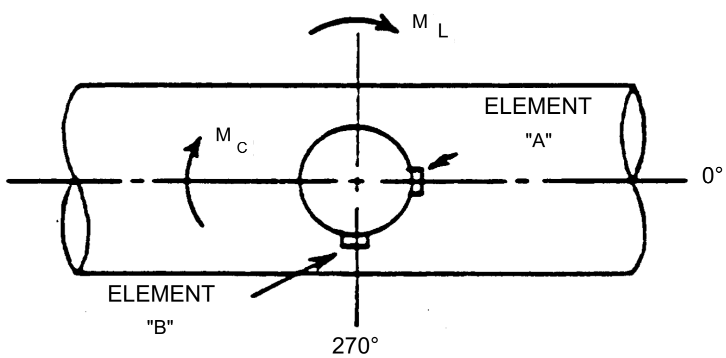


TABLE 3A.3–13 COMPARISON OF PILUG COMPUTER PROGRAM OUTPUT WITH HAND CALCULATIONS

Test Problems: Run Pipe O.D. = 17 in; Run Pipe Thickness = 0.812 in
Axial Length of LUG = 12 in;
Width of LUG along Circumf = 3 in
Loads: P = 3,399 lb; V_c = 1,788 lb; V = 2478 lb;
M_c = 81,834 in-lb; M1 = 103,320 in-lb
M_t = 76,284 in-lb

Stress in Circumferential Direction:

Figure		Stress From Hand Calculation	Computer Output	Remarks
3C	0.5485	387	330	Membrane stress due to P
1C	0.326	2,165	2,160	Bending stress due to P
3A	0.294	671	629	Membrane stress due to Mc
1A	0.388	18,976	19,904	Bending stress due to Mc
3B	0.467	3,014	2,961	Membrane stress due to M1
1B	0.416	6,143	5,969	Bending stress due to M1
Stress in Axial Direction:				
4C	0.4447	683	690	Membrane stress due to P
2C	0.4632	773	792	Bending stress due to P
4A	0.294	1,897	1,864	Membrane stress due to Mc
2A	0.550	6,357	5,942	Bending stress due to Mc
4B	0.467	2,365	2,328	Membrane stress due to M1
2B	0.582	4,989.7	4,842	Bending stress due to M1
Shear Stress:				
		1,304.8	1,304.8	Shear stress due to Mt
		-366.99	-366.99	Shear stress due to Vc
		127.15	127.16	Shear stress due to V1

TABLE 3A.3–14 SUMMARY OF COMPARISON OF SAVAL COMPUTER OUTPUT WITH HAND CALCULATION AS-DESIGNED CONDITION

Variable	Hand Calculation	SAVAL
Valve and Nozzle Weight (lb)	714.09	714.09
Run Pipe Stiffness (in-lb/Rad)	37,950,000	38,083,056
Nozzle Stiffness (in-lb/Rad)	1,120,000,000	1,120,619,008
Equivalent Stiffness (in-lb/Rad)	36,700,000	36,831,376
Nat. Rotational Frequency (cps)	22.0	22.19
Time Ratio	1.11	1.11
Dynamic Load Factor	1.22	1.22
Circumferential Moment x DLF (in-lb)	314,760	314,760
Net Vertical Force (lb)	14,423	14,436
Nozzle Stress (psi)	6,600	6,597*
PITRUST Stress Intensity (psi)	62,782	62,789*
Stress Intensification Factor	5.6	5.608
Equation (9) Stress (psi)	36,651.9	36,681*

NOTE:

* Allowable Stress = 1.2 S_h -1.2 (15,490) = 18,588 psi

TABLE 3A.3–15 SUMMARY OF COMPARISON OF SAVAL COMPUTER OUTPUT WITH HAND CALCULATION REINFORCED CONDITION (1 1/4” PAD)

Variable	Hand Calculation	SAVAL
Valve and Nozzle Weight (lb)	714.09	714.09
Run Pipe Stiffness (in-lb/Rad)	181,500,000	181,490,256
Nozzle Stiffness (in-lb/Rad)	1,120,000,000	1,120,619,008
Equivalent Stiffness (in-lb/Rad)	156,200,000	156,193,808
Nat. Rotational Frequency (cps)	45	45.7
Time Ratio	2.27	2.29
Dynamic Load Factor	1.13	1.13
Circumferential Moment x DLF (in-lb)	291,540	291,540
Net Vertical Force (lb)	14,436	14,436
Nozzle Stress (psi)	6.144	6.144
PITRUST Stress Intensity (psi)	17.046	17.046
Stress Intensification Factor	2.59	2.59
Equation (9) Stress (psi)	16,186	16,200

TABLE 3A.3–16 NODAL FORCE COMPARISON

Diameter D = 0.25 ft

Area $A = \pi D^2/4 = 0.0490874\ ft^2$

Nodal Force $= (p + \rho v^2 /g)\ A - P_{atm}\ A$

p = pressure lb/ft^2

ρ = density lb/ft^3

v = velocity ft/sec

g = gravitational constant 32.2 ft/sec^2

$P_{atm} = 14.7 \times 144\ lb/ft^2$

At time t = 0.00650 sec

Node No.	Pressure (psia)	Velocity (fps)	Density (lb/ft^3)	Force (lb) (STEHAM)	Force (lb) (Hand Calculation)
1	42.523	0.0	0.23954	186.57	196.67
5	42.785	5.7843	0.24076	198.43	198.53
10	44.231	31.219	0.24647	209.00	209.11
15	47.003	78.172	0.25737	230.62	230.73
20	50.214	129.89	0.26979	257.84	257.97
25	52.095	159.43	0.27697	274.93	275.06
30	52.209	161.97	0.27742	276.09	276.23
35	52.168	162.21	0.27731	275.83	275.97

TABLE 3A.3–17 INPUT DATA FOR WATHAM

Pipe No.	Total Length (ft)	Inside Dia (ft)	Friction Factor	No. of Nodes	Nodal Span (ft)	Thickness (in)	Velocity (fps)
1	2,000	3.0	0.030	7	333.33	0.30824	4.24413
2	3,000	2.5	0.028	9	375.00	0.44	2.92132
3	2,000	2.0	0.024	6	400.00	0.50026	4.98473
4	1,800	1.5	0.020	7	300.00	0.11108	3.59336
5	1,500	1.5	0.022	5	375.00	0.264	4.52142
6	1,600	1.5	0.025	6	320.00	0.13796	2.29183
7	2,200	2.5	0.040	8	314.29	0.21534	3.65878
8	1,500	2.0	0.030	6	300.00	0.14811	3.83245
9	2,000	3.0	0.024	7	333.33	0.30824	4.24413

NOTE:

The initial heads of all nodes are calculated by using the Darcy-Weisbach equation.

TABLE 3A.3–18 COMPARISON OF NODAL FORCE CALCULATION AT TIME = 2.34 SECONDS

Pipe No.	Node No.	Force, Kips (WATHAM)	Force Kips (Hand Calculation)
1	1	276.34	276.48
1	2	300.46	300.62
1	3	317.78	317.94
1	4	329.59	329.76
1	5	341.39	341.56
1	6	355.31	355.49
1	7	369.52	369.71

Nodal Force Calculation is based on the following equation:

$$F = A(pH + (\rho/g)V^2)$$

where:

F = nodal force (lb)

ρ = density (lb/ft^3)

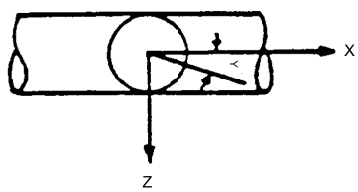
H = nodal head (ft)

g = 32.2 ft/sec^2

V = nodal velocity (fps)

A = pipe area (ft^2)

TABLE 3A.3–19 COMPARISON OF PITRIFE COMPUTER PROGRAM OUTPUT WITH STRUDL-II OUTPUT

Test Problem:		Size-on-Size	0.707 Size-on-Size
	Average Pipe Radius (in)	3.00	3.00
	Average Trunnion Radius (in)	3.00	2.12
	Pipe Wall Thickness (in)	0.30	0.30
	Trunnion Wall Thickness (in)	0.30	0.21

SIZE-ON-SIZE MAXIMUM STRESS INTENSITY--PSI ($\propto$= 30°)

Load	PITRIFE Output	STRUDL-II Output
FX = 10,000 lbs	5,763	5,768
FY = 10,000 lbs	7,844	7,846
FZ = 10,000 lbs	6,507	5,506
MX = 10,000 in-lb	1,329	1,329
MY = 10,000 in-lb	1,688	1,687
MZ = 10,000 in-lb	4,066	4,068

0.707 SIZE-ON-SIZE MAXIMUM STRESS INTENSITY--PSI ($\propto$ = 30°)

Load	PITRIFE Output	STRUDL-II Output
FX = 10,000 lbs	13,471	13,458
FY = 10,000 lbs	9,616	9,611
FZ = 10,000 lbs	20,105	20,030
MX = 10,000 in-lb	4,371	4,368
MY = 10,000 in-lb	2,467	2,467
MZ = 10,000 in-lb	6,178	6,176

TABLE 3A.3–20 COMPARISON OF PITRIFE COMPUTER PROGRAM OUTPUT WITH HAND CALCULATIONS

Test Problem:

Average Pipe Radius	=	1.5 in
Average Trunnion Radius	=	1.35 in
Pipe Wall Thickness	=	0.30 in
Trunnion Wall Thickness	=	0.27 in

LOADS FOR EACH LOAD TYPE COMBINED (DL, OBEI, THER, OCCU, ETC.)

FX = FY = FZ = 10,000 lbs

MX = MY = MZ = 10,000 in-lbs

MNS Stress = 200 psi

Internal Pressure = 100 psi

STRESS COEFFICIENTS--0.9 SIZE-ON-SIZE--FX LOADING ($\propto$ = 30°)

Stress Type	Coefficient By Hand Calculation	Coefficient From PITRIFE
Longitudinal--Inside Fiber	-1.2652	-1.2652
Circumferential--Inside Fiber	-0.2764	-0.2764
Shear--Inside Fiber	0.2041	0.2041
Longitudinal--Outside Fiber	0.7454	0.7454
Circumferential--Outside Fiber	1.3509	1.3509
Shear--Outside Fiber	0.2041	0.2041

MAXIMUM STRESS INTENSITY--0.9 SIZE-ON-SIZE ($\propto$ = 30°)

Load Condition	Maximum Stress Intensity--psi Hand Calculation	PITRIFE
P + DL + MNS_1	28,181	28,182
P + DL + SRSS (OBEI, OCCU) + MNS_2	73,220	73,220
P + DL + OBEA + THER + MNS_3	88,216	88,216
P + DL + OCCE + MNS_4	59,853	59,853
P + DL + SRSS (SSEI, OCCF) + MNS_5	73,220	73,220

TABLE 3A.3–21 BENDCORD PROGRAM--VERIFICATION PROBLEM

	Offsets Between Nodal Points (feet)	
	Manual	BENDCORD
X_1	4.27721	4.27715
Z_1	4.27721	4.27713
X_2	3.73046	3.73042
Z_2	4.76158	4.76151
X_3	3.73043	3.73041
Z_3	4.76152	4.76151
X_4	3.12932	3.12929
Z_4	5.17652	5.17642
X_5	3.12932	3.12929
Z_3	5.17652	5.17642
X_6	2.48254	2.48254
Z_6	5.51598	5.51588
X_7	2.48254	2.48254
Z_7	5.51598	5.51587
X_8	1.79956	1.77956
Z_8	5.77500	5.77489
X_9	1.79956	1.79959
Z_9	5.77500	5.77490

TABLE 3A.3–22 PIPE 1, SEGMENT 1 FORCE VERSUS TIME FOR RUN NUMBER S2807011

Time (sec)	Seg. Force (lb)
0.0	0.0
0.004	0.0
0.008	4415.83
0.012	9626.32 (Maximum Force)
0.016	6712.86
0.020	2681.74
0.024	965.24
0.028	2081.88
0.032	4794.40
0.036	6089.20
0.040	5097.82
0.044	3360.32
0.048	2580.63
0.052	3234.01
0.056	4383.75
0.060	4857.52
0.064	4390.47
0.068	3631.02
0.072	3333.10
0.076	3652.30
0.080	4135.11

FIGURE 3A.3–1 MATHEMATICAL MODEL FOR FLEXIBILITY ANALYSIS VERIFICATION

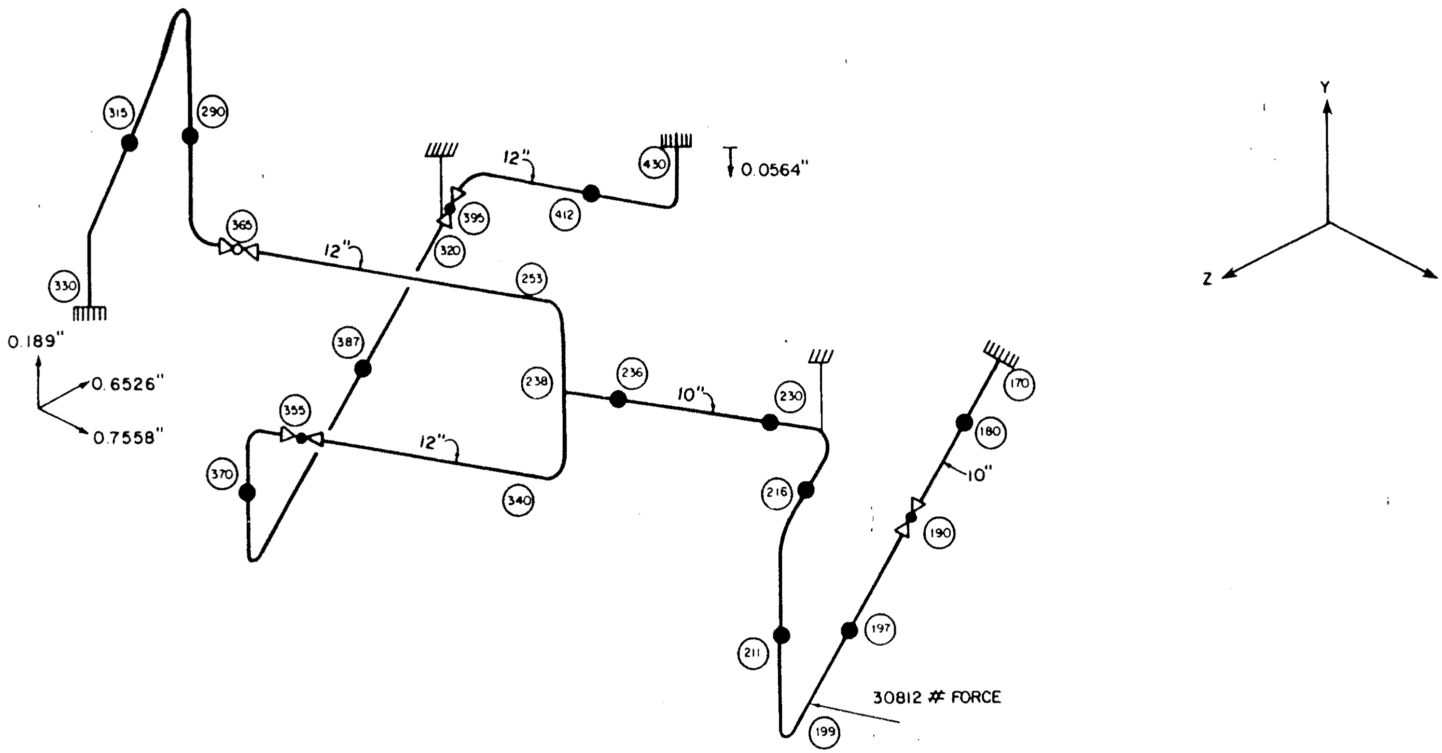


FIGURE 3A.3-1
MATHEMATICAL MODEL
FOR FLEXIBILITY
ANALYSIS VERIFICATION
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3A.3–2 NUPIPE PROGRAM FORCE TIME-HISTORY VERIFICATION

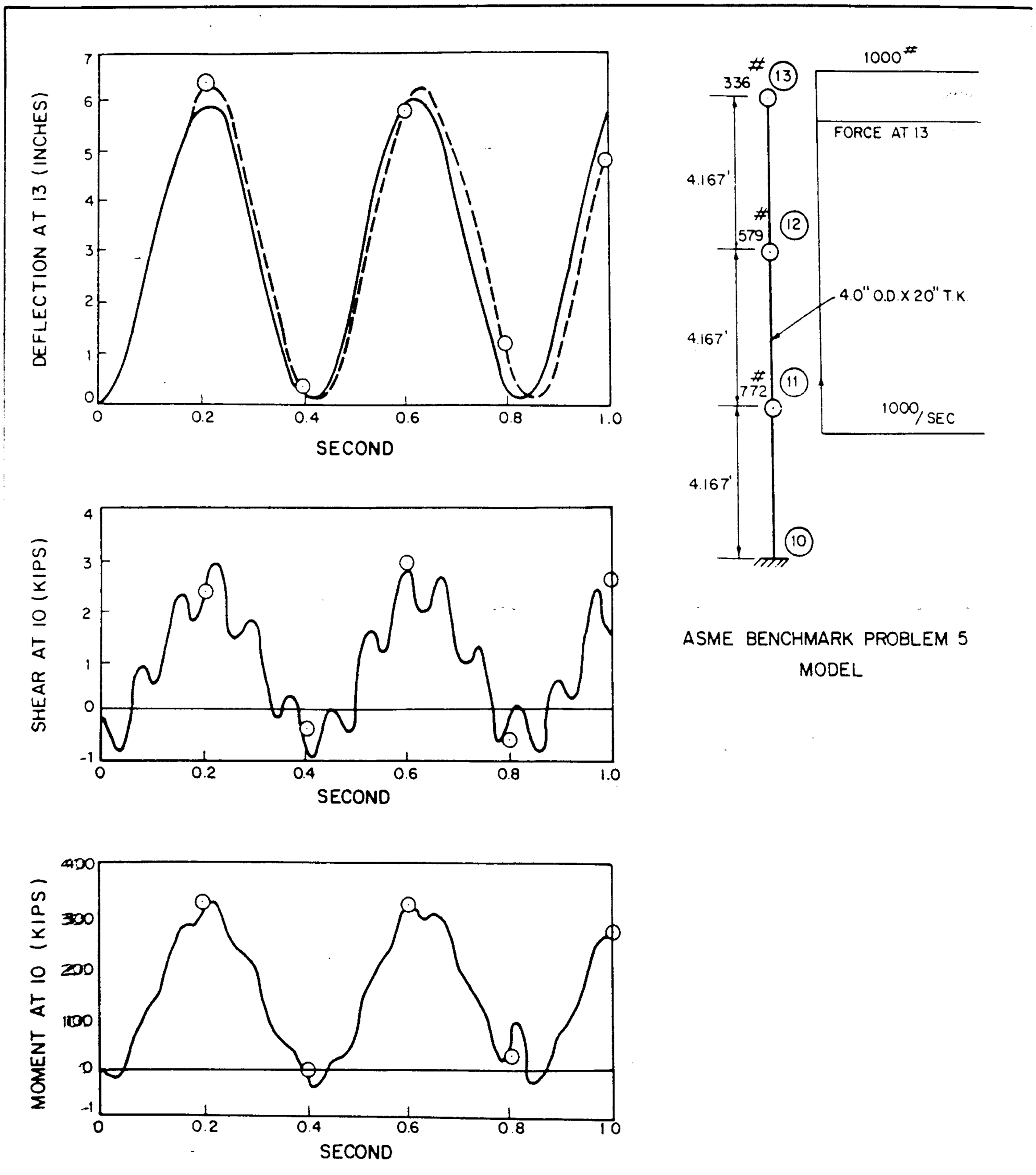
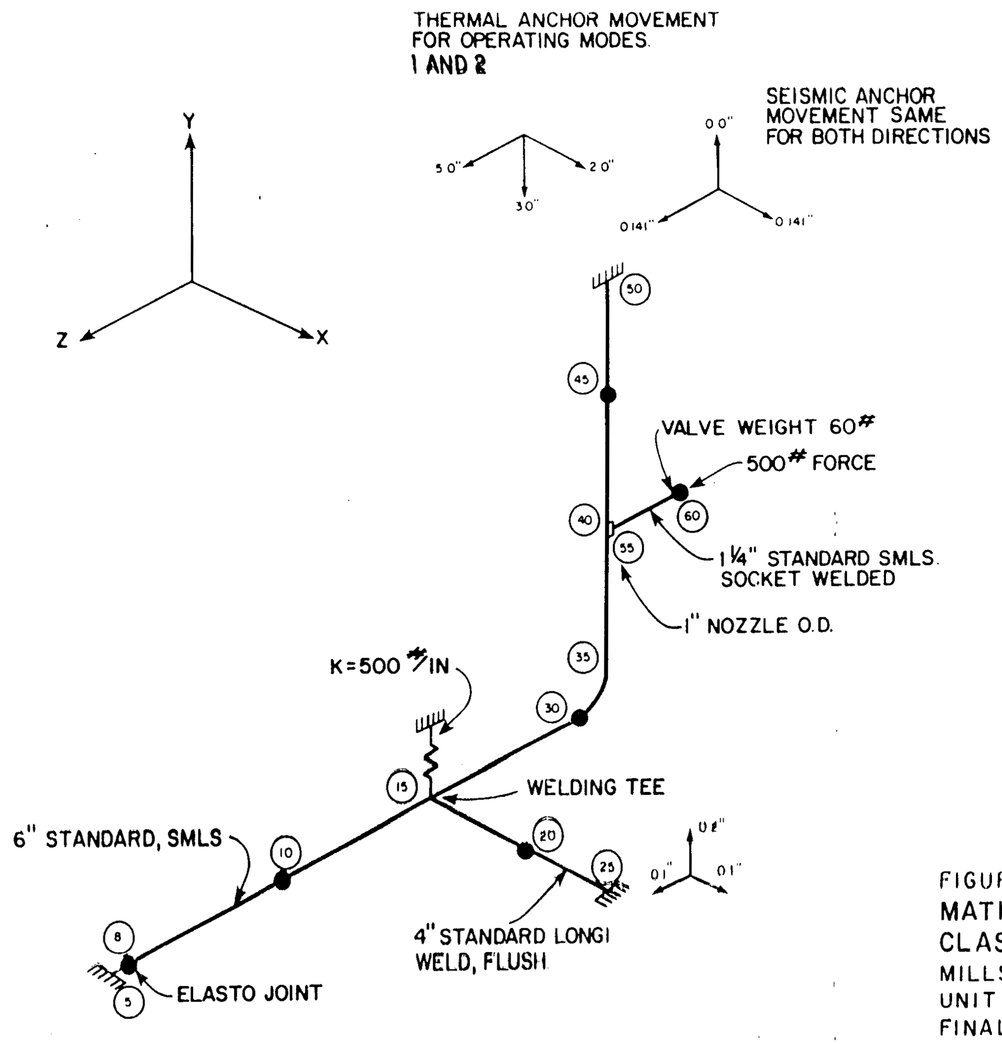


FIGURE 3A.3-2
NUPIPE PROGRAM FORCE TIME-HISTORY VERIFICATION
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3A.3–3 MATHEMATICAL MODEL FOR CLASS 1 STRESS VERIFICATION

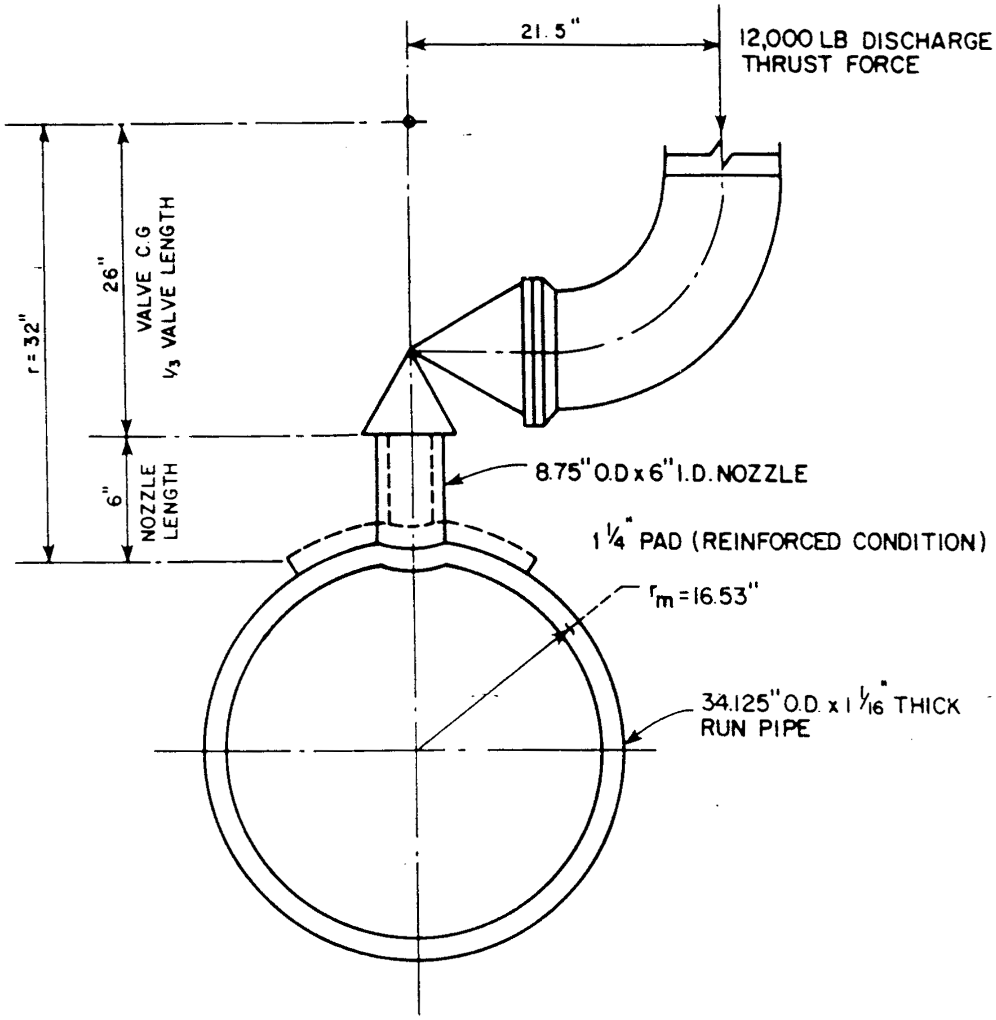


OPERATING CONDITIONS

OPER MODE	PIPE	PRESSURE (PSI)	TEMPERATURE °F
1	6"	200	400
	4"	"	"
	1¼"	"	"
$E/\alpha_1 Ta - \alpha b Tb / AT$ 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

FIGURE 3A.3-3
MATHEMATICAL MODEL FOR
CLASS I STRESS VERIFICATION
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3A.3–4 TEST PROBLEM - SAVAL

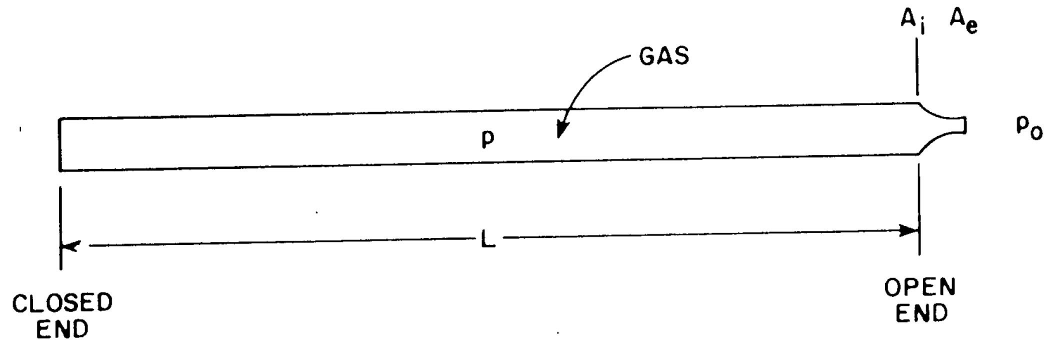


NOTE

VALVE WEIGHT = 660 LBS.
VALVE LENGTH = 78", CG = 1/3 x 78 = 26
VALVE OPENING TIME = 0.05 SECOND (ASSUME)
NOZZLE LENGTH = 6"
NOZZLE O.D. = 8.75" I.D. = 6.0"
VALVE SET PRESSURE = 850 psi

FIGURE 3A.3–4
TEST PROBLEM–SAVAL
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3A.3–5 SUDDEN DISCHARGE OF A GAS FROM A PIPELINE THROUGH A NOZZLE



CASE (A) FOR COMPARISON WITH ANALYTICAL RESULTS.

INITIAL CONDITIONS =

$p_1/p_o = 4.72$, $a_o/a_1 = 0.80$

DIMENSIONS =

$A_e/A_1 = 0.6$, L = PIPE LENGTH

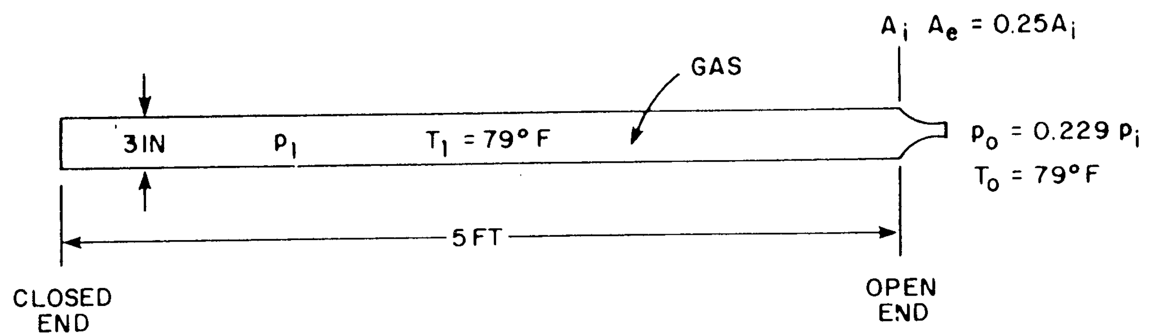
SPECIFIC HEAT RATIO =

$\gamma = C_p/C_v = 1.4$

p = PRESSURE, a = SOUND VELOCITY, A = FLOW AREA

FIGURE 3A.3-5
SUDDEN DISCHARGE OF A GAS FROM A PIPE LINE THROUGH A NOZZLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3A.3–6 SUDDEN DISCHARGE OF A GAS FROM A PIPELINE THROUGH A NOZZLE



CASE (B) FOR COMPARISON WITH EXPERIMENTAL DATA AND HAND CALCULATION.

PRESSURE = $P_o = 14.7$ psia, $P_1 = 14.7/0.229 = 64.2$ psia

AREA = $A_i = \frac{\pi}{4}(0.25)^2 = 0.0491\ ft^2$, $A_e = 0.25A_i = 0.0123\ ft^2$

GAS CONSTANT = $R = 53.35\ ft \cdot lb_f/lb°R$

TEMPERATURE = $T_o = 79°F = 539°R$, $T_1 = T_o$

DENSITY = $\rho_o = \frac{P_o}{RT_o} = 0.0736\ lb/ft^3$, $\rho_1 = \frac{P_1}{RT_1} = 0.3215\ lb/ft^3$

FIGURE 3A.3–6
SUDDEN DISCHARGE OF A GAS FROM A PIPE LINE THROUGH A NOZZLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3A.3–7 COMPARISON OF PRESSURE RESPONSE AT THE CLOSED END

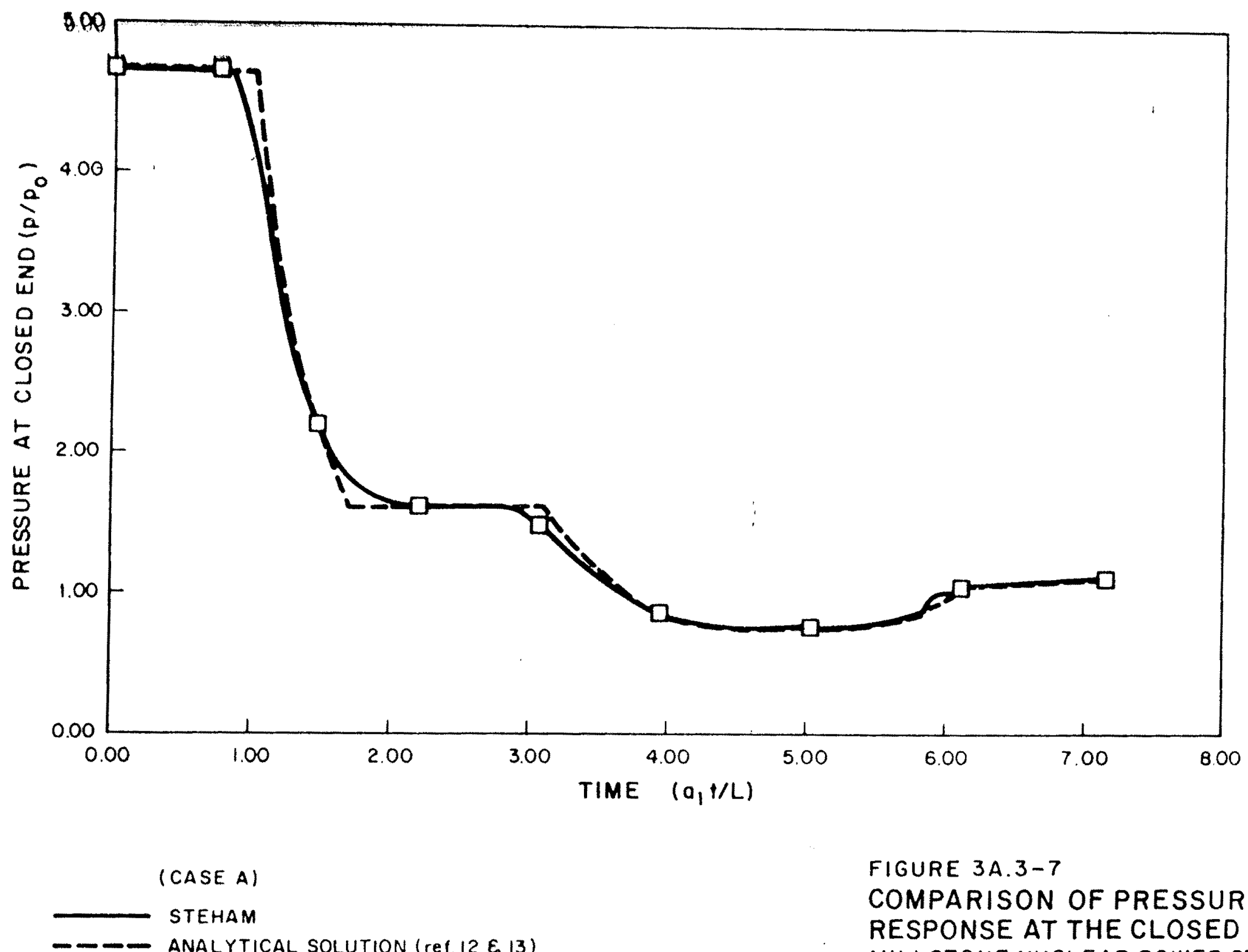


FIGURE 3A.3–7
COMPARISON OF PRESSURE RESPONSE AT THE CLOSED END
MILLSTONE NUCLEAR POWER STATION
UNIT 3

FIGURE 3A.3–8 COMPARISON OF PRESSURE RESPONSE AT THE OPEN END

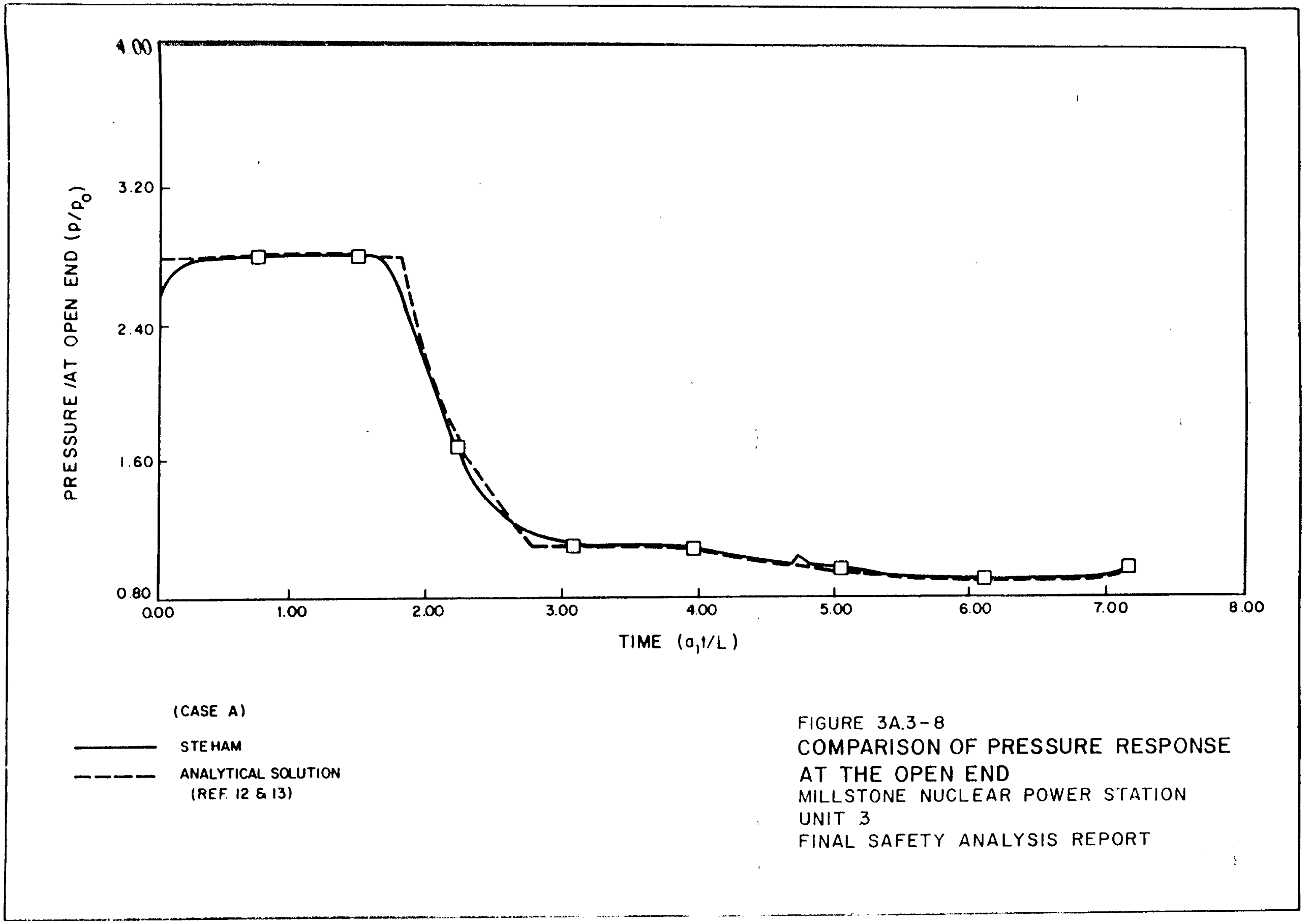


FIGURE 3A.3–9 COMPARISON OF PRESSURE RESPONSES BY STEHAM AND EXPERIMENT

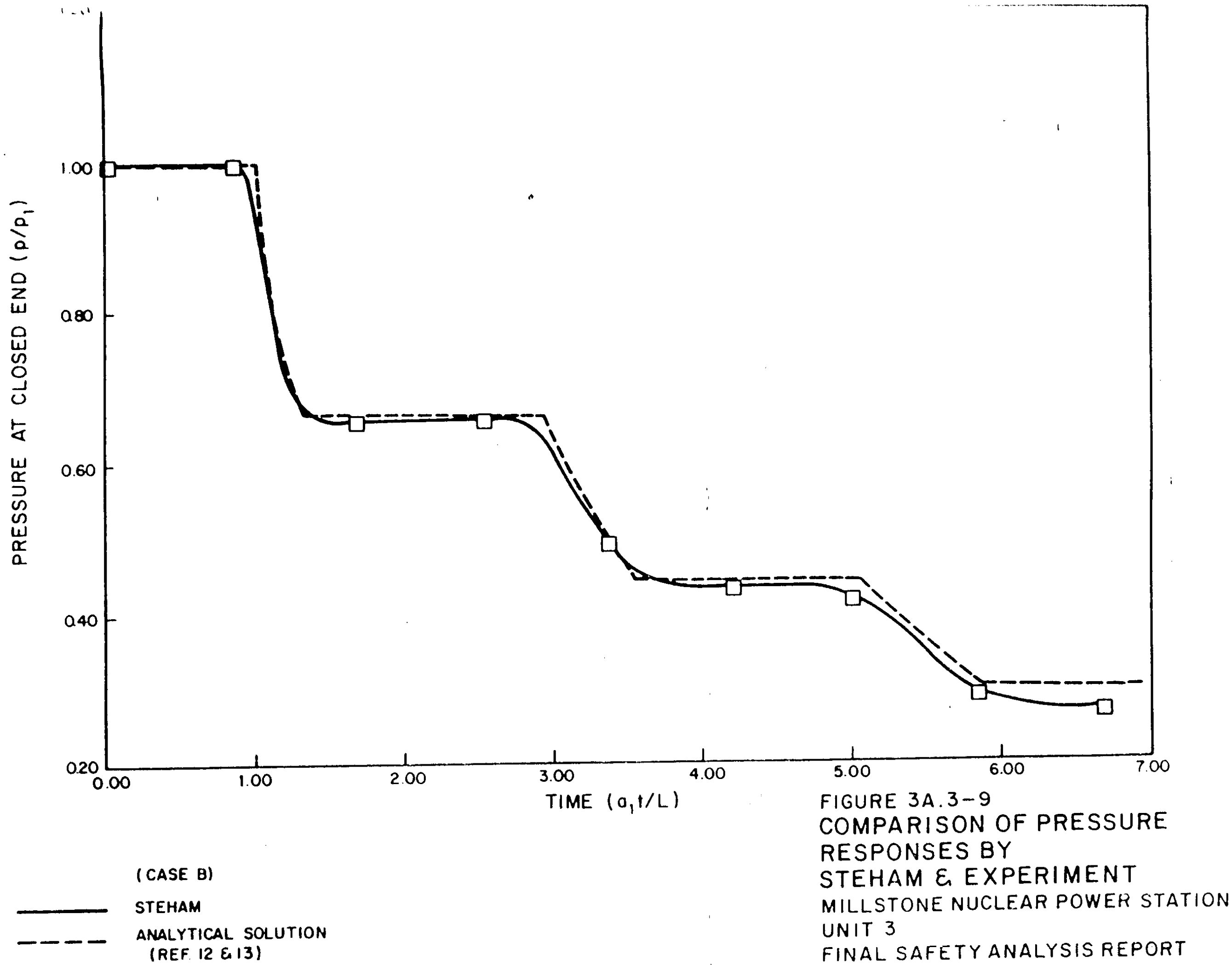


FIGURE 3A.3–9
COMPARISON OF PRESSURE
RESPONSES BY
STEHAM & EXPERIMENT
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3A.3–10 HYDRAULIC NETWORK FOR VERIFICATION PROBLEM

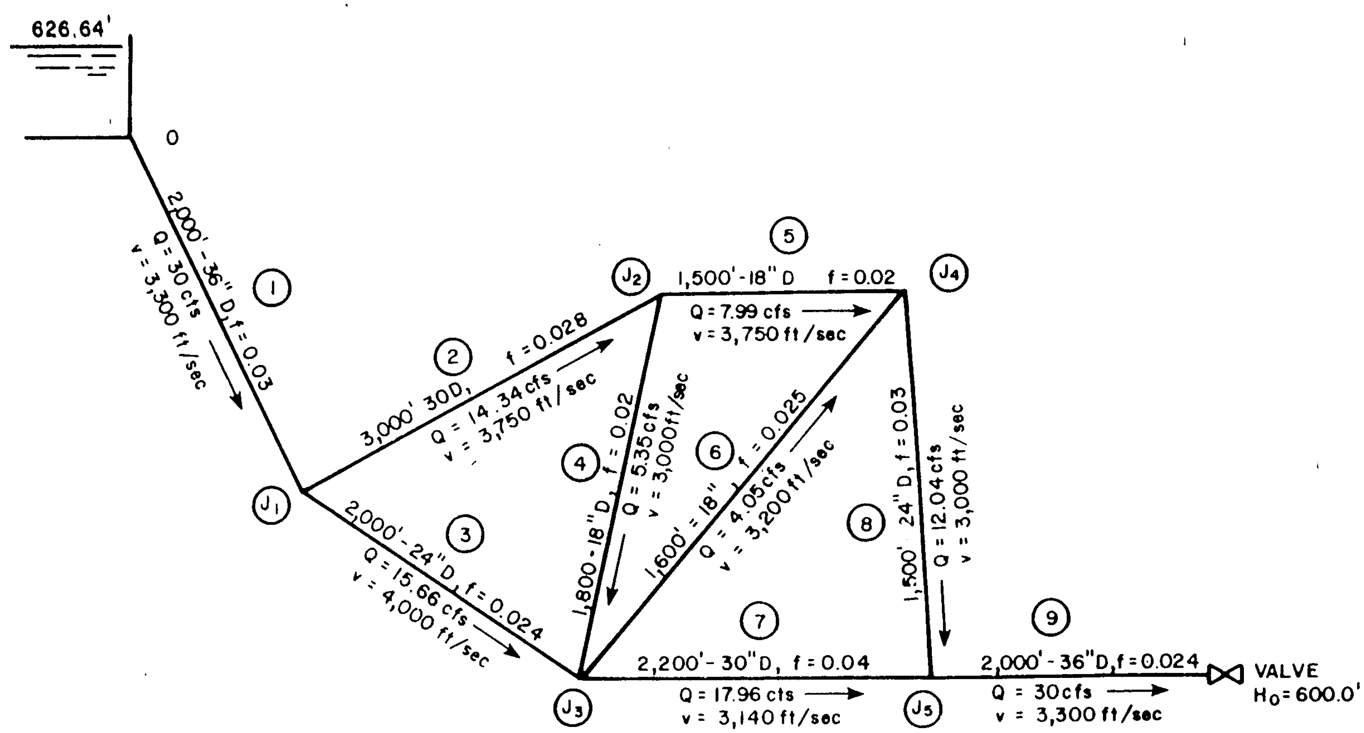


FIGURE 3A.3-10
HYDRAULIC NETWORK FOR
VERIFICATION PROBLEM
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3A.3–11 HYDRAULIC NETWORK FOR WATHAM VERIFICATION

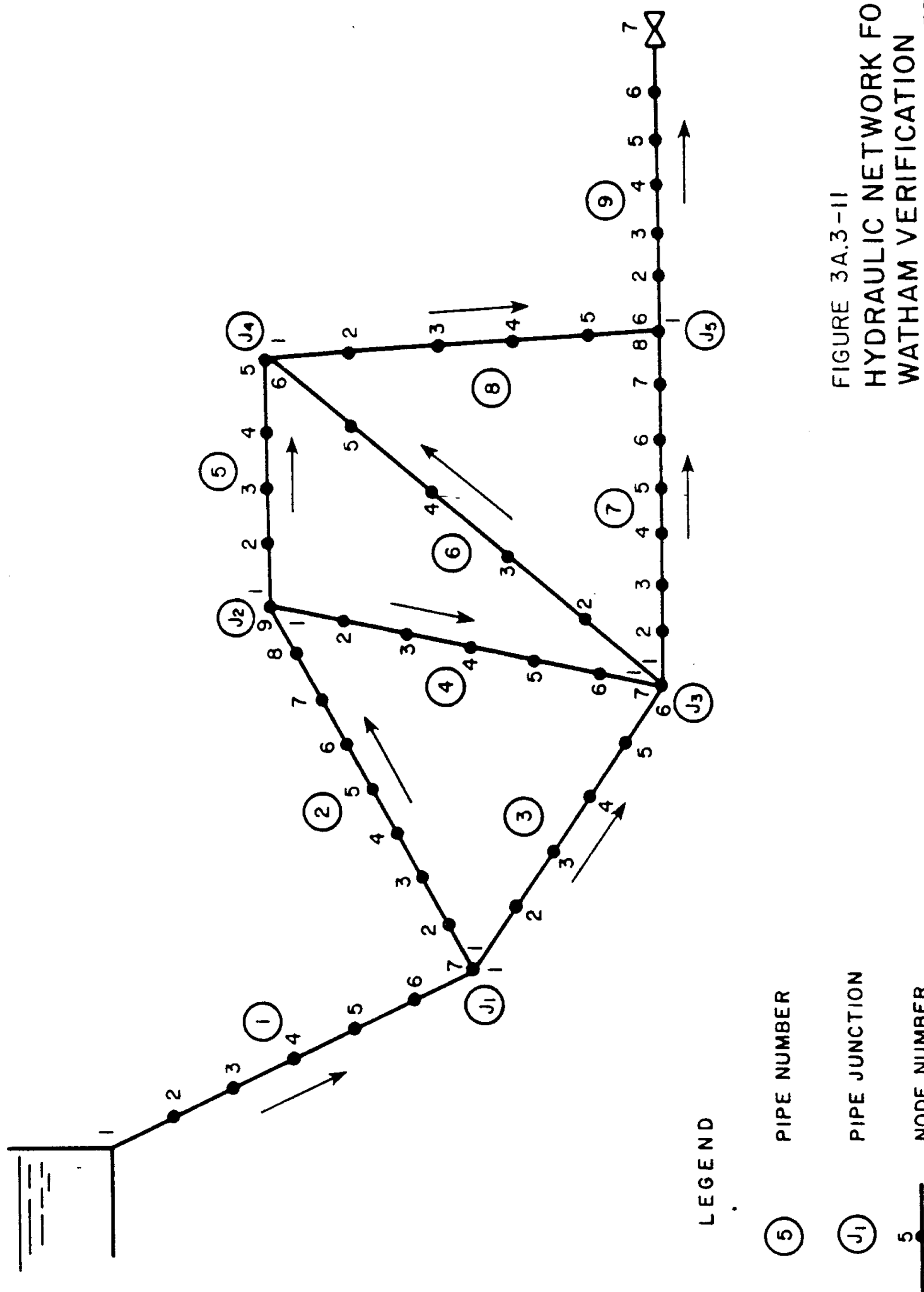


FIGURE 3A.3-11
HYDRAULIC NETWORK FOR
WATHAM VERIFICATION
MILLSTONE NUCLEAR POWER STATION

FIGURE 3A.3–12 HEAD VERSUS TIME PLOT FOR JUNCTION J

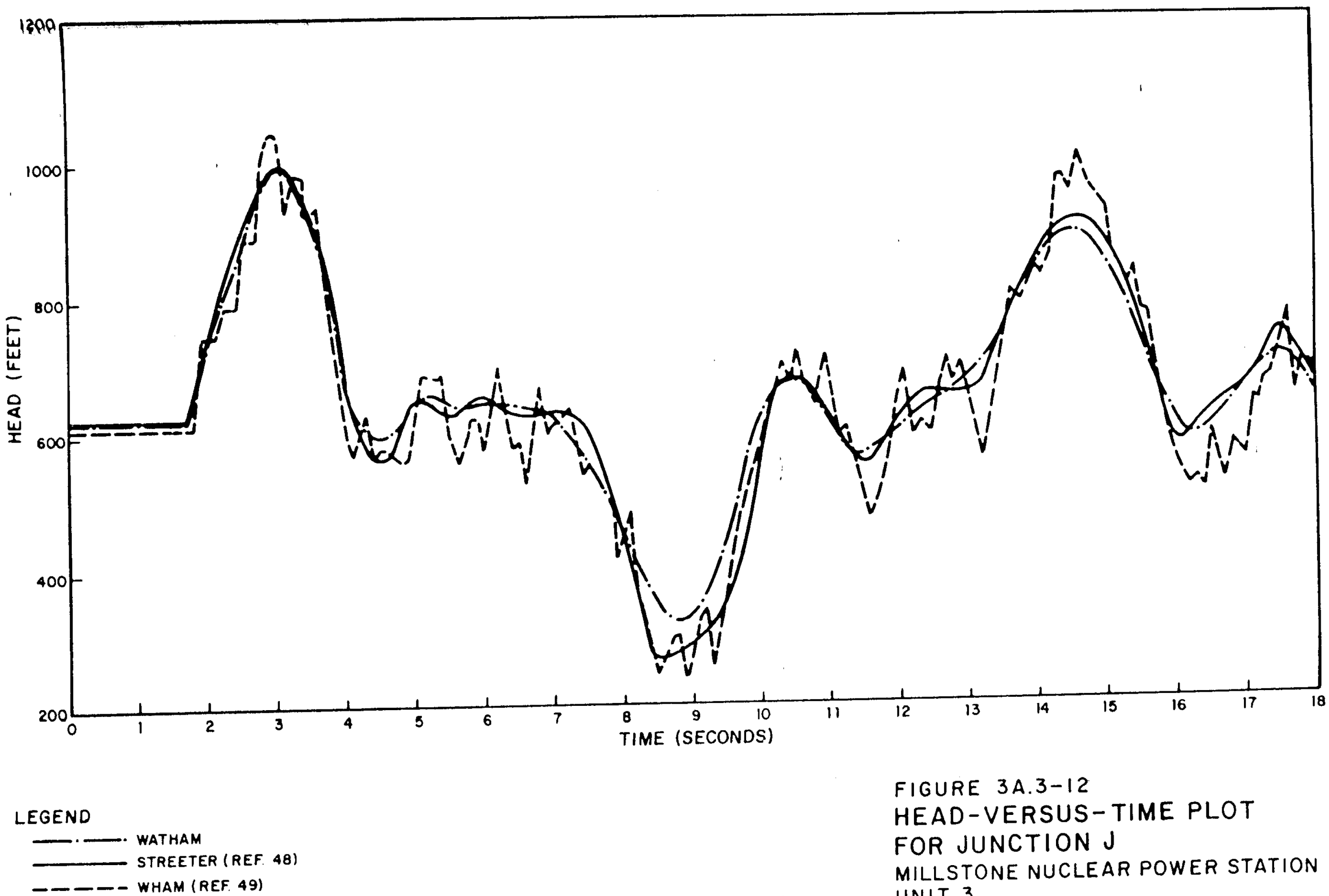


FIGURE 3A.3–12
HEAD-VERSUS-TIME PLOT
FOR JUNCTION J
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3A.3–13 HEAD VERSUS TIME PLOT AT VALVE

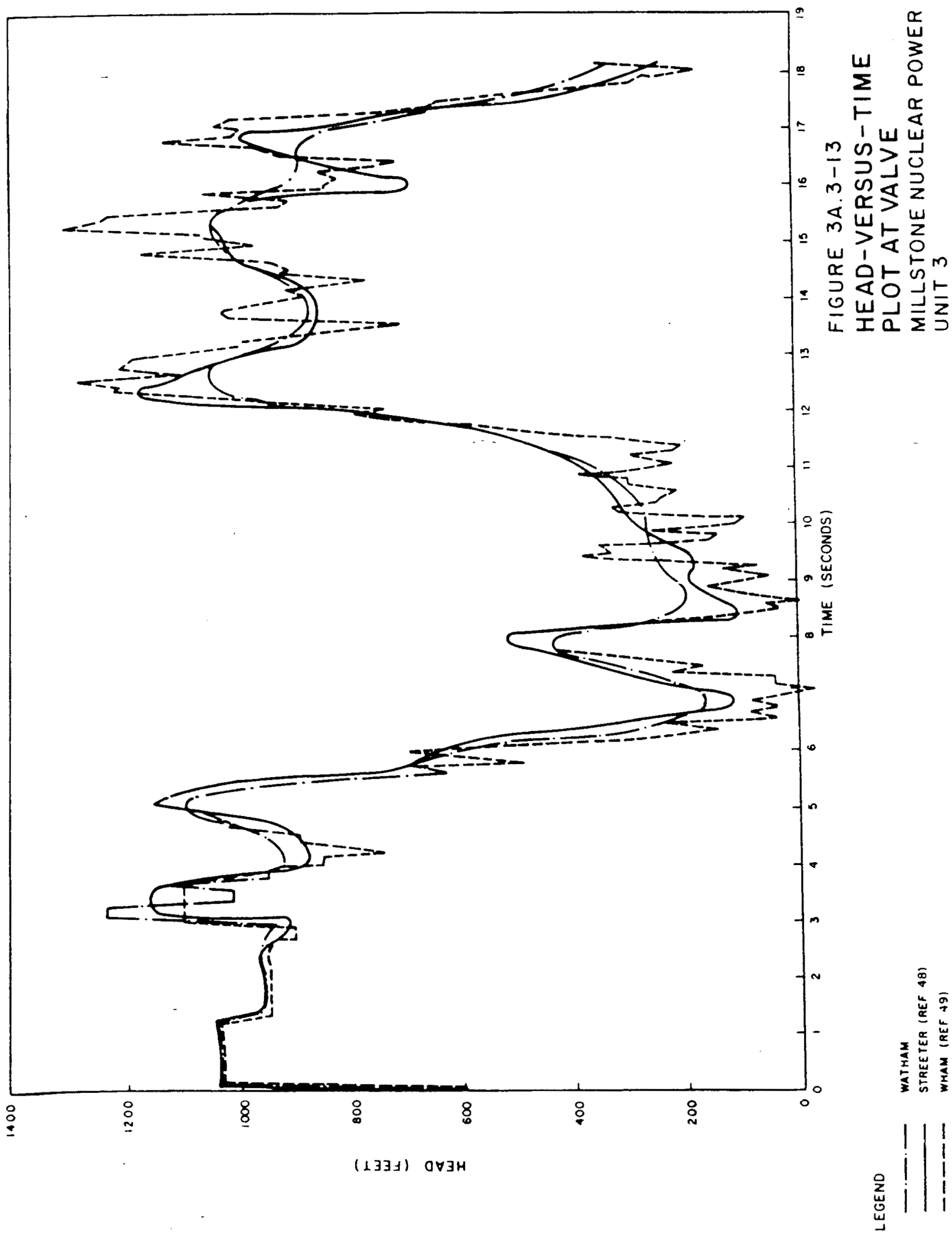
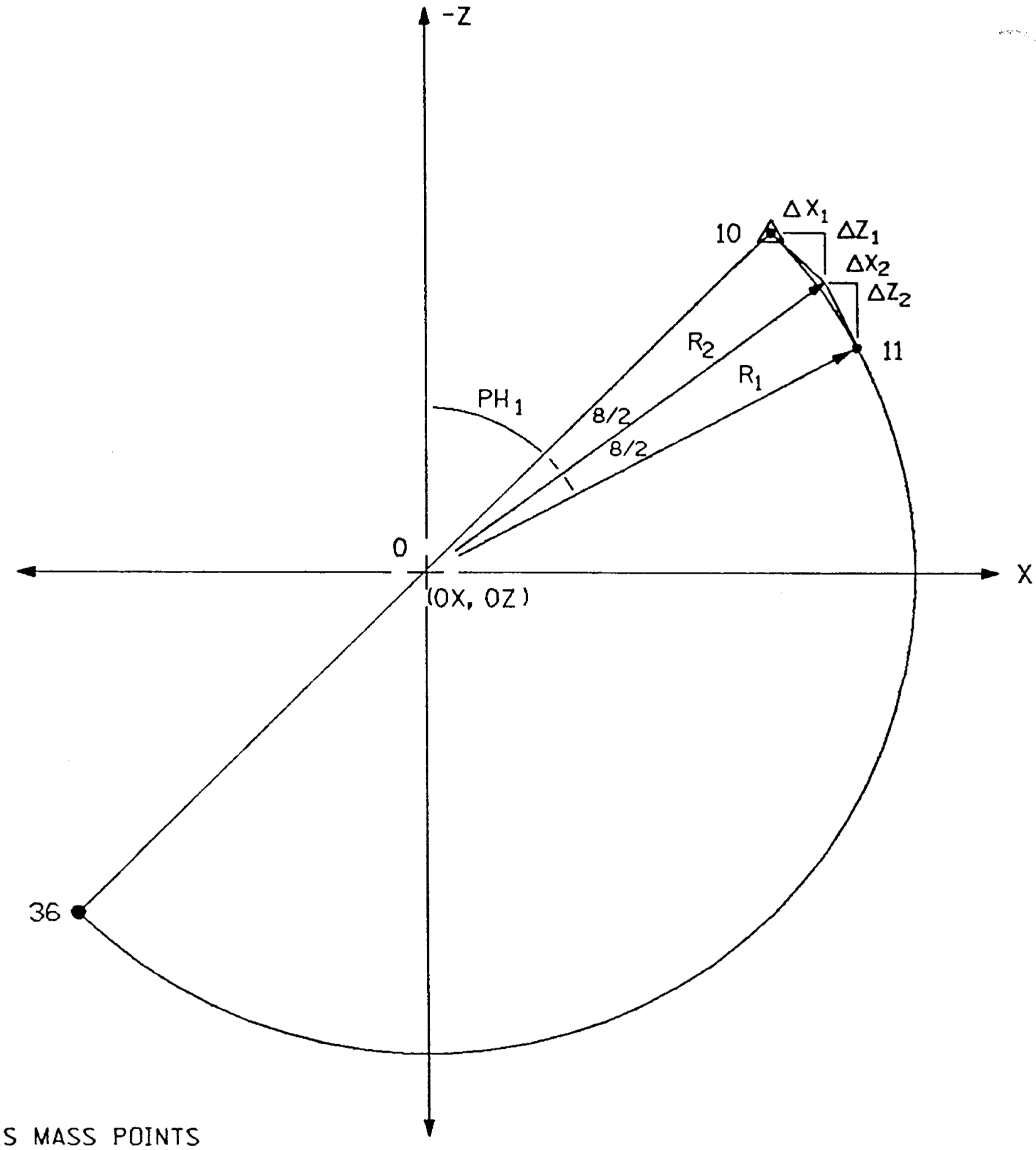


FIGURE 3A.3–13
HEAD-VERSUS-TIME
PLOT AT VALVE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3A.3–14 DESIGN INPUT TO A PROBLEM

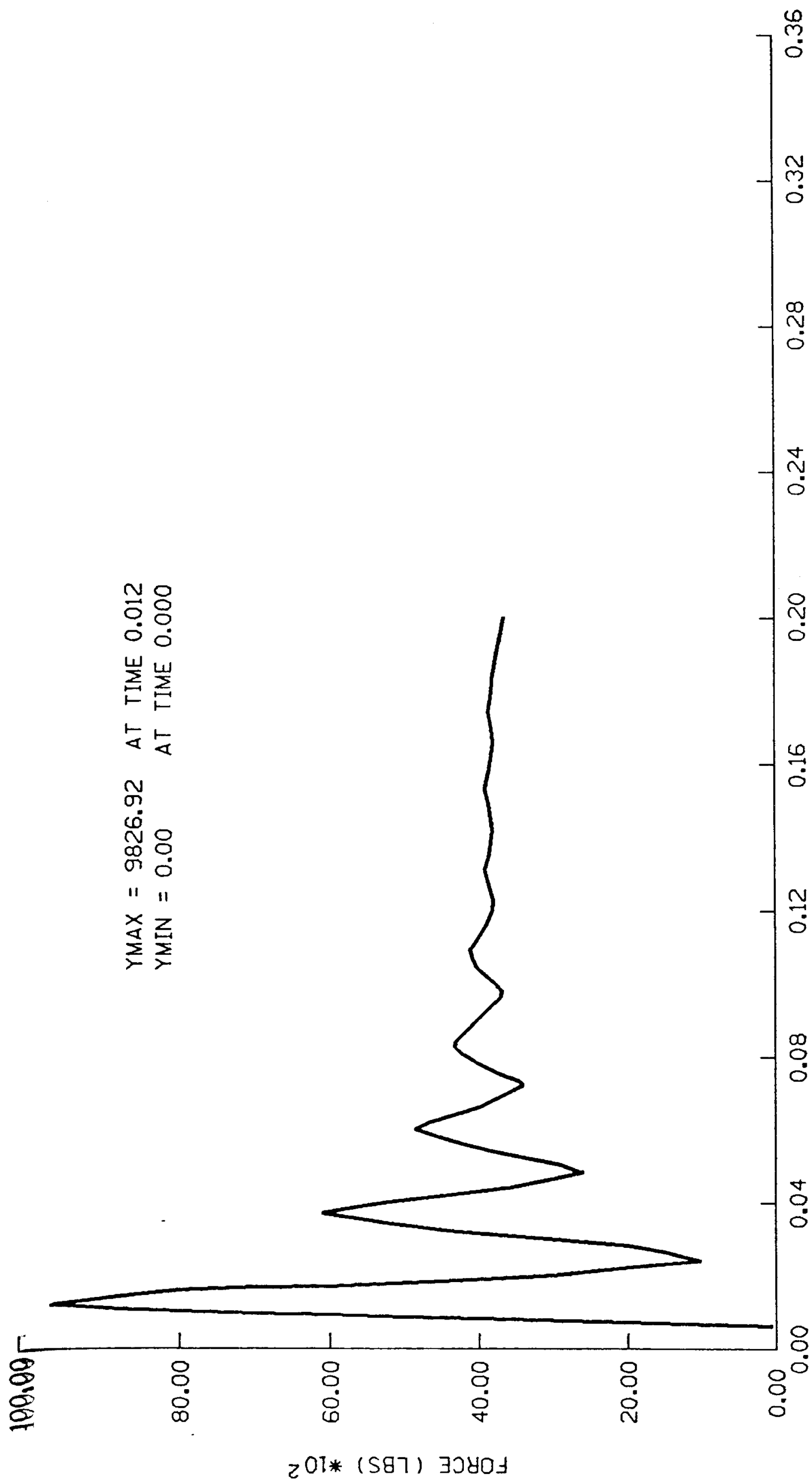


NOTES:

- △ = DENOTES MASS POINTS
- R = 100 FT.
- PH1 = START ANGLE=45°
- OX = X COORDINATE OF ORIGIN OF ARC = 0.0 FT.
- OZ = Z COORDINATE OF ORIGIN OF ARC = 0.0 FT.
- OD = 14.0 IN.
- θ = INCLUDE ANGLE BETWEEN NODES = 6.923°
- NANGLE = NO. OF INCLUDED ANGLES IN ARC = 26

FIGURE 3A.3.14
DESIGN INPUT TO A PROBLEM
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

FIGURE 3A.3–15 PIPE 1 SEGMENT 1 FORCE-TIME HISTORY



APPENDIX 3B - ENVIRONMENTAL DESIGN CONDITIONS

FOR ENVIRONMENTAL DESIGN CONDITIONS
SEE ENGINEERING SPECIFICATION SP-M3-EE-0333